MASTER

GULF GENERAL ATOMIC

GA-10338

STEAM GENERATORS FOR HIGH-TEMPERATURE
GAS-COOLED REACTOR PLANTS
IN THE U.S.A.

by

W. G. Schuetzenduebel

Prepared in part under
Contract AT(04-3)-633
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

Date Published - March 19, 1971

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

GULF GENERAL ATOMIC COMPANY
P.O. BOX 608, SAN DIEGO, CALIFORNIA 92112

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

GULF GENERAL ATOMIC

GA-10338

STEAM GENERATORS FOR HIGH-TEMPERATURE
GAS-COOLED REACTOR PLANTS
IN THE U.S.A.

by

W. G. Schuetzenduebel

Prepared in part under
Contract AT(04-3)-633
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Date Published - March 19, 1971

Gulf General Atomic Project 900

GULF GENERAL ATOMIC COMPANY
P.O. BOX 608, SAN DIEGO, CALIFORNIA 92112

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

ABSTRACT

Describes the development of steam generators for gas-cooled high-temperature reactors. Traces the development from the existing steam generators operating at the 40 MW(e) plant at Peach Bottom to those developed for the 330 MW(e) plant at Fort St. Vrain. Also includes data on the steam generator designs for an 1100 MW(e) commercial HTGR plant and for a 300 MW(e) gas-cooled fast reactor demonstration plant. Material covers description, design, and research. Included for the Peach Bottom plant are controls, operation, and failure history, and, for Fort St. Vrain, fabrication and controls.

CONTENTS

ABSTRACT	iii
1. INTRODUCTION	1
2. THE 40 MW(e) PEACH BOTTOM HTGR PROTOTYPE PLANT.	2
2.1. Steam Generators	2
2.1.1. Description	2
2.1.2. Design	4
2.1.3. Controls	17
2.1.4. Research and Development	17
2.1.5. Operation	19
2.1.6. Failure History	21
3. THE 330 MW(e) FORT ST. VRAIN POWER DEMONSTRATION PLANT	37
3.1. Steam Generators	39
3.1.1. Description	39
3.1.2. Design	48
3.1.3. Fabrication	67
3.1.4. Controls	68
3.1.5. Research and Development	77
3.1.6. Schedule	86
4. THE 1100 MW(e) COMMERCIAL HTGR PLANT	87
4.1. Steam Generator	91
4.1.1. Development.	91
4.1.2. Design	95
4.1.3. Research and Development	115
5. THE 300 MW(e) GCFR DEMONSTRATION PLANT	119
5.1. Plant Description	119
5.1.1. Steam Generator	124
6. OUTLOOK	129
7. ACKNOWLEDGEMENTS	133
8. REFERENCES	134

FIGURES

1. Peach Bottom Atomic Power Plant Schematic Cycle Diagram	3
2. Peach Bottom Atomic Power Plant Steam Generator Arrangement	5
3. Steam Generator Tubesheet - Original Design	16
4. Peach Bottom Atomic Power Plant Simplified Control Diagram	18
5. Steam Generator Performance - Predicted Versus Actual	22
6. Gas-Side Performance - Predicted Versus Actual	23
7. Steam Generator Performance at 98% Power	24
8. Peach Bottom Atomic Power Plant Operating History	27
9. Peach Bottom Steam Generator Tubesheet - New Design	34
10. Fort St. Vrain Nuclear Power Plant Schematic Cycle Diagram	38
11. Fort St. Vrain Nuclear Power Plant Reactor Arrangement	40
12. Fort St. Vrain Nuclear Power Plant Steam Generator Module	43
13. Primary Closure Assembly	46
14. Steam Generator Shop Assembly	47
15. Tube Design Temperatures	51
16. ASME Code, Section III Stress Limits	57
17. Stress Allowables for 210,000 Hours	60
18. Reheater Collector	62
19. Secondary Closure Thermal Sleeve - Superheater Tube	63
20. Tube Support Structure with Coiled Tubes	64
21. Sleeve and Wedge Assembly	66
22. Sulzer Feedwater Ringheader	69
23. Reheater Support Structure	70
24. Steam Generator Assembly	71
25. Primary Assembly	72
26. Secondary Assembly	73
27. Fort St. Vrain Nuclear Power Plant Simplified Control Diagram	75
28. Sulzer Vibration Test Arrangement	81
29. Boiling Stability Test Loop - Schematic	84
30. 1100 MW(e) HTGR Plant	88
31. 1100 MW(e) HTGR Plant Schematic Cycle Diagram	89
32. 1100 MW(e) HTGR Plant Nuclear Steam Supply System	90

FIGURES (continued)

33. 1100 MW(e) HTGR Plant Steam Generator Concept (Downflow)	92
34. 1100 MW(e) HTGR Plant Steam Generator Concept (U-Tube Reheater)	94
35. 1100 MW(e) HTGR Plant Steam Generator	96
36. ASME Code, Section III, Summer Addendum 1969, Stress Analysis . . .	103
37. 1100 MW(e) HTGR Plant Simplified Control Diagram	111
38. 300 MW(e) GCFR Plant	122
39. 300 MW(e) GCFR Plant - Schematic Cycle Diagram	123
40. 300 MW(e) GCFR Plant - Steam Generator.	127
41. Projected Addition to the U.S. Electric Power Generating Capacity	130

TABLES

1. Peach Bottom Atomic Power Plant - Steam Generator Material Selections	7
2. Peach Bottom Atomic Power Plant - Steam Generator Design Conditions	8
3. Peach Bottom Atomic Power Plant - Steam Generator - Applied Stress Limits	11
4. Peach Bottom Atomic Power Plant - Steam Generator - Transient Operating Conditions	14
5. Peach Bottom Atomic Power Plant - Cumulative Availability	25
6. Fort St. Vrain Nuclear Power Plant - Steam Generator Design Conditions.	42
7. Fort St. Vrain Nuclear Power Plant - Steam Generator Material Selection	54
8. Fort St. Vrain Nuclear Power Plant - Steam Generator - Transient Operating Conditions	59
9. Fort St. Vrain Nuclear Power Plant - Steam Generator - Applied Research and Development	79
10. 1100 MW(e) Commercial HTGR Plant - Steam Generator Design Conditions	97
11. 1100 MW(e) Commercial HTGR Plant - Steam Generator Material Selection	102

TABLES (continued)

12. 1100 MW(e) Commercial HTGR Plant - Steam Generator Operating Transients	105
13. 300 MW(e) GCFR Demonstration Plant - Supporting Utilities	120
14. 300 MW(e) GCFR Demonstration Plant - Steam Generator Design Conditions	125

STEAM GENERATORS FOR HIGH-TEMPERATURE GAS-COOLED
REACTOR PLANTS IN THE U.S.A.

by

W. G. Schuetzenduebel

Gulf General Atomic, Incorporated, San Diego, California U.S.A.

1. INTRODUCTION

During the early years of nuclear power generation the steam generator was somewhat neglected component in nuclear power engineering. Being considered a conventional plant component no special effort towards its development was expended. The experience gained from the operation of the initial generation of nuclear power plants, however, has resulted in much greater attention being paid to the steam generating equipment and industry has initiated costly development programs directed towards the achievement of the ultimate goal: A trouble free steam generator with high reliability and availability.

As it was painfully recognized, the steam generator is an extremely important element of power generation from nuclear sources. This applies equally to thermal as well as to breeder reactors. The only exceptions are the boiling water reactors and possible future gas-cooled reactors which will employ a direct gas turbine cycle.

The nuclear steam generator must keep pace in its development with the fast evolution of the nuclear power system. So far, most of the experience acquired in nuclear steam generator operation stems from pressurized water systems which are in use for a much longer time period than any of the other nuclear systems. The pressurized water reactor, however, which initially was developed for marine propulsion, operates at relatively modest steam conditions.

The development of nuclear power plants in the last few years has been marked by a significant increase in unit size. In addition, an extraordinary improvement in steam condition was achieved in the field of gas

cooled reactor plants. This is true for reactors operating with uranium and metal clad fuel, the AGR, as well as for the uranium-thorium fueled high temperature reactor, the HTGR. Unlike the pressurized water reactor, no standardization of the power plant cycle nor its components, has been achieved for gas cooled nuclear power plants. This lack of standardization essentially forces the designer of the plant and its components into costly and time consuming research and development programs specifically designed to meet the requirements of a given plant.

Very high demands are being made on the development and design of steam generators for gas cooled nuclear power plants. These high demands are due mainly to the high temperature conditions as well as the limited space available for these components inside the pre-stressed concrete pressure vessel, which is a common feature for all modern gas cooled reactor plants.

2. THE 40 MW(e) PEACH BOTTOM HTGR PROTOTYPE PLANT

In August 1959, the AEC signed contracts with the Philadelphia Electric Company, Bechtel Corporation and General Atomic, Inc. for the design and construction and operation of a 40 MW(e) gas-cooled reactor prototype plant. This plant was intended to demonstrate the feasibility of the HTGR concept and the U235-Thorium-U233 fuel cycle. The design life for this prototype plant is 30 years, although the power demonstration period under the AEC contract is only 5 years. The Peach Bottom reactor is housed in a steel pressure vessel. Two steam generators are used in parallel for exchanging the heat from the primary coolant helium to the secondary coolant water. Both steam generators are located outside of the pressure vessel and are connected to the reactor vessel by concentric helium ducts carrying the hot and cold helium to and from the steam generators and helium circulators. Figure 1 is a schematic cycle diagram of the Peach Bottom atomic power plant.

2.1. Steam Generators

2.1.1 Description

The Peach Bottom Atomic Power Station has two primary steam generating systems each consisting of a shell and tube helium to water/steam

LC25039

Figure 1. Peach Bottom Atomic Power Plant Schematic Cycle Diagram

heat exchanger, a steam drum, a recirculating pump, associated interconnecting water water and steam piping and controls. The shell and tube heat exchanger is the only component in the steam generating system exposed to the primary coolant flow.

The steam generator, as shown in Figure 2, is of the vertical U-tube design, carrying helium on the shell side and water or steam in the tubes. Its overall dimensions are 8 feet diameter and 30 feet height.

The tube bundles are separated into three distinct sections, namely economizer, evaporator and superheater. Feedwater at full load enters the steam generator at 425°F and superheated steam leaves at 1005°F and 1550 psig. The steam flow rate is 97,500 lb/hr per steam generator.

Helium from the reactor core at 1367°F and 329 psig enters the steam generators through the inner pipe of a concentric helium nozzle and first encounters a perforated impingement baffle, which in conjunction with a tube shroud, provides for uniform lateral and longitudinal distribution of helium flow to the tube bundles. The impingement baffle also prevents erosion of the superheater tubes. The helium flow is distributed along the length of the tube bundle in a distribution shroud and directed across the tubes by internal baffles. The helium flows through the superheater, evaporator and economizer section successively and leaves the steam generator at 628°F via the compressor inlet pipe after passing through the annulus formed by the shroud and steam generator shell. Helium from the compressor discharge is returned to a side connection in the annular part of the concentric inlet nozzle. The helium flow rate is 234,200 lb/hr to each steam generator.

2.1.2. Design

The steam generators were supplied under subcontract with C. F. Braun Co., who had complete design and fabrication responsibility. Fabrication was performed in part by Baldwin-Lima-Hamilton Co. and completed by Nooter Corp. under contracts with Braun. In areas of critical stresses General Atomic performed independent analyses to confirm the soundness of the detailed design. Throughout the design and fabrication General Atomic

LC84849

Figure 2. Peach Bottom Atomic Power Plant Steam Generator Arrangement

maintained a close surveillance of the engineering, the fabrication methods, and the quality control.

The shell side of the steam generator is a primary pressure vessel and is designed in accordance with Section VIII - Unfired Pressure Vessels - of the 1959 ASME Boiler and Pressure Vessel Code and various Case Interpretations in effect at the date of contract.

Parts on the water side, including the tubes, tubesheets, channels, piping, and steam drums, are designed in accordance with Section I - Power Boilers - of the 1959 ASME Boiler and Pressure Vessel Code. In addition, the design accords with the ASA Pressure Piping Code and the TEMA Standards.

The materials used in the fabrication of the steam generator vessels are listed in Table 1. All significant structural materials are in accordance with Section II - Materials Specification - of the 1959 ASME Boiler and Pressure Vessel Code.

In designing the steam generators, the best available data were used to evaluate the possible effects of corrosion and erosion. From this it was concluded that no provision need be included in the thicknesses of the steam generators for corrosion beyond that provided in Section I of the ASME Code. Dissolved impurities in the water will be carefully controlled during operation to prevent excessive formation of oxide films on the water side of the vessel. No corrosion allowance is added to wall thicknesses on the helium side of the vessel since helium is inert and no corrosion is expected. Flow of fluids throughout the system are held to rates that do not cause erosion. At the helium inlet where helium could impinge on the tube bundle at a high velocity, a diffuser is used on the duct and an impingement plate is provided to prevent direct impingement of helium on the tubes.

Table 2 outlines the design parameters for the steam generator. The maximum pressure surges will not exceed design conditions. However, under certain emergency conditions, temperature excursions may exceed design values. The maximum estimated temperatures during the excursion are included in the table.

TABLE 1
PEACH BOTTOM ATOMIC POWER PLANT
STEAM GENERATOR MATERIAL SELECTION

Part	Material
<u>Water Side:</u>	
Channel sphere	Hi-strength C-Si
Main tubesheet	C
S.H. outlet tubesheet	Cr-Ni stainless
Economizer tubes	C
Evaporator tubes	C
Superheater tubes	Cr-Ni stainless
Channel nozzles	C
S.H. outlet nozzles	1 Cr-1/2 Mo
<u>Helium Side:</u>	
Shell	C-Si
Helium inlet nozzle	C-Si
	1 Cr-1/2 Mo
Helium outlet nozzle	C
S.H. tube support plate	Cr-Ni stainless
Evap. tube support plate	5 Cr-1/2 Mo
Econ. tube support plate	1 Cr-1/2 Mo
Insulation	Solami C (laminated corrugated stainless steel foil)

NOTE: Various other materials were used in the construction of leak baffles, tie rods, drain and vent nozzles and similar components of minor significance.

TABLE 2
PEACH BOTTOM ATOMIC POWER PLANT
STEAM GENERATOR DESIGN CONDITIONS

	100% Steam Flow	30% Steam Flow
<u>Load</u>		
Generator Output (MW)	45.4	~ 8.0
Station Output (MW)	40.0	~ 5.7
<u>Helium</u>		
Flow (lbs/hr)	~ 500,000	~ 170,000
Outlet Pressure (psia)	329	329
Inlet Temperature (°F)	1367	1240
Outlet Temperature (°F)	628	495
<u>Feedwater/Main Steam</u>		
Flow (lbs/hr)	368,000	110,400
Outlet Pressure (psia)	1590	1470
Inlet Temperature (°F)	425	343
Outlet Temperature (°F)	1005	1005

2.1.2.1. Stress Analysis

The following parts were subject to a detailed stress analysis covering primary, secondary and local stresses due to pressure, thermal, and external loadings:

- a. Complete tubesheet
- b. Concentric helium inlet nozzle
- c. Helium outlet nozzle
- d. Interaction between concentric helium inlet nozzle, including Tee connection, and helium outlet nozzle
- e. Steam generator supports
- f. U-tubes
- g. Shell side manhole

Stress analysis in sufficient detail to prove the mechanical design has been made on the following items:

- a. Superheater outlet partition and nozzle
- b. Feedwater inlet nozzle
- c. Steam drum support
- d. Channel closure
- e. Interconnecting piping, including piping flexibility
- f. Helium inlet nozzle liner
- g. Steam generator shroud
- h. Tube supports
- i. Leak baffles
- j. Lifting lugs

All the basic thicknesses of the pressure components are based on ASME Code requirements. Thus all primary membrane stresses are lower than the allowable values listed in Section I and Section VIII of the ASME Code.

In addition to the ASME Code requirements, the structural reliability of the vessel was determined by computing primary and secondary stresses at local discontinuity areas. The codes existing at the time of the design of the steam generators did not cover the allowable limits for these stress categories. However, in the stress analysis two sets of allowable limits were satisfied: (1) the allowables shown in Table 3; and (2) the general philosophy of the ASME Section III "Rules for Construction of Nuclear Vessels", then in the draft stage, and subsequently adopted in 1963.

The summary of ASME Section III philosophy was as follows:

a. General Primary Membrane Stress Intensity. This is the stress intensity derived from the average value across the thickness of a section of the general primary stresses produced by design internal pressure and other specific mechanical loads, but excluding all secondary and local stresses. The allowable value of this stress intensity is S_m , as given in September, 1962 draft of ASME proposed Section III. For temperatures higher than 750°F, the values of S_m are taken from Section VIII, ASME Code.

b. Local Membrane Stress Intensity. This is the stress intensity derived from the average value across the thickness of a section of the local primary stresses produced by design pressure and specified mechanical loads but excluding all secondary stresses. The allowable value of this stress intensity is $1.5 S_m$.

c. Primary Membrane (General or Local) Plus Primary Bending Stress Intensity. This is the stress intensity derived from the highest value across the thickness of a section of the general or local primary membrane stresses plus primary bending stresses produced by design pressure and other specified mechanical loads, but excluding all secondary and peak stresses. The allowable value of this stress intensity is $1.5 S_m$.

d. Primary Plus Secondary Stress Intensity. This is the stress intensity derived from the highest value at any point across the thickness of a section of the general or local primary membrane stresses plus secondary stresses produced by specified operating pressure and other specified mechanical loads and by thermal effects. The effects of gross structural

TABLE 3
PEACH BOTTOM ATOMIC POWER PLANT
STEAM GENERATOR - APPLIED STRESS LIMITS

Type of Stress	Limit
Primary membrane stress	Allowable stress listed in ASME Code
Primary membrane plus-bending stress	1.5 times the allowable stress listed in ASME Code
*Combined primary and secondary stresses due to the combined effects of pressure and steady-state temperatures	Room-temperature yield strength plus the lower value of (1) metal temperature yield strength, (2) stress for a creep rate of 1 per cent in 100,000 hours at metal temperature

*NOTE: Either the Maximum Shear Theory (Tresca) or the Maximum Energy of Distortion Theory (Hencky-von Mises) shall be used in evaluating combined stresses. Transient stresses shall be combined with the steady-state stresses in accordance with the Modified Goodman diagram. Cumulative damage due to alternating stress which occur a finite number of times shall be determined using Miner's hypothesis, with a limiting damage factor of 0.8.

discontinuities but not of local structural discontinuities (stress concentrations) are included. The allowable value of this stress intensity is $3 S_m$.

e. Peak Stress Intensity. The peak stress intensity is that derived from the highest value at any point across the thickness of a section of the combination of all primary, secondary, and peak stresses produced by specified operating pressures and other mechanical loads and by general and local thermal effects and including the effects of gross and local structural discontinuities. The allowable value of this stress intensity is dependent on the range of the stress difference from which it is derived and on the number of times it is to be applied. Cumulative damage due to alternating stresses which occur a finite number of times was determined using Miner's hypothesis, with a limiting damage factor of 0.8. This is more conservative than Section III which allows a cumulative usage factor of 1.0.

f. Theory of Failure. The maximum shear stress theory (also known as the Tresca criterion), or the energy of distortion theory (also known as the octahedral shear theory and Mises criterion) is assumed in predicting the failure. The reason for adopting these theories is that they are much better than the maximum stress theory for predicting both yielding and fatigue in ductile metals. Maximum shear theory has been generally adopted in the analysis, as it is somewhat more conservative and more readily applied.

2.1.2.2. Loading Conditions

a. Design Piping and Support Loads. The steam generator is suspended from four 3-inch diameter hanger rods. This gives it freedom to move in the horizontal plane due to loading arising from the thermal expansion of the piping. The steam drum is supported by four constant support hangers which give it freedom to move vertically and horizontally due to loading arising from the thermal expansion of the water piping and movement of the steam generator. The water piping is supported in three places by constant support hangers. Because of these degrees of freedom of the

vessels, the principal loads which are imposed by the piping are low in magnitude.

All forces and moments due to the piping are assumed to act concurrently, and they are assumed to act in the direction most deleterious to the vessel. The loads occurring during Normal Operating Conditions are used in the steady state stress analysis. Loads occurring under Emergency Conditions are used only in the fatigue evaluation for the emergency cycles.

b. Dynamic Mechanical Loading Conditions. There are two sources for dynamic loading of the steam generator, earthquake and pipe rupture. The steam generation system and its supports are designed for a 0.1 g horizontal or vertical seismic load. A special study was made to determine whether a failure in a helium pipe would cause a coincident failure in a water-side pipe, or vice versa. The study showed that if the proper lateral restraints were applied to the steam generator vessel, no coincident failures would occur. Restraints using hydraulic cylinders for control of the dynamic loads have been provided.

c. Transient Operational Cycle Design Cases. The steam generator vessel is subject to various normal and emergency operational changes in steam conditions. The changes in steam load affect temperatures and pressures of helium, water, and steam. The rates of change of these parameters have effects on stress distribution patterns in the steam generator which are considered in the cyclic fatigue analysis.

Eight transient operational cycle design cases were assumed. These are listed in Table 4. Each cycle has the same initial and final steady-state conditions. The rates of change and peak temperatures and pressures are conservative estimates of actual operating values, taken wherever possible from the results of reactor analog model studies which give both helium and steam conditions as functions of time. Use of the analog results makes the stress values for these cycles as realistic as possible. Additional conservatism was allowed by provision for more operating cycles than are likely to occur during the 30 year design operating life of the plant.

TABLE 4
PEACH BOTTOM ATOMIC POWER PLANT
STEAM GENERATOR - TRANSIENT OPERATING CONDITIONS

Transient	No. of Cycles
Cold start-up and shut-down	100
Hot start-up and shut-down	370
Cold start-up and loss of electric power or reactor scram	100
Cold start-up and loss of helium pressure	1
Cold start-up and steam leak	20
Hot start-up and loss of one circulating pump	50
Load change	40,000
Control rod stutter	10×10^6

2.1.2.3. Thermal Loading

The design temperatures of fluids at various locations in the steam generator for 100 percent and 10 percent load normal operating conditions are shown in Table 2 . Transient changes in fluid temperature during normal and emergency operating cycles are discussed in the previous section. The temperature distributions cause unique and complex temperature distribution in two areas of the pressure vessel--the tubesheet and the helium inlet. All other areas of the pressure vessel have negligible thermal gradients during normal operation. During transients thermal gradients are set up due to heat flow through the walls and from components of different cross-section. However, these gradients do not significantly affect the fatigue strength of the vessel.

a. Tubesheet. The tubesheet is the most severely loaded part of the steam generator and experiences its most severe thermal loading condition during the cold start-up and steam leak transient. Figure 3 shows the temperature distribution in the tubesheet during this transient.

b. Tubesheet Bimetallic Joint. The superheater tubesheet is fabricated from stainless steel, while the main tubesheet is carbon steel. This causes a problem of differential expansion because of the difference in operating temperatures and the difference in coefficient of thermal expansion. To minimize this problem, the juncture between stainless and carbon steels is located in the cylindrical barrel connecting the two tubesheets.

c. Helium Inlet. The helium inlet pipe conducts helium at a design temperature of 1367°F into the shrouded bundle. The helium, after leaving the bundle flows over the shroud and out of the steam generator. The flow then goes to the compressor and is returned to the outer section of the concentric piping. To prevent cold helium from reentering the generator, a conical divider is welded in the annular position of the concentric nozzle. This divider is designed to withstand a 450 psi differential pressure in either direction at design temperature.

Thus, the divider restrains the inner pipe from changing diameter under all conditions. The conical divider has some flexibility, but still stresses are developed. During transients, the gradients can become more severe.

TEMPERATURE DISTRIBUTION IN THE TUBESHEET

LC84854

Figure 3. Steam Generator Tubesheet - Original Design

2.1.3. Controls

The Peach Bottom plant control system uses the HTGR characteristics with forced recirculation boilers in a manner analogous to fossil-fired steam generators of similar design and throttle conditions. Essentially, the plant is fired for pressure and has a controlled-temperature heat source for superheater control. For any given steam flow, as determined by manual opening of the turbine-generator throttle valves, only one combination of helium temperature and helium flow rate will give 1000°F steam at the required 1450 psig throttle pressure. This results from the physical characteristics of the reactor-steam generator loops, especially the fixed superheater surface of the drum-type boiler. The steam pressure controller output sets the demanded speed on the two helium compressor speed controllers, thus governing the flow of helium through the reactor and the two steam generators. The steam temperature controller sets the required temperature on the helium temperature controller, which in turn sets the required flux on the reactor neutron flux controller. The flux controller, operating one automatic control rod, then makes the reactor power level match the required plant power. The feedwater controllers are of the conventional three-element type, matching feedwater flow to steam flow with additional control as required to hold the drum level constant for all loads.

Figure 4 shows a simplified schematic of the Peach Bottom plant control system.

2.1.4. Research and Development

Only a limited amount of R&D work was carried out to aid in the design of the Peach Bottom steam generators. Most of the research and development programs executed prior to the manufacture were concerned with compatibility of the materials of construction with the primary and secondary flow media. A fair amount of work was also dedicated to the development of welding procedures for the high strength materials, especially for tube to tubesheet joints.

In addition, during the preoperational hot test runs, a vibration test was conducted on the steam generators.

Figure 4. Peach Bottom Atomic Power Plant Simplified Control Diagram

2.1.5. Operation

Power operation of the Peach Bottom Atomic Power Station began in January 1967 and full power was reached in May. The significant results of the start-up of the world's first power producing High Temperature Gas-Cooled Reactor plant was widely reported.

In addition to producing commercial power, Peach Bottom being a prototype nuclear power station requires that power changes, including shutdowns be performed to accommodate testing of plant systems and components under the AEC sponsored post construction research and development program. The approach to full power operation of the Peach Bottom station was programmed as a series of tests which took place during the first six months of 1967. These tests consisted of operational experiments and surveillance covering the disciplines of reactor physics, radiochemistry as well as chemical, mechanical, electrical and control engineering.

The engineering tests were designed to demonstrate the operational behavior of the major components and systems of the nuclear steam supply system and to verify that the overall plant performance was within the predicted limits. The Peach Bottom steam generators were the controlling items in planning the approach to power programs. In addition to imposing a number of restraints, they were the only major equipment items that could not be performance tested before start-up. Therefore, considerable analytical work was done utilizing digital computer codes to determine exactly how the plant should be operated.

In addition the reactor plant transient behavior was studied by use of the Peach Bottom Trainer-Simulator, an analog computer with operator and instructor consoles located at the plant site.

In the initial power-raising operation of January 1967 the reactor reached approximately 28% power before it was decided that a shutdown was necessitated because of excessively high cold helium temperatures. The analysis of plant behavior was made difficult by shifting instrument calibrations, failure of several important temperature detectors, and difficulty in controlling feedwater flow and temperature because of

leaking valves. With the data that were available, and in consultation with the original steam generator designer, it was agreed that a substantial (approximately 20%) hot-gas bypass existed underneath the tube bundles. A downstream blankoff baffle was installed on the bottom bay to correct this situation, and several new shell metal thermocouples were added to monitor the effects of this baffle on the vessel.

The March 1967 tests showed improvements in effective heat transfer and in the ability to measure and control temperatures, feedwater, recycle flow, and steam pressure to the desired conditions. These all combined to yield steam-side performance, which was very close to predicted values; however, cold helium temperatures were nearly 100°F too high, requiring higher helium flows to maintain steam at 1000°F and 1450 psig. A high temperature on the steam generator bottom head was also found, and although steam generation and control were excellent to approximately 60% power, a shutdown for modifications was again required. Additional baffling was added to the next two adjoining bays in each steam generator and to the clearance space between the tube bundles and the hot gas shroud. More thermocouples were added to the steam generator shells.

The May 1967 restart revealed another improvement in lowering the outlet gas temperature about 25°F at approximately 30% power, but shell metal temperatures were still excessive. To permit further study of the overall plant and steam generator behavior, a short-term operation was permitted at a maximum shell temperature of 775°F at 320 psig helium pressure. This yields a lower stress than 700°F at 450 psig, the design conditions of the steam generators. Steam pressure was lowered to 1250 psig and high helium flow rates were used, resulting in low steam temperatures, and, finally, the high-pressure feedwater heaters were turned off. Reactor power reached approximately 96% under these conditions before the bottom head temperature reached the limit of 775°F.

The data obtained were used to analyze the problem and also served to demonstrate that, except for the high temperature of the shell metal, the plant was capable of full power operation with no difficulty. It was

considered that a combination of radiative heating of the head by the shroud bottom and inadequate cooling flow caused the excessive temperature. This was remedied by removing the insulation from the steam generator bottom heads, passing room air over the surface and through a cooler, and returning it to the room. Late in May 1967, this system was installed, tested, and operated successfully. The plant was brought to essentially full power operation at design steam conditions (see Figure 5). The baffle additions made in April had been used to modify the earlier performance predictions to account for the changes in helium flow distribution. This had also been studied by use of a water flow-table mockup of a typical tube bay. Figure 6 shows the excellent agreement of the helium-side conditions with these revised predictions. Figure 7, a heat diagram for one steam generator at nearly full power, also shows the performance at one set of conditions that are very close to design numbers.

As of October 31, 1970 the Peach Bottom Atomic Power plant had operated a total of 546.6 effective full power days.

In this time it generated a total of 566,400,000 KW hrs. The cumulative plant and nuclear system availability since start-up are presented in Table 5.

The operating history is shown in Figure 8. As of now no steam generator failure or malfunction has forced a shutdown of the Peach Bottom plant.

2.1.6 Failure History

2.1.6.1 Failure Detection

When the steam generators were opened for inspection during the last week in September of 1965 the stainless steel 304 bellows in the superheater outlet section of No. 1 steam generator was found to have failed due to cracking. The corresponding bellows in No. 2 steam generator was subsequently found to be in the same condition. This cracking was identified as stress corrosion cracking. An examination of the ends of the SS 304 H superheater tubes as well as the stainless steel piping

LC53824

Figure 5. Steam Generator Performance - Predicted Versus Actual

LC53823

Figure 6. Gas-Side Performance - Predicted Versus Actual

ECONOMIZER DUTY	31.9×10^6 BTU/HR (16.3%)
EVAPORATOR DUTY	107.1×10^6 BTU/HR (54.8%)
SUPERHEATER DUTY	56.4×10^6 BTU/HR (28.9%)
<hr/>	
TOTAL	195.4×10^6 BTU/HR

LC53826

Figure 7. Steam Generator Performance at 98% Power

TABLE 5
PEACH BOTTOM ATOMIC POWER PLANT
CUMULATIVE AVAILABILITY

Year	Month	Plant Availability*	NSSS Availability*	Gross Plant Capacity Factor†
1967	June	100	100	91.4
	July	99.2	100	78.4
	August	89.5	97.8	86.2
	September	0	0	0
	October	67.2	67.2	56.0
	November	92.9	100	87.1
	December	100	100	83.7
	Cumulative	78.4	80.7	68.9
1968	June	98.0	100	90.0
	July	80.9	80.9	74.3
	August	88.0	88.0	78.8
	September	100	100	94.9
	October**	73.3	73.3	69.8
	Cumulative	88.0	88.4	81.6
1969	March	75.0	75.0	64.6
	April	100.0	100.0	93.3
	May	96.5	100.0	86.1
	June	36.7	40.3	34.3
	July	91.5	91.5	82.5
	August	98.8	98.8	64.1
	September***	93.3	93.3	46.0
	Cumulative	84.7	85.7	67.1

TABLE 5 (Cont.)
PEACH BOTTOM ATOMIC POWER PLANT
CUMULATIVE AVAILABILITY

Year	Month	Plant Availability*	NSSS Availability*	Gross Plant Capacity Factor†
1970	July	57.5	57.5	51.3
	August	100.0	100.0	91.5
	September	84.8	84.8	74.3
	October	90.6	90.6	84.3
	Cumulative ††	33.6	33.6	30.8

*(service hr/net period hr) x 100

†[MW-hr(e)/net period hr x 44.5 MW(e)] x 100

**Shutdown on October 23, 1968, at 300 EFPD

***Shutdown on October 3, 1969 at 452 EFPD for refueling with core #2.

††Through October 31, 1970

LC59484

Figure 8a. Peach Bottom Atomic Power Plant Operating History for 1967

LC67195

Figure 8b. Peach Bottom Atomic Power Plant Operating History for 1968

LC74528

Figure 8c. Peach Bottom Atomic Power Plant Operating History for 1969

Figure 8d. Peach Bottom Atomic Power Plant Operating History for 1970

connected to the failed bellows was then made to determine the extent of the stress corrosion attack. Since this examination did not disclose any evidence of cracking a pressure test was performed which resulted in three leaking superheater tubes in steam generator No. 1. No leaks were found in steam generator No. 2.

An extensive program to analyze the failure condition and cause of failure was initiated. During this examination evidence of typical stress corrosion cracking, transgranular as well as intergranular was found when specimens cut from the bellows were metallographically examined. Furthermore, traces of chlorine were found on the crack fracture surfaces as well as some uniform thin layer of a reddish-brown deposit on the inside and outside of the bellows.

Further investigations also revealed several cracks in the pipe elbow welded to one of the expansion bellows. Here again intergranular and transgranular cracking was found.

Metallographic examination of samples cut from several superheater tube ends, including the leaking tubes, showed cracks. These cracks were found in the following areas:

- a) Immediately below the seal weld
- b) Immediately below the rolled area
- c) Less severe cracks up to eight inches below the tubesheet.

Following this examination an additional 117 tubes (57 from unit 1 and 60 from unit 2) were examined at their inlet end and indications of cracking were found on all tubes.

In addition evidence of pitting attack was found on the inside surface of the superheater tubes and electron microscope analysis indicated the presence of chlorine in the cracks and surface imperfections on the inside tube surfaces. All tube cracks were found to have started on the inside of the tubes. None of the tube-to-tubesheet welds showed any evidence of cracks.

Since the steam generators had not seen any service other than pressure testing and hot test runs, during which the steam generators were

pressurized and heated by the heat of compression of the main compressors, an extensive search for the cause of the stress corrosion was conducted. This search covered fill water, possible oxygen ingress, inert gas blanketing, dirt and dust as well as manufacturing methods and materials such as weld electrodes, weld slag, cleaning agents, lubricants, etc.

No satisfactory answer to the question of contamination was found during this search. None of the investigated possible causes above appeared to result in sufficient contamination to induce the severe stress corrosion in evidence. Therefore, it was finally concluded that a combination of residual stresses with the presence of relatively low chloride and other contaminants under the presence of oxygen present at one time or another during fabrication, erection and testing had led to the described condition.

2.1.6.2. Corrective Measures

The foregoing analysis shows that a cracking condition existed in the superheater section of the steam generator. Although the only through cracks observed in the tubes were in the tubeheet area, other partial cracks were found below the tubesheets. Repairing the cracks that existed within the tubesheet and operating the steam generators with the crack indication known to exist in other areas was considered at the time. However, the evidence available suggested that crack propagation could occur in the partial cracks already present.

As an alternate to repairing the existing tubes, the feasibility of retubing the superheater sections of the steam generator was considered from both the metallurgical and design standpoint.

Weighing the probability and consequences of tube failures using repaired tubes against the option of using new tubes, it was concluded that retubing the superheater section of the steam generators was the optimum solution.

Knowledge concerning the selection of materials had developed significantly since the original design of the steam generators and additional materials for this type of application were approved in the ASME Code.

The 300 series stainless steels, including 304H, were considered carefully. During the initial design phase of the steam generators, also Incoloy 800 was considered, however, at that time this material did not have any superheated steam service record, nor did it have ASME Code approval.

Since then, service history of Incoloy 800 became available and ASME Code approval has been granted. When considering these reasons, together with the fact that Incoloy 800 is superior to SS 304 steel in strength, oxidation resistance and resistance to stress corrosion, it was decided that Incoloy 800 was the better choice for retubing the superheater. A repair design was then adopted employing Incoloy 800 superheater tubes and a raised Incoloy 800 tubesheet. The barrel section which provides the transition between the existing carbon steel superheater inlet tubesheet and the new Incoloy 800 superheater outlet includes a transition piece composed of Inconel 600 material. This material provides a reasonable gradual change in the thermal expansion coefficients of the adjacent members. Figure 9 shows the configuration of the redesigned tubesheet, which can be compared against the original design in Figure 3.

Supplementing the redesign effort was an extended research program whose primary objective was to define and establish the fabrication requirements for welded Incoloy 800 tube-to-tubesheet joints, which would exhibit superior resistance to stress corrosion cracking. To this end GGA initiated autoclave corrosion tests at Battelle Northwest Institute. These tests were designed to study the stress corrosion behavior of both 304H and Incoloy 800 superheater tubing by subjecting simulated tube-to-tubesheet joints to various environments, including a simulated "hot run" environment, as well as full load operation. This test work was supplemented by boiling magnesium chloride stress corrosion tests that were performed in GGA's own laboratories.

The results of these autoclave corrosion tests showed that stress corrosion cracking occurred in 304H SS tubes when subjected to a high oxygen-low chloride environment (air saturated solution with 2 ppm

0 1 2

A.F. MCHES
R SE A A JV

EM

DESCRIPTION

REACH BO TOM & OMIC POWER S & VON
POLAROID WALL RE TOPICS

FAM GENERATOR
HANNEL
N REMAINS
RIMWORK ELEV

32-5K 34 B

chloride). Incoloy 800 tubes showed no stress corrosion cracks in any of the tests, indicating that this alloy is much more resistant to stress corrosion cracking than the 304H SS material.

The failed SH outlet bellows and adjacent pipe elbow were replaced during the repair program by a single curved pipe also made of nickel alloy, Inconel 600.

Following the repair work the steam generators were chemically cleaned and flushed to ensure that all foreign material was removed. The feedwater and condensate cycle was likewise cleaned prior to operation.

2.1.6.3 Steam Generator Repairs Since Full Power Operation

- a. During the shutdown of September 1967, the No. 1 steam generator was found to have leaks in the channels between the evaporator and economizer sections. This problem was corrected by repairing the welds of the partition baffle where the leakage had occurred.
- b. During the early part of 1968 there were steam generator modifications that involved the installation of thermocouples on the internal shroud surrounding the tube bundle and the removal of shipping lugs from the inside of the shell. During operation, the lower head and shell sides of one steam generator unit had continued to run at a higher than desired temperature. The lugs, which were originally installed to support the shroud during shipment, were cut free from the shroud after installation and no longer served a useful purpose.
- c. On July 7 the reactor was shutdown to repair a broken gauge glass on the No. 2 steam drum.
- d. On August 17 the Peach Bottom plant was shut down to repair a leak in one of the steam generator head cooling systems. The reactor was returned to power on August 21.

- e. During a shutdown in March of 1969 a steam drum pressure relief valve flange was repaired.
- f. In June of 1969 the plant was brought down to remedy problems with the water level instrumentation.

As evidenced by this tabulation no significant problems or failures occurred in the steam generator proper. Most importantly, no tube failures or failures of the tubesheets and tube-to-tubesheet joints have been experienced with the new design utilizing stress corrosion resistant materials, although fluctuating moisture readings in one steam generator indicate the possibility of a tube-to-tubesheet weld crack.

3. THE 330 MW(e) FORT ST. VRAIN POWER DEMONSTRATION PLANT

In July 1965 GGA contracted with the Public Service Company of Colorado to erect a turn-key 330 MW(e) high temperature gas cooled reactor plant, known as the Fort St. Vrain plant.

The total project was jointly financed by PSC, GGA and the USAEC who supported the project under its Power Reactor Demonstration Program.

Under this program the plant serves to demonstrate the economics, reliability and operability of large HTGR power plants and is a further step towards the development and commercial utilization of nuclear power in gas cooled reactors of even larger sizes.

The Fort St. Vrain plant with its many improvements is a great advancement over the Peach Bottom prototype HTGR.

The use of a prestressed concrete reactor pressure vessel made the integration of all nuclear steam supply system components--including the steam generators, helium circulators and reactor core--practical. Integration of these items reduces the primary coolant inventory, eliminates external piping, removes the requirements for steam generator pressure shells, provides a single integral radiation shield and results in minimum building size.

The use of helium as the coolant and graphite as the core structural material allowed the steam cycle to be optimized as in conventional fossil fired power plants. Consequently the steam conditions at the turbine-generator were selected at 2400 psig at 1000°F and reheat to 1000°F. Overall plant net efficiency of the Fort St. Vrain plant will be over 39% for a turbine back pressure of 2-1/2 in. Hg. The turbine-generator is essentially a conventional reheat unit, except that steam from the high-pressure turbine exhaust will drive the helium circulator turbines before being reheated and returned to the intermediate pressure turbine. Figure 10 shows the schematic cycle diagram for the Fort St. Vrain Power Plant.

K53355

Figure 10. Fort St. Vrain Nuclear Power Plant Schematic Cycle Diagram

3.1 Steam Generators

The adoption of the PCRV concept and conventional steam conditions with reheat brought about a whole new area of problems in steam generator design. Firstly the available space inside the PCRV was rather limited requiring a very compact arrangement of heat transfer surfaces.

Piping to and from the main steam and reheat steam sections of the steam generators must penetrate the concrete pressure vessel and is required to have its own radiation shields. Furthermore, the steam generators must be removable for replacement, if necessary.

The larger heat duty for the reheat cycle required the adoption of higher primary coolant temperatures and pressures over the operating range of the unit. This together with the relatively high secondary coolant pressure required high strength materials and larger wall thicknesses of pressure parts. In addition, all steam and water connections must be sized to restrict water leakage into the primary coolant in the event of a tube or connection failure.

It was clear that the above requirements could only be satisfied by a once-through type steam generator system.

After evaluating designs from several companies it was decided to contract Sulzer Brothers of Switzerland to assist GGA in the development and design of the Fort St. Vrain steam generators, since they had a broad range of experience in the field of once-through steam generator design and manufacture in both the conventional and nuclear field.

3.1.1. Description

The plant has two identical steam generator loops arranged in parallel, each of which consists of six individual steam-generator modules, located within the cavity of the prestressed concrete reactor vessel (PCRV) and beneath the reactor core (see Figure 11). Each steam-generator module contains a heat-transfer section with main steam circuitry and an integral reheater, as well as a PCRV penetration section with suitable inlet and outlet connections and helium-pressure barriers.

The purpose of the steam generator is to transfer the heat from the primary coolant, helium, to the secondary coolant, water/steam. The

K52064

Figure 11. Fort St. Vrain Nuclear Power Plant Reactor Arrangement

helium flow that cools the reactor core enters the steam generator at high temperature and gives up its heat, first to the reheat steam section, then to the once-through main steam/water section. The sections produce main steam and reheat steam as demanded by the plant load and main turbine requirements.

The steam generator is designed for continuous operation between the maximum-guaranteed-load point and the one-quarter-load point having the parameters as shown in Table 6.

The modules (Figure 12) are divided into three bundles (heating surface sections) arranged one above the other, made up of helically wound tubes supported by perforated plates attached to the central support structure.

a. Helium Circuitry

The primary coolant for the reactor core is helium. Hot helium gas leaves the reactor core outlet plenum and enters the steam-generator modules at the top via individual circulator ducts to each module. The gas then passes over the reheater, finishing superheater, pre-superheater, evaporator, and economizer surfaces consecutively, transferring its heat to the secondary coolant in the tubes. The helium leaves the steam-generator modules at the bottom and enters a suction chamber from where the circulators recycle it.

b. Main Steam Circuitry

Feedwater is distributed to the 12 modules through flow-control valves and enters each module through a ringheader from where it is distributed to subheader supply tubes by means of the integral flow-trim valves. These subheader supply tubes penetrate the PCRV cavity and its closures. The heat-transfer surfaces of the economizer-evaporator-presuperheater (EES) and finishing superheater (SH II) bundles are composed of helically wound tubes originating

TABLE 6
FORT ST. VRAIN NUCLEAR POWER PLANT
STEAM GENERATOR DESIGN CONDITIONS

	100% Steam Flow	25% Steam Flow
<u>Load</u>		
Generator Output (MW)	342.0	81.2
Station Output (MW)	330.2	67.4
<u>Helium</u>		
Flow (lb/hr)	3,490,000	996,000
Outlet Pressure (psia)	686	589
Inlet Temperature (F)	1427	1236
Outlet Temperature (F)	742	569
Pressure Drop (psi)	3.47	0.36
<u>Feedwater/Main Stream</u>		
Flow (lb/hr)	2,305,326	576,342
Outlet Pressure (psia)	2512	2419
Inlet Temperature (F)	403	299
Outlet Temperature (F)	1005	1000
Pressure Drop (psi)	590	45
<u>Reheat Steam</u>		
Flow (lb/hr)	2,245,366	556,928
Outlet Pressure (psia)	600	151
Inlet Temperature (F)	673	570
Outlet Temperature (F)	1002	1000 •
Pressure Drop (psi)	42	11

LC72627-D

Figure 12. Fort St. Vrain Nuclear Power Plant Steam Generator Module

at the feedwater subheaders. Stabilizing nozzles are provided at the inlet of each tube.

In the bottom EES bundle, the water is preheated, evaporated, and presuperheated. The steam flow is countercurrent to the helium flow passing over these tubes, which is in keeping with conventional steam-generator practice. It also assures compliance with requirements for maximum utilization of the installed heat-transfer surface, as well as providing maximum stability with up-flow boiling.

Crossover tubes lead the presuperheated steam to the finishing superheater, which is arranged for co-current flow above the EES bundle. This arrangement is required to minimize tube metal temperatures and further enhances stability in this section. Superheated steam at controlled conditions leaves the SH II bundle via tubes arranged in an annular space between a support cylinder outside the reheat header, and a gas shield surrounding it. These tubes lead to the superheater subheaders located below the EES bundle. Superheater subheader extension tubes penetrate the PCRV floor through the primary and secondary closures and are collected in the main steam ringheader, which is connected to the main steam piping.

c. Reheat Steam Circuitry

Cold reheat steam enters each steam-generator module through the annulus between the cold outer and hot inner reheat pipes, the cold reheat pipe being used as the structural support for the reheat bundle. Both of these pipes penetrate the PCRV floor through the primary and secondary closures and extend to the top of the module assembly, where the steam is discharged from the annulus to the reheater (RH) bundle. The RH bundle, also composed of helically wound tubes, is the uppermost heating surface

in the steam generator and is arranged for countercurrent flow which, in this case, is dictated by structural and remote-tube plugging access requirements. Hot reheat steam is carried via the central reheat pipe in the steam generator through the primary and secondary closures and returns to the intermediate pressure (IP) turbine.

d. Primary Closure

The primary closure is a hemispherical head that forms the inner pressure barrier for the steam-generator module penetration in the PCRV (Figure 13). It is located at the inner face of the bottom head where it mates with the extension of the PCRV penetration liner. A gas-tight seal between the closure and the penetration-liner flange is required to prevent leakage of contaminated helium under accident conditions. The primary closure is penetrated by all feedwater and steam subheader-connecting tubes and the reheater connecting pipe, all of which utilize thermal sleeves to control thermal stresses during steady-state and transient operating conditions. The complete assembly of heat-transfer bundles is supported by the primary closure, which transfers the load through the penetration liner to the PCRV (Figure 14).

e. Secondary Closure

The secondary closure serves as a secondary containment for the reactor system and is attached to the lower end of the steam-generator penetration liner. The secondary closure has the configuration of a tube sheet and is penetrated by all high-pressure connecting tubes and the reheater connecting pipe. All tubes penetrating the secondary closure are equipped with thermal sleeves similar to those at the primary closure. The differential thermal expansion between the penetration liner, and the RH inlet-connecting pipe is accommodated by a set of bellows, which constitute a part

LC75981

Figure 13. Primary Closure Assembly

HT67916

Figure 14. Steam Generator Shop Assembly

of the secondary containment. The differential thermal expansion between the tubes and the penetration liner is accommodated with expansion loops in the tubing.

3.1.2. Design

3.1.2.1 Thermal Design

a. Main Steam Section Sizing

With the many parallel steam circuits and the high helium-to-steam temperature differences, equal heat transfer to each tube is desirable. Equal tube lengths are, therefore, required in each of the parallel heat-transfer circuits. The fulfillment of this requirement results in a modified in-line bundle configuration with the number of tubes in each row increasing stepwise with bundle radius and with tube back-spacing varying from cylinder to cylinder to produce a uniform bundle height. The bundle, then, cannot be treated with standard correlations for either staggered or in-line arrays. The calculated gas-side heat-transfer coefficients are based on modified correlations for in-line tube arrays and the calculated gas side coefficients were reduced for sizing purposes. Adequate fouling resistance factors are used for the evaporator region. The heat transfer to various lead and crossover tubes is included for sizing purposes.

In the event of tube failures in any of the heat surfaces sections, the failed tubes must be plugged, thereby reducing the effective heat-transfer surface, which must be considered in sizing each module. Generally, the criteria applied to plugging were that the plant should be capable of generating full power with a reasonable fraction of the main steam heat-transfer surface area plugged in each module. The plant is capable of operating at full power with 11% of the area plugged in each module, in the most severe plugging situation.

The crossover tubes, leading from the top of the EES bundle to the top of the SH II bundle, must pass between the SH II bundle and the outer shroud. The annular passage provided for these crossover tubes allows some flow to bypass the SH II bundle. Steam outlet tubes leading from the bottom of the SH II bundle pass through guide rings in an annulus formed by an inner gas baffle and the EES support cylinder. Here, too, gas bypass flow is required to be taken into account. This bypass flow must be considered for sizing purposes as it reduces the total flow available for heat transfer to the steam-generating section. Suitable flow restrictions are necessary to limit the gas bypass flow to tolerable limits over the full operating range.

b. Reheater Section Sizing

Sizing of the RH section is generally similar to that of the main steam sections. A gas-side heat-transfer uncertainty allowance, as well as an allowance for plugging tubes was selected as the design basis. The heat transfer in the lead-in and lead-out tubing to the bundle is included for sizing purposes. Regenerative heat transfer in the concentric pipe, which has the effect of decreasing the logarithmic mean temperature difference (LMTD) in the RH bundle for a given gas-inlet temperature, is an important factor in sizing and must be accounted for.

c. Tube Design Temperature

To determine tube-metal design temperatures for the main steam sections, hot-spot factors must be considered. In detail, these factors include:

1. Coil curvature effect on steam-side mass-flow/pressure-drop characteristics.

2. Tube length imbalance. (The inner row of tubes in the economizer evaporator presuperheater section is reduced in length to compensate for coil curvature effects.)
3. Circumferential gas-side film coefficient variations.
4. Control dead bands and instrument measurement uncertainties.
5. Plugged tubes.
6. Module-to-module deviations from the average gas flow.
7. Hot gas streaks.
8. Gas channeling.
9. Water-side sensitivity to variations in heat transfer among parallel tube circuits.
10. Gas-side mixing with gas channeling and plugged tube cases.

The resulting tube-metal design temperatures are shown for the main steam sections in Figure 15 along with profiles for mean wall temperature for the average tube. The tube length required for a once-through circuit of the main steam section was obtained by joining tubes of selected materials and wall thicknesses as required by design temperatures and pressures.

The permissible amount of SH II bypass flow is determined by the tube-metal temperatures in the EES bundle outer tube row.

The effect of coil curvature on circuit resistance is substantial and affects steam side stability. By taking advantage of variations in manifolding pressure losses at the subheader tube junction, the effect of coil curvature is

LC71784-A

Figure 15. Tube Design Temperatures

annulled. In this instance, the rows of tubes with smaller radii of curvature and higher flow resistance are connected to positions in the steam subheader manifold with minimal flow resistance, thereby equalizing parallel-circuit resistance characteristics.

Tubes between subheaders are intermixed between the EES and SH II bundles so that an inner cylinder tube in one bundle does not connect to an inner row tube in the other bundle. This arrangement minimizes the effects of plugging a subheader pair on the metal temperature of the tubes remaining in service.

The requirement for static and dynamic stability over the operating-load range between 25 and 100 percent of full load made is necessary to install stabilizing nozzles at the inlet of each once-through circuit.

Design temperatures for the reheater are based on hot-spot factors similar to those for the main steam section. The regenerative heating in the concentric reheat pipe must be limited to control tube-metal temperatures at the outlet of the reheater. This is achieved by a thin, nearly stagnant steam layer within baffles on the inner side of the hot reheat pipe.

3.1.2.2 Materials

The materials for pressure parts and attachments thereto used in the steam generator are required to conform to the rules of the ASME Code, Section III, paragraph N-310. These materials were procured under material specifications with requirements as to quality, sampling, and inspection that exceed the ASME Code requirements. The materials for major support structure parts must conform to the ASME Code, Section II requirements and were procured under equally strict specifications as the pressure part materials. Minor support structures and attachments that encounter no significant pressure, flow, or restraint are purchased under various commercial specifications.

The deciding factors in the selection of the PSC steam generator materials were their strength in the design temperature range, their weldability, their thermal conductivity, their metallurgical stability and foremost their corrosion resistance.

Table 7 lists all the major materials used in the manufacture of the pressure parts and structural components of the PSC steam generators. As can be noted in this table no stainless steel was used.

3.1.2.3 Stress Analysis

a. General Requirements

All pressure parts of the steam generator are required to be designed, analyzed, fabricated, and tested in accordance with the 1965 ASME Boiler and Pressure Vessel Code, Section III, including all addenda through 1966. Structural and nonstructural parts, although not covered by the requirements of the ASME Codes, were generally designed and analyzed in accordance with the principles of the same code to assure their structural integrity under all conceivable operating conditions for the design plant life. Various code cases which supplement the ASME code and were applicable to the design and fabrication of the steam-generator modules were considered in the design and analysis of steam generator components.

The PSC steam generators are designed in accordance with the ASME Nuclear Vessel Code for higher temperatures than any other commercial power plant vessel designed to the same Code in the U.S.A. In addition, the PSC steam generators are designed for an operating life of 210,000 hours over a 30 year design life rather than the typical Code life time of 100,000 hours. Special allowable stresses which reflect the 210,000 hour design life are being used for certain components which operate in

TABLE 7
FORT ST. VRAIN NUCLEAR POWER PLANT
STEAM GENERATOR MATERIAL SELECTION

Section	Material
Main steam/water tubing and piping and associated components	C C - 1/2 Mo C - Si 1 Cr - 0.2 Mo 2-1/4 Cr - 1 Mo 1/2 Cr - 1/2 Mo Ni-Fe-Cr
Reheater steam tubing and piping and associated components	1-1/4 Cr - 1/2 Mo 2-1/4 Cr - 1 Mo Ni-Fe-Cr
Main steam/water sections support structure and reheat sections support structure	2-1/4 Cr - 1 Mo Ni-Fe-Cr
Primary closure	2-1/4 Cr - 1 Mo Ni-Fe-Cr
Secondary closure	2-1/4 Cr - 1 Mo Ni-Fe-Cr C - 1/2 Mo
Shroud	2-1/4 Cr - 1 Mo Ni-Fe-Cr 1 Cr - 1/2 Mo
Baffles and cover sheets	1 Cr - 1/4 Mo Ni-Fe-Cr
Insulation	Metallic Kaowool Silica felt

the creep range. These allowable stresses are lower than the typical Code allowables. As a consequence of this high temperature, long design life and low allowable stresses in the design temperature range the designers of the steam generators were faced with material property, thermal stress and other problems.

Most of the Code approved materials have relatively low allowable stresses at the design temperature levels of the PSC steam generators. High design temperatures and low Code allowables result in relatively thick pressure parts. Large wall thicknesses are required to maintain the pressure stresses below Code allowables at those high temperatures, create, however problems in terms of thermal stresses. Compactness of the design restricts the ability to provide for tubing expansion thereby increasing end reactions on some components. In other areas a reduction of flexibility to limit vibrational stresses and amplitudes equally results in an increase of tube end loads.

This together with the thermal loadings during operating transients makes the design of many steam generator components highly critical.

Due to configuration of major support structures and gas baffles, substantial thermal gradients were encountered. Local strain concentrations and piping reactions added to thermal and primary loadings required particularly careful analysis to show the structural adequacy of the parts.

Additional requirements were that the steam generators withstand shock loads in any direction during shipping and erection and plant design earthquake loadings.

A great deal of complex and time consuming analysis was required to predict the structural behavior of the components in the high temperature range and satisfy the

ASME Code requirements for a Section III Stress Report for the pressure and structural parts in the steam generators. In many cases new analysis methods had to be developed because standard methods could not be applied.

b. Steam Generator Loads

The analysis of Code parts was performed in compliance with stress limits as illustrated in Figure 16 taken from Section III of the ASME Code.

The loadings considered in the analysis were the flow media pressures, helium drag, dead weight, bending stresses, vibration stresses, discontinuity stresses, pipe loads, thermal stresses, earthquake loads, as well as all dynamic loads imposed on the steam generator due to transient operating conditions.

In expectation of the ASME Code Section III revision, all operating transients were checked for the new definition of transients which establishes four categories of transient conditions for analysis purposes. These are:

1. Normal conditions (Start-up, steady-state operation and shut-down).
2. Upset conditions (operator error or equipment malfunction not requiring shut-down).
3. Emergency conditions (operator error or equipment malfunction/failure requiring shut-down).
4. Faulted condition (postulated accidents for plant safety analysis).

The stress allowables are unchanged from Code values for the first two conditions. For emergency conditions the primary stress limit is increased to yield strength and the primary plus secondary stress limit ($3 S_m$) is

Stress Category	Primary			Secondary Membrane plus Bending	Peak
	General Membrane	Local Membrane	Bending		
Description	Average primary stress across solid section. Excludes discontinuities and concentrations. Produced only by mechanical loads.	Average stress across any solid section. Considers discontinuities but not concentrations. Produced only by mechanical loads.	Component of primary stress proportional to distance from centroid of solid section. Excludes discontinuities and concentrations. Produced only by mechanical loads.	Self-equilibrating stress necessary to satisfy continuity of structure. Occurs at structural discontinuities. Can be caused by mechanical load or by differential thermal expansion. Excludes local stress concentrations.	(1) Increment added to primary or secondary stress by a concentration. (notch) (2) Certain thermal stresses which may cause fatigue but not distortion of vessel shape.
Symbol	P_m	P_L	P_b	Q	F
Combination of stress Components and Allowable Limits of Stress Intensities.	<pre> graph TD Pm[Pm] --> S1((Sm)) PL[PL] --> S2((1.5 Sm)) Pb[Pb] --> S3((1.5 Sm)) S1 --> S4((1.5 Sm)) S2 --> S4 S3 --> S4 S4 --> S5((3 Sm)) Q[Q] --> S5 S5 --> S6((Sa)) F[F] --> S6 </pre> <p>The flowchart illustrates the combination of stress components and allowable limits. It starts with three input boxes: P_m, P_L, and P_b. Arrows from P_m and P_L point to a circle containing S_m. An arrow from P_b points to a circle containing $1.5 S_m$. Arrows from these two circles point to a third circle containing $1.5 S_m$. An arrow from this circle points to a fourth circle containing $3 S_m$. An arrow from a box containing Q also points to the $3 S_m$ circle. An arrow from the $3 S_m$ circle points to a final circle containing S_a. An arrow from a box containing F also points to the S_a circle. The boxes for P_m, P_L, P_b, Q, and F are arranged in a row at the top, with arrows pointing down to the flowchart.</p>				

LC74429

Figure 16. ASME Code, Section III Stress Limits

eliminated. In the case of faulted condition, only primary stresses are controlled by limit analysis. This Section III Code revision was finally adopted by issue of the Summer 1969 Addendum to ASME Code, Section III. Table 8 gives an account of all the operating transients analyzed under the above Code.

Faulted conditions are not shown since the analysis conducted must only prove the possibility of a safe shutdown without danger to the public. No requirement as to further operability of the plant exists.

c. Reduced Allowables for 210,000 Hours Design Life

In general, allowable stresses are taken from the tables in the applicable ASME Code section, however for sustained loading of steam generator components at high temperatures the primary membrane stress allowables were reduced to assure structural adequacy of these components over 210,000 hours life. The reduced allowables were established by applying the normal Code criteria to 210,000 hour creep data. The creep data for 210,000 hours were obtained by a log-log extrapolation from the existing 100,000 hour curves. Plots of the reduced allowables for the extended life are presented in Figure 17 for two of the major materials used in the PSC steam generators.

3.1.2.4 Design Problems

a. Reheat Header

The external pressure design conditions for this part require, under code rules, a great deal more wall thickness than for a corresponding internal pressure design condition. The parts so designed are particularly sensitive to thermal transients. A careful and extensive analysis was required to assure that thermal stresses

TABLE 8

FORT ST. VRAIN NUCLEAR POWER PLANT
STEAM GENERATOR - TRANSIENT OPERATING CONDITIONS

Transient	No. of Cycles
<u>Normal Conditions</u>	
Start-up from refueling status	79
Start-up with full helium inventory	316
Shut-down to refueling status	40
Shut-down with full helium inventory	50
Normal load increase	18500
Normal load decrease	18500
<u>Upset Conditions</u>	
Reactor scram from full load	100
Turbine trip from full load	50
Various loop shut-downs	100
Various steam leaks with subsequent loop dump	30
<u>Emergency Conditions</u>	
Loss of helium circulator	25
Loss of steam driven feed pump	5
Various steam leaks with subsequent dumping of wrong loop	2
Simultaneous loop dump and system depressurization	5
Primary coolant system depressurization	5
Loss of helium circulator steam turbine drive	1
Loss of helium circulator steam turbine drive and primary system depressurization	1

LC84551

Figure 17. Stress Allowables for 210,000 Hours

in the RH header do not exceed the 3 Sa limit as defined by the applicable code case under the various postulated emergency transients.

Figure 18 shows the machined reheater collector in the shop.

b. Thermal Sleeves

Thermal-sleeve loadings include helium and steam pressure, tubing reactions, and local thermal stresses, particularly during transients. To reduce local temperature gradients and thermal stresses so that code allowables could be met, selective insulation of portions of the thermal sleeve was required. In addition, a special design had to be adopted for the superheater thermal sleeves at the secondary closure (Figure 19).

c. Structural Parts

The loadings on the tube-support structures include dead weight, helium drag, and differential thermal-expansion stresses in the tubing adjacent to the support structure. With respect to the support structure, the dead weight and helium drag are treated as primary stresses. Loads from differential thermal expansion between the tubes and support structure (Figure 20) are generally considered to be secondary stresses; however, for regions in the support plates where strain cannot be relieved, these thermal stresses must be treated as primary stresses. To establish the stress allowable for certain kinds of thermal expansion loadings, it is necessary to evaluate the relative stiffness of all parts transmitting the loads.

d. Nonstructural Parts

Relatively thin internal gas-baffles serve to guide the flow at the radial boundaries of the bundle and are

135-98-14

Figure 18. Reheater Collector

63

LC77835

Figure 19. Secondary Closure Thermal Sleeve - Superheater Tube

135-98-5

Figure 20. Tube Support Structure with Coiled Tubes

attached along all edges to structural parts. The baffle temperatures are significantly different from the temperatures of the adjacent structural parts during operation and transients. The baffles and their attachment welds were evaluated on the basis of an elastic-fatigue analysis and code-fatigue charts.

e. Wear Protection

The possibility of surface damage to certain components in motion under an inert helium environment suggested that protection be provided against possible damage from relative motion with other components.

As an added protection sleeve and wedge combinations were employed in all areas where heating-surface tubes penetrate their respective support structures. The design principle involved here is that the sleeve material must have a higher wear resistance than the support-plate material, and the wedge material, which is in contact with the tube, must be similar to the tube material. In addition, the sleeve material must have a coefficient of thermal expansion less than that of the tube material to avoid a loose fit between tube and sleeve under relative thermal expansion.

A typical sleeve and wedge assembly is shown in Figure 21.

Heat-transfer tubing that is subject to relative motion and in contact with baffles or thermal shields is flame-sprayed with chromium carbide to protect the tubes from possible damage.

Structural parts that are in sliding contact with each other are protected against significant damage by the application of a weld overlay of cobalt-base alloy in appropriate areas.

HT69284

Figure 21. Sleeve and Wedge Assembly

3.1.3 Fabrication

The fabrication of the steam generators for the Fort St. Vrain plant was executed by the GGA Manufacturing Division partly in GGA's own shops and partly at the General Iron Work shops in Denver, Colorado.

The feedwater ringheader of all steam generator modules with their integral trim valves were fabricated by Sulzer Brothers in Switzerland.

For installation purposes the fabrication of the individual modules was carried out in several major subassemblies. These were:

- a. The primary assembly, consisting of the primary closure, the shroud, the heat transfer bundles. (See Figure 14)
- b. The secondary assembly consisting of the secondary closure, the interspace tubing, both the feedwater inlet and superheater outlet ringheaders with their pipe connections.
- c. Various reheat pipe sections.

All subassemblies were joined in the field to their mating assemblies and to the power plant piping respectively.

3.1.3.1 Installation

The installation of a primary assembly of the steam generator module was accomplished by lifting it into the PCRV through the open central access penetration in the PCRV bottom head. A lifting rim was provided on the top of the steam generator shroud for this purpose. The total weight of the primary assembly is about 62,000 lbs. The assembly was then moved laterally to a position above its PCRV penetration and lowered onto the penetration liner flange, which supports the complete assembly.

Prior to installation the primary assemblies, the secondary assemblies were installed from below the PCRV and welded to their respective penetration liner extension. After bolting the primary closure down onto its flange, all necessary tube welds were made and the individual reheat pipe sections were installed.

3.1.3.2 Quality Assurance

Quality assurance for the shop and field assembly of steam-generator module fabrication meets the requirements of the ASME Code, Section III, Article 6. Specific quality assurance functions are supplemented by GGA quality assurance requirements, which are more stringent than those of the Code. One of these is the GGA requirement for helium leak-testing of all steam/water pressure parts and the primary-closure dome including welds thereon. The helium leak-testing is carried out in accordance with GGA procedures and limits the integrated leak rate for each module to 0.1 cu ft/hr of helium at 1 atm.

Figure 22 through 24 show the steam generator modules in various stages of fabrication, while Figure 25 shows a completed primary assembly prior to attaching the shroud at the left and another completed primary assembly at the right. Figure 26 shows a secondary assembly being readied for shipment.

3.1.4 Controls

3.1.4.1 Plant Control

The plant control system permits boiler-follow turbine action with the reactor automatically following the load demand. The plant is designed to automatically follow load from about 25% to 100% load at a maximum rate of change of about 5% per minute.

In addition to accommodating system disturbances resulting from normal load changes, the control system is also capable of handling transient conditions resulting from various abnormal and emergency conditions.

The overall plant control system is designed to permit automatic operation of the plant throughout its control range to maintain controlled steam temperatures and pressures at the turbine throttle valves. During normal operation a load demand signal will act upon the turbine throttle valves causing them to open or close depending on the direction of the load change. The change in throttle valve position causes a deviation in steam pressure from setpoint and the pressure error signal combined with

K85352

Figure 22. Sulzer Feedwater Ringheader

K75037

Figure 23. Reheater Support Structure

HT65991

Figure 24. Steam Generator Assembly

K84903

Figure 25. Primary Assembly

HT78193

Figure 26. Secondary Assembly

a feedforward signal indicating a change in turbine steam flow causes a change in feedwater flow setpoint which is adjusted until the desired load level and steam pressure are obtained. Main steam temperature anticipation is provided by a feedforward signal from feedwater flow, which is combined with the steam temperature error to produce the required change in helium circulator speed. The combination of change in main turbine first stage pressure plus the reheat temperature error will cause a change in the reactor power setpoint. Actual power output measured by means of a neutron flux instrument is compared against the new setpoint and the error signal will adjust the control rods resulting in a change in reactor power. A simplified plant control diagram is shown in Figure 27.

3.1.4.2 Steam Generator Control

The steam generator control system is designed for full automatic operation over the normal control range of the unit and is designed for boiler-follow-turbine mode of control where steam generator input and output is changed in accordance with turbine demand.

a. Main Steam Pressure

The pressure of the main steam at the inlet to the turbine is controlled to approximately 2400 psig over the control range of the unit by feedwater flow. This is accomplished by a pressure controller with load change anticipation from the first stage pressure of the turbine. Steam pressure is compared to the setpoint and an error signal is generated which in combination with the first stage pressure signal will cause a change in the feedwater flow setpoint.

b. Main Steam Flow (Feedwater Flow)

Feedwater flow control to the desired setpoint is accomplished by the combined action of flow control valves and feedpump speed control. The pressure drop across the flow control valves is measured and this signal is used to adjust the speed of the turbine driven pumps to produce the required valve pressure drop.

LC69601-A

Figure 27. Fort St. Vrain Nuclear Power Plant Simplified Control Diagram

c. Main Steam Temperature

The main steam temperature at each steam generator outlet is controlled over the control range at about 1000°F by varying the helium flow in each steam generator loop. This is accomplished by an individual loop temperature controller which determines a helium circulator speed setpoint for its associated loop. The individual circulator speed controllers adjust the throttle valves of the helium circulator drive turbines, thereby varying the circulator speed and helium flow. Anticipating signals from feedwater flow to circulator speed are also provided.

d. Reheat Steam Temperature

A temperature signal from average reheater outlet temperature is compared against the setpoint. Deviations from the setpoint generate an error signal which acts upon the reactor control rod controller causing a change in reactor power level. The change in the reactor power level causes a change in helium temperature in the desired direction. A feedforward signal proportional to turbine steam flow is used as an anticipation for reheat steam temperature error. In addition, manually controlled RH attemperators are provided at each module inlet for reheat steam temperature balance purposes with a remotely controlled main spray valve in each steam generator loop.

e. Helium Flow (Circulator Speed)

Under normal operating conditions helium circulator speed is automatically controlled as a function of main steam temperature. Under other conditions where helium flow is required for reactor core cooling without simultaneous demand for generator load, the helium flow is controlled independently from the steam generator control system either manually, semi-automatically or full automatically by the plant protective system, depending on the condition.

f. Helium Temperature

Helium temperature is not directly controlled. The temperature of the helium entering the steam generator is primarily a function of the helium flow rate, cold helium temperature and the reactor power level. Helium flow and reactor power are adjusted to maintain the desired superheat and reheat steam temperature with the cold helium temperature primarily determined by the steam generator performance, thus determining a unique hot helium temperature.

3.1.4.3 Emergency Controls

The power plant is equipped with a plant protective system in accordance with safety requirements and nuclear power plant practice. This system consists of instrumentation and controls required to initiate automatic corrective action upon the onset of an unsafe operating condition. These actions are directed towards either returning the plant to safe operating conditions or shutting down vital plant equipment in a safe manner without danger to the public. The plant protective system will override all other manual or automatic controls if activated.

Under abnormal and/or emergency conditions and during start-up, shut-down, emergency cooling, etc. the feedwater flow will be controlled independently from the main steam flow and pressure in accordance with other requirements than load demand. Control of feedwater flow under these conditions will be executed either manually, semi-automatically or automatically by the plant protective system, depending on the conditions.

3.1.5 Research and Development

With the advent of nuclear power plant operating experience having become available to a greater degree, certain problems with nuclear power plant components were recognized in general by the nuclear power industry. This fact together with GGA's own experience obtained from the Peach Bottom plant was a decisive factor in the decision to conduct a much more extensive research and development program to aid in the design and

performance of the PSC steam generators. In addition, the decision to utilize helically coiled forced flow steam generator surfaces necessitated a number of research programs because no design or operating experience with this type unit was available elsewhere.

To describe all R&D programs in support of the Fort St. Vrain steam generators would exceed the scope of this paper. Therefore, only the major programs, their reasons and their results shall be mentioned. A summary of all steam generator research programs is given in Table 9.

3.1.5.1 Vibration Tests

The overall vibration testing done for the PSC steam generator consists of many different phases. Some of these tests were done on test modules, others were done with a completed and heavily instrumented steam generator module on a test stand. In addition, vibration testing will still be conducted during the "hot tests" in the plant, prior to actual operation.

a. Tube Bundle Prototype Tests

This test was conducted by Sulzer Brothers on a full-scale mock-up of a portion of the steam generator module. Its objective was to measure the vibration response of the unit by determining tube frequency, response of the tube helixes to flow excitation, response of lead-in and lead-out tubes to flow excitation, Helmholtz effect on vortex shedding and Strouhal number. The results of the tests which were conducted with air flows up to 120% of the design condition indicated that vibratory stresses in the tube bundle of the Fort St. Vrain steam generator are well below the design limits and sufficient safety margin exists. Figure 28 gives an overall view of the Sulzer vibration test facility.

b. Lift Coefficient Tests

This test was also conducted by Sulzer Brothers and served to determine the lift coefficients and friction factors of

TABLE 9

FORT ST. VRAIN NUCLEAR POWER PLANT
STEAM GENERATOR - APPLIED RESEARCH AND DEVELOPMENT

Vibration Tests

1. Sulzer Tube Bundle Prototype Vibration Test
2. Lift Coefficient Test
3. Tube Support Damping Test
4. Development Tests of Steam Generator Sleeve and Wedge
5. Steam Generator Module Air Flow Test
6. Steam Generator Hot Flow Test

Performance Tests

1. Once-Through Boiling Stability Test
2. Gas Mixing Test
3. Reheat Flow Re-entrant Test
4. Tube Flow Resistance Test
5. Steam Generator Module Air Flow Test
6. Steam Generator Hot Flow Test

Materials Tests

1. Stress Rupture Tests on Ni-Fe-Cr Tubing
2. Elevated Temperature Tests for Ni-Fe-Cr Alloy
3. Autoclave Corrosion Tests
4. Development of Welding Procedures
5. Thermal Shock Resistance Tests of Flame Sprayed Coatings
6. Leak Tests of Primary Closure Thermal Sleeve Connectors
7. Thermal Sleeve Forging Development
8. Compatibility of Steam Generator Materials of Construction With the Primary Coolant

TABLE 9 (Cont.)

FORT ST. VRAIN NUCLEAR POWER PLANT
STEAM GENERATOR - APPLIED RESEARCH AND DEVELOPMENT

Materials Tests (Cont.)

9. Spotwelding Tests for Strain Gage Installation
10. Qualification Test for Brazed Thermocouple Penetrations
11. Cyclical Certification Tests for Steam Generator Bellows
12. Reheat Pipe Seal Welding Test
13. Chemical Cleaning Test of Feedwater Ringheader
14. Chemical Cleaning Test of Air Flow Test Module

K65975

Figure 28. Sulzer Vibration Test Arrangement

Karman vortex streets for four different tube bundles with staggered and in-line tube arrays. These tests were conducted in wind tunnels and proved that the tube arrangement selected for the Fort St. Vrain steam generators will not be adversely affected by the dynamic forces due to flow induced vibration, lift coefficients and friction factors.

c. Damping Test

This test was performed to evaluate the damping characteristics of two helical tube and support combinations. The results of the test resulted in the selection of the sleeve and wedge combination shown in Figure 21.

d. Air Flow Test on Steam Generator Module

A completed primary assembly of a steam generator module was tested with ambient air flow at velocities up to and above the operational helium velocity, thereby duplicating the range of forcing frequency which should occur in service. The objective of the test was to verify the predicted induced vibration amplitudes and frequencies of critical pressure parts and support components.

This test showed that actual vibration in all cases was well below predicted values and the analytical methods used for the predictions appear to be conservative.

e. "Hot Test" Run

The air flow test described above will be essentially repeated early this year during the hot test period of the plant, except that helium, heated by compression, will be the forcing medium.

3.1.5.2 Boiling Stability Tests

The objective of this test series was to provide information on steam/water flow stability in the Fort St. Vrain steam generator to

allow 1) to establish the inlet orifice requirements for stable operation over the full control range, 2) to determine the feasibility of plant start-up procedures, 3) to establish limits on various plant operating conditions as for instance minimum load, 4) to check and confirm various analytical methods used in the design, 5) to determine whether serious thermal stresses may exist due to unstable film boiling or tube curvature effect on boiling, 6) to evaluate possible water carry-over into the superheater, 7) to test steam generator materials and bimetallic weld under simulated steam generator operating conditions.

To this end a boiling stability test facility was constructed and operated (Figure 29).

In view of the information developed during this extensive test series, it was determined that the Fort St. Vrain steam generators have satisfactory performance characteristics in terms of static and dynamic stability over their specified operating range.

3.1.5.3 Gas Mixing Tests

This test was conducted to determine the amount of mixing between a hot and a cold gas stream in a tube bundle. The mechanism of gas mixing in a tube bundle involves the transport of heat from a hot gas channel to a cooler adjacent channel, thereby establishing axial temperature profiles through the tube bundle.

A test was conducted, where the actual mixing of hot and cool gas streaks was measured and results in the form of mixing coefficients were obtained.

3.1.5.4 Material Tests

a. Elevated Temperature Tests for Incoloy 800

The ASME Code Section III requires the use of fatigue curves in the analysis of pressure vessels. At the time of the PSC steam generator design the Code approved fatigue curves for Ni-Fe-Cr material only were valid up

K62379

Figure 29. Boiling Stability Test Loop - Schematic

to 800°F. For higher temperatures the Code requires that data be submitted to establish allowable fatigue stresses.

To this end GGA initiated high temperature strain controlled fatigue tests for Incoloy 800 which resulted in fatigue curves, which were used for the design of high temperature components of the PSC steam generators.

b. Autoclave Corrosion Tests

One thousand hour autoclave corrosion tests were performed to identify the effect of fabrication features of the Fort St. Vrain steam generators on the stress corrosion behavior of Sanicro 31 tubing material.

The tests were done using environmental conditions which, from previous experience relating to the Peach Bottom steam generators, were known to be very aggressive and capable of causing pitting and stress corrosion cracking of austenitic steels.

The results of these tests showed no evidence of stress corrosion cracking in any sample, although the Sanicro 31 did show pitting and occasional associated shallow intergranular attack after exposure to the high oxygen-chloride environment.

The distribution of this pitting and the intergranular attack could not be related to weld locations or presence or absence of residual stresses.

It was therefore concluded that the stress corrosion behavior of the Sanicro 31 tubing material was not adversely affected by the investigated fabrication features.

c. Thermal Shock Tests of Thermal Sleeve

The primary closure thermal sleeves are under certain emergency conditions subjected to severe thermal shocks.

These sleeves form a primary boundary against contaminated helium inside the steam generator. A multiple failure would lead to the loss of purified helium from the penetration interspace and in an extreme case would not permit to maintain the overpressure in the interspace.

Sleeve connectors of the feedwater and the superheater tubes were both subjected to thermal shock from operating temperature to ambient temperature and helium leak checked after various cycles.

From the results of the tests as well as from metallurgical examination it was concluded that no problems are envisaged with forged thermal sleeves, where the amount of inclusion is limited and the grain contour follows the design contours of the part.

3.1.6 Schedule

At present the primary assemblies of the steam generators are being installed in the concrete pressure vessel. All secondary modules are installed. The present schedule calls for chemical cleaning of the feedwater and condensate systems immediately following the steam generator installation. This will be followed by a complete hydrostatic pressure test of the steam generators and their associated piping systems. Thereafter the hot flow tests will be conducted, during which the steam generator will be checked for vibrations.

The steam generators proper and turbine piping will be chemically cleaned immediately prior to initial rise to power scheduled for the latter part of this year. Commercial operation of the Fort St. Vrain nuclear power plant is scheduled for early 1972.

4. THE 1100 MW(e) COMMERCIAL HTGR PLANT

The large commercial HTGR plant is a further step in the development of HTGR technology and its associated components. Principally the same as the 330 MW(e) Fort St. Vrain plant discussed before, there are a number of differences other than size, which have a marked effect on the steam generator design. An illustration of a typical 1100 MW(e) HTGR power plant is shown in Figure 30, and a schematic cycle diagram is shown in Figure 31.

The nuclear steam supply system is again housed within a prestressed concrete pressure vessel. Due to the size of this vessel and the required design for earthquake loading, it was not feasible to repeat the Fort St. Vrain vessel design, where the steam generators and circulators are located beneath the reactor core. Instead, the diameter of the vessel was increased and these components are arranged in individual penetrations surrounding the core.

In maintaining the safety philosophy of gravity-fed control rods and downward helium flow through the core, the primary coolant now enters the steam generator penetrations at the bottom, unless costly and complex ducting inside the PCRV would be used to direct the gas flow to the top of the steam generator penetration. Also, in anticipation of more stringent AEC regulations in terms of safety and to ease plant licensing procedures, the complete nuclear steam supply system is located within a conventional reinforced concrete building, which serves as secondary containment.

Furthermore, a completely independent emergency cooling system is provided, consisting of helium to water heat exchangers and helium compressors with associated ducting, piping controls, etc.

The nuclear steam supply system arrangement for the 1100 MW(e) HTGR power plant is shown in Figure 32. The system contains six completely independent primary coolant loops installed within wall cavities of approximately 13 feet diameter. Each primary loop consists of a helium

LC78344-B

Figure 30. 1100 MW(e) HTGR Plant

K70108

Figure 31. 1100 MW(e) HTGR Plant Schematic Cycle Diagram

LC85728

Figure 32. 1100 MW(e) HTGR Plant Nuclear Steam Supply System

circulator and a steam generator. Three additional auxiliary cooling loops are provided, which are located in wall cavities. Because of the multi-cavity design of the PCRVR, access for all major equipment installation and maintenance is through the top head of the PCRVR, rather than a central access penetration common to all equipment.

4.1 Steam Generator

4.1.1 Development

Initial development work on the 1100 MW(e) steam generating system started immediately after the conceptual design of the 330 MW(e) Fort St. Vrain system was completed.

Early during the development a decision was made to use six independent primary coolant loops. This decision was made in order to maintain the steam generator penetrations at a reasonable size, without increasing the overall PCRVR diameter significantly. Furthermore, a six-loop design constituted a much more reasonable size extrapolation for the power plant components, in particular the steam generators and circulators, from the 330 MW(e) PSC system.

The primary coolant entering the steam generator penetration at its bottom end presented a new challenge to the steam generator designer. In following the PSC design having a one-pass forced flow boiler the 1100 MW steam generator required its secondary coolant flow to be in the downward direction. A conceptual design of such a steam generator was pursued (Figure 33) and a great deal of analysis and test work proved its feasibility.

Although a downflow steam generator proved to be a workable solution, two factors led to the decision to abandon its development.

- a. The aversity of the U.S. power industry to downflow heating surfaces in conventional fossil fired units as for instance downflow superheaters, rear pass tubes, etc.
- b. The penalty to be paid in either water/steam side pressure drop or restrictions in low load operation to definitely assure

K75041

Figure 33. 1100 MW(e) HTGR Plant Steam Generator Concept (Downflow)

static and dynamic flow stability over the full operating range of the steam generators could not be justified.

With the single gas pass concept abandoned, a new design had to be developed which allowed for internal gas flow direction changes within the steam generator. In maintaining the helical tube-perforated support plate concept the most logical arrangement evolved with the center support spine of the steam generator heat transfer bundles serving as the gas duct leading the helium to the top of the steam generator.

Here, two different concepts were pursued with additional effort towards a detailed design and analysis. One steam generator was selected using a single tube bundle for the main steam section surrounding the center duct. The integral reheater in this steam generator then was located inside the center duct and consisted of a U-tube bundle (Figure 34). The advantage of this design lies in the reduced length of the steam generator, which in turn allows a reduction in thickness of the pressure vessel bottom head. Inherent disadvantages are the somewhat increased diameter of the steam generator, requiring an increase in penetration diameter, and the relative inefficiency of a U-tube reheater bundle in parallel flow arrangement. Also, the relative large tube size required for the reheater section, to accommodate structural and layout requirements poses problems in manufacturing as well as sizing, since the loss of a single large tube results in a greater percentage of ineffective heating surface than for a small diameter tube.

The other steam generator arrangement under consideration employed the same heat transfer arrangement for the main steam/water section. It also utilized a helically wound reheater bundle similar to the PSC steam generator, except that in this case the reheater bundle was located beneath the main steam tube bundle.

This arrangement resulted in somewhat greater length of the overall steam generator, requiring a corresponding greater thickness of the PCRV bottom head--a cost disadvantage.

LC77921

Figure 34. 1100 MW(e) HTGR Plant Steam Generator Concept (U-Tube Reheater)

On the other hand this design permitted the diameter of the penetration to be maintained and using a design similar to the Fort St. Vrain steam generator design, permitted closer reliance on design and manufacturing experience previously gained. It was this experience factor which affected the decision to adopt this steam generator concept, after weighing all pros and cons of the discussed arrangements.

4.1.2 Design

The detailed design of the 1100 MW(e) commercial HTGR plant steam generators is being carried out by GGA in cooperation with Foster Wheeler Corporation, of New York, a company with broad experience in the design and manufacture of steam generating equipment.

4.1.2.1 Description

Six identical steam generators are located in cavities symmetrically oriented in the PCRV around the reactor core. A sectional side view of a steam generator is shown in Figure 35, their design conditions are given in Table 10.

As shown in Figure 35, the heat transfer section for the main steam bundle consists of a helically coiled tube bundle. Feedwater enters this main steam section at 340°F and 3135 psig and leaves at approximately 955°F and 2500 psig. The steam leaves each unit through the superheat tubesheet to the steam pipe headers on the steam generators. The steam is then led to the high-pressure stage of the turbine. After expanding through the high-pressure turbine, the steam is piped back to the turbine drives of the helium circulators positioned above the six steam generators. After further expansion through the circulator turbine, the cold reheat steam, now at approximately 625°F and 620 psig, is piped directly to the reheater inlet pipe header. The heat transfer bundle in the reheater section is also of a helical coiled tube design. The hot reheat steam leaves the steam generators at 1001°F and 575 psig and flows to the intermediate-pressure section of the turbine.

The helium flow path through the steam generator begins in the cross ducts that radiate from the bottom of the core and extend into the

Figure 35. 1100 MW(e) HTGR Plant Steam Generator

TABLE 10
1100 MW(e) COMMERCIAL HTGR PLANT
STEAM GENERATOR DESIGN CONDITIONS

	100% Steam Flow	25% Steam Flow
<u>Load</u>		
Generator Output (MW)	1108.7	261.8
Station Output (MW)	1092	246.2
<u>Helium</u>		
Flow (lbs/hr)	10,464,000	3,200,000
Outlet Pressure (psia)	700	606
Inlet Temperature (°F)	1377	1182
Outlet Temperature (°F)	636	504
Pressure Drop (psi)	15.25	1.70
<u>Feedwater/Main Steam</u>		
Flow (lbs/hr)	7,302,000	1,827,600
Outlet Pressure (psia)	2515	2421
Inlet Temperature (°F)	340	255
Outlet Temperature (°F)	955	950
Pressure Drop (psia)	627	39
<u>Reheat Steam</u>		
Flow (lbs/hr)	7,230,000	1,794,000
Outlet pressure (psia)	590	149
Inlet Temperature (°F)	625	509
Outlet Temperature (°F)	1001	1001
Pressure Drop (psi)	45	11

steam generator cavities. The hot gas flows through a passage formed between the cross ducts and the steam generator inlet plenum that feed the reheat tube bundle. From this plenum, the gas, at 1377°F and 680 psig, flows downward through the helical coil reheat bundle. As the helium discharges from the reheater, it flows upward through an annular duct and then downward through the main heat transfer section of the steam generator. The helium exits at about 630°F and flows upward through an annulus formed between the steam generator shroud and the penetration insulation cover plate to the helium circulators.

a. Main Steam Tube Bundle

Feedwater enters each module via a tubesheet located below the primary closure. Orifices are installed at the feedwater tubesheet to provide a stable and uniform steam/water flow distribution through the main steam bundle. From the tubesheet, tubes are routed to the lower end of the main steam bundle. Each tube lead becomes one coil in the helical bundle. The tube diameter, wall thickness, and material, change to meet the design stress requirements and to give improved hydrodynamic stability characteristics. At the top of the main steam bundle, the tubes are collected and subheadered. Tube lead-outs from the subheaders are routed down to the superheater tubesheet at the lower end of the unit. In the helium inlet plenum, the superheater lead-outs are collected and pass through the penetration to the superheater tubesheet.

The main steam bundle is supported by perforated plates through which the tubes are coiled. These plates are secured to the main support cylinder which is supported off the penetration liner. The bundle leads are exposed to differential thermal expansion movements between their attachment at the tubesheet and support point at

the plates. To account for this, flexibility is designed into the superheater and economizer leads in the interspace region between the floor plate and primary closure.

b. Reheat Tube Bundle

Tubesheets are used to collect the reheat steam to and from the steam generator modules. Cold reheat tubes extend from the cold reheat tubesheet up through the floor plate and coil directly into the heat transfer bundle. At the top of the bundle the tubes are joined to return tubes leading to the bottom of the bundle and extend from this region through the floor plate and are collected into the hot reheat tubesheet.

The reheat bundle is supported by perforated plates through which the tubes are coiled. These plates are secured to the main support cylinder which is supported off the penetration liner. The bundle leads are exposed to differential thermal expansion movements between their attachment to the tubesheet and support point at the plates. To account for this, flexibility is designed into the hot reheat and cold reheat leads in the interspace region between the floor plate and tube sheet.

c. PCRV Penetration Closures and Steam Generator Access

Access for tube plugging of the steam generator is provided. Tube plugging is accomplished by isolating a steam generator, disconnecting the inlet and outlet piping to the main steam or reheat circuit, and manually repair welding or plugging any damaged tube at the outside surface of the tubesheet. This can be done on any of the steam generators during a plant shutdown.

Two types of closures are associated with each steam generator. The six cavity top closures are located within the top head of the PCRV and also form the support for the helium circulator units. The closures are concrete plugs bolted to the penetration liner.

The steam generator cavity bottom closures, located in the PCR/V bottom head, provide for entry of the water and main steam pipes to the steam generators. The penetration closure is mounted on a flanged thermal sleeve extension of the penetration liner. The closures are sealed by two concentric metal O-ring seals, with buffer helium between.

The means of transferring the pressure load from the closures to the concrete is provided by two independent sets of shear anchors in the concrete. Closures are designed to withstand combined accident and pressure loads. Low alloy steel is used for the flanges and closures.

4.1.2.2 Materials

The materials used in the steam generators range from carbon steel to nickel base alloys. The use of a specific material depends upon the particular design conditions of temperature, pressure, corrosion resistance, weldability, and long term creep properties.

Carbon steel is used in low design temperature items such as the feedwater tubesheet and feedwater tubes. Various grades of low alloy steel are used for the primary closure, cold reheat tubesheet, portions of the shroud, and economizer-evaporator tubes and their support structure. Nickel-iron-chrome alloy is used for the superheater tubes, reheater tubes, superheat tubesheet, hot reheat tubesheet and portions of the shroud. High strength nickel base alloy is used in the superheater and reheater support structures.

Materials for all parts covered by ASME Codes conform to the requirements of the applicable Codes. Some high strength nickel alloys which so far have not been granted Code approval in the high temperature range will be used for Non-Code components because of their favorable high temperature characteristics.

Table 11 summarizes the material selection for the steam generators of the 1100 MW(e) HTGR power plant.

4.1.2.3 Analysis

The basic thermal design of the large steam generator follows in principle, the ~~same~~ design rules that were established for the PSC steam generator design. Sufficient allowances in design temperature and material selection are being made for factors such as temperature unbalances, tube plugging, hot gas streaks, gas bypass, etc.

The stress analysis again is being done in strict accordance with the applicable Section III of the ASME Code as amended by the Summer 1970 Addendum. This addendum now includes design rules for the various types of transient conditions. The respective Code analysis rules are presented in the Code chart in Figure 36 (compare to Figure 16). The type of transient operating conditions encountered in the 1100 MW(e) HTGR power plant are very similar to the conditions in the Fort St. Vrain plant,

TABLE 11

1100 MW(e) COMMERCIAL HTGR PLANT
STEAM GENERATOR MATERIAL SELECTION

Section	Material
Main steam/water tubing, piping and associated components	1/2 Cr-1/2 Mo 1-1/4 Cr-1/2 Mo 2-1/4 Cr-1 Mo Ni-Fe-Cr
Reheater steam tubing, piping and associated components	C 1-1/4 Cr-1/2 Mo Ni-Fe-Cr Ni-Cr-Mo-Cb
Main steam/water sections support structure	1 Cr-1/2 Mo 18 Cr-8 Ni 2-1/4 Cr-1 Mo Ni-Fe-Cr Ni-Cr-Mo-Cb
Reheat sections support structure	18 Cr-8 Ni Ni-Cr-Mo-Cb
Primary Closure	18 Cr-8 Ni
Shroud	1 Cr-1/2 Mo 18 Cr-8 Ni
Baffles and Flow Shields	1 Cr-1/2 Mo
Insulation	Kaowool Silica felt

- USE DESIGN LOADS (N-441 AND N-448)
- USE OPERATING LOADS (N-441) SPECIFIED FOR NORMAL, UPSET AND TESTING CONDITIONS
N-412 (1) (1), (2) & (3)
- USE OPERATING LOADS (N-441) AND ACTUAL MECHANICAL LOADS (N-448) SPECIFIED
FOR EMERGENCY OR FAULTED CONDITIONS N-412 (1) (4) AND (5) RESPECTIVELY

Figure 36. ASME Code, Section III, Summer Addendum 1969, Stress Analysis

however the number of cycles of some transients and the thermal response of some components under such transients can vary appreciably. This, of course, shifts the severity of certain transient operating conditions to different steam generator components, than in the PSC plant. The transient operating conditions considered in the analysis for the commercial HTGR plant steam generators are tabulated in Table 12.

Steam generator parts which do not fall under the jurisdiction of the ASME Code, such as support structure, gas baffles, etc. are being analyzed by using Section III of the ASME Code as a guide in establishing the structural integrity of these parts for all operating conditions over a 30 year life of the plant.

TABLE 12
1100 MW(e) COMMERCIAL HTGR PLANT
STEAM GENERATOR OPERATING TRANSIENTS

Transient	No. of Cycles
<u>Normal Conditions</u>	
Start-up from refueling status	70
Start-up from full helium inventory	310
Shut-down to refueling status	40
Shut-down with full helium inventory	50
Load increase	20050
Load decrease	20050
<u>Upset Conditions</u>	
Reactor scram from full load	100
Turbine trip from full load	50
Loop shut-down followed by plant shut-down	150
Steam leak followed by loop shut-down	30
Loss of one feed pump	10
<u>Emergency Conditions</u>	
Primary system depressurization	1
Shut-down with three auxiliary cooling loops	3
Shut-down with two auxiliary cooling loops	1
Depressurization and shut-down with two auxiliary cooling loops	1
Rod withdrawal with high flux scram	1
Loop shut-down with helium valve failure	3

4.1.2.4 Design Details

There are various areas in the design of this large commercial steam generator which presented new challenges to the designer. These challenges and problems were due to the size, configuration and general layout of the steam generator.

Some of these problem areas are described below:

a. Effect of Size

Each steam generator has an overall height of approximately 65 feet between the steam and water pipe connections and the top of the shroud. The diameter of the unit is 12 feet. The total weight of the completely assembled steam generator is 364,000 lbs.

Provisions made in the design of this unit to accommodate thermal expansions, which are significant in a unit of this size, operating at high temperatures, included providing the necessary flexibility in the support structure for the heat transfer bundles.

This flexibility, on the other hand, poses problems in terms of support during assembly as well as shipping of the steam generator. Both final assembly and shipping must be executed in a horizontal position because of the steam generator height.

Necessary support features are provided for in the design by a support spine which is keyed to the heat transfer section support system, allowing radial motions and restraining tangential motions.

b. Tubing and Support

Large physical dimensions associated with large temperature differences inherent in modern steam generating equipment consequently result in large differential thermal expansions.

These thermal differentials are especially pronounced in the lead tubes between the tubesheets and the effective heat transfer tubing, since these tubes must absorb the expansion differences (up to 2 inches) generated by the long length of lead tubing without overstressing the tube-to-tubesheet joint or the subheaders above the tubesheets, to which they are connected. Arrangement of these leads therefore must provide for expansion loops with proper consideration given to available access space for making the required weld connections during assembly of the unit.

The heat transfer section tubing in the main steam bundle is composed of various materials ranging from SA-210 to Incoloy 800. The continuous plates supporting the main steam tube bundle must therefore be designed to accept the radial expansion difference in the tube coils due to non-linear temperature profiles and due to variation in tube metal thermal expansion coefficients.

c. Inlet Seal

The steam generator inlet seals provide a positive seal between the hot helium ducts embedded in the concrete pressure vessel and the individual steam generators. Due to lack of access to the seal area during installation of the steam generators, these seals which are an integral part of the steam generator assembly, must mate with their respective counterparts of the ducting as the unit is set in place.

The primary function of the seal is to prevent hot primary coolant at core outlet temperature from bypassing the steam generator and entering the PCRV cavity between penetration liner and steam generator shroud. This primary function of the seals must be accomplished with large thermal expansion movements of the steam generator under various transient operating conditions.

To this end a flexible seal, with positive cold gas purge preventing hot gas leakage, has been adopted.

d. Gas Bypass Flow Restrictors

The standard requirement of restricting water leakage into the primary coolant under tube rupture conditions results in a large number of individual tubes of relatively small size. Routing these tubes from the heat transfer bundles to and from the tubesheets, they are required to pass through internal flow baffles which define the gas flow path inside the steam generator. A flow restrictor is required at each of these locations to limit leakage of primary coolant flow from one gas region into another. These seals must be able to accommodate the flexibility requirements as discussed under paragraph (a) above.

Various flow restrictor designs, of either sliding or restraining contraction, are presently being evaluated by analytical and/or test methods.

e. Tubesheets and Tube-to-Tubesheet Joints

The major problem in the design of a tubesheet as well as a tube-to-tubesheet joint in a HTGR steam generator lies in the high operating temperatures of these components and the temperature changes experienced under certain transient operating conditions. These problems are predominantly caused by large variations in thermal response times between adjacent thin-walled and thick-walled components in the tubesheet areas, and also steady state temperature differences between inlet and outlet sections in the primary closure.

The tubesheets are attached to the primary closure by means of thermal sleeves which provide sufficient length to maintain temperature gradients at acceptable levels.

Tube-to-tubesheet joints must withstand the temperature gradients discussed above. In addition tube end reactions as discussed in paragraph (a) significantly influence the joint design. The joint design under development utilizes a partial penetration weld, where the weld thickness is greater than the tube wall thickness.

Thermal shielding of the tube-to-tubesheet joint may be required, pending final stress and fatigue analysis, in order to protect the actual weld connection from high cyclic stresses during transients.

4.1.3.1 Plant Control:

The control system design for the 1100 MW(e) commercial HTGR plant is based on studies conducted for the 330 MW HTGR plant, the 1100 MW(e) HTGR plant as well as the GCFR plant.

These studies consisted of the explicit mathematical modeling of the various plant subsystems and determination of their response characteristics. Various control concepts were evaluated. The 330 MW(e) HTGR plant control system concept is used as a basic reference because of the relatively large amount of design data available for establishing some model/hardware correlation.

The results of these studies indicate that the control system is used in the 330 MW demonstration plant is acceptable, in part, for the 1100 MW(e) commercial HTGR.

Figure 37 shows the Advanced HTGR Plant Control System in a simplified functional form.

The control of three major plant variables is achieved by the manipulation of reactor power, helium circulator speed (flow), feedwater flow and reheat attemperation flow. The selection of the manipulated variables was done by extensive evaluation of the fluid flow and heat transfer dynamics of the reactor and steam generator, and application of basic control theory towards obtaining:

1. Minimal coupling of the major interacting variables.
2. Highly responsive control action by optimal selection of presently available feedback and feedforward variables yielding a quasi-linear controlled plant.
3. Balanced controller compensation in feedback and feedforward loops to minimize difficulty of operator adjustment.

LC72287

Figure 37. 1100 MW(e) HTGR Plant Simplified Control Diagram

The most significant difference in the 1100 MW(e) HTGR control concept compared to the 330 MW(e) HTGR control system is the addition of controlled reheat steam attemperation and the removal of the cascaded reheat steam temperature controller from the reactor power control loop.

4.1.3.2 Steam Generator Control

Some important points to remember in the control of the main and reheat steam temperatures with a compact once-through steam generator are:

1. The active section steam residence times are very short, on the order of 0.5 secs. or less.
2. The active section steam volumes are very small, on the order of 50 ft³ or less.
3. The ratios of steam heat capacity (Btu/°F) to active tube bundle (plus weighted header) capacity can vary from about 1/2 (GCFR) to 1/10 (330 MW HTGR).
4. The heat conductance (Btu/deg-sec) from tubes-to-steam ranges from 2 to 5 times higher than that from the helium-to-tubes.

These conditions indicate that rate of change of heat transfer rates (Btu/sec²) are considerably higher through the steam/water side than through the helium side. In order to obtain better temperature control, adjustment of steam/water variables is not only desirable but necessary.

a. Main Steam Temperature

Main steam temperature is controlled by the manipulation of circulator speed (flow) using the measured loop main steam temperature as a feedback signal to the temperature controller. A feedforward signal derived from the loop feedwater flow is used to dynamically compensate the helium flow demand signal for feedwater (steam) flow changes to which steam temperature is quite sensitive. This control scheme is identical to that used in the 330 MW(e) Power Demonstration Plant.

The selection of this control mode is based on the evaluation of the dynamic characteristics of the main steam generator section and the helium turbine/blower.

b. Reheat Steam Temperature

Reheat steam temperature is controlled by manipulating reheat attemperation, instead of directly adjusting reactor power as it is used in the 330 MW(e) HTGR plant for reheater control. The major advantages of this control mode are:

1. Reheat steam temperature can be controlled better because of the more rapid response to an inlet temperature and reheat inlet steam flow variation than a helium temperature via reactor power and/or helium flow variation. Improvements of 200 to 1 in response time have been realized in some cases studied.
2. The adjustment of attemperation flow decouples reheat steam temperature from main steam temperature variations.
3. It provides higher margins of stability in all of the other plant control loops.

Another beneficial effect of reheat inlet temperature decoupling which is obtained now by reheat inlet attemperation is that the reheater even though integral with the main steam section, i.e. coupled through helium flow and temperature, will now have less of a coupling effect on the main steam temperature.

Further investigation into the effects of reheat attemperation flow and reheat inlet temperature variations on reheat and main steam outlet temperatures were performed with the aid of a linearized steam generator model. The basis for this approach was the immediate availability of the model used for the 330 MW plant control studies and the similarity of the 1100 MW HTGR and 330 MW HTGR steam generator designs.

The state variable formulation of the steam generator dynamics was used to develop transfer functions. From these transfer functions, the responses of the steam generator to various input variables were evaluated. The results indicate that the performance of the steam generators will be improved with reheat attemperation. In particular, changes in main and reheat steam outlet temperatures resulting from changes in reheater steam flow are smaller and respond more quickly yielding a more stable steam generator operation when reheat steam temperature is controlled by attemperation rather than reactor power.

c. Main Steam Pressure

Steam generator outlet pressure is not directly controlled. Instead the concept is to control the main turbine throttle pressure which is more convenient than to have a separate valve at the steam generator outlet. This control concept is similar to the 330 MW plant control system design.

Because of the relatively low steam flow capacity and low flow resistances of the piping, pressure changes will be rapid and large if a quick responding pressure control is not employed.

The manipulation of feedwater flow is an effective way of controlling the throttle pressure. Rapid changes in water flow through the boiler feedpump can be made and when combined with the fairly high heat storage capacity of the steam generator tubes and piping, sufficient additional energy can be transferred to or from the steam to obtain good pressure control.

The control loop consists of a throttle pressure setpoint which is compared to the measured throttle pressure. The resulting error signal is properly modified through the pressure controller and is combined with a feedforward signal derived from the turbine first stage pressure. This resulting signal is used as a cascaded setpoint to the boiler feedpump flow control loop.

d. Reactor Power

Reactor power will now be controlled by sensing the IP turbine inlet pressure and comparing this signal with a cascade setpoint derived from the plant load index. First stage pressure is shown as a feedforward signal and is intended to supply some dynamic compensation. The choice of the IP turbine inlet pressure is based on the fact that this signal is a good representation of actual turbine load. The principal advantages of this mode of control are:

1. Utilization of the very large heat capacity of the reactor core is made possible, i.e., the average power changes for a given change in plant operation can be done more smoothly without the need of rapid changes in helium temperature through reactor power changes.
2. Controller requirements are more compatible with the typical 3% to 5%/minute plant load changing requirements. In particular, large derivative action will not be required yielding a less noise sensitive control loop.

4.1.3 Research and Development

Again, in recognition of the extreme importance of the functional availability of the steam generator as a nuclear steam supply system component, a series of research and development programs have been planned to verify design, performance and analysis of the steam generator and/or its components. Many of these programs are being executed at present.

The following is a description of the more significant work being done in this area.

a. Materials Tests

Mechanical properties of structural materials at high temperatures are needed to verify the design of the steam generator. Materials which require investigation are Inconel 625 and Incoloy 8-0. Proposed evaluations include time dependent tensile tests, creep strain, stress relaxation and a series of variable loading tests on welded sections.

b. Boiling Stability Tests

A series of tests using the existing Boiling Stability Test Loop (BSTL) are being conducted in order to verify stable steam generator operation at low load conditions and to establish the feasibility of a reactor scram from full load with 10% minimum feedwater flow rate during plant cool-down.

The first phase of the experiment utilizes the existing BSTL helical coil, which models the PSC steam generator design, with only minor changes to loop instrumentation to get accurate data at low flow. A second phase of testing will follow in which the helical coil will be replaced with one representative of the 1100 MW(e) steam generator design. Inlet orifices necessary for stable steam generator operation over the full range of operating conditions will be sized from the test data.

c. Primary Coolant Flow Distribution

A test is being conducted to determine what type of flow distribution devices are required (vanes or baffles) to establish satisfactory primary coolant flow conditions through the reheater and superheater tube bundles. These tests are conducted on a scaled down, transparent-flow model, which employs air as the working fluid. Particular emphasis is being placed on flow conditions at the reheater

inlet, reheater outlet and superheater inlet sections where flow distribution problems are anticipated. Pressure loss characteristics of the steam generator will also be determined during this program.

d. Vibration Tests

Due to the size and cost of an actual steam generator of the commercial HTGR plant, preoperational testing of a unit, which was done for the PSC project, becomes impractical.

Therefore, it is anticipated that all vibration test work required to verify predicted vibrational stresses, frequencies and amplitudes will be conducted on scaled down steam generator models.

e. Inlet Seal Test

Tests will be conducted to verify seal effectiveness, fatigue and wear characteristics of the steam generator module seals. It will be necessary to perform mechanical cycling tests under simulated environmental conditions to determine wear, compatibility of surfaces, structural behavior and leakage flow.

f. Tubesheet Tests

An experimental program will be conducted to determine the adequacy of the tube-to-tubesheet joint. The program will be designed to evaluate the stress concentration factor for the weld joint, to demonstrate the ability of the joint to withstand thermal shock loading, and to evaluate the effectiveness of rolling the tubes into the tubesheet.

These tests will be conducted on actual component models under simulated operating conditions, except that the

number of cycles the components will be subjected to are many times the number of operational cycles.

Periodic inspections will be made during the tests. After completion of the tests, the samples will be sectioned, examined and photographed to evaluate the results.

5. THE 300 MW(e) GCFR-DEMONSTRATION PLANT

Gas-cooled fast reactor design studies began at General Atomic about 10 years ago, first under private sponsorship and later with AEC support. In 1965 the work was intensified in a cooperative program with the East Central Nuclear Group, a group of utilities which financially supported this development program. This work as well as studies abroad showed that cooling with helium at 70 to 100 atm pressure and with core pumping power of the order of 4% of the thermal power is a very attractive alternative to liquid metal cooling for fast breeder power reactors, particularly for plants of large size.

To explore further the promise of gas cooling for fast breeders GGA and 43 utility companies (Table 13) in 1968 began a joint program that is now scheduled to run for four years. This program is directed towards the development of a GCFR and is centered on a 300 MW(e) demonstration plant. A plant of this size is considered large enough to give realistic service experience with full scale reactor components on a utility system.

The design is based to the greatest extent possible on fuel, as it is being developed for liquid metal fast breeders, and on HTGR plant component technology. Thus, development requirements special to GCFR plants are minimized.

5.1 Plant Description

The design of the 300 MW(e) GCFR demonstration plant features the nuclear steam supply system to be completely enclosed by a prestressed concrete vessel similar to that used for an HTGR as shown in Figure 38. Conventional secondary containment is also provided. The helium coolant at 85 atm is circulated by steam driven single stage axial flow compressors in each of the main loops. Three auxiliary cooling loops with electrically driven circulators are used for emergency backup.

Steam generated at 2900 psi and 875°F is first expanded through the helium circulator drive turbines and then reheated to 925°F before being admitted to the turbine generator (see Figure 39).

TABLE 13

300 MW(e) GCFR - DEMONSTRATION PLANT
SUPPORTING UTILITIES

GCFR Utility Program Group

Central Illinois Light Company
Central Illinois Public Service Company
Cincinnati Gas and Electric Company
Dallas Power and Light Company
Duke Power Company
Gulf States Utilities Company
Houston Lighting and Power Company
Illinois Power Company
Kentucky Utilities Company
Northeast Utilities Service Company
 Connecticut Light and Power Company
 Hartford Electric Light Company
 Holyoke Water Power Company
 Western Massachusetts Electric Company
Pacific Power and Light Company
Philadelphia Electric Company
Portland General Electric Company
Public Service Company of Colorado
Public Service Company of New Mexico
Puget Sound Power and Light Company
St. Joseph Light and Power Company
San Diego Gas and Electric Company
Sierra Pacific Power Company
Texas Electric Service Company
Texas Power and Light Company
Union Electric Company
Utah Power and Light Company
Virginia Electric Power Company

East Central Nuclear Group

Appalachian Power Company
Cleveland Electric Illuminating Company
Columbus and Southern Ohio Electric Company
Dayton Power and Light Company
Indiana & Michigan Electric Company
Indianapolis Power & Light Company
Louisville Gas and Electric Company
Monongahela Power Company

TABLE 13 (Cont.)

300 MW(e) GCFR - DEMONSTRATION PLANT
SUPPORTING UTILITIES

East Central Nuclear Group (Cont.)

Ohio Edison Company
Ohio Power Company
Pennsylvania Power Company
Potomac Edison Company
Southern Indiana Gas and Electric Company
West Penn Power Company

Overseas Utility Participant

Stockholms Elverk (Sweden)

LC81109-A

Figure 38. 300 MW(e) GCFR Plant

LC83837

Figure 39. 300 MW(e) GCFR Plant - Schematic Cycle Diagram

The turbine cycle for this plant does not include a reheater, because earlier studies showed that the use of gas reheat is probably not economically justified in association with the low fuel cycle cost provided by the fast reactor.

5.1.1 Steam Generator

5.1.1.1 Description

The GCFR steam generators are once-through, helically coiled units fairly similar in concept to those already built for the Fort St. Vrain power station and under detailed design for the 1100 MW(e) commercial HTGR plant. These units are, however, much larger than the Fort St. Vrain steam generator modules: 3 modules for 300 MW(e) in the GCFR versus 12 modules for 330 MW(e) in Fort St. Vrain. On the other hand, the GCFR temperatures are somewhat lower on the steam as well as on the gas side, as evident from Table 14 which shows the steam generator design data. Lower temperatures tend to ease the design problems. In general, though, the GCFR steam generators are very similar to those for large advanced HTGR power plants.

Gas flow is downward through the core and it is also downward across the steam generator heat transfer surfaces to accommodate upflow of the secondary coolant in the steam generator. This necessitates appropriate flow reversals in the gas path, and these are obtained by leading the hot helium through the center region of the tube bundles to the top of the steam generators, in a similar manner as described for the 1100 MW(e) steam generator design.

The steam generator is made up of two separate tube bundles--the resuperheater tube bundle and the main steam tube bundle, which consists of the combined economizer, evaporator, and superheater sections. In this single tube bundle, the incoming high-pressure feed-water is converted into superheated steam that is then used to drive the main helium circulator turbine. The steam is subsequently returned to the steam generator and resuperheated to final steam conditions in the resuperheater before it flows to the main turbine of the power plant. Helium,

TABLE 14

300 MW(e) GCFR - DEMONSTRATION PLANT
STEAM GENERATOR DESIGN CONDITIONS

	100% Steam Flow
<u>Load</u>	
Generator Output (MW)	315.8
Station Output (MW)	310.7
<u>Helium</u>	
Flow (lbs/hr)	5,487,000
Outlet Pressure (psia)	1250
Inlet Temperature (°F)	1007
Outlet Temperature (°F)	570
Pressure Drop (psi)	6.1
<u>Resuperheater Steam</u>	
Flow (lbs/hr)	2,649,000
Outlet Pressure (psia)	1304
Inlet Temperature (°F)	680
Outlet Temperature (°F)	925
Pressure Drop (psi)	43
<u>Feedwater/Main Steam</u>	
Flow (lbs/hr)	2,649,000
Outlet Pressure (psia)	2923
Inlet Temperature (°F)	412
Outlet Temperature (°F)	875
Pressure Drop (psi)	254

the primary heat-transfer medium, enters the steam generator via the cross ducts from below the reactor passes over the heat-transfer surface in the steam generator and leaves the steam generator via the annulus between steam generator shroud and tube bundle to enter the suction side of the circulator.

The helium circuit of the plant is divided into three primary coolant loops, with each loop having one steam-generator module with its associated helium circulator. The general arrangement of the steam-generator is shown in Figure 40. The water-steam flow on the tube side is axially upward; the helium in almost pure crossflow passes downward across the tube bundles.

Feedwater enters the steam-generator cavity through penetrations in the bottom of the PCRV. The feedwater pipe for each steam-generator module discharges into three separate feedwater penetration channels containing tubesheets and then to lead-in tubes connecting the tubesheets to the economizer section of the helically wound steam-generator tube bundle; each steam-generator primary bottom closure has three feedwater tubesheets. To avoid instability in the steam-generator tube bundle during low plant load operations, the economizer and evaporator sections have smaller diameter tubes than the superheater section of the steam-generator tube bundle, thereby effectively distributing water/steam side pressure drop. In addition to the main helium circulator, each steam generator top closure has to accommodate nine separate tubesheets, three each for the superheater outlet and the resuperheater inlet and outlet. The available space in the top closure limits the size of the individual channel penetrations and thus subheadering is required.

Individual tubes are connected to subheaders in the cold helium area above the flow barrier located over the resuperheater. After the subheadering, the tubes are collected into nine bundles and routed in a partial helix to extend up into the closure-plug penetration and terminate at the tubesheets above the flow restrictors. The tube bundles are insulated to reduce the heat transfer from the hot steam tubes to the cold helium in the circulator inlet plenum. Insulation is also provided inside the closure-plug penetrations to reduce cooling requirements for the penetration liner.

LC89938

Figure 40. 300 MW(e) GCFR Plant - Steam Generator

The steam-generator primary closure is a composite steel and reinforced-concrete design that has nine steam pipe penetrations around the center penetration for the helium circulator. Each steam pipe penetration contains a tubesheet. The feedwater channels at the bottom of the PCRV and the steam channels at the top form an integral part of their respective primary closures and are readily accessible for tube plugging.

5.1.1.2 Design

Each steam generator module is 10 ft. 6 in. in diameter and approximately 37 ft. in length and is supported at its upper end by means of a bolted seal flange seating on a thermal sleeve flange extending from the penetration wall.

Despite the comparatively higher material cost for Incoloy 800, thinner tube walls may be used due to their higher strength and greater resistance to corrosion, and this results in an overall saving in the total steam generator cost. It also leads to a corresponding decrease in the number of tubes with associated labor cost reductions and also in an overall weight reduction of the steam-generator tube bundles. Therefore, Incoloy 800 Grade I tubing is used throughout the steam-generator tube bundles.

The detailed design will be executed in accordance with all requirements of the ASME Code, Section III. Equally material selection and fabrication will meet all requirements of applicable codes. It is expected that design details of the steam generators for a 300 MW(e) GCFR demonstration plant will follow closely the details of the HTGR plants, thereby allowing advantage to be taken of the experience gained during layout, design and analysis of the HTGR steam generators.

Details of tube supports, seals and tubesheets in particular will rely heavily on previous experience. As mentioned earlier, the elimination of an internal reheater, designed for external pressure, as well as the somewhat lower temperature level of this unit will most assuredly reduce the design problems.

6. OUTLOOK

The tremendous increase in power demand especially in the U.S.A. provide a great challenge to the utility supply industry. Present indications in the U.S.A. are that an additional 250,000 MW might be required by the end of this decade. (Figure 41). This translates into 250 plants of approximately 1000 MW size.

It is safe to assume that a great percentage of the new capacity to be added will be powered by nuclear sources especially in the light of more stringent air pollution control laws. With only five suppliers of nuclear steam supply systems in the U.S.A., we at Gulf General Atomic are confident that we will be able to do our part in supplying the equipment to meet the ever increasing demand. We are confident that we will capture a fair share of the utility market. At present the commercial 1100 MW(e) HTGR plant is being offered to the American utility market and a 300 MW(e) fast reactor plant is conceptually designed and could be built at any time. We are convinced that by the end of the decade a 300 MW(e) gas-cooled breeder will be in operation.

However, in spite of this achievement, work is continuously going on in order to improve the existing systems as well as in advanced designs.

With the advent of ever increasing power demand together with the shortage of suitable nuclear power plant sites, as well as in the interest of cost savings, still larger unit sizes are being studied. It probably can be stated as a fact that in not too far a future we will see nuclear power plants of 1500 MW(e) or even 2000 MW(e) size. Here, of course, the HTGR system with its prestressed concrete reactor pressure vessel offers a tremendous advantage over steel pressure vessels which would have to be field assembled.

With units in the 1500 MW(e) to 2000 MW(e) range it becomes an economical question if the nuclear steam supply system components should keep pace by increasing their size or if a greater number of proven

LC84570

Figure 41. Projected Addition to the U.S. Electric Power Generating Capacity

standardized modules should be employed. This problem is also under active study at GGA and its resolution will greatly depend on the reliability and the service history of the first few plants going into operation.

Improvement in the fuel cycle can be brought about by improving steam conditions. With present material technology it must be conceded that the steam temperature level of 1000°F will stay with us for a while. This, however, does not preclude the possibility of increased steam temperatures in the future, when better materials with high strength in the high temperature range become available.

The use of small tubing forced flow steam generating surfaces in the HTGR steam generator makes the adoption of a supercritical pressure steam cycle with main steam temperatures at 1000°F a natural choice and will undoubtedly be adopted in the future for large HTGR commercial power plants.

Other improvements which will increase the reliability, availability and maneuverability of large HTGR systems can be made in the control area by employing a control principle similar to the "direct energy balance" system, where demand is balanced to the supply by direct control of "fire" and "water" in response to the load demand signal, rather than having the boiler follow the turbine demand. Such a coordinated control system would improve the unit response to system changes, which is almost a necessity in the U.S.A. with its large intertie systems.

The notes made above concerning future development work on large commercial HTGR power plant systems apply equally for GCFR systems.

After their initial acceptance, fast breeder reactor systems will replace more and more conventional reactor systems for power generation in the utility industry. At this point then, the same problems arise: namely increase in size as well as improvement of existing designs for all components utilized in the system.

Some studies of a 1000 MW(e) gas cooled breeder reactor have been done at GGA. The plant will be a direct scale-up of the multicavity PCRV as it is proposed for the 300 MW(e) GCFR demonstration plant and it will house

the complete nuclear steam supply system. Six steam generator loops will be used in a similar manner as for the 1100 MW(e) commercial HTGR plant. In addition three emergency cooling heat exchangers are at present anticipated. These, together with their emergency circulators will be housed in their own individual cavities.

In conclusion, it should be stated that there is room for advancement and improvement in any component or system described in the paper and the utility industry can be assured that everything is being done in terms of research, development, design and fabrication to supply them with a final product which will give them many years of reliable service. Furthermore, we in the nuclear steam system supply industry will do our utmost at all times to meet the demands put on us by the increasing power requirement.

7. ACKNOWLEDGEMENTS

The author wishes to express his gratitude to his many colleagues at Gulf General Atomic, Incorporated and at Sulzer Brothers for their assistance in assembling the information contained in this report.

8. REFERENCES

Section 1:

- 1.1 J. Kägi: "The Principal Design of Steam Generators for Indirect Cycle Nuclear Power Plants", Sulzer Technical Review, Special No. "Nuclex" 1969.

Section 2:

- 2.1 M. E. Kantor, H. F. Menzel, R. W. Schlicht, W. T. Westlake: "Engineering Tests During the Initial Operation of the Peach Bottom HTGR", American Nuclear Society, 13 Annual Meeting, San Diego, June 11-15, 1967.
- 2.2 M. E. Kantor: "Recent Operating Experience at the Peach Bottom Atomic Power Station", American Nuclear Society, International Conference, Washington, D.C., November 10-15, 1968.
- 2.3 R. W. Green, J. S. Kemper: "Significant Operating Experience at the Peach Bottom Atomic Power Station", Symposium on Advanced and High Temperature Gas-Cooled Reactors, Jülich, October 21-25, 1968.

Section 3:

- 3.1 W. G. Schuetzenduebel, P. S. Hunt: "Review of Design Criteria for Steam Generators of HTGR Plant of Public Service Company of Colorado", ASME Winter Annual Meeting, Los Angeles, November 16-20, 1969.
- 3.2 W. G. Schuetzenduebel: "Konstruktionseinzelheiten des Dampferzeugers für ein 330 MW(e) HTGR-Kraftwerk", Energie 22, Nr. 4, April 1970.
- 3.3 W. Kalin: "The Steam Generators of Fort St. Vrain Nuclear Power Plant", Sulzer Technical Review, Special No. "Nuclex" 1969.
- 3.4 W. G. Schuetzenduebel, L. Meyer: "330 MW(e) HTGR mit Spannbetondruckgefäß", In preparation.
- 3.5 W. G. Schuetzenduebel, "Das HTGR-Kernkraftwerk Fort St. Vrain, Colorado, VDI-Nachrichten Nr. 9, March 9, 1970.

3.6 ASME Code Section III - Nuclear Vessels 1968.

3.7 L. C. Shao, P. S. Hunt, W. G. Schuetzenduebel: "The Effect of Thermal Stresses on the Design of the Steam Generators for PSC's 330 MW(e) HTGR Plant", In preparation.

Section 4:

4.1 T. A. Johnston: "Large HTGR Power Plants", American Power Conference, 32 Annual Meeting, Chicago, April 21-23, 1970.

Section 5:

5.1 W. C. Guyker, D. W. Ver Planck: "Special Development Problems in GCFR", American Nuclear Society, Topical Meeting, Williamsburg, September 1-2, 1970.

5.2 P. Fortescue, W. I. Thompson: "The GCFR Demonstration Plant Design", ORNL-Gas Cooled Information Meeting, Oak Ridge, April 27-29, 1970.

Section 6:

6.1 G. Seaborg: "A Review and Comparison of Selected U.S. Energy Forecasts" - American Power Conference, 32 Annual Meeting, Chicago, April 21-23, 1970.

6.2 F. C. Olds: "The Nuclear Power Struggle" - Power Engineering, September 1970.

6.3 G. Melese d' Hospital: "The Gas-Cooled Fast Breeder Reactor", Energie Nucleaire, October 1969.