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MASTER

# STATUS OF THE CRBRP STEAM-GENERATOR DESIGN

J. E. Schmidt (W-LRM)

R. S. Martinez (AI-ESG)

J. F. Murdock (CRBR-PO)

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## 1.1 INTRODUCTION AND BACKGROUND

The Clinch River Breeder Reactor Plant (CRBRP) steam generators have a long history of design, fabrication, and testing at Atomics International, a division of Rockwell International. The CRBRP hockeystick configuration was first conceived in the late 1960's with the design and fabrication of a 30-MWT test unit which was subsequently tested, under a cooperative agreement with DOE, at the Energy Technology Engineering Center in the early 1970's. The 30-MWT test unit was a 158-tube unit, approximately 19.8 m (65 ft long), 0.46 m (1.5 ft in diameter), and was constructed entirely of the low-alloy ferritic steel 2-1/4 Cr - 1 Mo.

Testing was initiated in mid-1973 and approximately 9000 hour of sodium side testing/4000 hour steaming operation were accumulated prior to termination of the test in mid 1974. Following testing, an extensive post-test examination of the unit was conducted. The results of that examination indicated the unit to be in excellent condition with no obvious deleterious effects resulting from the test environment on either the sodium side or waterside of the unit.

About the time the 30-MWT test was initiated, a contract was awarded to Atomics International by DOE for the conceptual and preliminary design of a hockeystick-type steam generator for possible use in the Clinch River Breeder Reactor Plant (CRBRP). The conceptual design was completed in late 1973. This design was based on the use of three evaporator/three superheater configuration per loop for CRBRP. In March of 1974, the decision was made to employ a two evaporator/one superheater configuration per loop using a recirculation ratio of 2:1, allowing departure from nucleate boiling (DNB) in the evaporator. In the following month, April 1974, a decision was made to continue with the hockeystick steam generator as the reference configuration for the CRBRP. By early fall of that year, a preliminary review design of the unit was completed.

Engineering final design of the unit was initiated in late Fall 1975. By the summer of 1976, some rather severe design problems had been encountered relative to showing that the design met the requirements of the equipment specification using the then-procured, fully annealed 2-1.4 Cr - 1 Mo material. The basic design problem revolved around the creep characteristics of the material. An intensive engineering effort has been needed to achieve acceptable creep damage in the design.

Extensive design studies were conducted in the late summer through early winter period of 1976, culminating in a CRBRP decision in January of 1977 to proceed with fabrication of the units.

The design of the superheater upper tubesheet, continued until early Summer 1977. Specifically, high stresses were calculated for the ligament areas near the peripheral steam/water holes in the tubesheet during reactor trip-type transients. Extensive analytical studies at Atomics International, coupled closely with system design studies by the General Electric Company, resulted in the identification of the need for mitigation of reactor trip-type transients by isolation of the steam side of the superheater following such events. The selected mitigation will use the protected air-cooled condensers (PACCS) on a routine basis following the trip with complete isolation occurring approximately 10-15 minutes after reactor trip. With this mitigation, the design of the superheater upper tubesheet was shown to be acceptable.

In the period from early September 1975 through July 1978, process development in support of manufacturing of the Clinch River steam generators was conducted. Two of the more important process development activities have been in the area of (1) tube-to-tubesheet welding and (2) the development of an automated metal-inert-gas (MIG) welding system to be used when the hockeystick-shaped units cannot be rotated under a submerged arc welding head during final manufacture. Those activities were successfully completed and have been implemented into the manufacturing sequence.

Two "Few Tube Test Models (FTTM)" were constructed by AI for performance verification testing as a superheater and evaporator by General Electric. These test models reflected the CRBRP steam generator configuration in most details. Shortly after being put into test in late 1978, excessive sodium bypass flow (flow bypassing the tube bundle outside of the shroud) was observed. X-ray examination of the units indicated that the shrouds had raised off the shroud support flange, thus providing a path for bypass flow. Additionally, tube deformations were observed in the elbow region which indicated jamming in the simulated vibration suppressors.

Subsequent disassembly revealed a high susceptibility to debris contamination and galling of the 2-1/4 Cr - 1 Mo to 2-1/4 Cr - 1 Mo material couple. The previously observed jamming of the tubes in the vibration suppressors was verified and, in addition, tubes were found to be locked to several of the 2-1/4 Cr - 1 Mo tubespacers. No locking or galling was found in the two (Inconel 718) tube spacers at each end of the unit.

As a result of these observations, Inconel 718 was adopted as the material choice for all tubespacers, lead-in angles which could collect debris were eliminated, and the tube-to-tubespacer hole clearances were increased in all tubespacers. The three sets of vibration suppressors in the elbow region were replaced by one set of Inconel 718 tube support bars.

Concurrently with the FTTM examination, it was discovered that the design could not accommodate the tube movements resulting from the E-specification transients. Revisions required to correct this problem included removal of the top and bottom tubespacers.

At the time that these decisions were made, the prototype steam generator was well into the tubing phase of fabrication and it could not be completely modified to the new configuration. All elbow area modifications were incorporated. The spacer plate changes, however, were not incorporated.

All design modifications developed from these studies are being incorporated into the plant unit configuration which will have the capability to meet all specification requirements and will reduce fabrication time. The changes include inversion of the shroud assembly and supporting it on a ledge above the outlet header, incorporation of a welded component support configuration, and elimination of the outlet thermal liner. The revised configuration permits welding of both inlet and outlet headers to the shell before installing the shroud/spacer assembly.

At the present time, the prototype unit is nearing completion. It will be pressure-tested and shipped to ETEC in August of 1981 for testing in 1982. The design and analysis of the plant unit configuration is far advanced and procurement of materials needed for the changes has been initiated.

# 2.0 DESIGN AND FABRICATION REQUIREMENTS

## 2.1 PROTOTYPE

The prototype steam generator will be tested in the 70 MW Sodium Component Test Installation (SCTI) at the Energy Technology Engineering Center (ETEC) to verify thermal/hydraulic performance under part load operating conditions and to evaluate structural integrity under steady-state and limited transient operation.

The construction (material, design, fabrication, examination, testing, inspection, and certification) of the prototype steam generator is controlled by a Westinghouse Equipment Specification. The prototype steam generator is constructed to the ASME B&PV Code, Section III, Division 1 as a Class 2 vessel, using selected Code Cases and supplemental requirements from selected RDT Standards and is designed to Class 2 rules.

The prototype steam generator is designed to be used as an evaporator under SCTI normal and faulted operating conditions. The E-specification defines a number of thermal transients that must be considered during design. The prototype steam generator structural design criteria is in accordance with the elevated temperature rules of Code Case N-253, Paragraphs -3300 and -3320.

The prototype steam generator design considers a design basis leak consisting of a single tube, double-ended guillotine rupture of a steam tube followed after 0.4 of a second by rupture of the adjacent six steam tubes. The prototype is designed to resist loading derived by the method described in the Uniform Building Code considering a Zone 4 seismic occurrence with ZIKCS = 0.20. This event and the design basis leak are considered faulted events. There are no upset or emergency events.

The prototype steam generator is thermally designed so that the steam generator system can operate at steady state from 40% to 100% of rated full load. Thermal hydraulic design conditions are shown in Table I. Structural design requirements are shown in Table II.

Pressure retaining materials must meet the requirements of Code Case N-253 (Construction of Class 2 of Class 3 Components for Elevated Temperature Service). However, materials that satisfy Code Case 1592-4 (Class 1 Components in Elevated Temperature Service) supplemented by other DOE requirements are actually used for construction of the prototype.

2-1/4 Cr - 1 Mo steel is the basic material specified for the pressure boundary with Inconel 718 being used for a few specific applications such as steam head bolting. Non-code boundary materials for such components as internals must comply with ASME, ASTM, and other specifications.

The tubesheets, tubing, and forgings comply with applicable RDT material standards for the application. The tubesheets are Vacuum Arc Remelt and the tubing is Electro Slag Remelt. Cleanliness and chemical requirements for these materials are tightly controlled by the RDT Standards.

## TABLE I
## STEAM GENERATOR THERMAL HYDRAULIC DESIGN CONDITIONS

| | Rated Full Load |
|---|---|
| Duty per loop | 325 Mwt |
| Sodium flow per loop, maximum | $6.1 \times 10^6$ kg/hr ($13.5 \times 10^6$ lb/hr) |
| Steam flow rate, superheater | $0.5 \times 10^6$ kg/hr ($1.11 \times 10^6$ lb/hr) |
| Water/steam flow rate, each evaporator | $0.5 \times 10^6$ kg/hr ($1.11 \times 10^6$ lb/hr) |
| Superheater outlet steam temperature | 485°C (905°F) |
| Superheater outlet steam pressure | 10.68 MPa (1550 psia) |
| Friction pressure loss, steam drum to superheater inlet | 0.24 MPa (35 psi) |
| Friction pressure loss, evaporator to steam drum | 0.04 MPa (6.3 psi) |
| Friction pressure loss, steam drum to pump inlet | 0.04 MPa (6.7 psi) |
| Friction pressure loss, pump outlet to evaporator inlet | 0.07 MPa (10.1 psi) |
| Superheater sodium inlet pressure | 1.43 MPa (208 psia) |
| Superheater sodium inlet temperature, max. | 502°C (936°F) |
| Feedwater temperature | 242°C (468°F) |
| Na friction pressure drop, maximum | |
| Superheater | 0.42 MPa (62 psi) |
| Evaporator | 0.11 MPa (16 psi) |
| Recirculation ratio | 2:1 |
| Continuous drain rate from steam drum | $0.5 \times 10^6$ kg/hr ($0.11 \times 10^6$ lb/hr) |

## TABLE II

## PROTOTYPE STEAM GENERATOR STRUCTURAL DESIGN REQUIREMENTS

Design Pressures

| | |
|---|---|
| Sodium to Air Boundary | 2.34 MPa (340 psia) |
| Sodium to Steam/Water Boundary | 0.10 MPa (15 psia) |
| Water/Steam Inlet | 16.55 MPa (2415 psia) |
| Water/Steam Outlet | 15.61 MPa (2265 psia) |
| External | 0.10 MPa (15 psia) |

Design Temperatures

| | |
|---|---|
| Sodium Inlet Nozzle | 518°C (965°F) |
| Sodium Outlet Nozzle | 343°C (650°F) |
| Steam/Water Inlet Nozzle | 343°C (650°F) |
| Steam Outlet Nozzle | 343°C (650°F) |

Design Mechanical Loads such as nozzle loads from piping thermal expansion, deadweight, sodium/water reaction thrust loads, etc., are included in the E-specification or on an Interface Control Drawing.

Fabrication Requirements

The fabrication and examination of the prototype steam generator pressure boundary must comply with the requirements of Articles NC-4000 and NC-5000 of Subsection NC (Class 2 components). However, the fabrication and examination of the prototype has been optionally upgraded to satisfy Subsection NB (Class 1 components) as supplemented by DOE requirements, Code Cases 1593 and 1594 and other specific requirements defined in the E-specification.

The tube-to-tubesheet welding is considered the most critical fabrication activity. Additional and more restrictive weld qualification and examination requirements were imposed for these welds to assure very high reliability. Radiography of the tube-to-tubesheet welds is performed with a rod anode machine capable of defining porosity of less than 0.127 mm (0.005) inch. The acceptance criteria for porosity are much more stringent than the ASME Code requirements for such a weld.

The fabrication and examination on non-pressure boundary parts and assemblies is performed to procedures that contain requirements equivalent to those of ASME Code Section VIII, Division 1. Instrumentation is temporarily or permanently attached to the pressure boundary in accordance with Code Cases N-226 and N-252, respectively.

## 2.2 PLANT UNITS

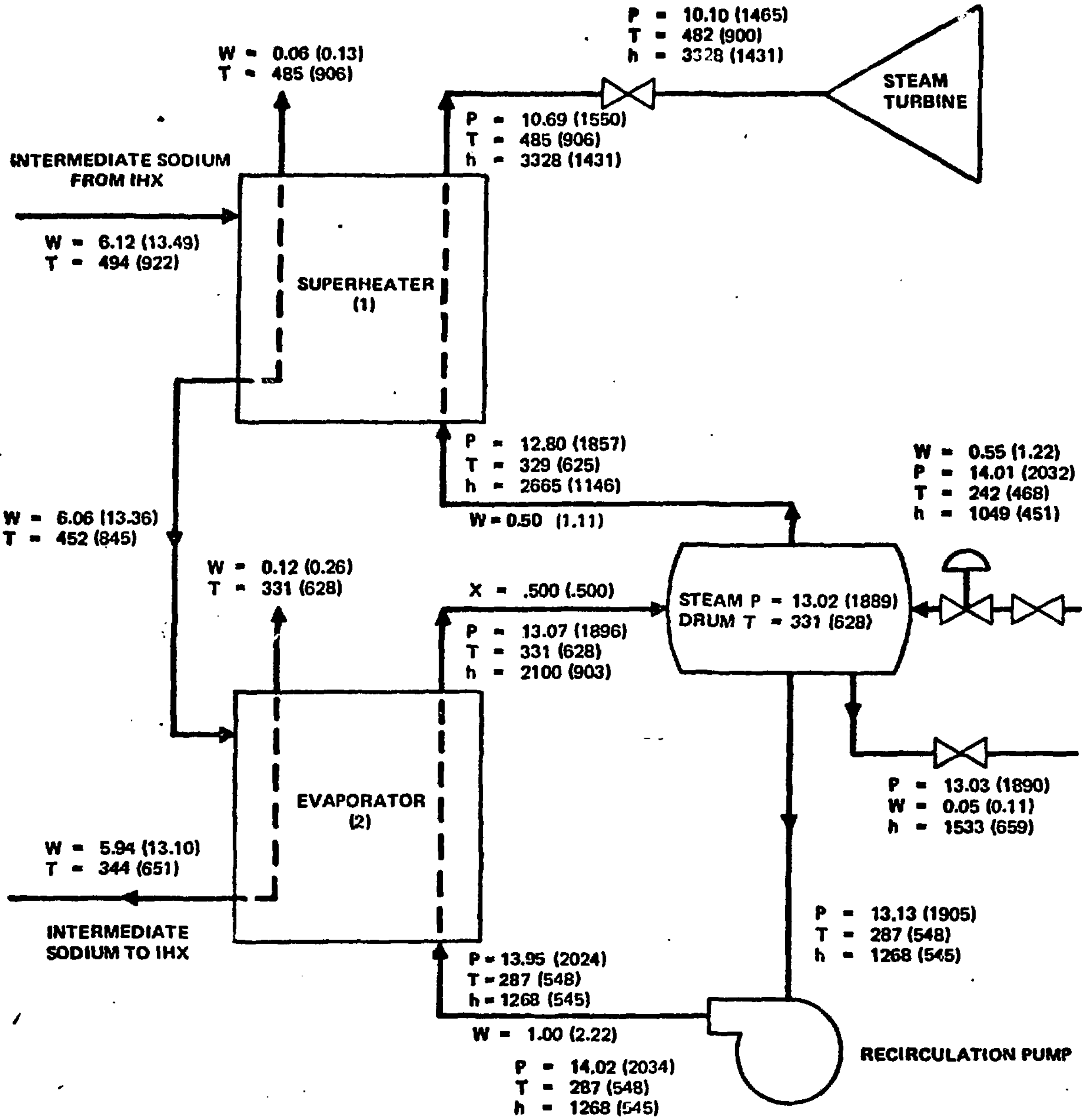
Steam generators will be installed in the three intermediate sodium heat transfer loops of the CRBRP. Two evaporators, in parallel, and one superheater will be used in each loop. Their purpose is to convert the energy (transferred from the reactor by the sodium) to water to form superheated steam for delivery to the steam turbine. The steam generator heat balance is shown on Figure 1.

The construction (material, design, fabrication, examination, testing, inspection, and certification) of the plant unit steam generators is controlled by a Westinghouse equipment specification.

The plant steam generator units are being designed and constructed to the ASME BPV Code, Section III, Division 1 as Class 1 vessels, selected Code Cases, and supplemental requirements from selected RDT Standards.

The steam generator modules are designed to be used as either evaporators or superheaters under normal, upset, emergency, and faulted operating conditions derived from plant and system operation. The E-specification defines a load histogram of transients categorized as normal, upset, and emergency events to show the assumed history of the occurrence of events throughout the service life of 30 years.

The steam generator structural design criteria are in accordance with the elevated temperature design rules of Code Case 1592-4 supplemented by DOE requirements. The steam generator design considers a design basis leak consisting of a single tube, double-ended guillotine rupture of a steam tube




| | | |
|---|---|---|
| W = kg/hr x $10^{-6}$ (lbm/hr x $10^{-6}$)<br>T = °C (°F) | h = J/g (Btu/lbm)<br>P = MPa (psia) | x = STEAM QUALITY |

| ASSUMPTIONS | EVAPORATOR | SUPERHEATER |
|---|---|---|
| TUBE SIDE SURFACE | FOULED | CLEAN |
| HEAT TRANSFER COEFF. | LOWER LIMIT | NOMINAL |
| PLUGGING ALLOWANCE | 5% | 5% |

FIGURE 1. CRBR STEAM GENERATOR HEAT BALANCE

followed after 0.4 second by rupture of six adjacent steam tubes. The unit in which the event occurs is considered to experience a faulted event and the other units in the loop to experience an emergency event.

The reactor plant is designed for ground accelerations of 0.25 g for Safe Shutdown Earthquake (SSE) and 0.125 g for Operating Basis Earthquake (OBE). The steam generator is designed for the combination of an SSE event with a water/steam pipe break and an OBE event in combination with the most severe thermal upset transient.

The steam generator is designed so that the steam generator system can operate at steady state from 40% to 100% of rated full load. Thermal hydraulic design conditions are shown in Table I. Structural design requirements are shown in Table III.

## TABLE III

## PLANT UNIT STEAM GENERATOR STRUCTURAL DESIGN REQUIREMENTS

Design Pressures

| | |
|---|---|
| Sodium to Air Boundary | 2.34 MPa (340 psia) |
| Sodium to Steam/Water Boundary | 0.10 MPa (15 psia) |
| Evaporator Water Inlet | 16.65 MPa (2415 psia) |
| Evaporator Water/Steam Outlet | 15.27 MPa (2215 psia) |
| Superheater Steam Inlet | 15.27 MPa (2215 psia) |
| Superheater Steam Outlet | 13.20 MPa (1915 psia) |

Design Temperatures

| | |
|---|---|
| Superheater Sodium Inlet | 518°C (965°F) |
| Superheater Sodium Outlet | 474°C (885°F) |
| Evaporator Sodium Inlet | 474°C (885°F) |
| Evaporator Sodium Outlet | 413°C (775°F) |
| Evaporator Water Inlet | 343°C (650°F) |
| Evaporator Water/Steam Outlet | 343°C (650°F) |
| Superheater Steam Inlet | 343°C (650°F) |
| Superheater Steam Outlet | 504°C (940°F) |

Design Mechanical Loads such as nozzle loads from piping thermal expansion, deadweight, pipe break thrust loads, sodium/water reaction thrust loads, etc., are included in the E-specification or on an Interface Control Drawing.

## 3.0 DESIGN DESCRIPTION

### 3.1 PROTOTYPE STEAM GENERATOR

The prototype steam generator, shown in Figure 2, is approximately 18.9 m (62 foot) long in its longest dimension with an offset "hockeystick" type end approximately 3.0 m (10 foot) long. The 1.23 m (49 inch) ID pressure boundary shell contains 757 15.9 mm (5/8 inch) OD x 2.8 mm (0.109 inch) wall steam tubes. The tubes are butt-welded to machined bosses on the sodium side of both upper and lower tubesheets using an internal bore, autogeneous single-pass weld.

The unit weighs approximately 94.5 metric tons (104 tons). The main shell is formed from 5.7 cm (2.25 inch) thick plate rolled and welded in five courses to form the main shell. The main support ring, machined from a ring-forged section, is used for the primary support for the unit in conjunction with a conical support cone between the main support ring and the building floor. The 10.2 cm (4.00 inch) thick inlet and outlet headers are formed in semicircular halves with integral nozzle pullout sections. One sodium inlet and two sodium outlets are formed on each unit. The prototype unit will be tested as an evaporator so only one sodium outlet nozzle will be used.

The elbow sections are also formed from plate material. The three 30° sections are formed in halves and seam-welded. Reducers are employed for the juncture between the tubesheets and the main shell sections. The final closure weld on the unit is made at the large V-groove weld located at the juncture of the larger and smaller diameter shell sections.

The internal structure of the unit consists primarily of the shroud subassembly which is supported from an internal ledge on the main support ring. The shroud subassembly serves to position the 21 tube spacers, located at varying positions inside the shroud, which form the lattice array for positioning of the steam tubes within the 14 m (46 foot) active length of the unit. Of the 21 tube spacers, four are fabricated from Inconel 718 material (specifically, the two upper most and two lowermost close-coupled tube spacers) with the remaining 17 tube spacers being fabricated from 2-1/4 - 1 Mo plate material. All tube spacers have an "hourglass" shaped tube hole configuration with an 0.20-0.43 mm (0.008-0.017 inch) diametrical clearance with the steam tubes.

Design conditions which are applied to the unit require heavy shell sections. To shield the heavy shell sections from sodium side thermal transients, extensive use of thermal baffling in the inlet header/nozzle region is employed. In addition, both tubesheets are extensively baffled and a close-fitting full diameter shroud is placed around the steam tube bundle in the elbow region.

A tube support assembly using Inconel 718 slat-type bars between tubes provides support for the tubes in the "hockeystick" region preventing excessive lateral motion of the tube bundle during seismic events. This assembly is fastened to the ID of the elbow shroud at its tubesheet end.

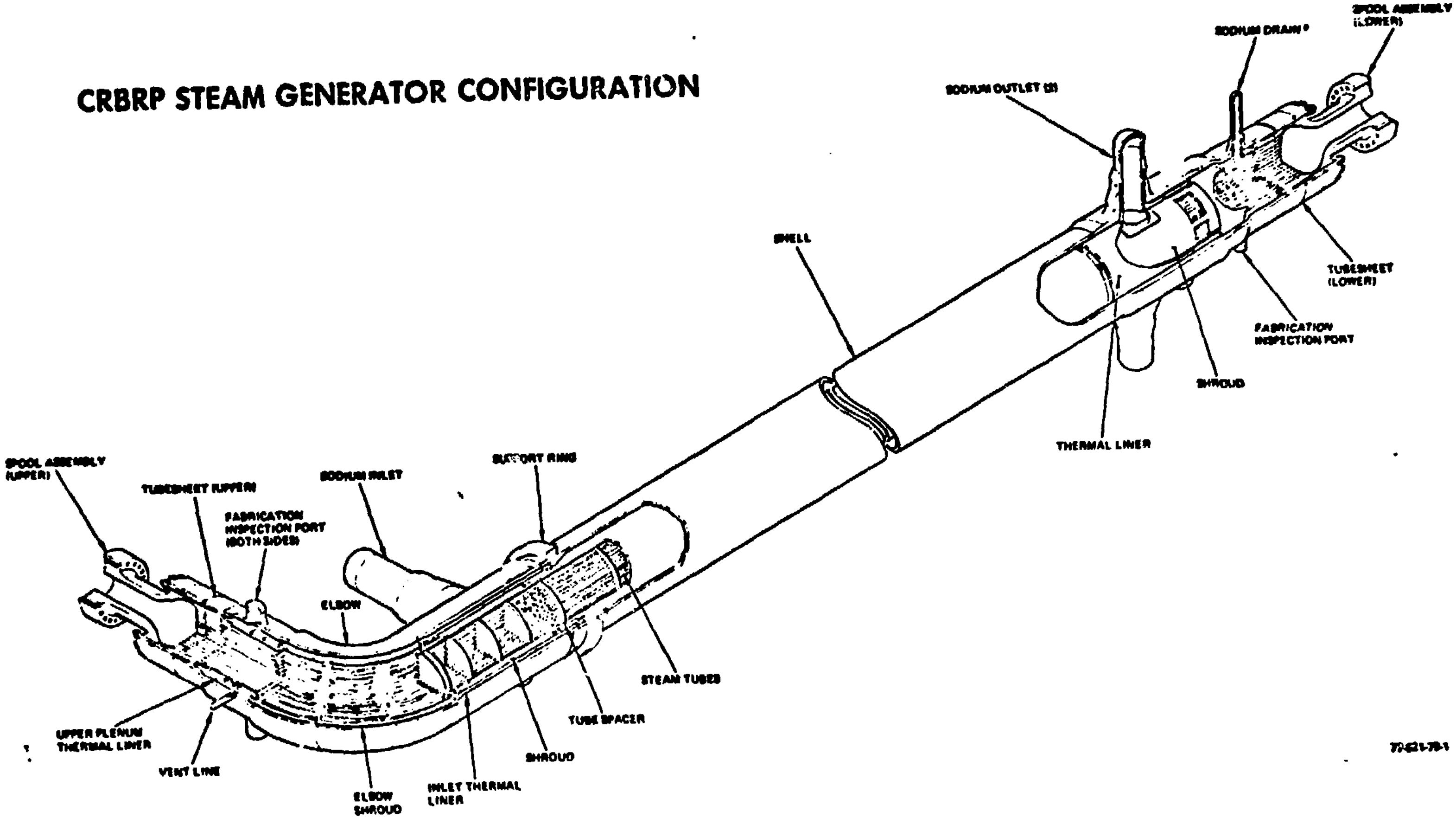



Figure 2. Prototype Steam Generator Module Cutaway

Removable spool assemblies are bolted to the upper and lower tubesheets of the unit. The spool assemblies are attached to the tubesheet using Inconel 718 studs and nuts. A spiral-wound gasket is employed at the joint between the tubesheet and spool assembly. These removable spool assemblies facilitate access to the water side face of the tubesheets for inservice inspection and/or tube plugging, should it be required during operation.

The main structural support for the unit is a cone which bolts to the units main support ring and is attached to the building. After installation of the unit, a lower restraint structure reacts out applied nozzle loadings and provides snubbing of the lower end of the unit during an earthquake event.

## 3.2 PLANT UNIT STEAM GENERATORS

The plant unit steam generators are identical to the prototype unit except for an integral cone-type component support and redesign of the internals in the following areas, as shown in Figure 3:

1) The number of tubespacers has been reduced from 21 to 19, eliminating one of the two close-coupled tubespacers at each end. The material of all tubespacers is Inconel 718. The tube hole configuration has been revised with the last two tubespacers at either end having a cone-shaped profile (large diameter down) and the remainder of the tubespacers having a cylindrical diameter. The diametrical clearance for all tube holes has been increased to .76 mm (.030 inch).

2) The shroud has been shortened and the window bars eliminated.

3) The outlet thermal liner assembly, shroud extension, and outlet and drain nozzle liners and extensions have been removed.

4) The shroud support location has been lowered from the main support ring elevation to a point just above the outlet header/main shell interface.

5) The inlet thermal liner assembly has been completely redesigned to move its support location from the shroud. The liner has been lengthened to extend below the main support ring/inlet header shell weld.

6) Piston ring seals on the OD of the inlet thermal liner above and below the inlet nozzle protect the nozzle region during thermal transients.

7) The inlet nozzle liner seal has been revised to reduce leakage into the inlet header nozzle region.

The only external change is the elimination of the bolted on component support. This one-piece cone and cylindrical component support is made integral with the main shell by a weld to a cone-shaped protrusion on the OD of the main shell.

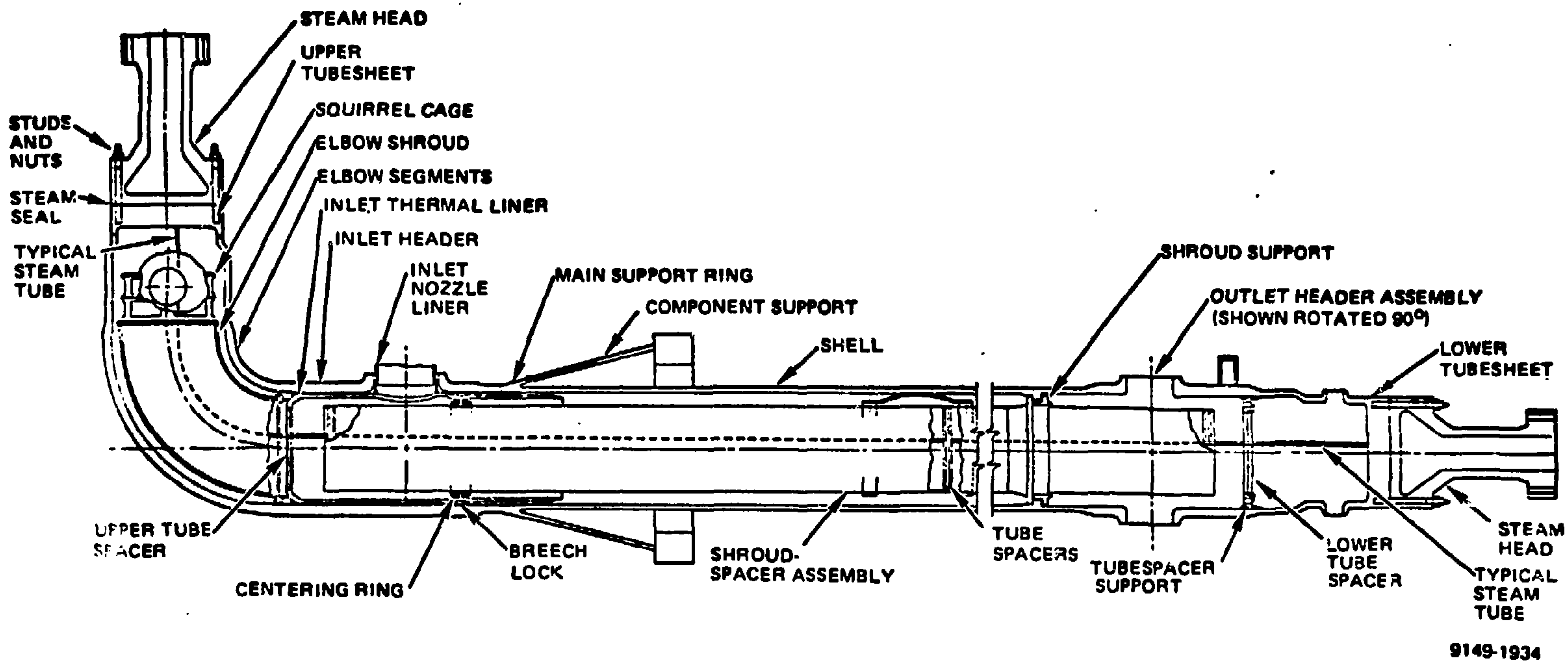



Figure 3. Plant Unit Steam Generator Configuration

# 4.0 SUPPORTING DEVELOPMENT TESTS AND RESULTS

## 4.1 FEW-TUBE TEST MODELS

Two few-tube test models were constructed by Rockwell International and tested as an evaporator and superheater combination by General Electric at San Jose, California. Excessive sodium flow between the shell and shroud was detected in both units soon after initiation of the test program. Testing was terminated and the models were destructively examined. The examination of the units yielded the following conclusions:

1) Shrouds were raised off their support rings and the holddown bolts were broken in both modules.

2) Shroud centering bolts were found to be tight against the shell. Sufficient clearance had not been provided during fabrication to permit free expansion of the shroud during transients.

3) Bent and buckled tubes were observed in the superheater.

4) Severe gouging was found on the vibration suppressors and tubes, and at some spacer locations, in the superheater; less severe damage was found at similar locations of the evaporator.

5) The vibration suppressors were misaligned, relative to each other and the plane of the tubes.

6) Thermocouple mounting foil bands were found wedged between the tubes and vibration suppressors.

7) Inconel 718 spacers did not exhibit the tendency for adhesive wear or galling that was noted for the 2-1/4 Cr - 1 Mo spacers.

8) Particulates were found in both units.

Figure 4 pictorially shows several of the findings.

As a result of these findings, the tube spacer plate material for the CRBR steam generator was changed to Inconel 718, tube spacer clearances were increased, the number sets of vibration suppressors was reduced and the tube support bar to tube clearances were increased.

## 4.2 THERMAL STRIPING

Two areas were identified in the steam generator with the potential to experience thermal striping. These are the steam tubes and shrouds in the stagnant regions (1) above the uppermost tube spacers in the elbow region of the evaporator and (2) below the bottom tube spacer in the lower stagnant region of the superheater. These two zones have a striping potential of 104°F (220°F) and 101°C (214°F), respectively.

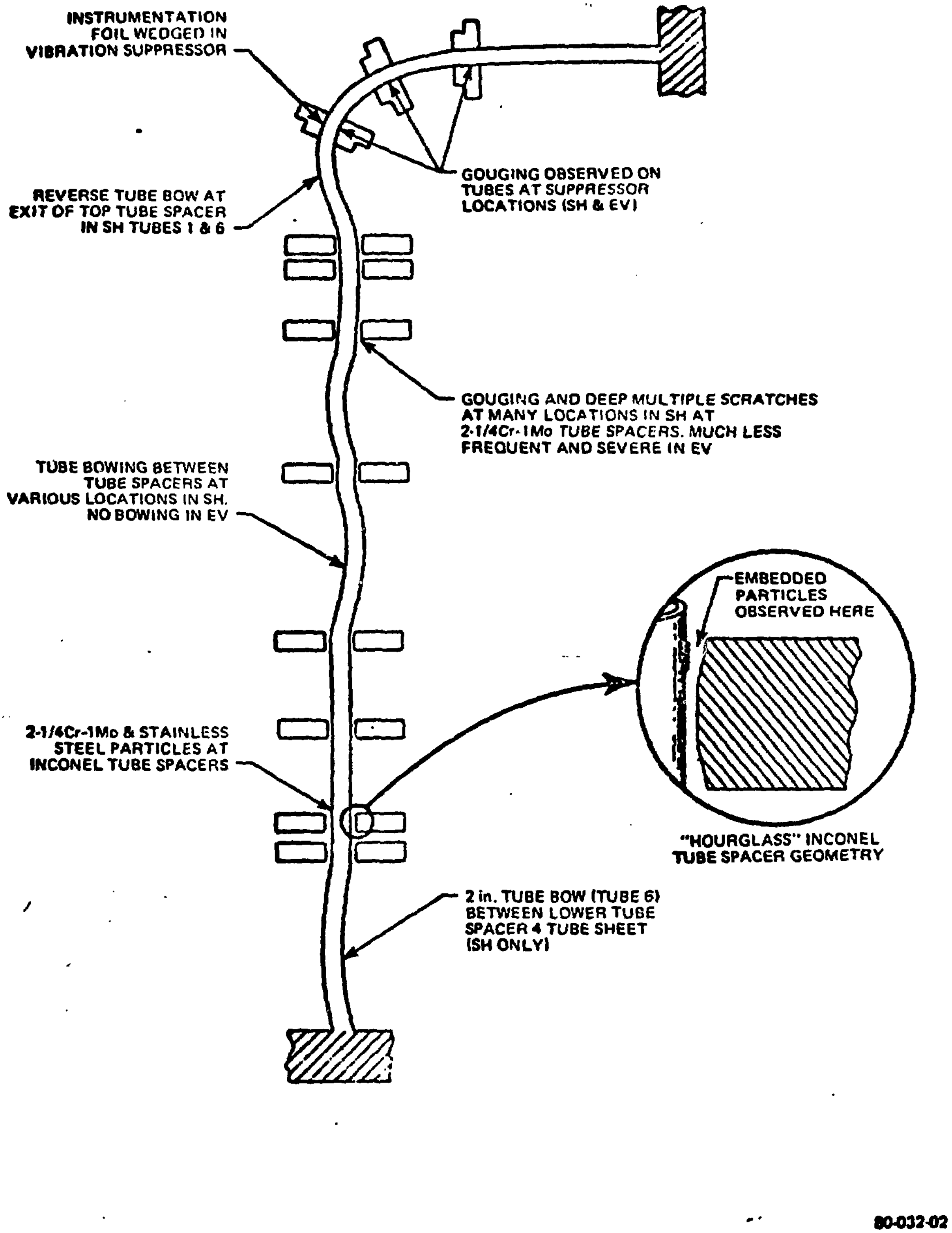



**Figure 4** EVAPORATOR AND SUPERHEATER FTTM POST-TEST EXAMINATION RESULTS

Scoping thermal striping tests were performed by Westinghouse Advanced Reactors Division (WARD) in the early part of 1980. Two 1/6-scale plexiglass models were constructed to simulate the lower end of the superheater (Figure 5) and the upper region of the evaporator (Figure 6). Water was used as a working fluid. Modelling criteria was established to satisfy hydraulic requirements, heat transfer simulation, and the flow capability of the available facility 2.8 l/second (45 gallon/minute). The heat transfer from the tubes to the stagnant sodium was simulated by injective cold water into the stagnant flow regions.

Figures 7 and 8 show temperature plots of the test results for the evaporator upper region and the superheater lower region, respectively. Data presented in these figures represent test data extrapolated to the evaporator and superheater operating temperatures. The calculated fatigue damage from these temperature oscillations was within the allowable limits.

## 4.3 INLET NOZZLE LINER AND THERMAL LINER LEAKAGES

The design of the inlet nozzle region incorporates a liner along the nozzle wall and another liner along the header. The original configuration is shown in Figure 9. The purpose of these liners is to provide thermal isolation between the incoming sodium and the nozzle wall and header during severe transients.

Due to fabrication and alignment tolerances, some bypass sodium could penetrate into the annulus between the liners and nozzle wall/header. This condition partially negates the effectiveness of the liners as a thermal barrier. Bypass flow direction is critical. If the flow were to penetrate directly into the nozzle liner cavity, it could increase the radial temperature difference across the nozzle knee to an unacceptable level during a transient. Therefore, the design must insure that bypass sodium flows upward in order that the sodium temperature reaching the knee is mitigated by the heat capacity of the structure.

For this reason, the design was modified by creating a step in the nozzle liner as shown in the sketch of Figure 10. The purpose of the step is to provide a lower static pressure in the nozzle than in the bottom of the annulus below the thermal liner. (Points A and B, respectively, in Figure 10.)

Due to the complex geometry of this region, hydraulic analyses were not considered accruate enough to predict the static pressures in the noted areas. In order to verify that the proposed design would indeed cause a lower static pressure at the nozzle, WARD performed scoping hydraulic tests. The water scale model included a variable height step on the inside of the nozzle (see Figure 6), similar to the proposed design. The test results indicated that a circumferential pressure gradient exists in the nozzle opening behind the step. This is caused by the unsymmetrical sodium flow distribution as it emerges from the nozzle and flows upward into the annulus between the shroud and header. Since this pressure gradient could induce undesirable recirculation behind the nozzle liner, a sealing device has been incorporated in the design which will prevent such a condition.

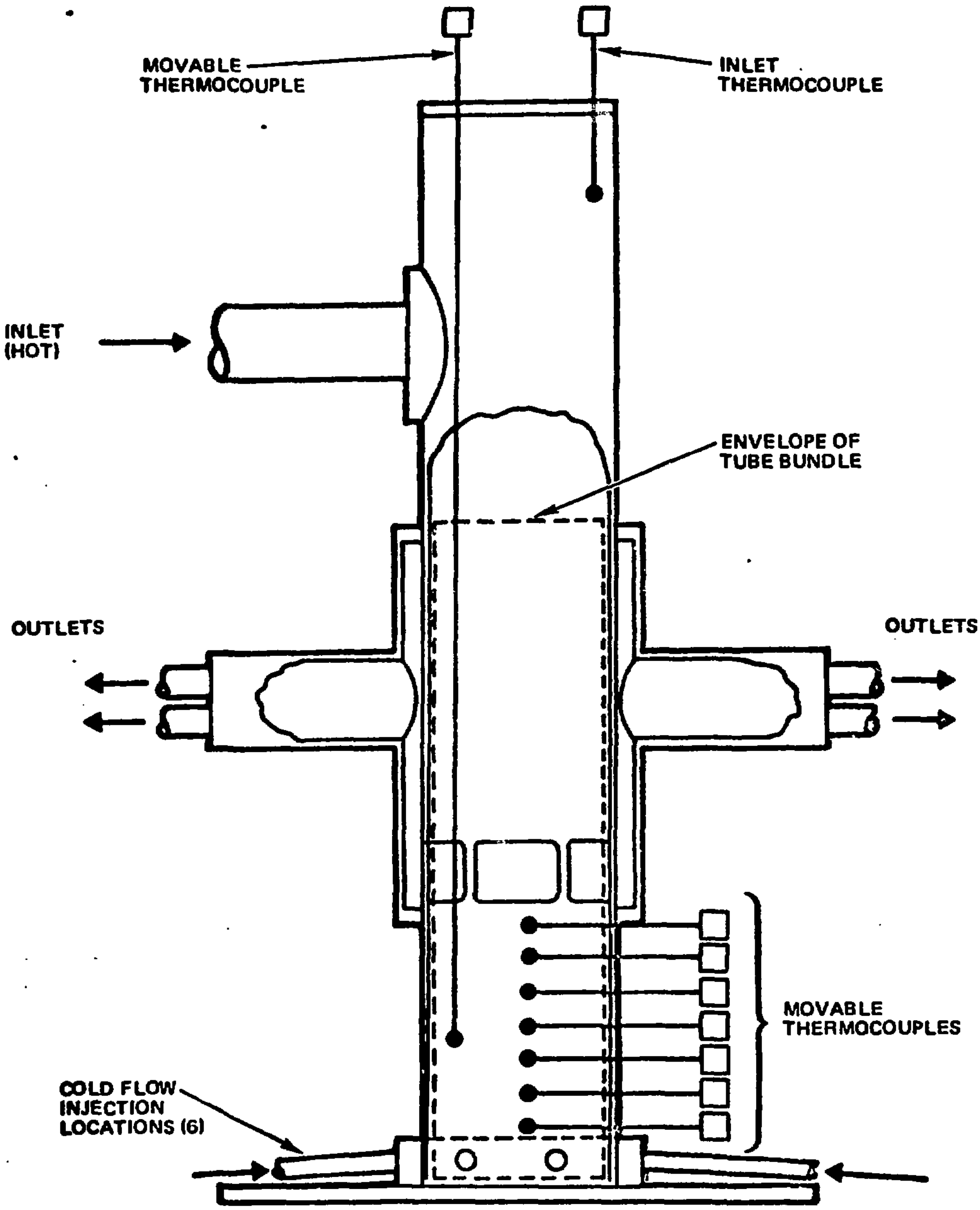



FIGURE 5. HYDRAULIC MODEL OF LOWER STAGNANT REGION

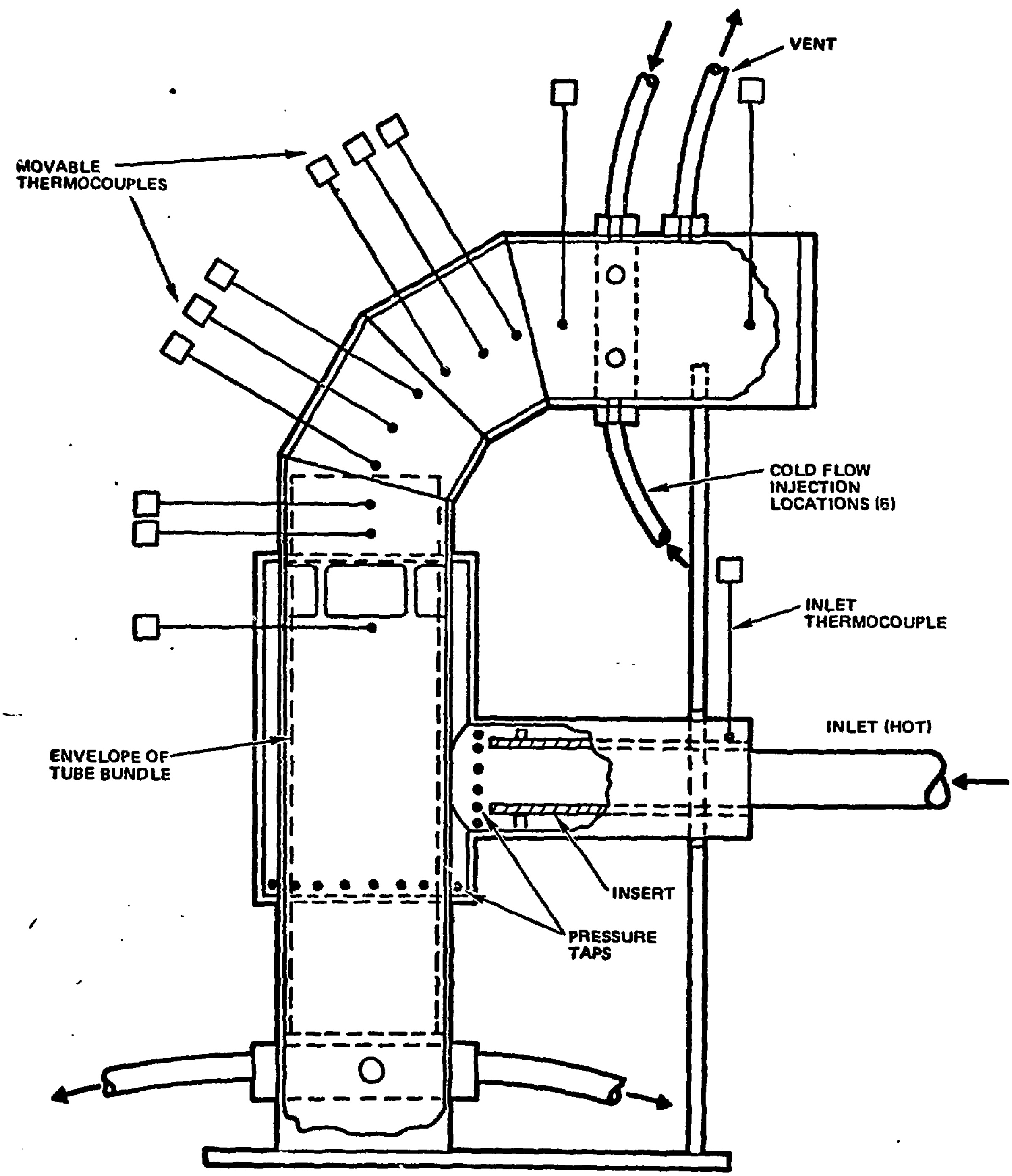



FIGURE 6. HYDRAULIC MODEL OF ELBOW REGION

NOTE: TEST TOTAL ADJUSTED POTENTIAL ~20°F (11.1°C) EXTRAPOLATED TO 220°F (122°C) FOR INPUT TO CRBRP SG ANALYTICAL MODEL

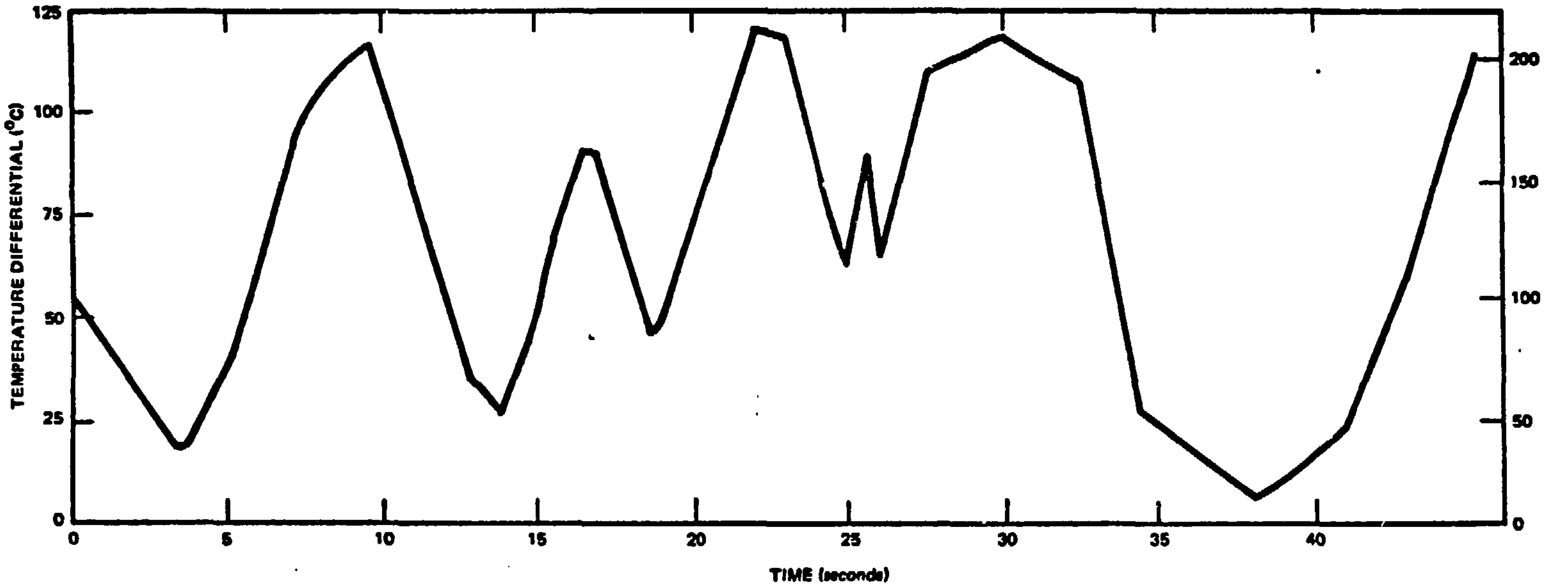



FIGURE 7. ELBOW REGION HYDRAULIC TEST DATA EXTRAPOLATED TO CRBR T POTENTIAL

NOTE: TEST TOTAL POTENTIAL ~35°F (19.4°C) EXTRAPOLATED TO 214°F (119°C) FOR INPUT TO CRBRP SG ANALYTICAL MODEL

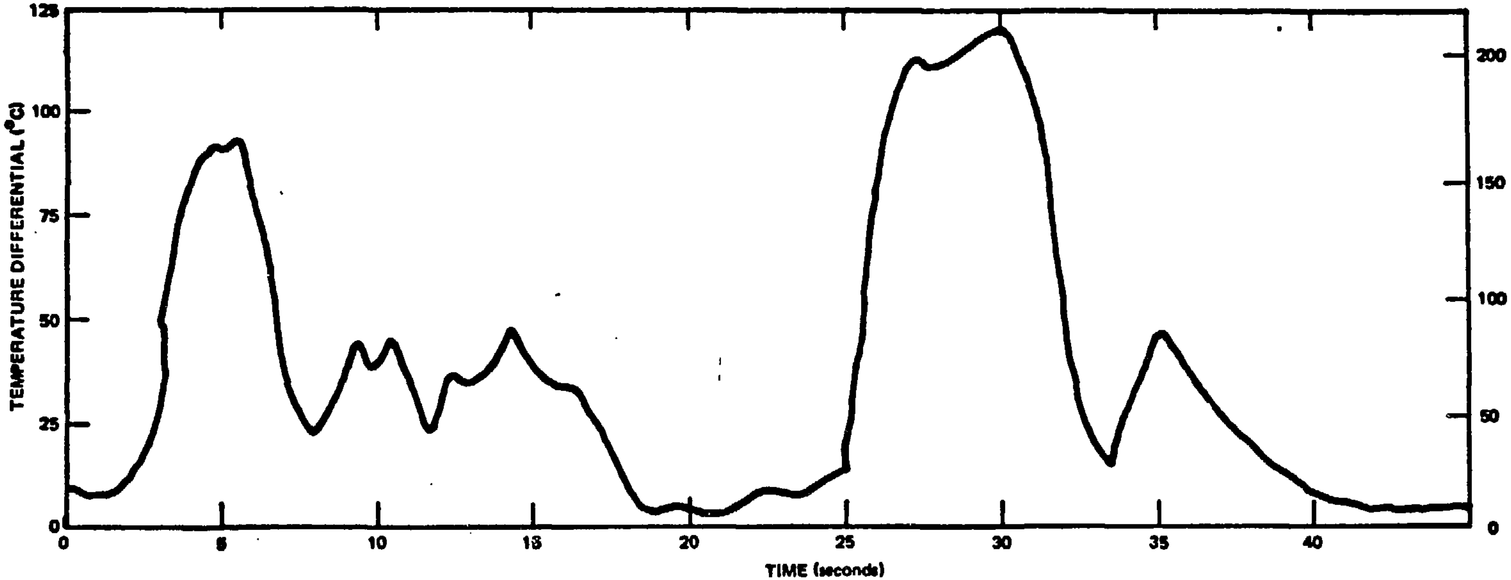



FIGURE 8. LOWER REGION HYDRAULIC TEST DATA EXTRAPOLATED TO CRBR T POTENTIAL

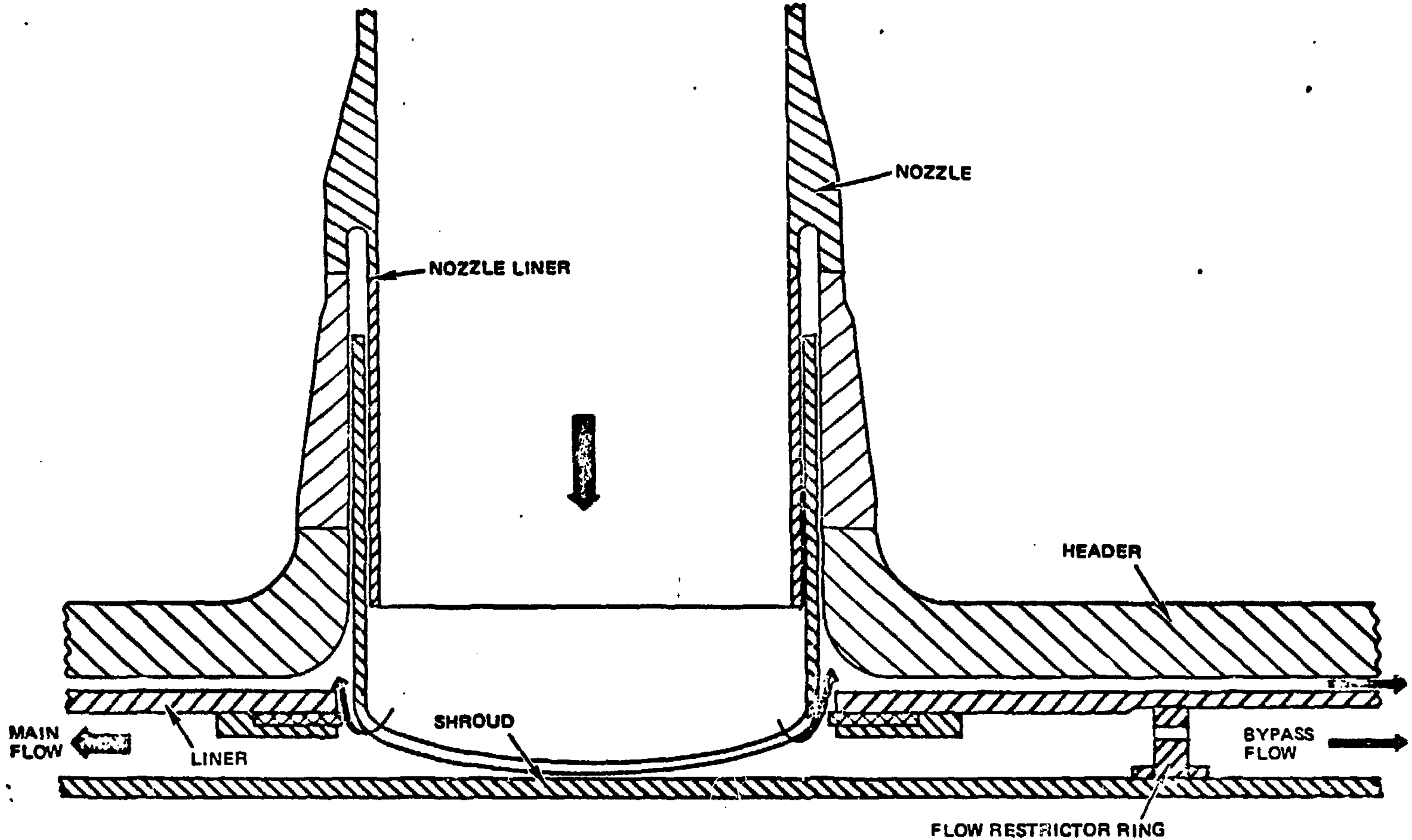



FIGURE 9. ORIGINAL CONFIGURATION OF INLET NOZZLE REGION

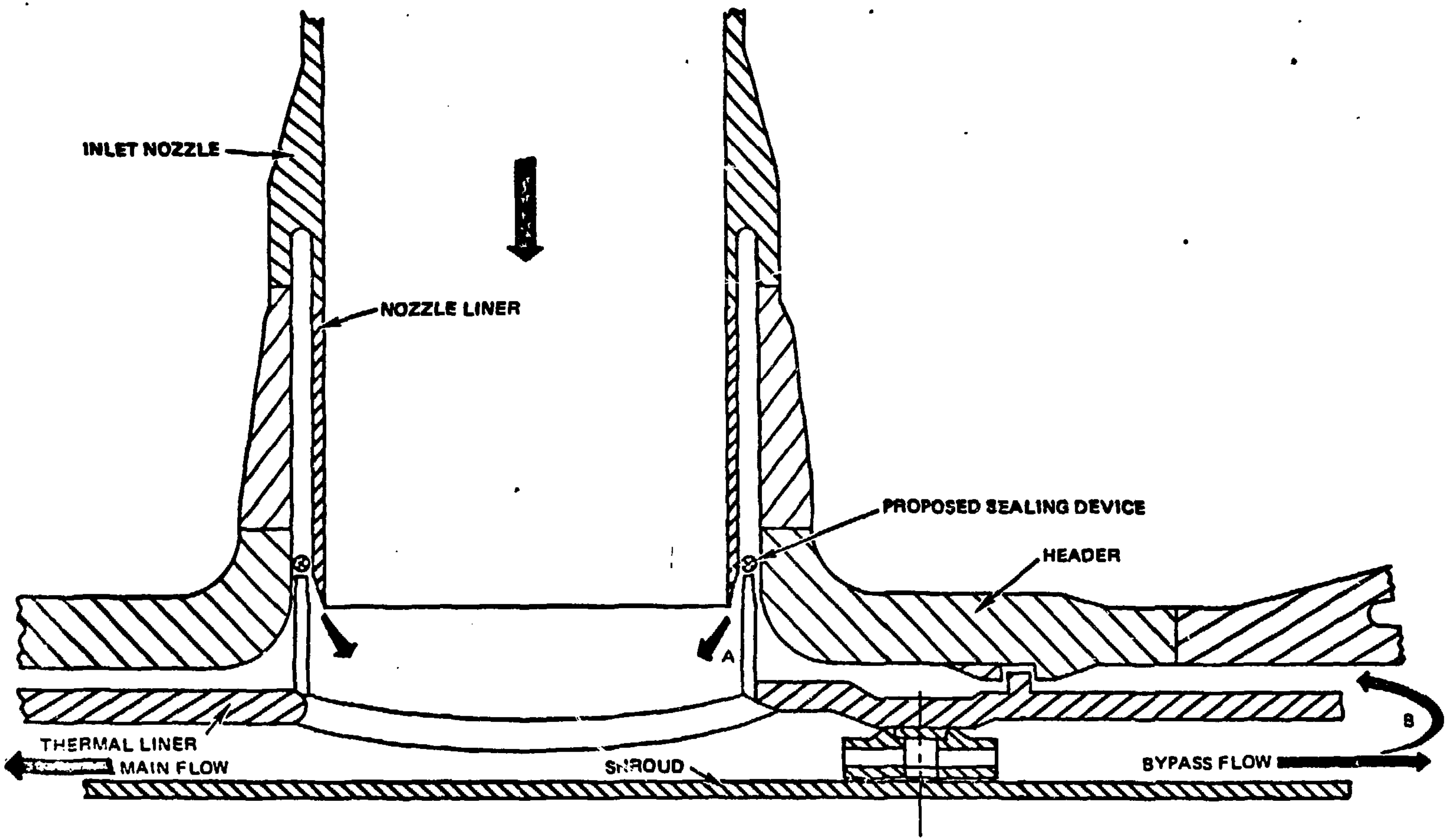



FIGURE 10. REVISED CONFIGURATION OF INLET NOZZLE REGION

## 4.4 ORIFICE STABILITY TESTS

A series of dynamic stability tests were performed at the Argonne National Laboratory (ANL) during the fall of 1980. The tests were carried out in the ANL single-tube steam generator facility (SGTF). The purpose of these tests was to obtain data to determine the orificing needed in the evaporator to assure its dynamic stability. Figure 11 shows the arrangement of the SGTF.

The test matrix includes comparison of the behavior of tubes without orifices to tubes with orifices having a 10 mm (0.41 inch) diameter hole. The following conditions were evaluated:

1) Simulated plant natural circulation during which sodium and water flow rates are very low

2) 40% power simulation

3) 100% power simulation

4) Prototype test simulation

The data collected from the tests are being evaluated.

## 4.5 SEAL (SPIRAL-WOUND GASKET) RECOVERY TEST

Spiral-wound gaskets are the reference design for sealing the upper and lower tubesheet to the steamheads in the CRBRP steam generator modules. The seal assembly consists of an Inconel Alloy 718 spiral-wound ribbon with a grafoil filler material mounted between inner and outer Inconel Alloy 718 stop rings.

Room temperature seal development tests are planned to measure seal leakage rate as a function of seal unloading deflection. The tests will be accomplished on subscale seals manufactured with gasket-winding tension which demonstrates load-deflection characteristics similar to that of the full-size seals.

## 4.6 ELBOW FLOW-INDUCED VIBRATION TEST

The observation of tubes jammed in the vibration suppressors of the FTTM (Section 4.1) and the possibility of two-point binding of tubes in the top two closely-coupled spacers led to a redesign of the internal structure of the steam generator in the region of the sodium inlet and the elbow. Major features of the redesign include elimination of the vibration suppressors at 30° and 60° locations of the elbow, replacement of the 90° vibration suppressor with a set of Inconel 718 tube supports having much wider clearances, and replacement of the top two tube spacers with a single spacer, also having larger clearances. Although the new design appears, by analysis, to have a low probability of damage by fatigue, a hydraulic test is contemplated.

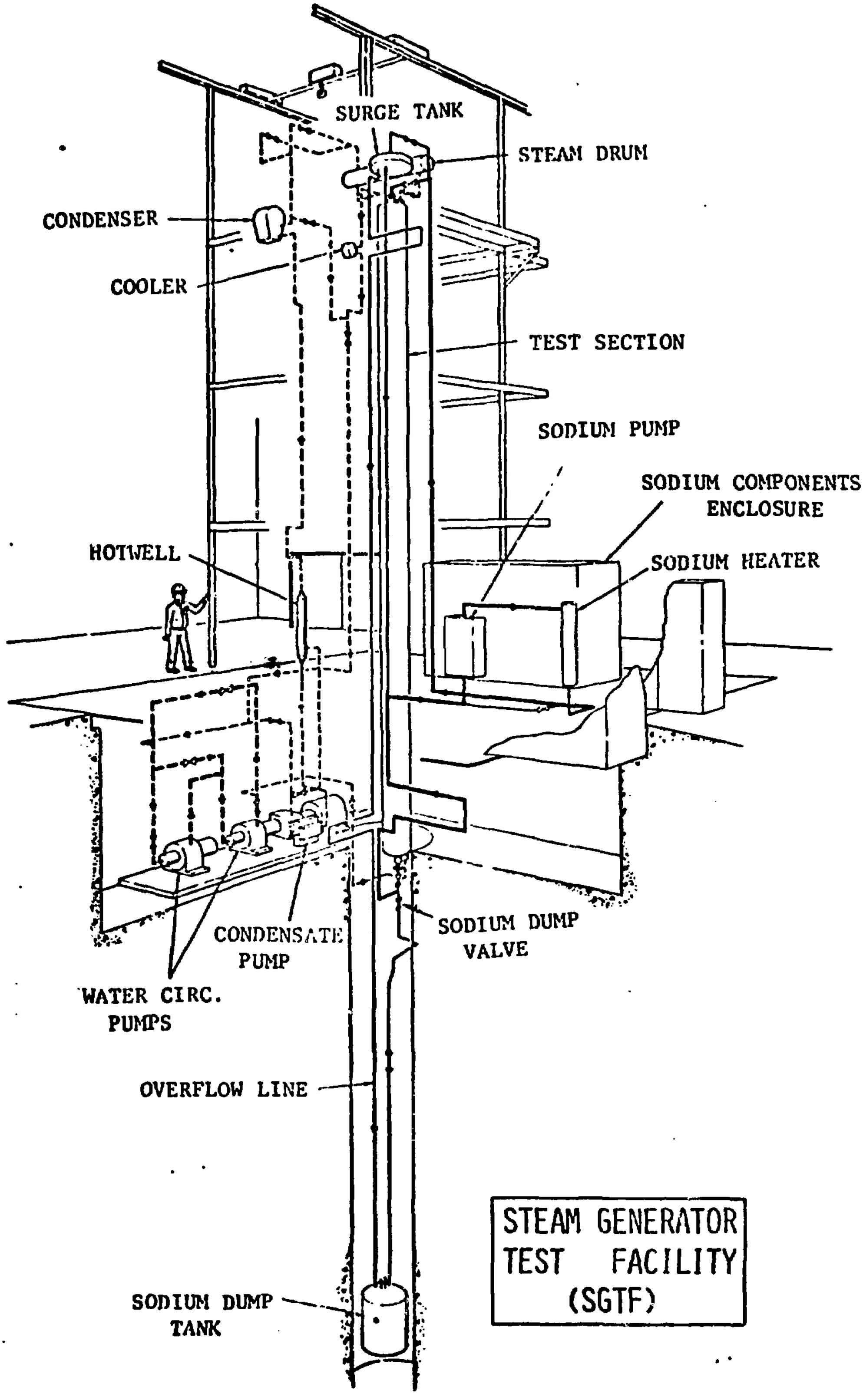



FIGURE 11. STEAM GENERATOR TEST FACILITY (SGTF)

## 4.7 MATERIAL PROPERTIES

A great deal of progress has been made over the past 4 years in obtaining an improved material model for 2-1/4 Cr - 1 Mo. The progress made manifested itself in improved understanding of inelastic behavior of 2-1/4 Cr - 1 Mo, vastly improved material behavior models for creep and plasticity, and a significant expansion in the state of the art and knowledge relating not only to 2-1/4 Cr - 1 Mo but other materials as well. A summary of the major improvements in the material behavior characterization follows:

1) A two-stage, ultimate stress-dependent creep equation has been adopted to give the best possible representation of the available creep data. The creep law allowed a very close matching of the creep behavior of specific heats and, in general, more accurate creep representation.

2) A strong dependence of creep hardening on plastic strains was discovered for 2-1/4 Cr - 1 Mo and incorporated into the material model.

3) It was discovered that the kinematic hardening model for plasticity gives erroneous trends in stress-strain behavior under mixed loading involving creep and plasticity. A modification consisting of introducing a dependence of plastic hardening on creep strains (see Figure 12) eliminate the obvious errors.

4) Subsequent use of the modified hardening rules uncovered other problems and after extensive studies and additional testing, led to the adoption of an a reset hardening model for 2-1/4 Cr - 1 Mo. This model accounts for observed time-dependent hardening recovery at temperature (see Figure 13).

In addition to the above material model related changes, two other material property issues impact steam generator design analysis.

1) The creep fatigue interaction diagram, although it was specified in the design specification, has been under extensive investigation by ORNL, ESG, and ASME code over the past 4 years. One of the major difficulties with attempts to characterize creep-fatigue damage interaction is the obvious presence of other damage mechanisms. Environment is probably the major of these secondary mechanisms and may be as important or more important than creep damage under certain loadings. Although a preliminary code approval has been obtained for a linear damage diagram with environmental correction factors for fatigue, this issue is not fully resolved. Fortunately, in the steam generator design, creep damage mechanisms dominates.

2) The impact of PWHT, thermal aging, and decarburization on mechanical properties of 2-1/4 Cr - 1 Mo have been under study for the past 4 years. The data shows some reduction in the strength properties.

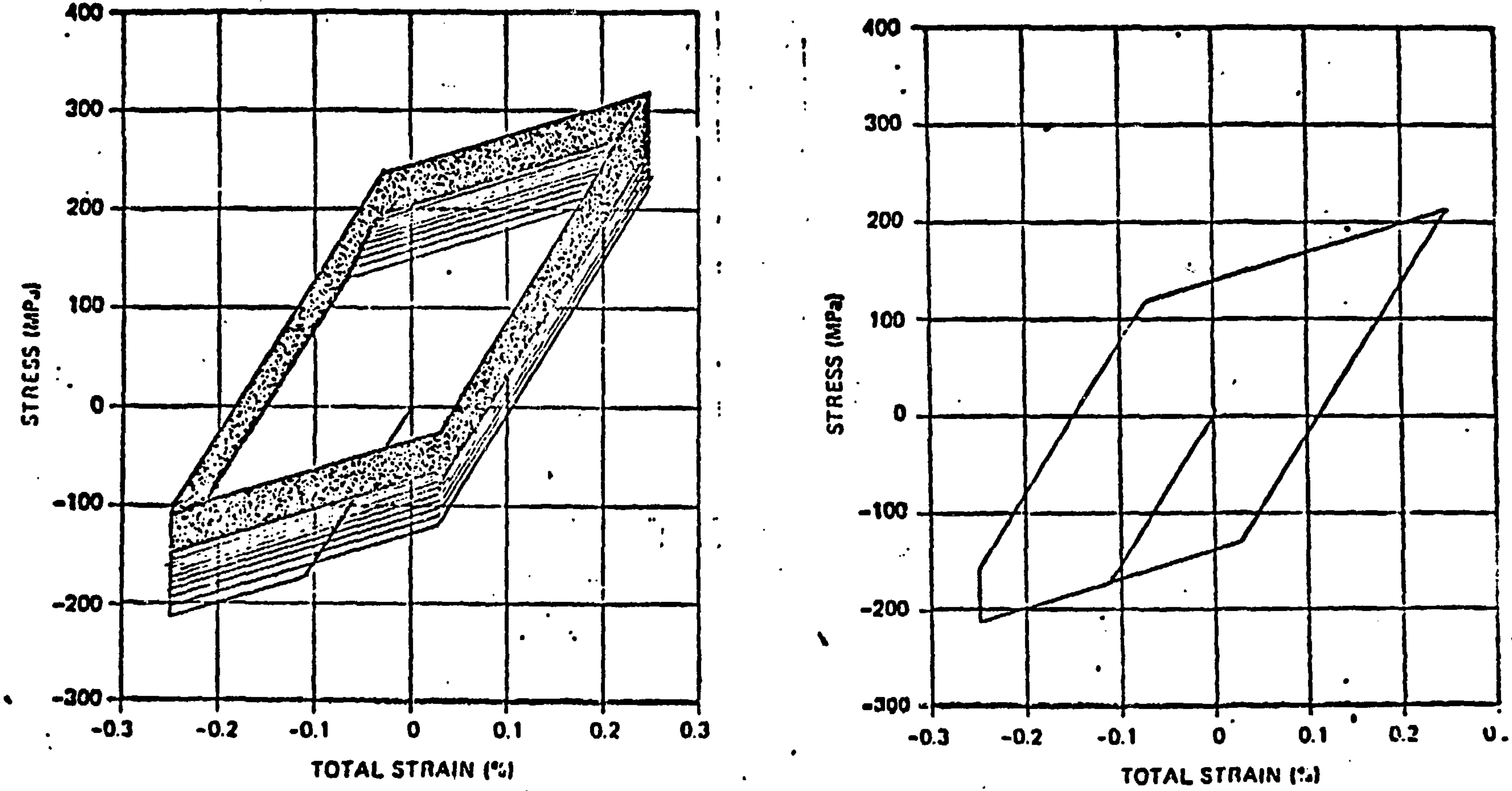



FIGURE 12

STRESS-STRAIN BEHAVIOR BEFORE AND AFTER INCORPORATION OF PLASTIC HARDENING DEPENDENCE ON CREEP STRAINS

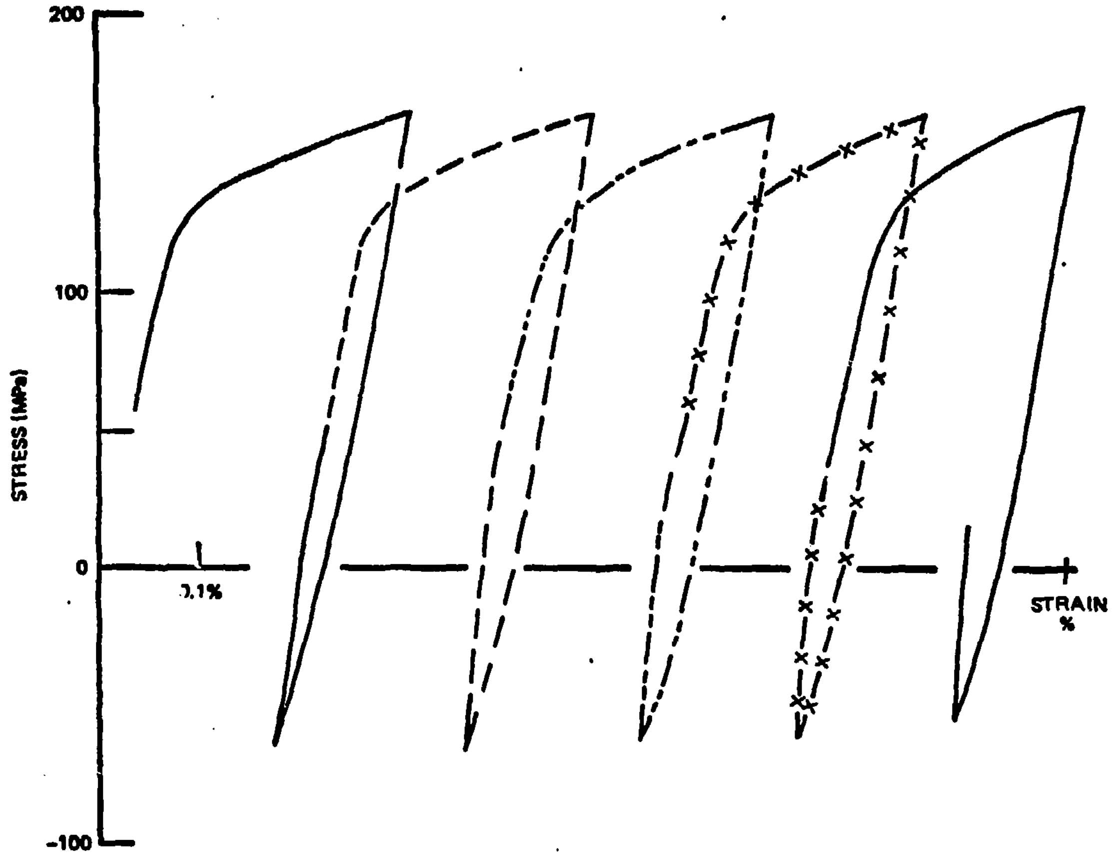



FIGURE 13

OBSERVED STRESS-STRAIN BEHAVIOR IN A TWO-BAR EXPERIMENT LEADING TO FULL α-RESET HARDENING RECOVERY MODEL

## 5.0 STATUS OF MANUFACTURING

Installation and welding of all 757 tubes into the prototype unit was completed on May 21, 1980. A photograph of the completed tube bundle, viewed from the upper end of the unit, is shown in Figure 14. The inspection shown in Figure 15 is examining the fit-up of a tube to a tubesheet stub prior to welding. A completed tube-to-tubesheet weld is visible at the top of the array of stubs. All the tube-to-tubesheet welds have passed rod anode x-ray, ultrasonic weld shape measurement, helium leak check, and liquid penetrant inspections, except for three tubes which have welds which are not completely satisfactory, and therefore, will be plugged at some convenient time prior to the final pressure test of the unit. The rod-anode x-ray machine is shown in Figure 16. Progress in achieving low defect rates was quite rapid initially and continued through prototype tube welding operations. Many of the defects were minor, such as incomplete penetration in a localized area of the weld. These were quickly and easily repaired by manual welding. Major defects, such as gross porosity, which necessitates removal and replacement of a tube, were reduced to about 1.5%.

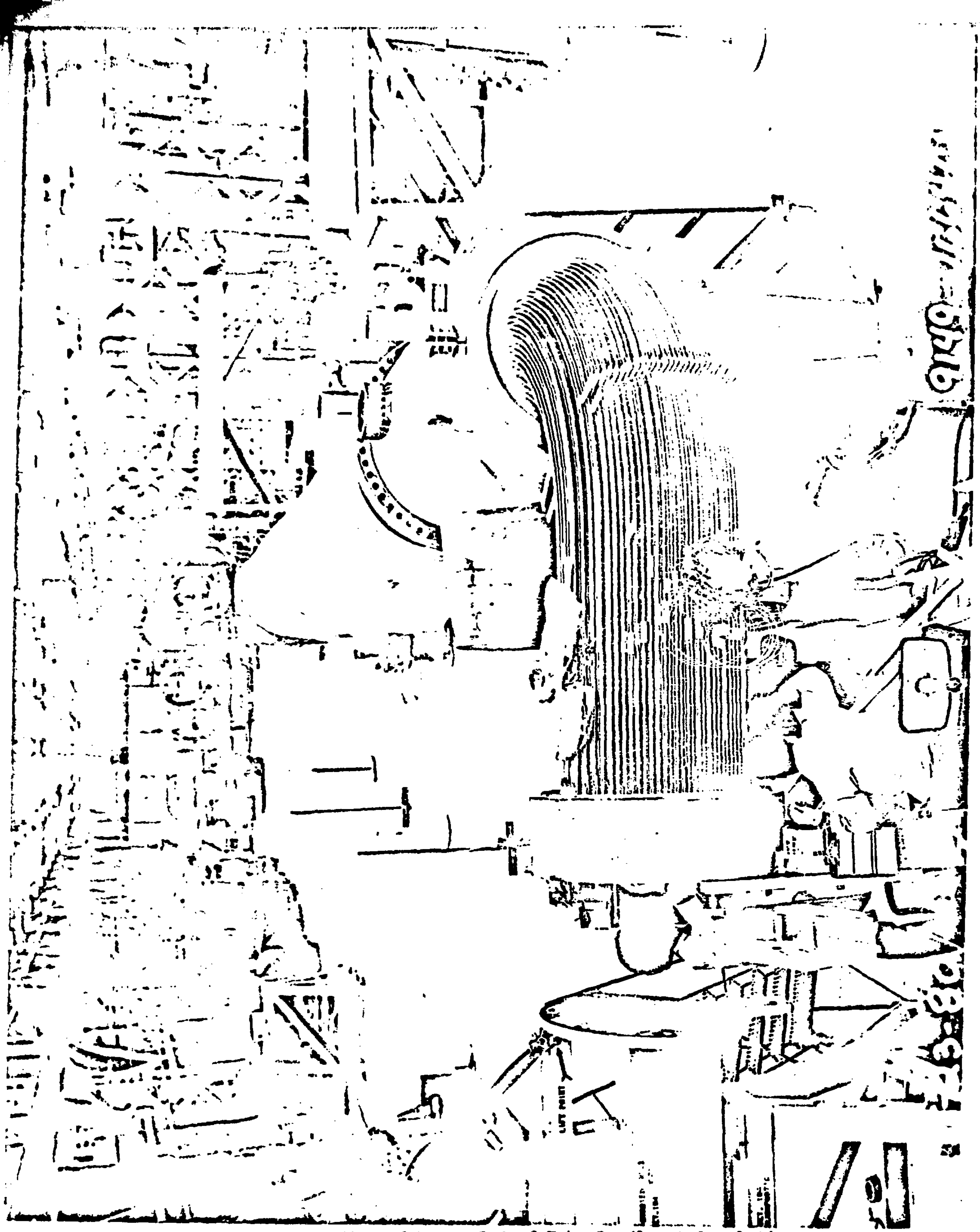


FIGURE 14 View of Upper End of Tube Bundle at Completion of Tube-to-Tubesheet Welding

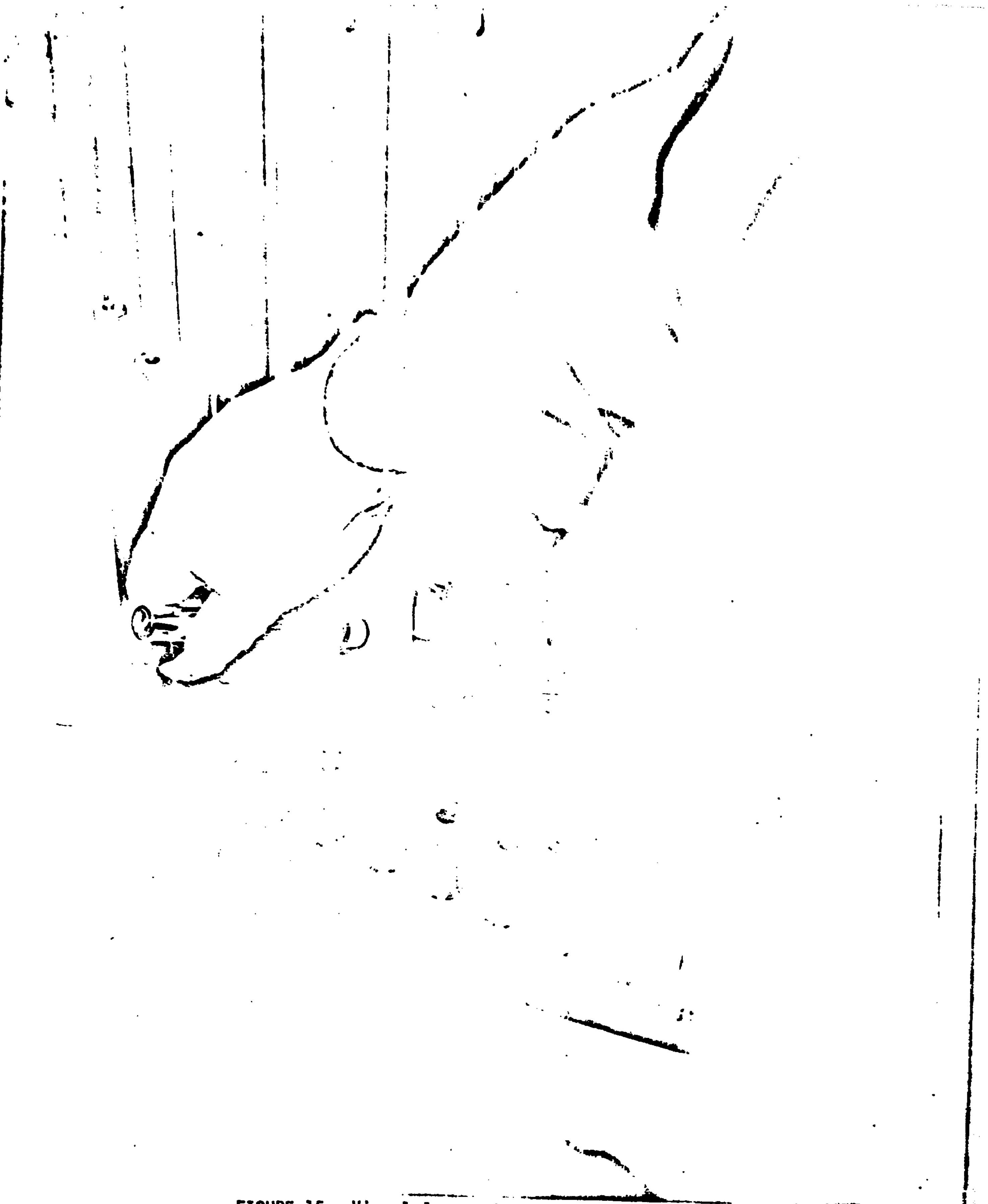


FIGURE 15 Visual Inspection of a Lower Tube-to-Tubesheet Weld

FIGURE 16

## CONCLUSION

Fabrication of the Prototype Unit is near completion and will be delivered to the test site in August, 1981.

The Plant Unit design is presently at an advanced stage and will result in steam generator units fully capable of meeting all the requirements of the CRBRP Power Plant.