QUARTERLY
PROGRESS REPORT
JULY, AUGUST, SEPTEMBER, 1965

BAW-1280-32

to

U.S. Atomic Energy Commission
Chicago Operations Office
Lemont, Illinois

OCTOBER 15, 1965

SODIUM-HEATED STEAM GENERATOR DEVELOPMENT

AEC CONTRACT NO. AT (11-1) - 1280
B&W CONTRACT NO. 610 - 0067

THE BABCOCK & WILCOX CO.
BOILER DIVISION

Barberton, Ohio

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Signed: Paul Perkert

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TABLE OF CONTENTS

ABSTRACT

The method of reporting on this Contract has been changed to a Quarterly Progress Report. This report is the first quarterly report.

The Preliminary Design of the Prototype Steam Generator is complete and the Preliminary Prototype Design Report will be issued within the next month.

Fabrication of the Prototype Steam Generator is proceeding. Coiling of tubes for the superheater bundle was started. Overall fabrication of the steam generator is approximately 20% complete. Some time was lost in starting of coiling, but it is hoped to regain this time as work progresses.

2. INTRODUCTION:

The method of reporting on this Contract has been changed to include an Informal Monthly Progress Report and a Formal Quarterly Progress Report. This report is the first quarterly Progress Report. It is planned to include enough background material in each Quarterly Progress Report that the report can be understood without reference to all previous reports. In general, each report will have a section describing the design and fabrication of the Prototype Steam Generator and the R&D Program, as outlined in the table of contents.

2.1 HISTORY OF CONTRACT:

This Contract was received on March 20, 1963, and signed on April 3, 1963. The overall objective of this Contract is to develop a large sodium-heated steam generator of improved design. This steam generator design will be available for use in the Atomic Energy Commission Sodium Reactor Development Program which has the overall objective of developing reliable, economical, large central station nuclear power plants.

2.2 SCOPE OF CONTRACT:

The scope of work covered under this Contract is briefly as follows:

Phase I

A - Preliminary Design of Full-Size Sodium-Heated Steam Generator.

B - Supporting Research and Development Work.

C - Preliminary Design of 30 MWT Prototype Steam Generator.

Phase II

A - Detail Design of Prototype Steam Generator

B - Fabrication of Prototype Steam Generator for Installation and Testing at SCTI, Santa Susana, California.

C - Final Design Report Relating the Performance of the Prototype to the Design of the Full-Size Steam Generator.

3. DESIGN WORK:

The preliminary design of the Full-Size of Steam Generator has been completed. This design is reported in BW 67-2, "Preliminary Design Report Full-Size Steam Generator" Volume 1 and 2. Additional work was done on analyzing problems of a sodium-water reaction in the Full-Size Steam Generator and this work was reported in BW 67-2(a), "Addendum to Preliminary Design Report Full-Size Steam Generator, Sodium-Water Reaction."

The Preliminary Design of the Prototype Steam Generator is complete and the preliminary draft of BAW-1280-28, "Preliminary Design Report Prototype Steam Generator" has been written. This design report is being revised to include the analysis work on sodium-water reaction problems in the Prototype Steam Generator.

The general arrangement of the Prototype Steam Generator is shown on Drawing 20957F. The design and performance data are tabulated on Tables 1 and 2. The Prototype Steam Generator is intended as a proof test of the solution of various problem areas in the design, manufactured,

and operation of the Full-Size Steam Generator. For this reason the Prototype Steam Generator in itself is not an optimum heat exchanger. It is larger in overall size for the amount of heat transfer surface contained than would normally be used. In modeling the problem areas in nozzles, tube sheets, etc. there is some wasted space within the Prototype Steam Generator.

In addition to modeling the temperature and pressure conditions of the Full-Size Steam Generator the Prototype Steam Generator will model the heat transfer and fluid flow conditions of the tubes of the Full-Size Steam Generator. The Prototype Steam Generator has the same diameter and length of tubes as the Full-Size Steam Generator. The feedwater inlet tube sheet is a full-size model of one of the similar tube sheets of the Full-Size Steam Generator so that the thermal and pressure stresses in this tube sheet assembly will be a full-size model of the Full-Size Steam Generator.

3.1 ANALYSIS OF SODIUM-WATER REACTION PROBLEMS:

In any steam generator heated by sodium one must face the possibility of a sodium-water reaction. Designing the boiler with double tube walls does not insure against a dosium-water reaction, and increases the cost of the heating surface in the steam generator by approximately a factor of four. It is considered credible that sometime during the thirty-year lifeof the Full-Size Steam Generator some type of leak will develop in which water will react with the sodium. This condition must be taken into account in designing the steam generator. The design of the Full-Size and Prototype Steam Generator is such that the steam

generator shells in each case are protected by a liner separated from the shell by an inert gas annulus. A gas cusion is provided over the sodium in each steam generator, and relief valves and rupture discs are provided for the safe relief of the products of a sodium-water reaction.

Generally, the problems of handling sodium-water reaction products in Full-Size and Prototype Steam Generators will be quite similar. The tubes in both steam generators are the same size so that the size of leak and quantity of reaction products to handle will be the same. The transient pressure peaks during the first few miliseconds will be less in the Prototype Steam Generator because it is not as tall as the Full-Size Steam Generator and therefore the inertia of the sodium between any leak site and the upper surface of the sodium is somewhat less.

A co-operative effort program has been carried out by Atomic Power Development Associates and the Babcock & Wilcox Company to analyze the Full-Size and Prototype Steam Generators under conditions of a sodium-water reaction accident. This program was divided into two parts. The first part was a study of the Full-Size Steam Generator to determine the credible maximum number of tubes that may fail as a result of an initial leak from a single tube, and to demonstrate that the products from this reaction can be safely contained and relieved. The second portion of this study was a somewhat easier task of demonstrating that the Prototype Steam Generator will be adequately safe to operate as SCTI.

The work done by APDA under sub contract No. 1 to this Contract is directed to analyze the dynamics of sodium-water reactions within the B&W Steam Generator and to predict pressures, temperatures, and volumes of the reaction products. B&W has analyzed the strength of the steam generators to determine if the steam generator shells can withstand the pressures and temperatures predicted from the APDA calculations.

The results of the APDA analyses are reported in the following APDA reports:

1. APDA 167-Ford, J.A., "Literature Review of Sodium-Water Reactions," March 15, 1965.
2. APDA 175-Ford, J.A., Chamberlaine. H.B., and Chase, W.L. "Analysis of the Babcock & Wilcox Prototype Steam Generator for a Sodium-Water Reaction," August 23, 1965.
3. APDA 176-Ford, J.A., Chamberlaine, H.B., and Chase, W.L., "Analysis of the Babcock & Wilcox Full-Size Steam Generator for a Sodium-Water Reaction."

Using hydrogen bubble pressure and volume predictions from the APDA work B&W calculated the dynamic strength of the steam generator shell for various times from the beginning of the assumed leak. Calculations were made of the strength of the complete shell and liner on an impulse-momentum method of analysis and on a work-energy method of analysis. In addition to these a separate analysis was made of the credibility of a bulge type failure due to local high pressures near the reaction site. These analyses show that the Full-Size Steam Generator can withstand at least the simultaneous failure of twenty tubes in the worst location in

the steam generator at the operating condition resulting in the highest leak rate of water to sodium. Very conservative assumptions were used throughout in the calculation of the expected pressures and in the calculation of the shell strength.

A separate analysis was made of the possibility of collapse of the centerpipe. These calculations show that the failure of one tube in the worst location and the boiler bundle may cause the centerpipe to collapse. Whether the centerpipe is full of sodium or arranged to be full of gas does not significantly affect the credibility of collapse. If the centerpipe does collapse and seal off the escape path through it the pressure predicted by the APDA Computer Program is almost unchanged. The reason for this is that the mathematical model is limited to two parallel escape paths. When it is assumed that the centerpipe is plugged another escape path can be selected for the computer solution and the net result is almost no change in the predicted pressure.

Separate analyses were made on the strength of the various coil supports and tube vibration supports and baffles. These analyses show that the support structure within the steam generator is strong enough to withstand the simultaneous rupture of at least twenty tubes in the worst location in the steam generator.

A question was raised whether the high temperatures generated by the reaction could cause overheating and failure of the vessel shell. For the time intervals when the pressure is high, (10^{-4}/1sec.) the high

temperature of the reaction will have penetrated the liner wall less than 5 per cent of the liner thickness, and will not have penetrated the main vessel wall at all. The strength of the liner and vessel will not have been significantly affected during this time. At longer times the rupture disc or relief valve will have operated to relieve the pressure within the steam generator.

The results of these analyses are reported more fully in BW 67-2(a) "Addendum to Preliminary Design Report, Full-Size Steam Generator, Sodium-Water Reaction," September 15, 1965.

3.2 DETAIL DESIGN AND FABRICATION OF PROROTYPE STEAM GENERATOR:

Detail drawings are proceeding as required to meet the fabrication schedule.

Welding of the straight length of superheater tubing into the proper lengths for the superheater coil bundle has been completed and welding has begun on the boiler tubes.

A summary of the operations performed on the superheater tubes so far is as follows:

1. Remove tubes from boxes and remove paper sleeve protectors.
2. Inspect tubes for tolerance on O.D. I.D. wall thickness.
3. Assign mark number to each tube corresponding to the mark number this tube will bear when assembled into the steam generator. Numbered plastic tape is being used to affix the mark numbers to the tubes.
4. Hydrostatic test tubes with demineralized water, drain and dry.
5. Machine weld preparation on ends of tubes and install plastic plugs to prevent foreign objects from getting into tubes.
6. Weld sections of tube together to proper length for superheater coils, using automatic TIG process with cold wire feed.
7. Grind weld to same OD as tube.
8. Dye check weld.
9. X-ray weld using angle shot from two positions 90° apart. (It is of interest that these x-rays have shown no weld defects to a sensitivity of 2% of the wall thickness or .010" minimum).

10. Store tubes on racks with ends plugged under plastic dust cover ready for coiling.

11. Coil tubes for superheater bundle.

Machining of the upper head and upper shell is proceeding. Fabrication of the tube supports and internal shrouding is complete ready for assembly with the tube bundle when it is complete.

There was some delay in starting the coiling of superheater tubes because of difficulties with the coiling fixture, but it is felt this time can be recovered as coiling proceeds.

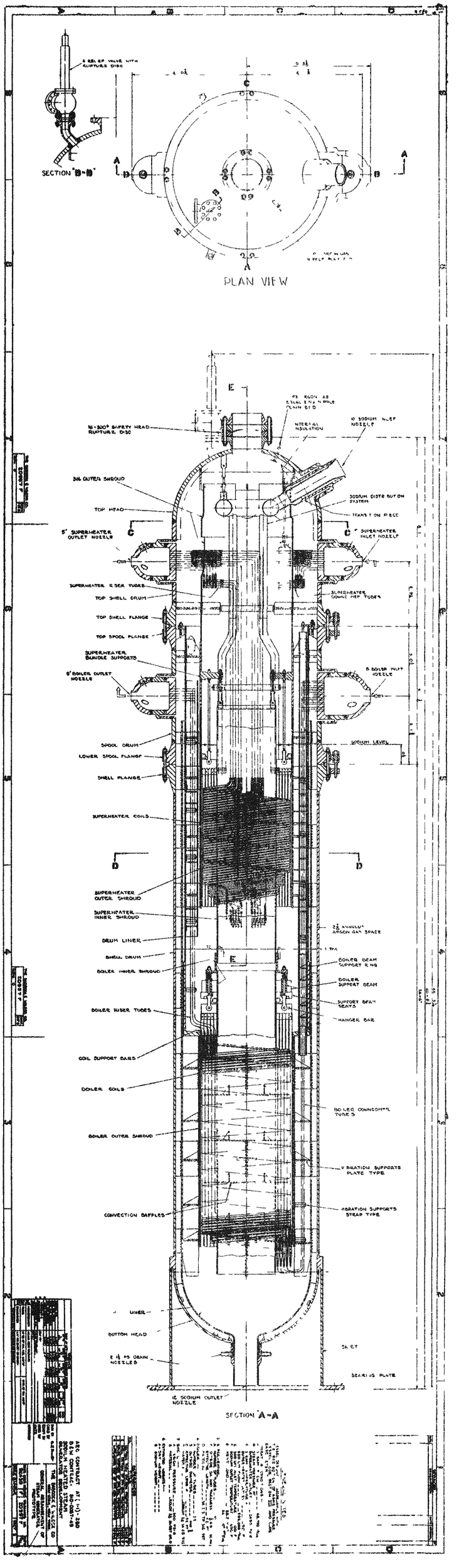

SECTION "B-B"
PLAN VIEW
SECTION "A-A"
TOP HEAD
TOP SHELL DRUM
TOP SHELL FLANGE
TOP SPOOL FLANGE
SPOOL DRUM
LOWER SPOOL FLANGE
SHELL FLANGE
SUPERHEATER COILS
SUPERHEATER OUTER SHROUD
SUPERHEATER INNER SHROUD
DRUM LINER
SHELL DRUM
BOILER INNER SHROUD
BOILER RISER TUBES
COIL SUPPORT BARS
BOILER COILS
BOILER OUTER SHROUD
CONVECTION BAFFLES
BOTTOM HEAD
SOCKET
BEARING PLATE
SODIUM LEVEL
HANGER BAR
SUPPORT BEAM SEATS
BOILER SUPPORT BEAM
BOILER BEAM SUPPORT RING
VIBRATION SUPPORTS PLATE TYPE
ABRATION SUPPORTS STRAP TYPE
SUPERHEATER INLET NOZZLE
SUPERHEATER OUTLET NOZZLE
BOILER OUTLET NOZZLE
BOILER INLET NOZZLE
SODIUM DISTRIBUTION SYSTEM
TRANSITION PIECE
INTERNAL INSULATION
SUPERHEATER RISER TUBES
SUPERHEATER DOWNCOMER TUBES
SUPERHEATER SUPPORTS
SODIUM OUTLET NOZZLE
SODIUM INLET NOZZLE
ARGON GAS SPACE
BOILER DOWNCOMER TUBES
SAFETY HEAD RUPTURE DISC
SAFETY VALVE WITH RUPTURE DISC
AEC CONTRACT AT (49)-280
D.E.M. CONTRACT
SODIUM REACTOR
SIMULATION TEST EQUIPMENT
THE BABCOCK & WILCOX CO.
GENERAL ARRANGEMENT OF PROTOTYPE

TABLE [illegible]

PROTOTYPE STEAM GENERATOR
BOILER SECTION

Design Conditions

[illegible] Temperature	[illegible]
Steam Temperature	8[illegible] F
Steam Pressure	[illegible]

Operating Conditions (Full Load)

Heat Load	[illegible] x 10^6 Btu/hr.
	[illegible] Mwt
Sodium Flow	[illegible],500 lb/hr
Sodium Inlet Temperature	[illegible] F
Sodium Outlet Temperature	[illegible] F
Sodium Pressure Drop	[illegible] psi
Steam Flow	[illegible]
Feedwater Temperature	[illegible]
Outlet Temperature	[illegible]
Outlet Pressure	[illegible]

Material

Tubes	[illegible] 22
Shells, Heads	[illegible]
Tube Sheets, Nozzles	[illegible]

Heat Transfer Results

	Number	Size	Average Length
Tubes: Economizer	2[illegible]	1" O.D. x .120" [illegible]	[illegible]
Nucleate & Film Boiling Sec.	25	1" O.D. x .145" [illegible]	[illegible].8 ft
Superheat Section	25	1" O.D. x .165" [illegible]	[illegible].0 ft
Total Length			[illegible] ft

Overall Heat Transfer Coefficient

Economizer		[illegible] Btu/hr ft²/F
Boiler	Nucleate Boiling	[illegible]
	Film Boiling	451 F [illegible] F
Superheater		[illegible]
Fouling Factor		[illegible]
Erosion Allowance		

BOILER SECTION
TABLE 1 (CONT'D)

Heat Transfer Surface	
Total	885.1 ft^2
Economizer	376.1 ft^2
Boiling Zone	371.6 ft^2
Superheater	137.4 ft^2
Boiler Inlet Tubes	
Size	5/8" O.D. x 0.076" MW
Boiler Outlet Legs	
Size	1" O.D. x 0.165" MW
Pressure Drop (Steam Side)	
Downcomers	14.82 psi
Economizer	6.21 psi
Boiling Zone	31.15 psi
Superheater	24.07 psi
Boiler Risers	27.10 psi
Tube Spacing	
Transverse Tube Spacing S_T	1.625"
Parallel Tube Spacing S_{11}	1.375"

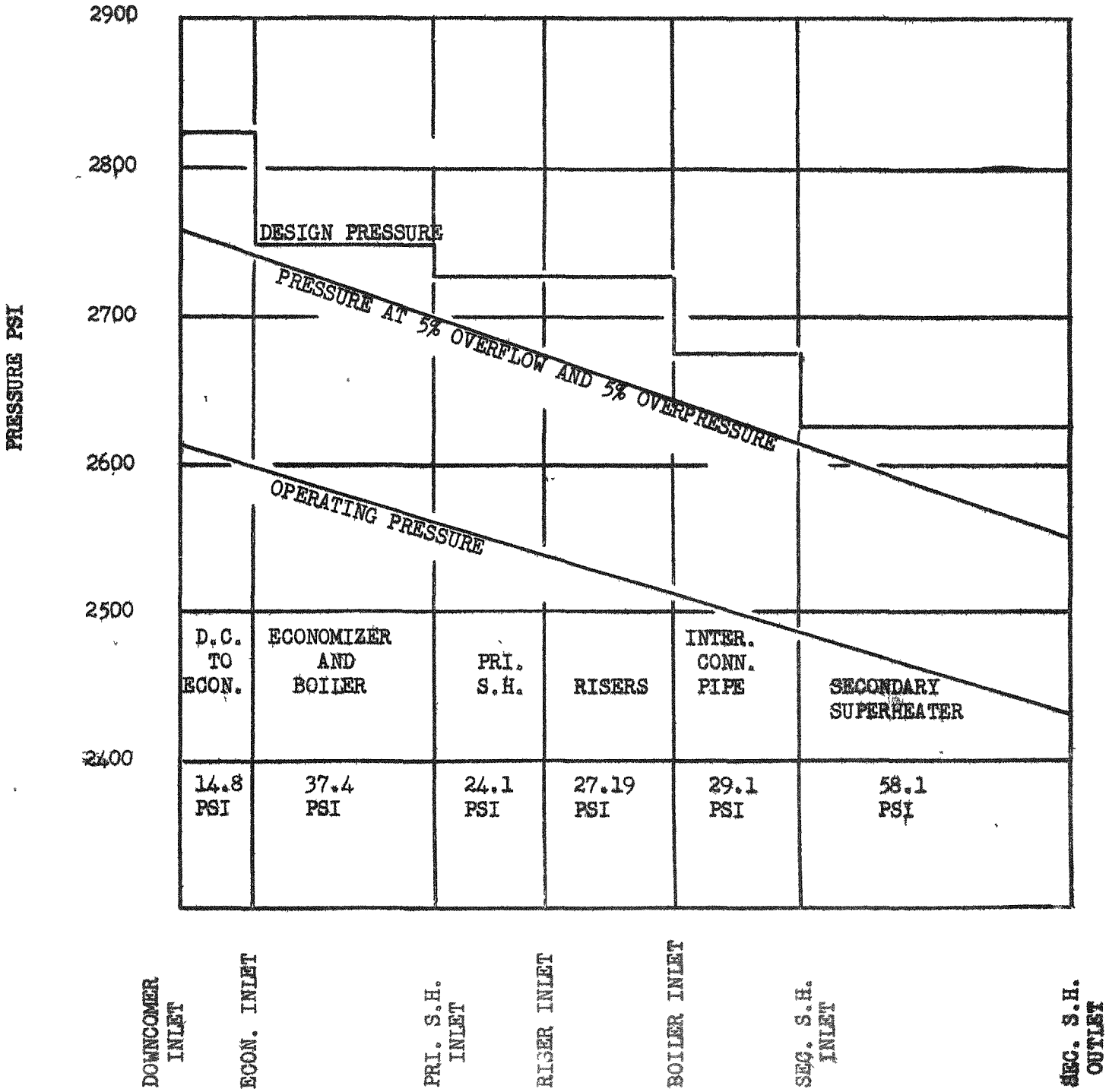
TABLE 2
PROTOTYPE STEAM GENERATOR
SUPERHEATER SECTION

Design Conditions	
Sodium Temperature	1200 F
Steam Temperature	1070 F
Steam Pressure	2625 psi
Operating Conditions (Full Load)	
Heat Load	21.82×10^6 BTU/hr.
	6,39 MWt
Sodium Flow	583,000 lb/hr
Sodium Inlet Temperature	1140 F
Sodium Outlet Temperature	1016 F
Sodium Pressure Drop	0.081 psi
Steam Flow	88,750 lb/hr
Steam Inlet Temperature	750 F
Steam Outlet Temperature	1050 F
Outlet Pressure	2425 psig
Material	
Tube	SA-213 TP 316 SS
Shell	Croloy 2-1/4
Tube Sheets, Heads	TP 316 SS
Heat Transfer Results	
Tubes - Number	45
Size	7/8" O.D. x .120" MW
Effective Length	43.1 ft.
Overall H.T. Coefficient	347.0 B/hr/ft^2/F
LMTD	162.5 F
Surface	444 ft^2
Superheater Inlet Tubes	
Size	7/8" O.D. x .120" MW
Material	TP 316 SS

PROTOTYPE
DESIGN PRESSURE - BOILER AND SUPERHEATER
STEPPED DESIGN PRESSURE FOR ONCE-THROUGH BOILERS
5% OVERFLOW, 5% OVERPRESSURE

DESIGN PRESSURES

DOWNCOMERS TO ECONOMIZER	2825 PSI
ECONOMIZER AND BOILER	2750 PSI
PRIMARY SUPERHEATER	2725 PSI
RISERS	2725 PSI
INTERCONNECTING PIPE	2675 PSI
SECONDARY S.H.	2625 PSI



4. RESEARCH AND DEVELOPMENT PROGRAM:

Research and development work has been approved under this Contract on the following projects:

1. Materials - Carbon transfer and effect on material properties simulating Full-Size Steam Generator
2. Corrosion of Croloy 2-1/4 in products of a sodium-water reaction.
3. Procedure for welding tubes to back side of tube sheet.
4. Sodium-Water Reaction - Engineering Analysis.

The problem to be solved and the general scope of each project is described on the following pages along with a short description of the present status of each project.

4.1 MATERIALS - Carbon Transfer and Effect on Material Properties: Background Information

There are actually two problems relating to materials and their application for a sodium-heated steam generator. There is the problem of carbon and mass transfer on the sodium side of a tube, and also the problem of possible accelerated corrosion at the location of Departure from Nucleate Boiling on the steam side of the tube.

It is well known that materials used in sodium systems above 900F undergo mass trnasfer whereby materials are dissolved from the tube walls in the hot zones and deposited in the cooler zones. In addition to this, there is a tendency for ferritic materials such as Croloy 2-1/4 to decarburize in sodium and for the carbon to deposit out on austenitic stainless steel surface in the same sodium circuit. Considerable work has been done on mass transfer and carburization-decarburization problems, but it is evident that further work is required to determine if these problems will be of concern in a closed loop with the relative quantities of Croloy 2-1/4 and Type 316 Stainless Steel involved in this design.

Tests have shown that accelerated corrosion can occur in boiler tubes at the location of DNB under certain conditions. Under some conditions a boiler tube can fail by corrosion at the DNB point in less than 24 hours.

This corrosion is controlled on fossil fuel fired boilers by careful control of water chemistry and by designing the boiler to have the DNB point occur in a zone of low heat flux. The relatively low heat flux at the DNB point comes about because of the necessity of protecting

the tube from over-heating ("Burnout") when nucleate breaks down and film boiling sets in. In this sodium-heated boiler there are no extremely high temperatures such as found on the gas side of a fossil fuel fired boiler, but because sodium is so much better a heat transfer medium than gas there can be very high heat fluxes with moderate temperature differences between sodium and water.

Tests which have been conducted on model once-through sodium-heated steam generators at temperature levels below those for which this boiler is designed showed a small amount of corrosion had taken place. It is felt that further experimentation is required at the temperature levels of interest for this steam generator to be able to say with assurance that serious corrosion will not occur.

Objective

The purpose of this program is to study the carbon and mass transfer effects in a sodium-heated steam generator containing both austenitic and ferritic materials, and to determine that these materials will be satisfactory on both the sodium and the water side for use under the planned operating conditions.

Scope

A test loop simulating the Full-Size Plant was operated for 8000 hours. The test boiler consisted of a one-tube model steam generator having a Croloy 2-1/4 boiler section and a Type 316 Stainless Steel Superheater Section. The portion of Croloy 2-1/4 and Stainless Steel were arranged to simulate the proportions of these materials in the Full-Size Steam Generator, and the ratio of surfaces exposed to volume of sodium were to duplicate the full-size boiler as closely as possible.

Sodium was circulated outside the tube in the model boiler at flow rates and temperatures simulating the full-load boiler, and steam was generated within the tube at pressures and temperatures similar to the full-load boiler. The sodium system consisted of an electromagnetic pump, flow meter, sodium heater, plugging indicator, cold trap, and pump. The water-side system included a reservoir, pump, heater, condenser-cooler section and a back pressure regulator.

The test operation extended over approximately one year. Samples were removed and examined during the test operation to get preliminary test results. Following the test, the unit was dissembled and the materials examined by appropriate techniques such as metallographic, electron microprobe, electron microscope, and spectrographic means.

PRESENT STATUS:

After completing 8000 hours of operation at full temperature the test loop was shut down and preliminary piping samples removed for examination. Metalographic examinations of the piping samples show that very little change had occurred in type 316 material and that the normal tempering of the Croloy 2-1/4 material due to temperature alone almost overshadowed any decarburization that had occurred. Based on thse examinations it was decided to proceed with the distructive examination of the model boiler and superheater and of selective samples from other parts of the loop.

Macrophotographs (3X) have been made of the sodium and water sides of the specimans removed from the boiler and superheater tube. Only thin rather uniform deposits are visible on the steam side of the

specimans, except for those located in the transition (boiling) zone which have a somewhat heavier, non-uniform deposit making the tube appear to be rough. These deposits at the transitions zone appeared heavier than are normally seen in a once through steam generator and were felt to be a characteristic ov something in the sedign or operation of this test facility. A complete analysis of the water side deposits was made. The deposit weight varied from 44.5 grams/ft^2 entering the transitions zone to 20.1 grams/ft^2 leaving this location. The deposit was of similar chemical composition at all locations and the principal depositing material was a magnesium silicate tentatively identified by x-ray diffraction as the compound $3MgO \cdot 2SiO_2 \cdot 2H_2O$ (The mineral serpentine). It was established that the source of this material was from the packing of the reciprocating feedwater pump. Minute quantities of this packing material were carried with the feedwater and deposited in the boiler tube. This kind of contamination would not be found in a Full-Size plant because it would not have a reciprocating feedpump and total amount of packing exposed to the feedwater would be extremely small when compared to the small scale test loop.

The destructive examination of materials removed from the model steam generator is continuing. Tensile testing of tubular specimans from the boiler and superheater has been completed. The changes in tensile properties for the Croloy 2-1/4 material can be attributed to tempering which occurred during service. The yield strength increase from 40,300 psi for the as received material to 47,800 psi at the lowest

service temperature with a gradual decrease in strength with increasing service temperature which is similar to the secondary hardening effects common in many alloy steels. The ductility of the specimans remained virtually constant at all service temperatures. For the Type 316 tubular specimans, both yield strength and tensile strength after exposure were slightly larger than the initial values. Ductility after exposure was reduced from about 55 per cent to appsozimately 44 per cent, (as measured by per cent elongation).

During the next three months it is expected that the examination of metalographic samples will be completed including the mocroprobe analysis of the specimans, and work will have begun on the final report on this project.

4.2 CORROSION RATES FOR CROLOY 2-1/4 STEEL IN PRODUCTS OF A SODIUM-WATER REACTION:

Background Information

Tube failures, due to sodium-water reactions, can result from stress corrosion cracking, apparent metal fatigue, and bulk transcrystalline corrosion. Considerable interest has been generated in this later type of metal failure, and little is known of its rates under the widely varying environmental conditions which can exist.

Examples of such corrosion can be found in the tube failures at Enrico Fermi Power Plant. In one area where a sodium-water reaction occurred, blisters with longitudinal cracks were noted. The wall thickness of the tube at the point of the reaction was reduced by about 1/2.

During normal operation, hot sodium is circulated through the steam generator. If a sodium-water reaction occurs with its associated products of reaction, the station operator has two choices in regard to sodium flow, (1) shut off the sodium flow to the steam generator and permit sodium temperature to decay to the isothermal condition, or (2) keep the sodium flowing and maintain a specific temperature in the unit. Since the products and rates of reaction are temperature dependent, it may be advantageous to maintain a predetermined sodium temperature to lessen the corrosion effects of a sodium-water reaction.

The program described below is directed toward the assembly of the necessary test equipment, and the determination of corrosion rates for Croloy 2-1/4 steel in hot aqueous solutions of sodium hydroxide, and in suitably equilibrated solutions of sodium hydroxide in molten sodium. Corrosion rates in pure sodium will be determined in order to establish base values for this work.

Croloy 2-1/4 steel is used for this project since it is more susceptible to gross corrosion in a sodium environment than any of the stainless steels to be used for other parts of the system. Although the austenitic steels are more susceptible to stress corrosion, extensive work has already been done in this area by B&W and others.

Objective

The objective of this program is to provide information pertaining to the rate of corrosion of Croloy 2-1/4 steel with reactor products of water-to-sodium leak. A second objective is to be able to postulate the mechanism of reaction between Croloy 2-1/4 steel and the sodium-

water reaction products. With such information, it will be possible to determine if there are operational procedures which will reduce the corrosion effect of a sodium-water reaction on the tube material. Also, the tests will indicate what products of corrosion will affect the tube material and to what extent.

PRESENT STATUS:

Testing work under this R&D project has been completed, and the final report is being written. The results from this investigation have been included, along with the results from a separate investigation of sodium-water reaction problems by APDA, in the Preliminary Design Report on the Full-Size Steam Generator BW 67-2 and in the Preliminary Design Report on the Prototype Steam Generator, BAW 1280-28.

It is expected that the final report on this or any project will be issued within the next two weeks.

4.3 PROCEDURE FOR WELDING TUBES TO BACK SIDE OF TUBE SHEET:

Background Information

An expanded and seal welded tube-to-tube sheet joint can develop a crevice exposed to the sodium side. Sodium vapor will collect in this crevice. When the steam generator is shut down for maintenance, moisture' laden air finding its way into the sodium side of the unit will result in the formation of a strong caustic solution concentrated in the crevice thus presenting the classical conditions for stress corrosion of stainless steel or "Caustic embrittlement" of ferritic alloys.

Objective

One way to eliminate the above problem is to machine a projection on the sodium face of the tube sheet at each tube and butt weld the

tube to this projection.

Scope

The purpose of this project is to develope the welding machine, welding processes, and Quality Control procedures to demonstrate sound welds for use in the Full-Size and Prototype Steam Generator.

A "flash-type" welding process was investigated and, although sound welds were produced, the inner surface of the weld was such that the integrity of the weld could not be varified by x-ray or other non-destructive techniques.

A change in scope of this project was suggested to stop work on the "flash-type" welding process and develop a TIG process utilizing a cold wire feed into the weld. (Filler wire is required for the thickness of tubes in the full-size and Prototype Steam Generator. This change in scope was approved, and work has begun on this TIG process with cold wire feed.)

PRESENT STATUS:

A twenty-five tube mock-up of the boiler inlet tube sheet is being fabricated so that the welding of these tubes to the tube sheet can be demonstrated and this mock-up can be sent to Los Alomos to prove out the x-raying of these welds using the x-ray tube that Los Alomos has developed.

To insure clean metal in the lips to which the tubes are welded, these lips will be machined from weld deposit cladding of the same material as the tube sheet. The first effort at cladding the tube sheet mock-up was not successful because the wire used for this automatic weld deposit process did not meet specifications. The wire

was rejected and new wire procured. The cladding of the tube sheet mock-up is proceeding without difficulty.

The fabrication of the welding torch for making the welds in the 7/8" and 1" OD tubes is proceeding. The first model of this torch had some mechanical defects and did not control the position of the wire being fed in correctly. It is believed that the modifications to this torch will cure this problem. It is planned that to have torch completed by mid November.

4.4 SODIUM-WATER REACTION - ENGINEERING ANALYSIS - (APDA Sub Contract):

Objective

The objective of this study is to determine whether the Babcock & Wilcox Full-Size Steam Generator and Prototype Steam Generator have a high probability of withstanding the reaction resulting from a leak of water into sodium.

Scope

The scope of work under this sub contract consists of a mathematical analysis of the sodium-water reaction problems and design consultation on the Babcock & Wilcox Company's 30 MWT Prototype Steam Generator and Full-Size Steam Generator using existing sodium-water reaction test data.

PRESENT STATUS:

The APDA analysis work, with the exception of consultation as required and up-dating of the mathematical model based on test data from another program is completed. These studies are recorded in the following three reports:

1. APDA 167-Ford, J.A., "Literature Review of Sodium-Water Reactions," March 15, 1965.

2. APDA 175-Ford, J.A., Chamberlaine, H.B., and Chase, W.L. "Analysis of the Babcock & Wilcox Prototype Steam Generator for a Sodium-Water Reaction," August 23, 1965.

3. APDA 176-Ford, J.A., Chamberlaine, H.B., and Chase, W.L. "Analysis of the Babcock & Wilcox Full-Size Steam Generator for a Sodium-Water Reaction."

These data have been used in the analysis of the strength of the Full-Size and Prototype Steam Generators as discussed in Section 3.1 of this report.