Test and Evaluation of Alco/BLH Prototype Sodium-Heated Steam Generator (Final Report)

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ABSTRACT

As a part of the U. S. Atomic Energy Commission's Liquid Metal-Cooled Fast Breeder Reactor program, the Liquid Metal Engineering Center tested a 30-Mwt prototype sodium-to-sodium intermediate heat exchanger and a 30-Mwt prototype sodium-heated steam generator in combined operation in its Sodium Components Test Installation. This report contains the results of test and evaluation of the steam generator.

The steam generator was designed by Alco Products, Inc., and manufactured by Baldwin-Lima-Hamilton Corporation. The steam generator is a once-through design in which water/steam flows vertically upward in tubes, being heated by sodium flowing downward on the shell side.

During plant performance tests, performance degradation was observed, which resulted in the initiation of a diagnostic test series. This test series revealed that under certain operating conditions, the thermohydraulic characteristic of the steam generator changed either suddenly or gradually, resulting in overall performance degradation. A structural failure, requiring retirement of the unit, occurred before the diagnostic test series and analytical support effort were completed.

This report describes the thermohydraulic and structural performance, including the structural failures, and related evaluation analyses of the Alco/BLH prototype steam generator performed prior to termination of the test and evaluation program. In addition, the report presents a post-test examination plan to obtain data that could possibly explain the cause of performance anomalies and structural failures experienced during testing.

I. INTRODUCTION

A. BACKGROUND

In 1957-58 the U. S. Atomic Energy Commission initiated a Sodium Components Development Program with the objective of developing the industrial capability to manufacture equipment for advanced sodium-cooled nuclear power plants. Highest priority was given to development of intermediate heat exchangers (IHX's) and steam generators.

Preliminary studies indicated that future power plants would require such units in very large sizes. For instance a 1000-Mwe plant might use three IHX's and three steam generators, each of 700 to 900-Mwt capacity. It was not feasible to perform component tests on units of such size. It was therefore decided to undertake a component test program based on scaled-down prototypes. A rating of 30 Mwt was selected.

In one phase of the prototype program, Alco Products, Inc., was contracted to design and manufacture scaled-down prototypes of an IHX and a sodium-heated steam generator. In a related phase of the program, Atomics International (AI) was contracted to construct and operate the Sodium Components Test Installation (SCTI), a gas-fired test facility suitable for testing IHX's and sodium-heated steam generators combined in a single system.

Alco completed design of both the IHX and steam generator and completed manufacture of the IHX. However, the concern terminated its heat exchanger business before manufacture of the steam generator had proceeded very far. Baldwin-Lima-Hamilton (BLH) Corp. was then contracted to complete manufacture of the steam generator. The units made under this program are designated the Alco IHX and the Alco/BLH steam generator.

The test units were installed in the SCTI in 1965. Checkout of the facility and test operation of the IHX and steam generator then proceeded.

United Nuclear Corp. (UNC) was contracted in 1965 to perform an in-depth evaluation of this test program, retaining this function from 1965 through 1968. An important task accomplished by UNC was a review of the Alco design and performance calculations. This task resulted in the finding that certain performance

correlations had been applied incorrectly. Correct application showed that the Alco design was conservative, and thus improved performance over Alco prediction was expected. The SCTI was assigned to the Liquid Metal Engineering Center (LMEC) following establishment of that organization. From 1969 on, LMEC performed the evaluation function under a project distinct from SCTI operations.

Testing of the IHX and steam generator in a combined system proceeded intermittently until May 1970 when a structural failure of the steam generator was discovered which necessitated its retirement and removal from the SCTI. A post-test examination of the steam generator has been planned and is scheduled for initiation by Oak Ridge National Laboratory (ORNL), early in 1971. The Alco IHX is undamaged as far as can be determined and remains installed in the SCTI for use in tests with future steam generators.

B. SCOPE OF THIS REPORT

This report covers the testing and evaluation of the Alco/BLH steam generator and presents the following information:

- 1) A description of the steam generator and IHX, and the steam generator installation and instrumentation.
- 2) A description of planned and performed tests.
- 3) Thermohydraulic results for all steady-state tests performed on the steam generator.
- 4) Work accomplished toward analysis of test results, analysis and interpretation of performance anomalies and deviations from predicted performance.
- 5) The operating history of the steam generator, including data which provides an indication of the chemical and thermal exposure to which the steam generator structure has been subjected. This includes fluid temperature exposure, chemical exposure, and structural temperature gradients.
- 6) A record of structural defects encountered in service, investigations of these defects, and resulting modifications and repairs.

- 7) A description of the condition of the steam generator at retirement, including all known and unrepaired defects, and changes which resulted from fabrication problems that occurred during manufacture.
- 8) Results of structural studies performed by LMEC to determine the safety of the steam generator against tube rupture hazards, and to define limits of operation.
- 9) The LMEC-recommended post-test examination plan (Appendix E).
- 10) Engineering drawings of the steam generator (Appendix F).

II. SUMMARY, CONCLUSIONS, AND RECOMMENDATIONS

A. SUMMARY AND CONCLUSIONS

The Alco/BLH steam generator was designed by Alco Products, Inc., and manufactured by Baldwin-Lima-Hamilton Corporation. The unit was installed in the LMEC-operated Sodium Components Test Installation (SCTI) in 1965 and had been tested intermittently from 1966 until May 1970, when a structural failure necessitated its retirement.

1. Operation

First steam operation on the Alco/BLH steam generator was in January 1966. It was operated intermittently for a total of 1300 steaming hours up to November 1966, at which time the SCTI was shut down because of chloride stress corrosion in the steam piping. During the following 2-yr down period, the Type 347 SS steam line was replaced with ferritic steel (2-1/4 Cr - 1 Mo) piping and components. Operation was resumed in February 1969. After further intermittent operation totaling 900 steaming hours, during which plant performance tests were run at 20 and 40% rated power, a leak was detected in the steam generator as indicated by hydrogen in the cover gas. A 6-month down period followed, of which 2 months were spent for inspection, evaluation, and repair of the tube leak. Damage to the water and steam manways was also detected and repaired during the remainder of this down period. Operation was resumed again in late December 1969, and was continued with several short interruptions until March 1970, accumulating 1100 additional steaming hours, during which the early part of the diagnostic test series was performed. Evidence of sodium-water reaction and subsequent discovery of tube leaks necessitated another down period. Repair activities were completed by May 20, 1970, and a hydrotest was initiated in preparation for resumption of operation. During this test, leakage was observed in the vicinity of the feedwater header; investigation of this incident disclosed the presence of cracks in the lower tube sheet area and necessitated the retirement of the unit.

2. Thermohydraulic Performance

A total of 40 steady-state test points were analyzed from the tests performed in 1969 and 1970. Two test points (20 and 40% rated power in May 1969) were

from the rise-to-power tests under the plant performance test program. In the diagnostic test series, 10 test points were associated with sudden performance changes while the remaining 28 test points were obtained without sudden performance changes. The following range for the steam generator parameters were encompassed in these tests:

1) Steam outlet temperature	700 to 1050°F
2) Water/steam flowrates	21,000 to 124,000 lb/hr
3) Water inlet temperature	600°F
4) Water/steam pressure	2200 psig
5) Secondary sodium ΔT (inlet-to-outlet)	70 to 450°F
6) Sodium inlet temperature	760 to 1180°F
7) Sodium flowrate	154,000 to 1,233,000 lb/hr
8) Thermal power	6 to 22 Mw.

Plant performance tests were run for nominal 20 and 40% rated power points. The former exhibited good overall performance, while in the latter a performance decrease resulted with an increased sodium inlet temperature of 1150°F. From the extrapolation of the results of these tests, it appeared that the sodium inlet temperature approached the 1200°F design limit at slightly over 40% rated power. It was concluded that it would not be possible to achieve 100% rated power using the test methods proposed by Alco.

Observance of decreased performance at 40% rated power under plant performance tests initiated a rather extensive diagnostic test series to investigate the performance decrease problem. Performance changes observed under various operating conditions were either gradual or sudden. Gradual changes, during which no indication of a problem existed except higher than expected sodium inlet temperature, were experienced in plant performance tests and Diagnostic Test Series 3; while sudden changes, which included those of the steam outlet temperature and sodium temperature profile, were observed in Diagnostic Test Series 1, 4, and 5. Both gradual and sudden performance changes, however, appeared to be of the same nature. It was noted that performance changes included both performance degradation and recovery. The average boiling zone

overall heat transfer coefficient was about 400 Btu/hr-ft²-°F under normal performance conditions, and between 100 and 200 Btu/hr-ft²-°F under degraded performance conditions.

Correlation of operating conditions, under which poor heat-transfer performance resulted, was attempted. Although perfect correlation was not obtained, a definite separation of the good and poor performance regions did exist.

The analysis for possible causes of performance degradation included hypotheses such as earlier inception of film boiling and occurrence of static and/or dynamic flow instability in the water/steam tubes. The study on the first hypothesis indicates that film boiling may not be the initial cause of performance decrease, although it may be the end result.

A bistable flow, digital computer model was formulated to investigate the possibility of static flow instability existing in the water/steam tubes. This model was applied at the 20 and 40% rated power conditions and for the pre- and post-sudden performance change conditions. All conditions exhibited a (monotonic) continuously positive slope for the water/steam ΔP -vs-flow curve. This is indicative of static stability in the system.

A preliminary investigation was performed of dynamic instability and related heat transfer mechanism change as the cause of the performance degradation. These preliminary studies indicate regions of dynamic instability in the unit. However, more extensive investigation in these areas is required.

Tests with tube inlet orificing were planned to investigate flow instability experimentally, but were precluded with retirement of the unit.

3. Structural Performance

During the fabrication of this unit, several manufacturing anomalies occurred that could have an effect on the structural integrity of the unit. The major ones concerned the tube-to-tube sheet and tube-to-guide ring joints. These resulted in a few tubes being plugged during manufacturing.

An inspection in December 1968 revealed a group of pits in the Inconel overlay localized in various areas of the lower feedwater header. Following discovery of the lower tube sheet cracks in May 1970, it was observed that one of the previous pitted areas coincided with a portion of the crack area on the

external tube sheet. However, these visual examinations did not confirm the existence of a leak path between the pitting and the external crack network.

On June 22, 1969, a crack was discovered in steam tube II-47. Inspection revealed that the crack was approximately 6-1/4 in. below the tube-to-tube sheet weld (within the tube sheet thickness of 7 in.). A crack in the fillet weld on tube IV-27 was discovered in addition to several minor cracks in the welds. These two tubes were plugged and the surface defects were polished out.

In August 1969, severe cracking of the metal hexagonal-ring groove on the upper manway flange was discovered. Additional inspection revealed that the inner lip of the flange groove of both the upper and lower manways was deformed radially inward. Metallurgical examinations of a sample removed from the failure area revealed subsurface microfissuring predominantly in the first layer of the overlay material. Structural evaluation of the joint revealed that the inner lip of the groove was inadequate to withstand the applied bolt loads. This factor in conjunction with the microfissures in the dilution zone apparently initiated the failures. The flange was repaired by utilizing a diaphragm seal at the joint.

In March 1970, the tubes were reinspected during shutdown following discovery of moisture in the cover gas. Leaks were discovered in tubes III-52 and III-53 in addition to partial defects in other tubes. The failed and partially failed tubes were plugged in preparation for the next series of tests.

On May 20, 1970, following completion of repair and modification activities within the steam generator, and closure of the manways, water was admitted to the steam generator. At a pressure of 600 psig, water was observed leaking from the region of the lower tube sheet. Visual inspection upon removal of the insulation revealed several large cracks on the tube-sheet extension edge and in the radius region above the weld to the lower head. In addition, deposits of sodium oxide were visible on the undersurface of the tube sheet. All testing of the steam generator was terminated at this time and plans were initiated for a failure investigation.

4. Materials Performance

While material degradation occurred in several areas of the steam generator, none were solely related to the basic material selections of Type 316 SS with Inconel overlay on all water/steam-side surfaces. Corrosion is considered to be the principal mechanism contributing to the premature deterioration and shut-down of the steam generator. Specific areas of apparent corrosive attack included tube and tube sheet failures (crevice and stress-corrosion) and superficial pitting corrosion of the nickel-base alloy overlay. Apparent fabrication defects were limited to microfissuring in the stainless-steel/overlay dilution layer. This subsurface fissuring, in combination with inadequate support for the compression loads imposed, contributed to gross cracking of the protective overlay at the upper manway ring joint groove.

Based on current information and observed performance, fabrication, principally welding, was, with only minor exception, of satisfactory quality for the containment of sodium, water, and steam at elevated temperature.

B. RECOMMENDATIONS

The design, fabrication, operation, and test experience of the Alco/BLH steam generator and its documentation is a major contribution to LMFBR steam generator technology. The proposed post-test examination of the Alco/BLH steam generator and evaluation of the examination results will contribute further to this critical technology area. The following recommendations are made to maximize benefits from the current, as well as future, LMFBR steam generator development programs.

- 1) Further analytical studies on the cause(s) of the steam generator overall performance changes are strongly recommended mainly because performance degradation limits the unit capability. These studies should include the analysis of dynamic flow instability in the water/steam tubes and related change in heat-transfer mechanism.
- 2) The fundamental difficulty in efforts to explain the phenomenon of performance changes was in the limitation of instrumentation. In the future testing of prototype steam generators, adequate instrumentation must be provided. The instrumentation under consideration

includes thermocouples, pressure gages, flowmeters, strain gages, and accelerometers. The state of water/steam flow, in terms of affecting the thermal performance of the unit, may be best indicated by having fluid-flow measuring devices at the tube inlet and thermocouples at the tube outlet. The instruments should be accurate, sensitive (fast response), and of high readout frequency.

- 3) Throughout testing, temperature asymmetry had been exhibited in the lower tube sheet, particularly in the vicinity of the drain slot. Since this thermal asymmetry is suspected to have been related to the cause of the cracks discovered in May 1970, some further analysis may be worth undertaking. The post-test examination should include careful observation in the crack area. The results may show adequacy of the protective thermal shield with sodium in contact with the tube sheet surface for thermal protection or the necessity for an alternate thermal protection feature.
- 4) In designing a steam generator, more realistic thermohydraulic characteristics should be defined along the entire flow length, particularly in the two-phase flow region. As soon as basic sizing and flow configuration is completed, flow stability analysis should be performed because this is one of the most likely causes for performance decrease. Later orificing of tube inlets is an expensive way of correcting flow instability and, in addition, requires increased pumping power.
- 5) A few water/steam tubes (bimetallic) are recommended to be cut normal and parallel to the flow direction, after casting ends to minimize springback effects, and the sections examined with particular observation made on the state of the bond (metallurgical, mechanical, or other types) between two tube materials. If any gap is found, the extent of separation should be measured.
- 6) During the December 1969 to March 1970 operating period, most of the internal thermocouples in the sodium stream failed. It is recommended that the cause(s) of thermocouple failure be investigated and that improved installation methods be developed.

- 7) Post-test examination should include detailed metallurgical failure analyses for tube leaks and lower tube sheet/water header cracks. Based on these analyses, the potential suitability of a front-face-welded, tube-to-tube sheet joint and the protective measures required for the lower tube sheet may be established.
- 8) The post-test examination results may provide information as to possible cause of specific steam generator malfunctions. These results should be subject to analytical verification where appropriate.

III. DESCRIPTION OF EXPERIMENTAL SYSTEM

A. STEAM GENERATOR

The Alco/BLH steam generator is a once-through, counterflow, combined evaporator-superheater of 30-Mwt rated capacity. A schematic of the steam generator is shown in Figure 3-1. The steam generator is of vertical shell-and-tube construction, carrying liquid sodium on the shell side and water/steam on the tube side. The basic material of construction is Type 316 SS. All water/steam-side surfaces are of nickel-base alloy. This was accomplished by using tubes containing a co-extruded Inconel 600 liner and by placing an Inconel 82 weld rod overlay on the steam and water headers, nozzles, and manway cover surfaces.

The feedwater enters the unit at the lower end nozzle and water header, flows through 300 parallel vertical tubes, and exits via the upper end steam header and nozzle.

The sodium enters via an inlet nozzle near the upper end, flows around and radially inward over the inlet shroud, and flows down the heat-transfer zone in a series of cross-flow passes created by 54 disk and doughnut-type baffles. At the lower end, the sodium flows radially outward under the exit shroud and around this shroud to the exit nozzle. Sodium flow over the tubes is considered to be cross flow except within the upper and lower shrouds where it is a cross-axial flow.

The tubes terminate at flat tube sheets at the upper and lower ends. The tube-to-tube sheet welds are fillet-type face welds of Inconel and include an Inconel overlay over the tube ends. The tubes are expanded into the tube-sheet passages for a distance of 1-1/2 in. below the welds.

The tubes contain 4-ft sine wave sections at their upper end for thermal expansion. An argon-pressurized gas space is maintained in the upper shell below the upper tube sheet, including the sine wave sections.

The shell and sodium nozzles are provided with thermal shields of 1/4 in. thickness with 1/8-in. gap separation for thermal shock protection on the sodium side. The gas-space surfaces do not have thermal shielding. The lower tube sheet is protected against thermal shock by a stack of ten 1/4-in. circular disks separated from one another by 1/8-in. gaps. The upper tube sheet is protected

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Figure 3-1. Alco/BLH Steam Generator Design

against thermal shock by the argon gas space. It was further intended that the gas space maintain the upper tube sheet in temperature equilibrium with the out-flowing steam, and free from significant thermal gradients due to sodium temperature transients.

Two level-sensing connections, spaced at a 12-in. axial interval, are provided in the shell at the bottom of the gas space. Four connections are provided at the upper end of the shell for argon fill and venting, and for hydrogen monitoring.

The water and steam headers contain manways with bolted manway covers for access. The headers are basically of hemispherical construction but contain large, flat flange sections at the manways.

Within the upper-end gas space is a sodium-water reaction relief nozzle sealed by a burst diaphragm. A second relief outlet was built into the facility piping immediately downstream of the sodium exit nozzle.

Penetration fittings for internal thermocouples are provided, including three in the steam header, four near the upper end of the shell, and three passages drilled through the lower tube sheet. These passages were originally intended to be sodium drains.

The shell contains 54 alternate disk and doughnut baffles. Each baffle contains a passage for every steam tube. The baffles thus also serve as tube supports. The baffles and the two shrouds are positioned by a system of support rods attached to the lower tube sheet. A guide ring (similar to a baffle) is located about 3 in. below the upper tube sheet. This component has no operational function but was required to position the tubes during assembly.

Table 3-1 contains the principal quantitative design data for the steam generator. Table 3-2 lists its rated performance characteristics as defined by the designers. A general outline and significant dimensions of the unit are shown in Figure 3-1. The engineering drawings are contained in Appendix F.

B. STEAM GENERATOR INSTALLATION AND INTERMEDIATE HEAT EXCHANGER

The steam generator was installed in the SCTI for its entire test program. This facility contains a natural gas-fired sodium heater to simulate the heating function of a nuclear reactor. The heater is installed in the primary sodium

TABLE 3-1

ALCO/BLH STEAM GENERATOR DESIGN CHARACTERISTICS

Manufacturer	Alco and Baldwin-Lima-Hamilton		
Operating date	1966 (initial)		
Unit type	Once-through		
Thermal output (Mw)	30		
Shell geometry			
Fluid	Sodium		
OD x wall (in.)	32-7/8 x 7/16 32-7/8 x 1-1/8		
Length (ft)	45.25		
Flow baffles	Disk and doughnut, 54 at 7-in. intervals		
Design temperature (°F)	1200		
Design pressure (psig)	150		
Material	SA 240 Gr M Tp 316		
Tube geometry			
Fluid	Water-steam		
Type of tube	Bimetallic welded and expanded		
Material	SA 213 Tp 316 and Inconel SB-163		
No. of tubes	300		
OD x wall (in.)	1/2 x 0.072 Type 316 SS (outer wall) 0.356 (nom.) x 0.032 Inconel 600 (inner wall)		
Pattern	Circular		
Design temperature (°F)	1200		
Design pressure (psig)	2500		
Expansion provision	Sine tube-bend		
Tube-tube sheet protection	Gas space/thermal shields		
	<u>Preheater</u>	<u>Evaporator</u>	<u>Superheater</u>
Liquid sodium data			
Temperature in (°F)	-	-	1175
Temperature out (°F)	777	-	-
Flow (lb/hr)		855,000	
Water and steam			
Temperature in (°F)	600	-	-
Temperature out (°F)	-	-	1050
Steam pressure (psig)	-	-	2200
Flow (lb/hr)		116,000	

TABLE 3-2

DESIGN RATED PERFORMANCE CHARACTERISTICS OF
STEAM GENERATOR

	Shell-Side	Tube-Side
Fluid circulated	Secondary sodium	Water-steam*
Flow	855,000 lb/hr	116,000 lb/hr
Temperature in	1175°F	600°F*
Temperature out	777°F	1050°F*
Flow arrangement	Cross-counter	Once-through
Velocity		
Minimum	2.70 ft/sec	4.95 ft/sec (feedwater inlet)
Maximum	4.22 ft/sec	79.0 ft/sec (steam outlet)
Pressure drop	12.3 psi	35.3 psi
Design pressure	150 psig	2500 psig
Proof test pressure	620 psig	6220 psig
Design temperature	1200°F	1200°F
Normal operation	Startup 1000 cycles Operating 1,000,000 cycles	
Heat transfer surface (effective)	1260 ft ²	
Duty (full load)	102,400,000 Btu/hr	
Overall heat transfer coefficients		
Superheating		278 Btu/ft ² -hr-°F
Boiling		495 Btu/ft ² -hr-°F
Preheating		395 Btu/ft ² -hr-°F

*Feedwater inlet at 600°F

Saturation at 650°F (approximately)

Superheat to 1050°F

loop while the steam generator is installed in the secondary sodium loop. The two loops are coupled by the Alco IHX, the companion test unit to the steam generator. The Alco IHX is a vertical, counterflow, shell-and-tube heat exchanger of generally similar construction to the steam generator. Each sodium loop includes a circulating pump, expansion tank, block valves, control valves, and cold trap. Figure 3-2 is a simplified schematic of the SCTI sodium system; it includes heat balance data for rated performance. The primary sodium leg also includes an interconnected system of lines, mixing tees, and valves to apply temperature ramps to the IHX. This system is not shown in Figure 3-2. Its capabilities were not utilized in the test program.

The water/steam system of the SCTI includes a series of pressure-reducing valves and desuperheaters, to dissipate the high pressure and temperature of the steam produced by the steam generator, and a steam condenser. The feedwater equipment in the loop includes condensate, booster, and feedwater pumps, deaerating heater, intermediate- and high-pressure feedwater heaters and a demineralizer. The system was originally provided with a demineralizer capable of handling 30% of rated flow. This was later replaced by a unit capable of handling full flow. Figure 3-3 is a schematic of the water/steam system of the SCTI with heat balance data for main components.

A cooling-water loop links the condenser to a wet cooling tower which rejects the system heat to the atmosphere. An auxiliary steam generator provides low-pressure steam for preheat of the steam generator and lines.

C. INSTRUMENTATION

The SCTI systems and the steam generator were provided with instrumentation to measure principal parameters of the working fluids. The location of the pressure and temperature measurements on the steam generator are shown schematically in Figure 3-4. The types of instruments used for the various parameters are described in the following text.

1. Temperature Measurement

Ungrounded chromel-alumel thermocouples were used for feedwater inlet, steam outlet, and sodium inlet/outlet temperature measurements. In addition, the steam generator was instrumented with approximately 80 thermocouples to

Figure 3-2. SCTI Sodium System Schematic and Heat Balance Diagram

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Figure 3-3. SCTI Steam System Schematic and Heat Balance Diagram

Figure 3-4. Steam Generator Instrumentation

measure internal fluid temperature and structural surface temperature. Most of the thermocouple locations are shown in Figure 3-4. Included in the quantity were 32 immersion thermocouples (T13 through T44 of Figure 3-4) spaced along the heat-transfer zone so as to define the sodium temperature profile through this zone and also through the gas space. Most of these thermocouples failed during operation of the steam generator in February 1970. To provide equivalent measurements, thermocouples were mounted externally on the shell surface as shown in Figure 3-5.

The internal thermocouples were sheathed with Inconel. Those measuring fluid temperatures were ungrounded. External thermocouples were welded directly to the structural surface at the point of measurement.

2. Pressure Measurement

Sodium inlet and outlet pressure was measured with Taylor volumetric gages in which a diaphragm pressure-sensing element at the pressure tap is connected to a Bourdon spring by a capillary tube filled with NaK. The sensing elements were insulated to maintain the NaK at the same temperature as the diaphragm, thus minimizing the temperature correction which would be required if the NaK were at a different temperature from the sodium contacting the diaphragm. These gages were calibrated in place with the system filled with sodium, making due allowance for the height from the gage tap to the sodium level. Feedwater and steam pressure measurements were made by Bailey Bourdon-type gages. The accuracy of these instruments was insufficient for measurement of the water/steam-side pressure drop in the steam generator. A Barton direct differential pressure gage was used for this purpose.

A gage was also provided for measuring the gas-blanket pressure in the steam generator.

3. Flow Measurement

Sodium flow was measured by an electromagnetic flowmeter. This instrument had not received a satisfactory calibration. It was not relied upon for the steady-state sodium flow (see Section V.B), but was useful for providing a record of changes in flow.

- 28 INTERNAL TC'S WERE USED TO DEFINE STEAM GENERATOR PERFORMANCE
- 26 OF THESE HAVE FAILED OR ARE OF A QUESTIONABLE CONDITION
- 13 OF THESE HAVE BEEN REPLACED BY EXTERNAL TC'S; LOCATION IS DESIGNATED BY XT

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Figure 3-5. Steam Generator Internal and External Thermocouple Locations

Feedwater flow was measured by a venturi-flowmeter which had received laboratory calibration. Another flowmeter was provided to measure outlet steam flow. This instrument was, however, less accurate.

Two instruments were provided for measurement of the steam generator sodium level (sodium-argon interface). A float gage was installed so as to measure the level with respect to the two level taps provided in the shell. An Accuray gamma radiation gage was also used. The latter instrument tended to have a high noise level. Either instrument could be used as the feedback signal source for level control.

4. Recording System

Most temperature measurements were recorded by 100-point Bristol strip-chart recorders. Table 3-3 indicates the type of recorder and the estimated accuracy for the various measurements. The recording system described in Table 3-3 consisted of an assortment of strip- and circular-chart recorders and dial indicators. This instrumentation was satisfactory for use in steady-state tests conducted in accordance with the specified test objectives (see Section V). However most of the recorders had poor frequency response and there was no time correlation among the various recorders except for time marks provided by the operators. This proved a disadvantage in trying to reduce data for the sudden performance changes described in Section V.

TABLE 3-3

SCTI INSTRUMENTATION ACCURACIES AND RANGES
AND TYPE OF RECORDERS

Steam generator temperatures - TR-103, 100-point strip chart recorder

Accuracy $\pm 7\text{-}1/2^\circ\text{F}$, range 400-1250°F

Sodium flow - FRC-106, strip chart recorder

Accuracy $\pm 3\text{-}1/2\%$, range 300,000 to 1,240,000 lb/hr

Feedwater flow - FRC-201, circular chart recorder

Accuracy $\pm 2\text{-}1/2\%$, range 22,000 to 130,000 lb/hr

Steam pressure - circular chart recorders, PRC-200

Accuracy $\pm 2\%$, range 0-3000 psi

Steam flow - FRC-200, circular chart recorder

Accuracy $\pm 3\%$, range 22,000 to 130,000 lb/hr

Steam temperature - TR-200, circular chart recorder

Accuracy $\pm 10^\circ\text{F}$, range 400-1100°F

Feedwater temperature - TRC-201, circular chart recorder

Accuracy $\pm 10^\circ\text{F}$, range 0-1000°F

Terminal sodium pressure PI-310 and PI-313, vertical indicating pressure gage

Accuracy $\pm 2\text{-}1/2\%$, range 0-100 psig

Steam generator gas blanket pressure - PI-610, Bourdon type pressure gage
calibrated using precision master gage for comparison

Accuracy $\pm 2\text{-}1/2\%$, range 0-100 psig

Steam generator sodium level, LR-103

Accuracy $\pm 3\%$, range ± 6 in.

IV. OPERATING HISTORY

A. CHRONOLOGICAL HISTORY

The Alco/BLH steam generator was installed in the SCTI in 1965. It was first filled with sodium in August 1965. The first steam operation was in January 1966. It was operated intermittently for a total of 1300 steaming hours up to November 1966, at which time the SCTI was shut down due to chloride stress corrosion in the steam piping (Type 347 SS). The steam and water headers were opened at that time for inspection. No damage to the steam generator was discernible. Slight pitting found in the lower tube-sheet/water-header overlay surfaces was considered a dormant condition related to manufacturing rather than service history.

Operation was resumed in February 1969 after a down period of over 2 yr, during which the entire Type 347 SS steam line was replaced with ferritic steel (Cr - Mo) piping and components. After further intermittent operation totaling 900 steaming hours, a leak was detected in a steam tube. The leak was detected during a start by indication of hydrogen in the cover gas. There was approximately 1 hr of operation in the leaking condition, during which time it was estimated that up to 60 lb of sodium reacted with water. The leak was within the gas space and reaction products were assumed to have collected at the sodium level. A 6-mo down period followed, of which 2 mo were spent for inspection, evaluation, and repair of this tube leak. Damage to the water and steam manways was also detected and repaired during the remainder of this down period.

Operation was again resumed in late December 1969. Restart was preceded by a sodium flush of the gas space, which was intended to remove any sodium-water reaction products remaining from the leakage incident. Operation was continued with several short interruptions until March 1970, accumulating 1100 additional steaming hours. During this period of testing, persistent but inconclusive evidence of sodium-water reaction products was observed. The evidence included abnormal hydrogen concentration in the cover gas, and intervals of high sodium plugging temperature. There was also difficulty in controlling sodium level because of plugging of the pneumatic level control system by sodium hydroxide. These indications were finally supplemented by the discovery of moisture in a vent line that had been removed for cleaning. Subsequent dewpoint checks of

the gas space verified the existence of moisture. It was then found that two additional tube leaks had developed and three other tubes contained apparent external defects. A down period then ensued to permit plugging of the defective tubes. This down period was also used for the installation of tube inlet orifices (see Section V.A.8).

During the December 1969 to March 1970 operating period, most of the internal thermocouples in the heat-transfer zone of the steam generator failed. A few had failed previously. Because it was not feasible to repair these, external thermocouples were mounted on the steam generator shell to provide equivalent measurements.

Repair activities and the installation of orifices was completed by May 20, 1970, and a hydrotest was initiated in preparation for resumption of operation. With water conditions of 600 psig and room temperature, leakage was observed in the vicinity of the water header. Removal of insulation disclosed the presence of cracks in the lower tube-sheet region, which necessitated termination of testing.

Figure 4-1 outlines the above chronology.

B. START AND SHUTDOWN

1. Methods

A total of 34 steam generator startups and shutdowns (17 in 1966, 10 in 1969, and 7 in 1970) was accumulated on the Alco/BLH steam generator. The startup procedure consisted of either:

- 1) Circulating the feedwater through the tube side of the steam generator and preheating the feedwater from a separate boiler to approximately 350°F, and then introducing the secondary sodium into the shell side of the steam generator at a temperature that is within 50°F of the feedwater temperature.
- 2) Circulating the feedwater through a steam generator bypass (if the secondary sodium system was already circulating), until it was within 50°F of the lower tube-sheet temperature; and then introducing the feedwater into the steam generator.

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Figure 4-1. Steam Generator Chronology

Shutdown was performed by initiating a plant SCRAM which automatically shut down the gas-fired heater, the sodium pumps, and the feedwater pumps. The feedwater inlet valve and the steam generator sodium inlet and outlet valves were also closed. Immediately after the SCRAM, the forced-draft fan for the gas-fired heater and the primary system sodium cooler were shut down. The systems were then allowed to cool naturally.

2. Cooling after Shutdown

After shutdown, unless the secondary sodium was drained, local internal temperatures in the steam generator tended to remain virtually constant at their values just before shutdown. This tendency is attributable to the positive, vertical sodium temperature gradient throughout the heat-transfer zone, which produces no natural circulation. The Alco IHX had a similar thermal tendency after shutdown.

It was necessary to bring the steam generator to an isothermal state at about 400°F to permit restart. To reach this condition in a reasonable time, a cooling procedure was adopted which involved the primary and secondary loop in combination. The primary loop was cooled by circulating sodium with an electromagnetic (EM) pump and employing the sodium heater as a cooler (blowing furnace air with no gas combustion). The secondary loop was permitted to undergo natural circulation, the IHX serving to cool the steam generator. The rate of circulation was controlled to avoid severe temperature transients.

C. FLUID TEMPERATURE EXPOSURE

1. Operating Limits

The following limits, recommended by Alco in their design report (Reference 1, Vol IV), were observed for the entire test program:

- 1) Steam outlet temperature: 1075°F for steady-state
1100°F for short periods
- 2) Lower tube-sheet fluid temperature difference: 175°F
- 3) Sodium and steam temperature change rates: Not to exceed 100°F/hr during adjustment period to achieve test condition. If the metal

temperatures had stabilized at some steady-state level, then a 150°F temperature change in 15 min or longer was permissible.

4) Sodium level: Within the readout range of the (float) level detector.

These limits were maintained. Tube-sheet ΔT exceeded 175°F marginally on several occasions by amounts within the accuracy of measuring instruments. At several times during thermal transients, the steam temperature changed at a rate of 0.16°F/sec (576°F/hr), but no design limit was specified for the transient rate of the steam temperature.

2. Cumulative Exposure

a. Steam Outlet Temperature

A summary of the number of operating hours at various steam outlet temperatures is given in bar chart form in Figure 4-2. Temperature was arbitrarily divided into 100°F segments. The total operating time for each year and the summation of all three years is shown. The extensive operating time at less than 400 to 500°F represents a combination of isothermal operation with only water and no steam in the system as well as initial warm-up during rise to power. The unit was operated for an approximate total of 4100 hr, of which 40% was logged at steam temperatures below 500°F, and less than 10% at temperatures greater than 900°F. The majority (55%) of operation at temperatures greater than 900°F was performed during 1970. Most of the large number of hours (787) at 700 to 800°F steam temperature were accumulated during the diagnostic test series in which the poor performance characteristics of the steam generator were investigated. The relatively low amount of superheat (50°F) at 700°F steam outlet temperature allowed testing to be done at a high sodium-to-water temperature difference. A chronological record of steam temperature is shown in Figure 4-3 for 1969 operation, and in Figures 4-4 and 4-5 for 1970 operations.

b. Feedwater Inlet Temperature

The feedwater inlet temperatures during the operating periods are presented in bar graph form in Figure 4-6. The feedwater temperatures had never exceeded 600°F. Half of the operating hours had been at temperatures between 500 and 600°F, which is the normal operating range while steaming.

Figure 4-2. Steam Generator Steam Temperature History

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Figure 4-3. SCTI Schedule of Events, January through June 1969

STEAMING HOURS: 547

CUMULATIVE SINCE STEAM GENERATOR REPAIR: 596

- 1 SHUT DOWN FOR SAFETY VALVE REPAIRS
- 2 SCRAM - LOW COMBUSTION AIR FLOW
- 3 SCRAM - Na HEATER CONTROL WIRE FAILURE
- 4 SCRAM - LOSS OF 480-V POWER

7693-59170

Figure 4-4. SCTI Operating History, January 1970

STEAMING HOURS: 478

CUMULATIVE SINCE STEAM GENERATOR REPAIR: 1072

- 1 SCRAM - LOSS OF ELECTRICAL POWER
- 2 SCRAM - STEAM GENERATOR OUTLET FLANGE GASKET LEAK
- 3 MAINTENANCE ON 4160-V SWITCHGEAR, CLEANING OF VAPOR TRAPS, AND INSTALLATION OF EXTERNAL THERMOCOUPLES

7693-59171

Figure 4-5. SCTI Operating History, February 1970

Figure 4-6. Steam Generator Feedwater Temperature History

c. Secondary Sodium Inlet and Outlet Temperatures

The thermal history of the secondary sodium inlet and outlet temperatures is presented in bar graph form in Figures 4-7 and 4-8, respectively. Temperature is divided into 100°F segments. Total operating hours at each segment are depicted in a similar fashion to the steam chart (Figure 4-2). The majority of the operating hours (60%) were in the less than 400 to 500°F range. This reflects isothermal operation during preheat and system cleanup activities. Approximately 25% of the sodium inlet operating hours were at about 700°F, the temperature necessary to disassociate NaH. Approximately 25% of the sodium outlet operating hours were at a temperature greater than 608°F, the temperature necessary to dissociate NaOH. These values have significance as related to sodium purity, which will be discussed in the next section.

3. Transients

The startup and shutdown procedures discussed in an earlier section were not significant with respect to introducing transients. Temperature changes during startup and shutdown were limited by the operating procedure to 150°F/hr for a 15-min period. Typical startup and shutdown transient information is listed in Table 4-1.

The most severe transients experienced, however, were not during startup or shutdown operations, but rather during testing such as when a "flip" or sudden and rapid degradation of steam generator performance occurred. Some of the more severe operating transients experienced are summarized in Table 4-1. A plot showing steam outlet temperature during these transient periods is included in Figure 4-9.

Although the steam generator flange leaks and loss of secondary sodium control were isolated incidents, the sudden performance changes (flips) were experienced a total of 18 times during the operation of the Alco/BLH steam generator. The February 27, 1970, flip presented in Figure 4-9 resulted in the greatest total steam temperature change. The total change in steam temperature for the flips ranged from ~5°F to 170°F. However, the rate of temperature change (0.75°F/sec) was approximately the same for all of the flip cases.

Figure 4-7. Steam Generator Sodium Inlet Temperature History

Figure 4-8. Steam Generator Sodium Outlet Temperature History

7693-59175

Figure 4-9. Steam Generator Outlet Steam Temperature - Representative Transients

TABLE 4-1

STEAM GENERATOR TYPICAL THERMAL TRANSIENTS

Transient Events	Date	Duration (sec)	Temperature Change (°F/sec)	
			Steam Out	Sodium Out
Typical startup	3-9-70	940	+0.04	+0.04
Typical shutdown	3-10-70	830	-0.01	+0.04
<u>Abnormal Transients</u>				
Flip and subsequent flange leak 1970	2-27-70	240	-0.75	+0.21
Steam generator flange leak 1969	5-16-69	3100	-0.15	-0.02
Secondary sodium control lost*	3-19-69	900	-0.15	~0

*Electronic failure in flow controller

D. CHEMICAL EXPOSURE

1. Feedwater Purity

The SCTI water purity requirements are set forth in Table 4-2. These requirements were established in 1964 with the aid of an industrial water consultant. Existing industry-wide steam plant practices plus the material specifications of the steam generator, the SCTI piping, and the SCTI components were considered to arrive at these requirements.

With few exceptions these requirements have been maintained. Those exceptions were:

- 1) During the 1966 operations there were procedural difficulties with pH control. Normally, the pH of the water was continuously monitored in the control room, utilizing a Beckman pH meter. It was controlled by the manual addition of morpholine to make the water more basic, as required. The mean pH was approximately 8.4 with short periods as low as 7.3 and as high as 9.8. The pH never dropped to an acid range. The out-of-limit pH condition was not considered to be significant. The procedural difficulties were resolved prior to the 1969 startup.

TABLE 4-2

STEAM GENERATOR INLET FEEDWATER PURITY

1. pH	8.8 - 9.0
2. Cation conductivity	1.0 μ mho max
3. Dissolved oxygen	0.000 - 0.005 ppm
4. Iron	10 ppb max
5. Copper	10 ppb max
6. Silica	20 ppb max
7. Chloride	0.1 ppm max

Exceptions:

1966 pH: 7.3 - 8.4 - 9.8

Manual control by the addition of morpholine.

May 1969 - D-2 demineralizer exhausted; increased to 4.4 μ mho during 3-hr 40-min shutdown period

June 1969 - Copper high

June 1969 - Feedwater header chloride concentration high

- 2) On May 15, 1969, demineralizer D-2 was exhausted (cation conductivity dropped to less than 1 μ mho). A decision was made to continue steaming operations while bypassing the demineralizer during its regeneration. When the demineralizer was returned to service the cation conductivity had increased to 4.4 μ mho's. The total period of the demineralizer shutdown was 3 hr 40 min. This relatively short period of outage was also considered to be insignificant.
- 3) On June 19, 1969, the content of copper in the feedwater exceeded the specification limits. The industrial water consultant was contacted to assist in resolving this problem. It was felt that this condition could be tolerated for long periods (weeks) without deleterious effects on the steam generator or the plant. The automatic shutdown of June 20, 1969, and the tube leak of June 22, 1969, interrupted attempts to locate the copper source. When circulation was reestablished through

the feedwater system in December 1969, copper was again high. Regeneration of the new D-3 polishing demineralizer corrected the problem.

- 4) In addition, following feedwater introduction to the steam generator, the chloride concentration increased temporarily before dropping to a steady low level. This rise was indicated in water samples taken from the steam generator outlet during the first hour after flow was established. It was due to the impurity concentration process which occurs as the steam generator boils and dries following shutdown, and the subsequent return of impurities into solution when feedwater is added. Chloride concentrations in excess of 10 ppm have been found in samples of this effluent water. A similar concentration of impurities occurred following the shutdown of June 22, 1969. However, in this instance, the sample was condensate which remained in the feedwater header and which was obtained following removal of the lower manway cover. The maximum chloride concentration in this residual water was 1.8 ppm.

2. Sodium Purity

The secondary sodium system has been in operation for a total of 7,609 hr. It has been filled and drained 19 times, as depicted in Table 4-3. The first three times, in early 1965, were for flushing purposes; those between August 1965 and October 1966 were for checkout type activities; those in early 1969 were done in conjunction with the plant performance testing, and those in 1970 in conjunction with the steam generator diagnostic test series. The longest single operating period was for 117 days ending in June 1969.

Figures 4-10 and 4-11 are graphs reflecting the sodium purity as a function of plugging temperatures for the 1969 and 1970 testing periods. It also contains specific information of analytical chemistry results of samples of sodium removed from the system. Agreement between the plugging temperatures and the sodium samples was not good. The plugging temperatures indicated the presence of a much higher level of Na_2O than the sodium sample analysis. It was felt that the reaction products left from the June 22, 1969, leak incident, such as sodium

TABLE 4-3

SECONDARY SODIUM FILL AND DRAIN HISTORY

Filled ⁽¹⁾	Drained	Duration ⁽²⁾ (day)
Initial Sodium Flush		
4-29-65	4-29-65	1
4-30-65	4-30-65	1
6-4-65	6-4-65	1
Plant Checkout		
8-20-65	8-25-65	6
11-29-65	12-9-65	11
11-11-65	12-24-65	13
12-29-65	1-15-66	18
12-24-65	1-28-66	5
2-6-66	3-1-66	24
3-8-66	3-17-66	10
3-9-66	6-18-66	41 ⁽³⁾
9-27-66	10-4-66	8
10-30-66	11-7-66	9
Plant Performance		
2-20-69	2-21-69	2
2-26-69	6-22-69	117 ^(3,4)
Diagnostic Test		
12-22-69	12-27-69	6
12-29-69	1-11-70	45 ⁽³⁾
2-15-70	2-28-70	14
3-8-70	3-11-70	4

(1) The secondary sodium system was filled and drained 19 times.

(2) Total hours of operation — 7,609 hours.

(3) Longest periods of operation in that year.

(4) Longest period of all operation.

7693-59176

Figure 4-10. Secondary Sodium Plugging Temperature History—1969

LMEC-Memo-70-20
64

7693-59177

Figure 4-11. Secondary Sodium Plugging Temperature History - 1970

hydrides, served to make the plugging temperatures higher and thus mask the true O_2 concentrations. The sample analysis results were assurance that the level of Na_2O impurities was below 12 ppm, which is the normal operating guideline.

The high points ($>$ than $300^\circ F$ plugging temperature) were all associated with startups or periods of time when high temperatures were reached for the first time. The high temperatures may have served to dissolve sources of Na_2O that existed in the sodium system.

3. Argon Cover Gas

A gas chromatograph was utilized to monitor H_2 concentrations in the cover gas of the steam generator during all testing. Figure 4-12 is a plot of the H_2 concentration and the sodium inlet and outlet temperature against time since the last startup in December 1969. There were no previous significant H_2 concentration levels recorded, except just before the first tube-leak incident on June 22, 1969.

The hydrogen concentration was high (6250 ppm) when the plant was first started up after the leak repairs were completed, as might be expected due to residual reaction products left in the steam generator and sodium system from the June 22, 1969, tube-leak incident. The concentration went up with temperature and then came down to a negligible amount as the temperature stabilized. This condition prevailed until after the first scram, which occurred on January 12, 1970. After that, the H_2 concentration levels had become much more significant. Generally, they rose with temperature and then decreased as the temperature stabilized. Peaks of 22,000 and 30,000 ppm were achieved. The average ranged between 6,000 and 15,000 ppm.

Helium checks were performed after each shutdown and/or SCRAM, but no leaks were found. The relatively high H_2 concentrations were attributed to reaction products remaining from the June 22, 1969, leak incident.

In the latter part of February 1970, the plant was shut down due to a leaking flange in the steam generator steam outlet line. During this outage, the steam generator cover-gas vent lines and a freeze trap were removed for cleaning. Moisture was found in the freeze-trap line. It was not clear at the time, however, whether or not the moisture was a result of the high humidity conditions which existed when the line was cut, or if it came from the cover gas.

Figure 4-12. Hydrogen Concentration History of Cover Gas
(Sheet 1 of 6)

Figure 4-12. Hydrogen Concentration History of Cover Gas
(Sheet 2 of 6)

Figure 4-12. Hydrogen Concentration History of Cover Gas
(Sheet 3 of 6)

Figure 4-12. Hydrogen Concentration History of Cover Gas
(Sheet 4 of 6)

Figure 4-12. Hydrogen Concentration History of Cover Gas
(Sheet 5 of 6)

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Figure 4-12. Hydrogen Concentration History of Cover Gas
(Sheet 6 of 6)

When the plant was restarted on March 9, 1970, the dewpoint of the cover gas was monitored. Moisture was indicated, so the plant was shut down. A subsequent helium leak check indicated a definite tube leak. The exact point in time that the leakage initiated could not be determined from the H_2 concentration readings because it was masked by the H_2 dissociated from the residual reaction products left from the June 22, 1969, leak incident.

E. STRUCTURAL TEMPERATURES

Some of the steam generator internal and external thermocouples served to give indication of temperature distribution over its structural elements. This section is a discussion of significant findings obtained with this instrumentation.

1. Lower Tube-Sheet Temperatures

Temperature distribution in the vicinity of the lower tube sheet was measured by an array including 6 external thermocouples, 3 thermocouples placed in holes in the tube-sheet flange extension, and 9 thermocouples placed on the upper surface of the lower tube sheet, of which 3 were inoperative. Throughout testing, these thermocouples had exhibited asymmetry and deviation from theoretical values, which could not be readily explained. Analytical temperatures were obtained by applying the measured fluid boundary temperatures to a finite element heat-transfer model of the tube-sheet-header area using the finite difference computer program (DEAP). Figure 4-13 shows the locations of these thermocouples, the analytical temperatures obtained at each thermocouple location, and the measured temperatures for a test at 40% power. The analytical values were obtained by applying the measured fluid temperatures as boundary conditions to the above computer model. The external thermocouples show a distinct asymmetry, with the set located 150 degrees from the sodium discharge nozzle having the highest temperatures, and the set 30 degrees from the nozzle having the lowest.

Of the 6 active thermocouples on the tube-sheet surface, those numbered 2 and 8 were located within the tube bundle. They were separated from the flowing sodium within the shell by the 10 shield plates in series. The analytical model indicated the temperature at these locations was 12°F above the feedwater temperature of 602°F . However, as shown in Figure 4-13, the measured temperatures were 74 to 77°F above the feedwater temperature.

Figure 4-13. Lower Tube-Sheet Temperature Data
(40% power; May 15, 1969; 1530 hours)

A sodium drain hole exists in the tube sheet, which is connected by a 1-in. pipe to the main sodium outlet line. During 1966 operations, this line was open. It was suggested that the high temperatures were caused by sodium flowing into the tube sheet-shield gap and draining via the 1-in. line. Prior to the 1969 operations this line was valved shut. However the temperature pattern exhibited in 1969 and 1970 operations was similar to that observed in 1966.

A possible explanation of the tube-sheet external temperature asymmetry would be the closeness of the high-reading set of thermocouples (66, 75, and 76) to the slot which drains the shell-thermal shield annulus. Sodium entered this annulus just below the upper shroud support ring through a labyrinth formed by overlapping plates. It could leave this space only at the aforementioned slot. When sodium was flowing, there was a pressure drop between the inlet and exit locations and some sodium flow might therefore exist through the slot. This might explain the higher temperatures near the slot.

It was further observed that the high-reading thermocouple set coincided closely in location with the cracks in the lower tube-sheet flange areas. The possibility was therefore considered that the temperature asymmetry might be related to the cause of failure. To investigate this possibility, a survey was made of tube-sheet temperatures in relation to changes of operating conditions. By analysis, the temperature at any location should be definable in a reduced form as follows:

$$(T_r)_i = \frac{(T_w)_i - T_{FW}}{(T_{Na})_O - T_{FW}} \cdot 100\%$$

where

T_r = reduced temperature

$(T_{Na})_O$ = sodium outlet temperature

T_{FW} = feedwater inlet temperature

T_w = local wall temperature

i = a particular location.

The reduced temperature should therefore be a constant for any combination of sodium and feedwater temperature if all factors affecting heat transfer are unchanged. Reduced temperatures as defined above were obtained for the thermocouple sets 67, 71, 76 and 68, 72, 75. The survey covered the initial vs the final test condition of each of the five test series, and the pre- vs post-conditions for three of the sudden performance changes. The results indicated some changes in reduced temperatures with test conditions, but these changes may be within the accuracy of measurements and their significance is not known. The general pattern of asymmetry, with the highest temperatures at an angular location 150 degrees from the sodium discharge nozzle and the lowest at 30 degrees, remained fairly constant.

2. Shell Temperatures

External shell temperatures at any height along the heat-transfer zone appeared to reflect the sodium temperature within the shell at the same height during steady-state operation despite the presence of the thermal shield and stagnant sodium annulus. This is shown in Figure 4-14 by a comparison of readings during Test 8R from the external thermocouples installed during February 1970, with readings of internal ones during Test 8. Conditions for the two tests were very similar (see Table 5-24). This plot illustrates the close correspondence between external and internal temperature. Both internal and external temperatures were used in data-analysis.

3. Gas Blanket Temperature Distribution

The 40% load point included a 100°F ΔT between inlet sodium and outlet steam temperature. It thus provided a good opportunity to test the effectiveness of the gas blanket, formed with the argon cover gas, which is intended to provide thermal protection for the upper tube sheet. The temperature profile in the gas blanket was plotted as shown in Figure 4-15; for most of its length the gas space was in thermal equilibrium with the steam outlet temperature, as was intended. This indicates that the gas blanket was very effective for thermal protection of the upper tube sheet. The temperature approached the sodium inlet temperature at the apparent sodium level.

Figure 4-14. Comparison of Internal and External Thermocouple Measurements

Figure 4-15. Temperature Profile in Gas Blanket
(40% rated power; May 15, 1969; 1530 hours)

F. LEVEL CONTROL

The steam generator sodium level was measured by two separate transducers, a float gage operating across the two level-sensing taps, and an Accuray radiation gage. As a matter of interest, these instruments exhibited a distinct bias in level height with respect to one another, possibly indicating that due to weir flow over the shroud, the radius-averaged level was below the float level. The level was controlled by a system which admitted cover gas to lower the level or vented gas to raise it, as required. Either of the two level instruments could be used as the feedback for this control system. The Accuray radiation gage tended to produce a noisy signal and the float gage was therefore used in preference.

A determining factor in setting the sodium level was the avoidance of gas entrainment. This phenomenon varied according to the height of sodium above the top of the inlet shroud. The effect of gas entrainment was not known; however, it was suggested that small or moderate amounts of entrained gas could affect heat transfer in the steam generator. Larger amounts could offset pump performance and render the entire secondary loop unstable. An investigation of gas entrainment was conducted in 1966 with sodium flowing in the secondary loop at minimum temperature and with the steam side dry. The report of this investigation is presented in Appendix B. It was found that with a level at 50% of the control range (6 in. above the lower level tap) significant gas entrainment could occur at approximately 1,100,000 lb/hr flow, while at 85% of the control range (10.2 in. above the lower level tap) significant gas entrainment could not occur until 1,250,000 lb/hr flow. Levels were set accordingly.

During the 1970 operating period, float gage operation became unsatisfactory, apparently caused by clogging of the float tube with sodium-water reaction products. There was an interval of erratic control on or about January 20, 1970. The radiation gage was used in its place.

V. THERMOHYDRAULIC PERFORMANCE

A. TEST PROGRAM

1. Planned Evaluation Tests

As part of their design and development task,⁽²⁾ Alco prepared a test program for combined operation of the steam generator and IHX. The program included steady-state tests, dynamic tests (start, load change, and shutdown), and transient tests (simulated reactor scram and other simulated casualties). The steady-state portion of the program was reviewed by AI, in its capacity as operator of SCTI, and by BLH and UNC. This review resulted in a final steady-state test program⁽³⁾ as shown in Tables 5-1 and 5-2. The principal objective of the tests was to obtain steam generator and IHX performance data which could serve to verify the thermal design of both units. Further review of the dynamic and transient tests was also planned but was not completed.

Rated water/steam conditions for the steam generator were 600°F inlet temperature, 2200 psig outlet pressure, and 1050°F outlet temperature. As shown in Table 5-1, the program included nine tests at these conditions. Tests A-1 through A-4 were to use unbalanced primary-to-secondary sodium flow while tests B-1 through B-5 were to use balanced flow. Unbalanced flow was defined as a primary-to-secondary flow ratio of 4/3. Balanced flow was defined as a one-to-one ratio. For the steam generator, unbalanced flow required a sodium inlet-outlet ΔT of 400°F, while balanced flow required 360°F ΔT at rated steam temperature. Table 5-2 shows the remaining tests for which the basic steam conditions were reduced below rated values. Tests C-1 through C-4 were at reduced superheat and reduced steam pressure, Tests D-1 through D-3 at reduced steam pressure, and Tests E-1 and E-2 at reduced feedwater temperature.

2. Plant Performance Tests

Before the evaluation tests could be started, it was necessary to conduct a plant performance test program which constituted the final checkout of the SCTI. These tests required operation of the steam generator, which couples the secondary sodium loop to the steam and feedwater loop. The first phase of the plant performance tests called for progressive rise of steam temperature at

TABLE 5-1
ALCO/BLH STEAM GENERATOR AND ALCO INTERMEDIATE HEAT EXCHANGER -
ESTIMATED STEADY-STATE CONDITIONS FOR UNBALANCED AND BALANCED FLOW TESTS

Test No. (1)	Case	Source	Water/Steam				Secondary Sodium				Primary Sodium				Tube Sheet ΔT 's (5)		
			Steam Pressure (psig)	Steam Flow (lb/hr)	Water Inlet (°F)	Steam Outlet (°F)	Flow Rate (lb/hr)	Temperature		ΔT (°F)	Flow Rate (lb/hr)	Temperature		ΔT (°F)	Steam Gen. Lower (°F)	IHX Lower (°F)	IHX Upper (°F)
								(°F)	(°F)			(°F)	(°F)				
A-1	Unbalanced flow, 20% load	(2)	2200	23,100	600	1050	171,000	1087	687	400	228,000	1089	789	300	87	102	2
		(3)	2200	23,100	600	1050	170,000	1065	665	400	227,000	1065	765	300	65	100	0
A-2	Unbalanced flow, 60% load	(2)	2200	69,400	600	1050	513,000	1134	734	400	683,000	1147	847	300	134	113	13
		(3)	2200	69,400	600	1050	510,000	1096	696	400	682,000	1100	800	300	96	104	4
A-3	Unbalanced flow, 100% load	(2)	2200	116,000	600	1050	855,000	1175	775	400	1,140,000	1200	900	300	175	125	25
		(3)	2200	116,000	600	1050	852,000	1128	728	400	1,138,000	1138	838	300	128	110	10
A-4	Unbalanced flow, 117% load	(2)	2200	140,000	600	1007	1,000,000	1170	770	400	1,333,000	1200	900	300	170	130	30
		(3)	2200	140,000	600	1007	995,000	1115	715	400	1,333,000	1129	829	300	115	114	14
		(3,4)	2200	136,000	600	1050	998,000	1140	740	400	1,334,000	1155	855	300	140	115	15
B-1	Balanced flow, 10% load	(3)	2200	11,550	600	1050	95,000	1055	695	360	95,000	1065	705	360	95	10	10
B-2	Balanced flow, 20% load	(2)	2200	23,100	600	1050	190,000	1074	714	360	190,000	1101	741	360	114	27	27
		(3)	2200	23,100	600	1050	189,000	1061	701	360	189,000	1075	715	360	101	14	14
B-3	Balanced flow, 60% load	(2)	2200	69,400	600	1050	570,000	1120	760	360	570,000	1168	808	360	160	48	48
		(3)	2200	69,400	600	1050	567,000	1087	727	360	567,000	1118	758	360	127	31	31
B-4	Balanced flow, 100% load	(2)	2200	119,000	600	1014	950,000	1135	775	360	950,000	1200	840	360	175	65	65
		(3)	2200	119,000	600	1014	948,000	1089	729	360	940,000	1133	773	360	129	44	44
		(3,4)	2200	116,000	600	1050	947,000	1116	756	360	947,000	1160	800	360	156	44	44
B-5	Balanced flow, 117% load	(2)	2200	141,000	585	965	1,110,000	1128	768	360	1,110,000	1200	840	360	183	72	72
		(3)	2200	141,000	585	965	1,103,000	1071	711	360	1,103,000	1120	760	360	126	49	49
		(3,4)	2200	136,000	600	1050	1,105,000	1127	767	360	1,105,000	1177	817	360	136	50	50

(1) Test designation in SCTI-C-1002 (internal operating procedure).

(2) Alco analysis data taken from Reference 2.

(3) Data calculated by UNC using corrected Alco methods.

(4) Steam outlet temperature raised to 1050 $^{\circ}$ F and steam flow changed to maintain same heat load.

(5) Tube sheet ΔT 's defined as the fluid terminal temperature differences.

TABLE 5-2
ALCO/BLH STEAM GENERATOR AND ALCO INTERMEDIATE HEAT EXCHANGER – ESTIMATED STEADY-STATE CONDITIONS
FOR REDUCED SUPERHEAT, REDUCED PRESSURE, AND REDUCED FEEDWATER TEMPERATURE TESTS

Test No. (1)	Case	Source	Water/Steam				Secondary Sodium				Primary Sodium				Tube Sheet ΔT 's (6)		
			Steam Pressure (psig)	Steam Flow (lb/hr)	Water Inlet (°F)	Steam Outlet (°F)	Flow Rate (lb/hr)	Temperature		ΔT (°F)	Flow Rate (lb/hr)	Temperature		ΔT (°F)	Steam Gen. Lower (°F)	IHX Lower (°F)	IHX Upper (°F)
								(°F)	(°F)			(°F)	(°F)				
C-1	Reduced superheat, 2200/700	(2)	2200	145,000	600	700	1,370,000	922	710	212					110		
		(3)	2200	145,000	600	700	1,365,000	870	658	212	1,365,000	904	692	212	58	34	34
C-2	Reduced superheat, 1800/672	(2)	1800	145,000	572	672	1,370,000	913	683	230					111		
		(3)	1800	145,000	572	672	1,364,000	860	630	230	1,364,000	897	667	230	58	37	37
C-3	Reduced superheat, 1400/638	(2)	1400	128,000	538	638	1,370,000	869	649	220					111		
		(3)	1400	128,000	538	638	1,340,000	817	597	220	1,340,000	852	632	220	59	35	35
C-4	Reduced superheat, 1000/596	(2)	1000	92,000	496	596	1,370,000	776	605	171					109		
		(3)	1000	92,000	496	596	1,330,000	724	553	171	1,330,000	751	580	171	57	27	27
D-1	Reduced pressure, 1800/950	(3)	1800	120,000	600	950	921,000	1030	670	360	921,000	1074	714	360	70	44	44
D-2	Reduced pressure, 1400/950	(3)	1400	109,500	550	950	923,000	1006	646	360	923,000	1050	690	360	96	44	44
D-3	Reduced pressure, 1000/950	(3)	1000	101,000	500	950	920,000	985	625	360	920,000	1029	669	360	125	44	44
E-1	Reduced feedwater temp (475°F)	(3)	2200	101,000	475	1050	969,000	1107	747	360	969,000	1152	792	360	272	45	45
		(3,4)	2200	101,000	475	1050	811,000	1124	694	430	811,000	1172	742	430	219	48	48
		(3,5)	2200	101,000	475	1050	696,000	1151	651	500	696,000	1201	701	500	176	50	50
E-2	Reduced feedwater temp (450°F)	(3)	2200	95,000	450	1050	936,000	1103	743	360	936,000	1147	787	360	293	44	44
		(3,4)	2200	95,000	450	1050	783,000	1120	690	430	783,000	1166	736	430	240	46	46
		(3,5)	2200	95,000	450	1050	673,000	1144	644	500	673,000	1193	693	500	194	49	49

(1) Test designation in SCTI-C-1002 (internal operating procedure).

(2) Alco analysis data taken from Reference 2.

(3) Data calculated by UNC using corrected Alco methods.

(4) Sodium temperature difference raised to 430°F to reduce steam generator tube sheet ΔT .

(5) Sodium temperature difference raised to 500°F to reduce steam generator tube sheet ΔT .

(6) Tube sheet ΔT 's defined as the fluid terminal temperature differences.

low power up to 1050°F rated temperature. The second phase called for progressive rise of plant power with unbalanced sodium flow and rated steam temperature and pressure as per tests A-1 thru A-4 of Table 5-1. It was planned to raise power, starting at 20% rated power, in 20% increments. Thus, in addition to the 20, 60, 100, and 117% load tests of Table 5-1, intermediate tests were planned at 40 and 80% load.

3. 2000-Hr Run

Prior to the start of the plant performance tests, LMEC was requested to interpose a 2000-hr run of the SCTI between the plant performance and steady-state evaluation tests. Selection of operating conditions was requested such that there would be minimum risk to the facility. Table 5-3 lists the conditions recommended for this run. Because of delays in completion of the plant performance tests it was not possible to accomplish the 2000-hr run at the time scheduled and plans for this run were cancelled.

TABLE 5-3
RECOMMENDED CONDITIONS FOR STEAM GENERATOR 2000-hr RUN

	Design Conditions	Recommended for 2000-hr Test
Power level (Mwt)	30	15
Steam pressure (psig)	2200	2200
Steam temperature (°F)	1050	950
IHX primary sodium inlet temperature (°F)	1175	1048
IHX primary sodium outlet temperature (°F)	775	724
Primary sodium flow (gpm)	2050	1200
Primary/secondary flow ratio	4/3	7/6
Terminal fluid ΔT :		
Steam generator lower tube sheet (°F)	175	62
IHX upper tube sheet (°F)	25	8
IHX lower tube sheet (°F)	125	62

4. System for Specifying Test Conditions

The following system of specifying test conditions was adopted by Alco for their original test program and generally retained in all subsequent revisions of the program. In this system the following parameters are fixed:

- 1) Feedwater flowrate
- 2) Feedwater inlet temperature
- 3) Steam outlet temperature
- 4) Steam pressure
- 5) Secondary sodium inlet-to-outlet temperature difference (steam generator)
- 6) Primary sodium inlet-to-outlet temperature difference (IHX).

Under this system the sodium inlet and outlet temperatures are not fixed but are permitted to find a level such that heat transfer is sufficient to produce the specified steam conditions. Primary and secondary sodium flowrates, while not basic test parameters, were inherently fixed by the power balance relationship (this assumes no appreciable sodium piping heat losses):

$$W_{Na} = \frac{(H_V)_O - (H_L)_I}{(c_p)_{Na} (\Delta T)_{Na}} \cdot W_L$$

where

$(H_V)_O$ = steam outlet enthalpy (Btu/lb)

$(H_L)_I$ = water inlet enthalpy (Btu/lb)

W_L = water flowrate (lb/hr)

$(\Delta T)_{Na}$ = sodium inlet-to-outlet temperature difference ($^{\circ}$ F)

W_{Na} = sodium flowrate (lb/hr)

$(c_p)_{Na}$ = specific heat of sodium (Btu/lb- $^{\circ}$ F).

This system was used for the entire evaluation test program and for all phases of the diagnostic test program except Series 5. In Series 5, sodium inlet and outlet temperatures were fixed, while steam outlet temperature was allowed to vary as a function of heat transfer.

5. Diagnostic Tests

During the plant performance tests, a decrease in heat-transfer performance was observed at 40% rated power compared to the performance exhibited at 20% power. It was then decided to defer the steady-state evaluation tests and start on an experimental program to investigate the heat-transfer performance problem. The diagnostic test program was accordingly formulated. Its objectives were as follows:

- 1) To confirm the decrease in performance observed in the plant performance tests with increasing power, and obtain better definition of the phenomenon.
- 2) To investigate the boiling phenomena associated with poor performance.
- 3) To determine the maximum power of the steam generator at the designed mode of operation.

The test program was made up of four test series as described in the following text.

a. Series 1

It was first suggested that the high sodium-to-water ΔT near the vapor end of the boiling zone might be the primary factor responsible for poor heat transfer (sodium-to-water ΔT refers to the temperature difference between sodium and water/steam at a given axial point of the heat-transfer zone). This high ΔT was an inherent feature when operating the unit at rated superheat. However, if the unit were operated as an evaporator it became possible to reduce this parameter. This concept is illustrated in Figure 5-1.

With all steam-side parameters (inlet and outlet temperatures, flowrate and pressure) fixed, the flexibility of the SCTI sodium loops permitted a further manipulation of the sodium-to-water ΔT . Constant power could be achieved with

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Figure 5-1. Sodium-Water ΔT Concept Used in Planning Diagnostic Tests

a wide range of sodium flowrate-sodium ΔT combinations (sodium ΔT refers to the sodium temperature difference between the inlet and outlet).

Approximating the specific heat of sodium as a constant, the system follows the relationship:

$$(\Delta T)_{Na} = \frac{P_w}{(c_p)_{Na} W_{Na}}$$

where P_w is the steam generator thermal power (Btu/hr). Thus, increasing sodium flow would tend to reduce sodium ΔT . The effect on sodium-to-water ΔT would be to reduce it at the vapor end of the boiling zone and raise it at the liquid end. This concept is illustrated in Figure 5-2.

Series 1 of the diagnostic tests was thus planned to investigate the effect of varying sodium-to-water ΔT by use of the above concepts. Steam outlet temperature was fixed at 700°F, allowing 50°F superheat over the 650°F saturation temperature at 2200 psia. This was considered the minimum superheat permissible for safe operation of the facility. Sodium ΔT was varied in four increments from 275 to 75°F. Feedwater flow was fixed at 46,900 lb/hr. Specified test conditions for Series 1 are given in Table 5-4.

b. Series 2

Series 2 of the diagnostic tests was intended to investigate the effect of water flowrate on heat-transfer performance. Test conditions were planned so as to minimize the sodium-to-water ΔT at the vapor end of the heat-transfer zone. As in Series 1, steam outlet temperature was fixed at 700°F. Sodium flow was fixed at 1,240,000 lb/hr, the maximum permitted by the facility. Sodium ΔT was varied as required to provide the power for each value of water flowrate. Water flowrate was increased in 3 increments from 78,000 to 140,000 lb/hr. Test 4 of Series 1, in which the water flowrate was 46,900 lb/hr, constituted in effect the initial test of this series, thus providing four increments of flowrate. Table 5-5 lists test conditions for this series.

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Figure 5-2. Concept of Variation of Sodium Temperature Difference at Constant Power

TABLE 5-4

DIAGNOSTIC TEST SERIES NO. 1 CONDITIONS - PLANNED

Test Number	1	2	3	4
Feedwater flow (lb/hr)	46,900	46,900	46,900	46,900
Feedwater inlet temperature (°F)	600	600	600	600
Steam outlet temperature (°F)	700	700	700	700
Steam pressure (psia)	2,200	2,200	2,200	2,200
Steam generator secondary sodium ΔT (°F)	275	200	150	75
Secondary sodium flow (lb/hr)	344,000	465,000	620,000	1,240,000
Primary sodium ΔT (°F)	207	150	113	72
Primary sodium flow (lb/hr)	459,000	620,000	826,000	1,300,000
Power (Mwt)	8.2	8.2	8.2	8.2

c. Series 3

Series 3 of the diagnostic tests was planned for the purpose of verifying the anomalous heat-transfer results observed in the plant performance test. It was planned to start the series at 20% power and increase power in 5% increments until the plant limitation of 1200°F sodium temperature was approached. The standard operating conditions were planned as per the unbalance flow phase of the evaluation program and the power-rise phase of the plant performance tests. Steam outlet temperature was thus fixed at 1050°F and secondary sodium ΔT at 400°F. Table 5-6 lists the specified conditions for Series 3 covering the 7 power increments from 20 to 50% rated power. The test plan provided, however, that the series should not proceed beyond the test condition which would result in a steam generator sodium inlet temperature of 1175°F.

TABLE 5-5

DIAGNOSTIC TEST SERIES NO. 2 CONDITIONS - PLANNED

Test Number	5	6	7
Feedwater flow (lb/hr)	78,000	109,000	140,000
Feedwater inlet temperature (°F)	600	600	600
Steam outlet temperature (°F)	700	700	700
Steam pressure (psia)	2,200	2,200	2,200
Steam generator secondary sodium ΔT (°F)	125	175	225
Secondary sodium flow (lb/hr)	1,240,000	1,240,000	1,240,000
Primary sodium ΔT (°F)	120	168	214
Primary sodium flow (lb/hr)	1,300,000	1,300,000	1,300,000
Power (Mwt)	13.7	19.2	24.7

d. Series 4

In Series 1, it was planned that the series would start with sodium ΔT , the independent variable, at its maximum value and the parameter would be progressively reduced. It was considered possible that the direction of change of the sodium ΔT might be significant in its effect on heat transfer. For this reason, Series 4 was planned as a reverse execution of Series 1, starting with sodium ΔT at its minimum value and proceeding to progressively higher values of this parameter. Test conditions for Series 4 are listed in Table 5-7.

6. Followup Diagnostic Tests

After observing the performance decrease exhibited in Series 1 of the diagnostic tests, it was decided that a more extensive test program was required to define the envelope of conditions under which the phenomenon of performance degradation would take place. The operating variables which might be expected

TABLE 5-6

DIAGNOSTIC TEST SERIES NO. 3 CONDITIONS - PLANNED

Test Number	8	9	10	11	12	13	14
Feedwater flow (lb/hr)	23,400	29,200	35,100	41,000	46,900	52,700	58,600
Feedwater inlet temperature (°F)	600	600	600	600	600	600	600
Steam outlet temperature (°F)	1,050	1,050	1,050	1,050	1,050	1,050	1,050
Steam pressure (psia)	2,200	2,200	2,200	2,200	2,200	2,200	2,200
Steam generator secondary sodium ΔT (°F)	400	400	400	400	400	400	400
Secondary sodium flow (lb/hr)	170,000	213,000	257,000	300,000	344,000	387,000	430,000
Primary sodium ΔT (°F)	300	300	300	300	300	300	300
Primary sodium flow (lb/hr)	227,000	284,000	343,000	400,000	459,000	516,000	573,000
Power (Mwt)	6.0	7.5	9.0	10.5	12.0	13.5	15.0

TABLE 5-7

DIAGNOSTIC TEST SERIES NO. 4 CONDITIONS - PLANNED

Test Number	15	16	17	18
Feedwater flow (lb/hr)	46,900	46,900	46,900	46,900
Feedwater inlet temperature ($^{\circ}$ F)	600	600	600	600
Steam outlet temperature ($^{\circ}$ F)	700	700	700	700
Steam pressure (psia)	2,200	2,200	2,200	2,200
Steam generator secondary sodium ΔT ($^{\circ}$ F)	75	150	200	275
Secondary sodium flow (lb/hr)	1,240,000	620,000	465,000	344,000
Primary sodium ΔT ($^{\circ}$ F)	72	113	150	207
Primary sodium flow (lb/hr)	1,300,000	826,000	620,000	459,000
Power (Mwt)	8.2	8.2	8.2	8.2

to affect performance decrease included water flowrate, superheat, sodium outlet temperature and feedwater inlet temperature.

The information required to understand the observed performance changes included answers to the following questions:

- 1) Can satisfactory heat-transfer performance be obtained at high superheat by maintaining a low sodium outlet temperature?
- 2) How does the heat-transfer response to decreasing sodium ΔT vary with feedwater flowrate?
- 3) How does the heat-transfer response to decreasing sodium ΔT vary with steam superheat?
- 4) Can changes in feedwater temperature produce a similar heat-transfer effect to changes in sodium ΔT ?
- 5) How does level of heat transfer vary with steam pressure?

To answer the above questions, it was planned to apply several different modes of changing operating conditions:

- 1) Starting at minimum superheat, increase superheat by increasing sodium inlet temperature, while maintaining constant sodium outlet temperature. This requires that plant power and sodium flow be adjusted simultaneously.
- 2) Holding plant power constant (steam outlet temperature approximately constant), increase sodium weight flow thus decreasing sodium ΔT and increasing sodium outlet temperature. This mode of change was followed in Series 1.
- 3) Holding steam outlet temperature constant at the minimum superheat level, and holding sodium flow constant at maximum level, increase feedwater flow thus increasing power. Raise sodium ΔT as required to meet increased power. This mode was previously used in Series 2.
- 4) Holding sodium outlet temperature and steam outlet temperature constant, vary feedwater inlet temperature.

The followup diagnostic test program consisted of five series in which the above modes of change were to be applied over a range of steam-side conditions. The tests were planned such that wherever possible the final test point of a series would constitute the initial point of the next series. Table 5-8 describes the program in planned sequence of tests.

The program was also designed so that the data to be obtained could be combined with the data from the previous diagnostic tests to form a coherent data matrix. Table 5-9 presents the followup and original diagnostic tests organized in terms of the variables being tested.

Tables 5-10 through 5-18 list the planned conditions for the individual test series.

7. Steady-State Criterion

To obtain valid steady-state performance data, a specific criterion for steady-state operation was employed in the SCTI tests. The criterion was that none of the steam generator or IHX terminal temperatures changes by more

TABLE 5-8
FOLLOWUP DIAGNOSTIC TEST PLAN
(Sheet 1 of 2)

Test Series	Constant Parameters	Value	Variable Parameters				Description	Remarks
			Controlled	Value	Floating	Value*		
5A	Water inlet temperature	600°F	Sodium inlet temperature	800°F	Steam outlet temperature	700°F	Raise sodium inlet temperature and power, thus raising steam outlet temperature. Control sodium outlet temperature. This mode of raising power expected to minimize chance of flip.	Expect to avoid flip by holding sodium outlet temperature down as superheat rises.
	Sodium outlet temperature	650°F		875		775		
	Water flowrate	47,000 lb/hr		950		845		
	Steam pressure	2,200 psia		1025		920		
				1100		990		
5B	Water inlet temperature	600°F	Sodium inlet-outlet ΔT	450°F	Sodium inlet and outlet temperature		Final test point of Series 5A is initial test point of 5B. In 5B, sodium flowrate is increased at constant power, thus reducing sodium ΔT . This mode of changing sodium-side conditions expected to maximize chance of flip.	Will observe whether raising sodium outlet temperatures induces flip at high superheat (as has already been observed at low superheat).
	Steam outlet temperature	Final value of 5A		375				
	Water flowrate	47,000 lb/hr		300				
	Steam pressure	2,200 psia		225				
6A	Water flowrate	47,000 lb/hr	Water inlet temperature	600°F	Sodium inlet temperature	905	Establish initial condition (600°F water inlet temperature, 700°F steam outlet temperature, 625°F sodium outlet temperature). Lower water inlet temperature. Increase power to maintain 700°F steam outlet temperature. Regulate sodium flow to maintain 625°F sodium outlet temperature. After 500°F water inlet temperature point is reached, reverse the above procedure to raise water inlet temperature back to 550°F.	Observe effect of feedwater inlet temperature on heat transfer. Sodium outlet temperature (the other feedwater end variable) held constant.
	Steam outlet temperature	700°F		575				
	Sodium outlet temperature	625°F		550		850		
	Steam pressure	2,200 psia		525		825		
				500				
				525				
				550		850		
6B	Water flowrate	47,000 lb/hr	Sodium inlet-outlet ΔT	225°F	Sodium inlet and outlet temperature		Final point of 6A is initial point of 6B. Reduce sodium ΔT as in Group 5B.	Observe whether flip is induced by raising sodium outlet temperature at lower feedwater inlet temperature (550°F vs rated 600°F).
	Steam outlet temperature	700°F		185				
	Water inlet temperature	550°F		145				
	Steam pressure	2,200 psia		120				
7	Water inlet temperature	550°F	Water flowrate	47,000 (initial)	Sodium inlet temperature		Establish initial point (final point of 6B may satisfy all conditions), increase water flow. Increase power to maintain 700°F steam outlet temperature. Regulate sodium flow to maintain sodium outlet temperature under 665°F.	Observe effect on heat transfer of increased water flowrate at low feedwater temperature.
	Steam outlet temperature	(keep below 665°F)						
	Sodium outlet temperature							
	Steam pressure	2,200 psia		92,000 (final)				

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TABLE 5-8
FOLLOWUP DIANOSTIC TEST PLAN
(Sheet 2 of 2)

Test Series	Constant Parameters	Value	Variable Parameters				Description	Remarks
			Controlled	Value	Floating	Value*		
8A	Steam outlet temperature Sodium outlet temperature Water flowrate Steam pressure	700°F 665°F 92,000 lb/hr 2,200 psia	Water inlet temperature	550°F 575 600	Sodium inlet temperature	870 (final)	Raise water inlet temp. Final point of 7 is initial point of 8A. Decrease power to maintain 700°F steam outlet temp. Regulate sodium flow to maintain 665°F sodium outlet temp.	Observe response to changing feedwater inlet temperature at high water flowrate (vs lower water flowrate in Group 6A).
8B	Water inlet temperature Sodium outlet temperature Water flowrate Steam pressure	600°F 665°F 92,000 lb/hr 2,200 psia	Sodium inlet temperature	870°F 950 1030 1115	Steam outlet temperature	700 915	Final point of 8A is initial point of 8B. Raise sodium inlet temperature as in 5A.	Parallels 5A (raise superheat at constant sodium outlet temperature at high feedwater flowrate).
8C	Water inlet temperature Steam outlet temperature Water flowrate Steam pressure	600°F (final value of 8B) 92,000 lb/hr 2,200 psia	Sodium inlet-outlet ΔT	450°F 380 310 240 175	Sodium inlet and outlet temperature		Final point of 8B is initial point of 8C. Reduce sodium ΔT as in Groups 5B and 6B.	Parallels 5B (raise sodium outlet temperature at high feedwater flowrate).
9	Water inlet temperature Water flowrate Sodium inlet-outlet ΔT	575°F 92,000 lb/hr 175°F	Steam pressure (psia)	2200 2100 2000 1900 1800	Sodium inlet and outlet temperature.		Establish initial condition by lowering water inlet temperature after final point of 8C at constant power, then lower steam pressure.	Tests effect of reduced pressure. (Data useful if orificing required.)

*Prediction only. Not a specified test parameter.

TABLE 5-9

DIAGNOSTIC TEST PLAN LOGIC TABLE

Mode of Changing Operating Conditions in Series	Fixed Conditions for Series			
	Series No.*	Feedwater Inlet Temp (°F)	Feedwater Flowrate (lb/hr)	Steam Superheat
1) Raise sodium inlet temp and steam outlet temp, holding sodium outlet temp constant	5 8B	600 600	47,000 92,000	- -
2) Reduce sodium ΔT (thus raise sodium outlet temp) at constant power	1 5B 6B 8C	600 600 550 600	47,000 47,000 47,000 92,000	Minimum High Minimum High
3) Change feedwater temperature	6A 8A	- -	47,000 47,000	Minimum Minimum
4) Increase feedwater flow at constant sodium flow	2 7	600 550	- -	Minimum Minimum

*Actual test numbers (see Section V.B.2)

TABLE 5-10

DIAGNOSTIC TEST SERIES NO. 5A CONDITIONS - PLANNED

Test Number	19	20	21	22	23
Feedwater flow (± 300 lb/hr)	47,000	47,000	47,000	47,000	47,000
Feedwater inlet temp ($\pm 5^\circ\text{F}$)	600	600	600	600	600
Steam outlet temp ($\pm 5^\circ\text{F}$)	700	← FLOATING VALUE →			
Steam pressure (± 10 psig)	2,200	2,200	2,200	2,200	2,200
Steam generator secondary sodium outlet temp ($\pm 5^\circ\text{F}$)	650	650	650	650	650
Steam generator secondary sodium inlet temp ($\pm 10^\circ\text{F}$)	As required	875	950	1,025	1,100

TABLE 5-11

DIAGNOSTIC TEST SERIES NO. 5B CONDITIONS - PLANNED

Test Number	23 [*]	24	25	26
Feedwater flow (± 60 lb/hr)	47,000	47,000	47,000	47,000
Feedwater inlet temperature ($\pm 5^\circ\text{F}$)	600	600	600	600
Steam outlet tem- perature ($\pm 5^\circ\text{F}$)	†	†	†	†
Steam pressure (± 10 psig)	2,200	2,200	2,200	2,200
Steam generator secondary sodium ΔT ($\pm 5^\circ\text{F}$)	450	375	300	225

*Final test point of Test Series 5A

†Value of Test 23 from previous series

TABLE 5-12

DIAGNOSTIC TEST SERIES NO. 6A CONDITIONS - PLANNED

Test Number	27	28	29	30	31	32	33
Feedwater flow (± 300 lb/hr)	47,000	47,000	47,000	47,000	47,000	47,000	47,000
Feedwater inlet temperature ($\pm 5^\circ\text{F}$)	600	575	550	525	500	525	550
Steam outlet tem- perature ($\pm 5^\circ\text{F}$)	700	700	700	700	700	700	700
Steam pressure (± 10 psig)	2,200	2,200	2,200	2,200	2,200	2,200	2,200
Steam generator secondary sodium outlet temperature ($\pm 5^\circ\text{F}$)	625	625	625	625	625	625	625
Sodium inlet tem- perature ($\pm 5^\circ\text{F}$)	905		850 [*]		825 [*]		850 [*]

*Estimated only

TABLE 5-13

DIAGNOSTIC TEST SERIES NO. 6B CONDITIONS - PLANNED

Test Number	33*	34	35	36
Feedwater flow (± 300 lb/hr)	47,000	47,000	47,000	47,000
Feedwater inlet temperature ($\pm 5^\circ\text{F}$)	550	550	550	550
Steam outlet tem- perature ($\pm 5^\circ\text{F}$)	700	700	700	700
Steam pressure (± 10 psig)	2,200	2,200	2,200	2,200
Steam generator secondary sodium ΔT ($\pm 5^\circ\text{F}$)	225	185	145	120

*Final test point of Test Series 6A

TABLE 5-14

DIAGNOSTIC TEST SERIES NO. 7 CONDITIONS - PLANNED

Test Number	36*	37	38	39
Feedwater flow (± 300 lb/hr)	47,000	†	†	92,000
Feedwater inlet temperature ($\pm 5^\circ\text{F}$)	550	550	550	550
Steam outlet tem- perature ($\pm 5^\circ\text{F}$)	700	700	700	700
Steam pressure (± 10 psig)	2,200	2,200	2,200	2,200
Steam generator secondary sodium outlet temperature ($^\circ\text{F}$)	<665	<665	<665	<665

*Final test point of Test Series 6B

†To be determined during test operation

TABLE 5-15
DIAGNOSTIC TEST SERIES NO. 8A CONDITIONS –
PLANNED

Test Number	39*	40	41
Feedwater flow (± 300 lb/hr)	92,000	92,000	92,000
Feedwater inlet temperature ($\pm 5^\circ\text{F}$)	550	575	600
Steam outlet tem- perature ($\pm 5^\circ\text{F}$)	700	700	700
Steam pressure (± 10 psig)	2,200	2,200	2,200
Steam generator secondary sodium outlet temperature ($\pm 5^\circ\text{F}$)	665	665	665

*Final test point of Test Series 7

TABLE 5-16
DIAGNOSTIC TEST SERIES NO. 8B CONDITIONS – PLANNED

Test Number	41*	42	43	44
Feedwater flow (± 300 lb/hr)	92,000	92,000	92,000	92,000
Feedwater inlet temperature ($\pm 5^\circ\text{F}$)	600	600	600	600
Steam pressure (± 10 psig)	2,200	2,200	2,200	2,200
Steam generator secondary sodium outlet temperature ($\pm 5^\circ\text{F}$)	665	665	665	665
Steam generator secondary sodium inlet temperature ($\pm 5^\circ\text{F}$)	870	950	1,030	1,115

*Final test point of Test Series 8A

TABLE 5-17

DIAGNOSTIC TEST SERIES NO. 8C CONDITIONS - PLANNED

Test Number	44*	45	46	47
Feedwater flow (±60 lb/hr)	92,000	92,000	92,000	92,000
Feedwater inlet temp (±5°F)	600	600	600	600
Steam outlet temp (±5°F)	†	†	†	†
Steam pressure (±10 psig)	2,200	2,200	2,200	2,200
Steam generator secondary sodium ΔT (±5°F)	450	450	450	450
Sodium flow (lb/hr)§	4.48×10^5	5.94×10^5	7.30×10^5	9.39×10^5
Sodium inlet temp (±5°F)§	1,115	1,055	995	935
Sodium outlet temp (±5°F)§	665	675	685	695

*Final test point of Test Series 8B.

†Final point of 8B - calculated to be 840°F.

§Target values only - not absolutely necessary to satisfy test conditions.

TABLE 5-18

DIAGNOSTIC TEST SERIES NO. 9 CONDITIONS - PLANNED

Test Number	48	49	50	51	52
Feedwater flow (±300 lb/hr)	92,000	92,000	92,000	92,000	92,000
Feedwater inlet (±5°F)	575	575	575	575	575
Steam pressure (±25 psig)	2,200	2,100	2,000	1,900	1,800
Steam generator secondary sodium outlet temp	*	*	*	*	*

*Value of last data point in Test Series 8C.

than $\pm 5^{\circ}\text{F}$ within a 10-min period. This criterion was required for each test point of the planned steady-state tests. The test procedure required that each test point be maintained for 1 hr after being attained.

Parameters covered by this criterion were the feedwater inlet temperature, steam outlet temperature, secondary sodium hot-side temperature (steam generator shell-side inlet and IHX tube-side outlet), secondary sodium cold-side temperature (steam generator shell-side outlet and IHX tube-side inlet), the primary sodium hot-side temperature (IHX shell-side inlet) and primary sodium cold-side temperature (IHX shell-side outlet).

8. Performance Tests with Inlet Orifices

Following the decision to modify the steam generator with tube inlet orifices, a test program was planned for the modified steam generator. The test plan was as follows:

- 1) Repeat Diagnostic Test Series 3, carrying power level to 100% if possible with conditions as in Table 5-6.
- 2) Repeat Diagnostic Test Series 1 with conditions as in Table 5-4. This plan was accompanied by a decision to defer those phases of the diagnostic test program which were not yet performed (Series 6 through 9).

B. HEAT-TRANSFER RESULTS

1. Test Chronology

Of the planned tests listed in the previous sections, the following were accomplished.

- 1) The temperature rise phase of the plant performance test was completed. No steady-state data were recorded for this phase.
- 2) The 20 and 40% levels of the power rise phase of the plant performance test were conducted on May 14 and 15, 1969. The 40% test was repeated on June 21, 1969. The tube leak detected on June 22, 1969, precluded reaching higher power levels.
- 3) Series 1 through 4 of the originally planned diagnostic tests and Series 5 (including Subseries 5A and 5B) of the followup diagnostic tests were

conducted between December 31, 1969, and February 27, 1970. There was considerable repetition of test points for reasons which included failure to meet target test conditions, test facility problems, and failure to record required data due to internal thermocouple failure.

Table 5-19 lists the tests in chronological order.

2. Steady-State Heat-Transfer Results

Overall steam generator and IHX performance at each test point was determined by reference to the outputs of the water flowmeter and the sodium and water/steam temperature measurements. Secondary sodium flow was calculated by heat balance with steam-side and secondary sodium inlet and outlet temperature measurements. Primary sodium flow was calculated by heat balance with secondary flow and the terminal temperatures. This method of calculation avoids placing reliance on the sodium EM flowmeters which had not received a satisfactory calibration.

a. Plant Performance Tests

Table 5-20 lists actual operating conditions for the nominal 20 and 40% power points of the plant performance tests. The second 40% point reached on June 21, 1969, which was similar to the first, is not listed.

IHX shell-side inlet and steam generator sodium inlet temperatures measured in these tests were compared with the values of these temperatures predicted by the Alco design method, which were listed in Table 5-1. The comparison is plotted in Figure 5-3. The limiting temperature for the IHX and the primary sodium loop was 1200°F. (There was no specific limit for the secondary loop but it must inherently be less than the primary loop.) As shown in Figure 5-3 it appeared that the inlet temperatures approached the 1200°F limit at slightly over 40% rated power. It was concluded that it would not be possible to achieve 100% rated power using the test methods proposed by Alco.

b. Diagnostic Tests

While the diagnostic tests were originally planned as steady-state tests, execution of the planned progression of test conditions resulted in several abrupt performance decreases (flips) or increases (reverse flips). The data for these incidents have been treated as valid test results and will be presented at their logical position with respect to the test conditions.

TABLE 5-19
CHRONOLOGICAL SUMMARY OF TEST POINTS
(Sheet 1 of 2)

Date	Time (hr)	Series	Independent Parameter	Test No.	Value of Independent Parameter
Plant Performance Tests					
5-14-69	1330	-	Power (% rated)	Nominal 20%	18.5
5-15-69	1530	-	Power (% rated)	Nominal 40%	38.1
Diagnostic Tests					
12-31-69	1530	1	Sodium ΔT ($^{\circ}F$)	1A	323
1-1-70	1200	1	Sodium ΔT ($^{\circ}F$)	2	194
1-1-70	2130	1	Sodium ΔT ($^{\circ}F$)	3	147
1-2-70	0330	1	Sodium ΔT ($^{\circ}F$)	Flip (pre)	86
1-2-70	0445	1	Sodium ΔT ($^{\circ}F$)	Flip (post)	77
1-9-70	1600	1	Sodium ΔT ($^{\circ}F$)	4A	95
1-9-70	1745	1	Sodium ΔT ($^{\circ}F$)	Flip (pre)	91
1-9-70	1930	1	Sodium ΔT ($^{\circ}F$)	Flip (post)	79
1-10-70	1430	2	Water flow (lb/hr)	5A	75,300
1-19-70	2315	1	Sodium ΔT ($^{\circ}F$)	1	280
1-20-70	0430	1	Sodium ΔT ($^{\circ}F$)	2R	151
1-21-70	1030	2	Water flow (lb/hr)	5AR	77,190
1-21-70	1745	2	Water flow (lb/hr)	5B	91,200
1-28-70	1415	2	Water flow (lb/hr)	6A	99,900
1-28-70	1950	2	Water flow (lb/hr)	6B	114,800
1-28-70	2330	2	Water flow (lb/hr)	7A	124,100
2-6-70	1405	3	Power (% rated)	8	20.3
2-8-70	0300	3	Power (% rated)	9	24.2
2-17-70	2314	3	Power (% rated)	8R	19.6
2-18-70	0630	3	Power (% rated)	9R	25.3
2-18-70	1515	3	Power (% rated)	10B	32.2
2-19-70	0415	3	Power (% rated)	12	40.2

TABLE 5-19

CHRONOLOGICAL SUMMARY OF TEST POINTS
(Sheet 2 of 2)

Date	Time (hr)	Series	Independent Parameter	Test No.	Value of Independent Parameter
Diagnostic Tests (cont)					
2-24-70	0825	4	Sodium ΔT ($^{\circ}F$)	15A	91
2-24-70	1230	4	Sodium ΔT ($^{\circ}F$)	15	85
2-24-70	1900	4	Sodium ΔT ($^{\circ}F$)	Flip (pre)	83
2-24-70	2345	4	Sodium ΔT ($^{\circ}F$)	Flip (post)	71
2-25-70	0310	4	Sodium ΔT ($^{\circ}F$)	Reverse flip (pre)	116
2-25-70	0515	4	Sodium ΔT ($^{\circ}F$)	Reverse flip (post)	173
2-25-70	0655	4	Sodium ΔT ($^{\circ}F$)	16	147
2-25-70	1055	4	Sodium ΔT ($^{\circ}F$)	17	199
2-25-70	1609	4	Sodium ΔT ($^{\circ}F$)	18	274
Followup Diagnostic Tests					
2-25-70	2335	5A	Sodium inlet temp ($^{\circ}F$)	19	821
2-26-70	0420	5A	Sodium inlet temp ($^{\circ}F$)	20	873
2-26-70	1210	5A	Sodium inlet temp ($^{\circ}F$)	21	948
2-26-70	2345	5A	Sodium inlet temp ($^{\circ}F$)	22	1024
2-27-70	0545	5A	Sodium inlet temp ($^{\circ}F$)	23	1100
2-27-70	1000	5B	Sodium ΔT ($^{\circ}F$)	Flip (pre)	401
2-27-70	1040	5B	Sodium ΔT ($^{\circ}F$)	Flip (post)	377

TABLE 5-20
PLANT PERFORMANCE TESTS
TARGET 20 AND 40% RATED POWER
OPERATING CONDITIONS

Operating Conditions	Target - 20% Rated Power	Target - 40% Rated Power
Test Date	5-14-69	5-15-69
Test Time	1330	1530
Water Conditions:		
Flow (lb/hr)	21,470	44,800
Steam temp (out) (°F)	1,049	1,032
Water temp (in) (°F)	602	602
Pressure (psia)	2,200	2,205
Secondary Sodium Conditions for Steam Generator:		
Flow (lb/hr)	154,400	327,800
SG inlet (IHX outlet) temp (°F)	1,065	1,150
SG outlet (IHX inlet) temp (°F)	660	755
ΔT (°F)	405	395
Primary Sodium Conditions for IHX:		
Flow (lb/hr)	196,100	406,800
IHX inlet temp (°F)	1,070	1,152
IHX outlet temp (°F)	750	833
ΔT (°F)	320	319
Power (Mwt)*	5.6	11.4
Percent Rated Power	18.5	38.1

*Independent parameter of the test series

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Figure 5-3. Predicted vs Actual Sodium Inlet Temperature

(1) Series 1

In terms of sodium ΔT , the independent variable, the following test points were achieved:

<u>Test Point</u>	<u>Sodium ΔT (°F)</u>
1A	323
1	280
2	194
2R	195
3	147
4A	95

The letters A and R in the test numbers refer to alternate and repeated runs. Table 5-21 lists terminal parameters for these points. Further reduction of sodium ΔT below 95°F resulted in a sudden performance decrease. There were two executions of this test in the low ΔT range. In the first instance, the

TABLE 5-21
DIAGNOSTIC TESTS
TEST SERIES 1
RESULTS OF TESTS 1A-4A

Operating Conditions	1A	1	2	2R	3	4A
Test Date	12-31-69	1-19-70	1-1-70	1-20-70	1-1-70	1-9-70
Test Time	1530	2315	1200	0430	2130	1600
Water Conditions:						
Flow (lb/hr)	44,670	48,270	45,050	48,150	44,680	44,680
Steam temp (out) (°F)	705	709	703	700	700	700
Water temp (in) (°F)	595	597	600	604	599	598
Pressure (psia)	2200	2205	2210	2210	2210	2230
Secondary Sodium Conditions for Steam Generator:						
Flow (lb/hr)*	280,500	350,400	461,400	482,800	600,400	926,800
SG inlet (IHX outlet) temp (°F)	941	909	838	842	800	770
SG outlet (IHX inlet) temp (°F)	618	629	644	647	653	675
ΔT (°F)*	323	280	194	195	147	95
Primary Sodium Con- ditions for IHX:						
Flow (lb/hr)	390,700	458,000	648,300	622,700	782,400	1,061,500
IHX inlet temp (°F)	952	915	842	842	801	770
IHX outlet temp (°F)	719	700	704	691	688	687
ΔT (°F)	233	215	138	151	113	83
Power (Mwt)	8.1	8.8	8.0	8.4	7.9	7.9
Percent Rated Power	27.0	29.2	26.7	28.1	26.4	26.3

*Independent parameters of the test series

incident occurred at 86°F sodium ΔT , in the second instance at 91°F. Figure 5-4 is a plot of principal steam generator performance parameters vs time covered from 1 hr before to 1 hr after the first occurrence. Table 5-22 lists terminal performance data approximately 1/2 hr before and after the event; the closest points at which reasonably steady data could be obtained.

Further reduction of sodium ΔT following occurrence of the flips was not attempted and the target value of 75°F for the final test point was not achieved.

(2) Series 2

Series 2 had been planned to use the same sodium flowrate as the final point of Series 1 which constituted its initial point. It was decided to revise Series 2 to use the sodium flowrate of test point 4A. This was the highest sodium flowrate (lowest sodium ΔT) of Series 1 at which satisfactory heat-transfer performance was achieved. This required a sodium flowrate approximately 80% of that originally planned. Series 2 was executed smoothly, yielding five usable test points. Test 4A of Series 1 also served as a point of this series. In terms of the independent parameter, water flowrate, they are as follows:

<u>Test Point</u>	<u>Water Flow (lb/hr)</u>
4A	44,680
5A	75,300
5A-R	77,190
5B	91,200
6A	99,900
6B	114,800
7A	124,100

Table 5-23 lists terminal performance parameters for these tests.

(3) Series 3

The first attempt to execute Series 3 was marred by failure of many internal thermocouples. The series was repeated after installation of substitute external thermocouples.

Increase of power was terminated at the 40% point when primary sodium inlet temperature reached 1176°F. At this point, sodium flow was reduced, at

Figure 5-4. Time Plot for Sudden Performance Decrease in Test Series 1 (1-2-70: 0410 hours)

TABLE 5-22
DIAGNOSTIC TESTS
TEST SERIES 1
PERFORMANCE CHANGES

Operating Conditions	Before Performance Decrease		After Performance Decrease	
Test Date	1-2-70	1-9-70	1-2-70	1-9-70
Test Time	0330	1745	0445	1930
Water Conditions:				
Flow (lb/hr)	44,180	44,670	44,430	43,440
Steam temp (out) (°F)	697	704	658	713
Water temp (in) (°F)	598	594	598	597
Pressure (psia)	2210	2215	2205	2240
Secondary Sodium Conditions for Steam Generator:				
Flow (lb/hr)*	1,010,000	1,000,500	1,008,000	1,122,000
SG inlet (IHX outlet) temp (°F)	756	769	800	817
SG outlet (IHX inlet) temp (°F)	670	678	720	729
ΔT (°F)*	86	91	80	88
Primary Sodium Conditions for IHX:				
Flow (lb/hr)	1,008,900	988,900	969,600	1,006,900
IHX inlet temp (°F)	767	775	805	817
IHX outlet temp (°F)	681	685	728	738
ΔT (°F)	86	90	77	79
Power (Mwt)	7.8	8.1	6.9	7.9
Percent Rated Power	26.0	26.9	23.1	26.4

*Independent parameters of the test series

TABLE 5-23
DIAGNOSTIC TESTS
TEST SERIES 2
RESULTS OF TESTS 5A-7A

Operating Conditions	5A	5AR	5B	6A	6B	7A
Test Date	1-10-70	1-21-70	1-21-70	1-28-70	1-28-70	1-28-70
Test Time	1430	1030	1745	1415	1950	2330
Water Conditions:						
Flow (lb/hr)*	75,300	77,190	91,200	99,900	114,800	124,100
Steam temp (out) (°F)	694	699	697	712	704	702
Water temp (in) (°F)	592	602	600	608	601	601
Pressure (psia)	2,200	2,215	2,210	2,250	2,250	2,250
Secondary Sodium Conditions for Steam Generator:						
Flow (lb/hr)	898,600	945,200	928,200	969,300	966,700	948,000
SG inlet (IHX outlet) temp (°F)	836	830	863	884	913	934
SG outlet (IHX inlet) temp (°F)	670	670	670	680	678	676
ΔT (°F)	166	160	193	204	235	258
Primary Sodium Con- ditions for IHX:						
Flow (lb/hr)	1,176,700	901,200	911,000	961,900	945,000	1,060,900
IHX inlet temp (°F)	837	847	887	910	950	944
IHX outlet temp (°F)	710	679	690	704	790	713
ΔT (°F)	127	168	197	206	241	231
Power (Mwt)	13.4	13.5	16.0	17.7	20.3	21.8
Percent Rated Power	44.5	45.1	53.4	58.0	67.5	72.7

*Independent parameter of the test series

constant power, to increase secondary sodium ΔT from 400 to 430°F, on the possibility that improved heat transfer might result. However, this resulted in raising the primary inlet temperature to 1198°F, only 2°F below the 1200°F operating limit. The test series was then terminated. The series yielded six data points. In terms of the independent parameter (power) and the limiting parameter of secondary sodium inlet temperature, they are as follows:

Test Point	Rated Power (%)	Power (Mwt)	Steam Generator Sodium Inlet Temperature (°F)
8R	19.6	5.9	1068
8	20.3	6.1	1065
9	24.2	7.3	1093
9R	25.3	7.6	1114
10B	32.2	9.7	1146
12	40.2	12.1	1176

Table 5-24 lists terminal parameters for this series. It is noted that 40.2% rated power obtained in this series is the maximum power achieved using the operating conditions of the originally planned evaluation test program.

(4) Series 4

Following preliminary evaluation of Series 1, it was decided that, in order to meet the objective of Series 4, the series should be started with the post flip or low heat-transfer condition reached at the end of Series 1. The increase of sodium ΔT would ensue from that condition. For this purpose it was planned to first establish conditions as per point 4A of Series 1, then reduce sodium ΔT to induce a flip to provide the starting point for the series.

Results were somewhat unexpected, causing further deviations from the original test plan. The sequence of events of this series was as follows:

- 1) Good performance was established per the revised plan with sodium ΔT of 94°F.
- 2) Reduction of sodium ΔT to about 85°F failed to induce a flip. ΔT was raised back to about 90°F.

TABLE 5-24
DIAGNOSTIC TESTS
TEST SERIES 3
RESULTS OF TESTS 8-12

Operating Conditions	8	9	8R	9R	10B	12
Test Date	2-6-70	2-8-70	2-17-70	2-18-70	2-18-70	2-19-70
Test Time	1405	0300	2314	0630	1515	0415
Water Conditions:						
Flow (lb/hr)	23,579	28,667	22,710	29,412	37,230	46,662
Steam temp (out) (°F)	1,050	1,040	1,050	1,043	1,053	1,053
Water temp (in) (°F)	603	610	600	600	600	603
Pressure (psia)	2,189	2,209	2,214	2,219	2,214	2,224
Secondary Sodium Conditions for Steam Generator:						
Flow (lb/hr)	169,404	202,312	168,134	220,614	297,810	345,461
SG inlet (IHX outlet) temp (°F)	1,065	1,093	1,068	1,114	1,146	1,176
SG outlet (IHX inlet) temp (°F)	660	688	673	725	755	780
ΔT (°F)	405	405	395	389	391	396
Primary Sodium Conditions for IHX:						
Flow (lb/hr)	239,441	240,287	236,536	255,815	326,115	412,825
IHX inlet temp (°F)	1,078	1,102	1,075	1,115	1,147	1,177
IHX outlet temp (°F)	790	760	793	779	811	845
ΔT (°F)	288	342	282	336	336	332
Power (Mwt) *	6.1	7.3	5.9	7.6	9.7	12.1
Percent Rated Power	20.3	24.2	19.6	25.3	32.2	40.2

*Independent parameter of the test series

- 3) It was then observed that water flow was 47,000 lb/hr whereas it had been 44,000 lb/hr during the Series 1 flip incidents. Water flow was adjusted accordingly.
- 4) Reduction of sodium ΔT was again attempted and a flip was induced at a ΔT of about 80°F.
- 5) Increase of sodium ΔT was started. When ΔT reached 150°F (the planned second point of Series 4) the unit was still performing in the low heat-transfer post-flip condition. Further sodium ΔT increase to 170°F resulted in a reverse flip.
- 6) Sodium ΔT was reduced to 150°F and the planned steady-state point was established.
- 7) Sodium ΔT was raised to establish the planned 200 and 275°F points in sequence.

Five test points were obtained during the above events, meeting the steady-state data criteria. Terminal performance data are tabulated in Table 5-25. In terms of the independent variable, sodium ΔT , and their place in the sequence of events the test points are as follows:

	<u>Test Point</u>	<u>Sodium ΔT (°F)</u>
Before performance decrease	15A	91
	15	85
After performance increase	16	147
	17	199
	18	274

The plot of the reverse flip parameters vs time is shown in Figure 5-39, Section V.D. Terminal performance data points shortly before the flip, during the low heat-transfer period and after the reverse flip, are listed on Table 5-26.

(5) Series 5

- 1) Phase A — Test Series 5 was executed according to plan in an uninterrupted sequence of tests. The test plan required that sodium outlet temperature be held constant while sodium inlet and steam outlet

TABLE 5-25
DIAGNOSTIC TESTS - TEST SERIES 4 -
RESULTS OF TESTS 15A-18

Operating Conditions	15A	15	16	17	18
Test Date	2-24-70	2-24-70	2-25-70	2-25-70	2-25-70
Test Time	0825	1230	0655	1055	1609
Water Conditions:					
Flow (lb/hr)	47,280	47,160	44,680	44,430	44,800
Steam temp (out) (°F)	699	699	702	712	697
Water temp (in) (°F)	600	602	601	599	596
Pressure (psia)	2,240	2,235	2,240	2,210	2,200
Secondary Sodium Conditions for Steam Generator:					
Flow (lb/hr)*	1,014,800	1,079,600	596,900	452,900	324,600
SG inlet (IHX outlet) temp (°F)	764	762	802	840	900
SG outlet (IHX inlet) temp (°F)	673	677	655	641	626
ΔT (in-out) (°F)*	91	85	147	199	274
Primary Sodium Conditions for IHX:					
Flow (lb/hr)	856,200	967,400	523,200	507,400	389,600
IHX inlet temp (°F)	787	782	830	850	910
IHX outlet temp (°F)	679	687	662	672	681
ΔT (in-out) (°F)	108	95	168	178	229
Power (Mwt)	8.3	8.2	7.9	8.1	8.0
Percent Rated Power	27.6	27.5	26.2	26.9	26.5

*Independent parameters of the series

TABLE 5-26
DIAGNOSTIC TESTS - TEST SERIES 4 -
PERFORMANCE CHANGES

Operating Conditions	Performance Decrease		Performance Increase	
	Before	After	Before	After
Test Date	2-24-70	2-24-70	2-25-70	2-25-70
Test Time	1900	2345	0310	0515
Water Conditions:				
Flow (lb/hr)	44,680	44,550	44,680	44,680
Steam temp (out) (°F)	696	699	692	720
Water temp (in) (°F)	601	600	605	601
Pressure (psia)	2225	2240	2240	2240
Secondary Sodium Conditions for Steam Generator:				
Flow (lb/hr)*	1,044,300	1,232,500	734,100	527,730
SG inlet (IHX outlet) temp (°F)	760	818	838	825
SG outlet (IHX inlet) temp (°F)	677	747	722	652
ΔT (in-out) (°F)*	83	71	116	173
Primary Sodium Conditions for IHX:				
Flow (lb/hr)	943,100	952,200	852,900	537,860
IHX inlet temp (°F)	775	841	849	840
IHX outlet temp (°F)	683	749	749	670
ΔT (in-out) (°F)	92	92	100	170
Power (Mwt)	7.8	7.8	7.6	8.2
Percent Rated Power	25.9	26.1	25.3	27.3

*Independent parameters of the test series

temperatures were raised. This mode of control, which had never been tried previously, was achieved quite successfully. A nearly constant sodium outlet temperature of 650°F was maintained during all changes of test conditions as well as during the steady-state data intervals. The series yielded five steady-state points. In terms of the independent test variable (sodium inlet temperature), and the floating variable (steam outlet temperature), the points were as follows:

Test	Sodium Inlet Temperature (°F)	Steam Outlet Temperature (°F)
19	821	700
20	873	771
21	948	851
22	1024	953
23	1100	1029

The steam temperature at the final test point was 1029°F, only 21°F below the rated 1050°F temperature. Terminal parameters are listed in Table 5-27.

- 2) Phase B – As planned, Phase 5B was initiated with the steam generator in the final condition of Phase 5A. Reduction of sodium ΔT from 449 to 401°F resulted in a flip with a large magnitude drop in steam temperature. The test plan provided that if a flip occurred in Series 5 the series would terminate at the point of occurrence. The series was accordingly concluded. Table 5-28 lists pre- and post-flip data and the time plot of the flip parameters is shown in Figure 5-40, Section V.D.

c. Sodium Temperature Profiles

The main source of internal data on steam generator performance was the series of internal thermocouples shown in Figure 3-4 (later replaced by external thermocouples per Figure 3-5). These data may be plotted vs percent of tube area (or tube length) to show the sodium temperature profile in the heat-transfer zone. Plots of these data are shown in Figures 5-5 through 5-14 for the initial

TABLE 5-27
DIAGNOSTIC TESTS - TEST SERIES 5A -
RESULTS OF TESTS 19-23

Operating Conditions	19	20	21	22	23
Test Date	2-25-70	2-26-70	2-26-70	2-26-70	2-27-70
Test Time	2335	0420	1210	2345	0545
Water Conditions:					
Flow (lb/hr)	47,280	47,160	47,410	46,790	46,410
Steam temp (out) (°F)	700	771	851	953	1029
Water temp (in) (°F)	600	599	598	600	604
Pressure (psia)	2240	2240	2260	2240	2230
Secondary Sodium Conditions for Steam Generator:					
Flow (lb/hr)*	542,100	468,500	383,100	336,300	295,600
SG inlet (IHX outlet) temp (°F)*	821	873	948	1024	1100
SG outlet (IHX inlet) temp (°F)	650	648	643	648	651
ΔT (in-out) (°F)	171	225	305	376	449
Primary Sodium Conditions for IHX:					
Flow (lb/hr)	566,000	574,200	523,800	467,600	358,900
IHX inlet temp (°F)	831	877	957	1025	1105
IHX outlet temp (°F)	667	693	733	753	734
ΔT (in-out) (°F)	164	184	224	272	371
Power (Mwt)	8.3	9.4	10.4	11.3	11.8
Percent Rated Power	27.7	31.4	34.7	37.6	39.3

*Independent parameters of the test series

TABLE 5-28
DIAGNOSTIC TESTS
TEST SERIES 5B
PERFORMANCE CHANGES

Operating Conditions	Performance Decrease	
	Before	After
Test Date	2-27-70	2-27-70
Test Time	1000	1040
Water Conditions:		
Flow (lb/hr)	47,280	47,530
Steam temp (out) (°F)	1001	818
Water temp (in) (°F)	602	600
Pressure (psia)	2225	1715
Secondary Sodium Conditions for Steam Generator:		
Flow (lb/hr)*	331,040	312,010
SG inlet (IHX outlet) temp (°F)	1060	1073
SG outlet (IHX inlet) temp (°F)	659	696
ΔT (in-out) (°F)*	401	377
Primary Sodium Conditions for IHX:		
Flow (lb/hr)	375,870	370,830
IHX inlet temp (°F)	1070	1080
IHX outlet temp (°F)	716	762
ΔT (in-out) (°F)	354	318
Power (Mwt)	11.8	10.4
Percent Rated Power	39.3	34.8

*Independent parameters of the test series

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Figure 5-5. Steam Generator Temperature Profile –
Test Series 1, Test 1A

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Figure 5-6. Steam Generator Temperature Profile –
Test Series 1, Test 4A

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Figure 5-7. Steam Generator Temperature Profile –
Test Series 2, Test 5A

7693-59186

Figure 5-8. Steam Generator Temperature Profile –
Test Series 2, Test 7A

7693-59187

Figure 5-9. Steam Generator Temperature Profile –
Test Series 3, Test 8

7693-59188

Figure 5-10. Steam Generator Temperature Profile –
Test Series 3, Test 12

7693-59189

Figure 5-11. Steam Generator Temperature Profile –
Test Series 4, Test 15A

7693-59190

Figure 5-12. Steam Generator Temperature Profile -
Test Series 4, Test 18

7693-59191

Figure 5-13. Steam Generator Temperature Profile –
Test Series 5A, Test 19

7693-59192

Figure 5-14. Steam Generator Temperature Profile –
Test Series 5A, Test 23

and final test points of each series. Pre- and post-test profiles for the sudden performance changes during Series 1, 4, and 5 are given in Section V.D. Each figure includes a water/steam-side temperature profile estimated by the one-dimensional analysis method discussed in the following section.

The steady-state sodium temperature plots shown are as follows:

<u>Series</u>	<u>Test</u>	<u>Figure No.</u>
1	1A	5-5
	4A	5-6
2	5A	5-7
	7A	5-8
3	8	5-9
	12	5-10
4	15A	5-11
	18	5-12
5A	19	5-13
	23	5-14

3. Heat-Transfer Analysis Method

Using the sodium temperature profile faired from data, each test point was analyzed for quantitative heat-transfer performance.

All calculations of heat-transfer parameters were one-dimensional and were based upon the following assumptions:

- 1) Water/steam flowrate was uniform among all tubes (this was confirmed by the results of static flow stability analysis which is discussed in a later section).
- 2) At any cross section normal to the steam tubes, steam enthalpy and heat flux were uniform among all tubes.

It is noted that, when delivered from the manufacturer, the steam generator had three defective tubes which were plugged at the factory. Following the June 22, 1969, tube leak incident, two additional tubes were plugged in July 1969 (see Sections VI.A.1 and VI.A.3). This means that the number of active water/steam tubes was 297 for the plant performance tests and 295 for the diagnostic tests (Series 1 to 5B).

The above assumptions furnished the simplest basis for test-to-test comparison of results. Based upon the one-dimensional assumptions, the following reduced data were obtained using the relationships of Table 5-29:

- 1) The local heat flux for each increment of the heat-transfer area bounded by a pair of adjacent thermocouples.
- 2) The local steam enthalpy at each thermocouple location. All other steam properties (temperature, etc.) may thus be defined using the system pressure as measured at the steam outlet. This is a reasonable assumption in view of the low pressure drop in the steam generator.
- 3) The location of the saturated liquid and saturated vapor points. The areas encompassing the preheat, boiling and superheat heat-transfer zones may thus be defined. Local steam quality may be obtained in the boiling zone (equilibrium quality).
- 4) The following parameters for the preheat, boiling and superheat zones:
 - Sodium-to-water ΔT at each end of the zone.
 - Property-adjusted mean temperature difference (PAMTD)
 - Average heat flux
 - Overall heat transfer coefficient in the zone.
- 5) The following properties may be obtained for the entire heat-transfer area of the steam generator:
 - PAMTD
 - Average heat flux
 - Overall heat transfer coefficient.

The principal equations employed in the one-dimensional analysis are listed in Table 5-29. The PAMTD defined in the table was used in place of the more commonly used log mean temperature difference (LMTD). The LMTD is derived with the assumption that both fluids in a heat exchanger have constant specific heat. LMTD has little meaning when applied to the combined boiling and superheat zones of a once-through steam generator.

TABLE 5-29

HEAT-TRANSFER RELATIONSHIPS USED IN ONE-
 DIMENSIONAL HEAT-TRANSFER ANALYSIS
 (Sheet 1 of 2)

Definitions

A	Heat-transfer area (ft ²)
c _p	Specific heat (Btu/lb-°F)
H	Water/steam enthalpy (Btu/lb)
N	Total number of discrete increments
$\frac{q}{A}$	Heat flux (Btu/ft ² -hr)
T	Temperature (°F)
U	Overall heat transfer coefficient (Btu/ft ² -hr-°F)
W	Weight flowrate (lb/hr)
Σ	Sum of a series
Δ	Finite difference over a small interval

Subscripts

C	Cold end of a zone
H	Hot end of a zone
H ₂ O	Water/steam
I	Inlet
i	One length segment (one of a series)
f	Saturated liquid state
Na	Sodium
O	Outlet
g	Saturated vapor state
Z	Heat-transfer zone

TABLE 5-29

HEAT TRANSFER RELATIONSHIPS USED IN ONE-
DIMENSIONAL HEAT-TRANSFER ANALYSIS
(Sheet 2 of 2)

Local heat flux for area

interval in which sodium
temperature increases from
(T_{Na})_{ci} to (T_{Na})_{hi}

$$\left(\frac{q}{A}\right)_i = \frac{c_{pNa} [(T_{Na})_{Hi} - (T_{Na})_{Ci}]}{A_i} \cdot W_{Na}$$

Location of saturated vapor
point on sodium profile

$$(T_{Na})_g = (T_{Na})_O + \frac{H_g - H_I}{H_O - H_I} [(T_{Na})_I - (T_{Na})_O]$$

Location of saturated liquid
point on sodium profile

$$(T_{Na})_f = (T_{Na})_O + \frac{H_f - H_I}{H_O - H_I} [(T_{Na})_I - (T_{Na})_O]$$

Average heat flux for a zone
(applies for preheat, boiling
and superheat zones and for
entire heat transfer area of
the steam generator)

$$\left(\frac{q}{A}\right)_Z = c_{pNa} \frac{[(T_{Na})_H - (T_{Na})_C]}{A_Z} \cdot W_{Na}$$

Property-adjusted mean tem-
perature difference (PAMTD)
(applies for a zone or entire
heat transfer area) with given
N incremental points at which
 T_{Na} , T_{H_2O} , H_{H_2O} have been
established

$$PAMTD_Z = \frac{H_H - H_C}{\sum_{i=1}^N \left[\frac{\Delta H_i}{(T_{Na})_i - (T_{H_2O})_i} \right]}$$

Average heat transfer coeffi-
cient (applied for a zone or
entire heat transfer area)

$$U_Z = \frac{\left(\frac{q}{A}\right)_Z}{PAMTD_Z}$$

4. One-Dimensional Analysis Results

The one-dimensional analysis results were used to obtain plots of heat flux and overall heat transfer coefficient vs percent heat-transfer area. Such plots are presented for the plant performance tests in Figure 5-15 for heat flux, and for the initial and final steady-state points of each series in Figures 5-16 through 5-20 for heat flux, and Figures 5-21 through 5-30 for the overall heat transfer coefficient between sodium and water/steam. Additional heat flux curves are shown in Appendix D for all other runs not reported in this section.

Tables 5-30 through 5-35 list the preheat, boiling and superheat areas obtained by the one-dimensional analysis and the average heat-transfer properties thus obtained for each zone, for all the test runs.

5. Heat-Transfer Characteristics

a. General Characteristics of Heat Flux Curve

As shown in Figures 5-15 through 5-20, the heat-flux curves show non-uniform distribution along the flow length as contrasted with the uniform heat flux associated with electrical heating of the water tube — the method frequently used in a laboratory test for boiling studies. In the preheating zone, the heat flux begins with a rather large value and decreases rapidly with the flow length (indicative of thermal entrance effect) to a single-phase, liquid-flow, forced-convection value, followed by a gradual increase near the end of preheating (subcooled boiling). In the bulk boiling zone, the heat flux rises in low-quality nucleate boiling to the critical value (peak point), decreases in intermediate-quality transition boiling to the Leidenfrost point (minimum heat flux), and increases slowly in high-quality dispersed film boiling to a pure vapor value. The heat flux in the superheating zone generally decreases with flow length as the sodium-to-steam temperature difference decreases. Heat-transfer characteristics in the steam generator are discussed in detail in the following section.

b. Boiling Phenomenon

The phenomenon under consideration is essentially forced-convection vaporization with vapor superheat, and the flow is single-component two-phase flow with steam quality increasing with flow length. The heat-transfer mechanism in this type of flow depends mainly on the flow pattern of each local two-phase flow regime.

Figure 5-15. Heat-Flux Curves – Plant Performance Tests

Figure 5-16. Heat-Flux Curves - Diagnostic Test Series 1, Tests 1A and 4A

Figure 5-17. Heat-Flux Curves — Diagnostic Test Series 2, Tests 5A and 7A

Figure 5-18. Heat-Flux Curves — Diagnostic Test Series 3, Tests 8 and 12

Figure 5-19. Heat-Flux Curves – Diagnostic Test Series 4, Tests 15A and 18

Figure 5-20. Heat-Flux Curves — Diagnostic Test Series 5A, Tests 19 and 23

Figure 5-21. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 1, Test 1A

Figure 5-22. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 1, Test 4A

Figure 5-23. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 2, Test 5A

Figure 5-24. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 2, Test 7A

Figure 5-25. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 3, Test 8

Figure 5-26. Overall Heat Transfer Coefficient Between Sodium and Water/Steam — Test Series 3, Test 12

Figure 5-27. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 4, Test 15A

Figure 5-28. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 4, Test 18

Figure 5-29. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 5A, Test 19

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Figure 5-30. Overall Heat Transfer Coefficient Between Sodium and Water/Steam - Test Series 5A, Test 23

TABLE 5-30

STEAM GENERATOR PERFORMANCE SUMMARY -
PLANT PERFORMANCE TESTS

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
Target 20% rated power	5-14-69 1330	Preheat	47	58	1.70	128	13,260	51	262
		Boiling	242	47	9.17	185	49,440	118	418
		Superheat	16	242	8.09	965	8,380	119	71
		Total	16	58	18.96	1278	14,820	106	140
Target 40% rated power	5-15-69 1530	Preheat	141	105	3.58	36	100,300	145	690
		Boiling	333	141	19.06	716	26,630	223	119
		Superheat	118	333	16.41	522	31,430	244	129
		Total	118	105	39.05	1274	30,660	221	139

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TABLE 5-31
STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 1 (Sheet 1 of 2)

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
1A	12-31-69 1530	Preheat	14	23	4.0	544	7,321	16.6	441.0
		Boiling	237	14	19.1	551	34,588	81.5	424.2
		Superheat	236	237	4.6	171	26,827	240.3	111.6
		Total	236	23	27.6	1266	21,815	56.0	389.2
1	1-19-70 2315	Preheat	18	32	4.2	403	10,430	22.6	461.1
		Boiling	210	18	20.5	654	31,381	79.2	396.4
		Superheat	200	210	5.2	201	25,841	209.0	123.6
		Total	200	32	29.9	1259	23,791	63.7	373.5
2	1-1-70 1200	Preheat	20	44	3.7	317	11,819	29.2	405.3
		Boiling	156	20	19.1	722	26,520	66.9	396.3
		Superheat	135	156	4.5	228	19,707	148.7	132.5
		Total	135	44	27.4	1266	21,619	61.6	351.1
2R	1-20-70 430	Preheat	22	43	3.7	302	12,318	30.0	410.3
		Boiling	161	22	20.5	743	27,546	70.1	393.0
		Superheat	142	161	4.6	214	21,499	154.3	139.3
		Total	142	43	28.8	1259	22,863	64.6	354.0
3	1-1-70 2130	Preheat	23	54	3.8	203	18,629	35.1	530.0
		Boiling	127	23	19.0	760	24,986	61.3	407.6
		Superheat	100	127	4.3	304	14,048	116.1	121.0
		Total	100	54	27.0	1266	21,344	59.6	358.4
4A	1-9-70 1600	Preheat	38	77	4.0	164	24,107	53.3	452.5
		Boiling	101	38	18.8	783	23,969	65.0	368.8
		Superheat	70	104	4.2	316	13,399	88.9	150.6
		Total	70	77	27.0	1264	21,345	65.7	325.1

TABLE 5-31

STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 1 (Sheet 2 of 2)

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
Before performance decrease	1-2-70	Preheat	32	72	3.80	101	37,500	48.1	780
		Boiling	93	32	18.77	734	25,600	57.3	446
	0330	Superheat	59	93	4.03	431	9,400	77.8	121
		Total	59	72	26.60	1266	21,000	58.0	362
	1-9-70	Preheat	42	82	4.12	102	40,400	58.4	692
		Boiling	104	42	18.94	814	23,300	67.7	343
	1745	Superheat	65	104	4.51	356	12,700	86.6	146
		Total	65	82	27.56	1272	21,700	68.5	316
After performance decrease	1-2-70	Preheat	83	122	3.80	32	1,201,000	100	1205
		Boiling	147	83	18.90	1121	16,900	112	151
	0445	Superheat	142	147	0.97	114	8,500	-	-
		Total	142	122	23.68	1266	18,700	114	163
	1-9-70	Preheat	90	132	3.95	32	1,248,000	108	1158
		Boiling	149	90	18.19	728	24,980	117	213
	1930	Superheat	104	149	4.87	507	9,620	130	74
		Total	104	132	27.01	1266	21,300	118	181

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TABLE 5-32

STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 2

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
5A	1-10-70 1430	Preheat	45	78	7.2	266	27,061	58.3	464.0
		Boiling	162	45	31.9	861	36,994	91.5	404.2
		Superheat	142	162	6.5	139	46,769	154.4	303.0
		Total	142	78	45.6	1266	35,984	88.7	405.6
5AR	1-21-70 1030	Preheat	41	68	6.2	247	25,257	52.0	485.6
		Boiling	155	41	32.7	836	39,152	85.7	456.9
		Superheat	131	155	7.3	184	39,503	145.5	271.5
		Total	131	68	46.2	1266	36,493	83.8	435.6
5B	1-21-70 1745	Preheat	47	70	7.6	247	30,624	55.9	547.4
		Boiling	184	47	38.8	882	43,960	99.9	440.1
		Superheat	166	184	8.3	140	59,676	177.5	336.2
		Total	166	70	54.7	1269	43,088	95.8	449.6
6A	1-28-70 1415	Preheat	53	72	7.7	229	33,566	60.6	554.3
		Boiling	194	53	41.5	891	46,621	108.5	429.9
		Superheat	172	194	11.1	153	72,341	186.8	387.3
		Total	172	72	60.3	1273	47,357	105.9	447.2
6B	1-28-79 1950	Preheat	59	77	10.0	261	38,334	65.7	583.4
		Boiling	221	59	47.7	884	53,956	122.5	440.4
		Superheat	209	221	11.4	127	89,897	218.3	411.7
		Total	209	77	69.1	1272	54,347	116.4	466.9
7A	1-28-70 2330	Preheat	60	75	10.8	274	39,514	65.6	602.7
		Boiling	239	60	51.6	878	58,753	129.8	452.7
		Superheat	232	239	12.0	121	99,380	238.8	416.2
		Total	232	75	74.4	1273	58,476	121.5	481.5

TABLE 5-33

STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 3

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
8	2-6-70 1405	Preheat	46	57	1.8	103	17,533	49.9	351.7
		Boiling	242	46	10.1	237	42,628	118.1	361.0
		Superheat	15	242	8.9	943	9,424	117.4	80.2
		Total	15	57	20.8	1283	16,215	105.4	153.9
9	2-8-70 0300	Preheat	70	78	2.0	38	51,115	72.6	704.2
		Boiling	268	70	12.2	448	27,192	147.2	184.8
		Superheat	53	268	10.6	794	13,424	168.8	79.5
		Total	53	78	24.8	1280	19,373	143.4	135.1
8R	2-17-70 2314	Preheat	60	73	1.9	95	19,976	64.4	310.3
		Boiling	248	60	9.6	247	38,928	132.0	294.9
		Superheat	18	248	8.6	926	9,278	123.6	75.1
		Total	18	73	20.1	1268	15,862	117.0	135.6
9R	2-18-70 0630	Preheat	111	125	2.5	42	58,991	116.3	507.4
		Boiling	297	111	12.4	588	21,144	189.1	111.8
		Superheat	71	297	11.0	643	17,111	195.0	87.7
		Total	71	125	25.9	1273	20,356	180.6	112.7
10B	2-18-70 1515	Preheat	141	154	3.1	50	61,655	145.6	423.4
		Boiling	328	241	15.8	638	24,741	220.8	112.1
		Superheat	93	328	14.2	586	24,163	224.4	107.7
		Total	93	154	32.1	1274	25,915	212.1	122.2
12	2-19-70 0415	Preheat	165	177	3.8	46	82,474	169.3	487.2
		Boiling	354	165	19.7	664	29,666	247.2	120.0
		Superheat	123	354	17.8	557	31,884	257.4	123.9
		Total	123	177	41.3	1267	32,543	241.2	134.9

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TABLE 5-34

STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 4
(Sheet 1 of 2)

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
15A	2-24-70 0825	Preheat	34	73	4.1	77	53,150	50	1071
		Boiling	98	34	19.8	900	22,000	61	363
		Superheat	65	98	4.4	309	14,260	84	171
		Total	65	73	28.3	1286	22,010	61	360
15	2-24-70 1230	Preheat	37	75	3.9	58	67,870	53	1293
		Boiling	97	37	19.8	996	19,880	62	318
		Superheat	63	97	4.4	231	19,000	82	232
		Total	63	75	28.1	1285	21,880	63	347
16	2-25-70 0655	Preheat	24	54	3.8	274	13,900	35	394
		Boiling	127	24	18.7	739	25,300	62	412
		Superheat	100	127	4.4	261	16,640	116	144
		Total	100	54	26.9	1274	21,080	60	353
17	2-25-70 1055	Preheat	18	42	3.8	280	13,480	27	506
		Boiling	154	18	18.8	713	26,400	63	416
		Superheat	128	154	5.0	280	17,670	145	122
		Total	128	42	27.6	1273	21,640	58	371
18	2-25-70 1609	Preheat	16	30	3.9	553	7,090	20	347
		Boiling	209	16	19.1	572	33,410	76	440
		Superheat	203	209	4.1	146	28,080	209	135
		Total	203	30	27.1	1271	21,350	59	365

TABLE 5-34

STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 4
(Sheet 2 of 2)

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
Before Perfor- mance Decrease	2-24-70 1900	Preheat	38	76	3.7	64	58,250	53	1,095
		Boiling	97	38	18.8	947	19,890	63	318
		Superheat	64	97	4.0	269	14,830	82	180
		Total	64	76	26.5	1,280	20,750	63	328
After Perfor- mance Decrease	2-24-70 2345	Preheat	105	147	3.9	39	100,300	124	813
		Boiling	155	105	18.6	938	19,870	129	155
		Superheat	119	155	4.1	309	13,430	140	96
		Total	119	147	26.6	1,286	20,740	130	160
Before Perfor- mance Increase	2-25-70 0310	Preheat	86	117	3.6	32	111,520	99	1,124
		Boiling	170	86	18.7	1,006	18,590	123	151
		Superheat	146	170	3.7	236	15,640	160	98
		Total	146	117	26.0	1,274	20,370	123	166
After Perfor- mance Increase	2-25-70 0515	Preheat	23	51	3.8	273	13,940	34	411
		Boiling	140	23	18.7	633	29,540	65	453
		Superheat	105	140	5.4	365	14,820	126	117
		Total	105	51	27.9	1,271	21,960	63	347

TABLE 5-35

STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 5A
(Sheet 1 of 2)

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
19	2-25-70 2335	Preheat	23	50	4.1	357	11,500	33	350
		Boiling	142	23	19.8	669	29,610	65	454
		Superheat	121	142	4.5	248	17,990	134	134
		Total	121	50	28.4	1274	22,270	61	363
20	2-26-70 0420	Preheat	25	49	4.2	317	13,110	34	387
		Boiling	163	25	19.8	615	32,100	74	436
		Superheat	102	163	8.3	336	24,640	141	175
		Total	102	49	32.3	1268	25,370	72	355
21	2-26-70 1210	Preheat	26	45	4.4	292	15,020	33	454
		Boiling	194	26	19.6	533	36,820	85	434
		Superheat	97	194	11.6	444	26,080	160	163
		Total	97	45	35.6	1269	28,050	82	344
22	2-26-70 2345	Preheat	35	48	4.1	204	19,930	40	504
		Boiling	226	35	19.6	490	39,930	103	387
		Superheat	71	226	14.8	580	25,540	165	155
		Total	71	48	38.5	1274	30,180	100	301
23	2-27-70 0545	Preheat	41	47	3.7	197	18,680	42	444
		Boiling	258	41	19.6	503	38,870	118	331
		Superheat	71	258	17.0	573	29,600	181	164
		Total	71	47	40.3	1273	31,570	116	273

TABLE 5-35
STEAM GENERATOR PERFORMANCE SUMMARY -
TEST SERIES 5A
(Sheet 2 of 2)

Operating Condition	Date and Time	Water Phase	Sodium-to-Water ΔT		Heat Flow (Btu/hr $\times 10^{-6}$)	Area (OD) (ft ²)	Heat Flux (OD) (Btu/hr-ft ²)	PAMTD (°F)	U (PAMTD) (OD) (Btu/hr-ft ² -°F)
			Hot (°F)	Cold (°F)					
Before performance decrease	2-27-70 1000	Preheat	46	57	3.9	190	20,520	50	412
		Boiling	244	46	19.9	487	40,920	119	344
		Superheat	59	244	16.4	588	27,940	164	170
		Total	59	57	40.2	1,265	31,820	116	274
After performance decrease	2-27-70 1040	Preheat	93	96	1.1	10	55,140	24	585
		Boiling	353	93	24.7	899	27,470	195	141
		Superheat	255	353	9.9	348	28,300	317	89
		Total	255	96	35.7	1,266	28,120	211	133

Figure 5-31 shows a flow pattern and associated heat-transfer characteristics in a vertical, forced-convection boiling tube. Slightly subcooled water enters the bottom of a vertical tube which is heated externally by the hot liquid sodium. As the subcooled water flows upward, the bulk fluid temperature and the inner tube wall temperature increase as a consequence of the heat transferred. When the inner tube wall temperature reaches a certain degree of wall superheat (inner wall temperature above the saturation point), small vapor bubbles begin to form at the wall even if the water in the core is at a temperature below the saturation temperature which corresponds to local fluid pressure. This is so-called bubble nucleation in subcooled boiling and corresponds to point A in Figure 5-31. Up to this point, the heat transfer is due to single-phase, liquid-flow convection and the heat transfer coefficient is more or less constant. As the water passes point A, more vapor bubbles are generated at the wall and entrained in the continuous liquid flow with some of the bubbles condensing in the subcooled liquid. This flow pattern is called bubbly flow; the heat-transfer mechanism is essentially that of nucleate boiling and, therefore, the heat-transfer coefficient rises above the liquid-phase value. Meanwhile the bulk fluid temperature continues to increase and finally reaches the saturation point. Thereafter bulk boiling takes place with net vapor generation.

As the quality or flow length increases, the individual bubbles agglomerate and grow to form slugs or plugs of vapor, which periodically pass a given point on the wall. This is the slug-flow regime. At higher quality, the vapor flows in a high-velocity core in the center of the tube, and a thin film of liquid covers the tube wall. Liquid droplets are dispersed in the vapor core. This flow pattern of vapor core and liquid annulus is called annular flow. In this flow pattern, both vapor and liquid velocities increase with flow quality because of constant flow area and larger specific volume of the vapor state over the liquid state. With increased two-phase flow velocity, the wall superheat required for initiation of nucleate boiling increases. In other words, chances of bubble formation on the wall decrease with increasing vapor-induced liquid velocity. This phenomenon is sometimes called "suppression" of nucleate boiling. This changes the character of the heat-transfer mechanism; namely, heat is transferred from the hot wall to the liquid in the annulus by forced convection and evaporation takes

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Figure 5-31. Flow Pattern and Heat-Transfer Characteristics in a Vertical Boiling Tube

place at the liquid-vapor interface instead of at the tube wall. The heat transfer coefficient increases with quality as the two-phase flow velocity increases.

As the quality increases further, liquid volume decreases through vaporization, and thus the liquid film in the annulus breaks off the wall, resulting in dry patches, until the entire flow becomes a mist or dispersed flow in which the vapor state is continuous and the liquid is dispersed in the vapor as droplets. In this transition from annular to mist flow, the heat transfer coefficient decreases because of the poor heat-transfer characteristics associated with vapor flow as compared to liquid flow. In mist flow, heat is transferred from the vapor to the liquid droplets. Once mist flow is established, the heat-transfer coefficient increases with increasing quality because continuous vapor generation accelerates the vapor flow velocity. With all the liquid evaporated, the flow becomes entirely vapor-saturated and further heat input will generate superheated steam.

In the general literature, the entire flow with phase change is generally divided into three regions for data-correlation purposes: liquid-wetted, nucleate boiling region I (low quality); vapor-covered, dry wall, dispersed film boiling region III (high quality); and transition boiling region II (intermediate quality).

6. Boiling Zone Overall Heat Transfer Coefficient

The average overall heat transfer coefficient between sodium and water/steam, computed by the one-dimensional approach, appears to be the best general indicator of good vs poor performance. The trend of this parameter in the boiling zone for each series is summarized below in terms of the independent variable of the series, and is plotted in Figures 5-32 through 5-36.

In Test Series 1, the heat-transfer coefficient remained at approximately $400 \text{ Btu/ft}^2\text{-hr-}^\circ\text{F}$ over a sodium inlet-outlet ΔT range of 320 to 150°F . It declined to $376 \text{ Btu/ft}^2\text{-hr-}^\circ\text{F}$ at a ΔT of 95°F and dropped to below 200 after flips when ΔT was reduced below 90°F (Figure 5-32).

In Test Series 2, the heat transfer coefficient increased from $370 \text{ Btu/ft}^2\text{-hr-}^\circ\text{F}$ at 45,000 lb/hr water flow to over 400 at 75,000 lb/hr; the heat transfer coefficient then continued to increase to approximately 450 at 124,000 lb/hr water flow (Figure 5-33).

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Figure 5-32. Boiling Zone Heat Transfer Coefficient vs Sodium Inlet-to-Outlet ΔT - Test Series 1

Figure 5-33. Boiling Zone Heat Transfer Coefficient vs Water Flowrate — Test Series 2

Figure 5-34. Boiling Zone Heat Transfer Coefficient vs Power –
Test Series 3

Figure 5-35. Boiling Zone Heat Transfer Coefficient vs Sodium Inlet-to-Outlet ΔT – Test Series 4

Figure 5-36. Boiling Zone Heat Transfer Coefficient vs Sodium Inlet Temperature - Test Series 5A

In Test Series 3 (including the 1969 plant performance results) the heat-transfer coefficient declined from 420 Btu/ft²-hr-°F at a 5.6 Mwt power level to 185 at 7.3 Mwt, and decreased further to approximately 120 Btu/ft²-hr-°F through a further power increase up to 12.1 Mwt (Figure 5-34).

In the initial attempt to establish starting conditions for Test Series 4, with water flow at 47,000 lb/hr, a reduction of sodium ΔT from 91 to 85°F produced a heat transfer coefficient drop from 363 to 318 Btu/ft²-hr-°F, and no flip was induced. With water flow reduced to 44,000 lb/hr, a flip was induced at a sodium ΔT of 83°F, with reduction of the heat transfer coefficient to 155 Btu/ft²-hr-°F. After the sodium ΔT was raised to 170°F, a reverse flip occurred with an increase of the heat transfer coefficient from 151 to 453 Btu/ft²-hr-°F.

A further rise in sodium ΔT yielded a change in the heat transfer coefficient as shown in Figure 5-35. Results for Test Series 1 are also shown in this figure. One purpose of Test Series 4 was to determine whether any hysteresis would be found in performance vs sodium ΔT when decreasing this parameter in Series 1 compared to increasing it in Series 4. Hysteresis was evident in the fact that the flip was induced at approximately 85°F ΔT and the reverse flip was induced at approximately 170°F ΔT . However, no hysteresis is shown after the unit returned to the good performance level via the reverse flip.

In Test Series 5A, an increase of sodium inlet temperature to 1100°F (rise of sodium ΔT to 450°F) resulted in an increase of steam temperature of 1029°F. The heat-transfer coefficient dropped from 454 to 331 Btu/ft²-hr-°F during this rise (Figure 5-36). In Series 5B, a decrease in sodium ΔT from the final point of Test 5A to 401°F had little effect on the heat transfer coefficient. At this point a flip occurred with a decrease in the heat transfer coefficient to 141 Btu/ft²-hr-°F.

7. Comparison of Analytical and Test Results

A comparison was made of experimental and theoretical overall heat transfer coefficients for each of the characteristic heat-transfer regions discussed in Section V.B.5. The test points of Series 2, which did not exhibit any performance degradation either gradually or suddenly, were used in the comparison, while in some cases good performance data points of Series 1 were also used.

a. Average Overall Heat Transfer Coefficient in Preheating Region

In the preheating region, the water-side heat-transfer coefficient begins with a rather large value, decreases rapidly with the flow length, and remains more or less constant, followed by a gradual increase near the end of the preheating region.

A rapid decrease of the water-side heat transfer coefficient from a high value in the beginning of the preheating region is typical of thermal entrance effect. The water entering the tubes flows about 13 in. (lower tube sheet plus a number of thermal-protective shields) without much heat transferred from sodium. In this initial flow of 13-in. length, water flow is more or less fully established hydrodynamically, but the thermal boundary layer hardly develops due to low heat transfer in this region. The thermal boundary layer begins to develop actively when positive thermal contact between sodium and water develops at a 13-in. point from the feedwater inlet. The effect of this type of prolonged development of the thermal boundary layer with respect to the hydrodynamic boundary layer is shown in Reference 4. The result is a rapid decrease of the heat transfer coefficient from an indefinitely high value.

It is noted, however, that the data reduction method (in which the heat transfer in the area of the lower tube sheet and thermal shields was neglected) tended to increase the magnitude of the negative slope of the heat-flux curve at the entrance region of the water/steam tubes.

As the thermal boundary layer (as well as the hydraulic boundary layer) develops fully, the heat transfer coefficient ceases to decrease and remains approximately constant thereafter until inception of boiling occurs.

The gradual increase in the heat transfer coefficient near the end of the preheating region is due to subcooled boiling (boiling in a subcooled liquid). Incipience of boiling occurs when the water-side wall temperature reaches a certain degree of superheat above the saturation temperature which corresponds to the local pressure. The wall superheat required for bubble nucleation is a function of heated surface condition, surface-fluid combination, heat flux, pressure, etc. This is a local (surface) phenomenon independent of the bulk fluid temperature, and usually precedes bulk net boiling. Some vapor bubbles

in subcooled boiling may detach from the heated wall and collapse in subcooled bulk stream, thus increasing the heat transfer coefficient over the single-phase value.

The overall heat transfer coefficient in the preheating region was computed by taking an inverse of the sum of thermal resistances of the sodium side, the bimetallic tube wall, and the water side including the fouling factor. The sodium-side heat transfer coefficient was computed from the equation.

$$Nu = 0.458 (Pe)^{1/2} \text{ (for sodium)}$$

where

Nu = Nusselt number based on tube outside diameter

Pe = Peclet number based on tube outside diameter and minimum cross-flow area.

This equation was obtained by correlating test data of the IHX shell side whose configuration is very close to that of the steam generator shell side.

The water-side fouling factor of $0.0001 \text{ hr-ft}^2\text{-}^\circ\text{F/Btu}$ (based on the inner tube area) was used while sodium side fouling was neglected. For the water-side heat transfer coefficient, the Dittus-Boelter type equation was used:

$$Nu = 0.023 (Re)^{0.8} (Pr)^{0.4} \text{ (for water)}$$

where

Nu = Nusselt number based on tube inside diameter

Re = Reynolds number based on tube inside diameter

Pr = Prandtl number.

The calculated results for the average overall heat transfer coefficient in the preheating region, based on the outside tube area, are compared to the experimental values for Series 1 and 2 as follows:

Test Number	U (experimental)	U (theoretical)
	(Btu/hr-ft ² - °F)	

Series 1

1A	441	343
1	461	359
2	405	357
2R	410	364
3	530	359
4A	452	358

Series 2

5A	464	429
5AR	486	435
5B	547	455
6A	554	471
6B	583	483
7A	603	493

The theoretical predictions are less than the experimental indications for all the cases considered. This is mainly a result of the effects of developing thermal boundary layer and subcooled boiling being neglected in the prediction. A refined analysis should include these effects.

b. Average Overall Heat Transfer Coefficient in Pre-Critical Heat Flux Boiling Region

This is a region from inception of bulk boiling to critical heat flux (CHF) point. The water-side heat transfer is characterized by nucleate boiling with wetted-wall condition. The heat transfer coefficient rises from the subcooled boiling value to a peak value at the CHF point.

The average overall heat transfer coefficient in this region was computed just as in the preheating region except that the water-side heat transfer coefficient was evaluated from Chen's equation.⁽⁵⁾ The results are compared to experimental values for Series 1 and 2 as follows:

Test Number	U (experimental)	U (theoretical)
	(Btu/hr-ft ² - °F)	

Series 1

1A	583	642
1R	542	691
2	776	675
2R	805	675
3	648	700
4A	594	688

Series 2

5A	618	734
5B	693	720
6A	732	738
6B	793	695
7A	771	738
5AR	807	708

A spread of data around theoretical values is apparent.

c. Average Overall Heat Transfer Coefficient in Post-CHF Boiling Region

This is a region from the CHF point to the 100% quality point. The water/steam-side heat transfer is characterized by dispersed (partial) film boiling. The heat transfer coefficient decreases from the CHF value to a minimum at the Leidenfrost point, then increases slightly to a pure vapor value at the 100% quality location.

The average overall heat transfer coefficient in the region was computed with the use of Swenson's equation⁽⁶⁾ for the water/steam side of the following form:

$$Nu = 0.024 (Re_v)^{0.8} (Pr)^{0.4} \text{ (film boiling)}$$

where

Nu = Nusselt number based on tube inside diameter

Re_v = Reynolds number for vapor fraction, based on tube inside diameter.

The results are compared to experimental values for Series 2 only as follows:

Test Number	U (experimental)	U (theoretical)
	(Btu/hr-ft ² - °F)	
<hr/>		
<u>Series 2</u>		
5A	242	255
5AR	296	260
5B	329	293
6A	294	299
6B	298	319
7A	329	312

A fair agreement is evident between theory and experiment.

d. Average Overall Heat Transfer Coefficient in Superheated Region

This is the region with dry steam. The steam-side heat transfer coefficient generally decreases with flow length, mainly due to the effect of increasing inner wall temperature.

The average overall heat transfer coefficient in this region was computed with use of Heineman's equation for superheated steam.⁽⁷⁾

$$Nu = 0.0133 (Re)^{0.84} (Pr)^{1/3} \text{ (superheated steam),}$$

where the characteristic length is the diameter of the inner tube, and the physical properties are evaluated at the average temperature between the inner tube wall temperature and the steam temperature. The results are compared to experimental values for Series 2 only as follows:

Test Number	U (experimental)	U (theoretical)
	(Btu/hr-ft ² - ° F)	
<u>Series 2</u>		
5A	303	300
5AR	272	304
5B	336	329
6A	387	336
6B	412	358
7A	416	364

A fair agreement between theory and experiment is evident.

8. Correlations of Critical and Leidenfrost Point Heat Fluxes

It is always interesting to observe the maximum and minimum values of heat flux in the boiling zone. The magnitude of these values, as well as the locations where they occur, are very important as regards heat transfer/structural design. Attempts were made to correlate these values.

a. CHF Correlation

The CHF is the peak heat flux at the end of nucleate boiling. This value is in general a function of local flow quality, flowrate, system pressure, upstream heated length-to-diameter ratio, fluid properties, heat flux distribution, and others. Various correlating forms have been reported in the literature. In this analysis, the following form from Reference 8 was chosen:

$$\frac{(q/A)_{\text{CHF}}}{10^6} = C_1 (H_f + X_{\text{CHF}} H_{fg})^{C_2} \left(1 + \frac{G}{10^7}\right)^2 e^{-0.0012 \frac{l}{d}}$$

where

$\left(\frac{q}{A}\right)_{\text{CHF}}$ = CHF (Btu/hr-ft²) based on water-side heat-transfer area

H_f = saturated liquid enthalpy (Btu/lb)

X_{CHF} = CHF steam quality

H_{fg} = latent heat (Btu/lb)

G = water/steam flowrate per unit area (lb/hr-ft²)

l = heated length from tube entrance to CHF point

d = tube inner diameter.

The CHF steam quality varied from 0.25 to 0.55 and the average value was 0.4. The coefficients C_1 and C_2 were determined by least square fitting, using experimental data. The experimental data used in the correlations are those in which the CHF values were clearly shown (26 runs). The results show that $C_1 = 290.0$ and $C_2 = -1.092$. This correlation predicts 25 out of the 26 data

points used within a 35% deviation. A further refinement with different correlating equations is needed.

b. Leidenfrost Point Heat-Flux Correlation

The Leidenfrost point heat flux is the minimum heat flux at the end of transition boiling. This value is strongly affected by the local flow quality and surface temperature. In the analysis, an attempt was made to correlate the Leidenfrost heat flux in terms of local flow quality only. Various correlating forms were assumed and the correlating coefficients were determined by least square fitting. The experimental data used in the correlation were those in which the minimum heat-flux points were clearly shown (17 runs). Two resulting equations are listed below:

$$\left(\frac{q}{A}\right)_{LP} \cdot 10^{-4} = 1.97 (X_{LP} - 0.052)^{-1}$$

$$\left(\frac{q}{A}\right)_{LP} \cdot 10^{-4} = 8.16 e^{-1.33 X_{LP}}$$

where

$\left(\frac{q}{A}\right)_{LP}$ = Leidenfrost point heat flux, based on water-side heat-transfer area (Btu/hr-ft²)

X_{LP} = Leidenfrost point steam quality.

The Leidenfrost point quality varied from 0.50 to 0.94 and the average value was 0.7. Both of these equations correlate 14 out of 17 data points used within 40%.

C. PRESSURE DROP RESULTS

1. Water/Steam-Side Pressure Drop

The water/steam pressure drop was measured directly by a differential pressure transducer connected between the inlet and outlet lines. For correct zero setting of this transducer, it was necessary that the sensing lines be filled with water as they would be during operation. It is not certain that this procedure was followed consistently; consequently, the measurements may contain

unknown biases. During most of the test period, pressure drop was not recorded but was read from a dial panel indicator at each data point. This device tended to have a large amplitude oscillation, further compromising accuracy. From February 17, 1969, until the conclusion of testing, pressure drop was recorded on a chart. Recording the measurement on a chart minimized the oscillation problem and provided a continuous ΔP record which could be correlated with changes in other performance parameters.

To obtain an estimate of friction and momentum pressure loss in the steam generator it was necessary to correct the raw ΔP measurement for static head, using the relationship:

$$\text{Static } \Delta P = \sum \rho_i \Delta Z_i$$

where

ρ_i = local water/steam density

ΔZ_i = local increment of tube length.

The local value of ρ_i and other water/steam properties could be obtained by the heat balance method of Section V.B.3. In the wet-steam region, density was estimated directly as a function of local steam quality. This method does not account for velocity slip in the two-phase flow region. The static head contribution within the wet-steam region is a function of velocity slip as shown in Appendix A. Most of the test results obtained were at flows of 40% of rated (47,000 lb/hr) or less power. In this flow range, static head tends to be of the same order of magnitude as friction loss (momentum loss was very small). Estimates of friction and momentum loss in the low flow range showed considerable scatter and inconsistency, as shown in Figure 5-37. In view of the sources of error discussed above, this may not be surprising.

At higher flows, static head correction and recorder biases are less significant. In Test Series 2, in which the higher flows were investigated, pressure drop results were as follows:

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Figure 5-37. Water/Steam-Side Friction/Momentum Pressure Drop

Test	Water Flow (lb/hr)	Estimated Friction Loss (psid)
5A	75,329	8.50
5AR	77,190	7.50
5B	91,213	9.90
6A	99,900	12.70
6B	114,792	15.70
7A	124,100	17.20

These tests were performed with 50°F superheat; therefore, the pressure drop at the rated steam temperature (400°F superheat) would be higher due to the larger percentage of superheated steam in each tube. An attempt to fit an equation to these points yielded the following:

$$\Delta P = 0.00208 w^{1.6} T_s / P_s$$

where

w = water/steamflow per tube (lb/hr)

T_s = steam outlet temperature (°R)

P_s = steam pressure (psia)

A number of flip and reverse flip incidents occurred after recording of ΔP was initiated. The recordings revealed that there was a drop in ΔP from the normal pre-flip performance state to the post-flip state and a corresponding gain in ΔP during reverse flips. This is either partly or totally caused by the change in the length of the superheated region associated with performance shift.

2. Sodium-Side Pressure Drop

Sodium-side pressure drop in the steam generator was measured by obtaining the difference of the separately measured sodium inlet and outlet pressures. Specified accuracy of the measurements is 2-1/2% (see Table 3-3). However a difficult procedure of in-place calibration is required and as-calibrated accuracy is uncertain. The measured pressure drop represents the combined

effect of static head and friction loss. The static head may be estimated, including correction of sodium density based on the steam generator temperature profile. Pressure drops for the steady-state points of the diagnostic tests were corrected for static head, yielding the frictional pressure drop plotted in Figure 5-38. In the higher flow region, the plotted points yield a reasonable fit to the power law. At lower flows, there is a considerable scatter about the power law curve. Contributions to this scatter may include uncertainty of pressure gages, and inaccuracy in the water flow and temperature measurements used to calculate sodium flow by the heat balance method.

D. PERFORMANCE SHIFTS

Sudden performance shifts occurred on five occasions during the planned operation of the diagnostic tests. In addition, 13 sudden performance shifts were observed during operations to bring the steam generator up to power, or to set up initial conditions of a test series. Other incidents of this type were observed during early SCTI checkout operations in 1966.

In the plant performance tests and in the Diagnostic Test Series 3, a gradual drop in performance took place as power was increased from 20%. Smaller magnitude decrease in heat-transfer performance with operating conditions occurred in the pre-flip regions of Series 1 and Series 5 (see Figures 5-32 and 5-36).

All the performance changes appear to represent a common problem. For this reason the incidents outside the planned tests have been studied as well as those within the tests. For identification purposes, some of the incidents have been arbitrarily numbered performance change incident 1, 2, etc.

1. Description

A sudden drop in performance (or flip) is characterized by the following:

- 1) Steam outlet temperature drops.
- 2) Sodium outlet temperature rises, accompanied by a rise of the entire sodium temperature profile of the heat-transfer zone.
- 3) The steam generator water/steam pressure drop decreases. The change in pressure drop is appreciable, typically amounting to 30% or more of the initial pressure drop.

Figure 5-38. Sodium-Side Friction Pressure Drop

During the diagnostic tests, the above changes occurred in a period of about 5 min.

Figure 5-4 of Section V.B shows changes of terminal parameters with time during the flip observed on January 2, 1970. The changes are typical for most flips observed.

A sudden performance increase or reverse flip is characterized by an almost exact reversal of the above effects. Steam temperature increased, sodium temperature profile dropped, and water/steam pressure drop increased. Figure 5-39 is a typical time plot for the reverse flip during Series 4. Figure 5-40 is the time plot for the flip which terminated Series 5B. This was the largest magnitude steam temperature change observed.

The ΔP depicted in these figures is the as-measured value. The transducer is located at an elevation slightly above the steam generator outlet. The pressure in the inlet and outlet lines is transmitted directly to the transducer through a water column. Since the elevation between the main steam/water line and the transducer is different for the high- and low-pressure sides, the magnitude shown in these figures is not representative of the steam generator pressure loss, but is indicative of the relative changes.

A gradual performance decrease, which occurred as power was gradually increased from 20%, did not reveal any indication of a problem except higher than expected sodium inlet temperature.

2. Heat-Transfer Effects

The typical change in sodium temperature profile through the heat-transfer zone between the pre- and post-flip conditions is shown in Figure 5-41. In addition to a general rise in sodium temperature, there is a distinct change in the shape of the profile.

Figures 5-42 and 5-43 show the corresponding changes in heat flux and heat transfer coefficient vs area as computed by the one-dimensional method of Section V. B.3. To further clarify these changes, Figures 5-44 and 5-45 show sodium temperature profile and heat flux within the boiling zone plotted vs steam quality for the same flip of Figures 5-41 to 5-43. Note that the number of thermocouple points encompassed by the post flip boiling zone is much larger than the pre-flip number. This reflects the increase in length of the boiling zone.

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Figure 5-39. Time Plot for Sudden Performance Increase in Test Series 4 (2-25-70, 0431 hours)

Figure 5-40. Time Plot for Sudden Performance Decrease in Test Series 5B (2-27-70, 1025 hours)

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Figure 5-41. Sodium Temperature Profiles Before and After Performance Decrease - Test Series 1, 1-2-1970

Figure 5-42. Heat-Flux Curves Before and After Performance Decrease - Test Series 1, 1-2-1970

Figure 5-43. Overall Heat Transfer Coefficient Curves Before and After Performance Decrease - Test Series 1, 1-2-1970

Figure 5-44. Comparison of Pre- and Post-Flip Sodium Temperature Profiles — Test Series 1, 1-2-70

Figure 5-45. Comparison of Pre- and Post-Flip Heat Flux Curves in Boiling Zone - Test Series 1, 1-2-1970

Figures 5-46 to 5-57 show temperature and heat-flux curves for other incidents of performance shift under the planned test conditions of the diagnostic tests (see Table 5-19). As can be seen from these curves and from Figures 5-41 to 5-45, heat-transfer performance in the pre-flip and post-reverse flip conditions, when plotted vs heat-transfer area, exhibits the expected characteristics of tube boiling as discussed in Section V.B.5.

In the post-flip condition, maximum heat-transfer rate occurred at very low steam quality and in some cases in the preheat zone. The heat transfer coefficient dropped rapidly with quality and leveled off at a very low value.

The low performance condition, which was achieved gradually by raising power to 40% in Series 3 and in the plant performance tests, exhibits the same characteristics as those achieved after sudden performance degradation. Figure 5-15 of Section V.B.4 shows heat flux vs heat-transfer area for the 20% (high performance) and 40% (low performance) test conditions of the plant performance tests. The heat-flux curves of Series 3 are shown in Figure 5-18 and Appendix D. Note that the curves resemble the pre- and post-flip curves of Figure 5-42. After the completion of Series 3, operating conditions were changed in order to set up the initial conditions of Series 4. During this operation, the steam generator returned to the high heat-transfer state via a reverse flip. This incident further supports the belief that the gradually achieved and suddenly achieved low performance are of the same nature.

The average boiling zone heat transfer coefficient is approximately 400 to 450 Btu/ft²-hr-°F under high performance conditions. Under low performance conditions, this value is between 100 and 200 Btu/ft²-hr-°F. Intermediate values of about 300 Btu/ft²-hr-°F were also attained at a number of test points in the diagnostic test series.

3. Investigations of Performance Change Incidents

Some of the sudden performance change incidents were investigated in an effort to determine the change in operating conditions which caused them. Results of these investigations are described below.

a. Incident No. 1

Incident No. 1, a flip, occurred on January 2, 1970, at 0425 hours during execution of Test Series 1. The flip occurred with sodium ΔT at 86°F. There

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Figure 5-46. Fluid Temperature Profiles Before Performance Decrease - Test Series 1, 1-9-70, 1745 hr

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Figure 5-47. Fluid Temperature Profiles After Performance Decrease - Test Series 1, 1-9-1970, 1930 hr

7693-59218

Figure 5-48. Fluid Temperature Profiles Before Performance Decrease - Test Series 4, 2-24-70, 1900 hr

7693-59219

Figure 5-49. Fluid Temperature Profiles After Performance Decrease - Test Series 4, 2-24-70, 2345 hr

7693-59220

Figure 5-50. Fluid Temperature Profiles Before Performance Increase — Test Series 4, 2-25-70, 0310 hr

7693-59221

Figure 5-51. Fluid Temperature Profiles After Performance Increase - Test Series 4 (2-25-70, 0515 hr)

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Figure 5-52. Fluid Temperature Profiles Before Performance Decrease - Test Series 5B, 2-27-70, 1000 hr

7693-59223

Figure 5-53. Fluid Temperature Profiles After Performance Decrease - Test Series 5B (2-27-70, 1040 hr)

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Figure 5-54. Heat-Flux Curves Before and After Performance Decrease - Test Series 1 (1-9-70)

Figure 5-55. Heat-Flux Curves Before and After Performance Decrease - Test Series 4, 2-24-70

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Figure 5-56. Heat Flux Curves Before and After Performance Increase - Test Series 4 (2-25-70)

Figure 5-57. Heat-Flux Curves Before and After Performance Decrease - Test Series 5B, 2-27-70

had been no sodium flow increase or other plant control change for a half hour preceding the flip. It may have been triggered by either of two events which slightly preceded it: (1) a slight increase in primary sodium flow, which caused an increase in steam generator sodium inlet temperature; or (2) an instability which developed in the first-stage pressure reducer and caused a low-frequency oscillation of all steam system pressures. A plot of the time change in principal parameters is given in Figure 5-4.

b. Incident No. 2

Incident No. 2 occurred on January 9, 1970, at 1833 hours during repeat execution of Test Series 1. Operating conditions preceding the flip were almost identical to those for Incident No. 1 except sodium ΔT was 91°F. The flip occurred within 2 min after an increase was made in sodium flow in accordance with the test procedure. It was apparently triggered by the slight change in sodium temperature conditions caused by this event. Pre- and post-flip data were presented under Test Series 1 results.

c. Incident No. 3

Incident No. 3 occurred on January 16, 1970, at 2241 hours during rise to power following a plant shutdown. Preceding the flip, the water flow was at 46,900 lb/hr; steam outlet temperature was 650°F; feedwater inlet temperature was 520°F; and steam pressure was 1200 psig. One minute prior to the flip the steam pressure setpoint was raised to 1600 psig and the feedwater temperature setpoint was raised to 570°F. Saturation temperature is 567°F at 1200 psig and 605°F at 1600 psig. This flip was followed within 15 min by a reverse flip. Depending on the relative rate of temperature and pressure rises in response to set point changes, it is possible that the inlet temperature may have approached saturation temperature. It is the reported cause of several flips which occurred during initial SCTI checkout operation in 1966.

d. Incident No. 4

Incident No. 4 occurred on January 19, 1970, at 0305 hours during another rise to power. Steam pressure was 674 psig and steam outlet temperature was 665°F. The flip occurred within 2 min after an increase in sodium flow. This in turn caused a rise in sodium outlet temperature, which appears to have been

the triggering event. When pressure reached its new set point, restoring the margin between inlet and saturation temperature, the reverse flip apparently occurred.

e. Incident No. 5

Incident No. 5 occurred on February 5, 1970, at 0440 hours during operations to establish the initial point of Test Series 3. Steam pressure was 2194 psig (close to the 2200 psig rated). Steam outlet temperature was 980°F. The flip appears to have been triggered by an increase in power with no adjustment in sodium flow. This resulted in a rise in sodium outlet temperature, which is believed to be the triggering disturbance.

f. Incident No. 6

Incident No. 6, a reverse flip, occurred on February 5, 1970, at 1220 hours. Following Flip No. 5, all operating conditions were left unchanged for several hours. Then the steam generator sodium ΔT was slowly increased by reducing sodium flow, with no control change made in any other operating condition. At the start of this process the sodium outlet temperature was 700°F. When the sodium outlet temperature dropped to 682°F, the reverse flip occurred.

g. Incident No. 7

During operations to establish initial conditions for the repeat run of Series 3, several flips and reverse flips occurred in succession. The initial flip was produced by raising power with no adjustment of sodium flow as in Incident No. 5. A regular cycle of flips and reverse flips was then produced by alternately increasing and decreasing sodium flow by 1% of flowmeter full scale.

h. Incident No. 8

Following completion of Series 3, the steam generator was at 40% power in its low-performance condition. Operating condition changes were initiated to establish the start of Series 4. This included reduction of sodium inlet temperature and increase in sodium flow. The low-performance condition persisted through these changes until 1605 hours on February 23, 1970. At that time, high performance was restored by a reverse flip. The incident appears to have been preceded by a slight decrease in power with no change in water or sodium flow.

i. Other Incidents

Flips and reverse flips occurred during tests of Series 4 and 5, as discussed in Section V.B.2.

j. General Observations

From the observations of the incidents stated above, the following appear to hold true:

- 1) Raising sodium outlet temperature with constant feedwater flow and inlet temperature has been shown to produce a sudden performance decrease. Such a temperature rise may be produced either by increasing sodium flow at constant power or by increasing power at constant sodium flow. A temporary sodium outlet temperature increase sufficient to induce a flip may be produced by an increase in primary sodium flow at constant secondary flow and power.
- 2) Reducing overall sodium temperature level with constant feedwater flow while the steam generator is operating at low heat transfer has been shown to produce a reverse flip.
- 3) Raising overall sodium temperature level while concurrently increasing feedwater flow has been shown to result in a gradual performance decrease. As feedwater flow increases, a higher sodium outlet temperature is required to induce a flip. It is not known whether a flip will occur at flows greater than 40% rated.
- 4) Two modes of increasing power have been shown to be capable of avoiding sudden performance decreases. Increasing steam temperature at constant sodium outlet temperature and constant water flow (in Series 5A) resulted in a gradual and partial performance decrease. Increasing water flow with constant steam outlet temperature and approximately constant sodium outlet temperature caused no performance decrease in Series 2.

4. Correlation of Operating Conditions

As has been stated, under certain operating conditions the steam generator heat-transfer characteristics, particularly in the boiling zone, changed either suddenly or gradually. The normal heat-transfer characteristic in the boiling

region is that the heat flux (or heat-transfer coefficient) rises from the single-phase liquid value to the critical heat flux in the low-quality nucleate boiling region, decreases to the Leidenfrost value in the intermediate quality-transition boiling region and increases slowly to the single-phase vapor value in the high quality-film boiling region. The heat-transfer performance associated with the above mechanism is termed here "good" performance. On the other hand, under some run conditions, when the heat flux was plotted against the steam quality or flow length with the assumption of uniform water/steam flow distribution, the heat-transfer curve differed from that of the normal characteristics, particularly in that the nucleate boiling region is not clearly defined, resulting in degradation in overall heat-transfer performance. This latter type of performance is termed here "poor" performance.

All the test runs taken at a water-steam pressure of 2200 psia and with a feedwater inlet temperature of 600°F were classified as either good or poor performance, depending on the shape of the heat-flux curve in the boiling zone. The major variables were then cross-correlated to define the boundaries of regions of good and poor performance.

The major test variables were sodium flow, sodium inlet-to-outlet temperature difference, sodium inlet temperature, and water/steam flow. Careful examination of test results revealed that heat-transfer performance tends to decrease with increasing sodium flow, increasing power (power is proportional to the product of sodium flow and sodium inlet-to-outlet temperature difference), decreasing water flow, and increasing temperature difference between sodium and water near the beginning of boiling. Therefore, the sodium flow (W_{Na}), water/steam flow (W_{H_2O}), sodium inlet to outlet temperature difference $[(\Delta T)_{Na}]$ and temperature difference between sodium and water at the beginning of bulk boiling $[(\Delta T)_B]$ were selected as the major performance variables. The first three variables were the test variables. $(\Delta T)_B$ was calculated by one-dimensional energy balance in the preheating region.

A plot of W_{Na}/W_{H_2O} vs $(\Delta T)_{Na} \cdot (\Delta T)_B$ gave the best grouping of data in terms of performance for all of the test runs, as shown in Figure 5-58. The light and dark symbols represent good and poor heat transfer performance, respectively. The symbols with a flag refer to the cases of sudden performance increase. The 20% and 40% rated plant performance tests taken in May 14-15, 1969, were included in Series 3.

Figure 5-58. Steam Generator Performance Diagram

From this diagram, the following observations may be made:

- 1) When W_{Na}/W_{H_2O} and $(\Delta T)_{Na} \cdot (\Delta T)_B$ were below line A, heat-transfer performance was good. Line A may be represented by

$$\left(\frac{W_{Na}}{W_{H_2O}} \right) [(\Delta T)_{Na} \cdot (\Delta T)_B]^{0.71} = 8.0 \times 10^3$$

This means that the condition of good performance is

$$\left(\frac{W_{Na}}{W_{H_2O}} \right) [(\Delta T)_{Na} \cdot (\Delta T)_B]^{0.71} < 8.0 \times 10^3$$

where W_{Na}/W_{H_2O} is dimensionless, and $(\Delta T)_{Na} \cdot (\Delta T)_B$ is in $(^\circ R)^2$.

- 2) When W_{Na}/W_{H_2O} and/or $(\Delta T)_{Na} \cdot (\Delta T)_B$ increased to above line B, poor performance resulted. This region may be expressed by

$$\left(\frac{W_{Na}}{W_{H_2O}}\right) \left| (\Delta T)_{Na} \cdot (\Delta T)_B \right|^{1.0} > 2.1 \times 10^5$$

- 3) In the region between lines A and B, performance may be in a transitional state and very unstable.
- 4) When the process was reversed, i.e., when W_{Na}/W_{H_2O} and/or $(\Delta T)_{Na} \cdot (\Delta T)_B$ decreased from very large values at which performance was poor, the region of poor performance extended to line C (passing line B) due to hysteresis effects, and stable good performance points were found at line D. Lines C and D may be expressed by

$$\text{Line C: } \left(\frac{W_{Na}}{W_{H_2O}}\right) \left| (\Delta T)_{Na} \cdot (\Delta T)_B \right|^{1.0} = 1.6 \times 10^5$$

$$\text{Line D: } \left(\frac{W_{Na}}{W_{H_2O}}\right) \left| (\Delta T)_{Na} \cdot (\Delta T)_B \right|^{0.44} = 4.5 \times 10^2$$

Since only four data points were considered for this reversed process, no conclusive observations can be made.

The above observations indicate that as W_{Na}/W_{H_2O} and/or $(\Delta T)_{Na} \cdot (\Delta T)_B$ increase, chances of heat-transfer performance being poor increase for the ranges of test conditions considered for the steam generator.

E. ANALYSIS OF CAUSES OF PERFORMANCE SHIFTS

This section discusses hypotheses offered to explain the heat-transfer performance shifts described in the previous section. The fundamental difficulty in efforts to explain these shifts is that the internal instrumentation, while

capable of providing a well-defined profile of the mixed-mean sodium temperature, could not indicate either tube-to-tube variations in performance or rapid fluctuations in performance with time. The limited test data permitted several alternate hypotheses including the following:

- 1) Film boiling — Inception of film boiling at very low steam quality can result in poor heat transfer.
- 2) Static flow instability — Gross flow maldistribution in the steam/water tubes can cause overall performance drop. This means that the exit steam temperatures of some tubes decreases extensively from the temperature which would be attained under a uniform flow distribution condition, while other increase slightly. The net effect is a decrease in the mixed-mean exit steam temperature and decrease in overall performance.
- 3) Dynamic flow instability — When the water/steam flow oscillates between the lower and upper limit values (flow oscillation), the CHF point in the boiling zone tends to move upstream, thus shortening the nucleate boiling region. This, of course, lengthens the film boiling region to induce a performance decrease.

These hypotheses are discussed further in the following sections.

1. Film Boiling

Inception of steady-state film boiling at low steam quality could be expected if the tube inner wall temperature were too high to permit nucleate boiling. This possibility was investigated by calculating the trend of the inner wall temperature in Series 1 as flip conditions were approached.

The wall temperature at the point of the maximum heat flux (critical point) was calculated from the temperature profiles of Tests 1A and at the condition immediately preceding the flip. By all indications, the conditions in Test 1A were remote from the pre-flip conditions. One dimensional parameters at the critical point for the two cases were as follows:

Parameters	Test 1A	Pre-flip (1-2-70)
Heat Flux (Btu/ft ² -hr)	84,500	53,750
Location (% area)	68	32
Sodium-water (ΔT)	115.7	60.8
Steam quality (%)	46.5	47.5

The inner, tube wall temperature was calculated for the two cases on the basis of equal assumed values of the tube wall and sodium film thermal properties; results were as follows:

Test 1A

Wall temperature = 696.5°F

Wall superheat = 46.5°F

Pre-flip (1-2-70)

Wall temperature = 666.0°F

Wall superheat = 16.0°F

Thus the comparison indicates that the wall temperature and wall superheat at conditions preceeding a flip were considerably lower than at conditions remote from a flip.

Similar comparisons were made for other series with the same results being found.

For the post-flip conditions, similar calculations indicated that relatively high tube inner wall temperature exists over most of the boiling zone, with typical wall superheat values between 100 and 200°F. Film boiling could be expected at this conditions.

This study thus indicates that film boiling may not be the initial cause of performance decrease although it may be the end result. It is noted, however, that film boiling may be induced with the occurrence of dynamic flow instability (discussed in Section V.E.3).

2. Static Instability

It has been known for many years that under certain operating conditions the pressure drop-flowrate relation of boiling fluid has the shape shown in

Figure 5-59. The significance of this curve is that, under a certain range of pressure drops, there are three possible flowrates (two stable points and one unstable point) for a given pressure drop. In the region between points A and B, an increase in the flowrate decreases the pressure drop, which is an opposite phenomenon to a single-phase pressure drop-flowrate relation. In this region of the negative derivative of the pressure drop-flowrate relation, the flow is unstable (Ledinegg stability criterion). For example, the unstable operating point C in Figure 5-59 moves to either point D or point E where the derivative of the pressure drop-flowrate relation is positive. This means that for a multiple tube flow system in which a constant pressure drop from the inlet to outlet phenums is imposed, some tubes may take more (or less) flowrate than others, resulting in a flow maldistribution. This is known as static flow instability.

Static flow instability was considered as a possible explanation of low performance which may have been the mixed-mean resultant of maldistribution of heat transfer among tubes.

The steam generator contains no tube inlet orifices. Orificing is the method generally used to increase the slope of the pressure drop-flow characteristic and thus attain stability.

To investigate static instability, a bistable flow computer model was formulated. The purpose of this model was to determine whether the measured test data could be explained by bistable operation, i. e., by the existance of two groups of tubes having different (stable) flows and temperatures but the same pressure drop. For this purpose the model simulated two families of tubes having independent water/steam-side conditions but transferring heat with a common sodium stream. This was considered a good approximation of the Alco/BLH steam generator in which mixing at each baffle pass tends to eliminate any sodium-side asymmetries.

Figure 5-59. Pressure Drop-Flowrate Relation Under Certain Two-Phase Flow Conditions

It was assumed that both families of tubes follow a common set of heat-transfer and pressure-drop laws. Details of the model are presented in Appendix A. The model could also be operated in a single tube mode in which it could generate the conventional pressure drop vs flow stability characteristic of the type shown in Figure 5-59.

A limited number of runs of the model were made in both the single tube and bistable versions. The single tube version was used to calculate the flow vs pressure drop characteristics for four sets of operating conditions. They are:

- 1) 20% rated power run obtained at 1330 hr on May 14, 1969, in the plant performance test series, which resulted in good performance;
- 2) 40% rated power run obtained at 0415 hr on February 19, 1970, Run 12 of Diagnostic Test Series 3, which resulted in poor performance;
- 3) Pre-flip conditions taken at 0330 hr on January 2, 1970, occurred when decreasing sodium inlet-to-outlet ΔT reached 86°F in Diagnostic Test Series 1 (see Incident 1 of Section V.D.3); and
- 4) Post-flip conditions taken at 0445 hr, January 2, 1970, for the same flip described above.

The results, covering 80 to 120% of nominal water flowrate (W_O), are shown in Figures 5-60 to 5-63. Each figure has two curves, one for total pressure drop and the other for frictional and momentum pressure drop. The difference accounts for the pressure drop due to elevation. These figures show the positive slope in the ΔP -flow curve, indicating the stable state of flow. It is interesting to note that the water/steam flow could have been statically slightly unstable at the 20% rated plant performance test if the unit had been mounted horizontally, as indicated from the curve of friction/momentum pressure drop in Figure 5-60. It is evident that vertical upward flow orientation strongly stabilizes the water/steam flow. All the curves of Figures 5-60 to 5-63 show that the pressure drop due to elevation is a substantial portion of overall pressure drop, which helps stabilize the flow. Also no significant change in stability was found between pre-flip and post-flip conditions.

Figure 5-60. Flow- Δp Characteristic for 20% Rated Power Run (5-14-70, 1330 hour)

Figure 5-61. Flow- ΔP Characteristic for 40% Rated Power Run (Test 12, 2-19-70, 0415 hours)

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Figure 5-62. Flow- Δp Characteristic for Pre-Flip Conditions in Test Series 1 (1-2-70, 0330 hours)

Figure 5-63. Flow- Δp Characteristic for Post-Flip Conditions in Test Series 1 (1-2-70, 0445 hours)

It should be noted that the water-side heat-transfer correlations used in the model are those based on normal boiling characteristics (Section V.B.5 and Appendix A). Therefore, the absolute values of Figures 5-61 and 5-63 for which poor performance resulted, may be different from the actual values if poor performance was accompanied by changed heat-transfer characteristics.

The bistable version was applied at the low-performance 40% power condition reached at the end of Series 3 (Test 12). Various divisions of flow and number of tubes between the two families of tubes were investigated. Several assumptions of tube roughness were made as this factor affects tube pressure drop. For all conditions investigated, the model yielded higher pressure drop for the tube family having higher flow per tube. This result indicated stability.

3. Dynamic Instability

Dynamic instability is defined as periodic oscillation of pressure drop, flow, etc., within a tube. It appears possible for this phenomenon to occur in the steam generator without being detected by the available instrumentation. This would be especially true if many tubes were oscillating at varying phase and frequency so that average heat transfer to the sodium showed very little fluctuation.

Analysis of dynamic instability for the Alco unit was limited to two brief studies, one by Gulf General Atomics (GGA), the other by Westinghouse. Both studies used computer methods in which the basic equations of conservation were linearized, with respect to time, for small disturbances from the steady-state conditions, and the resulting equations were Laplace-transformed. The linearized, transformed equations were then used to construct a system transfer function which determines the system stability according to the Nyquist criterion. This method can predict approximate frequency of an instability but cannot predict amplitude. Both of these studies indicated regions of unstable operation within the envelope of operating conditions. The operating conditions covered in these studies and the degree of simulation to the actual configuration were too limited to permit any firm conclusions. However it appears likely that a more extensive analysis would demonstrate instability with greater certainty.

Dynamic instability may not account for low performance if steam parameters oscillate slightly about a normal heat-transfer condition. Time-averaged

heat transfer would vary little from its steady-state value. However, if the magnitude of the oscillation is appreciable, the CHF point may tend to move upstream. To illustrate this, assume the water/steam flow oscillates between the upper and lower limit values. Let Z_{U1} be the CHF location, measured from the feedwater inlet, when the water/steam flow is at its upper limit value. This means that at this moment the water/steam-side wall temperature is relatively low upstream of Z_{U1} and relatively high downstream of Z_{U1} , because the evaporation process is characterized by nucleate boiling upstream of the CHF location and by film boiling downstream of it. As the flow decreases from its upper to lower limit values, the CHF point moves upstream to the new location Z_{L1} ($Z_{L1} < Z_{U1}$) because of earlier evaporation with decreasing water flow. The wall temperature in the region between Z_{L1} and Z_{U1} , which was relatively low previously, attains a relatively high temperature due to its new location downstream of the new CHF point. When the flow increases to its upper value, the CHF location does not return downstream fully to its original location of Z_{U1} . This is because the region between Z_{L1} and Z_{U1} , which had a relatively high wall temperature associated with (partial) film boiling at reduced flow, tends to keep its low-performance film boiling status even with increasing flow. In other words, the transition process from film to nucleate boiling is more difficult to bring about than the reverse transition from nucleate to film boiling. The new CHF location Z_{U2} with the water flow at its upper limit value in the second cycle, tends to be less than Z_{U1} . The CHF location Z_{L2} with the water flow at its lower limit value in the second cycle is approximately $Z_{U2} - (Z_{U1} - Z_{L1})$ and thus less than Z_{L1} . In this fashion, the CHF location gradually moves upstream with oscillatory flow. The consequence is lengthening of the film boiling region and reduction of heat-transfer performance.

In view of the fact that analysis has not supported alternate hypotheses, this theory of dynamic flow instability-induced film boiling appears to merit further investigation.

F. INTERNAL INSTRUMENTATION

The internal instrumentation was limited to thermocouples which measure the mixed-mean sodium temperature at any level in the heat-transfer zone.

There was no capability of measuring transverse maldistribution/oscillation of flow or temperature on either the steam or sodium side.

Serious consideration was given to installation of internal instrumentation to measure water/steam-side flow maldistribution/oscillation. Instruments considered include hot-wire anemometers at tube inlets to measure flow maldistribution, and thermocouples at tube outlets to measure temperature maldistribution/oscillation. However, following the discovery of recurrent tube leaks in March 1970, it was concluded that life expectancy of the steam generator did not justify the cost of such instrumentation.

G. TUBE ORIFICING

The steam generator as built and tested contained no tube inlet orifices. Inlet orificing is the most straightforward method of correcting flow instability. By installing orifices with sufficient pressure drop it should be possible to provide assured uniformity of flow among tubes, and to subdue any flow oscillation. Orifice installation is a much simpler and less costly procedure than adding internal instrumentation.

It was therefore decided to install inlet orifices. An orifice diameter of 0.0625 in. was selected. The orifices were manufactured and installed after first being subjected to individual flow tests. These tests indicated that the orifices would produce a pressure drop of approximately 120 psia at rated flow and inlet temperature, which is about three times the calculated tube pressure drop. Statistically, the flow tests indicated a $\pm 4\%$ variation of flow tube-to-tube. Discovery of cracks and retirement of the steam generator precluded conducting tests with orifices. Consequently, experimental resolution for flow instability as a cause of performance shift has not been possible. Details of the orifice installation are described in Section VI.A.6.

VI. STRUCTURAL PERFORMANCE

A. STRUCTURAL FAILURES, REPAIRS, AND MODIFICATIONS

1. Manufacturing Anomalies

In assembling the tubes into the guide ring, it was first attempted to expand the tubes into the guide-ring holes. However, after this was done for some tubes in the inner row, expanding tools broke and became lodged in a tube. The tools had to be removed with nitric acid. This malfunction occurred in tubes I-2, I-5, I-21, and I-28 (see Figure 6-1 for tube identification). It was then decided to tack weld the tubes to the guide ring rather than expand them.

Two holes in the upper tube sheet (II-1 and IV-35, Figure 6-1) were found to be out of tolerance in diameter. They were repaired by counterboring to a depth of 1/2 in., inserting a section of 5/8-in. tubing, expanding it into the counter bore, applying Inconel overlay, and machining the inside diameter to size.

While evacuating the shell for leak tests, a small amount of vacuum pump oil was found in the piping and in the steam generator. After draining the oil (which amounted to less than a cupful) the interior was flushed four times with a naphtha solvent. The vacuum pump oil was analyzed for chlorides and found to contain 340 ppm. The solvent contained less than 10 ppm after flushing.

In the steam generator assembly process, several tube internal anomalies occurred as a result of tack welds and tube rolling operation (refer to Figure 6-1):

- 1) Tubes I-2, I-5, I-6, I-21, and I-28 showed indentation in the inside walls caused by broken rollers or cages during the tube expanding operation. In three of the tubes, the broken tooling had to be removed by acid, but the records do not specify which three of the five required acid.
- 2) Tubes I-22, I-27, and IV-41 had fuse-through conditions in the tack weld to the guide ring assembly. These tubes were plugged at the factory.

Baldwin-Lima-Hamilton Corporation, the fabricator of the steam generator, has prepared a report (Reference 9) which outlines the fabrication and inspection

Figure 6-1. Upper Tube-Sheet Diagram Showing Orientation of Defective Tubes and Plugged Tubes (SRI Survey)

techniques used on the unit. The report also includes photographs of the unit and subassemblies during various stages of fabrication.

It is also noted that internal thermocouple ends were unsupported in the sodium flow areas of the tube bundle. The 1/16-in. diameter thermocouple lead was bent on a long radius (2 to 3-in. radius) to place the thermocouple end in the sodium flow path. The length of the unsupported thermocouple end was 6 to 12 in.

2. Water Header Pitting

An inspection of the steam generator in December 1968, included visual and dye-penetrant examination of the inside surfaces of the feedwater header. These indicated a group of pits localized in various areas: (1) at the periphery of the access port; (2) the water inlet; and (3) the tube sheet/water header joint. The nominal depth of this localized and random pitting was 0.015 in., with none exceeding 0.020 in. The shallow pits were smooth in appearance, either round or flat-bottomed, with no evidence of adjacent cracking. Based on the passive (dormant) appearance of the pits, it was concluded that the potential for any significant depth increase was very low (this observation was concurred with by a representative from the International Nickel Company). The condition was not considered significant since the Inconel overlay thickness is nominally 1/8 in.

Following discovery of the lower tube-sheet cracks in 1970, it was observed that one of the previous pitted areas (item 3 above) coincided with a portion of the crack area on the external tube-sheet surfaces. However, these visual examinations did not confirm the existence of a leak path between the pitting and the external crack network.

3. Tube Leaks

a. 1969 Tube Leaks

On June 22, 1969, following an indication of high hydrogen concentration in the cover gas, a leak was discovered in steam tube II-47 (Figure 6-1) at a point approximately 6-1/4 in. below the upper tube-to-tube sheet weld. The leak was pinpointed first through a helium leak check and later an eddy-current test technique. The tube crack location was further defined using an optical borescope

and a siligel casting technique. In addition to inspection of the defective tube, the various tube-to-tube sheet welds were dye-penetrant inspected; this revealed a crack in the fillet weld on tube IV-27, plus several minor flaws in the tube-sheet surface and a minor weld crack in tube II-48 adjacent to the leaking tube.

The discrepancies in tubes II-47, and IV-27, identified above, were repaired by machining the tube ends flush with the tube sheet, and sealing the two tubes with welded Inconel plugs. Figure 6-2 shows the plug design and installation. The

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Figure 6-2. Tube Plug Design

surface defects were polished out with no welding required. A complete record of the defects and the associated repairs is contained in an LMEC topical report (Reference 10). It was not feasible to remove the tubes for examination as no cutting tool small enough to fit the tube inside diameter could be found.

The location of the leak in tube II-47 with respect to the upper tube-sheet thickness, is clearly shown in Figure 6-3. It is noted that the leak occurred at a position within the tube-sheet thickness.

b. 1970 Tube Leaks and Cracks

In March 1970, when the steam generator was shut down after moisture was found in the cover gas, the tubes were again inspected for leaks. Leaks were discovered in tubes III-52 and III-53. In both cases the leak location was slightly below the end of the 1-1/2-in. tube section which is expanded into the tube sheet. The Southwest Research Institute (SwRI) eddy-current equipment, which had proved to be highly successful in locating the leaks, also indicated the existence of three tubes with exterior cracks (or material deficiencies) which did not penetrate to the inside diameter. A summary of the defects found is as follows:

Tube Identification	Defect Location (in. below top of tube)	Defect Magnitude (% of tube wall)
III-52	1-1/2 to 1-3/4 (leak check) 1-7/8 (ec)*	100 (leak check) (ec) [†]
III-53	1-3/4 (leak check) 1-7/8 (ec)	100 (leak check) (ec) [†]
II-48	6-3/8 (ec)	90 (ec)
III-51	8 (ec)	60 (ec)
I-42	8 (ec)	60 (ec)

*ec = eddy current

†Magnitudes could not be established by eddy-current tests since the defects occur at the bottom of the tube roll into the tube sheet. The changing geometry in this region perturbs the output signal, permitting only the identification of an anomalous condition through comparisons with signals obtained from normal tubes.

Figure 6-3. Axial Locations of Tube Defects

Figure 6-3 shows the various defects located in elevation. Figure 6-1 shows their orientation in plane view. Note that the three leaking and three cracked tubes lie in a compact cluster, and that all the defects lie within a short span of tube bounded by the expanded section and the guide ring. Also, three of the tubes involved also had tube-sheet weld cracks.

The tube ends and the tube sheet were also dye-penetrant inspected. Approximately 25 surface defects were noted, all but 3 of which were superficial. Based on an apparently limited test life remaining for the generator, it was determined that a simple "butter-welding" technique would be utilized to repair these three defects. This effort was completed without incident.

Plugging of the faulty tubes (identified above) was begun, using the same technique previously employed. The lower end of tube III-53 was successfully plugged; however, the remainder of the tubes could not be plugged because of the presence of sodium in the tube-to-tube sheet annulus, which melted and ran along the tube wall into the weld area. Several different approaches to alleviating the contamination problem were tried. A "hollow-head" plug was fabricated to provide a small containment volume for extruded sodium, and allow additional time for completion of the seal weld. This was not successful because of the small size of the plug and the limited volume which could be made available without extensive plug redesign. A second approach involved local heating of the tube-to-tube sheet annulus in an attempt to melt out and eliminate residual sodium. This too was unsuccessful. An alternate design, in which 3/8 in. of the tube end was removed, the tube-sheet inside diameter cleaned, and a larger plug inserted, was utilized for the other four tubes. The upper tube ends were plugged as before. During this downtime, design and fabrication of special orifices for the feedwater inlet end of the steam generator tubes was completed. The lower tube ends were reamed to receive the tapered surface of the orifice bodies, and all tube orifices were installed and tack welded in place.

Upon completion of the orificing and tube plugging activity, the steam generator was cleaned and closed in preparation for hydrotesting. It was during initiation of the hydrostatic test that the feedwater header cracks were discovered.

4. Manway Defects

Early in August 1969, as the tube repair effort was nearing completion, the ring-joint metal O-rings were removed from the steam generator manway flange openings. These rings had been left in place during the early repair activities to protect the ring-joint grooves. Following removal, inspections of the ring-joint grooves revealed extensive circumferential cracking at the upper flange. Crack distribution and severity is illustrated in Figures 6-4 and 6-5. Cracks occurred at the base of the groove and near its inner diameter. The inner lip of the flange was found to be deformed (radially inward) increasing the side angle of the groove by an estimated 3 to 6 degrees. Similar deformation was evident on the inner lip of the lower manway flange ring joint, but no cracking was found in this region.

Various inspection techniques were applied to establish the severity of cracking. Radiographic inspections were unsuccessful; but ultrasonic testing did indicate a crack penetration of approximately 125 mils. Similar penetration was obtained from the insertion of 3-mil wire probes into the largest of the cracks. Subsequent to these observations LMEC recommended controlled machining of selected crack regions to establish the actual depth and direction of propagation of the cracks. Representative photographs obtained during the incremental machining procedure are included as Figures 6-6 and 6-7.

Initially, three crack regions were selected for exploratory machining. Machining was accomplished in increments of 10 mils and later in larger increments of approximately 25 mils. After the first few exploratory studies the cleanup machining was extended to encompass all the cracks evident in the manway flange. With few exceptions, most of the cracks were eliminated near a depth of 125 mils. However, the largest crack, at the 2 o'clock position, extended to a maximum depth of approximately 195 mils. Throughout the machining operation, chip samples were retained and were submitted for analyses to determine the extent of penetration through the transition zone of the Inconel overlay material, and to protect against penetration into base metal. The information in Table 6-1 is a representative survey of the chip constituents as a function of penetration depth. Data obtained at the 2 o'clock position were indicative of results associated with maximum penetration. As can be seen,

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Figure 6-4. View of Top Manway Flange Showing Cracks in the Bottom of the Groove

6UN37-8/29/69-S1A

Figure 6-5. Closeup of Top Manway Flange Crack at the 2 O'Clock Position

6UN37-8/29/69-61B

Figure 6-6. Closeup of Top Manway Flange Crack at 2 O'Clock Position After Machining to a Depth of 0.025 in.

6UN37-8/29/69-S1G

Figure 6-7. Closeup of Top Manway Flange Crack at 2 O'Clock Position After
Machining to a Depth of 0.125 in.

the crack removal operation was accomplished entirely within the overlay transition zone. The maximum concentration of stainless steel in the chip samples was of the order of 40 to 45%.

During the course of the crack elimination efforts, a boat sample containing a complete crack segment was obtained. The sample was procured from a crack at the 5 o'clock position (Figure 6-8). Metallurgical examinations of the sample showed subsurface microfissuring, predominantly in the first layer of overlay material. A macroscopic view of the boat sample and several photomicrographs are included as Figures 6-9 through 6-11. The large, tensile-type crack which propagated through to the overlay surface (Figure 6-10) apparently was initiated from one of these original microfissures.

Machining of the ring joint for crack elimination was accomplished using a 1/4-in.-diameter end mill. Following crack removal, the machined surfaces were hand ground to eliminate vertical walls and sharp corners and establish an acceptable configuration for the overlay repair welds. These areas were then covered with an Inconel overlay to reestablish a full protective layer above the base metal. The final appearance of the overlay repair is shown in Figure 6-12. As described below, the method of sealing the manway flanges was changed to eliminate the ring joint. Consequently, no machining of the overlay repair areas was required, other than surface polishing to permit completion of dye-penetrant weld acceptance tests. Following completion of the welding activities, the manway flange surfaces, the overlaid region of the manway covers, and the inside surfaces of the manway flange openings were inspected. High resolution, post-emulsified fluorescent dye penetrant was used to investigate the possibility of crack network propagation within the overlay material. Results were satisfactory; no cracks or crack networks were detected.

Consideration was given to several alternatives for repair and sealing of the steam generator manway flanges. Initial attention was concentrated on rehabilitation of the original ring-joint seal, and on the addition of a stiffener ring inside the manway flange to avoid recurrence of the distortion problem. This approach would have necessitated extensive field machining on the steam generator and would have been extremely costly and time-consuming to implement. The second approach, which was finally selected, was based on utilization of a

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Figure 6-8. View of the O-Ring Seal Groove on Top Manway Flange

TABLE 6-1

ANALYSIS RESULTS ON TOP MANWAY
FLANGE CHIP SPECIMENS

Depth (mil)	Chip Constituents (%)					
	Mo	Nb	Ni	Fe	Cr	SS*
0.25	0.4	2.0	68	15	14	18
0.75	0.5	2.0	67	16	14	20
0.100	0.7	1.9	62	20	15	26
0.110	0.9	1.5	55	27	15	37
0.125	0.9	1.6	54	28	16	38
0.137	0.9	1.8	57	25	15	34
0.150	0.9	1.7	55	27	16	37
0.180	1.0	1.5	51	31	15	43
0.195	0.9	1.5	54	29	15	40

*The percentage of stainless steel was obtained from measured percentages of iron and molybdenum in the chip analyses, ratioed to the expected percentages of these constituents of Type 316 SS.

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Figure 6-9. Macrograph of Crack in the O-Ring
Groove of Top Manway Flange (Mag: 8X)

LMEC-Memo-70-20

Figure 6-10. Microstructure of the Inconel Overlay of Top Manway Flange Crack Segment (10% Oxalic Etchant, Mag: 100X)

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Figure 6-11. Microstructure of the Inconel Overlay of Top Manway Flange Crack Segment in a Section Without any Visible Surface Crack (Mag: 75X)

seal-welded diaphragm. Several diaphragm designs were analyzed, including convoluted designs which would avoid shear loadings on the seal weld, but the final selection was a flat plate with a reduced-thickness central section to minimize radial loads. In addition, the sealing area was designed to provide sufficient frictional resistance to minimize shear loadings at the seal weld. The final diaphragm configuration is illustrated in Figure 6-13. This design utilized 16-mil-deep serrations on the sealing surface to improve sealing capability and to enhance the frictional resistance between the diaphragm and the mounting flange. The outside surfaces in contact with the manway covers received a fine finish and were lubricated with graphite powder, prior to placement of the manway covers, to accommodate relative motion caused by differential thermal expansion. The sealing surfaces of the manway flanges also were improved by lapping to increase the probability of achieving an effective seal. The final condition of this sealing surface (outside the ring-joint groove) can be seen in Figure 6-12.

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Figure 6-12. Final Appearance of the Overlay Repair of Top Manway Flange

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Figure 6-13. Diaphragm Cover Seal Installation of Top Manway Flange

Two manway closure diaphragms were fabricated and, following extensive cleaning of the steam generator tubes and the steam and feedwater headers, were tack welded to the manway flanges. Steam and feedwater header conditions prior to diaphragm installation are shown in Figures 6-14 and 6-15. The manway covers were then installed and a hydrostatic test was performed to a maximum pressure of 2225 psig to demonstrate the sealing effectiveness of the diaphragm as a simple gasket. The hydrostatic test was successful; no evidence of leakage was observed. The covers then were removed to permit completion of the full seal weld (Figures 6-16 and 6-17). The high resolution, dye-penetrant examinations were repeated after seal welding; again, no significant indications were obtained. Flange repair efforts were completed with the final installation of the manway covers.

5. Thermocouple Failures

Of the approximately 28 sodium temperature-sensing thermocouples covering the heat-transfer zone, all were apparently operating during the 1966 testing period. Approximately 7 failed before or during the 1969 operating period; approximately 18 were working at the start of the 1970 diagnostic test period; and most of these failed within a period of less than 12 hr during February 8 and 9. The few remaining thermocouples in this group failed within the next few weeks. As noted in Section III.C, they were replaced for functional purposes by externally mounted thermocouples. The time of the mass thermocouple failure coincided with an interval of operation at high sodium-inlet temperature, low heat-transfer performance, high cover-gas hydrogen concentration and high secondary-sodium plugging temperature. The temperature recorder was down during the period, precluding a more exact fixing of the time of failure.

6. Orifice Installation

During the 1970 tube repair interval, special attention was given to modification of the steam generator to procure the maximum information during a minimum test period. This requirement resulted from the assumption that additional tube failures could be anticipated and that the remaining life of the steam generator was limited. To satisfy this objective, emphasis was given to the installation of individual tube orifices in the feedwater end of the steam generator. This action was intended to provide an orifice pressure drop significantly higher than

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Figure 6-14. Steam Header Condition Prior to
Manway Diaphragm Installation

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Figure 6-15. Feedwater Header Condition
Prior to Manway Diaphragm Installation

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Figure 6-16. Manway Flange Diaphragm Installation

7693-5994

Figure 6-17. Manway Flange Diaphragm Seal Weld

that associated with friction loss in the tube, thereby assuring more uniform tube-to-tube flow distribution and damping any flow oscillation that might have existed in the previous tests. This, in turn, would reduce or eliminate possible instabilities which might have caused adverse thermal performance.

Extensive effort was devoted to the design of an orifice and an installation method. The final configuration of the orifice is illustrated in Figure 6-18. This design was predicated on achieving a full-flow pressure drop of approximately 120 psi; the equivalent of three times the tube pressure drop at full flow (116,000 lb/hr). This characteristic was verified in an orifice calibration test installation. Detailed hydraulic data were obtained for an initial set of 10 orifices to define a mean characteristic. Additionally, single point flow checks were performed on all orifices to verify that deviations from the mean flow value did not exceed 5%.

Installation of the orifices was based on a tapered press fit, with a single tack weld after installation to assure orifice retention during thermal transients. Only one orifice was defective after installation, the result of a slightly oversized shank and a non-axial insertion into the tube end. This orifice was removed and a replacement of the correct size was installed. The new orifice was seal welded to the tube end to counteract the possibility that the stainless steel tube end might have been exposed during removal of the defective orifice.

The subsequent discovery of cracking in the steam generator feedwater header during the initial stages of the hydrostatic test, and the decision to remove the unit for destructive examination, precluded any operational testing or evaluation under orificed conditions.

7. Lower Tube-Sheet Cracks

On May 20, 1970, following completion of repair and modification activities within the steam generator and closure of the manways, water was admitted to the steam generator. Water temperature was 90°F and internal pressure was 600 psig. No increase in shell-side hydrogen concentration occurred following the initial fill. Hydrogen levels in the cover gas were 10 ppm or less. At this point, the upper and lower diaphragm regions were visually inspected to verify proper seating before increasing system pressure. The upper manway diaphragm was found to be leaktight; however, water was observed leaking from

Figure 6-18. Tube Orifice Installation (Typical)

the region of the lower tube sheet at the lower end of the steam generator. Since the water source was above the diaphragm location, it was apparent that a significant problem existed within the steam generator. Water circulation was terminated immediately and the steam generator was drained. Local insulation was removed to provide access to the tube-sheet region and to permit visual observations. Several large cracks were observed on the tube-sheet edge and in the radius region above the weld to the lower channel head. Deposits of sodium oxide were also visible on the undersurface of the tube sheet. These deposits and several of the major cracks are shown in Figures 6-19 and 6-20. A developed view of the tube sheet, showing the approximate extent and location of the cracks, is included as Figure 6-21.

The damage observed while the steam generator was in place is described as follows:

- 1) A predominantly vertical crack, approximately 5 in. long, was evident on the edge of the tube sheet at approximately 340 degrees. (This crack extended upward from bottom of the tube sheet and showed several branches from the main stem approximately 1 in. from top. A second crack, approximately 0.5 in. long, was also present near the bottom edge at 345 degrees).
- 2) A major crack network of 6 or 7 segments extended approximately 60 degrees on either side of 0 degree at the fairing between the tube sheet and weld to the lower channel head. The overall network was circumferential, but individual segments also showed vertical components. Two significant deposits of yellow material existed in this region below the tube sheet. The details of chemical analysis of these deposits are shown in Appendix C. Other cracks gave evidence of past water leakage in the form of rust-colored stains running down vertically from the crack location.
- 3) There was no visual evidence of cracking in the steam generator shell directly above the tube sheet.

On June 3, 1970, a preliminary metallurgical examination was performed with the primary objective of characterizing the large crack on the periphery of the tube sheet. In addition, the Brinell hardness number (BHN) of the tube

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Figure 6-19. Lower Tube-Sheet Deposits
and Crack Indication

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Figure 6-20. Cracking on Undersurface of
Lower Tube Sheet

Figure 6-21. Location and Extent of Lower Tube-Sheet Cracking

sheet was measured at two locations: adjacent to the crack, and in an unaffected area. The examination procedure and findings are described in the following paragraphs.

a. Preparation (Metallurgical)

All operations were performed by hand and confined to a small area around the crack. Preparation methods included rough grinding and polishing, using a range of abrasive paper (240 to 600 grit), and final polishing with 3- μ alumina. Marble's reagent, the etchant used was applied by swabbing at ambient temperature. To remove the maximum of disturbed surface metal, alternate polishing (alumina) and etching procedures were employed. Distilled water rinses followed each stage of preparation.

The area prepared for examination was approximately 2 in. by 1 in. where the long dimension paralleled the crack. Optical equipment employed included a 20X-power portable microscope (Brinell) and a pocket magnifying glass (2X).

Preparation for hardness testing involved hand polishing two 3-in.² areas with 320-grit abrasive paper. This was effective for removing minor surface irregularities (machining grooves) and minor discoloration produced during operation.

b. Observations and Measurements

The character of the main crack, with respect to intergranular or transgranular fracture, could not be determined. However, the absence of deformation/ductility in the fracture was evident.

Numerous small secondary cracks, branching from the main crack were found. One localized network originating from a secondary crack appeared to be intergranular in character.

The general appearance of the entire crack system and the absence of pitting indicate that the cracks terminated, rather than originated, at the external surface of the tube sheet.

The overall appearance of the crack system is similar to a type associated with corrosion in austenitic stainless steels, specifically the branchlike cracking associated with stress corrosion.

The hardness measurements do not indicate any significant variation with respect to location. In the crack area, the BHN ranged from 140 to 153, an average of 145.6 (79 Rb); in an unaffected area, the average BHN was 143.6 (79 Rb) and a range from 139 to 148 BHN.

c. Conclusions

On the basis of these observations and measurements, it is concluded that:

- 1) The fracture type (intergranular or transgranular) of the main crack could not be positively identified.
- 2) The fracture is brittle rather than ductile in character.
- 3) Cracking did not originate at the external tube-sheet surface.
- 4) The hardness in failed and unfailed areas is similar.

8. Steam Generator Support Anomalies

The following conditions regarding the steam generator support interface were observed during removal of insulation and investigated in the post-test examination.

- 1) The lower tube-sheet extension and support ring were out of contact for a portion of the periphery.
- 2) Three of the four holddown pads were not seated on the lower tube-sheet extension.

The damage and defects of the steam generator at the time of retirement are summarized in Appendix E.

B. INPLACE EXAMINATION

1. Plan

The following inplace examinations or operations were planned (most of these, and some additional examinations, were performed; they are discussed in the subsequent section):

- 1) Remove all insulation from steam generator.
- 2) Dye-penetrant inspect region of cracks on exterior; use "chloride-free" penetrant.

- 3) Perform helium leak checks for sodium-side-to-water-side leakage, sodium-side-to-exterior leakage and water-side-to-exterior leakage.
- 4) Perform overall visual inspection.
- 5) Measure clearances to shell for three holddown pads which do not contact shell.
- 6) Measure lower tube-sheet extension support-ring interface, noting contacting segments and measuring clearance of non-contacting sections.
- 7) Remove holddown screw assemblies, noting the number of turns required to release the one contacting holddown pad.
- 8) Sample the following metal surfaces for chlorides or other contaminants:
 - a) Steam header
 - b) Feedwater header
 - c) Feedwater tube-sheet extension exterior
 - d) Shell exterior where there are deposits or evidence of corrosion
- 9) Clean the above areas for dye-penetrant test; also all nozzle-to-piping welds.
- 10) Dye-penetrant inspect all cleaned areas.
- 11) Perform chemical analyses of samples of removed insulation and of surface deposits.
- 12) Photograph all dye-penetrant-inspected areas.
- 13) Cut, inspect, add inert gas, and seal all lines. The cuts will be made outboard of the nozzle-to-pipe welds. Inspect and photograph the interior surface areas accessible through these openings.
- 14) Verify locations of recorded identification of all thermocouples prior to disconnect of the facility recorder.
- 15) Remove sodium relief rupture disk; inspect and photograph shell interior and tube bundle areas which may be seen through this opening.

Note: All openings were sealed as soon as inspections were completed.

2. Results

To help identify specific locations discussed in the following text, the direction opposite to the sodium outlet nozzle is designated 0 degrees. Other locations are measured clockwise from this point as viewed from above. The sodium and water nozzles are at 180 degrees.

a. Dye-Penetrant Examination of Region of Cracks on Exterior (item 2 of plan)

The examination was performed using dye-penetrant which met the total halogen limits of NavShip Specification 250-1500. The examination located an additional crack at 0 degrees just below the 1-in. sodium drain line. This location is within the region of the cracks first observed. A mark (or possibly a crack) was also found above the flange at 180 degrees (below the sodium inlet nozzle). Previously observed cracks were confirmed.

b. Helium Leak Check (item 3 of plan)

A shell-side-to-atmosphere leak test yielded negative results. A shell-side-to-tube-side test yielded negative results. A tube-side-to-atmosphere leak test was inconclusive because of the inability to achieve complete sealing of the lower manway against leakage.

c. Overall Visual Inspection (item 4 of plan)

Overall visual inspection revealed no additional evidence of damage.

d. Measurement of Clearance to Flange of Noncontacting Holddown Pads (item 5 of plan)

It was visually observed that three of the four holddown pads provided to retain the flange against the support ring did not touch the flange. The clearances were measured with the following result:

<u>Pad Location</u>	<u>Angle Coordinate (deg)</u>	<u>Clearance (in.)</u>
SE	45	0.033
SW	135	0.050
NE	315	0.050

e. Measurement of Tube-Sheet Flange-Support Ring Interface (item 6 of plan)

The steam generator flange was supported by a ring composed of fitted segments of Johns-Mansville "Marinite," a hard insulating material. This ring was in turn supported by the steel ring of the facility support structure. Several regions were observed at which the flange did not contact the Marinite surface. Detailed measurement defined the condition as follows:

- 1) The flange contacted the support ring over a segment between 296 and 347 degrees and over a second segment between 119 and 182 degrees.
- 2) Maximum clearance of the 182- to 296-degree gap was 0.157 in. at 220 degrees.
- 3) Maximum clearance of the 347- to 119 -degree gap was 0.110 in. at 56 degrees.

No conclusion has been made whether this condition represents irregularity of the flange or the support ring. The support ring structure separates into two sections and the lack of contact indicated above occurs about the line of separation. It was necessary to separate the support structure to remove the steam generator.

f. Removal of Holddown Screw Assemblies (item 7 of plan)

In removing the clamp of the one contacting holddown pad at 225 degrees, it was observed that the pad was released after a bolt turn of under 1/64th of a revolution.

g. Sampling of Metal Surfaces for Chlorine and Other Contaminants
(item 8 of plan)

A green stain on the shell exterior below the sodium relief nozzle was found to be slightly caustic. The samples showed chlorine concentrations of between 1 and 10 ppm. The highest concentration was at the stained area mentioned above.

h. Dye-Penetrant Inspection of Cleaned Areas (item 10 of plan)

No defects were found beyond the original cracks and the defect below the sodium outlet nozzle reported in 2.a.

i. Chemical Examination of Insulation and Deposits (item 11 of plan)

Various insulation samples and surface deposits from feedwater header cracks were subjected to chemical analysis. Insulation samples were found to contain 18 to 367 ppm of chlorides. Detailed chemical analysis is shown in Appendix C.

j. Cut, Inspect, and Seal Lines (item 13 of plan)

Figure 6-22 shows the line-cut locations. Inspection of internal areas visible through the sodium nozzles after lines were cut revealed that the upper and lower shroud exterior surfaces were clean. A glove box mirror arrangement was installed at the bottom nozzle to permit viewing toward the shell bottom. It was found to be filled with solidified sodium to a level of 14 in. below the nozzle. The sodium surface was slightly rippled.

k. Verification of Thermocouple Identifications (item 14 of plan)

Thermocouple locations were found to be correctly identified except for thermocouples 75 and 76 which were adjacently located on the lower tube sheet (see Figure 4-13). Test records which include output of these thermocouples have been corrected accordingly.

l. Inspection of Areas Visible Through Sodium Relief Nozzle (item 15 of plan)

The sine wave section of all tubes, shell inside surface, nozzle inside surface, and the burst diaphragm itself were covered with a heavy layer of a gray-white material (see Figure 6-23). A sample of this material taken from the rupture disk was found to be soft and pliable, as is characteristic of metallic sodium. Chemical analysis of a sample showed it to be mainly metallic sodium with small quantities of sodium hydroxide, sodium hydride, and other sodium compounds.

The material and the manner in which it was built up resemble the "sodium frost" which has been observed in gas spaces over sodium reactors which were subject to air leakage. Sodium frost forms in the following manner: sodium vapor diffuses from the liquid sodium surface and condenses on surfaces of the gas space. Then the condensed sodium partly oxidizes, forming a porous structure which adheres to the surfaces. This structure serves to retain further

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Figure 6-23. View of Steam Generator Tube Bundle after 1970 Shutdown

condensing sodium which in turn partly oxidizes, forming a further retention structure. A thick layer can accumulate by continuation of this process. Personnel of AI who have investigated sodium frost in reactors confirmed the similarity.

The sodium relief nozzle was also opened for diaphragm replacement during the July-to-December 1969 down period which followed the first tube leak. It was observed at that time that the steam tubes were clean except for a few pieces of white material adhering to tubes at the level of the sodium surface (Figure 6-24). As mentioned in Section IV.A, there was probably 1 hr of operation with steam leakage before the 1969 shutdown. The comparison suggests that leakage took place for a considerable time before the 1970 shutdown.

m. Radiography of Tube Sheet and Header Crack Areas (item added to plan)

A successful radiograph of the crack was obtained with the cobalt radiation source positioned in the center of the feedwater header. A pattern of hairline cracks was shown, which was in general agreement with the pattern visible on the surface but possibly includes additional cracks. Full details of this radiograph can only be seen when it is placed in an illuminator. It has not been included in this report. The radiography confirmed the presence of the sodium slug in the bottom of this shell.

n. Measurement of Steam Generator Deflection During Line Cuts
(item added to plan)

Deflection measurements were made to determine whether the steam generator was deflected by residual forces in steam or sodium lines. Prior to starting the line cutting, two displacement transducers were installed at a right angle to one another to measure the distance between the shell and fixed points in the facility structure. They were positioned horizontally, contacting the steam generator at the radiation transducer housing, which is about 5 ft below the top of the shell. In addition, a plumb bob was attached to the bottom of the water header and referenced to a fixed point on the facility floor. No deflection was indicated during line cuts by either of the above devices.

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Figure 6-24. View of Steam Generator Tube Bundle
During 1969 Down Period

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C. FAILURE THEORIES AND ANALYSES

As this report was prepared prior to accomplishing the destructive examination, positive identification of modes of failure is not possible. However, preliminary failure theories which may serve as a guide in planning the examination are discussed in this section.

1. Manway Defects

While further examination appears desirable, a highly probable underlying cause of the manway failure is apparent. Bolt loading appeared to press the sealing ring into the groove. As shown in Figure 6-25, the ring pressed against the side of the groove. The resulting wedging action accounts for the lip deformation. The stress imposed on the bottom of the groove by this deformation, further concentrated by microfissures, produced the cracks in the upper manway groove.

A finite element plastic stress-strain analysis indicated that the 0.056-in. measured lip deformation could be explained by an 80,000-psi bearing pressure at the ring and groove engagement. The maximum recorded bolt torque applied

Figure 6-25. Pressure Force in Top
Manway Flange Seal Ring-Groove
Engagement

to the flanges was 7450 ft-lb. The resulting bearing pressure would depend on the bolt friction coefficient. A bearing pressure of 82,000 psi corresponds to a friction coefficient of 0.32, which is well within the range of possible friction coefficients for this bolting combination. Thus the analysis gave a fair agreement between measured and theoretical lip deformation.

The same analysis predicted 2.6% effective strain at the crack location on the bottom of the seal groove. Yield of stainless steel is considered to start at 0.2% effective strain. The results thus predicted considerable yielding at the crack location. However, the failure did not occur at the area of maximum predicted strain. Computed strain at the corners of the seal ring groove is approximately 8.2%. No explanation of this effect other than possible lack of microfissures at the corners has been advanced.

The manway inside diameter was 15 in. Standard ring seal designs for this dimension (per ANSI 51D-B165, Taylor Forge Catalog, etc.) call for a 23-in.-diameter ring and groove vs the 16 in. adopted by Alco. The standard distance from groove to inside diameter is approximately 3-1/2 in., as opposed to 1/2 in. used in this design. Thus this failure appears to be related to a design deviation.

A complete explanation for this apparent design deviation could not be obtained from available records. However, it apparently was the result of scaling the steam generator design down to 30 Mwt from an earlier 70-Mwt design. In scaling down it was necessary to retain a functionally adequate manway inside diameter. In the 70-Mwt design, with a much larger header, the manway may have been a minor anomaly in an essentially hemispherical header. In the 30-Mwt scale-down, if proper design had been followed (including a larger bolt circle), the manway flange would have been so large as to preclude any resemblance to a hemispherical structure. The reduced seal-groove diameter was thus a compromise which permitted some resemblance between the prototype and the large-scale design.

2. Tube Leaks

The two tubes with leaks and the three with nonpenetrating cracks, detected in 1970, were located in a compact cluster which also includes the one leak detected in 1969. The proximity of these leaks suggests that a localized

environmental condition (such as concentration of a media corrosive to stainless steel) existed. Two potential sources include the initial manufacturing/pre-installation phase, and the operational phase. The 1969 leak incident may have initiated a secondary corrosion process due to accumulated sodium-water reaction products. The contributing role of preoperational factors could relate to results recently developed after examination of an HNPF IHX tube-to-tube sheet, where cracks in both tube and tube sheet were discovered.

A second possible cause of failure is thermal fatigue. This possibility was investigated in connection with the 1969 leak. It was suggested that the unequal expansion moduli of Inconel and stainless steel could produce stress under temperature changes. An estimate was made of the total number of temperature cycles to which the tubes had been subjected. This estimate was based on the recorded starts, shutdowns, and steam outlet temperature changes. The resulting cumulative stress cycle was estimated to be less than 1% of the allowable fatigue life. These results are reported in Reference 10.

However, a much more severe thermal fatigue accumulation might be possible if dynamic instability had taken place during operation. As discussed in Section V.E.3, it is considered possible that undetected dynamic instability occurred and this phenomenon may explain the anomalous heat-transfer performance. Dynamic instability would result in periodic oscillation of the steam outlet temperature. It could also result in an oscillating thermal gradient across the upper tube-sheet ligaments separating adjacent tubes with different steam temperature. Growth and contraction of ligaments with temperature change could cause an alternating bending moment on the tube span between the weld and the guide ring, the section in which all tube defects occurred. This mode of loading would also put a stress on the tube-sheet weld. This hypothesis might account for the coincident tube leaks and weld cracks in three of the tubes. Where starts, shutdowns, and operational changes could account for only a few dozen thermal cycles in the limited test life of the unit, dynamic instability could produce hundreds of thousands of cycles in the same period. The possibility of exceeding cycle life to failure would be greatly increased.

If this hypothesis is correct, evidence of thermal fatigue may be present in upper tube-sheet ligaments as well as the tubes.

3. Lower Tube-Sheet Cracks

Existing evidence points most strongly to stress corrosion as the mode of failure. The corrosive agent might be either water-side chloride or sodium-side caustic. In either case it is necessary to seek a mechanism which could initiate the process. The lines of investigation which have been suggested include the following.

a. Accumulated Corrosion Products on Lower Tube Sheet

Caustic stress corrosion cracking of the Type 316 SS tube sheet is a strong possibility. For this mechanism to occur would require the introduction or formation of a caustic solution in the shell side of the steam generator. During the 1969 leak incident (failed tube), a significant amount* of sodium-water reaction products were produced. After final 1970 drainage, it was observed that a deposit of sodium remained on the lower tube sheet. This demonstrates that incomplete drainage of the unit could have occurred. Thus, caustic generated during the 1969 leak may not have been completely drained following the initial leak. Instead, this residual caustic could have collected and remained on the lower tube sheet and in the 1-in. drain line.

b. Coincidence of Pitting and Cracks

While no leak path was established, it appears possible that the pits observed in the water header may have penetrated the Inconel overlay and thus provided sites for the initiation of chloride stress-corrosion cracking which propagated to the outside of the header.

Since deposits of sodium oxide were found on the external tube-sheet surface (Section VI.A.7), cracking originating on the water side would, of necessity, have to communicate with the shell side (sodium side) in a manner that would permit a sodium release to the outside of the unit.

c. Tube-Sheet Drain Passage

This passage is in proximity to the crack region. For a corrosion path starting on the water side, this could furnish a short path to contact with sodium. Once contact was established, sodium-water reaction products could form and a process of caustic corrosion could start.

*Estimated to be approximately 60 lb of sodium reacted with water (see Section IV.A.)

d. Asymmetric Temperature in Tube Sheet

This condition was described in Section IV.E.1. The high-temperature position in the asymmetry pattern coincided with the location of the large tube-sheet crack. This suggests some relationship between the cracks and thermal effects. No specific mechanism connecting the two has been formulated.

e. Thermal Shield Drain Slot

As also noted in Section IV.E.1., the slot which drained the shell thermal shield annulus was close to the high-temperature location in the asymmetry pattern, as well as to the tube-sheet crack region. The possibility of a mutual relationship among the slot, cracks, and temperature distribution pattern is suggested.

VII. STRUCTURAL STUDIES

A. LOWER TUBE-SHEET TERMINAL TEMPERATURE DIFFERENCE LIMIT

Alco specified a maximum terminal temperature difference (ΔT) between fluids of 175°F for the steam generator lower tube sheet, based on theoretical analyses. The structural design of the unit was in accordance with the ASME Unfired Pressure Vessel Code, Section VIII, and TEMA Class R Standards, where applicable. Thermal stresses were calculated in accordance with the requirements of the Navy Code (Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components).

The UNC review of the Alco design calculations (Reference 3) indicated that an increase beyond 175°F was possible. However UNC did not consider it their responsibility to make a recommendation to this effect.

To resolve this difference, LMEC began an independent analysis to establish the allowable ΔT . This analysis utilized Section III, Code for Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code. The results showed that the allowable ΔT for this tube sheet is 210°F. This allowable was based on the stress intensity at the barrel-to-tube sheet junction inner surface. Utilizing a stress concentration factor (K_t) of 2.0 for the effect of the probable intergranular fissures in the dilution zone between the Inconel weld overlay and the Type 316 base material, resulted in an allowable life at this ΔT limit of 2.6×10^4 cycles.

Since an analysis of the sort utilized by Alco and UNC would probably result in a repetition of their results, and the state-of-the-art of analysis techniques has advanced since these analyses were performed, it was concluded that a finite-element analysis technique would result in a better simulation of the actual stresses. A linear finite-element analysis was used to determine the displacements, strains, and stresses on the tube sheet and shell junction area. The computer code utilized for this analysis is the "Finite Element Stress Analysis of Two Dimensional Structures with Orthotropic Temperature-Dependent Material Properties."⁽¹¹⁾ Implicit in the use of this analysis are the restrictions on any analysis using the equations of linear elasticity and the displacement finite element methods; these are:

- 1) The material is assumed to be linear elastic.
- 2) The deflections are sufficiently small that changes in geometry do not introduce nonlinear behavior. (Superposition of stresses is applicable.)
- 3) The structure and loading is assumed to be axisymmetric, i. e., all variables are independent of the circumferential angle.
- 4) The displacement function within each triangular subelement is assumed to be linear (i. e., constant strain elements).

To obtain the tube-sheet and shell wall temperature distributions, a finite-element thermal model of the tube sheet was developed and utilized with a thermal analyzer computer code to define the temperature distributions at various steady-state terminal ΔT 's.

The details of the analysis are presented in the following text.

1. Thermal Analysis

A finite-element thermal model of the lower tube sheet (Figure 7-1) was developed and used in connection with a thermal analyzer code (DEAP) to define the temperature distribution in the tube sheet for the various overall tube-sheet ΔT 's. This model used the standard relaxation method for solving the steady-state heat transfer of a finite-element grid.

The model consisted of 980 elements. This included a representation of the 10 thermal shield plates which separate the tube sheet from the flowing sodium, as shown in Figure 7-1. The tube-bundle region of the tube sheet was treated as a homogeneous material with bulk heat transfer to or from a source within each node to represent heat transfer to the fluids within the tubes. The conductivity of the internal heat source was based on the calculated radial conduction/convection path for a single tube and the square region of metal surrounding it.

The 100% rated power/flow conditions were used for all cases as these yield the highest convective heat transfer coefficients and thus the highest temperature gradients within the stressed sections. The heat transfer coefficients used at fluid boundaries were tabulated with the assumed heat transfer mechanism and source references for the correlations used in Table 7-1. Figure 7-1 shows these coefficients as well as directions of fluid flow. The surface

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Figure 7-1. Lower Tube-Sheet Geometry and Flow Configuration

TABLE 7-1
LOWER TUBE-SHEET CONVECTIVE HEAT TRANSFER
COEFFICIENTS FOR THERMAL MODEL

Location	Mechanism	h (Btu/hr-ft ² - °F)	Reference
Horizontal surface of tube sheet in barrel	Axisymmetric flow, forced convection	154	12 and 13
Vertical surface of tube sheet in barrel	Turbulent free convection along a vertical plate	107	14
Inside tubes	Turbulent forced convection in sodium (heating)	2600	15
Top surface of shield	Turbulent free convection for a horizontal plate (cooled)	675	16
Vertical surface of liner	Turbulent free convection for vertical plate	875	16

exposed to fluids were divided into a number of regions with each region having a uniform boundary temperature. These boundary temperatures are defined in Table 7-2. Note that the exposed shell-side surface of the tube sheet is divided into three regions: inboard of the tubes, encompassing the tubes, and outboard of the tubes.

The results of this analysis for the temperature distribution in the tube sheet are shown in Figures 7-2 to 7-5 for three different values of overall tube-sheet ΔT between sodium and water. These temperature distributions were incorporated in the structural analysis which is discussed next.

2. Structural Analysis

For the structural model, the tube sheet, dome, and shell were divided into a finite-element mesh. The dome section of the tube sheet was treated as a constant thickness section, since it was felt that the reinforcing of the nozzle

TABLE 7-2

LOWER TUBE-SHEET BOUNDARY TEMPERATURES
FOR THERMAL MODEL

Locations	Temperatures (°F)		
	Case I 150 ΔT	Case II 250 ΔT	Case III 350 ΔT
1) Water temperature in barrel	600	600	600
2) Water temperature in tubes in tube sheet	600	600	600
3) Water temperature in tubes in shield	601.5	603.5	605
4) Sodium temperature on top of shield inboard of tubes	764	873	980.5
5) Sodium temperature on top of shield in tube bank	757	861.5	965.3
6) Sodium temperature on top of shield outboard of tubes	750	850	950

Note: The location numbers of this table correspond to those of Figure 7-1.

openings was removed a sufficient distance from the dome-to-tube sheet junction so that its effect would be negligible. The shell section was extended a sufficient distance from the junction so that any end conditions applied to the shell end would not affect the stresses at the junction. Refinement of the mesh size was performed in areas of expected discontinuities and high displacement and strain gradients.

Article I-9 of Section III of the ASME Boiler and Pressure Vessel Code was utilized to establish the effective radii for the ligament area of the tube sheet. The equivalent elastic modulus and equivalent Poisson's ratio of the ligament were also determined from the rules of this article. This usage presupposes that the hole pattern of the tube sheet is applicable to the rules of this article. Conditions that are not satisfied are:

- 1) The holes be arranged in an array of equilateral triangles.
- 2) The hole pattern extend to the plate center.

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Figure 7-2. Temperature Distribution in Lower Tube Sheet —
Case I ($\Delta T = 150^\circ \text{F}$)

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Figure 7-3. Temperature Distribution in Lower Tube Sheet –
Case II ($\Delta T = 250^{\circ}\text{F}$)

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Figure 7-4. Temperature Distribution in Lower Tube Sheet —
Case III ($\Delta T = 350^{\circ}\text{F}$)

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Figure 7-5. Temperature Distribution in Lower Tube Sheet —
Case III (Detail)

No credit for the stiffening effect of the tubes in the perforations was taken for this analysis, and the stiffening effect of the tubes on the tube sheet was not considered.

The models were analyzed for the effects of pressure and thermal loading cases separately to determine their individual effects. These linear elastic cases were then combined by superposition, and the rules of ASME Section III as amended by Code Case 1331-4, where applicable, to determine thermal gradient limits under the applied pressure.

The ligament stress intensities were evaluated by the rules of Article I-9 of Section III of the ASME Code, as modified by Code Case 1331-4 where applicable.

Figure 7-6 presents the tube-sheet stress classifications as required for an analysis per Section III of the ASME Code; symbols shown in the figure are defined in that work. Shown here are the analytical requirements for each of the areas of the tube-sheet structure. Shown also are the allowables for each of the stress intensity classifications. A major area of concern was establishment of the F stress intensity requirement. As the F stresses are compared to the material fatigue allowables, the design life must be established to obtain the number of cycles and hence allowable stress intensity. The intent of this study was to establish the allowable steady-state ΔT utilizing the conservative approach of basing the fatigue on the endurance-allowable stress intensity. The cumulative usage factor,

$$U_c = \frac{n_1}{N_1} + \frac{n_2}{N_2} + \dots + \frac{n_n}{N_n} ,$$

where n_n is the number of repeated stress cycles and N_n is the allowable number of cycles at stress level corresponding to n_n , must be equal to or less than unity to preclude a fatigue failure. By utilizing the endurance limit for the steady-state ΔT allowable stress, one gets

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Figure 7-6. Tube-Sheet Stress Classification

$$\frac{n}{N} \approx \frac{n}{\infty} \rightarrow 0$$

This results essentially in an allowable steady-state ΔT independent of any additional cumulative damage requirements.

The load cases evaluated for the tube sheet were:

- 1) Tube-side maximum design pressure: 2500 psi
- 2) Shell-side maximum design pressure: 150 psi
- 3) Three tube-sheet thermal gradient cases: there were 150, 250, and 350°F differentials.

These analysis parameters are shown in Figure 7-7. For this study, the steam temperature was held constant at 600°F. The secondary sodium bulk temperature was 600°F plus the ΔT .

The finite-element model of this tube sheet is shown in Figure 7-8. The isotherm plots for the three temperature gradient cases are shown in Figures 7-2 to 7-4, with Figure 7-5 for the 350°F ΔT case at a reduced thermal scale to obtain greater resolution. The maximum shear stress isostress plot for this temperature condition is shown in Figure 7-9 to illustrate the stress distributions.

This tube sheet has an Inconel overlay on the tube-side surface. As the coefficient of expansion for this overlay material is less than the parent Type 316 SS material, at the operating temperature of 600°F the overlay would force a reduction in the tensile stresses in the parent material. From the rules of Section III of the ASME Code, for an overlay or cladding of less than 10% of the nominal component thickness, the cladding may be neglected completely. Since the cladding is less than 10% of the tube sheet thickness, and this would result in the worst stress condition, the cladding thickness was not considered in the structural analysis. The thermal analysis did consider this overlay.

An evaluation was made of the stresses obtained from the finite-element analysis at the various discontinuity areas and the ligament area to obtain the plots of combined stress intensity vs bulk ΔT shown on Figures 7-10 and 7-11.

Figure 7-7. Lower Tube-Sheet Stress Analysis Parameters

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Figure 7-8. Finite Element Stress Model of Lower Tube Sheet

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Figure 7-9. Stress Distribution in Lower Tube Sheet for $\Delta T = 350^\circ F$

Figure 7-10. Lower Tube-Sheet Ligament Stress
(No Overlay)

Figure 7-11. Lower Tube-Sheet Junction Stress (No Overlay)

A plot of the stress distribution through the wall thickness of the barrel-to-tube sheet junction indicated a very nonlinear stress gradient. As a very fine nodal mesh had been utilized in this analysis, essentially the total structural discontinuity stress had been determined with this analysis. Therefore the stresses at the barrel-to-tube sheet junction were treated as the Q stress intensities with no additional geometry factor to obtain the peak stress intensities.

The allowable Q stress intensity at 600°F for Type 316 SS is $3 S_m$ or 48,000 psi. The maximum stress intensity for this tube sheet did occur at the barrel-to-tube sheet junction. This resulted in an allowable ΔT of 210°F, as shown in Figure 7-11.

In a previous evaluation of a failure in the upper steam generator tube-sheet area, the existence of intergranular fissures in the dilution zone between the Inconel overlay and base Type 316 SS had been discovered. Therefore a stress concentration factor of 2.0 was utilized with the parent material stress

Figure 7-12. Lower Tube-Sheet Tube-Side Inner Junction Discontinuity Stress for Cracked Overlay

intensities to determine a peak stress intensity in this area. This resulted in a reduced operating life of 2.6×10^4 cycles, as shown in Figure 7-12.

Figure 7-13 summarizes the stress intensities for the $210^{\circ}\text{F } \Delta T$ condition at the discontinuity and ligament areas.

B. PIPING LOADS

The loads imposed on the steam generator nozzle by the SCTI piping were analyzed by Bechtel Corporation and AI as part of the SCTI construction tasks. Results of those analyses were not published.

LMEC conducted a review of these analyses to provide assurance that the steam generator was safe in this respect, and to document the results.⁽¹⁷⁾

The original Alco structural analyses⁽¹⁾ provided allowable nozzle loads of 40,000 ft-lb moment and 4000-lb thrust.

BLH also performed an external design load test on the steam generator in conjunction with its hydrostatic test. These tests are reported in Reference 18.

*SHELL SIDE PRESSURE = 150 psi
 †LIFE = 2.6×10^4 CYCLES

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Figure 7-13. Lower Tube-Sheet Stress Summary for $\Delta T = 210^\circ F$
 Under Maximized Pressure Conditions

Comparison of the calculated piping load with the steam generator design and test data indicates that the piping loads would not stress the steam generator to a high level. Figure 7-14 shows the calculated forces and moments for the two water/steam nozzles and two operating sodium nozzles of the steam generator.

C. TUBE-RUPTURE SAFETY STUDIES

A series of studies were conducted to provide a preliminary indication of the safety of the steam generator following a tube rupture. Three separate hazard aspects were considered as follows:

- 1) The transient pressure imposed on the shell by a sudden guillotine rupture of a single tube.
- 2) The pseudo steady-state pressure imposed on the shell during expulsion of sodium through the two burst disks and sodium relief pipes.
- 3) The short-term burst strength of the shell.

The transient pressure study utilized data from a tube rupture test conducted by AI in a simulated sodium steam generator under the ESADA program.⁽¹⁹⁾ The burst strength study considered material properties in the region of high plastic strain, including strain hardening effects. Results are summarized in the following paragraphs.

The static burst pressure of the shell was calculated to be 1200 psi for uniform internal pressure. For the internal pressure applied as a non-decaying step pulse, the burst pressure magnitude is 1130 psi. The burst pressure for a linearly decaying step pulse is shown in Figure 7-15 as a function of pulse duration. These results were obtained using the analysis of C. J. Costantino, Reference 20. Additional protection (if any) provided by the shell linear and stagnant sodium layer was not considered.

The initial pressure effect following a sudden tube rupture would be a shock wave originating at the point of rupture. The shock pressure might exceed the tube steam pressure at the point of rupture but would decrease rapidly with distance from this point. The highest pressure at the shell would result from rupture of a tube in the outer row of tubes, which is 4 in. from the shell. This pressure is estimated as 2000 to 3000 psi at the point on the shell closest to the

LOCATION	F_X (lb)	F_Y (lb)	F_Z (lb)	M_X (ft-lb)	M_Y (ft-lb)	M_Z (ft-lb)
SECONDARY SODIUM INLET	-508	+1420	+553	-3340	-2110	+4560
SECONDARY SODIUM OUTLET	+338	-1297	-9	+6250	+1307	+3730
FEEDWATER* INLET	+8	+148	+41	-811	-20	-274
STEAM* OUTLET	-1488	+80	+35	+640	+6854	-3954

* LOADS GIVEN IN PLANE OF NOZZLE TO PIPE INTERFACE,
X AXIS COINCIDENT WITH NOZZLE CENTERLINE

1. RIGHT HAND RULE USED FOR SIGN CONVENTION
2. LOADS ARE COMPATIBLE FOR STEADY STATE THERMAL
CONDITION

Figure 7-14. Piping Loads on Steam Generator Inlet and Outlet Nozzles

Figure 7-15. Steam Generator Shell Burst Pressure vs Pulse Duration
(Interpolated from Reference 20 for a single linearly –
decaying impulse)

rupture, decreasing elsewhere as a function of distance from the rupture. Its duration is under 1 msec. The initial effect would be followed by a pressure, created by expansion of the steam bubble, opposed by the inertia and flow resistance of the displaced sodium. For a burst near the water inlet end of a tube (the worst case) the pressure would be under 300 psi.

The pseudo steady-state pressure would next be reached. This is estimated to have a maximum value of 152 psi, taking account of possible liquid, gas, or two-phase flow in the relief system. By reference to the time-dependent burst strength relationship of Figure 7-15, applicable to durations under 2 msec, it appears that the shell may be only marginally safe against the initial shock pressure. It is safe by a large margin against the lower pressure effects which follow.

This study could have been carried to greater mathematical exactitude by more extensive calculations. However, such an effort was not considered justified because of the limited accuracy of available data on the shock pressure accompanying sodium-water reactions.

D. STRAIN GAGE INSTALLATION PLANS

1. History

The Alco test program included use of strain gages during transient tests. However, no suitable gages were available and these plans were deleted. (Strain gages were employed during the hydrostatic test at the BLH plant, as reported in Reference 18.) At the recommendation of UNC, plans for installation of strain gages were reinstated together with plans for strain gage installation on the Alco IHX.

Because of the high temperatures imposed on the steam generator and IHX structures it was necessary to use special high-temperature strain gages. The Type 420 and 425 gages made by Microdot, Inc., were selected as the most suitable. Both are weld-attached gages intended for service up to 900°F.

Evaluation of these gages, and development of calibration and installation methods for them was conducted under a separate LMEC project. Results are reported in part in References 21 and 22 and further documented internally at LMEC.

A second type of high-temperature gage considered for use in this program was the Armour-D gage developed at LMEC.⁽²³⁾ This is a ceramic spray-bonded gage. Though considered to have limited potential with respect to accuracy of measurement, it was smaller than the Microdot gages and thus less susceptible to strain gradients. It was planned to place Armour-D gages within Microdot gage rosettes to provide a check on strain gradient effects.

2. Location Plans

A plan of specific strain gage locations was formulated by UNC. There were several revisions of this plan by LMEC. Figures 7-16 and 7-17 show the plan in its most extensive form. Table 7-3 describes the purpose of each gage rosette of the figures. In subsequent revisions, the quantity of gages was reduced on the basis of budgetary limits and further consideration of the value of information certain gages would provide. All of the Armour-D gages were deleted from the plan after a production lot proved unsatisfactory.

Figure 7-16. Proposed Strain Gage Locations on Lower Tube-Sheet Region

Figure 7-17. Proposed Strain Gage Locations on Upper Tube-Sheet Region

TABLE 7-3
DESCRIPTION OF STRAIN GAGE LOCATIONS ON STEAM GENERATOR
(Sheet 1 of 5)

Gage	Category	Type	Quantity	Location	Purpose
150	Primary	Microdot 420	2	Gage 150-1 is to be located on the 180° reference plane, 18 in. above the transition line. Gage 150-2 at the same location, to be positioned on longitudinal gage's filament center. Thermocouple to be located on the centerline of the circumferential gage, as close as possible to intersection of gages (~1/2 in.).	To correlate a relatively known calculated stress value with a test value. Located 180° apart in order to eliminate incorporating any unknown bending stresses. Located at this height so as to be isolated from discontinuity stresses imposed by tube sheet restraints or nozzles.
151	Primary	Microdot 420	2	Same as gage 150, except located on 0° reference plane.	
152	Primary	Microdot 420	4	Gages are to be located as follows: 152-1 at 0° (top of pipe) 152-2 at 90° 152-3 at 180° 152-4 at 270° Thermocouple to be located adjacent to gage 152-1.	To verify piping load imposed on nozzle by thermal growth of piping system.
153	Primary	Microdot 420 and Armour D	3 1	Gage 153-1 to be located on the 180° reference plane, 4 in. above toe of fillet. Other 2 gages at same location to be positioned on longitudinal gage's filament center. The Armour D gage is to be located adjacent to the longitudinal Microdot gage. Thermocouple to be located on centerline of circumferential gage as close as possible to intersection of gages (~1/2 in.).	To evaluate wall stresses as close to tube sheet as possible. Located 180° apart in order to eliminate incorporating any unknown bending stresses. Located 180° apart to determine the effect of transients on stresses in this structurally critical area. A test method by which to indirectly determine tube sheet stresses.

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TABLE 7-3

DESCRIPTION OF STRAIN GAGE LOCATIONS ON STEAM GENERATOR
(Sheet 2 of 5)

Gage	Category	Type	Quantity	Location	Purpose
154	Primary	Microdot 420 and Armour D	3 1	Same as gage 153, except located on 0° reference plane.	
155	Primary	Microdot 420 and Armour D	2 1	Gage 155-1 to be located on 180° reference plane, at toe of fillet. Other 2 gages at same location positioned on longitudinal gage's filament center. Armour D gage to be located adjacent to longitudinal Microdot gage. Thermocouple to be located on centerline of circumferential gage, as close as possible to intersection of gages (~1/2 in.).	Same as gages 153 and 154.
156	Primary	Microdot 420 and Armour D	3 1	Same as gage 155, except located on 0° reference plane.	
157	Primary	Microdot 420 and Armour D	3 1	Gage 157-1 to be located on 180° reference plane at toe of fillet. Other 2 gages at same location positioned on longitudinal gage's filament center. Armour D gage to be located adjacent to longitudinal Microdot gage. Thermocouple to be located on centerline of circumferential gage, as close as possible to intersection of gages (~1/2 in.).	Same as gages 153 and 154.
158	Primary	Microdot 420	3	Same as gage 157 except located on 0° reference plane.	

TABLE 7-3
DESCRIPTION OF STRAIN GAGE LOCATIONS ON STEAM GENERATOR
(Sheet 3 of 5)

Gage	Category	Type	Quantity	Location	Purpose
159	Primary	Microdot 420, and Armour D	3 1	Gage 159-1 to be located on 180° reference plane, 2 in. below the toe of fillet. Other 2 gages at same location positioned on longitudinal gage's filament center. Armour D gage to be located adjacent to longitudinal Microdot gage. Thermocouple to be located on centerline of circumferential gage, as close as possible to intersection of gages (~ 1/2 in.).	Same as gages 153 and 154.
350	Backup	Microdot 425	3	Gage 350-1 to be located on 180° reference plane. Height to be determined in the field, and is to be at the first location above the tube sheet clear of obstructions in the circumferential direction. Other 2 gages at same location positioned on longitudinal gage's filament center. Thermocouple to be located on centerline of the circumferential gage, as close as possible to intersection of gages (~ 1/2 in.).	Being installed to measure the possibility of unsymmetrical strain in the steam generator. This condition is caused by circumferential variation of thermal mixing which was noticed by UNC as a previous test. Possible causes are: a drain line was left open, the thermal shields may have an unsymmetrical thermal gradient, a pressure drop across the thermal shields causing flow through the annular gap between the shield and steam tubes.
351	Backup	Microdot 425	3	Same as gage 350 except located on the 135° reference plane.	
352	Backup	Microdot 425	3	Same as gage 350 except located on the 90° reference plane.	
353	Backup	Microdot 425	3	Same as gage 350 except located on the 45° reference plane.	

TABLE 7-3

DESCRIPTION OF STRAIN GAGE LOCATIONS ON STEAM GENERATOR
(Sheet 4 of 5)

Gage	Category	Type	Quantity	Location	Purpose
354	Backup	Microdot 425	3	Same as gage 350 except located on the 0° reference plane.	
355	Backup	Microdot 425	3	Same as gage 350 except located on the 315° reference plane.	
356	Backup	Microdot 425	3	Same as gage 350 except located on the 270° reference plane.	
357	Backup	Microdot 425	3	Same as gage 350 except located on the 225° reference plane.	
358	Backup	Microdot 425	3	Same as gage 153 except located on the 90° reference plane and no Armour D gage.	Same as gages 350 to 357.
160	Primary	Microdot 420 and Armour D	3 1	Gage 160-1 to be located on 180° reference plane at the vessel-to-nozzle juncture. Other 2 gages at same location positioned on longitudinal gage's filament center. Armour D gage to be located adjacent to longitudinal Microdot gage. Thermocouple to be located on centerline of circumferential gage, as close as possible to intersection of gages (~1/2 in.).	To evaluate secondary bending and stress concentration effects in the circumferential direction in the area of the nozzle.
161	Primary	Microdot 420 and Armour D	3 1	Same as gage 160 except located on the 0° reference plane, at edge of manway reinforcement weld.	To evaluate secondary bending and stress concentration effects in the circumferential direction in the area of the manway.

TABLE 7-3

DESCRIPTION OF STRAIN GAGE LOCATIONS ON STEAM GENERATOR
(Sheet 5 of 5)

Gage	Category	Type	Quantity	Location	Purpose
162	Primary	Microdot 420	4	Same as gage 152.	Same as gage 152.
163	Primary	Microdot 420 and Armour D	3 1	Same as gage 157 except located at the upper tube sheet.	To evaluate wall stresses as close to the upper tube sheet as possible. Only test method by which to indirectly determine tube sheet stresses.
164	Primary	Microdot 420	4	Same as gage 152.	Same as gage 152.
250	Secondary	Microdot 425	3	Gage 250-1 to be located on the 180° reference plane, 2 in. above toe of fillet. Other 2 gages at same location positioned on longitudinal gage's filament center. Thermocouple to be located on centerline of circumferential gage, as close as possible to intersection of gages (~ 1/2 in.).	Deemed necessary because of failure rate of Microdot gages and is located in most critical area.
251	Secondary	Microdot 425	3	Same as gage 250 except located on the 0° reference plane.	

Following discovery of the tube leaks in 1970, it was decided to defer strain gage installation except for two rosettes (Figure 7-18), which were installed at the following locations in May of 1970:

- 1) Rosette 1: 90 degrees from steam outlet nozzle, 9-1/2 in. below lower tube sheet.
- 2) Rosette 2: 90 degrees from steam outlet nozzle, 7-1/8 above lower tube sheet.

It was intended to use these rosettes to gain experience in field use of the strain gages and data reduction. The locations were selected to yield high strain levels which would minimize the effect of certain errors and permit clear interpretation of results.

3. Rosette Configuration Problems

In addition to selecting the location of each rosette, it was necessary to select a rosette configuration. Two problems had to be considered in making this selection:

- 1) In obtaining the combined strain of a rosette, the mathematical relationships employed could magnify the measurement error of any single gage in the rosette. This effect depends on the ratio of strains on mutually perpendicular axes. For large ratios this effect could cause a large inaccuracy. A study of this problem by UNC is presented in Reference 3. A delta rosette was found to minimize this source of error.
- 2) The Microdot gages are relatively large, with an active element length of 1-1/2 in. For certain rosette configurations, this results in a large separation between gage centers. Effects of strain gradients would therefore be amplified.

A final resolution of this problem was not achieved. Rosette No. 2, installed in May 1970, was a dual rosette (Figure 7-18) to obtain a comparison between the delta rosette and the rectangular rosette. Gages 1, 2, and 3 of this pattern constitute a delta rosette. The gage centers were located on a common circumferential line. On the basis of axisymmetric stress analysis, all three gages

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Figure 7-18. Dual Strain Gage Rosette

should thus be at points of equal stress. However, the centers of gages 2 and 3 were 2-5/8 in. apart, which would induce some error if there were nonaxisymmetric loadings. Gages 1, 4, and 5 constitute a rectangular rosette with all gage centers approximately 1-1/4 in. apart. This would reduce strain errors from nonaxisymmetric loadings, however it would amplify the effect of the longitudinal strain gradient in the areas of discontinuities. Rosette No. 1 included only the delta pattern (gages 1, 2, and 3).

4. Measurement Accuracy Problems

The principal source of error in the Microdot gages was variability of the apparent strain (or bias). The gages were first welded to a laboratory test bar and calibrated. They were then removed from the test bar and welded to the structure under test in the field. (In laboratory investigations of accuracy they were rewelded to a test bar.) The apparent strain, which varies nonlinearly with temperatures, was measured during the test-bar installation. It was found that reinstallation resulted in a highly variable drift of the original apparent strain. This drift variability would constitute an error in measurements made in the field, as its magnitude could only be estimated statistically. Figure 7-19 shows the predicted value of this error as a function of temperature. The maximum error of 45.0 microstrain approaches the absolute strain level predicted at some locations on the steam generator.

Therefore, studies were conducted of techniques which would reduce the reinstallation strain error by avoiding complete reliance on laboratory calibration. It was concluded that it might not be possible to estimate strain at a given operating condition with reference to its value in the nonheated and nonpressurized state before startup. However when any change is made in operating conditions it should be possible to measure the resulting change in strain. For a properly planned sequence of tests, such measurements would be largely free of the reinstallation error. The effect of the reinstallation error may be analyzed by assuming that the steam generator operates at two consecutive test conditions, A and B. If a particular gage location is at the same temperature at conditions A and B, there would be no change in apparent strain between the two conditions. The measured difference in strain would thus contain no apparent strain error. More generally, a change in test conditions from A to B

Figure 7-19. Apparent Strain Error Due to Reinstallation Bias

will cause a temperature change from T_A to T_B at a given location. The resulting error would depend on the variability of the parameter (apparent strain at T_A minus apparent strain at T_B). Evaluation of laboratory reinstallation data indicated that for temperature intervals of under 100°F the variability of this parameter tended to be small. Figure 7-20 shows the expected accuracy of the gages if used to measure changes in strain accompanied by temperature changes of $\pm 50^\circ\text{F}$ at the point of measurement. This figure includes the combined effect of all anticipated sources of gage error. (It is thus not directly comparable with Figure 7-19 which includes only apparent strain error.) Figure 7-20 shows total strain error of under 15 microstrain for any $\pm 50^\circ\text{F}$ increment between 200 and 750°F .

5. Test Planning Problems

Laboratory evaluation indicated that it was necessary to put the Microdot gages through three complete strain cycles between zero strain and the maximum value measured in order to stabilize their output. Prior to stabilization, the gage outputs showed considerable hysteresis. In terms of steam generator

Figure 7-20. Expected Strain Error vs Temperature

operation, this requirement meant three cycles of (1) starting the unit, (2) bringing it to maximum operating conditions, and (3) shutting it down. Only after a fourth start could an accurate measurement be obtained. This procedure was considered feasible but it was expected to require a considerable expenditure of test time and effort.

The Microdot gages were only intended for use to 900°F. At rated power conditions the sodium inlet nozzle, upper tube sheet, steam header, and a large portion of the shell would be at temperatures over 900°F. As shown in the location plan, most gages were located on the lower parts of the steam generator, which would remain below 900°F. It also appears possible to plan test conditions under which the entire steam generator is below 900°F. (This was accomplished in some tests of Series 1 as described in Section V.A.5.)

After strain gages are installed, it appears advisable to conduct such low temperature tests before proceeding to tests at rated conditions. Chances of gage failure were considered to increase at higher temperatures.

For gage locations in the vicinity of the lower tube sheet, large variation in strain was predicted as a function of tube-sheet ΔT . Investigation of structural

properties in this region should thus include systematic variation of tube-sheet ΔT . A method of accomplishing this by varying sodium flow at constant steam conditions was demonstrated in Series 1 and 4.

In summary, it was found that a test program of specially formulated test conditions would be necessary to realize the maximum accuracy of the strain gages. For greatest value, such a program should provide systematic variation of the principal stress-inducing factors, which include steam pressure, sodium pressure and lower tube-sheet ΔT .

NOMENCLATURE

A	heat-transfer area
c_p	specific heat at constant pressure
D_o	tube outer diameter
D_i	tube inner diameter
F	flowrate; and peak stress in Section VII
G	water/steam flowrate per unit area (lb/hr-ft^2)
g	acceleration of gravity
H	enthalpy
H_f	saturated liquid enthalpy
H_{fg}	latent heat
h	convective heat transfer coefficient
K_t	theoretical stress concentration factor
l	heated length from tube entrance to CHF point
Nu	Nusselt number
OD	outside diameter of tube
P	pressure
PAMTD	property adjusted mean temperature difference
P_w	steam generator thermal power (Btu/hr or Mwt)
Pe	Peclet number
Pr	Prandtl number
$\frac{q}{A}$	heat flux (Btu/hr-ft^2)
Re	Reynolds number
Re_v	Reynolds number for vapor fraction
S_R	two-phase flow velocity slip ratio
T	temperature

T_r	reduced temperature
T_w	wall temperature
U	overall heat transfer coefficient (Btu/hr-ft ² - °F)
U_c	cumulative usage factor
u	flow velocity
W	weight flowrate (lb/hr)
w	water/steam flow per tube (lb/hr)
X	steam quality in two phase flow region
Z	flow length

Greek Symbols

α	steam void fraction
Δ	differential segment
ΔP	pressure drop
ΔT	temperature difference
$(\Delta T)_B$	temperature difference between sodium and water at beginning of bulk boiling
ΔZ	increment of tube length
$(\Delta T)_{Na}$	sodium inlet-to-outlet temperature difference
ρ	density
σ	stress
Σ	sum of a series
τ_w	wall shear

Subscripts

CHF	critical heat flux
f	saturated liquid state
FW	feedwater

g saturated vapor state
H₂O water/steam
I inlet
i index
L water (liquid state)
LP Liedenfrost point
Na sodium
O outlet
s outlet steam
V steam (vapor state)
w wall
Z local heat transfer zone

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APPENDIX A

STATIC STABILITY MODEL

A static flow stability model was developed to determine whether the measured test data with poor performance could be explained by bistable flow operation; i. e., the existence of two groups of tubes having different flowrates and outlet temperatures, but the same pressure drop. Such a condition would result from static instability. For this purpose, the model simulated two families of steam tubes having independent steam-side conditions (except inlet temperature) and exchanging heat with a single, common sodium stream. This is assumed to be a close approximation of the sodium side of the steam generator in which mixing at each baffle pass would tend to eliminate any sodium temperature asymmetry. The model is described in the mathematical form in the following.

The entire water/steam flow was divided into six regions:

- 1) Single-phase subcooled liquid region
- 2) Subcooled boiling region
- 3) Bulk boiling region with liquid-wetted wall (nucleate boiling)
- 4) Transition boiling region from wetted to dry wall conditions
- 5) Bulk boiling region with vapor-covered dry wall (film boiling)
- 6) Single-phase superheated vapor region.

In each of these regions, the basic equations of continuity, momentum, energy and state, proper relations for the wall-shear, heat transfer, velocity slip, and flow quality were written. Boundaries between various regions were properly designated. These equations are shown below in the two-phase boiling regions. However, the equations for liquid and vapor phase regions can be obtained by putting $\alpha = 0$ and $\alpha = 1$, respectively, into appropriate equations.

1. Continuity

$$\frac{d}{dz} \left[\rho_L (1 - \alpha) u_L + \rho_V \alpha u_V \right] = 0 ,$$

where

ρ = density

u = flow velocity

α = void fraction

Z = flow length measured from the tube entrance

subscripts

V = vapor phase

L = liquid phase

2. Momentum

$$\frac{d}{dZ} \left[\rho_L (1 - \alpha) u_L^2 + \rho_V \alpha u_V^2 \right] = -g \left[\rho_L (1 - \alpha) + \rho_V \alpha \right] - \frac{dP}{dZ} - \frac{4\tau_w}{D_i} ,$$

where

P = static pressure

τ_w = wall shear

D_i = tube inner diameter.

3. Energy

$$\frac{d}{dZ} \left[\rho_L (1 - \alpha) u_L H_L + \rho_V \alpha u_V H_V \right] = \frac{4(q/A)_w}{D_i} ,$$

where H = enthalpy and $(q/A)_w$ = wall heat flux and the viscous heat dissipation was neglected. The wall heat flux was determined by the sodium-side resistance, wall resistance, and water-side resistance including the fouling factor, and thus is a function of the location.

4. State

Both liquid and vapor states were assumed to be at saturation condition when both coexisted (except in subcooled boiling zone). In this case, the thermodynamic variables such as density, temperature, and enthalpy are a function of the local pressure only. For the states of subcooled liquid and superheated

vapor, the thermodynamic variables are a function of local pressure and local temperature. The 1967 ASME Steam Tables were used.

5. Flow Quality, X

$$X = \frac{\alpha S_R}{\alpha S_R + (1 - \alpha) \frac{\rho_L}{\rho_V}}$$

6. Velocity Slip, S_R

The modified Bankoff model was used.

$$S_R = \frac{u_V}{u_L} = \frac{1 - \alpha}{(K_S - \alpha) + (1 - K_S)\alpha^r}$$

where

$$K_S = 0.71 + \frac{0.29}{0.32062} \left(\frac{P}{10^4} \right) \quad (P \text{ in psia})$$

$$r = 3.33 + 0.18 \left(\frac{P}{10^3} \right) + 0.46 \left(\frac{P}{10^3} \right)^2$$

7. Wall Friction

In the single-phase flow region, the Moody chart was used, while in the two-phase flow region, the modified Martinelli-Nelson correlation was used: ⁽²⁴⁾

$$\tau_w = \phi_{M-N}^2 \Omega f_f \left(\frac{1}{2} \cdot \rho_f \cdot u_f^2 \right)$$

where

f_f = liquid phase friction coefficient = f (Reynolds Number, surface roughness)

subscript

f = saturated liquid state

$$\phi_{M-N}^2 = \text{Martinelli-Nelson two-phase flow friction factor} \\ = f(X, P)$$

Ω = modification factor

$$= 1.36 + 0.5 \left(\frac{P}{10^3} \right) + 0.1 \left(\frac{G}{10^6} \right) - 0.714 \left(\frac{P}{10^3} \right) \left(\frac{G}{10^6} \right) \text{ if } G \leq 0.7 \times 10^6$$

$$= 1.26 - 0.4 \left(\frac{P}{10^3} \right) + 0.119 \left(\frac{10^6}{G} \right) + 0.28 \left(\frac{P}{10^3} \right) \left(\frac{10^6}{G} \right) \text{ if } G > 0.7 \times 10^6$$

G = mass flow per unit cross-section area in lb/hr-ft², and P in psia.

8. Heat-Transfer Relation — Sodium Side

The heat-transfer relation between the outer tube wall and sodium flow was obtained by analyzing the test data of the Alco IHX whose shell-side flow geometry is similar to that of the Alco/BLH steam generator. The following equation was established by curve fit:

$$Nu = 0.411 \left(\frac{\phi_1}{D_o} \right)^{1/2} \left(1 - \frac{D_o}{P_o} \right)^{1/2} (Pe)_{\max}^{1/2} \quad (\text{cross sodium flow})$$

or

$$Nu = 0.458 (Pe)_{\max}^{1/2} \text{ for the tube geometry of the steam generator,}$$

where

Nu = Nusselt number based on the tube outside diameter

ϕ_1 = unit hydrodynamic potential at rear stagnation point of a tube

D_o = tube outside diameter

P_o = tube pitch between tube centers, and

Pe = Peclet number using maximum local velocity.

9. Heat-Transfer Relation – Water/Steam Side

The heat-transfer relations between the inner tube wall and water/steam flow varied from region to region as shown in the following table:

Subcooled liquid	Dittus-Boelter equation
Subcooled boiling	Jens-Lottes equation ⁽²⁵⁾
Bulk boiling with wall wetting (nucleate boiling)	Chen equation ⁽⁵⁾
CHF quality	0.4
Transition boiling	McDonough-Millich-King equation ⁽²⁶⁾
Leidenfrost point (LP) quality	0.7
Bulk boiling with dry wall (dispersed film boiling)	Sieder-Tate equation for vapor fraction
Superheated steam	Heineman equation ⁽⁷⁾

10. Energy Balance Between Sodium and Water/Steam

$$d(W_A H_A) + d(W_B H_B) = W_{Na} c_{pNa} dT_{Na}$$

where

$$d(WH) = N \cdot d[\rho_L (1 - \alpha) u_L H_L + \rho_V \alpha u_V H_V] \cdot (\text{Area}) \text{ in two-phase region}$$

W_{Na} = total sodium flowrate

N = number of tubes in each family of tubes, and subscripts A and B refer to two families of tubes.

11. Region Boundaries – Water/Steam Side

The subcooled water enters the tube, thus forming the single-phase liquid region. The subcooled boiling region starts where the inner wall temperature reaches a value designated by the Jens-Lottes equation and ends where the bulk fluid reaches the saturation temperature ($X = 0$). Then the bulk boiling region with wetted wall condition begins and culminates at the critical heat flux

point, followed by the transition boiling region which ends at the Leidenfrost point. The bulk boiling region with dry wall condition then starts, followed by the superheated vapor region at the 100% quality location.

Numerical solution was attempted by use of the computer [the program is call Static Flow Instability Analysis (SFIA)]. The differential equations were put into finite difference forms and a set of nested iteration loops was used to solve the above equations simultaneously.

APPENDIX B

GAS ENTRAINMENT

Tests were conducted in 1966 to determine the extent and magnitude of gas entrainment in the SCTI secondary sodium system and its effects on the steam generator test program.⁽²⁷⁾ The results indicate that with a level of 50% of the control range, significant gas entrainment could occur at a sodium flow of about 1,100,000 lb/hr (2600 gpm) while with 85% of the level control range, significant gas entrainment would not occur until 1,250,000 lb/hr (3000 gpm) flow. Details of the test program and its findings are described below.

1. System Description

The SCTI secondary sodium system consists of a free-surface pump, the tube side of the intermediate heat exchanger (IHX), the shell side of the steam generator and an expansion tank (T-5). Figure B-1 shows the arrangement of these components and their relative locations.

Figure B-2 is a representation of the upper part of the steam generator showing the location of the inlet sodium nozzle, the upper tube sheet, the rupture-disk nozzle, and the sodium level gages. A gas space was maintained in the upper part of the steam generator to protect the upper tube sheet and the upper rupture disk from the high temperature, and rapid temperature changes, of the incoming sodium. Since the steam generator was not at the high point of the system, the steam generator sodium level was controlled by regulating the gas pressure in this gas space. This gas pressure was controlled, and the level maintained, by a controller which utilized the sodium level signal from the displacer-type level gage to position an argon supply and vent valve.

2. Test Method

Previous observations indicated that gas was being entrained in the steam generator and transported, at least partially, to the secondary expansion tank. Therefore, an argon bottle was temporarily connected to supply the gas space above the sodium free surface where gas entrainment occurred. The remaining argon supplies and vents in the secondary sodium system were closed. The vents were checked to verify that there was no significant argon leakage from the secondary sodium system. Thus the amount of gas entrained could be determined by the pressure change in the argon bottle.

Figure B-1. SCTI Secondary Sodium System

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Figure B-2. Upper Steam Generator Outline

For the tests which determined the rate at which gas was entrained and transported from the steam generator to the secondary expansion tank, system parameters were allowed to reach an equilibrium condition before data was collected. In the last test, which was to determine volume of gas continuously circulating throughout the system, data was taken prior to and immediately following the flow increase.

3. Test Results

The test data are presented in Figures B-3 and B-4. The secondary expansion tank level was measured with a precision level probe and the measurement given was from an arbitrary reference point down to the sodium level. Thus, a larger number means a decrease in sodium level in the expansion tank. The steam generator sodium-level measurement was from the displacer-type level gage and was given in percent of its 12-in. range.

Sodium flow was recorded in percent of full scale of a 0 to 30-mv recorder. Translation of this millivolts signal to a sodium flow in gpm was based upon rather limited heat balance data taken during low-power operation. Based upon this limited data, a flow calibration of 5600 gpm full scale on the 30-mv recorder has been derived. This calibration, however, could be in error by as much as 10%.

The gas transfer rates were based on an argon bottle content of 330 scf at a bottle pressure of 2640 psig. These values were obtained from Air Reduction Company, the argon supplier. This means that each 100-psig reduction in argon bottle pressure corresponds to 12.5 scf of argon being supplied to the steam generator via the level-control system.

The first data were taken at 1400 gpm, with no gas transport from the steam generator to the expansion tank as evidenced by the constant expansion tank pressure. The gas consumption of 6.7 scfh was assumed to be system leakage. Later tests at flows below the rate at which gas entrainment occurred showed system leakage somewhat below this value - in the range of 2 to 4 scfh. This difference was attributed to greater leakage at the temporary argon bottle connection.

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Figure B-3. Argon Consumption vs Sodium Flow

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Figure B-4. Argon Consumption vs Sodium Level

The second run was attempted at 2800 gpm, however, system operation became very erratic and large volumes of gas were being entrained (as shown by a rapid increase in the secondary expansion tank level). The flow was then reduced to about 2000 gpm and data were taken at that level. During this run, the expansion tank level decreased nearly to the level observed in run no. 1, indicating that the gas entrained at 2800 gpm was being de-entrained. During the time of the major de-entrainment, no gas was supplied from the argon bottle. The argon bottle pressure increased slightly during this time, apparently due to the recovery of the bottle temperature after being cooled somewhat, due to the heavy gas usage at 2800-gpm sodium flow.

Subsequent runs at various flows provided the data shown as Curve 1 in Figure B-3. As shown by this curve, gas entrainment starts to climb rapidly above about 2000 gpm when the steam generator sodium level is maintained at 50% on the displacer-type level gage. Operation at more than 2600 gpm with the above steam generator level produces excessive gas consumption rates and erratic system operation.

Four runs were then made to determine if operation with a higher steam generator sodium level would appreciably affect the gas entrainment rate. The results of these tests are shown in Figure B-4. These tests showed that an increase in steam generator level from 50% to 93% (only about 5 in.) caused an 80% reduction in the rate of gas entrainment at 2400 gpm.

In view of these results, it was decided to determine the relationship between sodium flow and rate of gas entrainment with the steam generator sodium level maintained at the highest value felt to be practical: 85%. The data from these runs are shown on Curve 2, Figure B-3. Operation at this steam generator sodium level allows operation to about 3000 gpm.

The final test was performed to determine the approximate volume of gas circulating around the system as opposed to the rate of gas transport from the steam generator to the secondary expansion tank. The test was started at the secondary sodium flow of about 1700 gpm, that was the flow at which gas entrainment begins with the steam generator sodium level at 50%. After all system parameters had stabilized, data were taken and then the flow was increased to about 2500 gpm in about 1 min. Over the next 20 min, the secondary expansion

tank level increased 4 in., indicating that about 9 ft³ of gas was being circulated around the system with the sodium. The sodium inventory of the secondary sodium system is about 460 ft³; thus the gas constituted about 2% by volume. This level of gas content should not have a significant effect on the heat-transfer characteristics of the IHX or steam generator.

APPENDIX C
CHEMICAL EXAMINATION OF FEEDWATER HEADER CRACK DEPOSITS
AND INSULATION

Deposited material was removed from three locations near the cracks on the feedwater header of the steam generator following its failure. One sample was taken near the top of the cracked area, and two near the bottom. The chemical analysis results are presented in Table C-1.

TABLE C-1
CHEMICAL ANALYSIS RESULTS OF FEEDWATER HEADER
CRACK DEPOSITS

Element	Emission Spectrograph Analysis (wt %)		
	Bottom Yellow Sample	Bottom Brown Sample	Top Sample
Al	-	-	4
Ca	-	-	8
Cr	3	3	1
Fe	4	6	4
Mg	-	-	10
Mn	0.2	0.6	0.4
Mo	2	2	0.4
Na	30	25	8
Ni	1	2	-
Si	0.2	0.2	15
Ti	-	-	0.2

Top Sample: X-ray diffraction (XRD) analysis showed that the top sample contained $\text{NaAlSi}_3\text{O}_8$ plus other compounds which were not identified.

Bottom Sample: XRD analysis of the yellow portion of the bottom sample identified this material as Na_2CO_3 . The brown portion was identified as a mixture of Na_2CO_3 and NaFeO_2 .

Samples of the thermal insulation taken from the lower end of the steam generator were also submitted for chemical analysis. Three of these (A, B and D) were taken from the feedwater header, below the tube sheet. The fourth (C) was taken from just above the tube sheet. Results of the emission spectrograph analysis of the insulation material are given in Table C-2.

TABLE C-2
CHEMICAL ANALYSIS OF FEEDWATER HEADER
INSULATION MATERIAL

Element	Emission Spectrograph Analysis (wt %)			
	Sample A	Sample B	Sample C	Sample D
Al	0.2	0.2	0.2	0.2
Ca	0.6	0.3	0.4	0.4
Fe	3	3	3	3
Mg	5	5	5	5
Mn	0.3	0.3	0.3	0.3
Na	0.5	0.5	0.3	0.6
Si	30	30	30	30

Emission spectrograph analysis indicated that samples A, B, C, and D were essentially the same, so only sample A was examined in detail by XRD, and X-ray fluorescence (XRF). XRF analysis showed the sample contained less than 50 ppm Cl (not detected). XRD indicated that the fibers in the sample are an amphibole asbestos of the cummingtonite-grunerite series characterized by the presence of iron, magnesium silicate. The remainder of the sample included amorphous material plus several forms of silica which are quartz, and cristobalite. Microscopic examination showed that the amorphous component is an opaline silica known as diatomaceous earth.

Analysis of cations and anions detected in insulation samples dissolved in water heated to $\sim 200^{\circ}\text{F}$ is presented in Table C-3.

TABLE C-3

ANALYSIS OF CATIONS AND ANIONS IN FEEDWATER HEADER INSULATION MATERIAL

Originator's Sample Number	p H	Fe (ppm)	Cl ⁻ (ppm)	Ca (ppm)	F ⁻ (ppm)	Mg (ppm)	SO ₄ ⁼ (%)	K (%)	Na (%)
No. A	9.7	50	367	26	18	58	0.11	0.03	0.30
No. B Beige	9.4	25	64	10	12	19	0.03	0.05	0.05
No. B Black	8.7	22	18	12	3	30	0.05	0.02	0.01
No. C	9.4	73	259	16	12	34	0.02	0.05	0.05
No. D	9.1	93	110	11	6	28	0.15	0.09	0.05

NOTES

- 1) Cl⁻ and F⁻ are by specific ion electrodes
- 2) Fe and Mg are by atomic absorption spectrophotometry (AAS)
- 3) Ca, K, Na are by flame photometry
- 4) SO₄⁼ are by turbidity measurements
- 5) ppm = $\frac{\text{Mg X}}{\text{gm of material leached}}$

APPENDIX D

HEAT-FLUX CURVES

Heat-flux curves which are not reported in the main text are collected in this appendix. These are the runs between the initial and final steady-state points of each diagnostic test series. The heat-flux curves of the latter runs are reported in Section V.B.4 and those of performance changes are in Section V.D.2.

Figure D-1. Heat-Flux Curves - Diagnostic Test Series, Tests 1 and 2

Figure D-2. Heat-Flux Curves – Diagnostic Test Series, Tests 2R and 3

Figure D-3. Heat-Flux Curves – Diagnostic Test Series,
Tests 5AR and 5B

Figure D-4. Heat-Flux Curves – Diagnostic Test Series, Tests 6A and 6B

Figure D-5. Heat-Flux Curves - Diagnostic Test Series, Tests 9 and 8R

Figure D-6. Heat-Flux Curves — Diagnostic Test Series, Tests 9R and 10B

Figure D-7. Heat-Flux Curves - Diagnostic Test Series, Tests 16 and 15

Figure D-8. Heat-Flux Curves — Diagnostic Test Series, Tests 17 and 20

Figure D-9. Heat-Flux Curves – Diagnostic Test Series, Tests 21 and 22

APPENDIX E

ALCO/BLH STEAM GENERATOR POST-TEST EXAMINATION PLAN

Post-test examination, which includes failure analyses and compilation of operational history, is part of the program for performance and structural evaluations of the Alco/BLH steam generator. The following sections describe the objectives and examinations of interest based on experienced performance anomalies and failures.

A. OBJECTIVES

1. General Objectives

The major objectives of the post-test examination are as follows:

- 1) Determine the metallurgical nature and basic cause of all defects which occurred during service.
- 2) Provide data on the condition of steam generator materials after service which may be of interest to current LMFBR steam generator designers, manufacturers, and developers.
- 3) Provide data which may help resolve heat transfer and temperature distribution anomalies observed in testing.

2. Specific Objectives

The examination plan has the following specific objectives:

- 1) Obtain visual, metallurgical, and chemical data necessary to define the mode of failure of the tube sheet, and to determine the underlying cause.
- 2) Obtain corresponding data for previously discovered defects, including:
 - a) Tube leaks
 - b) Tube cracks
 - c) Manway lip deformation
 - d) Upper manway groove cracks

- e) Microfissures in Inconel overlay
 - f) Failed internal thermocouples.
- 3) Determine whether other deterioration or defects have developed in the unit; this includes but is not limited to the following:
- a) Tube-to-tube sheet welds
 - b) Tube-to-guide ring tack welds
 - c) Inconel-stainless steel interface of steam tubes
 - d) Tube-baffle contact points
 - e) Inconel overlay.
- 4) Determine whether corrosion or foreign deposits, or both, have occurred on any internal and external surfaces of the steam generator. Of particular interest are inner and outer tube surfaces, tube cavities in tube sheets, shield plate-shell gap space, lower tube-sheet shield stack, inner shell side of lower tube sheet, and outer shell.
- 5) Verify location and readout circuitry of lower tube-sheet thermocouples which have yielded anomalous temperature indications.
- 6) Verify location of shell shield-plate drain slot, and determine whether it is related to the tube-sheet cracks.
- 7) Check condition of tubes reported to have presented particular problems during manufacture.
- 8) Check condition of three shell thermocouples and the drain hole in the lower tube sheet, and determine whether they are related to the tube-sheet cracks.
- 9) Check for evidence of vacuum pump oil siphoned into the shell side and any possible contribution of this contaminant to structural failure.
- 10) Check for corrosion of tubes and shell at sodium/cover-gas interface surfaces.

B. SUMMARY OF DAMAGE AND DEFECTS

The presently known unrepaired defects and damage in the steam generator are summarized in this section (see Section IV.A for details).

1. Tube Leaks and Cracks

See Figure 6-1 for tube locations.

- 1) Tube II-47 contains a leak which appears as an approximately circumferential crack 6-1/2 in. below the tube-sheet surface. This location is within the tube sheet.
- 2) Tubes III-52 and III-53 contain leaks located 1-3/4 in. below the tube-sheet upper surface. This location is just below the tube end section which is expanded into the tube-sheet hole.
- 3) Tube II-48 contains an external crack estimated to penetrate 90% of the wall thickness, 6-3/8 in. below the tube-sheet upper surface.
- 4) Tubes III-51 and I-42 contain external cracks estimated to penetrate 60% of the wall thickness located 8 in. below the tube-sheet upper surface. This location is below the tube sheet and above the guide ring.

All data on nonleaking cracks were based on eddy-current test results.

- 5) Tubes I-22, I-27, and IV-41 contain burn-through at the guide ring.

2. Manway

- 1) Upper and lower manway lips (section between ring groove and inside diameter) are deformed inward.
- 2) A boat specimen from the upper manway groove contained microfissures at the Inconel-stainless steel interface.

3. Water Header

A group of pitting indications exist in various areas of the water inlet header.

4. Thermocouples

Most of the internal thermocouples in the heat-transfer zone (TC numbers 13 through 44) are inoperative. A number of thermocouples in the steam header

and other areas are broken at the outside of the thermocouple penetration fittings of the header.

5. Tube-Sheet Crack

The damage visible on the external surface of the lower tube sheet is a branch-like crack network situated in an area approximately 1 ft^2 . This network is located nearly opposite the water inlet line and is shown in Figure 6-21.

A preliminary metallurgical examination conducted to characterize this crack network indicated:

- 1) Brittle rather than ductile fracture.
- 2) Cracking terminated rather than originated on the external surface.
- 3) Tube-sheet hardness is uniform with no significant variations between the cracked (failed) and unfailed areas.

This examination is discussed in greater detail in Section VI.A.7.

C. DISASSEMBLY AND EXAMINATION

1. Proposed Examination

Table E-1 lists the proposed NDT activities for the steam generator. These tests and analyses will provide information on the condition of the unit, including manufacturing defects and those incurred as a result of operation. The NDT examinations will be performed before and during disassembly. The results will be used to select areas for more detailed scrutiny, and to select specimens for metallurgical examination.

Table E-2 lists examination requirements which involve detailed sectioning, removal of specimens, and metallurgical examination. Included are estimated quantities of the specimens which must be obtained and examined to furnish the required information.

Table E-2 covers only presently known defects and problems of interest. It must be anticipated that the examinations listed in Table E-1 may reveal other possible defects which will require additional sectioning and metallurgical examination.

TABLE E-1

NDT PRE-DISASSEMBLY AND DISASSEMBLY EXAMINATIONS

(Sheet 1 of 2)

Activity	Type of Examination										
	V	T	Ph	F	P	A	E	B	He	Cl	Ra
Pre-disassembly Tests											
1. Complete steam generator	X		X								
2. Thermocouple checkouts		X									
3. Tube flow tests (air)				X							
4. Lower tube sheet	X		X		X	X					X
5. Lower support surface	X					X					
6. Shell and support	X					X					
7. Tube tests (emphasize anomalous tubes)	X						X	X			
8. Leakage test of shell									X		
9. Interior examination	X		X								
Disassembly and examination											
1. Remove upper and lower heads											
a. Head interiors	X		X		X					X	
b. Tube-sheet front faces	X		X		X						
c. Tube interiors	X		X					X			
2. Remove upper shell											
a. Upper tube-sheet backface	X		X								
b. Guide ring (including tack welds)	X		X		X					X	
c. Tube sine wave area	X		X							X	
d. Shield plate (cylindrical)	X		X		X					X	
e. Thermocouples	X	X	X								
f. Inner surfaces of shell	X		X		X					X	
g. Shroud and support ring	X		X		X					X	
h. Cover gas/sodium interface tube surfaces	X		X							X	
3. Remove lower shell											
a. Inner surfaces of shell	X		X		X					X	
b. Shield plate (cylindrical)	X		X		X					X	
c. Outlet nozzle shield plate	X		X		X					X	
d. Upper shroud and ring	X		X		X						
e. Auxiliary annulus shell drain	X		X					X		X	
4. Remove shield plates											
a. Tubes	X		X								
b. Lower shroud exterior	X		X		X					X	
c. Shield plate and spaces	X		X		X					X	
d. Baffles and support rods	X										
e. Lower shield plates	X		X								
f. Thermocouples	X	X	X								
5. Cut upper and lower shrouds											
a. Shroud interior and exterior	X		X		X					X	
b. Tube sections	X		X		X						

TABLE E-1

NDT PRE-DISASSEMBLY AND DISASSEMBLY EXAMINATIONS

(Sheet 2 of 2)

Activity	Type of Examination										
	V	T	Ph	F	P	A	E	B	He	Cl	Ra
6. Remove tube sheets											
a. Guide ring lower face	X		X		X					X	
b. Upper tube-sheet lower face	X		X		X					X	
c. A portion of the lower tube-sheet shield plates	X		X								
7. Remove tubes from baffles											
a. Tube exteriors (contact evidence)	X		X		X					X	
b. Baffles	X		X		X					X	
c. Thermocouples and harnesses	X	X	X								
8. Disassemble lower shield plates											
a. Shell-shield plate	X		X		X					X	
b. Backface of lower tube sheet	X		X		X					X	
c. Circular shield plates	X		X		X					X	
d. Tubes in circular shield plates	X		X		X					X	
e. Tube-sheet backface	X		X		X					X	X
f. Thermocouple junctions	X	X	X		X						
9. Remove lower shell-shield plate											
a. Tube-sheet extension	X		X		X					X	
b. Shell	X		X		X					X	
10. Cut tubes											
a. Guide ring and tack welds	X		X		X					X	
b. Backface of upper tube sheet	X		X		X					X	
c. Tube ends						X					

Legend:

V - Visual	E - Eddy current
T - Thermocouple checkouts	B - Borescopic
Ph - Photographic	He - Helium leak test
F - Flow	Cl - Area cleaning
P - Liquid penetrant	Ra - Radiographic
A - Alignment verification	

NOTE:

1. Visual examination will consist of inspection for cracks, corrosion, erosion, fretting, pitting, wear, distortion, deposits, loose parts, evidence of fatigue, tube-to-tube contact, tube-to-shell/ shield plate contact. Detected anomalies will be subjected to appropriate additional examination.
2. Visual, liquid penetrant, radiographic, and helium leak tests will be performed in accordance with applicable sections of RDT Standard F3-6T.
3. Cleaning methods should not obscure evidence of chemical attack or contaminate areas of interest with cleaning solution.

TABLE E-2

DETAILED SECTIONING AND METALLURGICAL EXAMINATION REQUIREMENTS

(Sheet 1 of 3)

Part or Subassembly	Description of Specimen	Estimate Number of Specimens	Type of Examination	Information Required
Upper tube sheet (including tube ends, tube welds, header weld and shell weld)	Leaking tubes II-47, III-52 & III-53	12	M H	Nature and cause of leak defects.
	Cracked tubes II-48, III-51 & I-42	12	M H	Nature and cause of crack defects.
	Tubes I-2, I-5, I-6, I-21 & I-28; remove expanded section from the tube sheet	5	D	Check condition of tubes in which roller breaks occurred during manufacture; and check effect of acid where used for removal.
	Tube plugs installed by fabrica- tor; tubes I-22, I-27, IV-41	2	M	Condition of plug weld (in service throughout life of steam generator).
	Tube plugs installed by LMEC; Tubes II-47 & IV-27	2	M	Condition of plug weld (in service since December 1969).
	Inconel overlay surfaces			
	Tube-sheet surface	5	M	Condition of overlay and interface; examine for microfissures.
	Header extension	3	M	
	Header weld	3	M	
	Ligament sections between tubes, including ligament between II-47 and II-48	12	D H EM	Presence of residual stresses and/or warping of ligaments; evidence of fatigue damage or cracking.
	Axial section through tube, weld and ligament (un- damaged tubes)	12	SE C M	Corrosion and foreign matter in tube cavity; condition of weld; condition of expanded tube section.
	Shell-side surfaces			
	Flat surface	4	M,C	Corrosion, hydrogen, embrittlement, any adverse effects of sodium- water reaction products.
	Shell extension	3	M,C	
	Shell weld	3	M,C	
Lower tube sheet (including tube ends, tube weld, header and shell welds)	Crack area of tube sheet and header (to be sectioned as required to trace cracks)	50	SE M D C R	Cause, nature, and extent of cracks; this problem to take priority over all others in allocation of specimens.
	Inconel overlay areas			
	Flat surface	5	M	Condition of overlay and interface
	Header extension	3	M	
	Header weld	3	M	Comparison of lower vs upper header overlay condition.

TABLE E-2

DETAILED SECTIONING AND METALLURGICAL EXAMINATION REQUIREMENTS
(Sheet 2 of 3)

Part or Subassembly	Description of Specimen	Estimate Number of Specimens	Type of Examination	Information Required
Lower tube sheet (continued)	Axial section through tube ends, tube weld and ligament (undamaged tubes); some near tube-sheet crack area	12	SE C M	Corrosion and foreign matter in tube cavities; condition of welds; condition of expanded tube section.
	Tube sheet, thermocouple, and drain holes (section to expose full length).	4	SE C	Corrosion and foreign matter; determine if drain was plugged while in service; check for relationship of holes to cracks; check for evidence of vacuum pump oil contamination.
Tube gross length (including heat transfer section and sine wave)	Undamaged tubes; section axially entire straight length	15	SE C H	1. Steam-side surface condition; corrosion, fouling & deposits (their distribution (if any) in relation to axial position in heat transfer zone). 2. Determine if mechanical strength varies with length due to temperature exposure.
	Tube I-23, I-27; section axially entire straight length	2	SE C	Steam-side surface condition; compare tube plugged throughout service with operating tubes.
	Undamaged tubes; section each normally at following heights above tube sheet 2, 7, 14, 22, and 33 ft.	5 (25 total cuts)	M T	Condition of tube bond with Inconel liner.
	Undamaged tubes; section encompassing sodium/cover gas interface.		SE M	Corrosion at sodium/cover gas interface.
	Tube sections at ends of sine wave bend		M EM	Vibrational fatigue.
Steam header (including steam nozzle & upper manway)	Ground and overlay-restored seal groove areas	8	M	1. Condition of overlay and interface 2. Determine whether cracks remain 3. Comparison of LMEC vs Alco overlay.
	Deformed manway lip area	8	D M	1. Extent of deformation 2. Nature of damage.
	Inconel overlay surfaces			
	Dome	6	M	Condition of overlay and interface; presence of microfissures.
	Nozzle	3	M	
	Manway neck	3	M	

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TABLE E-2

DETAILED SECTIONING AND METALLURGICAL EXAMINATION REQUIREMENTS
(Sheet 3 of 3)

Part or Subassembly	Description of Specimen	Estimate Number of Specimens	Type of Examination	Information Required
Feedwater header (including water nozzle & lower manway)	Seal groove	8	M	Compare overlay and interface for lower vs upper manway seal groove; help determine why only upper had cracks.
	Deformed manway lip areas	8	D M	1. Extent of deformation 2. Nature of damage 3. Comparison of lower vs upper lip damage.
	Inconel overlay surfaces			
	Dome	6	M	Condition of overlay and interface; presence of microfissures.
	Nozzle	3	M	
	Manway lip	3	M	

LEGEND:

- H - Hardness testing
- D - Dimensional check
- M - Metallurgical examination
- SE - Surface examination for deposits and corrosion
- SG - Strain gage and remove material to measure residual stress
- C - Chemical
- R - Radiography
- EM- Electron microscope
- T - Thermograph

2. Discussion of Specialized Examination Techniques

Specialized examination techniques may be used to increase knowledge about areas of the steam generator. These techniques, for complete material characterization, would involve the application of electron microscopy, electron diffractometry, mass spectrometry, X-ray, and electron-beam microanalysis. Metallurgical analyses and NDT methods in the form of radiographic, ultrasonic, and xenon autoradiographic testing will also be employed.

A specialized technique, such as time-domain reflectometry, may be applied to determine the location and nature of thermocouple defects: an electrical pulse is applied to a thermocouple lead and the signal reflected back provides defect information through its wave shape and timing.

Another approach, the thermograph, may be used to determine whether steam tubes have any nonuniformity of heat-transfer properties due to separation at the stainless-Inconel interface or due to other causes. In this technique, the tubes are coated externally with temperature-sensitive materials and heated internally by an electrical heating element which provides uniform heat flux per unit length of tubing.

3. Sequence of Disassembly and Examination

Disassembly is to be accomplished in 10 steps, as described in this section. The areas exposed by the operation are listed in each step. These areas are to be visually examined and photographed. Also listed with each step are examination items of special interest, and special examination procedures which must be prepared prior to the next step of disassembly. Detailed procedures covering the examination requirements shall be prepared. Thermocouple inspections required at intermediate stages of disassembly may be determined by the time-domain reflectometry technique.

An examination bed will be used to provide access to portions of the steam generator circumference while it is in a horizontal position.

Plastic enclosures will be used wherever possible in cutting and NDT operations to maintain an inert atmosphere over sodium-contaminated components. All cuts are to be made by cutting tools of adequate precision. No torch cutting is to be done. Dry cutting tools will be used to avoid lubricant contamination.

In the course of disassembly, examination should be made of (but not limited to) the following categories of defects and damage: cracks, corrosion, erosion, fretting, distortion, pitting, wear, deposits, loose parts, fatigue, evidence of contact between active and inactive tubes, between active tubes, and between tubes and shell.

All stagnant spaces, including liner-shell and liner-sodium nozzle gaps, guide ring-to-upper tube sheet space, lower tube-sheet thermal shield gaps and tube-tube sheet cavities will be checked for foreign matter at appropriate stages of disassembly; and the surfaces of such spaces should be checked for corrosion.

Cleaning techniques will be selected with a view toward avoiding contamination or removal of evidence.

The 10-step disassembly procedure is listed as follows:

Step 1. Remove upper and lower headers (with integral nozzles). Parting line to be on the header side of the header-nozzle weld.

Areas Exposed for Examination: head interiors, tube-sheet front faces, tube interiors.

Step 2. Make local circular shell cutouts around each thermocouple penetration near upper end of shell and around the level and sampling ports. Remove two shell segments as follows:

Each segment bounded by a circumferential cut just above the upper tube sheet-to-shell weld; a second circumferential cut just above the upper shroud support ring (8-1/2 ft below the first cut); and two axial cuts 110 degrees apart. The two segments are separated by a 20-degree uncut segment equally spaced between the sodium inlet and rupture disk nozzles. The shell shield plate should not be cut in this step. The sodium inlet and relief nozzles will each be included in one of the cutout segments.

This step will leave a 120-degree segment of shell for support on the examination bed, and an opposite (top) 20-degree segment to maintain rigidity of the assembly.

Areas Exposed for Examination: Upper tube-sheet back face, guide ring, tube sine wave area, exposed outside surface of liner, sodium nozzle liner, radial-running portions of thermocouple bundles, inner surface of removed shell segments.

Problems of Special Interest:

- 1) Condition of liner-shell gap.
- 2) Tube-to-tube sheet interfaces.
- 3) Sodium-water reaction products at thermocouple-tube sheet interfaces and/or surface of tubes.
- 4) Sodium-water reaction products in guide ring-tube sheet space.
- 5) Foreign matter on shroud support ring.
- 6) Upper shell vent connection (in tube sheet).

Step 3. Circumferentially cut the sodium outlet nozzle shield plate from the nozzle near the exit. Circumferentially cut the shell 6 in. from the inboard side of the shroud support ring to sever the shroud ring from the section of shell to be removed. Extend the axial cuts of Step 2 to a point just above the lower tube sheet-shell weld. Cut circumferentially at this point to release two 110-degree shell segments in line with the two previous segments. A 120-degree support segment, and a 20-degree stabilizing segment are left in line with those of Step 2. The upper shroud will also remain intact.

Areas Exposed for Examination: Shell interior, remainder of shell shield plate, sodium outlet nozzle shield plate, upper shroud and ring.

Problem of Special Interest: Evidence of tube and shell contact.

Step 4. Cut out shield plate segments corresponding to the previously cut out shell segments.

Areas Exposed for Examination: Full length of tubes except within shrouds, lower shroud exterior, shield plate interiors, shield

plates, and shell shield plate spaces, baffles, support rods, lower circular shield plates, thermocouples.

Problem of Special Interest: Evidence of tube-baffle fretting.

Step 5. Cut out upper and lower shrouds. Cut the upper shroud in segments corresponding to the previously cut out shell segments, and allow small sections of shell and shroud to remain attached to the guide ring. Cut lower shroud in line with axial termination of shell cuts.

Areas Exposed for Examination: Shroud interiors, tube sections within shrouds.

Step 6. Remove the tube sheets and shell segments. Cut through entire tube bundle and shell segments: (1) just below the guide ring; and (2) in line with the lower end circumferential shell cut. Do not cut through the axially-running thermocouple bundle which passes within the tube bundle. This step leaves three separable subassemblies:

- 1) The upper tube sheet and guide ring with tube ends.
- 2) Tube bundle.
- 3) Lower tube sheet and circular shield stack with tube ends.

In separating these, the thermocouple harness remains with the tube bundle. The six radial bundles at the upper end will be severed inside the shell penetrations and withdrawn through the guide ring.

Areas Exposed for Examination: Guide ring lower face, portions of upper tube-sheet shell side, top shield of lower tube-sheet shield plate stack.

Problems of Special Interest:

- 1) Tube-to-guide ring tack welds.
- 2) Leaky, burned through, plugged, internally damaged, and cracked tubes, and conditions in their vicinity. Tubes in inner row which were expanded into guide ring in manufacture. Broken roller expander was removed from tube inside diameter with nitric acid in tubes I-11 and I-20 (see Figure 6-1).

Step 7. (Applies to tube bundle subassembly) Withdraw the tubes individually from the baffle system, observing each tube for fretting damage at the baffles as it becomes visible. Maintain baffle system in alignment by support rods until inspection of thermocouple system is completed, then separate rods and baffles.

Areas Exposed for Examination: Tube exteriors, baffles, thermocouple junctions and harness. Examine for evidence of contact between active and inactive tubes or contact among active tubes.

Step 8. (Applies to lower tube-sheet subassembly) Disassemble the lower tube-sheet shield plates one by one, observing foreign matter in inter-shield spaces.

Areas Exposed for Examination: Lower end interior surface of shell shield plate, back face of lower tube sheet, lower tube-sheet shield plate surfaces, tube surfaces within the circular shield plate, stack, tube-sheet face thermocouple junctions, shell side of tube sheet.

Problems of Special Interest:

- 1) Position of shield plate drain slot. Any apparent relationship between this slot and the crack system.
- 2) Evidence pertaining to the cracks.
- 3) Presence of corrosive contaminants and their relationship to crack network.
- 4) Verification of location of lower tube-sheet thermocouples 1 through 9 and their condition.
- 5) Condition of the lower tube-sheet drain hole and its relation (if any) to cracks.
- 6) Condition of the tie rod and shield bolt holes in tube sheet.

Step 9. (Applies to lower tube-sheet subassembly) Remove the lower end of the shell shield plate. Shell side of tube-sheet extension is exposed. Tube sheet is ready for detailed examination.

Step 10. (Applies to upper tube-sheet subassembly) Cut through individual tubes just above guide ring. Measure deformation of tubes when cut. Continue this procedure for inner rows as they are exposed so as to sample every row. When entire guide ring is removed, inspect for evidence of sodium-water reaction products and distribution pattern of leakage and reaction products. Make further observation by photographic and dye-penetrant checks of condition of tube-guide ring tack welds as they appear on upper side of guide ring. The upper tube sheet is now ready for detailed examination.

Problems of Special Interest: Tubes which were rolled into guide ring and those within this group from which rollers were removed with acid.

D. SAFETY PRECAUTIONS

The proposed test plan for examination of the Alco/BLH Steam Generator should not present restrictive health and safety hazards to either personnel or facilities.

The following guides shall be applied to examination plans when there may be personnel exposure to hazard:

- 1) The steam generator shall be rested in a cradle stressed for the weight involved and for anticipated rotational loads.
- 2) Sodium-handling protective clothing (hardhats, face shields, coveralls, gloves, goggles) shall be available and used by the personnel involved when cuts are made into the sodium side.
- 3) Provide fire protection service during sodium-side cuts.
- 4) Adequate quantities of dry calcium carbonate in metal cans shall be used for scrap sodium disposal. These cans of scrap shall be disposed daily. Met-L-X extinguishers shall be available.
- 5) Hoisting and rigging and material handling equipment shall be available and currently proof loaded.
- 6) Cuts shall be made in accordance with written procedures approved by project, operations, and health and safety personnel. On-the-spot

deviations shall be written in and approved by project, operations, and health and safety personnel.

- 7) Hand tools, cutting equipment, plastic sheet, tape, and inert gas supply shall be available for the start of each job.
- 8) Small sodium cleaning jobs may safely be performed in well ventilated areas using butyl alcohol. The use of Dowanol-EB will require special handling.
- 9) Steam cleaning at a suitable disposal area will be required for large amounts of sodium removal.
- 10) Sodium-contaminated parts to be examined shall be wrapped in plastic and identified as containing sodium.
- 11) Personal cars shall not be used to transport sodium-contaminated parts.
- 12) Openings in the sodium side of the steam generator shall be sealed with plastic sheet and taped at the conclusion of each operation. An inert cover gas shall be used between opening operations and examinations.
- 13) An inert cover gas shall be used on the sodium side of the generator when left overnight.
- 14) A qualified crane operator shall man all lifts.
- 15) Sections removed that are not needed for examination may be disposed of if they contain no sodium. Sodium-containing sections shall be steam cleaned and verified as clean prior to disposal.
- 16) Visitor control will be necessary. Observers shall be limited to those immediately concerned with the examination process. When cuts, lifts, or sodium removal is in progress, observers shall not be allowed in close proximity to the steam generator or working crews.

E. QUALITY ASSURANCE REQUIREMENTS

1. Purpose

The Quality Assurance (QA) requirements are established to:

- 1) Assure that the post-test examination of the steam generator is accomplished in a planned and systematic manner, and that adequate documentation (photographs, logs, disassembly procedures, inspection records, analyses, correspondence, and examination procedures) is accumulated and maintained.
- 2) Assure that required information is obtained at each stage of destructive examination.

2. Requirements

The specific QA requirements are as follows:

- 1) The organization performing the examination shall provide QA support having the authority to:
 - a) Monitor all examination activities to determine adequacy of any compliance with requirements, plans, and procedures.
 - b) Initiate nonconformance reports if necessary.
 - c) Perform or supervise all inspection and NDT activities, including preparation of inspection and test records.
- 2) If packaging and storage of the steam generator is required this should similarly be monitored by QA.
- 3) Before the start of examinations and evaluations, a list shall be prepared defining the responsibilities of key individuals associated with this task. The list shall designate the individual responsible for providing written approval before any inspection, test, removal, or disassembly operation.
- 4) A log book shall be maintained summarizing daily activities.
- 5) A file shall be maintained containing as a minimum:
 - a) Photographs
 - b) Inspections performed
 - c) Analyses
 - d) Correspondence

- e) Disassembly procedures
 - f) Examination procedures
 - g) Disposition of parts, samples, subassemblies
 - h) Requests for support.
- 6) Nondestructive tests performed for permanent data purposes shall be accomplished by QA personnel. Certain tests (e. g., eddy current) may be delegated to other qualified organizations.
 - 7) Nondestructive tests to be performed for preliminary evaluation purposes may be performed by other personnel known to be competent. The QA engineer will provide approval for any NDT's in this category.
 - 8) An identification system shall be originated to assure that all reports, photographs, and inspection results referring to specific locations of the steam generator or associated interfaces are identified in a consistent manner to preclude the possibility of any doubt as to axial, radial, interior, exterior or other position orientation of the subject part, sample, or subassembly.
 - 9) Disassembly and inspection instructions shall be in writing and shall be approved by appropriate supervision. The instructions shall contain pertinent cautionary or safety notes, including requirements for protection of parts.
 - 10) Access to the steam generator shall be controlled to preclude the possibility of damage or obliteration of evidence.
 - 11) The QA engineer shall evaluate all inspection data and provide input for inclusion in a final report by all participating functions.

APPENDIX F
ENGINEERING DRAWINGS OF ALCO/BLH STEAM
GENERATOR

DETAIL OF 7\"/>

CALCULATED WEIGHTS (LBS.)	
ENTIRE UNIT (DRY)	33,000
ENTIRE UNIT (OPERATING)	42,000
ENTIRE UNIT (WITH WATER)	46,500
MANHOLE COVER EACH	960

	DESIGN PRESS.	DESIGN TEMP.	TEST PRESS.
SHELL SIDE	150 P.S.I.	1200° F.	620 P.S.I.
CHANNELS	2500 P.S.I.	1075° F.	6220 P.S.I.
TUBES ONLY	2500 P.S.I.	1200° F.	6220 P.S.I.

REFERENCE DRAWINGS	
TUBE STUDY	D666-4-4
X-SECTION	D666-4-5
NOZZLE CONN.	B666-4-6
NOZZLE (RUPTURE DISC)	B666-4-7
DETAILS	C666-4-8

REVISIONS	NO.	DATE	BY	CHKD.	DESCRIPTION
1	1	10/1/68	W.M.	W.M.	GEN. REV. SHEET 1
2	2	10/1/68	W.M.	W.M.	GEN. REV. SHEET 2
3	3	10/1/68	W.M.	W.M.	GEN. REV. SHEET 3
4	4	10/1/68	W.M.	W.M.	GEN. REV. SHEET 4
5	5	10/1/68	W.M.	W.M.	GEN. REV. SHEET 5
6	6	10/1/68	W.M.	W.M.	GEN. REV. SHEET 6
7	7	10/1/68	W.M.	W.M.	GEN. REV. SHEET 7
8	8	10/1/68	W.M.	W.M.	GEN. REV. SHEET 8
9	9	10/1/68	W.M.	W.M.	GEN. REV. SHEET 9
10	10	10/1/68	W.M.	W.M.	GEN. REV. SHEET 10

Figure F-1. Steam Generator Outline
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- 1.00^{+0.06}-.00 DIA. C'BORE TUBE
EXTENSION & WELD BEAD TO
INCONEL OVERLAY PARENT METAL
TUBE END TO BE MACHINED FLUSH OR
NO MORE THAN .030 BELOW C'BORE SURFACE.

TUBE SHEET SST ASTM A182 GRADE TP316
TUBE - SST ASTM A213 GRADE TP316

DETAIL A
PREPARATION OF TUBE SHEET
PRIOR TO PLUG WELDING
TYPICAL FOR UPPER & LOWER TUBE SHEETS
SCALE 2/1

SECTION C-C
LOCATION OF PLUGGED TUBES

NOTES: UNLESS OTHERWISE SPECIFIED

1. BREAK ALL SHARP EDGES
2. TWO PLACE DECIMAL TOLERANCES
3. THREE PLACE DECIMAL TOLERANCES
4. ANGULAR TOLERANCE $\pm 0.30^\circ$
5. USE THIS DRAWING WITH AI PROCESS PROCEDURE
110.WP-001-14-001
① 6. WELD PER AI PROCESS PROCEDURE 110.WP-001-14-001
7. INSPECT PER AI PROCESS PROCEDURE 110.WP-001-14-001

DETAIL 8
TYPICAL FOR UPPER & LOWER TUBE SHEETS
SCALE 2/1

➤ BREAK CORNERS .015

DET 4-27 UPPER PLUG IREQD C'DIA. = .312 +.000
DET 4-27 LOWER PLUG IREQD C'DIA. = .316 -.001
DET 2-47 UPPER PLUG IREQD C'DIA. = .305 +.001
DET 2-47 LOWER PLUG IREQD C'DIA. = .319 -.001
MAT'L = INCONEL 600
SCALE 2/1

Figure F-9. Tube Plug Repair and Installation