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1170 MW_t HTGR STEAMER COGENERATION PLANT
DESIGN AND COST STUDY

August 1980

Work Performed Under Contract No. AC02-78ET34222

United Engineers & Constructors Inc.
Philadelphia, Pennsylvania

TECHNICAL INFORMATION CENTER
UNITED STATES DEPARTMENT OF ENERGY

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by
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SECTION 1.0

EXECUTIVE SUMMARY

1.1 INTRODUCTION

The work reported herein is a conceptual design and cost study of an intermediate size high temperature gas-cooled reactor (HTGR) for industrial energy applications performed by United Engineers & Constructors Inc. (UE&C) and The General Atomic Company (GAC). The UE&C work was performed for the Department of Energy (DOE) under Contract No. DE-AC02-78ET34222, and the GAC work was performed under Subcontract No. 4188-7 for the Oak Ridge National Laboratory (ORNL) operated by the Union Carbide Corporation. The study is part of a program at ORNL and has the objective to provide support in the evaluation of the technical and economic feasibility of a single unit 1170 MW_t HTGR steam cycle cogeneration plant (referred to as the Steamer plant) for the production of industrial process energy. Inherent in the achievement of this objective, it was essential to perform a number of basic tasks such as the development of plant concept, capital cost estimate, project schedule and annual operation and maintenance (O&M) cost.

The groundrules for the study were set generally in accordance with the requirements of the Gulf Oil Company refinery/chemical complex at Port Arthur, Texas. The design also has broad general applicability to other industrial locations. Gulf Oil Company provided actual process steam and electric requirements, site data, etc., to create a realistic design basis.

HTGR plants are considered to be particularly suited to cogeneration of process steam and electric power. Since they produce steam at elevated conditions of temperature and pressure, electric power can be efficiently produced, and the HTGRs can substitute fairly directly for the large oil/gas-fired industrial boilers currently used for cogeneration of process steam and electric power. The Steamer plant design is based heavily on the large HTGR steam cycle (HTGR-SC) electric power plant design developed for the Gas Cooled Reactor Associates (GCRA) utility association. The GCRA power plant design consists of a 2240 MW_t core with four primary reactor coolant loops. For the Steamer design, the core was reduced in size to 1170 MW_t, and two reactor coolant loops were used. Design and cost information related to the Nuclear Steam Supply System (NSSS) was developed by the General Atomic Company, whereas the balance-of-plant data were developed by UE&C.

1.2 STUDY GROUND RULES

The groundrules developed for this study were based generally in accordance with the requirements of the Gulf Oil Company refinery/chemical complex at Port Arthur, Texas. An important exception is the parameters related to soil characteristics and foundations. The soils along the Gulf Coast area generally require special considerations unique to the area. Standard soil/foundation parameters (Table 1.2-1) were assumed for this study to allow it to retain a generic character and thus make it more readily applicable to other industrial locations.

The following is a summary of major groundrules used in this study:

- a. The reference site parameters are as listed in Table 1.2-1.
- b. Cost data are presented at the three digit level of detail in accordance with an expanded AEC code-of-accounts (USAEC Report NUS-531). The estimate was developed for January 1, 1978 prices using the labor craft rates associated with the Port Arthur, Texas area and reflects nth-of-a-kind (NOAK) plant design. Adders for first-of-a-kind (FOAK) plant design are discussed in Subsections 5.1 and 7.2.
- c. The licensing and regulatory requirements are based on those in effect on January 1, 1978.
- d. The plant design is based on a single unit with no provisions for sharing.
- e. The balance-of-plant (BOP) design is based on the following HTGR Steamer Cogeneration Plant documents provided by GAC:
 1. Balance of Plant Requirements (BOPR) Document No. 903727, Draft 1
 2. Plant Technical Description-Document No. 903733, Issue 1 - 9/15/78
 3. Expected NSSS Performance-Document No. 903737, Issue 1 - 9/15/78
 4. Plant Layout Requirements-Document No. R-1101, 9/15/78
 5. System Description-Document No. 903750, Issue 1 - 9/15/78
 6. Reactor Internals Layout-Drawing No. 024506, Issue 1 - 9/15/78

7. GAC Drawing No. 024454 "PCRV General Arrangement Plan and Section, Rev. 1, Preliminary dated 8/24/78

Equipment sizes, other than for the PCRV and CACW Systems, are based on the GCRA Plant Layout Requirements, Document No. 903605, dated 7/15/78.

- f. The fuel storage facility is designed for 1.3 cores using a design basis of 1832 fuel blocks per core.
- g. The process steam system is designed to deliver steam at the plant boundary at 650 psia, 750 F, with condensate returning to the plant boundary at 25 psia, 95 F. While the refinery requires steam at several pressure and temperature conditions, a single steam condition was selected to allow this study to retain a generic character.
- h. It is assumed that secondary steam from the steamer plant can be transmitted to the industrial user without requiring isolation barriers. Also the condensate returning from the process is assumed to be equal in chemistry to that normally found in a utility condenser.
- i. The design utilizes two 69 kV independent offsite sources of power. The electrical systems terminate at the plant boundary and do not include the switchyard or transmission facilities.
- j. Normal service water is provided by closed evaporative cooling tower systems.
- k. Plant makeup water is provided from a surface water source.

1.3 NUCLEAR STEAM SUPPLY SYSTEM

The NSSS design considered here is based on a large HTGR-SC electric power plant design developed for the Gas Cooled Reactor Associates (GCRA), an association which includes many leading U.S. electric utilities. The design for the GCRA group is a 2240-MW(t) plant producing 900 MWe. This design, referred to as the Reference 900 HTGR-SC, is the result of more than 20 years of HTGR development plus experience acquired from construction and operation of the HTGR power plants at Peach Bottom, Pennsylvania, and Fort St. Vrain, Colorado.

The 1170 MW_t HTGR Steamer NSSS design is identical to that of the Reference 900 HTGR-SC electric power plant, except for modifications required to reduce the thermal capacity from 2240 MW_t to 1170 MW_t. Since the HTGR-SC power plant has four primary reactor coolant loops, the capacity reduction was accomplished by using two loops of the same design and a reduced diameter core.

1.4 BALANCE OF PLANT

Figure 1.4-1 shows a site Plot Plan for the Steamer Plant. This plot plan includes a reactor containment building which houses the PCRV and other NSSS components; a turbine building, which houses the turbine generator and its auxiliary systems; a reactor service building, which houses reactor auxiliary equipment such as facilities for fuel handling, radioactive waste management, etc.; a fuel storage building, which houses all equipment related to new and spent fuel shipping, receiving and storage; a control, auxiliary and diesel generator building; an administration and service building; a containment annulus building; and other miscellaneous structures.

Table 1.4-1 includes key parameters for the Steamer plant. Figure 1.4-2 shows a turbine heat balance for the steamer plant designed to provide 2,715,500 lbs/hr of process steam at 650 psia and 750 F and 144.5 MWe (107.5 MW net) of electricity concurrently. The mix of design process steam and electrical output can be varied in a significant range depending upon the process steam temperature, pressure and other requirements of the particular user.

1.5 CAPITAL AND O&M COST ESTIMATES

1.5.1 Capital Cost Estimates

Table 1.5-1 includes a capital cost summary for the Steamer plant at the two digit account level. The cost estimate reflects nth-of-a-kind (NOAK) plant design. The capital costs are in January 1978 dollars and reflect labor craft rates associated with the Port Arthur, Texas area.

Capital costs related to the Nuclear Steam Supply System (NSSS) were provided by the General Atomic Company, and the balance-of-plant costs were developed by UE&C. The estimate does not include items listed in Table 1.5-2. Generally, these items are extremely sensitive to the particular policies of the individual plant owner, and to the specific plant and site being considered.

The capital cost estimate includes capital costs associated with a waste heat rejection system designed to provide the cooling requirements during reactor startup and hot standby operation; and also to reject normal service water heat. This system may not be required for hot standby operation, depending upon the requirements of the process steam user and other operational considerations. In such cases, the size of the system will be reduced and the capital cost estimate presented in Table 1.5-1 will be lower by up to \$3,000,000.

1.5.2 O&M Costs

Table 1.5-3 includes nonfuel annual operating and maintenance costs for the Steamer plant, based upon January 1978 dollars. Staff requirements for the plant manager's office, operations, maintenance, technical and engineering, and plant security are estimated at 203. Insurance and fees include commercial liability insurance, government liability insurance and inspection fees. Administrative and general expenses include owner's offsite salaries and expenses directly allocable to the Steamer plant.

1.6 PROJECT SCHEDULE

Figure 1.6-1 shows the overall project schedule for nth-of-a-kind HTGR Steamer plant.

The schedule from project start to limited work authorization was developed by the General Atomic Company (GAC). It is based on the assumption of a standardized plant design and preselected site, which allow the time required for the licensing phase to be minimized. It also assumes prebidding of material for the PCRV liner, steam generators, and other long lead components.

The construction schedule from limited work authorization to commercial operation was developed by UE&C. It is based primarily on the schedule previously developed for the 3360 MW (thermal) HTGR and supplemented by drawings developed for the Steamer plant. This schedule indicates approximately 67 months from receipt of the limited work authorization (LWA) to commercial operation, determined by the critical path which runs through the reactor containment building and PCRV.

1.7 RECOMMENDATIONS

In the course of this study no barriers were identified which would seriously place in question the feasibility of the intermediate size HTGR for industrial energy applications. However, due to budget limitations, the design effort and cost estimate relied heavily on UE&C's past experience with HTGR's, using engineering judgment and assumptions to arrive at the final product. Therefore, additional cost studies could be undertaken to evaluate the economic viability of this concept when applied to specific industrial complexes utilizing the actual site conditions and process energy requirements. The results could be compared to alternate methods for providing the process energy, and, if encouraging, additional technical and economic studies could be performed.

TABLE 1.2-1

SITE PARAMETERS

A. Site Conditions

1. Soil Characteristics	Firm Soil	
Shear Wave Velocity (fps)	1000-1500	
Soil Shear Modulus (psi)	1.7×10^4 - 4.4×10^4	
Allowable Static Bearing Capacity (KSF)	10	
2. Foundation	No special compensating foundation required. Assume shallow embedment of reactor building.	
3. Water Table	Normal groundwater elevation is approximately eight ft below grade	
4. Wind Velocity	110 mph Basic Wind Speed per ANSI A58.1-1972 using 100 Year Mean Recurrence Interval and Exposure C.	
5. Air Temperature	Dry Bulb	Wet Bulb
Hottest Day of Record	102 F	82 F
Coldest Day of Record	20 F	N/A
6. Meteorological	Per NRC Regulatory Guide 1.4	
Atmospheric Dispersing X/Q Values		
7. Makeup Water Source	Nonbrackish Water (less than 500 ppm TDS)	
8. Service Water	Service Water will be provided by closed loop evaporative cooling tower systems	

B. Site Events

1. Seismic	
Maximum Horizontal Ground Acceleration:	Firm Soil
Safe Shutdown Earthquake	0.25g
Operating Basis Earthquake	0.125g

TABLE 1.2-1 (Cont'd)

SITE PARAMETERS

2. Tornado

Designation (RG 1.76)	Zone I
Tangential Velocity	290 MPH
Translational Velocity	70 MPH
External Pressure Drop	3.0 psi @ 2 psi/sec.

3. Flooding

No special provisions required.

Maximum probable flood level
below plant grade.

4. Aircraft Hardening,
Industrial/Transportation
Accidents

No special provisions required.

TABLE 1.4-1
KEY PLANT PARAMETERS

A. Primary Coolant System	
Number of Loops	2
Power Generated by Core, MW _t	1170
NSSS Thermal Power, MWt	1,180
Total Helium Flow Rate, 10 ³ lb/hr	4,843
Core Inlet Average Temperature, F	605
Core Outlet Average Temperature, F	1,266
System Pressure, psia	1,050
B. Steam and Power Conversion System	
Total Steam Flow, 10 ⁶ lbm/hr	3,834
Steam Generator outlet Pressure, psia	2,515
Steam Generator Exit Temperature, F	1,005
Feedwater Inlet Pressure, psia	3,003
Feedwater Inlet Temperature, F	430
C. Overall Plant Performance	
High Pressure T-G Output, MWe	90.7
Low Pressure T-G Output, MWe	53.8
Total Gross Plant Output, MWe	144.5
Total Net Plant Output, MWe	107.5
Steam Flow to Process, 10 ⁶ lbm/hr	2.715
Process Steam Pressure, psia	650
Process Steam Temperature, F	750
Process Heat Load, MWt	1,046

TABLE 1.5-1

TWO DIGIT SUMMARY CAPITAL CCST ESTIMATECOST BASIS
01/76

ACCT NO	ACCOUNT DESCRIPTION	FACTORY EQUIP. CCSTS	SITE LABOR HOURS	SITE LABOR COST	SITE MATERIAL COST	TOTAL COSTS
*****	*****	*****	*****	*****	*****	*****
20 .	LAND AND LAND RIGHTS				268,800	268,800
21 .	STRUCTURES & IMPROVEMENTS	3,410,879	3300936 MH	35,605,432	30,762,274	69,778,585
22 .	REACTOR PLANT EQUIPMENT	159,679,992	1245465 MH	19,016,079	8,678,082	187,394,153
23 .	TURBINE PLANT EQUIPMENT	24,457,808	581562 MH	6,731,247	2,588,324	33,777,379
24 .	ELECTRIC PLANT EQUIP	12,100,000	1241215 MH	15,245,657	7,898,598	35,244,355
25 .	MISC. PLANT EQUIP	5,005,553	234972 MH	2,767,187	784,359	8,556,999
26 .	MAIN COND HEAT REJECT SYS	3,004,333	87275 MH	1,042,574	695,323	4,742,030
2 .	TOTAL DIRECT COSTS	207,678,365	6691925 MH	80,408,176	51,675,760	339,762,301
91 .	CONSTRUCTION SERVICES	13,380,000	1180000 MH	11,775,000	17,015,000	42,170,000
92 .	HOME OFFICE ENGRG. & SERVICE	64,145,000				64,145,000
93 .	FIELD OFFICE ENGRG. & SERVICE	16,000,000			1,920,000	17,920,000
9 .	TOTAL INDIRECT COSTS	93,525,000	1180000 MH	11,775,000	18,935,000	124,235,000
	TOTAL BASE COST	301,203,365	7871925 MH	92,183,176	70,610,760	463,997,301

TABLE 1.5-2

ITEMS NOT INCLUDED IN COST ESTIMATE

1. Switchyard and transmission facility costs.
2. Owner's costs, including consultants, site selection, etc.
3. Waste disposal costs.
4. Fees and permits - federal, state and local.
5. State and local taxes.
6. Spare parts.
7. Nuclear liability and other insurance (except as noted).
8. Initial fuel loading.
9. Interest during construction.
10. Escalation.
11. Contingency.

TABLE 1.5-3
ANNUAL OPERATING AND MAINTENANCE COSTS
JANUARY 1978 DOLLARS

Staff	\$4,570,000
Maintenance Materials	
Fixed	1,734,000
Variable	0
Supplies and Expenses	
Fixed	3,558,000
Variable	112,000
Insurance and Fees	400,000
Administrative and General	1,479,000
Total Fixed Costs	\$11,741,000
Total Variable Costs	112,000
Total Annual O&M Costs	\$11,853,000

FIGURE 1.4-1

SITE PLOT PLAN

 united engineers MEMBERSHIP

SCALE: 1" = 50' 6712-F-11000

Figure 1.6-1
NOAK Project Schedule
1170 MWt HTGR Steamer
Cogeneration Plant

SECTION 2.0

INTRODUCTION AND SITE EVALUATION

2.1 INTRODUCTION

Industrial process steam generating plants and cogeneration (process steam and electric power) plants constitute more than 16 percent of the total energy usage in the U.S. today. Most of the energy is being supplied in the form of natural gas and oil. For reasons of economics, dwindling, resources, conservation, and federal legislative prohibitions, these fuels are expected soon to be unavailable as boiler fuels for many new industrial plants and may, in fact, be phased out for many existing plants.

Viable alternatives for natural gas and oil through the near future are restricted to coal and nuclear fissionable material. The viability of these substitute energy sources as cogeneration plant fuels has been discussed in various technical literature. The advantages of the HTGR cogeneration plant in particular have been discussed in several publications. General Atomic report GA-A14712* describes a 1200 MW_t HTGR Steamer Cogeneration plant based on the currently operating Fort. St. Vrain, Colorado HTGR - steam cycle (HTGR-SC) electric power plant design.

HTGR plants are particularly well suited for process steam/electric power cogeneration applications because of their ability to produce high temperature, high pressure steam conditions identical to those of fossil-fired boilers. This provides maximum flexibility in establishing heat cycles to produce steam at any condition required for process use. It also permits fairly direct substitution for existing large oil/gas-fired industrial boilers.

This report is the culmination of a study performed to develop a concept, capital cost estimates, construction schedule and operating and maintenance (O&M) costs for an 1170 MW_t HTGR Steamer Cogeneration plant designed to produce process steam and electricity. The study ground rules developed and discussed in subsection 1.2 are based generally upon the requirements of the Gulf Oil Company refinery/chemical complex at Port Arthur, Texas. However, the objective has been to keep the design as generic as possible, to broaden its applicability to other industrial locations.

* Conceptual Design of a 1200-MW_t HTGR Steamer Cogeneration Plant. Economics and Program Report", General Atomic Report GA-A14712, v.3, April 1978.

2.2 SITE EVALUATION

This subsection contains the results of the preliminary evaluation of the suitability of the Gulf Refinery site for siting a nuclear power plant such as the 1170 MWt HTGR. The objective was to evaluate the actual site and to note the specific problems it presents in contrast to the generic or standard soil/foundation conditions assumed in this study. The scope of this evaluation was limited to review of readily available geologic data.

The soils along the gulf coast area generally require special considerations unique to the area. This is also the case with the site under evaluation. It should be noted that there may be other sites in the Port Arthur area which are more suitable from the standpoint of soil/foundation conditions, site elevation, etc. The benefits associated with such potential sites would have to be evaluated against increase in steam transmission costs.

The following summarizes the results of the site evaluation:

1. The surficial soils are soft with minimal load bearing capability. The absence of a good foundation material, or of a uniformly developed soil of well known characteristics, raises the question of the controllable medium on which to place the foundation. A box-type floating foundation placed on the upper (60-80 ft) pleistocene soils could represent a possible solution. This foundation concept was developed by UE&C in the early 1970s for a then-planned two-unit HTGR station. In that case, the additional costs for the special foundations etc. would have added a few percent to overall plant capital costs.
2. The structural geology of the area is controlled by the presence of salt domes and associated faults, mostly normal and inactive. Another system known as growth faults (dipping south) is forced within the sedimentary structure and allows a slow movement so that strain accumulation leading to sudden movement generating seismic energy cannot take place.
3. The presence of surface faulting in the area is indicated by pavement and foundation breaks, topographic scraps, etc. It is not known whether these faults will remain inactive.
4. The magnitude and area of development for subsidence are difficult to define. Extensive research will be required to correlate the man-made activities with the development of subsidence before any judgement can be made.
5. The elevation of the refinery area seems to be well below the open coast surge expected to be induced by the "Probable Maximum Hurricane" at the Gulf of Mexico. Protection against flooding due to hurricanes will be required to meet NRC requirements.

6. Potentially hazardous facilities and activities within five miles of the site would have to be identified and analyzed for their possible effect on the HTGR plant. Examples of these hazards are flammable vapor clouds, toxic chemicals, missiles and aircraft.
7. Considerations related to population densities have been evaluated in accordance with NRC siting criteria for nuclear power plants. Results indicated that the steamer plant could be located adjacent to the refinery.
8. Other siting factors would have to be evaluated in accordance with NRC siting criteria for nuclear power plants.

The evaluation of many of the items listed above were beyond the scope of this study. In general, the problems appear to have engineering solutions but no attempt was made to estimate additional costs associated with these special considerations. Instead, standard soil/foundation parameters (Table 1.2-1) were assumed for this study. This also allows this study to retain a generic character and thus make it more readily applicable to other industrial locations.

In view of the early 1970s work mentioned in Note 1. above, a few percent increase over generic costs might be expected. Such costs could be determined by more detailed study of this or any other specific site.

SECTION 3.0

NUCLEAR STEAM SUPPLY SYSTEM (NSSS) DESCRIPTION

3.1 INTRODUCTION

This section presents a design description for the nuclear steam supply system (NSSS) of an HTGR Steamer cogeneration plant, designed to meet the requirements of the Gulf Oil Company refiner/chemical complex at Port Arthur, Texas. The design also has broad general applicability to other industrial locations.

The NSSS design considered here is based on a large HTGR-SC electric power plant design developed for the Gas Cooled Reactor Associates (GCRA), an association which includes many leading U.S. electric utilities. The design for the GCRA group is a 2240-MW(t) plant producing 900 MW(e). This design, referred to as the Reference 900 HTGR-SC, is the result of more than 20 years of HTGR development plus experience acquired from construction and operation of the HTGR power plants at Peach Bottom, Pennsylvania, and Fort St. Vrain, Colorado.

The 1170 MWt HTGR Steamer NSSS design is identical to that of the Reference 900 HTGR-SC electric power plant, except for modifications required to reduce the thermal capacity from 2240 MW(t) to 1170 MW(t). Since the HTGR-SC power plant has four primary reactor coolant loops, the capacity reduction can be accomplished by using two loops of the same design and a reduced diameter core.

3.2 NSSS DESCRIPTION AND GENERAL ARRANGEMENT

The HTGR NSSS is characterized by a graphite-moderated, helium-cooled thermal reactor core within a multicavity PCRV.

The use of an inert single-phase gas for reactor coolant, combined with the use of graphite for the fuel cladding and core structure, allows steam production at the high temperatures and high pressures of modern fossil-fired electric power plant boilers, i.e., 17.34 MPa (2515 psia) at 540C (1005 F). Figure 3.2-1 shows the heat cycle diagram for the proposed HTGR process steamer cogeneration plant.

Figure 3.2-2 shows the arrangement of the PCRV. Primary coolant helium flows downward through the reactor core and divides equally between two identical coolant loops, each containing a steam generator, helium circulator, and helium shutoff valve. In each loop, the heated helium first passes upward through the steam generator and is then pumped back to the core inlet plenum by the helium circulator, completing the circuit. The helium shutoff valve is located at the circulator discharge and prevents flow through the loop when the loop is not operating.

Table 3.2-1 outlines the steamer plant NSSS performance conditions. Helium conditions at the core inlet are 318 C/7.239 MPa (605 F/1050 psia); core outlet temperature is 686 C (1266 F), and steam generator outlet temperature is 314 C (597 F). Total primary circuit pressure drop is 0.127 MPa (18.4 psi).

The steam generators are once-through units with helical coils in the economizer/evaporator section and straight tubes in the superheater section. The helium circulators are single-stage centrifugal compressors with electric motor drive. Speed control is provided by a variable frequency power supply.

In addition to the two primary coolant loops, three core auxiliary coolant loops are provided to insure safe shutdown of the reactor. Each auxiliary loop contains a water-cooled heat exchanger, a helium circulator, and a helium shutoff valve. The arrangement of the auxiliary loops is similar to that of the primary loops, with helium coolant passing downward through the core, upward through the heat exchangers, and then being pumped by the circulators back to the core inlet plenum. The shutoff valve is located between the heat exchanger and auxiliary circulator section.

The PCRV is a vertical cylinder 29.6 m (97 ft) in diameter by 28.3 m (93 ft) high. The core is located in a cylindrical cavity 4.1 m (31 ft-6 in) in diameter by 15.6 m (51 ft-2 in) high, which is offset from the vertical centerline of the PCRV. (See Figure 3.3-2). On one side of the core cavity are two 5.1 m (16 ft-7 in) diameter vertical cavities, each containing the steam generator and helium circulator for one of the two primary coolant loops. Three 2.1-m (7-ft) diameter cavities, each containing the heat exchanger and circulator for one of the three auxiliary loops, are located on other sides of the core cavity. Horizontal prestress is applied to the PCRV by circumferential wire winding, and vertical

prestress is applied by linear tendons. Horizontal ducts connect the primary and auxiliary loop cavities with the core inlet and outlet plenums. The PCRV is designed for a maximum cavity pressure of 7.79 MPa gage (1130 psig), and it has a water-cooled steel liner with a thermal barrier on the gas-side surfaces.

3.2.1 Safety/Licensing Criteria

The HTGR steamer incorporates all of the inherent safety characteristics of the HTGR power plants and is compatible with urban or industrial siting. Safety characteristics include low reactor power density, high core heat capacity, use of ceramic materials for the core and moderator, single-phase reactor coolant, and employment of a PCRV.

In addition, engineered safety features reduce the likelihood and mitigate the consequences of radioactivity release accidents. Three safety-class core auxiliary coolant loops provide for safe reactor shutdown, and a secondary containment structure contains releases from postulated primary coolant depressurization accidents. Analysis shows that very low risks are associated with locating an HTGR on an urban or industrial site.

3.2.2 Auxiliary Cooling and Heat Rejection

The HTGR steamer has the same reactor core cooling design philosophy as HTGR-SC power plants. In addition to the primary reactor coolant loops, there are three safety-class core auxiliary cooling system (CACS) loops.

The CACS provides an independent means of cooling the reactor core with the primary coolant system pressurized or depressurized. The system is designed to maintain the PCRV core and component temperatures within safe limits.

Each CACS loop includes a water-cooled CAHE, an electric motor-driven auxiliary helium circulator, and a shutoff valve. Heat is rejected from each CAHE by a separate, independent core auxiliary cooling water system (CACWS) loop. The CACWS provides pressurized high-purity cooling water to the CAHEs. Heat is rejected to the atmosphere through forced-draft air-cooler heat exchangers.

3.2.3 Other Design Specifications

The HTGR steamer incorporates reactor auxiliary systems and equipment following the same designs as the HTGR-SC plant. These systems include the reactor neutron and region flow control, fuel handling and storage, helium services (purification and storage), plant protection, plant control, data acquisition, reactor service equipment, PCRV pressure relief,* PCRV seal and test,* and radioactive gas waste.*

* Included in the balance of plant (BOP) scope for ninth plant.

3.3 NSSS SYSTEMS

3.3.1 Introduction

The NSSS of the 1170-MW(t) HTGR steamer cogeneration plant consists of the power generation systems located within the nuclear reactor containment building. These systems function to operate and control the power generated within specified safe limits. The steam generator surface, circulator turbine power, and core size have sufficient design margins to assure that nominal NSSS output of rated steam conditions is achieved. The NSSS includes the following systems, which are briefly described in the succeeding sections:

1. System 11 - Prestressed concrete reactor vessel (PCRv)
2. System 12 - Neutron and region flow control
3. System 13 - Fuel handling
4. System 15 - Fuel shipping
5. System 16 - Reactor service equipment and storage walls
6. System 17 - Reactor internal components
7. System 18 - Reactor core
8. System 21 - Primary coolant
9. System 22 - Main and auxiliary circulator service
10. System 23 - Helium services
11. System 28 - Core auxiliary cooling system (CACS)
12. System 32 - Plant protection system (PSS)
13. System 33 - Plant control
14. System 35 - Plant data acquisition, processing, and display (DAP)
15. System 63 - Gas waste system (GWS)

3.3.2 System 11 - Prestressed Concrete Reactor Vessel (PCRv)

System 11, the PCRv (Figure 3.2-2, includes cavity liners, penetrations, closures, the thermal barrier on the gas-side surfaces of the liner, and two independent pressure relief trains, as well as the concrete-embedded portions of the liner cooling water system, PCRv structural instrumentation and data acquisition equipment, and a leased wire winding machine which is used during vessel construction.

The PCRV functions as the primary containment for the reactor core, the primary coolant system, and portions of the secondary coolant system. The PCRV also provides the necessary biological shielding and minimizes heat loss from the primary coolant system. The prestressed concrete portion of the PCRV and those portions of the penetrations unbacked by concrete, including their closures, form the primary coolant pressure-resisting boundary. The cavity and penetration liners, including closures, form the continuous gas-tight boundary of the PCRV. Penetrations and closures also restrict the leakage flow area from the vessel to acceptable limits in the event of postulated failures. Liner and penetration anchors transmit loads from internal equipment support structures to the PCRV concrete. The PCRV, in turn, transmits these loads to the vessel support structure. During construction, the liners serve as formwork for the concrete.

The PCRV is a thick-walled, cylindrical, prestressed and reinforced concrete structure. It has a central cavity which contains the reactor core. The core cavity is surrounded by two cylindrical steam generator/main helium circulator cavities and three CAHE/auxiliary circulator cavities. Ducts at the top and bottom of the core cavity connect to each steam generator and auxiliary cooling cavity. Three reactor plant cooling water system (RPCWS) header pipe chases, two with pressure relief valve pits at the top, are spaced symmetrically around the core cavity. In addition, several pipe chases are contained in the PCRV wall. The PCRV employs high-strength concrete to satisfy structural and biological shielding requirements. Bonded reinforcing steel in PCRV concrete prevents crack propagation and gives the PCRV ductility under hypothetical overpressure conditions.

The linear prestressing system (LPS), consisting of high-capacity tendons, provides longitudinal prestressing to the PCRV to counteract stresses from vertical pressure loads in the cavities and penetrations and thermal loads in the concrete. The circumferential prestressing system (CPS), consisting of bands of prestressed strand wound in concrete channels on the exterior PCRV cylindrical surface, provides circumferential prestressing to the PCRV to counteract stresses from horizontal pressure in the cavities and penetrations and thermal loads in the concrete.

A liner provides a gas-tight primary coolant boundary covering the surfaces of cavities, communicating ducts, and access openings inboard of any penetration or closure anchorage system. The liners are continuously welded, 12.7 to 25.4 mm (1/2 to 1 in) thick carbon steel membranes. Welded studs are attached to the outside surface of the core cavity liner and the flat bottom head of the steam generator cavity liner. Welded studs are used, as required, in other regions to anchor the liner to the PCRV concrete.

Penetrations, including the shear anchors which are embedded in the surrounding concrete, are constructed from carbon steel. The penetrations are joined to the adjacent liners to maintain the continuity of the membrane enclosing the primary coolant. The embedded portions of penetrations have continuously welded cooling tubes at a pitch similar to that of the adjacent liner. Each penetration includes a gas-tight closure which may be of steel, lined concrete, or composite steel and concrete. The closure may be bolted or welded to the penetration or anchored to the vertical prestress tendons. Penetrations with closures which are expected to be removed are fitted with double-gasketed gas-tight closures. Purified helium is supplied to the annulus between the closure gaskets at 0.10 MPa (15 psia) above the PCR_V upper plenum pressure so that any leakage of the inner gasket will be purified helium into the reactor primary coolant. Any outward leakage will also be purified helium. Closures not normally removed are welded to the liner. Major parameters of the PCR_V liner and penetrations are given in Table 3.3-1.

Flow restrictors are provided, as required, on penetrations and closures to limit the leakage flow area in the unlikely event of failure of the penetration/closure. This is accomplished by plugs held in place by structural members either external or internal to the pressure boundary.

The PCR_V cooling water system consists of 25.4 and 31.8 mm (1 and 1-1/4 in) Sch 40 tubes connected to two independent loops normally operating in parallel. Adjacent tubes are not connected to the same loop. Where possible, counterflow exists between adjacent tubes. The cooling water tubes are continuously welded to the outside surface of the liner, except for such local discontinuities as field welds. Cooling tubes have a pitch range of 101.6 to 203.2 mm (4 to 8 in). Additional PCR_V cooling tubes are embedded in the concrete at approximately 203.2 mm (8 in) from the core cavity sidewall liner to reduce the effect of nuclear heating. Supply and return piping connects both the embedded and welded tubes to the BOP headers located in the cooling water pipe chases.

The PCR_V pressure relief train consists of the block valve, pressure relief valve, and rupture disc. A simplified schematic of a single relief train is shown in Figure 3.3-1. The block valve is capable of isolating a pressure relief train under the following conditions:

1. in the event of relief valve leakage,
2. to permit in-place testing of a relief valve,
3. to permit maintenance of relief train components during plant shutdown, and
4. to terminate blowdown if a relief valve should fail to reset.

The pressure relief valve is closed during normal plant operations. It is automatically opened for pressure relief of the PCR_V when its preset

pressure is reached, and it remains open until its (reduced) reseating pressure is attained. The spring-loaded, pilot-actuated safety relief valve is automatically opened by the process fluid (PCRVR helium) pressure. It can also be manually operated by a remote, helium-actuated pilot valve (e.g., for valve testing and calibration). During normal plant operations, the rupture disc holds negligible pressure on the sealed disc. When activated by a PCRVR overpressure, the disc is ruptured (after opening of the relief valve) and primary coolant gases are discharged directly into the containment building. Figure 3.3-2 shows the PCRVR pressure relief system setpoints.

The thermal barrier typically consists of layers of fibrous mineral insulation compressed against the PCRVR liner by metal cover plates and seal sheets. The compression is affected by attachment fixtures which are connected to the liner. A standard cover plate utilizes one central attachment fixture and four midedge retainers. Thermal protection is provided wherever necessary to satisfy the PCRVR temperature limits. This necessitates the employment of different grades or classes of thermal barrier, depending on the specific heat duty requirements within a designated area or zone. Fifteen thermal barrier zones, shown in Figure 3.3-3, are presently defined as dictated by temperature environment and geometric considerations. Table 3.3-2 defines the major parameters of the thermal barrier. The following three classifications of thermal barrier are employed:

1. Class A. This insulation class is designed for service on the core inlet side of the primary coolant loop, where the hot face is generally exposed to helium at about 318 C (605 F) during normal full power operation. The Class A thermal barrier employs high purity alumina-silica fiber blankets compressed against the PCRVR liner by carbon steel cover plates, seal sheets, and attachment fixtures.
2. Class B. This class of thermal barrier is designed for service on the core outlet side of the primary coolant loop, where the hot face is generally exposed to helium at about 686 C (1266 F) (703 C (1298 F) max) during normal operation at full power. The basic method of insulating is similar to that described for Class A; however, the nickel-base alloy Hastelloy-X replaces carbon steel as the material for cover plates, seal sheets, and attachment fixtures. The Class B insulation blanket is a composite of high- and low-temperature fibrous materials. The cooler portions employ the Class A blankets, while the hotter portions are fabricated of high-purity silica or alumina.
3. Class C. This class of thermal barrier is designed for the same service environment as Class B in normal (mean temperature) operation, but it has the capability to withstand higher local temperatures. The Class C system is used only beneath the core on the bottom head of the PCRVR. The portion of the Class C installation which transmits the weight of the reactor core to

the vessel proper is built up from a composite assembly of hard ceramic insulating blocks. The materials employed are silica, alumina, and silicon-nitride. The nonload-bearing portion of the Class C assembly consists of a two-layered fibrous insulation arrangement plus a hard layer ceramic. The bottom layer (i.e., adjacent to the liner) is essentially identical to a normal Class A thermal barrier and consists of carbon steel cover plates and compressed insulation. The intermediate layer consists of lightly compressed insulation with a Hastelloy seal sheet. The top layer consists of cast-fused silica blocks which sustain the direct gas impingement and diffuse the hot streaks.

The wire winding machine (WWM) traverses the outer circumference of the PCRV while transferring a continuous strand of steel strand from a storage reel placed on the WWM to a channel on the PCRV. The strand is manually anchored at one side of the channel and is of sufficient length to make one solidly wrapped layer within the channel. The machine applies the strand at the design tension and maintains tension until the wrapped strand is manually anchored at the opposite side of the channel. The machine places additional layers in the same manner until the prestressing requirements for the channel are satisfied. If needed, the WWM can remove strand from the channel and rewind it on the storage reels. The WWM is supported from a monorail at the top of the PCRV. Hoists are used to position the WWM at the desired elevation and to install new strand storage reels. The WWM fills two adjacent channels without vertical repositioning.

PCRV structural instrumentation verifies PCRV response to temperature and pressure loads to prove the adequacy of structural material and construction. PCRV instrumentation includes all the sensors and recording and alarm instruments at selected locations on the PCRV structure to provide the following data:

1. temperature of the liner and concrete
2. strain of the liner and concrete
3. linear and circumferential prestressing loads.

3.3.3 System 12 - Neutron and Region Flow Control

The neutron and region flow control system (System 12) consists of two major subsystems:

1. The neutron control subsystem, which comprises:
 - a. The normal control and shutdown system of control rod pairs, power rods, and neutron detectors (see Figure 3.3-4).

- b. The reserve shutdown system (RSS).
 - c. The movable in-core flux mapping and startup detector system.
2. The primary coolant flow control subsystem, consisting of variable orifices and outlet temperature thermocouples for 31 core regions.

System 12 utilizes excore flux detectors, controllers, power rods, control rods, and/or the reserve shutdown material to adjust core reactivity, as required, to meet the demands of the plant control system (PCS, System 33), the plant protection system (PPS, System 32), or the plant operator. System 12 also adjusts the helium flow through regions of the core by incrementally positioning each adjustable core region inlet orifice valve when commanded by the plant operator. Reactor in-core flux distribution and startup-range neutron measurements are also performed by System 12 using movable detectors in selected core locations. This in-core instrumentation system also measures core region outlet temperatures.

Appropriate controls, together with indications of individual rod position, rod motion, rod limit of travel and slack cable, reactor power (flux), helium flow control orifice valve position, reserve shutdown system status, in-core and startup detector position and flux measurements, and core region outlet temperatures are provided in the control room.

Penetrations into regions which do not require neutron or flow control are equipped with a shield plug assembly. This assembly provides neutron and gamma shielding, thermal protection, and gas flow restriction for the penetration and liner. Table 3.3-3 lists major system parameters associated with system 12.

3.3.4 System 13 - Fuel Handling

The fuel handling system comprises all the equipment and subsystems required for the remote handling of both fuel and reflector elements and the various reactor components associated with refueling the HTGR. Included is the fuel sealing and inspection equipment, which seals spent core elements and other solid waste into shipping containers prior to storage and subsequent handling by the fuel shipping system (System 15), as well as the equipment for the transfer of shipping containers between the fuel sealing and inspection facility and the fuel storage wells. Figure 3.3-5 shows the fuel element handling cycle and equipment.

Refueling operations are predicated on a four-year core life, whereby one quarter of the reactor core is replaced each year with new fuel; replaceable reflector elements that reside adjacent to active fuel elements are replaced at eight-year intervals. Both fuel and reflector elements are transferred through refueling penetrations in the top head of the PCR. With a few exceptions, these penetrations contain control rod drives which

have to be removed prior to installation of the fuel handling machine (FHM) on the refueling floor.

Each refueling region, consisting normally of seven columns of fuel and removable reflector elements, is entirely emptied of spent fuel before the placement of new fuel.

Spent fuel and reflector elements are transferred to the fuel container loading facility where reflector elements to be returned to the reactor are retained. The balance and the removed fuel elements are transferred to the fuel sealing and inspection facility (FSIF) where they are sealed in helium filled containers which are subsequently placed into storage.

After reactor shutdown and depressurization of the PCRV, refueling penetration hold down plates and a refueling penetration closure plate are removed. A reactor isolation valve (RIV), which serves as a radiation and atmospheric barrier between the reactor cavity (or refueling penetration) and working personnel on the refueling floor, is attached to the refueling penetration.

The auxiliary service cask (ASC), which is designed to handle both the control rod drive (CRD) unit and the high-temperature filters and absorbers (HTF/A), then removes the CRD assembly. The ASC is then transported to the reactor service building (RSB) by the fuel handling equipment transporter (FHET) and the CRD placed in the CRD storage facility.

The FHM is then secured to the refueling penetration and sealed to the preplaced RIV. A fuel transfer mechanism within the FHM lowers through the refueling penetration into the reactor, picks up a fuel or reflector element, and lifts it back into the shielded portion of the machine. From there it is transferred to a fuel transfer cask (FTC). When loaded with spent fuel elements (five max), the FTC is transported by the fuel transfer cask dolly (FTCD) to the RSB, which is adjacent to the reactor containment building (RCB), placed to mate with a plug actuator above the fuel container loading equipment, and then passed on to the storage facility after sealing. Two FHETS and two rail systems are provided to facilitate the transfer of fuel and reflector elements between the RCB and RSB. Reflector elements are handled in the same manner as fuel elements.

When the working refueling region has been emptied, new fuel and reflector elements are brought in FTCs from storage to the FHM, which transfers the elements into the core. After the region has been refueled and the RIV closed, the FHM is detached and moved to the next refueling region. The ASC then installs a CRD assembly into the refueling penetration. The ASC and RIV are then removed, and connection of service lines and installation of the closure plate complete the process for a refueling region.

The fuel sealing facility, which operates in a helium atmosphere, inserts each fuel element into a storage canister and seals the can before it is transferred to the storage facility.

The equipment comprising the fuel handling system and their functions are listed below.

Fuel Handling Machine (FHM)

Removes and replaces fuel and reflector elements; the machine forms a leak-tight containment with integral shielding.

Fuel Handling Control Station (FHCS)

Located in the reactor service building. The FHCS provides fully automated remote control of all refueling operations within the PCRV together with an in-core viewing capability, control of some ex-vessel refueling operations while having capability of monitoring others.

Fuel Transfer Cask (FTC)

Shielded casks used to convey five full-sized elements at a time between the FHM and the reactor service building (RSB). They are handled by the Fuel Handling Equipment Positioner in the RCB and the building crane in the RSB and Fuel Transfer Cask Dolly.

The casks provide leak-tight containments and feature an integral gate valve interfacing with both the floor valves and the FHM.

Auxiliary Service Cask (ASC)

Primarily used for removing, transferring and installing control rod drives, the cask also accommodates high temperature filters and adsorbers and the reserve shutdown vacuum tool. The cask is conveyed between the RSB and the Reactor Containment Building (RCB) in a horizontal position by the Transporter and is handled by a Fuel Handling Positioner in the RCB and the building crane in the RSB. The cask provides a leak-tight containment with an integral gate valve interfacing with both the floor valves and the Reactor Isolation Valve.

The cask also interfaces with the Control Rod Drive Storage Facility.

Reactor Isolation Valve (RIV)

Placed over a penetration immediately following removal of the penetration closure, the valve provides both biological shielding for personnel on the refueling floor and a means of maintaining continuity of the helium atmosphere before and after installation of the Fuel Handling Machine or the Auxiliary Service Cask.

Fuel Handling Equipment Transporter (FHET)

A self-propelled vehicle running on a set of elevated rails between the RCB and RSB which can accommodate either the Fuel Handling Machine or the Auxiliary Service Cask in a tilted position, or a circulator cask in an upright orientation.

Fuel Transfer Cask Dolly (FTCD)

A special purpose rail mounted dolly operating between RSB and the RCB and which can accommodate a Fuel Transfer Cask in a vertical position. Two dollies and two rail systems are required for the plant.

Fuel Handling Equipment Positioner (FHEP)

Operating in the Reactor Containment Building, each of two rail supported FHE positioners is capable of lifting, transporting, and positioning the fuel transfer machine, the auxiliary transfer cask, and a fuel transfer cask on any one of the penetrations related to fuel handling and located in the refueling floor area.

FHE Positioner Adapter

Used in the RCB on the FHE positioner when fuel transfer casks are handled. This device is required due to the height differential between the FHM, ASC and the fuel transfer casks in combination with the short lift distance of the positioner.

Floor Valve (FV)

Performing a similar function to the Reactor Isolation Valve, one of these valves is permanently installed over each of the transfer chutes leading to the Fuel Container Loading Facility.

Fuel Transfer Chute

The Fuel Transfer Chutes, a BOP installation, connect the floor valves in the Reactor Service Building with the Fuel Container Loading Equipment.

Fuel Container Loading Equipment

Contained in a tank type structure filled with helium, and located below the fuel transfer chutes, the fuel container loading equipment provides for temporary storage of fuel and reflector blocks. Sorting of blocks is accommodated by a conveyor type system and a turntable, while two hoists facilitate grappling and lowering of a fuel or reflector block into the Fuel Sealing and Inspection Facility. This equipment is a BOP installation.

Fuel Sealing Equipment (FSE)

The Fuel Sealing and Inspection Facility (FSIF), a BOP installation, incorporates the following fuel sealing and inspection equipment:

Fuel Transfer Port: Connects the Fuel Container Loading Equipment with the helium chamber in the FSIF.

Helium Chamber: Provides a continuity of atmosphere between the floor valve, the fuel container loading equipment, and the conveyor chamber. A grapple arrangement inside the chamber permits removal and storage of shipping container lids within the controlled environment.

Conveyor: A continuous chain of water-cooled chambers accommodating a number of shipping containers, the conveyor provides both a storage facility and means of moving containers within the FSIF as well as to and from a transfer position between the FSIF and the fuel storage area.

Canning Machine: Located adjacent to the helium chamber, the machine removes, inserts and torques the shipping container flange bolts and performs a seal leak test on the bolted connection.

Storage Area Hoist: Basically an overhead travelling crane, the equipment has a special grapple and guide sleeve arrangement to handle shipping containers, while maintaining them in a proper orientation, between the FSIF access port and the fuel storage area, and between the storage pool and the shipping cask. The hoist also handles the storage area closure plugs.

Control Station: All equipment in the FSIF is remotely controlled from a control station located beyond the shield walls of the FSIF. Control is normally automated but each function can be manually controlled and viewed through shielded windows adjacent the control station.

3.3.5 System 15 - Fuel Shipping

The spent and recycle fuel shipping system provides complete fuel shipping capability. The system provides equipment for the transport of spent and recycle uranium fuel elements between the reactor site and storage and/or recycle plant.

The system equipment is designed to meet all applicable requirements of the regulations set forth by the Nuclear Regulatory Commission (NRC), the U.S. Department of Transportation (DOT), the Association of American Railroads (AAR), and the Interstate Commerce Commission (ICC).

The shipping package design philosophy is based on double containment. Six spent fuel elements or five recycle fuel elements will be packaged in a spent-and-recycle fuel shipping container. The container will be sealed with a bolt-on lid incorporating both an elastomer and a metallic seal. Twelve shipping containers fit into a railroad shipping cask.

3.3.5.1 Fuel Shipping Container

Diameter	0.521 m (1 ft-8-1/2 in)
Length	4.978 m (16 ft-4 in)
Weight (empty)	544 kg (1,200 lb)
Capacity	6 spent fuel elements 5 recycle fuel elements

The fuel shipping container provides containment and facilitates handling of multiple fuel elements. The container has both a metallic and an elastomer O-Ring which provide an effective double seal at the bolt-on lid.

The shipping container is loaded with up to six spent fuel elements and the cover (with seals) is then bolted in place. These operations are done by the sealing equipment at both the reactor site and the recycle plant. The sealed containers are stored in the cask loading area to await shipment inside the shipping casks.

Fresh fuel elements are shipped to the reactor site within the shipping containers. The container holds up to five fresh fuel elements plus the required protective packaging.

3.3.5.2 Shipping Cask

Diameter	2.717 m (8 ft-11 in)
at impact limiters	3.023 m (9ft-11 in)
Length	5.436 m (17 ft-10 in)
with impact limiters	6.655 (21 ft-10 in)
Weight (without containers)	136,078 kg (300,000 lb)
loaded with spent fuel	152,407 kg (336,000 lb)
Capacity	72 spent fuel elements 60 recycle fuel element

The shipping cask provides the structural, thermal, subcriticality, and shielding integrity of the shipping system. The sealed cask, with a removable cover, serves as the secondary enclosure vessel necessary to meet the double containment design philosophy.

The shipping cask holds 12 shipping containers for a total of 72 spent or 60 fresh fuel elements. Structural and shielding integrity is achieved through the use of a heavy steel outer shell, a depleted uranium gamma shield, and an inner steel shell. An inner aluminum basket aids in heat transfer and also maintains a subcritical shipping container spacing.

The cask is transported by a special-purpose railroad car. The cask is nearly horizontal on the railroad car during transport, being held by front and rear supports built into the railcar. It is erected to a vertical position prior to the loading and/or unloading operations. The

entire cask is enclosed by a removable personnel barrier. The barrier restricts access to the cask by unauthorized personnel.

The cask loading operations start with removal of the personnel barrier and top impact adsorber and erection of the cask to the vertical position. With the collar in place and the loading port cover opened, the shipping containers are loaded/unloaded into/from the cask. The loaded cask with seal and cover in place is checked for radiation dose rate and leak rate prior to shipment to ensure that requirements of the design criteria have been met.

3.3.5.3 Rail Car

Width	3.15 m (10 ft-4 in)
Length	24.38 m (80 ft-0 in)
Height (includes cask and barrier)	4.60 m (15 ft-1 in)
Weight	65,317 kg (144,000 lb)
with loaded cask	217,724 kg (480,000 lb)

The special-purpose railroad flat car used to transport the shipping cask has unrestricted interchange service as specified by the AAR and conforms to the regulations set forth by the NRC and the DOT.

3.3.6 System 16 - Reactor Service Equipment and Storage Wells System

The reactor service equipment and storage wells system consists of a collection of NSSS equipment which is used to service selected areas of a large HTGR.

Each category of equipment is designed to perform a particular function and has only minimal impact upon the design of other equipment in the system. The equipment involved are the CRD storage wells, the reflector storage wells, the circulator handling equipment, the in-core thermocouple service equipment, core service tools, and service facility tools.

3.3.7 System 17 - Reactor Internals Components

The graphite and metal parts of the reactor internals system form a high-temperature and radiation-resistant inner container for the reactor core, which is raised above the PCRV bottom head by posts and separated from the core cavity sidewall by an array of springs. The system provides location, support, and restraint for the reactor core, shielding for the PCRV and liner, and seals to prevent bypass of the primary core coolant gas. The support posts form a bottom plenum for mixing and distribution of the hot coolant to the steam generators and enable relative movements between the core and the PCRV.

The reactor internals system includes all the graphite components of the core support floor, permanent side reflector, and the core peripheral seal; the metal peripheral seal support structure, including those items which attach the structure to the PCRV liner and those which provide the interface with adjacent thermal barrier; the metal core lateral restraint and side shield assemblies; and the metal plenum elements fitting over the top permanent side reflector (PSR) blocks. System parameters are listed in Table 3.3-4.

3.3.7.1 Permanent Side Reflector (PSR)

The PSR consists of columns of stacked graphite blocks which form a cylinder surrounding the outer hexagonal reflector columns of the reactor core. The blocks are shaped to make the transition from the irregular outer boundary, formed by the hexagonal reflector columns, to a polygonal cylinder, which is nominally concentric with the PCRV core cavity liner. In the plan view, the PSR is divided into 24 equally spaced circumferential segments. Each segment is ~ 93 cm (~ 45 in) wide and the radial thickness varies from 41 to 91 cm (16 to 36 in) with an average of ~ 66 cm (~ 26 in). Vertically, each column of the PSR is composed of six equal-height blocks, making a total of 144 PSR blocks. The pattern around the core repeats every 60 deg; so there are four different block plant forms. The blocks in each column are all identical, except the top block which has a metal plenum element fitted over its upper portion. There are eight different block designs in the PSR and four different plenum element designs.

The outer faces of the PSR blocks are identical and form a series of 24 chordal facets of a polygonal cylinder. There are vertical grooves in the corners of all blocks which form channels ~ 76.2 mm (~ 3 in) deep for interfacing with the core lateral restraint (CLR) spring packs. This keying of the CLR with the PSR provides rotational restraint to the PSR and to the core, and the depth of the grooves assures that disengagement cannot occur during extreme seismic movements. Radial inward stability of the PSR blocks is provided by keying of vertical interfaces with adjacent blocks. Horizontal alignment between stacked PSR blocks is maintained by shear ties which permit small relative rotations but no relative horizontal movement between blocks.

During normal power operation, the PSR is subjected to nuclear radiation and to heating by irradiation and heat transfer from the core. To reduce the temperatures of the CLR load-bearing pad and side shield to acceptable values, vertical bypass coolant holes may extend through the full heights of the PSR columns. To accommodate thermal expansion of the PSR assembly, the reflector blocks are installed with 2.0 mm (0.08 in) gaps between adjacent columns. The gaps between PSR blocks close when the blocks attain an average temperature of about 538 C (100 F). As the average temperature increases beyond 538 C (1000 F), the PSR segmented rings expand against the spring packs of the lateral restraint structure. In addition, the temperature difference in the radial direction causes each

block to bow slightly. To accommodate both bowing and thermal expansion, the inner horizontal surfaces of the blocks are relieved.

The PSR blocks will be machined from a medium grade of graphite (similar to Grade HLM of Great Lakes Carbon Company) of proven structural capacity. The feedstock is a petrochemical filler and a coal tar pitch binder. The method of formation is extrusion or molding to produce graphite monoliths with anisotropic properties.

3.3.7.2 Core Support Floor (CSF)

The CSF consists of graphite blocks supported by graphite posts which are, in turn, supported on graphite seats atop ceramic bases on the PCRVR bottom head liner. Each seven-column fuel region is supported and located on a single graphite core support block having the same star shape in plan view. PSR columns are supported on individual peripheral CSF blocks, which also provide the inner face for the core peripheral seal (CPS) and the interfacing slots for the CLR spring packs. Every region CSF block is supported by three graphite support posts arranged in an equally-spaced, triangular pattern. Peripheral blocks are also supported on triangular arrangements of graphite posts, but the spacing cannot always be equilateral for those cases. The ends of the core support posts (CSPs) are spherical and fit into similarly shaped recesses in post seat inserts in the CSF blocks and in post supports at the bottom.

The top of the CSF blocks contains circular recesses where similarly shaped protrusions on the hexagonal bottom reflectors interface with the CSF. Two large dowels in each peripheral CSF block interlock with the associated PSR blocks. Each fuel region CSF block also contains coolant flow passages that direct the core exit coolant gas streams from the various columns of the supported region into a central mixing and temperature measurement chamber within the block. The coolant is then exhausted through a large hole in the block bottom into the lower plenum of the PCRVR. Graphite sleeves across the mixing chamber in each star-shaped block provide a heat-conducting housing for the thermocouples, but protect the thermocouples from direct, high-velocity coolant impingement.

3.3.7.3 Core Lateral Restraint (CLR)

The CLR is composed of metal support assemblies located in a regular array between the PSR and PCRVR liner, and it includes the thermal neutron side shield. There are 168 identical, radially oriented spring pack assemblies attached to the PCRVR through the thermal barrier. There is one spring pack at each intersection of four adjacent PSR or PSR-and-CSF blocks. This results in seven horizontal rows of spring packs with 24 units in each row. The spring packs transfer lateral core loads from the PSR into the liner, and they support the steel plates which form the side shield.

Each spring pack contains seven helical coil springs arranged in parallel inside a cylindrical spring housing with a pair of stiffer disk springs in series at the outer end of the housing. A face plate attached to the inner ends of the coil springs interfaces with the PSR and transfers loads into the spring. Axial and lateral deflections of the coil springs are limited by physical-limit stop features built into the face plate and coil spring housing. Thus, slow relative movements due to thermal expansion and PCRV shrinkage and creep are accommodated by the coil springs without developing excessive static loads.

The clearance between the two disk springs controls the maximum deflection of these springs, and their combined spring rate is approximately 20 times greater than that of the coil springs. Thus, in an earthquake, the coil springs quickly bottom-out, absorbing only a small percentage of the seismic energy, and the disk springs absorb the remaining energy in a relatively small deflection. Low carbon plates 25-mm (1-in) thick form up to 101.6-mm (4-in) thick sandwich layers which, in turn, form the thermal neutron side shield. Each plate has a central hole which fits over the coil spring housing for support, and the shield assembly is bolted to the housing with standard flat washers as spacers between plates to allow room for relative thermal bowing. The edges of the shield plates are bevelled, and on assembly, interleave with the edges of adjacent plates, so that there are no gaps in the shield where neutrons or gamma rays can stream through. There are 672 plates in the side shield.

3.3.7.4 Core Peripheral Seal (CPS)

The CPS is formed by 24 triangular cross-section graphite logs, which fit in the annular space between the CSF and the thermal barrier. A sloping shelf in the outer face of each peripheral CSF block provides the inner seal seat. The outer seat is provided by a metal structure, supported by cantilever beams from the PCRV liner, and is enclosed within the thermal barrier. A corrugated sheet-metal collar with radially-oriented pleats sits over the cantilever beams to prevent bypass coolant flow through the fibrous insulation. The corrugations enable the collar to flex to accommodate the relative thermal expansion, resulting from the large thermal gradient between the inner and outer edges of the collar.

As relative motion occurs between the CSF and the PCRV, the seal logs slide up or down the sides of the V-shaped trough formed by the seal seats to maintain its sealing function. During reactor operation, the core pressure drop acts across the seal to force the seal logs down against the seats. Coil spring retainers are installed from brackets above each log to prevent its being dislodged should an earthquake occur during an unpressurized condition.

A small amount of bypass coolant flow is permitted, primarily between the ends of the seal logs, to keep the CLR, side shield, and sidewall thermal barrier temperatures within design limits.

3.3.8 System 18 - Reactor Core

The reactor core (see Figure 3.3-6) provides for both generation heat and efficient removal and transport of this heat to the steam supply system via the helium coolant. The core is operable in a controlled manner at all times, maintains mechanical integrity throughout its design lifetime, and limits release of fission products to within specified limits. The ability to terminate the fission chain reaction and maintain a shutdown condition is available at all times.

The quantity of uranium and thorium fuel in the core is chosen to maintain the neutron chain reaction at the design power level for the design lifetime. The distribution of this fuel in the core and the reloading sequence of the refueling regions is chosen to meet design constraints on power distributions, fuel and coolant temperatures, fission product release, and fuel element stresses.

The neutron chain reaction is controlled by the movement of the control rods. The control-rod pairs are used to ensure permanent shutdown; the small control rods are used to control normal power operation and load change transients; and the reserve shutdown system is used as a backup safety system for the control-rod pairs.

Helium coolant flow and, hence, core gas outlet temperatures are controlled by flow control valves (part of System 12) in each refueling region, consisting of four or more fuel columns. These valves are adjustable from the control room during core operation. The flow to several one-, two-, and three-column fuel regions on the core reflector boundary is set by fixed orifices in the plenum elements on top of these regions.

A safe, controlled startup is produced by the use of several neutron sources located in the top layer of fuel elements. They provide a continuous neutron source for the fission chain reaction at near-critical and just-critical zero-power conditions. This system includes the fuel elements, the hexagonal reflector elements, the top layer/plenum elements, and the startup neutron sources.

The fuel element is a graphite block with the dual function of containing the fuel and acting as a moderator. Each fuel element consists of a hexagonal graphite block containing drilled coolant passages and also parallel fuel channels into which the individual fuel rods are inserted. The individual fuel rods contain the fuel particles distributed in a graphite matrix. The initial core elements and the reload elements, whether containing fresh or recycle fuel, are identical in geometry.

For the reference cycle with high enrichment, the fissile kernel is uranium carbide surrounded by a buffer layer of low-density pyrolytic carbon, a thinner layer of high-density pyrolytic carbon, a layer of silicon carbide which provides containment of gaseous and solid fission products, and an outer layer of high-density pyrolytic carbon which adds strength to the

coating. The fertile kernel is throrium oxide surrounded by a buffer layer of low-density pyrolytic carbon and an outer coating of high-density pyrolytic carbon, which provides the containment.

For the low-enrichment fuel cycle, the fuel element design would be essentially unchanged except for the fuel rod diameter, which would most likely be reduced to accommodate the typically lower fuel loadings of this cycle. The fuel particle designs would be similar in character to those used in the reference high-enrichment cycle, but specific designs have not yet been selected.

The fuel elements and hexagonal reflector elements are arranged in columns supported on the core support blocks. Each support block under the major portion of the active core corresponds to one fuel region which has a central control column and six surrounding fuel columns. The fuel regions are surrounded by two rows of hexagonal reflector columns which are, in turn, surrounded by the PSR. The core elements follow.

1. Fuel Elements. Graphite hexagonal right prisms with arrays of fuel and coolant holes and a hole for burnable poison shim. Control rod and reserve shutdown holes are included where required.
2. Refueling Region. A section of the core containing adjacent fuel elements which are normally loaded or reloaded in the same time interval.
3. Reflector Elements Graphite hexagonal right prisms. Reflector elements may have coolant holes, control rod and reserve shutdown holes, and shielding material as required, but they do not contain fuel.
4. Top Layer/Plenum Elements. Alloy steel hexagonal components which provide the flow plenums for distributing the flow from the region flow control valves to the individual columns, lateral restraint during refueling, and the flow control valve/lower guide tube assembly support.
5. Startup Neutron Sources. A material, Cf-252, with a suitable container which is inserted into core fuel elements to provide a source of neutrons of sufficient strength to ensure a safe, controlled approach to reactor criticality.

3.3.9 System 21 - Primary Coolant

The principal function of the primary coolant system is to transfer heat from the reactor core to the steam generators to produce steam for process and power generation. In addition, the system must, under all normal and abnormal conditions,

1. Provide parts of the boundary for:
 - a. Containing the primary coolant
 - b. Confining radioactive material
 - c. Limiting release to acceptable values
2. Remove stored heat and decay heat from the core during normal shutdown, upset, emergency, and faulted conditions, as long as the cause of the emergency or fault has not affected main loop cooling availability.

The major system components involved in the heat transfer function are the steam generator, the main helium circulator, and the main circulator helium shutoff valve.

The system uses a constant inventory of helium to transfer heat from the reactor core to the steam generator. The primary coolant system utilizes two steam generator modules in series with two main helium circulators, situated in cavities within the PCRV. The primary coolant helium is forced downwards through the reactor core by means of the two main helium circulators, which derive their power from coaxial synchronous electric motors. The helium exits the core through the core support blocks, where mixing occurs to attenuate hot fuel column temperature streaks. The helium then traverses the lower plenum and enters each of two steam generator cross ducts. From there it flows over the steam generator surfaces and through the circulator to complete the circuit.

The temperature of helium is measured at the exit of each core support block and at the steam generator inlet. These temperatures are controlled by adjustment of the flow control valve or control rod configuration. The control valve for manual adjustment of region flow is situated at the plenum element at the entrance to the core. Steam temperature is used to automatically regulate control rods.

The transfer of heat from the core to the steam generator is accomplished by having the helium pass through both the core and the steam generator. The plant design, however, has various paths whereby the helium bypasses the core; these include a buffer helium flow to the steam generator, auxiliary helium circulator leakage, and a designed bypass flow between the segmented permanent side reflector and the core cavity thermal barrier.

The expected operating conditions of the primary coolant are summarized in Table 3.3-5.

3.3.9.1 Steam Generator (SG) The SGs are of a once-through type and have a helical coil configuration in the economizer-evaporator-superheater I (EES) region of the module, followed by straight tube superheater II (STSH) in the central core of the module. Figure 3.3-7 gives a diagram

of the SG module. Table 3.3-6 gives the design and operating parameters of the SGs.

Feedwater is supplied to the bottom of the EES via a single side penetration duct. It enters the economizer through tubes extending from the feedwater tubesheet located in the cold helium flow. The feedwater flow is then directed upward through the economizer portion of the main steam section and is converted to steam in the evaporator.

The tubes exit from the top of the helical bundle and are routed to the perimeter of the bundle. From this perimeter, the tubes are routed within a stagnant gas volume, which is utilized for the location of the bimetallic weld and of the expansion geometry. They are turned downward into the STSH region, where the steam is raised to its final temperature in the superheater. (The superheater extends below the bottom of the helical bundle.) At a location just below the superheater heat transfer section, the tube bundle diameter is reduced in a conical fashion, and the individual tubes enter a single superheater tubesheet.

The end of each tube is accessible at the feedwater and superheater tubesheets; there is no internal subheadering.

Hot helium emerging from the core flows to the SG via a lower crossduct and enters the STSH radially through a right cylinder flow screen. As the gas enters the tube bundle, it turns 90 degrees and flows upward through the STSH section, parallel to the tubes. The gas exits at the top of the STSH and is turned 180 degrees by a turning cone assembly containing a second cylindrical flow screen and a flat plate screen. It then proceeds downward over the helically coiled tubes in the EES section in counterflow to the steam and water.

The gas flow leaves the EES at its lower end, turns, flows radially outward through windows of the outer shroud extension, and enters the annular region between the SG module and the cavity thermal barrier. The helium then exits at the top end of the annulus and flows to the main circulator via the circulator inlet ducting.

3.3.9.2 Main Helium Circulators

The circulator consists of a single-stage, centrifugal compressor, synchronous electric motor drive, water-lubricated compressor bearing system, and helium buffer seal system. Each circulator is located in the same cavity as its SG. (See Figure 3.3-8.) The circulator motor is designed for variable speeds up to a maximum of 2300 rpm. Power is supplied from a variable frequency power supply.

The circulator is rigidly mounted on the SG cavity PCRV closure. The compressor is overhung from a central housing that contains the water-lubricated bearings and seals. The motor is located vertically above and outside the primary coolant pressure boundary.

Helium enters the compressor from an annulus around the SG through an annular inlet section and discharges through an axial diffuser to a duct leading to the reactor core cavity upper plenum. The helium then flows to the core. Parameters of the main helium circulator are given in Table 3.3-7.

A primary coolant shutoff valve is installed in series with each circulator. This valve limits the reverse flow of the helium coolant through the circulator when that circulator is shut down with the other circulator operating. Open- and closed-valve position signals are displayed in the control room.

3.3.10 System 22 - Main and Auxiliary Circulator Service

The principal function of the main circulator service system is to provide water to the integral pump on the circulator shaft. This water is used for circulator bearing lubrication and cooling. Another function is to provide the circulator labyrinth seals with purified buffer helium to prevent inleakage of bearing water to the primary coolant and outleakage of primary coolant to the service system. The system also provides high-pressure helium to actuate the circulator static seals and air to actuate the circulator brakes.

A main circulator service module for each circulator contains all of the service system equipment associated with one circulator. Major equipment includes a surge tank, two bearing water boost pumps (one standby), two bearing water filters (one standby), a bearing water cooler, a helium/water drain cooler, an auxiliary jet supply pump, a bearing water make-up pump, two helium dryers (one being regenerated), and a regeneration heater.

The auxiliary circulator service system provides a supply of purified buffer helium to prevent inleakage of motor bearing lubricant to the primary coolant system or leakage of primary coolant into the motor casing. It removes oil vapor carried over in purge helium from the auxiliary circulators and also removes and replaces the motor bearing lubricant when an auxiliary circulator is shut down.

The helium purge flow and bearing oil change-out preparation for each auxiliary circulator is performed by its associated service module. The three service modules are piped to a common oil adsorber module where purged helium and entrapped oil are separated. Purged helium is then returned to the helium purification system. There is one bearing-oil module per plant which is portable, and which as a maintenance item is attached to a service module to change the bearing oil in the auxiliary circulator upper and lower bearing cavities.

3.3.11 System 23 - Helium Services System

The helium services system consists of the helium purification system (System 23-1) and the PCRV seal system (System 23-2). These systems

perform chemical purification of helium flow circuit for use as seal, buffer, and purge gas.

The helium purification system is designed to remove helium from the primary coolant loop and process it to remove particulates, chemical impurities, and radioactivity, so that the resulting gas can safely be used as a clean gas purge where needed throughout the plant. The helium purification system will not serve as the main radioactivity-level control within the primary coolant, except for long-lived gaseous isotopes. The helium purification system will, however, serve as the main chemical-impurities-level control in the primary coolant. The normal flow requirements for purified helium from the helium purification system are established by the various clean helium purge requirements throughout the reactor plant.

When the PCRV helium inventory is transferred to the helium storage system, the helium purification system removes radioactive impurities, and to as great an extent as possible, all chemical impurities. The helium purification system compresses purified helium recycled from the main and auxiliary helium circulator service systems to be used as purge gas. The system also periodically processes radioactive gas waste to separate the chemical and radioactive impurities and to concentrate the radioactive impurities.

The purified helium requirements for the 1170-MWt steamer cogeneration plant at normal rated power operations are given in Table 3.3-8.

The helium purification system consists of filters, adsorbers, oxidizers, coolers, heaters, compressors, and associated piping, valving, instrumentation, and controls necessary to purify primary coolant and return the purified helium to the PCRV:

1. There are a total of four high-temperature filters located in penetrations on the top head of the PCRV. Each filter has a dust storage capacity of $\sim 0.42 \text{ m}^3$ ($\sim 15 \text{ ft}^3$) (72.6 kg (160 lb) of graphite dust).
2. There are two high-temperature adsorbers located in penetrations on the top head of the PCRV. Each adsorber contains impregnated charcoal to provide for the chemisorption of volatile electro-negatives such as iodine.
3. The oxidizer-cooler module contains the following equipment:
(a) an electrical preheater, an oxidizer, which contains $\sim 1588 \text{ kg}$ ($\sim 3500 \text{ lb}$) of copper oxide catalyst, (b) an aftercooler, which has a duty of 393.95 kJ/s ($1,345,000 \text{ Btu/hr}$), and (c) a liquid drain tank, which has a capacity of 0.085 m^3 (3 ft^3).

4. The dryer module consists of two molecular sieve dryers, each containing 1814 kg (4000 lb) of molecular sieve.
5. The low-temperature adsorber module contains (a) a low-temperature heat exchanger with a duty of 334.20 kJ/s (1,141,000 Btu/hr), (b) two low-temperature adsorbers, which contain 0.949 m³ (33.5 ft³), 454 kg (1000 lb) of charcoal, (c) two liquid nitrogen tanks, and (d) one filter.
6. There are two oil-free reciprocating compressors with a capacity of $\sim 47,195$ cm³/s (100 actual ft³/min) at suction conditions of 6.72 MPa gage (975 psig) and 43 C (110 F) and discharge conditions of 7.41 MPa gage (1075 psig) and 60 C (140 F). Each compressor is equipped with inlet and outlet pulsation bottles and a water-cooled aftercooler.
7. There is a regeneration oxidizer module which is yet to be selected and will be sized by a qualified supplier.
8. There is a regeneration module which consists of (a) a water cooler with a design duty of 91.4 KJ/s (312,000 Btu/hr), (b) a diaphragm compressor with a design capacity of 9439 cm³/s (20 actual ft³/min) at 4.83 MPa (700 psi), 48.9 C (120 F) suction, (c) molecular sieve dryer which contains 1814 kg (4000 lb) of type 5A molecular sieve, and (d) a regeneration heater with a design duty of 120 KJ/s (410,000 Btu/hr).

The PCRV seal system comprises the seal services to selective PCRV penetration closures and includes the piping, valves, and instrumentation outside of these penetrations. This system detects penetration closure seal leakage and includes capabilities for optional purge services to selective penetrations. The system pressurizes the seal interspaces of selective penetration closures to prevent leakage of primary coolant, and it depressurizes seal interspaces, as required, for maintenance and/or PCRV depressurized conditions (inner seal not designed for full reverse pressure differential). The system continuously monitors the integrity of these seals and provides the means for leak testing newly installed seals. PCRV penetration closures are secured with a pair of elastomer seals, an inner one generally sealing against PCRV leakage and an outer one usually preventing leakage to the (ambient) containment building.

During normal reactor operations the seal interspaces are continuously pressurized with helium from the helium purification header. The interspace pressure is 0.10 kPa (15 psi) higher than the PCRV upper plenum pressure. Helium flow, compensating for the total integrated leakage from all penetration closures, is continuously indicated at the seal test module. When the preset leakage rate has been exceeded, an alarm is actuated and a solenoid-operated valve automatically shuts off the supply of pressurizing helium to all seal interspaces. The same alarm is registered on a DAP system unit, located in the control room. Unless the leak is excessive

or is accompanied by radiation alarms in the containment building, no emergency action need be taken; an operator can manually activate the solenoid valve to repressurize the seal system. This leakage would be clean helium flow to the PCRV or to the containment building. As convenient, the leaking closure would be traced and subsequently repaired (e.g., during a planned shutdown).

During abnormal operations and prior to PCRV pressure/leak testing, each of the penetration closures would be leak tested, including those not directly serviced by this system. A portable test rack would be connected to the seal test module (located in the containment building), which contains a helium bottle, pressure regulating equipment, and recording instrumentation for pressure differential measurements. The seal test module equipment would be used to certify the integrity of each penetration closure. Valves on the seal test module would be adjusted to permit testing of a single closure, and its seal interspace would be accurately pressurized (e.g., to 0.10 kPa (15 psi) above PCRV pressure). Integrity would be verified after valve-off from the pressure source by an observed absence of pressure decay. Pressure loss to PCRV pressure indicates an inner seal leak, and pressure decay below this point would suggest an outer seal leak. The same procedure would be used after the replacement of new seals.

Following an excess leakage indication, the following leak detection and isolation procedure would be performed using the seal test module located in the containment building:

1. The penetration closures would be repressurized with helium from the purified helium header by manually opening the valve which had previously been shut automatically.
2. A valve subsection, serving 20 or fewer closures, would be independently isolated (valved-off) to determine which subsection was leaking (i.e., the leak rate will be reduced when the faulty section is valved-off). The flow indicator will be continuously monitored during testing.
3. After detection, the leaking subsection would be valved-open, then one closure at a time would be successively closed (then opened) to isolate the leaking interspace. (Note: Several closures could leak simultaneously, contributing to the overall leak.) The leaking closure can be isolated by noting the reduction in flow when its feed valve is closed. If the leakage were excessive, the reverse procedure would be used, i.e., all closure valves in the leaking subsection would be closed and one valve at a time would be opened (then closed) to isolate the leaking interspace.

4. Leakage in the closure can be confirmed by testing with the portable rack, as described previously, and inner or outer seal leakage can be identified.

3.3.12 System 28 - Core Auxiliary Cooling System (CACS)

The NSSS portion of the CACS includes the auxiliary helium circulators and their drive motors, motor controls, diffusers, and valves (Subsystem 28-1), the CAHEs (Subsystem 28-2), and control instrumentation and hardware (Subsystem 28-3).

The function of the CACS is to provide an independent means of cooling the reactor core with the primary system pressurized or depressurized. The CACS cooling maintains the temperatures of all PCRV components within safe limits.

The CACS consists of three cooling loops, each capable of removing 100 percent of the core residual and decay heat for safe cooldown from 102 percent of reactor steady-state power level under pressurized conditions. Two loops are required in the depressurized condition, each with 50 percent capability. The auxiliary circulator forces circulation of the primary coolant with air or water inleakage. The core coolant gas is circulated through CAHE, which delivers the heat to the CACWS for disposal to the atmosphere.

Table 3.3-9 summarizes overall CACS performance at three design points which correspond to peak duties in three transient cases. These cases include (1) core cooling with depressurized PCRV and pure helium, (2) with depressurized PCRV and air ingress, and (3) with pressurized PCRV and water ingress.

The CACWS (System 47) is a BOP system providing water and heat rejection to the CAHEs and is part of CACS. The overall CACS performance described herein is based on the assumption that the NSSS portions of CACS interface with a CACWS. Table 3.3-10 summarizes CACWS parameters.

The CAHE (Figure 3.3-9) is a straight-bayonet-tube heat exchanger. Hot gas from the lower cross duct enters the bottom of the straight-bayonet-tube bundle and flows upward through the tube bundle parallel to the tubes. Water flows countercurrent to the helium flow inside the tubes. Smaller diameter core tubes convey the water to the upper end of the heat transfer tubes. The water exits the core tube and flows through the annulus between the two tubes to the bottom of the tube bundle, gaining heat on the way.

The straight-bayonet tubes extend upward from a tubesheet, which is welded into the bottom of the CACS cavity in the PCRV; the tubes are welded to the tubesheet. At the opposite or upper end, the tubes terminate in a cap welded to the end of the tubes. This system forms the primary coolant pressure boundary of the heat exchanger. The core tubes are connected to a tubeplate inside the water box of the heat exchanger, which separates

the hot and cold water flow streams and serves as a support for the core tubes. A closure head is then bolted on to the bottom of the heat exchanger, forming the lower end of the water box and the water/containment pressure boundary. The connections for the inlet and outlet water piping penetrate the bottom head. The outlet connection is in the center of the head. Heated water emerging from the bayonet tubes flows to the center of the water box and through an opening in the core tube tubeplate into an internal conduit to the outlet water piping connection. The inlet nozzle is off center; flow to the heat exchanger enters the plenum below the core tube tubeplate and then flows directly into the core tubes. A summary of the CAHE module design parameters is given in Table 3.3-11.

The auxiliary circulators are electric-driven axial flow compressors. The electric drives are variable-speed induction motors. The motor power supplies include static inverters to provide variable frequency control. Each motor is operated by an independent speed controller. For commonality of hardware with larger HTGR plant designs, the circulator motor is sized to a higher design rating than the maximum required in this application. The power supply sizing corresponds to the lesser figure.

An auxiliary circulator helium shutoff valve is installed in series with each circulator. This valve limits the reverse flow of the helium coolant through the circulator when the main circulators are operating. Open-and-closed-valve position signals are displayed in the control room. The valve is a butterfly check valve which is opened by the auxiliary circulator pressure when the CACS is actuated.

Table 3.3-12 summarizes the auxiliary circulator parameters.

3.3.13 System 32 - Plant Protection System (PPS)

The PPS includes all of the equipment from the sensors to the input terminals of the actuated devices involved in a public protection function.

The PPS prevents any unacceptable releases of radioactivity which could constitute a hazard to the health and safety of the public by initiating actions to protect the fission product barriers and to limit the release of radioactivity if failures occur in the barriers. To accomplish these functions, the PPS provides the subsystems listed below:

1. Reactor Trip System This system initiates rapid reduction in reactor power level following reactivity excursions, loss of adequate core cooling, or other events, to limit damage of fuel coatings and preserve the integrity of the primary coolant barrier.
2. SG Isolation and Dump System This system limits the quantity of water that can leak into the PCRV following failure in the SGs to limit damage to the fuel and to protect the reactor vessels internals and the PCRV. This system responds to signals

from the moisture monitoring system (below) to isolate and dump an SG if a high moisture level should occur in a primary coolant loop.

3. SG Shutdown System This system limits the temperatures of the SG tubes following mismatches of primary and secondary coolant flows to maintain the integrity of the primary coolant boundary.
4. CACS Initiation System This system initiates auxiliary core cooling and trips the reactor following the loss of all main loop cooling to preserve the integrity of or to minimize damage to the fuel coating and the primary coolant barrier.
5. Containment Isolation System (CIS) This system limits the pressure buildup in the containment due to secondary system leaks to preserve the integrity of the containment structure. The CIS also limits release of radioactivity from the containment if the primary coolant boundary should fail. The environmental conditions inside the containment which result from these postulated events set the design bases for equipment qualification. The CIS is a BOP system.
6. PCRV Pressure Relief Block Valve Closure Interlock This system prevents the simultaneous closure of both PCRV relief block valves to ensure that at least one PCRV relief valve is available to protect the PCRV and primary coolant boundary.
7. Rod Bank Withdrawal Interlock This system prevents simultaneous withdrawal from the reactor core of more than one control rod bank or one power rod bank to restrict the possible reactivity insertion events that can be initiated by rod withdrawals.
8. CAHE Isolation System This system detects conditions which indicate a leak between the CACWS and the primary coolant system, blocks the main loop moisture monitor signals, and initiates action as appropriate to either alarm this condition to the operator for orderly shutdown, or to automatically close the CACWS isolation valves. CAHE isolation is a BOP system.
9. Moisture Monitoring System This system consists of two moisture monitor modules, one for each SG. The system samples the primary coolant from the associated SG cavity in the PCRV, detects high moisture content due to inleakage, and provides signals to (a) the SG isolation and dump system for isolation and dump of the appropriate SG and (b) the reactor trip system to trip the reactor on detection of high moisture level. It is comprised of the sample rakes, sample lines, filters, coolers, hygrometers, tankage, and compressors necessary to provide the safety function.

Plant availability and equipment protection are not the main functional requirements of the PPS, but lie primarily within the scope of the plant control system.

3.3.14 System 33 - Plant Control

The plant control system is an integrated system that includes the instrumentation and equipment associated with the monitoring and control of the plant. Included in the plant control system are the overall plant control loops to maintain rated steam conditions during normal operation and systems which provide protection of major equipment items and serve as a first line of protection for incidents which could otherwise result in the need to PPS (System 32) action. The design and supply of control room consoles and boards is provided as Subsystem 33-2. Also included is the nonsafety-related analytical instrumentation for the NSSS (Subsystem 33-3).

Subsystem 33-3 includes both analytical instrumentation and the associated piping and controls needed for gas sampling, gas conditioning, and related operations.

The plant control system provides for safe plant operation and high plant availability, regulates reactor power, and controls the pressure and temperature of steam. The system has the capability of automatic load following over the range and rates of change specified for the plant. It also remains operational during abnormal occurrences, such as loop trip, turbine generator trip, boiler feedpump loss, and electrical load and process steam rejection.

The plant control system also provides automatic actions for protection of major equipment components and protective actions during certain incidents which would result in the need for PPS response. These control actions include those required as a result of failure of an active NSSS component. (Failure of these protective functions does not jeopardize public health or safety.)

The control room portion of this system (Subsystem 33-2) provides for the design and supply of the plant control room with continuous power production controls operated from a centralized control and monitoring station. The design provides for a control operator's console meeting the above requirements, together with associated boards and consoles for control and monitoring of other plant systems.

The analytical instrumentation portion of this system (Subsystem 33-3) provides the nonsafety-related NSSS analytical instrumentation required to monitor primary coolant purity and gas purification and recovery system performance by measuring and analyzing chemical impurities in several specified process fluids.

3.3.15 System 35 - Plant Data Acquisition, Processing, and Display (DAP)

The DAP system is a dual-computer-based interface between the plant instrumentation and the plant operator. The DAP services both the NSSS and BOP and replaces conventional plant alarms, indicators, and recorders. Redundancy of computers and critical peripheral equipment is used for maximum availability. In addition to the two computers, this system includes redundant disc memories, programmer consoles, magnetic tape units, line printers, color cathode ray tube displays, plant operator function keyboards, and trend recorders.

This system converts certain instrument signals to engineering units and provides visual and audible alarms, periodic logs, point trending, sequence-of-event recording, post-trip review, and displays of various operator information and procedural instructions on multicolor cathode ray tubes (CRTs). Various applications programs are executed in the system computers to provide operational or plant performance information. Categories of these applications programs are:

1. Core reactivity status
2. Core temperature and power distribution
3. Heat balance calculations
4. Plant performance calculations
5. Operator guides

The DAP system performs data acquisition, processing, alarming, engineering calculations, plant performance calculations, periodic logs, and operator guidance. The DAP system does not perform any control or safety function, but monitors both of these functions automatically and independent of the operator and the PPS. The DAP system informs the operator whether the limits exceeded are safety or operation related.

The DAP system is completely integrated with the overall plant instrumentation and the control room panels. Information is presented in formats which are easily assimilated by the operator. Error is reduced and operating efficiency is increased because the operator is less occupied with data recording and analysis. The operator has more time to observe and direct his attention to the overall plant operation. The operator can readily obtain information concerning all monitored and calculated variables in one central location, greatly reducing the need for roving data collection. The functions provided in the DAP system include:

1. Operator/computer communication
2. Analog point processing and alarm

3. Digital point processing and alarm
4. Alarm suppression
5. Sequence of events monitoring
6. Operator demand functions
7. CRT information displays
8. Analog and numerical trend recording
9. Data logging
10. Post trip review
11. Diagnostics and self checking
12. Nuclear applications programs
13. Performance calculations
14. Operator guides

The DAP system is designed for exceptional reliability and continuous operation. The system consists of two digital computers, each with its own main memory, auxiliary memory, system interfaces, peripheral input/output, clock, interrupt logic, watchdog timer, and programmer console. Input signals are connected to both computers, providing complete duplication of input signals. Output signals are connected to both computers, providing complete duplication of output circuits; although, only one set of outputs is used at one time. Peripheral equipment is either dedicated to a computer or is shared by being switched from one computer to the other.

3.3.16 System 63 - Gas Waste System (GWS)

The GWS collects all radioactive and potentially radioactive gas wastes generated in the reactor plant, excluding PCRV leakage and other equipment leakage. The system also provides sample connections for radioactivity analysis of the contained gas.

The system consists of 13 major equipment items, including a process module, vacuum tank module, compressor control modules, compressors, and storage tanks. The process module contains the piping and valving necessary to perform the following functions:

1. Send potentially radioactive gas to the system vacuum tank.
2. Discharge low-activity gases from the high-pressure surge tanks to the reactor service ventilation system.
3. Direct high-activity gases from the high-pressure surge tanks to the system compressors for transfer to the helium purification system for further processing.

The vacuum tank module contains a gas waste vacuum tank and liquid drain tank plus associated piping and valving. The gas waste vacuum tank collects radioactive gas which is to be compressed and stored in the gas waste surge tank.

The compressor control modules contain the piping, valving, and some of the instrumentation necessary to operate the compressors.

The compressors are diaphragm machines. Three compressors each are used for first- and second-stage compression. The first and second-stage relationships are fixed and exclusively paired. The compressors have a capacity of 23,600 cm³/s (50 ft³/min) at 0.083 MPa (12 psia). The compressors take suction at 0.052 MPa (7.5 psia) (min) and discharge at about (later) MPa (psia) (max).

3.4 FUEL

3.4.1 Fuel Strategy

Considerable flexibility in the HTGR fuel cycle means that different fuel cycles can be used with essentially no change in the plant design. The fuel material, whether uranium, thorium, or plutonium, is contained in very small ceramic-coated particles, which occupy a relatively small and readily adjustable fraction of the fuel element volume, and thus, do not affect its structural integrity. The use of helium, which has no effect on the reactor physics characteristics, as the coolant, combined with the relatively low-power density, allows the fuel element design and heterogeneous fuel material within the moderator to be modified as appropriate to a particular fuel cycle.

The graphite fuel element block is the same basic unit whatever the fuel cycle. These elements are grouped into reload regions consisting of seven columns of fuel blocks. A fraction (typically one-quarter to one-sixth) of the reload regions are reloaded annually. The transition from one fuel cycle to another is done simply by progressively loading new elements appropriate to the new cycle at each annual reloading. The loadings, during the transition between cycles, can be tailored to match the characteristics of the regions they replace to avoid excessive power peaking or reactivity swings.

From an economic standpoint, the optimal fuel cycle in the HTGR is the HEU-Th cycle, where HEU indicates high-enriched uranium (HEU), 93 percent enriched in the fissile U-235 isotope. The economic advantages of the HEU/Th cycle are greatest when the fissile U-233, produced by neutron captures in the thorium, can be recycled in subsequent reloads. Even without recycle, the HEU/Th cycle maintains a significant advantage over alternative cycles.

The use of high enriched uranium has recently caused increasing concern because of the possible proliferation of materials useful to the manufacture of nuclear weapons. Alternative fuel cycles have therefore been designed for the HTGR which use lower enrichment uranium.

If all the thorium is eliminated, the HTGR can be run on low-enrichment uranium of about 10 percent to 15 percent U-235. If the enrichment is raised to 20 percent, which is the current export limit, it is possible to include a significant amount of thorium in the cycle and retain the economic advantages of U-233 recycle. This alternative is designated the MEU/Th cycle, where MEU indicates medium-enriched uranium (MEU). From the standpoint of proliferation resistance, the recycle U-233 can be mixed with natural uranium or depleted U-238 to reduce the fissile enrichment to a safe level (typically 12 percent for U-233).

It is presently expected that any HTGR built in the near term would probably begin operation on the MEU/Th cycle. As more extensive international arrangements are developed to safeguard nuclear materials, a transition to the more economical HEU/Th cycle could be carried out with no effect on the reactor performance as a steam supply system.

Data is given below for both cycles. This data is based on extensive optimization studies for the 2240-MW(t) HTGR-SC. Since the 1170-MW(t) HTGR steamer design described in this report differs only in core size, the results of these optimization studies provide a valid basis for estimating fuel loadings.

3.4.2 HEU/Th Cycle

The HEU/Th cycle optimizes at significantly different thorium loadings, depending on whether the bred U-233 is recycled. For recycle, the thorium loading at equilibrium is characterized by a carbon/thorium atom ratio (C/Th) of 180. For no recycle, the optimum is C/Th = 335. In either case, the cycle length is four years with annual reloading of one quarter of the core. The initial core and the first several reloads typically have special loadings to accommodate the buildup of fission products and the beginning of recycle. In a mature HTGR economy in which a recycle plant is built and operating, recycle in a new reactor could begin with the second reload. If this were not the case, the bred U-233 in the discharged fuel could be stored for future recycle. Alternatively, if recycle were not anticipated, the plant could be run on the lighter C/Th = 335 loadings for which less U-233 is produced. Almost any possibility in between these alternatives could also be handled.

The smaller core size of the steamer leads to greater neutron leakage than in the larger core assumed in the optimization studies. Since the uranium and the thorium are in separate fuel particles and independently adjustable, the U-235 loading per block would likely be raised rather than changing the C/Th ratio.

3.4.3 MEU/Th Cycle

With the lower enrichment of 20 percent U-235 assumed in the MEU/Th cycle, less thorium can be loaded. The C/Th ratios selected from the optimization studies are 600 and 850 for recycle and nonrecycle, respectively. The difference, relative to the HEU/Th cycle, is the large amount of U-238 associated with the U-235, which represents an additional fertile material and produces some plutonium. Adjustments to this cycle to accommodate a reactivity change, such as the increased leakage of the smaller core, would likely require a change in the C/Th ratio, as well as in the uranium, since the latter has a lower worth at its lower enrichment. The optimum C/Th ratios are, therefore, expected to be larger, perhaps 700 and 900 for recycle and nonrecycle in the steamer.

As with the HEU/Th cycle, a variety of recycle options could be exercised, including no recycle, recycle of bred U-233, and recycle of all uranium. It is assumed that recycle of the plutonium produced from U-238 would not be allowed in any situation that did not also allow the use of the HEU cycle. For the HTGR, the amount of plutonium discharged in the MEU/Th cycle is relatively small and does not represent a large penalty to the fuel cycle economics.

TABLE 3.2-1

HTGR STEAMER NSSS PERFORMANCE DATA AT 1170 MWt

	<u>Heat MW(t)</u>
Generated by core	1170
NSSS power	1180
Transferred by steam generators	1180
Added by circulators	17.5
Loss to CACS	2.26
Loss to PCRV liner cooling system	
Steam generator cavity	1.79
Core cavity	1.46
CAHE cavity	0.66
Miscellaneous losses	1.26

	<u>Helium Pressure</u>	
	<u>(MPa)</u>	<u>(psia)</u>
Operating pressure	7.239	1050
Pressure drop	0.127	18.38

	<u>(kg/s)</u>	<u>(lb/hr)</u>
Total circulating helium flow	610.21	4,843,800

TABLE 3.3-1

MAJOR PCRV LINER AND PENETRATION PARAMETERS

Overall dimensions	
Diameter, m (ft-in)	26.51 (87-0)
Height, m (ft-in)	28.35 (93-0)
Core cavity	
I.D., m (ft-in)	9.60 (31-6)
Height inside, m (ft-in)	15.60 (51-2)
Offset from PCRV centerline, m (ft-in)	2.79 (9-2)
Steam generator cavities	
I.D. at midheight, m (ft-in)	5.05 (16-7) (5.21 (17-1) at top head)
Height including top closure, m (ft-in)	25.35 (83-2) (2.13 (7-0) above PCRV)
Auxiliary loop cavities	
I.D. at barrel midheight, m (ft-in)	2.13 (7-0) (2.44 (8-0) at top)
Height including top closure, m (ft-in)	28.65 (94-0)
Max cavity pressure/design, MPa (psig)	7.791 (1130)
Circumferential channels	24
Total strand layers	336
LPS strand tendons	252
PCRV support	Integrated with containment
Refueling penetrations	45
Peripheral supports	18
Hot gas inlet ducts	
Diameter, m (ft-in)	1.83 (6-0)
Cold gas inlet ducts	
Height, m (ft-in)	1.52 (5-0)
Auxiliary loop inlet and outlet ducts	3
Side feedwater penetrations pipe chases	
Diameter, m (ft-in)	1.68 (5-6)

TABLE 3.3-1 (Continued)

Relief valve pits	2
Dimensions of cavity, m (ft-in)	1.83 x 2.90 (6-0 x 9-6)
Height of cavity, m (ft-in)	4.88 (16-0)
Reactor plant C.W. system pits	3
Diameter at midheight, m (ft-in)	2.44 (8-0)
Height of pit, m (ft-in)	23.47 (77-0)
Core outlet thermometer penetrations	7
Steam generator inlet thermowell penetrations	2 (1 per steam generator)
CAHE inlet thermowell penetrations	3 (1 per CAHE)
Minor penetrations (top head)	4
Material surveillance	2
Neutron detectors	6
Minor pens (BOT head)	5
Thermal barrier viewing	
Liner/concrete interface temperature, °C (°F)	66 (150)
Normal operation (effective), °C (°F)	121 (250)
Normal operation (hot spot), °C (°F)	121 (250)
PCRV cooling and heating rate (max rate based on mass average), °C/week (°F/week)	7.8 (14.0)
Design primary coolant leakage, %/yr	1
Max allowable primary coolant leakage, (% total inventory/yr)	3.65
PCRV major penetrations - barrel and lower head	
Feedwater side entry	2
Feedwater access cavity	1
Main steam	2
PCRV major penetrations - upper head	
Refueling	45
Auxiliary loops	3
Access, optional	1
Diameter, m (ft-in)	1.45 (4-9)
Steam generator	2

TABLE 3.3-2

MAJOR THERMAL BARRIER PARAMETERS

Zone	Class	Thermal Barrier Areas (cm ² (ft ²))
1 Steam generator cavity	A	658.7 (7,090)
2 Main circulator plug	A	84.0 (904)
3 Upper main cross duct	A	58.5 (630)
4 Top head	A	71.2 (766)
5 Upper sidewall	A	67.7 (707)
6 Center sidewall	A/B	193.2/116.5 (2080/1254)
7 Lower sidewall	B	56.5 (608)
8 Bottom head	C	80.3 (864)
9 Lower main cross duct	B	52.6 (566)
10 Steam generator lower penetration	A/G	26.2/27.3 (282/294)
11 Steam generator cavity	B	78.8 (848)
12 Upper auxiliary cross duct	A	45.7 (492)
13 CAHE cavity	A/B	254.5/44.3 (2739/477)
14 Lower auxiliary cross duct (hor.)	B	53.8 (579)
15 Lower auxiliary cross duct (vert.)	B	129.3 (1392)
17 Peripheral seal		18.8 (202)
		Heat Load To Cooling Water (CW) System (KJ/s (Btu/hr))
Design		10,250 (35 x 10 ⁶)
Expected		5,860 (20 x 10 ⁶)
		Temperature (°C (°F))
Maximum steady state	A	371 (700)
	B	
	C	
Maximum transient	A	371 (700)
	B	
	C	
Maximum 10-hr emergency condition	A	482 (900)
	B	982 (1800)
	C	1371 (2500)
Maximum 1-hr faulted condition	A	593 (1100)
	B	1093 (2000)
	C	1649 (3000)

TABLE 3.3-3

MAJOR NEUTRON AND REGION FLOW CONTROL SYSTEM PARAMETERS

Total regions requiring control and orificing (C&O) assemblies	31
Regions requiring control rod drive and mechanism (CRDM) assemblies ^(a)	28
CRDM assemblies which will be equipped with an in-core flux mapping unit (IFMU)	6
Reflector control rod drive (RCRD) assemblies ^(b)	0, 3 or 6 ^(c)
Regions requiring control rod drive and instrumentation (CRD & Instr.) assemblies ^(d)	3
Ex-core neutron detectors providing input to the flux controller (System 33)	6
Length between top and bottom of the poison material in control rod pairs and power rods	Same as active core length
Ex-core neutron detectors providing input to protection system (System 32)	12 (2 per detector well)

(a) This type of C&O assembly is equipped with a control rod pair and drive, a power rod and drive, an RSS unit, a flow control orifice and drive, and the capability to install an IFMU.

(b) This type of assembly is equipped only with a control rod pair and drive and an RSS unit.

(c) This data will be established later.

(d) This type of C&O assembly is equipped with a control rod pair and drive, a power rod and drive, a flow control orifice and drive, and a startup detector and drive.

TABLE 3.3-4

MAJOR REACTOR INTERNALS SYSTEM PARAMETERS

Permanent outer reflector columns	24
Block shapes in permanent side reflector	8
Core support blocks	79
Block shapes in core support floor	6
Core support posts - seats assemblies	237
Peripheral seal segments	24
Lateral restraint springpacks	168
Thermal neutron shield plates	672
Permanent side reflector thickness, cm (in)	
Max	91 (36)
Min	41 (16)
Av	66 (26)
Lateral restraining spring type	Bilinear
Tangential deflection limit, mm (in)	38 (1.50)
Total radial deflection limit, m (in)	31 (1.225)
Permanent side reflector, kg (lb)	(405,000)

TABLE 3.3-5

MAJOR PRIMARY COOLANT EXPECTED SYSTEM PARAMETERS

Primary coolant loops	2
Reactor power, MWt	1170
Nominal electrical output (net), MW(e)	172
Helium inventory, kg (lb)	
Total	7865 (17,340)
Circulating	5792 (12,770)
Helium flow rate, kg/s (lb/hr)	610.21 (4,843,000)
Helium pressure at circulator discharge, MPa (psia)	7.239 (1,050)
Pressure Drop Data	
Total primary circuit, MPa (psi)	0.1267 (18.38)
Steam generator, MPa (psi)	0.0512 (7.42)
Core inlet gas temperature, °C (°F)	318 (605)
Steam generator inlet gas temperature, °C (°F)	686 (1266)
Circulator inlet gas temperature, °C (°F)	314 (597)
Thermal power to NSSS from other sources (circulator power), MW(t)	17.5
Heat losses from NSSS heat transfer system, MW(t)	
Total	7.43
Standby parasitic CACS heat loss	2.26
Loss to liner cooling system	3.91
Miscellaneous heat losses	1.26
Power to helium circulator motors, total, MW(e)	17.5
Core power density, MW/m ³	7.15

TABLE 3.3-6

MAJOR STEAM GENERATOR MODULE PARAMETERS

Type	Once-through
Pressure drop allowances	
Steam side, MPa (psi)	
Helium side, MPa (psi)	+15% (design evolution)
Support type/location	Welded support skirt; support junction with cavity liner 231 cm (91 in) above center level of lower cross duct.
Purge flow, % of circulator discharge flow	1.5
Helium side	
Flow rate per SG, kg/h (lb/hr)	1,098,340 (2,421,500)
Inlet pressure, MPa (psia)	7.2 (1041)
Outlet pressure, MPa (psia)	7.1 (1033)
Inlet temperature, °C (°F)	685.6 (1266)
Outlet temperature, °C (°F)	313.3 (596)
Pressure loss, MPa (psi)	0.05 (7.42)
Water/main steam	
Flow per SG, kg/h (lb/hr)	869,513 (1,916,950)
Min flow, kg/h (lb/hr)	208,607 (459,900)
Feedwater temperature, °C (°F)	221.1 (430)
Feedwater pressure, MPa (psia)	20.6 (2991)
Main steam temperature, °C (°F)	540.6 (1005)
Main steam pressure, MPa (psia)	17.3 (2515)

TABLE 3.3-7

MAIN HELIUM CIRCULATOR PARAMETERS

Number	2
Weight(each), g (lb)	NA
Circulator speed (rated), rpm	2300
Motor shaft speed (each), kW (hp)	NA
Efficiency, %	
Compressor net	NA
Motor net	NA

NA - Not Available

TABLE 3.3-8

PURIFIED HELIUM REQUIREMENTS - 1170 MW(t) HTGR STEAMER COGENERATION PLANT
(At Normal Rated Power Operations)

User	No.	CFM at Helium Header to Each User (7.343 KPa, 49°C (1065 psia, 120°F))	Mass Flow to Each User (kg/s (lb/hr))	Total Flow	
				(cm ³ /s (CFM))	(kg/s (lb/hr))
Main circulators	2	3490 (7.4)	0.037 (294)	6,980 (14.8)	0.0741 (588) ^(a)
Auxiliary circulators	3	2830 (6)	0.0300 (238)	8,500 (18.0)	0.0900 (714) ^(b)
PCRV pressure relief valve penetrations	2	3490 (7.4)	0.0370 (294)	6,980 (14.8)	0.0740 (587)
PCRV seal system (23-2)	--	--	--	0	0
Reserve ^(c)	--	--	--	13,600 (28.8)	0.1438 (1141)
TOTAL				36,000 (76.4)	0.3818 (3030) ^(d)

(a) Includes 0.0370 kg/s (294 lb/hr) helium recycle flow from main circulator service system (903755); 50% recycle.

(b) Includes 0.0675 kg/s (536 lb/hr) helium recycle flow from auxiliary circulator service system (903758); 75% recycle.

(c) Purified helium in excess of design purge requirements will be used to increase the relief valve purge flow for return to the PCRV.

(d) 0.2772 kg/s (2200 lb/hr) purified primary coolant plus 0.1046 kg/s (830 lb/hr) recycled from main and auxiliary circulator service systems.

TABLE 3.3-9

CACS SYSTEM PERFORMANCE

	Depressurized PCRV		Pressurized PCRV
	Pure Helium	Air Ingress	
CACS loops operating	2	2	1
Primary coolant pressure, MPa (psia)	0.163 (23.6)	0.163 (23.6)	7.239 (1050)
Primary coolant molecular weight	4	12	5.9
Heat duty per CACS loop, MJ/s (Btu/hr)	12.9 (44.0 x 10 ⁶)	11.0 (37.6 x 10 ⁶)	37.2 (127.0 x 10 ⁶)
Primary coolant circuit			
Flow per CACS loop, kg/s (lb/hr)	3.846 (30,521)	9.113 (72,325)	17.651 (140,093)
CAHE inlet temperature, °C (°F)	952 (1,746)	752 (1,746)	860 (1,580)
Core inlet temperature, °C (°F)	307 (585)	363 (685)	371 (700)
Pressure drop, MPa (psi)	NA	0.0026 (0.38)	NA
CAHE	NA	0.0019 (0.27)	NA
Core	NA	0.0005 (0.07)	NA
Ducts and plena	NA	0.0003 (0.04)	NA
Secondary (CACWS) circuit			
Flow per CACS loop, kg/s (lb/hr)	65.5 (0.52 x 10 ⁶)	65.5 (0.52 x 10 ⁶)	65.5 (0.52 x 10 ⁶)
CAHE inlet temperature, °C (°F)	79 (174)	74 (165)	164 (328)
Air blast heat exchanger inlet temperature, °C (°F)	128 (263)	116 (241)	284 (544)
Pressure drop, MPa (psi)	NA	NA	1.51 (219)
CAHE	NA	NA	0.10 (15)
Piping and air blast heat exchanger	NA	NA	1.41 (204)

NA - Not Available.

TABLE 3.3-10

CACWS DESIGN DATA

Water pressure in pressurized case, MPa (psia)	10.34 (1500)
Air blast heat exchanger area, per loop, m ² (ft ²)	365 (3929)
Air blast fan power requirements, per loop, kW (hp)	338 (453)
CACWS pumping requirement, per loop, kw/hp	129 (173)
Air blast heat exchanger configuration	
Airflow, kg/s (lb/hr)	213 (1.69 x 10 ⁶)
Approach air velocity, cm/s (ft/min)	679 (1337)
Vane-axial fans	2
Diffuser length, m (ft)	3.05 (10)
Fan efficiency, %	77
Tube rows	9
Tube diameter, mm (in)	25.4 (1)
Tube pitch, mm (in)	60.3 (2-3/8)
Fin density, per cm (per in)	4 (11)
Fin height, mm (in)	15.9 (5/8)
Fin thickness, mm (in)	0.51 (0.020)
Ambient conditions	
Max temperature, °C (°F)	43 (110)
Min temperature, °C (°F)	-34 (-30)
Min atmospheric pressure, MPa (psia)	0.093 (13.5)

TABLE 3.3-11

MAJOR CORE AUXILIARY HEAT EXCHANGER (CAHE) MODULE PARAMETERS

CAHE modules	3
Heat transfer tubes	234
Tube size (o.d. x t - mm (in))	38.1 x 3.05 (1.5 x 0.12)
Heat transfer capability (UA)	
Depressurized, pure helium, W/°C (Btu/°F)	15.4 (29.2)
Depressurized, air ingress, W/°C (Btu/°F)	11.9 (22.5)
Pressurized, W/°C (Btu/°F)	49.6 (94.0)
Heat transfer area, effective, m ² (ft ²)	189.0 (2,034)
Type	Bayonet tube
Penetration location in PCRV	Bottom

TABLE 3.3-12

MAJOR CORE AUXILIARY CIRCULATOR PARAMETERS

Core auxiliary circulators	3
Compressor type	Two-stage axial flow
Drive type	Electric induction motor
Speed control	Solid state inverter system
Design power, circulator motor, kW (hp)	671 (900)
Maximum required power, motor power supply, kW (hp)	116 (156)

Figure 3.2-1 Heat Cycle Diagram 1170 MWt HTGR Steamer Cogeneration Plant

GENERAL NOTES:

PCRV DESIGN IS IN COMPLIANCE WITH THE ASME BOILER AND PRESSURE VESSEL CODE, SECTION III, DIVISION 2, "CODE FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS" (ACI-353)

PCRV MAXIMUM CAVITY FRESSURE (MCP) = 7.79 MPa (1133 PSIG)

LEGEND:

- TENDON BEARING PLATE
- REFUELING PENETRATION
- HIGH TEMPERATURE ADSORBER PENETRATION
- HIGH TEMPERATURE FILTER PENETRATION

SECT. A-A

Figure 3.2-2 PCRV general arrangement

Figure 3.3-1 Single Pressure Relief Train, Schematic

Figure 3.3-2 PCRV Pressure Relief System Setpoints

ZONE NO.	CLASS	NAME	ZONE NO.	CLASS	NAME
1	A	STEAM GENERATOR CAVITY-UPPER	11	B	STEAM GENERATOR CAVITY - LOWER
2	A	MAIN CIRCULATOR PLUG - OUTER	12	A	UPPER AUXILIARY CROSS DUCT
3	A	UPPER MAIN CROSS DUCT	13	A	AUXILIARY HEAT EXCHANGER CAVITY
4	A	TOP HEAD - CORE CAVITY	14	B	LOWER AUXILIARY CROSS DUCT (HORIZONTAL)
5	A	UPPER SIDEWALL	15	B	LOWER AUXILIARY DUCT (VERTICAL)
6	A,B	CENTER SIDEWALL			
7	B	LOWER SIDEWALL			
8		BOTTOM HEAD - CORE CAVITY			
9	B	LOWER MAIN CROSS DUCT			
10	A,B	STEAM GENERATOR LOWER PENETRATION			

Figure 3.3-3 Thermal Barrier System Zone Description

Figure 3.3-4 Complete Control Rod and Drive Mechanism Assembly

Figure 3.3-5 Fuel Element Handling Cycle

Figure 3.3-6 Layout Reactor Internals 1170 MWt HTGR Steamer

Figure 3.3-7 Steam Generator Module Diagram

Figure 3.3-8 HTGR Main Circulator Installation, Electric Drive

Figure 3.3-9 Basic Mechanical Arrangement of the Bayonet-Tube CAHE

SECTION 4.0

BALANCE OF PLANT DESCRIPTION

4.1 INTRODUCTION

This section describes the Steamer BOP design which was used as the basis for the cost estimate presented in Subsection 5.1. The descriptions which follow correspond to the two-and three-digit level account structure of the cost estimate. Summary descriptions at the two digit level are presented in Subsection 4.2, and detailed descriptions at the three digit level are presented in Subsection 4.3.

A listing of the key plant parameters is given in Table 1.4-1.

Drawings supporting the balance of plant design are included in Subsection 4.4

4.2 TWO DIGIT SUMMARY DESCRIPTION

DIRECT COST ACCOUNTS

ACCOUNT 21 - STRUCTURES AND IMPROVEMENTS

The structures for nuclear power plants are normally divided into two categories: Seismic Category I structures and non-Seismic Category I structures. The Seismic Category I structures house all safety related equipment essential for safe plant operation, shutdown and control. These structures generally consist of foundation mats, exterior walls, interior walls, floor slabs and roof slabs, all of reinforced concrete. The floor and roof slabs are normally supported on heavy structural steel framing.

The non-Seismic Category I structures house equipment and components not required for plant safety or safe shutdown. These structures generally consist of structural steel framing, metal siding, and a roof comprised of either concrete plank or metal decking with insulation and builtup roofing, and are founded on reinforced concrete spread footings. When required by proximity, these structures are designed to assure that their failure will not cause a loss of function of a Seismic Category I structure.

The gross building volumes given in the structure descriptions are based on the overall outside dimensions of the building.

ACCOUNT 22 - REACTOR PLANT EQUIPMENT

The design of the reactor plant is based on the General Atomic Company two-loop 1170 MWt HTGR Steamer NSSS design discussed in Section 3.0. The NSSS scope includes the reactor internal components, reactor core, neutron and region flow control system, fuel handling system, PCRV service system, primary coolant system, control rod drive storage wells, special shipping equipment, main and auxiliary circulator service systems, helium purification system, core auxiliary cooling system, reactor plant protection system, overall plant control system, and plant data acquisition and processing (DAP) system.

The reactor balance-of-plant (BOP) systems include the liquid, gaseous and solid waste management systems, PCRV erection, core auxiliary cooling water system, fuel storage and viewing equipment, fuel pool cooling system, fuel handling auxiliary system, helium storage and transfer system, gaseous and liquid nitrogen systems, nuclear service water system, reactor plant cooling water system, diesel cooling water system, NSS process cooling water system, decontamination system, and associated instrumentation and controls for these systems.

ACCOUNT 23 - TURBINE PLANT EQUIPMENT

The turbine plant functions to provide 2,715,500 lbs/hr of process steam at 650 psi and 750 F. At the same time the turbine plant produces 144.5 MW

of electricity. The turbine plant also functions to deliver feedwater at proper conditions irrespective of process steam demand or electrical load generation. The turbine plant equipment includes the turbine generators, auxiliary condensate system, feedwater heating system, main steam supply and bypass system, heat dump system, turbine plant closed cooling water system, demineralized water system, chemical treatment system and all the associated instrumentation and controls for these systems and equipment.

ACCOUNT 24 - ELECTRIC PLANT EQUIPMENT

The electric plant equipment transfers the power generated in the plant to the high voltage switchyard through the generator stepup (GSU) transformers; controls and meters the electric energy and protects the power carrying components. It is the source of power for the plant auxiliaries, plant control, protection and surveillance systems and the engineered safety features equipment during normal operation, abnormal accident conditions, and during plant shutdown and refueling.

There are three station auxiliary transformers (SATs) feeding into two 13.8 kV and two 4.16 kV non Class 1E buses and three 4.16 kV Class 1E buses, and two reserve auxiliary transformers (RATs). The electric plant design reflects all the applicable regulatory and technical requirements. Physical and electrical separation of equipment and systems is provided to assure the availability of the minimum required safety featured equipment to mitigate the consequence of any design basis event. Physical separation of equipment and circuits is achieved in such a way that the single failure criterion is met. Drawing No. 6712-M-310100 shows basic design features for the electric plant equipment.

ACCOUNT 25 - MISCELLANEOUS PLANT EQUIPMENT

The miscellaneous plant equipment provides water, compressed air, auxiliary steam and general maintenance and service equipment for the overall plant. It includes crane systems, compressed air system, service water system, fire protection system, potable water system, auxiliary steam system, communications system, fire detection system, security system, laboratory equipment, office furnishings, and environmental monitoring equipment.

ACCOUNT 26 - WASTE HEAT REJECTION SYSTEM

The waste heat rejection system provides cooling for the service water system during normal operation and for the condenser during startup, partial load and upset operation. It includes the structures and equipment for the circulating water and service water systems.

INDIRECT COST ACCOUNTS

ACCOUNT 91 - CONSTRUCTION SERVICES

Includes the costs associated with temporary construction facilities, construction tools and equipment and payroll insurance and taxes.

ACCOUNT 92 - HOME OFFICE ENGINEERING AND SERVICES

Includes costs associated with home office services, quality assurance and construction management.

ACCOUNT 93 - FIELD OFFICE ENGINEERING AND SERVICE

Includes costs associated with field office expenses, job supervision and quality assurance.

4.3 THREE DIGIT DETAILED DESCRIPTION

DIRECT COST ACCOUNTS

ACCOUNT 21 - STRUCTURES AND IMPROVEMENTS

ACCOUNT 211 - Yardwork

The yardwork account includes site clearing, excavation and backfilling, roads, grading and landscaping, parking areas, fencing, a sewage treatment facility, storm sewers, road and yard lighting, settling basins, and a railroad spur for a hypothetical site in the Port Arthur, Texas area.

The account only includes the excavation and backfilling work associated with the main nuclear and turbine plant areas. For all other structures this work is included with each of the individual miscellaneous structures.

It is assumed that highway access is provided to the site boundary by existing roads. Railroad access is assumed to be provided by constructing a five mile spur which connects to an existing railroad line.

ACCOUNT 212 - Reactor Containment Building (RCB)

The reactor containment building houses the PCRV and other NSS components and is designed to protect them against normal, abnormal and environmental conditions and external missiles. It is designed to limit fission product release during normal and accident conditions including the design basis depressurization accident (DBDA) and the postulated maximum hypothetical fission product release (MHFPR).

The RCB is a reinforced concrete, steel-lined structure comprised of a foundation mat, cylindrical shell and hemispherical dome. The gross volume inside the structure above the bottom floor is 2,314,400 cu ft, and the design pressure is 60 psig. The shell is 119 ft-0 in I.D., 170 ft-11 in high and 4 ft-6 in thick, the dome has a 59 ft-6 in inside spherical radius and is 3 ft-6 in thick, and the overall height of the structure above the foundation mat is 233 ft-11 in. The mat is 180 ft-0 in diameter and 14 ft-0 in thick and its bottom is founded 32 ft-10 in below grade. The mat is extended in one area to provide support for the containment penetration building. The RCB is completely lined on the interior with steel plate to provide a pressure-tight boundary. The liner thickness is 1/4 inch on top of the foundation mat, 3/8 inch on the shell and 1/2 inch on the dome. The liner on the foundation mat is protected by a 2 ft-0 in thick concrete fill slab above, which also acts as the RCB floor. The RCB contains two hatches; one a 28 ft-0 in I.D. equipment hatch with personnel lock and the other an 8 ft-0 in I.D. personnel hatch. Inside the RCB, the foundation mat supports the PCRV and its support. The PCRV is 87 ft-0 in diameter and 93 ft-0 in high and is supported on a reinforced concrete ring wall and center core which bears on the RCB foundation mat. The ring wall is 84 ft-0 in O.D., 5 ft-3 in thick and

12 ft-6 in high above the fill slab, and contains three large full-height openings to provide for routing of major piping through the wall. The center core is solid, 22 ft-0 in diameter and 12 ft-6 in high. The PCRV is located concentrically in the RCB, thus providing a 16 ft-0 in annular clearance between it and the RCB shell. Within this annular space and supported on the fill slab is a steel structure extending up to the top of the PCRV for support of major equipment, piping, electrical trays, HVAC equipment and access platforms and stairs.

The RCB is a Seismic Category I structure.

ACCOUNT 213 - Turbine Building (TB)

The turbine building (TB) houses the turbine generators, condensate system, feedwater system, and other associated equipment. The TB is a metal structure, 153 ft-0 in long and 116 ft-0 in wide, supported on reinforced concrete spread footings and framed with structural steel. The turbine hall portion of the building is 68 ft-0 in high, with the ground floor at grade and operating floor 22 ft-0 in above. The heater bay portion of the building is 80 ft-0 in high and has intermediate floors at elevations 22 ft-0 in and 44 ft-0 in. The ground floor is a concrete slab on grade, and upper floors are concrete slabs supported by metal decking on structural steel framing. The roof is metal decking on steel framing, with insulation and builtup roofing. The building walls are covered with insulated metal siding. A 75-ton overhead travelling crane is located in the turbine hall as is the turbine-generator pedestal, a high-tuned reinforced concrete structure supported on a mat foundation. A rail-truck bay is located in one end of the building. A switchgear and battery room is located in a lean-to structure framed to the side of the turbine building. The lean-to is 103 ft-0 in long, 32 ft-0 in wide and 23 ft-6 in high and is similar in construction to the turbine building. The lean-to has only one floor, a concrete slab on grade. The gross volume of the turbine building is 1,239,900 cu ft and the gross volume of the lean-to is 77,500 cu. ft.

The TB is a non-Seismic Category I structure.

ACCOUNT 214 - Security Building (SB)

The security building (SB) provides a controlled means of access to the nuclear island buildings. It contains a waiting room, search area, locker room and guard room. The guard room contains the plant surveillance equipment which is used to monitor the activities within and surrounding the plant.

The SB is a single-story masonry structure, 72 ft-0 in long, 69 ft-0 in wide and 13 ft-6 in high at the eaves, supported on reinforced concrete wall footings. The exterior walls are 12 in thick concrete block and the floor is a reinforced concrete slab on metal decking supported by structural steel framing. Roofing is comprised of insulation and builtup roofing. Interior partition walls are concrete block and the floor is a reinforced

concrete slab on grade. The entire building has a suspended ceiling and resilient tile flooring, except toilet rooms which have ceramic tile floors and wainscot. Security areas inside the building have bullet-resistant doors and windows. The gross volume of the building is 67,100 cu. ft.

The SB is a non-Seismic Category I structure.

ACCOUNT 215 - Reactor Service Building (RSB)

The reactor service building (RSB) houses the facilities, systems and components necessary for fuel handling and inspection, control rod storage, equipment decontamination and inspection, radioactive waste management, and other auxiliary equipment associated with the operation and maintenance of the reactor.

The RSB is a six-story structure that is located at one end of, and is integral with, the fuel storage building. The RSB portion is 96 ft-6 in wide and 81 ft-6 in long. The lower five stories are of reinforced concrete, with the top floor 91 ft-6 in above the top of the foundation mat which is located at grade and serves as the bottom floor. The foundation mat is 6 ft-0 in thick and exterior walls are 3 ft-0 in thick. Interior wall thicknesses vary. The upper floors are concrete slabs, generally supported by metal decking on structural steel framing. The building contains two steel stairways and one passenger/freight elevator which serve all floors and are enclosed by reinforced concrete fire towers. An enclosed reinforced concrete truck dock projects from the building at grade. The dock is 31 ft-0 in wide, 36 ft-6 in long and 25 ft-0 in high, with 2 ft-0 in thick walls and roof. It is tornado missile resistant but is non-Seismic Category I. The top floor of the RSB is enclosed by a 67 ft-0 in high non-Seismic Category I steel framed structure having insulated metal siding and a roof comprised of metal deck, insulation and builtup roofing. A 100 ton overhead travelling crane is located within and supported by the structure. One wall of the enclosure is open to the abutting containment penetration building (CPB) and both the concrete and steel portions of the building are structurally free of the CPB. The gross volume of the RSB is 797,400 cu ft and the gross volume of the metal structure is 526,900 cu. ft.

The RSB is a Seismic Category I structure.

ACCOUNT 216 - Main Circulator Controller Building (MCB)

The MCB is a single-story metal structure, 110 ft-0 in long, 35 ft-0 in wide and 16 ft-0 in high at the eaves, supported on reinforced concrete spread footings. The building is framed with structural steel, and has insulated metal siding and a roof comprised of metal decking, insulation and builtup roofing. The floor is a reinforced concrete slab on grade. The gross volume of the building is 61,600 cu. ft.

The MCB is a non-Seismic Category I structure.

ACCOUNT 217 - Fuel Storage Building (FSB)

The fuel storage building (FSB) houses all equipment related to new and spent fuel shipping, receiving and storage, and is designed to provide storage for 1.3 cores of spent fuel. The building is capable of handling either truck or rail shipping of spent fuel and has the facilities necessary to handle either shipping cask.

The FSB is a reinforced concrete structure that is located at one end of, and is integral with, the reactor service building. The FSB portion is 96 ft-6 in wide, 83 ft - 0 in long and 69 ft-10 in high above the top of the 6 ft-0 in thick foundation mat which is located at grade and serves as the bottom floor. A portion of the mat is depressed to form the bottom of the fuel storage pool which is fully enclosed with 5 ft-0 in thick concrete walls and roof. The walls of the FSB are 3 ft-0 in thick, and the roof is 2 ft-0 in thick with elastomeric membrane roofing. The roof slab is supported by metal decking on structural steel framing. A 5-ton overhead fuel handling crane is supported from the building walls and a 160-ton semi-gantry fuel cask crane is located in the shipping bay at one end of the building. An extension of the shipping bay projects from the building. This extension is of reinforced concrete and is 33 ft-6 in wide, 25 ft-0 in long and 50 ft-6 in high above the mat. Walls are 3 ft-0 in thick, and the roof is 2 ft-0 in thick with elastomeric membrane roofing. A 3 ft-0 in thick reinforced concrete shield slab runs the full length of the shipping bay 14 ft-0 in clear above the railcar tracks. The gross volume of the FSB is 724,900 cu ft.

The FSB is a Seismic Category I structure.

ACCOUNT 218A - Control, Auxiliary and Diesel Generator Building (CADB)

The control, auxiliary and diesel generator building (CADB) houses the control and electrical equipment required for plant operation, which includes the main control room, over and under cable spreading areas, switchgear area, and diesel generators. A remote shutdown area is also provided which is isolated from the main control room and containment.

The CADB is a six-story reinforced concrete structure with a 5 ft-0 in thick mat foundation, 2 ft-0 in thick above-grade exterior walls and roof slab, and intermediate floor slabs supported by metal decking on structural steel framing. The building is 93 ft-0 in wide, 130 ft-0 in long and 123 ft-3 in high above the top of the foundation mat which is located 32 ft-6 in below grade and serves as the bottom floor. The exterior walls below grade are 3 ft-0 in thick. Interior partitions are 1 ft-0 in thick reinforced concrete. Two separate full-height steel stairs are provided on the exterior of the building in reinforced concrete fire towers. One tower is provided with an adjacent passenger elevator. Concrete foundations for the diesel generators are located on the bottom floor, and on the

roof are located package-type cooling towers for building cooling. The gross volume of the CADB is 1,550,600 cu ft.

The CADB is a Seismic Category I structure.

ACCOUNT 218B - Administration and Service Building (ASB)

The administration and service building (ASB) houses the general offices, conference rooms, storage areas, locker and change rooms, toilets, non-radioactive laboratories and machine shop, and spare parts rooms.

The ASB is a two-story metal structure, 160 ft-0 in long, 150 ft-0 in wide and 28 ft-0 in high, supported on reinforced concrete spread footings. The building is framed with structural steel and has insulated metal siding and a roof comprised of metal decking, insulation and builtup roofing. The first floor is a reinforced concrete slab on grade and the second floor is a concrete slab on steel framing. Interior walls are of concrete block, metal partitions or plaster board on studs, and finished rooms have suspended ceilings and resilient tile or carpeted floors. Toilets have ceramic tile floors and wainscot. The gross volume of the building is 672,000 cu ft.

The ASB is a non-Seismic Category I structure.

ACCOUNT 218C - Auxiliary Boiler and Makeup Demineralizer Building (ABDB)

The auxiliary boiler and makeup demineralizer building (ABDB) is a single-story metal structure located along one side of the turbine building. The building is supported on reinforced concrete spread footings, is framed with structural steel, and has insulated metal siding and a roof comprised of metal decking, insulation and builtup roofing. The floor is a reinforced concrete slab on grade.

The ABDB is a non-Seismic Category I structure.

ACCOUNT 218D - Fire Pump House (FPH)

The fire pump house (FPH) contains the electric motor-driven and diesel motor-driven fire pumps and associated equipment. It is a single-story metal structure, 75 ft-0 in long, 30 ft-0 in wide and 12 ft-0 in high at the eaves, supported on reinforced concrete spread footings. The building is framed with structural steel and has insulated metal siding and a roof comprised of metal decking, insulation and builtup roofing. The floor is a reinforced concrete slab on grade. Interior walls are of concrete block. The gross volume of the building is 27,000 cu ft. The foundations for the two 500,000 gal. steel fire protection storage tanks located adjacent to the pump house are reinforced concrete ring walls with sand beds for support of the tank bottoms.

The FPH is a non-Seismic Category I structure.

ACCOUNT 218E - L.P. Helium Storage Area (HSA)

The L. P. helium storage area (HSA) is a 150 ft-0 in by 200 ft-0 in area on grade which provides for the storage of 30 horizontal low pressure helium storage tanks. The tanks are supported on saddle-type reinforced concrete piers with continuous spread footings. The tanks are protected by an open-sided, steel framed canopy structure having corrugated-metal roofing. The ground area below the canopy is surfaced with six inches of deep crushed stone.

The HSA is a non-Seismic Category I structure.

ACCOUNT 218H - Diesel Cooling and Fuel Oil Storage Building (DFB)

The diesel cooling and fuel oil storage building (DFB) houses the diesel generator dry cooling towers and the seven-day fuel oil storage tanks. It is a two-story reinforced concrete structure supported on a 3 ft-0 in thick mat foundation. The structure is 68 ft-0 in long, 52 ft-0 in wide and 72 ft-0 in high above the top of the foundation mat which is located 31 ft-0 in below grade and serves as the bottom floor. The exterior walls below grade are 3 ft-0 in thick. All interior walls and the exterior walls above grade are 2 ft-0 in thick. The second floor is the roof over the fuel oil tanks and is a concrete slab supported by metal decking on structural steel framing. The roof of the building protects the cooling towers and is a concrete slab with portions made in removable sections to permit heat exchanger replacement. The gross volume of the structure is 254,600 cu ft.

The DFB is a Seismic Category I structure.

ACCOUNT 218I - Access Building (AB)

The access building (AB) houses the liquid nitrogen system, helium compressors, locker rooms, radioactive shop, and health physics check point to monitor personnel leaving potentially radioactive areas.

The AB is a two-story metal structure, 130 ft-0 in long, 90 ft-0 in wide and 40 ft-0 in high, supported on reinforced concrete spread footings. The building is framed with structural steel and has insulated metal siding and a roof comprised of metal decking, insulation and builtup roofing. The first floor is a reinforced concrete slab on grade and the second floor is a concrete slab on steel framing. Interior walls are of concrete block, metal partitions or plaster board on studs, and finished rooms have suspended ceilings and resilient tile floors. Toilets have ceramic tile floors and wainscot. The gross volume of the building is 468,000 cu ft.

The AB is a non-Seismic Category I structure.

ACCOUNT 218J - Containment Annulus Building (CAB)

The containment annulus building (CAB) houses the RPCW system, CACW pumps, portions of the liquid waste management and helium purification systems, and penetration areas.

The CAB is an annulus-shaped, two-story reinforced concrete structure that surrounds the RCB and utilizes the projected portion of the RCB base mat as a common foundation. The exterior wall of the structure is cylindrical, 180 ft-0 in O.D., and extends completely around the RCB along the periphery of the foundation mat, except where it butts into the containment penetration building, to form a 26 ft-0 in wide annular structure. The wall extends 48 ft-10 in above the mat up to the roof, and is 3 ft-0 in thick below grade and 2 ft-0 in thick above. The top of the foundation mat serves as the bottom floor of the structure. The roof and intermediate floor are reinforced concrete slabs supported on steel framing, and are structurally free of the RCB shell which forms the interior wall of the structure. The roof slab is 2 ft-0 in thick for missile resistance. The annular space in both stories is partitioned by various radially-oriented concrete walls to provide functional separation of areas. Located on the roof are two identical structures housing the CAB ventilation equipment. These structures are non-Seismic Category I. Each structure is full annulus width, 50 ft-0 in long and 14 ft-0 in high, steel framed, and has insulated metal siding and a roof comprised of metal deck, insulation and built-up roofing. A reinforced concrete enclosure inside each structure provides missile protection for the ventilation duct penetration through the RCB wall. The gross volume of the CAB (above the foundation mat) is 491,000 cu ft.

The CAB is a Seismic Category I structure.

ACCOUNT 218K - Containment Penetration Building (CPB)

The containment penetration building (CPB) houses the electrical penetration areas, instrument areas, helium purification system and liquid waste management system.

The CPB is a five-story structure that abuts the RCB and utilizes the projected and extended portion of the RCB base mat as a common foundation. The structure is 97 ft-0 in long, and 36 ft-0 in wide at its narrowest point widening as it follows the periphery of the RCB wall. The lower four stories are of reinforced concrete, with the top floor 110 ft-4 in above the top of the foundation mat which serves as the bottom floor. The exterior walls are 3 ft-0 in thick and the floors are 4 ft-6 in deep slab and beam construction. The interior partition walls are, in general, 3 ft-0 in thick, with heavier walls used where additional radiation shielding is required. The top floor of the building is enclosed by a 41 ft-0 in high non-Seismic Category I steel framed structure having

insulated metal siding and a roof comprised of metal deck, insulation and builtup roofing. This enclosure is open to a similar structure at the top of the adjacent reactor service building (RSB). Both the concrete and steel portions of the RSB are structurally free of the abutting RCB shell. The gross volume of the concrete structure (above the foundation mat) is 466,700 cu ft and the gross volume of the metal enclosure is 173,400 cu ft.

The CPB is a Seismic Category I structure.

ACCOUNT 218S - Holding Pond (HP)

The holding pond (HP) collects effluent from non-radioactive plant drains. It is an open top, reinforced concrete basin, with the top located 1 ft-0 in above grade level. The basin is 80 ft-0 in square and 8 ft-0 in deep and is divided in half by a concrete partition wall to create two compartments. An 8 ft-0 in wide, 10 ft-0 in long and 10 ft-0 in high, steel framed, metal sided and roofed control house is located on grade at one end of the basin.

The HP is a non-Seismic Category I structure.

ACCOUNT 218T - Ultimate Heat Sink Structures (UHSS)

The core auxiliary cooling water/nuclear service water (CACW/NSW) ultimate heat sink structure (UHSS) houses two of the three train CACW system dry towers, and the two trains of the Nuclear Service Water (NSW) system which includes wet towers, pumps and associated electrical equipment. It is a two-compartment reinforced concrete structure on a 4 ft-0 in thick mat foundation. Each compartment is 62 ft-0 in wide and 56 ft-0 in long and is comprised of 2 ft-0 in thick exterior and interior walls, and floor and roof slabs. The floors and roof of the portion of the structure housing the CACW equipment are supported on steel framing. A wet basin is located below each compartment to provide 15 days of water storage, 30 days total. A wet cooling tower for each compartment is located at one end of the structure in a 22 ft-0 in long by 24 ft-0 in wide reinforced concrete chamber. The gross volume of the structure is 424,800 cu ft.

An underground two-compartment rectangular reinforced concrete piping tunnel connects the CACW/NSW area in the containment annulus building with this ultimate heat sink structure. The tunnel is comprised of two exterior walls and a top and bottom slab. A vertical concrete wall separates the CACW and NSW functions. Each compartment is 10 ft-0 in wide and 8 ft-0 in high inside.

The CACW ultimate heat sink structure houses the third train CACW system dry tower. It is a reinforced concrete structure on 4 ft-0 in thick mat foundation. The structure is 62 ft-0 in wide and 25 ft-0 in long and is comprised of 2 ft-0 in thick exterior and interior walls and roof slab. Gross volume of the structure is 57,400 cu ft.

An underground single-compartment rectangular reinforced concrete piping tunnel connects the CACW area in the containment penetration building with this ultimate heat sink structure. The tunnel is comprised of two exterior walls and a top and bottom slab and is 8 ft-0 in wide and 8 ft-0 in high inside.

The UHSS are Seismic Category I structures.

ACCOUNT 218V - Control Room Emergency Air Intake Structure (EAS)

Two separate intake structures are provided to assure 30-day control room habitability. Each EAS is a reinforced concrete structure, 12 ft-0 in long and 12 ft-0 in wide, supported on a 2 ft-0 in thick foundation mat with 2 ft-0 in thick exterior walls and roof slab. The gross volume of each structure is 2160 cu ft..

The EAS are Seismic Category I structures.

ACCOUNT 22 - REACTOR PLANT EQUIPMENT

ACCOUNT 221 - Reactor Equipment

The reactor equipment includes the PCRV and its associated supports and systems including the PCRV seal and PCRV pressure relief systems. The PCRV internals include the core, core support and reflector elements, thermal barriers, plenum elements, and the steam generator and core auxiliary heat exchanger (CAHE) cavity liners and ducts.

The reactor is a high temperature gas cooled reactor employing a U-235/thorium fuel cycle, helium cooling, and graphite moderation. The reactor core consists of vertical columns of hexagonal graphite fuel and reflector elements arranged to approximate a right circular cylinder. Core heat is removed by the downward flow of the helium coolant through the core and reflector elements.

The core is located and supported within the prestressed concrete reactor vessel (PCRVR) by three structures: the core support structure, the permanent side reflector and borated shield, and the core lateral restraint.

Reactor control is provided by the control rods which are operated by the control rod drives. The drives are located in PCRVR penetrations above the center column of the refueling region.

ACCOUNT 222 - Main Heat Transport System

The reactor coolant system encompasses the reactor coolant and the primary cooling loops. The ducting and structures involved in directing the coolant flow through the reactor and cooling loops are included in Account 221. Each of the two 50 percent primary cooling loops includes a steam generator, a main helium circulator, and a primary coolant shutoff valve.

The reactor coolant system uses helium as the reactor coolant since it is chemically inert, is stable, has excellent heat transfer characteristics, and does not undergo a phase change. The helium flow is forced downwards through the reactor core by means of the two main helium circulators which are driven by co-axial synchronous electric motors. The helium flows through the core, steam generator, and helium circulator and then back to the core to complete the circuit. The steam generators and helium circulators are situated in cavities within the PCRV.

ACCOUNT 223 - Safeguards Cooling System

The safeguards cooling system (SCS) consists of the core auxiliary cooling system (CACS), which is part of the reactor coolant system, and the core auxiliary cooling water system (CACW).

The CACS system is a Safety Class 2 system which includes the auxiliary circulators and their drive motors, motor controls, diffusers and valves, the core auxiliary heat exchangers (CAHE), and control instrumentation and hardware.

The CACS system provides an independent means of cooling the reactor core with the primary system pressurized or depressurized. It consists of three cooling loops, each capable of removing 100 percent of the core residual and decay heat for safe cooldown from 102 percent of reactor steady state power level under pressurized condition. Two loops are required in the depressurized condition, each with 50 percent capability.

The CACW system is a Safety Class 3 system which circulates cooling water through the CAHE to remove stored and decay heat from the primary coolant and reject this heat to the atmosphere. The system consists of three independent cooling water loops, each capable of providing 100 percent heat removal requirements for one of the three CAHEs. Each loop consists of two 100 percent auxiliary pumps, one 100 percent main pump, one 100 percent air blast heat exchanger and associated equipment.

The main and auxiliary circulator service systems, which are also included in this account, provide buffer helium for the circulator seals. The main circulator service system also provides a supply of high-pressure water for bearing lubrication and cooling.

ACCOUNT 224 - Radioactive Waste Processing Systems

The radioactive waste process system consists of the liquid, gaseous and solid waste management systems. The liquid waste management system is NWS and includes sumps and tanks for collection of liquid effluent and utilizes filtration, demineralization, evaporation and reverse osmosis singularly or in combination for processing.

The gaseous waste management system is NNS and has the capability to selectively release or retain gaseous effluent for a suitable period prior to a controlled release.

The solid waste management system is NNS and capable of low level compacted waste drum storage and complete remote handling of high level solidified waste.

ACCOUNT 225 - Nuclear Fuel Handling and Storage

The fuel handling and storage facility utilizes the fast refueling equipment supplied by GAC. A Seismic Category I storage pool provides water cooled storage for 1.3 cores. Fuel pool cooling is provided by two Safety Class 3 cooling-trains, each equipped with one 100 percent pump and one 100 percent heat exchanger. The fuel handling auxiliary system is a single NNS system which provides purge and evacuation services during refueling.

ACCOUNT 226 - Other Reactor Plant Equipment

The helium storage system is a NNS system which provides storage capacity for the entire primary coolant inventory and one year makeup requirements. The helium storage system also provides PCRV depressurization and pressurization capabilities and maintains a sub-atmospheric PCRV pressure during refueling.

The helium purification system is a Safety Class 3 system which consists of two 100 percent redundant loops that purify helium from the primary coolant system, removing fission products and chemical impurities, and return the purified helium to the primary loop.

The liquid nitrogen system is a NNS system which supplies refrigeration to the helium purification system low temperature adsorbers. The system consists of two 100 percent liquid nitrogen recondensers for operating reliability, one phase separator storage tank and two 100 percent transfer pumps.

The nuclear service water system is a Safety Class 3 system which provides cooling for the reactor plant cooling water system, fuel pool cooling system and other reactor plant auxiliaries. The system consists of two 100 percent redundant and independent loops containing one circulating water pump and one cooling tower. The system is designed to function for 30 days following a postulated accident without requiring makeup water to the tower basins.

The reactor plant cooling water system is a Safety Class 3 system which provides cooling water to the PCRV cooling coils and other reactor plant auxiliary equipment. The system consists of two 100 percent redundant, independent, closed loops which contain two 100 percent pumps and one 100 percent heat exchanger. The NNS equipment cooled by the system can be isolated during abnormal plant conditions.

ACCOUNT 227 - Reactor Plant Instrumentation and Controls

The reactor plant instrumentation and control system is designed to ensure that the unit can be safely and efficiently operated. And, in the event of an abnormal or accident condition, it can be shut down and maintained in a safe-shut-down condition. The system consists of automatic and manually initiated protection systems for safety under accident conditions, safety related display systems required during normal, upset, emergency, and faulted conditions, computer based data acquisition and display system, and regulating systems used for normal operation of the unit.

The instruments and controls are located in the main control room which provides remote operation of the unit. In the event that access to the main control room is lost, equipment is provided outside the main control room to shut the reactor down and maintain it in a safe-shut-down-condition.

ACCOUNT 23 - TURBINE PLANT EQUIPMENT

ACCOUNT 231 - Turbine Generator

The turbine generator unit is composed of two completely independent turbine generator sets. The high pressure turbine is designed to deliver 90,700 kW with throttle steam conditions of 2415 psia at 1000 F. The high pressure turbine exhausts 650 psia steam at 750 F to both process and the low pressure turbine. The low pressure turbine is designed to deliver 53,800 kW with throttle steam conditions of 650 psia at 750 F, while supplying extraction steam to three stages of feedwater heating and exhausting to the fourth stage (Number 1) feedwater heater. The generators have a rating of 101 MVA and 60 MVA with 0.90PF, 25,000V, 3 phase 60 Hz output. Each generator is totally enclosed hydrogen cooled.

The turbine generator units function to produce electricity as a byproduct of the process system. The plant is designed to supply process steam whether or not the turbine generator units are on the line.

Figure 4.4-3 shows a heat balance for the cogeneration plant. It should be noted that the process steam pressure/temperature requirements, which are the High Pressure (HP) turbine exhaust conditions, impose a low efficiency blade path design on the HP turbine. Alternate methods of achieving the required process steam conditions include (1) exhaust from HP turbine at design process steam enthalpy but higher pressure and throttle the steam as necessary (2) exhaust from HP turbine at design process steam pressure but lower temperature and mix with appropriate amount of main steam. These alternates are not expected to have a significant impact on total plant output and capital costs but the exact approach is dependent on the requirements of the particular user.

ACCOUNT 233 - Condensing System

The condenser is a single stage, single pass type which condenses at 8 psia. The condenser is designed to handle the heat rejection load for the plant (approximately 25 percent of reactor thermal rating) on startup until the steam quality is satisfactory for process and turbine operation, or for other reactor "hot standby" functions. As such, it takes suction from the main steam line through the flash tank and is used for startup, shutdown and maintaining the NSSS at minimum load during periods of very low demands by the turbines and/or the process system.

The condensate from the condenser is further cooled in a separate heat exchanger before being returned to the condensate storage.

Two mechanical vacuum pumps are supplied for removing non-condensable gases from the lowest stage (8 psia) feedwater heater. The vacuum pumps are motor driven rotary two stage units.

The condensate transfer pumps take suction from the condenser (when in operation) and discharge condensate to the condensate storage tank. The process return is to the condensate tank. The condensate tank serves as the process and power condensate collection point. Three 50 percent capacity condensate pumps taking suction from the condensate storage tank, will pump the condensate through a full flow condensate polishing system. The system is designed to operate at 200 psig and 130 F.

ACCOUNT 234 - Feed Heating System

Feedwater Heaters

Four stages of feedwater heating is utilized to heat the feedwater being returned to the steam generators. Heaters are placed in series and operate under increased pressure of various stages of extraction steam from the low pressure turbine. The Numbers 1, 2, and 4 heaters are of the closed type, horizontal, U-tube arrangement. The Number 3 heater is of the direct contact, deaerating type. The Number 2 heater has an integral drain cooler and its drains cascade to the Number 1 heater which drains to a drains receiver tank. The drains receiver tank provides a reservoir for the feedwater heater drains which are pumped forward into the condensate line. The Number 4 heater has an integral drain cooler and its drains cascade to the Number 3 deaerating feedwater heater. The deaerator storage tank, which is part of the deaerating feedwater heater, is kept at a temperature that does not differ by more than 70°F from the final feedwater temperature. The boiler feed booster pump takes suction from the deaerator storage tank. Two parallel trains of closed feedwater heaters, each sized for 50 percent capacity, are utilized for the first, second, and fourth stages of heating. To allow for maintenance bypass, valves are provided around the feedwater heaters.

Boiler Feed Pump and Boiler Feed Booster Feed Pump

Two 60 percent capacity motor driven boiler feed pumps of the centrifugal type are provided. Each boiler feed pump is driven by a 9000 HP electric motor.

Three 50 percent capacity motor driven boiler feed booster pumps are provided. These pumps are also of the centrifugal type.

ACCOUNT 235 - Other Turbine Plant Equipment

Turbine Building Closed Cooling Water System

A closed cooling water system, with three 50 percent capacity motor driven water pumps, head tank and two 50 percent capacity heat exchangers, provides cooling to various turbine plant components. This system dissipates heat to the main cooling tower.

Demineralized Water Makeup System

A demineralized water makeup system is provided to supply the required high purity water for makeup to miscellaneous cooling loops and laboratory processes and for initial plant cleaning and flushing operations.

Chemical Treatment System

An all volatile chemical treatment system is used to maintain feedwater chemistry. System components include two hydrazine feed pumps, two ammonia feed pumps and one hydrazine and one ammonia storage tank. The hydrazine chemically removes the dissolved oxygen from the feedwater and the ammonia controls the pH.

Main Steam and Process Piping and Control

Includes all main steam piping, all process steam piping and control valves, all desuperheating and pressure reducing stations as shown on the cogeneration flow diagram (Figure 4.4-2).

ACCOUNT 236 - Turbine Plant Instrumentation and Control

The turbine plant instrumentation and control system is designed to ensure that the turbine and its related systems can be safely and efficiently operated during normal, upset, emergency and faulted conditions. The instruments and controls are located in the main control room and function in conjunction with the reactor plant controls.

ACCOUNT 24 - ELECTRIC PLANT EQUIPMENT

ACCOUNT 241 - Switchgear

A generator load break switch is provided to facilitate rapid disconnection of the generator from the offsite power system allowing the auxiliary power system to be fed through Generator Step-up (GSU) and Station Auxiliary Transformers (SATs). There are two 15 kV and two 5 kV non-Class 1E metal-clad switchgear buses and three 5 kV Class 1E switchgear buses to provide the power sources for the plant auxiliary loads. In the event of a loss of total off-site power supplies, all engineered safety features equipment are automatically sequenced into the Class 1E buses being fed from the diesel generators.

Non-Class 1E and Class 1E 460 volt motor control centers are provided for power distribution to motors up to 100 hp, lighting loads and other miscellaneous loads such as motor-operated valves, resistance heaters, heat tracing and space heaters.

ACCOUNT 242 - Station Service Equipment

Three SAT's and two Reserve Auxiliary Transformers (RATs) are provided to furnish power to the plant auxiliary power system. Each transformer is sized to carry, with sufficient margin, the plant auxiliary loads under heavily loaded conditions. Transformer impedances are selected to limit the available short circuit currents on the switchgear buses without adversely affecting the acceptable voltage regulations during extreme plant operating conditions. Appropriate protections are provided for the transformers.

Unit substations are provided to furnish power sources to the low voltage (460V) Class 1E and non-Class 1E distribution systems. Motors rated 101 hp through 250 hp are connected to the unit substations. The unit substations for the cooling towers are fed from a loop feeder.

The d-c system comprises the plant non-Class 1E and Class 1E batteries and battery chargers. Each Class 1E d-c bus is supplied from a Class 1E battery and a Class 1E battery charger. During normal operation, d-c power is supplied from the battery chargers. During emergency operation, d-c power is supplied from the batteries in the absence of any a-c source to the battery chargers. During startup and shutdown, d-c power is supplied from whichever source is available. Non-Class 1E 125/250 V DC buses are fed from two Non-Class 1E batteries and two Non-Class 1E chargers.

Three independent diesel generators are provided to furnish the onsite a-c power source to the Class 1E 4.16 kV buses. Diesel generators are properly sized such that any two units have the capability of operating all protection systems and the engineered safety features to mitigate the consequence of a design basis depressurization accident (DBDA) concurrent with a loss of offsite power.

Class 1E and non-Class 1E solid state inverters are utilized to provide an uninterruptible power sources for miscellaneous vital and non-vital a-c plant loads.

ACCOUNT 243 - Switchboards

Appropriate number of Class 1E and non-Class 1E a-c power distribution panels are provided to distribute a-c power from the inverters to the 120 volt uninterruptible loads.

Class 1E and non-Class 1E d-c power distribution switchgear line-ups are provided to distribute d-c power from the batteries and their associated chargers.

A control benchboard is provided in the main control board lineup for control and data acquisition of the main generator and the auxiliary electric power system.

Relay panel(s) are provided for protection and metering of the main generator, the generator stepup transformers, and the station and reserve auxiliary transformers. Appropriate protective schemes and devices are utilized to protect the generator, generator stepup and other auxiliary transformers.

ACCOUNT 244 - Protective Equipment

The station grounding system provides the means for maintaining an effective ground at equipment and metal structures, protecting them from galvanic corrosion and personnel from dangerous potentials. Lightning protection schemes are provided for the containment structure.

ACCOUNT 245 - Electrical Structures and Wiring Containers

The equipment provides mechanical protection and physical separation of wire and cable routed between equipment and buildings. The bulk of the raceways consists of cable trays of various types. Raceways are routed in accordance with the same criteria as for cable routing. Fire stops are placed in cable trays wherever they penetrate floors or firewalls, and in other areas where their installations reduce the hazard of fire propagation.

ACCOUNT 246 - Power, Control and Instrumentation Wiring

The plant wire and cable system consists of single/three conductor and triplexed single conductor power cable, multi-conductor control cable, coaxial, triaxial, shielded twisted pair, multi-shielded twisted pair and shielded quad-instrument wire and containment electrical penetrations. Ethylene-propylene rubber (EPR) insulation with chloro-sulfonated

polyethylene (CSPE) based jacket is selected to provide optimum system performance at the area of physical stability, tensile strength, flexibility, aging characteristics, resistance to abrasion, ozone (where required), water/oil absorption, heat distortion, solvent extraction, irradiation, self-extinguishing and non-propagating fire characteristics and resistance to corona effects (where required). Wires and cables are assigned to a load group irrespective of their safety classification in order to avoid the hazard of non-safety related cables being inadvertently routed between two redundant load groups. All wires and cables are qualified as Class 1E irrespective of their application, to eliminate the possibility of inadvertent routing of non-Class 1E cables in Class 1E systems and to minimize the storage/inventory problems at the construction site. In addition to separation by load groups, wires and cables are also separated by energy levels to reduce heating and arcing fault problems.

ACCOUNT 25 - MISCELLANEOUS PLANT EQUIPMENT

ACCOUNT 251 - Transportation and Lifting Equipment

Cranes and Hoists

Two major cranes are provided within the plant. Servicing the reactor service building is an overhead crane with an auxiliary hoist. Within the turbine building is an overhead traveling bridge crane. The bridge span covers the main operating floor area. The containment building crane is provided by General Atomic Co.

ACCOUNT 252 - Air Water and Steam Service System

Compressed Air System

The compressed air system supplies service and instrument air for the entire plant. It consists of two 100 percent oil free compressors, two air receivers and two air dryers.

Service Water System

The service water system supplies cooling water from the cooling tower basin to the turbine plant closed cooling water system and the nuclear service water system during normal plant operation. The system has two vertical, 100 percent capacity wet pit pumps.

Fire Protection System

The fire protection system is designed to minimize the probability and effect of the occurrence of an in-plant fire. The system has four fire pumps, two diesel driven and two motor driven, two 100 percent capacity water storage tanks, and one jockey pump.

Potable Water System

Potable water is required for drinking, sanitary, and washing purposes at the plant. The water is supplied by the local water supply system.

Auxiliary Steam System

An auxiliary steam system is provided, consisting of one 100 percent oil fired boiler and appurtenances. The system is used to provide steam for feedwater preheating and deaerating during plant startup.

ACCOUNT 253 - Communications System

The communications system consists of an intercommunication and paging system, a telephone system, and a sound-powered telephone system. These systems are designed to provide communication between various parts of the plant for all conditions of operation.

ACCOUNT 254 - Furnishings and Fixtures

Instrument shop apparatus are provided for testing, calibration, repairing and routine maintenance of the plant instrumentation and control devices.

Monitoring equipment is provided for the off site radiological, meteorological, water quality, thermal effluent, and seismic monitoring systems.

ACCOUNT 26 - WASTE HEAT REJECTION SYSTEM

ACCOUNT 261 - Structures

ACCOUNT 261.1 - Makeup Water Intake and Discharge Structure

ACCOUNT 261.11 - Makeup Water Intake Structure (MWIS)

The makeup water intake structure (MWIS) is a waterfront structure consisting of a below-grade intake basin comprised of a reinforced concrete mat and walls, and an adjacent battery room, switchgear room and pump room at grade supported on reinforced concrete spread footings. These rooms are single-story and have concrete block exterior and interior walls, and a roof comprised of metal decking, insulation and builtup roofing supported on structural steel framing. Interior floors are reinforced concrete slabs on grade.

ACCOUNT 261.12 - Discharge Structure (DS)

The discharge structure (DS) is a concrete pipe running underground from the circulating water pump house into the waterway. The pipe is supported on steel piles in the waterway, and wrapping is placed around the pipe end to prevent scouring. Wood marker piles are installed in the waterway to mark the pipe location.

ACCOUNT 261.2 - Circulating Water Pump House (CWPH)

The circulating water pump house (CWPH) contains the circulating water pumps, service water pumps and associated equipment. It is a single-story reinforced concrete and metal structure. The concrete portion of the structure is below grade.

ACCOUNT 261.4 - Chlorination Building (CB)

The chlorination building (CB) houses the chemical equipment for the circulating water system. It is a single story masonry building, supported on reinforced concrete wall footings. The walls are concrete block and the roof is comprised of metal decking, insulation and builtup roofing supported on structural steel framing. The floor is a reinforced concrete slab on grade.

ACCOUNT 261.5 - Cooling Tower Switchgear Building (CTSB)

The cooling tower switchgear building (CTSB) is a single-story metal structure located at one end of the circulating water pump house. The building is supported on reinforced concrete spread footings, and is framed with structural steel, with insulated metal siding and a roof comprised of metal decking, insulation and builtup roofing. The floor is a reinforced concrete slab on grade.

ACCOUNT 262 - Mechanical Equipment

Circulating Water Pumps

There are four 25 percent capacity circulating water pumps provided. The pumps are of the vertical mixed flow type which deliver water to the condenser when it is utilized during startup and other load shedding operations. The cooling water for the turbine building closed cooling water heat exchanger is supplied from the circulating water pump basin.

Cooling Tower

A mechanical draft, wet cooling tower is provided. The tower is sized to provide the cooling requirements during the 25 percent "hot standby" operation and also to reject normal service water heat.

Cooling Tower Makeup and Blowdown Systems

Two 100 percent mixed flow vertical pumps are provided for the makeup system. The pumps are located in the intake structure adjacent to the river. Two traveling screens are provided, each suitable for 50 percent of the flow requirements with an approach velocity of 1/2 ft per second. Servicing the traveling screens are two 100 percent capacity screen wash pumps. The screens are protected by a bar rack and trash rake.

INDIRECT COST ACCOUNTS

ACCOUNT 91 - CONSTRUCTION SERVICES

ACCOUNT 911 - Temporary Construction Facilities

This account includes the cost of all temporary structures, janitorial services and maintenance of temporary facilities, guards and security, roads, parking lots, laydown areas, etc., and temporary electrical and piping, temporary heat, air, steam and water systems, general cleanup, etc.

ACCOUNT 912 - Construction Tools and Equipment

This account includes the cost of rental and/or purchase of construction equipment, small tools, consumables (fuel, lubricants, etc.) and maintenance of construction equipment.

ACCOUNT 913 - Payroll Insurance and Taxes

This account includes insurance and taxes related to craft labor such as social security taxes and state unemployment taxes. Workmen's compensation insurance and Public Liability and Property Damage Insurance are also included.

ACCOUNT 914 - Permits, Insurance and Local Taxes

This account includes an allowance based upon in-house experience for the cost of Builders All-Risk Insurance for the project during the construction phase.

ACCOUNT 92 - HOME OFFICE ENGINEERING AND SERVICES

ACCOUNT 921 - Home Office Services

This account includes services associated with home office engineering and design, procurement and expediting activities, estimating and cost control, engineering planning and scheduling, home office reproduction services as well as expenses associated with performance of the above functions (i.e., telephone, postage, computer use, travel, etc.). These costs include salaries of personnel, direct payroll-related costs (DPC), overhead loading, expenses and fee for these services consistent with contractual terms.

ACCOUNT 922 - Home Office Quality Assurance

This account includes services of home office quality assurance engineers and staff personnel engaged in work on the project. Services include reviews, audits, vendor surveillance, etc. as required for design and

construction of the nuclear safety-related portion of the facility. Costs included are salaries, DPC, overhead loading and expenses (i.e., travel) of these individuals. Manhours required for these services and their costs are based upon UE&C experience in this area.

ACCOUNT 923 - Home Office Construction Management

This account includes the services of the construction manager and his assistants. Services of construction planning and scheduling, construction methods, labor relations, safety and security personnel are utilized as required. Costs include salaries, DPC, overhead loading, and expenses.

ACCOUNT 93 - FIELD OFFICE ENGINEERING AND SERVICE

ACCOUNT 931 - Field Office Expenses

This account includes costs associated with purchase and/or rental of furniture and equipment (including reproduction), communication charges, postage, stationery, other office supplies, first aid and medical expenses.

ACCOUNT 932 - Field Job Supervision

This management function includes the resident-construction superintendent and his assistant, craft labor supervisors, field accounting, payroll and administrative personnel, field construction schedules, field purchasing personnel, warehousemen, survey parties, stenographers and clerical personnel. Costs include salaries, DPC, overhead loading, relocation costs of key personnel, and fee. The estimates assume that size of supervisory forces is a function of total direct employed craft labor.

ACCOUNT 933 - Field Quality Assurance

This account includes services of personnel located at the job site engaged in equipment inspection, required documentation of nuclear safety-related equipment and inspection of construction activities. Costs include salaries, DPC, overhead loading, and expenses.

ACCOUNT 934 - Test and Startup Engineering

This account includes services associated with preparation of startup and plant operation manuals and test procedures, direction and supervision of all testing of equipment and systems as the plant nears completion and direction of startup of the facility. Costs include salaries, DPC, overhead loading, and miscellaneous related expenses. Costs of any craft labor required for startup and testing activities are included in the appropriate Direct Cost line items.

4.4 REFERENCE FIGURES

This section contains the figures for the 1170 MWt HTGR-SC steamer balance-of-plant described in Section 4.0. Included are a plot plan, general arrangement drawings, a flow diagram and a one line diagram.

FIGURE 4.4-1

United Engineers & Constructors

SCALE: 1" = 50' 6712-F-11000

LEGEND
 FEED WATER —————
 HP STEAM - - - - -
 LP STEAM
 HTR DRAINS - - - - -

DEC. 1978
 1170 MW(t) HT&R-SC STEAMER STUDY
FIGURE 4.4-2
 CO-GENERATION
 FLOW DIAGRAM
 united engineers
 SCALE - NONE 6712-F-210001

LEGEND
 h - ENTHALPY - BTU/LB.
 # - FLOW - LBS/HR.
 P - PRESSURE - PSIA
 F - TEMP. - °F

PERFORMANCE :
 GROSS OUTPUT 144,500 KW
 AUXILIARY POWER 37,000 KW
 NET POWER TO TRANSFORMER 107,500 KW
 PROCESS HEAT LOAD 1,046,000 KW_t

JULY 1979
 1170 MW_t HTGR-SC STEAMER STUDY

FIGURE 4.4-3

HEAT BALANCE DIAGRAM

united engineers & consultants inc.

SCALE - NONE 6712 - F - 210002

DEC. 1978
1170 MW (L) HTGR-SC STEAMER STUDY
FIGURE 4.4-4
GENERAL ARRANGEMENT
TURBINE BUILDING
PLANS
united engineers & constructors inc.
SCALE 1/8" = 1'-0" 6712-F-210003

SECTION A-A
(F-210003)

DEC. 1978
1170 MW (b) HTGR-SC STEAMER STUDY

FIGURE 4.4-5

GENERAL ARRANGMENT
TURBINE BUILDING
SECTION A-A

United engineers & constructors inc.

SCALE $\frac{1}{8}'' = 1'-0''$

6712-F-210004

EL (+) 1'-0"

EL (+) 26'-0"

DEC. 1978
1170MW(5) HTGR - SC STEAMER STUDY
FIGURE 4.4-8
GENERAL ARRANGEMENT
C.A.D. BLDG
PLAN @ ELEVATIONS
G) 1'-0" & (+) 26'-0"
 UNITED ENGINEERS & CONSTRUCTORS
SCALE: 1/8"=1'-0" 6712-F-310202

4-35

DEC. 1978 1170MW(3) HTGR-SC STEAMER STUDY	
FIGURE 4.4-9	
GENERAL ARRANGEMENT C.A.D. BUILDING PLAN @ ELEVATIONS (+)48'-3" & (+)67'-6"

SCALE: 1/8" = 1'-0"	6712-F-310203

DEC. 1978
1170MW(1) HTGR-SC STEAMER STUDY

FIGURE 4.4-10

GENERAL ARRANGEMENT
C.A.D. BLDG.
PLAN @ ELEVATIONS
(+) 90'-9"

united engineers & constructors inc.

SCALE: 1/8" = 1'-0"

6712-F-310204

SECTION 'A-A'

SECTION 'B-B'

DEC. 1978
1170MW (6) HTGR - SC STEAMER STUDY

FIGURE 4.4-II

GENERAL ARRANGEMENT
C.A.D. BLDG.
SECTIONS

united engineers a consortium inc.

SCALE: 1/8" = 1'-0" **6712-F-310300**

GENERAL NOTES

1. PLANT LAYOUT REQUIREMENTS FOR PCRV ARE BASED ON GENERAL ATOMIC CO. DRAWING NO. 004454, ENTITLED "PCRV GEN. ARCHT. PLAN & SECTION, REV. 1 (PRELIM), DATED 8-24-78, GAC PLANT LAYOUT REQUIREMENTS EQUIP. NO. R-101, DATED 9-15-78, ENTITLED "SYSTEM DESCRIPTION, PCRV, SYSTEM II, 1170 MW (H) STEAMER COGENERATION PLANT." OTHER NUCLEAR STEAM SUPPLIER (NSS) EQUIPMENT IS BASED ON GAC PLANT LAYOUT REQUIREMENTS DOCUMENT NO. 903605 - ISSUE "DRAFT" DATED 7-15-78.

2. NSS EQUIPMENT NUMBERS SHOWN ARE AS PER G.A. PLANT LAYOUT REQUIREMENTS DOCUMENT NO. 903605.

DEC. 1978
1170 MW (H) HTGR-SC STEAMER STUDY

FIGURE 4.4-12

GENERAL ARRANGEMENT
NUCLEAR ISLAND
PLAN & SECTIONS

united engineers consultants inc.

SCALE 1/8"=1'-0" 6712-F-81 0000

8 7 6 5 4 3 2 1

D
C
B
A

SECTION "A-A" 4-40

DEC. 1978
1170 MW (B) HTGR - SC STEAMER STUDY
FIGURE 4.4-14
GENERAL ARRANGEMENT
REACTOR CTMT. BLDG.
SECTION "A-A"
 united engineers CONSULTING INC.
SCALE: 1/8"=1'-0" 6712-F-81 0002

PLAN @ EL. 18'-10"

DEC. 1978
1170 MW (U) HTGR-SC STEAMER STUDY

FIGURE 4.4-15

GENERAL ARRANGEMENT
CTMT ANNULUS BLDG.
LOWER LEVEL PLAN & SECTIONS

United Engineers

SCALE: 1/8"=1'-0" 6712-F-81 0003

FIGURE 4.4-21
GENERAL ARRANGEMENT
RSB FSB & CTMT. PENETR. BLDG.
SECTION "A-A"

e united engineers a construction inc.

SCALE: 1/8" = 1'-0" 6712-F-81 0009

SECTION 5.0

COST ESTIMATES AND PROJECT SCHEDULE

5.1 CAPITAL COST ESTIMATES

The cost summary presented in this section reflects the conceptual 1170 MWt HTGR Steamer Plant design described in Sections 3.0 and 4.0. The summary is presented at both the two and three digit account levels in Tables 5.1-1 and 5.1-2 respectively. A direct craft labor rate summary is presented in Table 5.1.3.

The cost estimate does not include normal contingency costs for equipment, material, and labor components of the total base construction cost. Also, no specific allowance is made for indeterminates associated with the lack of detail information and design/construction experience. Other items not included in the cost estimate are listed in Table 5.1-4.

The cost estimate was developed from UE&C inhouse information. Conceptual structural material quantities were developed by performing takeoffs for major structures, where practical, and by scaling for all other structures using similarly designed buildings as a basis. Costs for equipment and systems were also adjusted using assumptions and extrapolations based on known component costs.

The capital cost estimate presented in Tables 5.1-1 and 5.1-2 reflect a nth-of-a-kind (NOAK) Steamer plant. For first-of-a-kind (FOAK) Steamer plant, the capital costs are expected to be higher. The FOAK adders, in January 1978 dollars, are estimated at roughly $\$6 \times 10^6$ for BOP costs and $\$5.8 \times 10^6$ for NSSS costs. It is assumed that Steamer plant will follow the construction of a similar 2240 MWt HTGR plant. The FOAK allowance for the balance of plant costs provides for first-of-a-kind analysis, design (for unique components), labor productivity, developmental costs, etc. The FOAK allowance for the NSSS is discussed in greater detail in Subsection 7.1.

The capital cost estimate includes capital costs associated with a waste heat rejection system designed to provide the cooling requirements during reactor startup and hot standby operation and also to reject normal service water heat. This system may not be required for hot standby operation, depending upon the requirements of the process steam user and other operational considerations. In such cases, the size of the system will be reduced considerably and the capital cost estimate presented in Table 5.1-1 will be lower by up to \$3,000,000.

5.2 PROJECT SCHEDULE

Figure 5.2-1 shows the overall project schedule for nth-of-a-kind HTGR Steamer plant.

The twenty one months schedule from project start to limited work authorization was developed by the General Atomic Company (GA). It is based on the assumption of a standardized plant design and preselected site, which minimizes the time required for the licensing phase. It also assumes prebidding of material for the PCRV liner, steam generators, and other long lead components.

The construction schedule from limited work authorization to commercial operation was developed by UE&C. It is based primarily on the schedule previously developed by UE&C for a 3360 MWt HTGR and supplemented by drawings developed for the steamer plant. This schedule indicates approximately 67 months from receipt of the limited work authorization (LWA) to commercial operation, determined by the critical path which runs through the RCB and PCRV.

A detailed construction schedule is shown in Figure 5.2-2.

Major assumptions and methods considered in the preparation of this schedule were as follows:

1. Work on the PCRV, setting of the steam generators and CAHEs, and erection of the polar crane are completed prior to the dome being placed on the containment structure. The containment and annulus walls are brought up allowing work within to proceed.
2. A heavy piece of rigging equipment is utilized to reach over the containment and annulus walls. In order to have the derrick in place for lifting the bottom head, and all subsequent heavy lifts, the annulus wall must be placed up to grade and backfilled.
3. The annulus mat and bottom liner are scheduled concurrently in order to get an earlier start on the annulus walls. The annulus walls and the containment liner are scheduled concurrently, this will allow the derrick to be ready to set the bottom head.
4. Included in the "bottom head forms, shoring and set lower steam generator sections" line on this schedule are:
 - a. Erect bottom head forms and shoring.
 - b. Set, align, brace, tie off two lower steam generator liner sections.

- c. Install all instrument conduit and sleeves.
 - d. Install bottom head spare penetrations.
 - e. Install cooling water headers.
 - f. Erect tendon tubes.
 - g. Install sample lines and rebar adjacent to lower steam generator liners.
 - h. Erect outer rebar curtain walls to first splice.
 - i. Erect precast panel rings to get above pour.
5. Without appropriate information concerning the number of pours for this PCRV, that available for the 3360 MWt PCRV was utilized. The 3360 MWt PCRV is 89 ft high requiring 23 pours. The 1170MWt PCRV is 93 ft high; using the sketch provided by GAC this would indicate that the 1170MWt PCRV has one additional pour. The durations and sequences developed for the 3360 MW PCRV were used for this schedule plus seven days for the additional pour.
 6. Wire winding and tendon tensioning is scheduled concurrently with installation of the steam generators, as approved by GAC for the 3360 MWt schedule.
 7. Rigging of the steam generators, CAHEs and other equipment going into the PCRV cavities is essentially a continuous operation. It should be re-iterated here, as was stated in the 3360 MWt schedule study, that GAC should determine what effect the sequence of installation has on the load balance of the PCRV.
 8. Structural steel between the PCRV and the containment cannot be installed until after wire wrapping is completed, due to space requirements. The steel supports and restrains equipment, piping, etc. Piping, equipment, etc. can be started earlier, as shown, within the annulus area once the walls are completed.
 9. The PCRV start-up schedule (from PCRV warm up to start fuel load) is assumed to be 32 weeks, the same as that for the 3360 MWt HTCR.

5.3 CASH FLOW

Figure 5.3-1 shows percent of cash vs. time for the NSSS (Account 220) and balance-of-plant capital cost estimates for the steamer plant. The NSSS cash flow was provided by GAC and the balance-of-plant cash flow is based on UE&C's experience with nuclear power plants. The percent of cash flow is based on constant dollars and does not reflect the effects of inflation.

5.4 OPERATION AND MAINTENANCE COSTS

Nonfuel annual operating and maintenance costs are summarized in Table 5.4-1. Values are shown in terms of January 1978 dollars. Staff requirements for the plant manager's office, operations, maintenance, technical and engineering, and plant security are estimated at 203 people. Insurance and fees include commercial liability insurance, government liability insurance and inspection fees. Administrative and general expenses include Owner's offsite salaries and expenses directly allocable to the Steamer plant, and are estimated at 15 percent of the total fixed costs of staff, maintenance materials, and supplies and expenses.

TABLE 5.1-1

TWO DIGIT SUMMARY CAPITAL COST ESTIMATE

COST BASIS 01/76						
ACCT NO	ACCOUNT DESCRIPTION	FACTORY EQUIP. COSTS	SITE LABOR HOURS	SITE LABOR COST	SITE MATERIAL COST	TOTAL COSTS
*****	*****	*****	*****	*****	*****	*****
20 .	LAND AND LAND RIGHTS				268,800	268,800
21 .	STRUCTURES & IMPROVEMENTS	3,410,879	3300936 MH	35,605,432	30,762,274	69,778,585
22 .	REACTOR PLANT EQUIPMENT	159,699,992	1245965 MH	19,016,079	8,678,082	187,394,153
23 .	TURBINE PLANT EQUIPMENT	24,457,808	581562 MH	6,731,247	2,588,324	33,777,379
24 .	ELECTRIC PLANT EQUIP	13,100,100	1241215 MH	15,245,657	7,898,598	35,244,355
25 .	MISC. PLANT EQUIP	3,005,433	234972 MH	2,757,167	784,359	8,556,999
26 .	MAIN COND HEAT REJECT SYS	3,004,133	87275 MH	1,042,574	695,323	4,742,030
2 .	TOTAL DIRECT COSTS	207,678,365	6691925 MH	80,408,176	51,675,760	339,762,301
91 .	CONSTRUCTION SERVICES	13,380,000	1180000 MH	11,775,000	17,015,000	42,170,000
92 .	HOME OFFICE ENGRG & SERVICE	64,145,000				64,145,000
93 .	FIELD OFFICE ENGRG & SERVICE	16,000,000			1,920,000	17,920,000
9 .	TOTAL INDIRECT COSTS	93,525,000	1180000 MH	11,775,000	18,935,000	124,235,000
	TOTAL BASE COST	301,203,365	7871925 MH	92,183,176	70,610,760	463,997,301

TABLE 5.1-2
(Sheet 1 of 4)

THREE DIGIT SUMMARY CAPITAL COST ESTIMATE

COST BASIS 01/78						
ACCT NO	ACCOUNT DESCRIPTION	FACTORY EQUIP. COSTS	SITE LABOR HOURS	SITE LABOR COST	SITE MATERIAL COST	TOTAL COSTS
*****	*****	*****	*****	*****	*****	*****
20 .	LAND AND LAND RIGHTS				268,800	268,800
211.	YARDWORK	161,258	215035 MH	2,133,732	2,406,930	4,721,920
212.	REACTOR CONTAINMENT BLDG	649,975	1096857 MH	12,133,440	11,904,125	24,687,540
213.	TURBINE BUILDING	488,488	193706 MH	2,198,040	3,055,660	5,742,188
214.	SECURITY BUILDING	34,560	13405 MH	154,425	144,640	333,625
215.	REACTOR SERVICE BLDG	438,143	299211 MH	3,164,456	2,240,539	5,843,138
216.	MAIN CIRC CONTROL BLDG		4842 MH	53,237	339,065	392,302
217.	FUEL STORAGE BLDG	49,993	216081 MH	2,301,352	1,467,472	3,818,817
218A.	CONTROL, AUX & D.G. BLDG	1,194,473	466586 MH	4,995,121	2,877,867	9,067,461
218B.	ADMIN + SERV BLDG	196,997	84028 MH	957,016	896,455	2,050,470
218C.	AUX BOILER & H.V. DEMIN BLDG				148,000	148,000
218D.	FIRE PUMP HOUSE	17,513	10737 MH	115,771	83,539	216,823
218E.	HELIUM STORAGE AREA		32453 MH	335,508	274,594	610,102
218H.	DIES CLG + FL OIL STG BLDG	11,130	65337 MH	677,940	341,482	1,030,552
218I.	ACCESS BUILDING	90,880	53757 MH	618,443	613,746	1,323,069
218J.	CONTAINMENT ANNULUS BLDG		86138 MH	917,200	927,775	1,844,975
218K.	CONTAIN PENETRATION BLDG		266796 MH	2,810,191	1,809,262	4,619,453
218S.	HOLDING POND		8198 MH	64,090	43,080	127,170
218T.	ULTIMATE HEAT SINK STRUCT	37,469	179650 MH	1,878,084	1,150,545	3,066,098
218V.	CTL RM EMG AIR IN STR	20,000	8119 MH	77,384	37,498	134,882
21 .	STRUCTURES & IMPROVEMENTS	3,410,879	3300936 MH	35,605,432	30,762,274	69,778,585

TABLE 5.1-2
(Sheet 2 of 4)

COST BASIS
01/75

ACCT NO *****	ACCOUNT DESCRIPTION *****	FACTORY EQUIP. COSTS *****	SITE LABOR HOURS *****	SITE LABOR COST *****	SITE MATERIAL COST *****	TOTAL COSTS *****
220A.	NUCLEAR STEAM SUPPLY (NSSS)	135,200,000				135,200,000
220B.	NSSS OPTIONS					
221.	REACTOR EQUIPMENT	525,942	579264 MH	11,156,631	7,656,898	19,339,471
222.	MAIN HEAT TRANS. SYS.	86,738	59727 MH	714,101	86,830	887,669
223.	SAFEGUARDS COOL. SYS.	1,951,975	112245 MH	1,318,254	167,335	3,447,564
224.	RAD WASTE PROCESSING	2,542,066	53745 MH	630,797	124,501	3,297,364
225.	NUCLEAR FUEL HANDLING + ST	11,870,635	47263 MH	554,482	60,395	12,485,513
226.	OTHER REACTOR PLANT EQUIP	5,657,635	226278 MH	2,660,611	326,524	8,844,770
227.	INSTRUMENTATION + CONTROL	1,352,060	63363 MH	834,932	15,313	2,202,245
228.	REACTOR PLANT MISC ITEMS	303,000	99080 MH	1,146,271	240,286	1,689,557
22 .	REACTOR PLANT EQUIPMENT	159,699,992	1245965 MH	19,016,079	8,678,032	187,394,153
231.	TURBINE GENERATOR	13,540,393	191747 MH	2,132,993	792,534	16,465,920
233.	CONDENSING SYS.	1,501,113	80395 MH	952,320	210,029	3,063,462
234.	FEED HEAT. SYS.	4,013,058	95907 MH	1,134,664	148,256	5,295,978
235.	OTHER TURB. PLANT EQUIP	4,290,244	125148 MH	1,460,839	456,818	6,207,901
236.	INSTRUMENTATION + CONTROL	713,000	47650 MH	581,954	506,014	1,800,968
237.	TURBINE PLANT MISC ITEMS		40715 MH	468,477	474,673	943,150
23 .	TURBINE PLANT EQUIPMENT	24,457,808	581562 MH	6,731,247	2,588,324	33,777,379

TABLE 5.1-2
(Sheet 3 of 4)

COST BASIS
G1/78

ACCT NO *****	ACCOUNT DESCRIPTION *****	FACTORY EQUIP. COSTS *****	SITE LABOR HOURS *****	SITE LABOR COST *****	SITE MATERIAL COST *****	TOTAL COSTS *****
241.	SWITCHGEAR	4,901,000	62464 MH	764,572	82,794	5,748,366
242.	STATION SERVICE EQUIP	5,746,800	85930 MH	1,027,761	197,480	6,972,041
243.	SWITCHBOARDS	667,300	11015 MH	135,553	12,175	815,028
244.	PROTECTIVE EQUIP		82784 MH	1,027,764	491,600	1,519,364
245.	ELEC STRUC + WIRING CNTNRS		484062 MH	5,897,653	1,867,665	7,765,318
246.	POWER & CONTROL WIRING	240,000	509960 MH	6,331,152	5,240,764	11,811,916
247.	STEP-UP TRANSFORMERS	545,000	5000 MH	61,202	6,120	612,322
24 .	ELECTRIC PLANT EQUIP	12,100,100	1241215 MH	15,245,657	7,898,598	35,244,355
251.	TRANSPORTATION+LIFT EQUIP	1,193,008	17732 MH	210,861	21,086	1,424,955
252.	AIR WTR+STEAM SERV SYS	1,820,351	185463 MH	2,179,499	706,072	4,705,922
253.	COMMUNICATIONS EQUIP	1,055,428	23400 MH	289,018	24,383	1,368,829
254.	FURNISHINGS + FIXTURES	936,666	7877 MH	87,809	32,818	1,057,293
25 .	MISC. PLANT EQUIP	5,005,453	234972 MH	2,767,187	784,359	8,556,999
261.	STRUCTURES	53,970	38823 MH	505,562	288,812	848,344
262.	MECHANICAL EQUIPMENT	2,950,163	48452 MH	537,012	406,511	3,893,686
26 .	MAIN COND HEAT REJECT SYS	3,004,133	87275 MH	1,042,574	695,323	4,742,030
2 .	TOTAL DIRECT COSTS	207,678,365	6691925 MH	80,408,176	51,675,760	339,762,301

TABLE 5.1-2
(Sheet 4 of 4)

COST BASES
01/78

ACCT. NO	ACCOUNT DESCRIPTION	FACTORY EQUIP. COSTS	SITE LABOR HOURS	SITE LABOR COST	SITE MATERIAL COST	TOTAL COSTS
*****	*****	*****	*****	*****	*****	*****
911.	TEMPORARY CONSTRUCTION FAC		1022000 MH	9,880,000	4,940,000	14,820,000
912.	CONSTRUCTION TOOLS & EQUIP		156000 MH	1,895,000	11,550,000	13,445,000
913.	PAYROLL INSURANCE & TAXES	13,380,000				13,380,000
914.	PERMITS, INS. & LOCAL TAXES				525,000	525,000
915.	TRANSPORTATION					
91 .	CONSTRUCTION SERVICES	13,380,000	1180000 MH	11,775,000	17,015,000	42,170,000
921.	HOME OFFICE SERVICES	60,940,000				60,940,000
922.	HOME OFFICE W/P	2,140,000				2,140,000
923.	HOME OFFICE CONSTRUCTION MGMT	1,065,000				1,065,000
92 .	HOME OFFICE ENGINEERING & SERVICE	64,145,000				64,145,000
931.	FIELD OFFICE EXPENSES				1,920,000	1,920,000
932.	FIELD JOB SUPERVISION	12,160,000				12,160,000
933.	FIELD GA/OC	2,030,000				2,030,000
934.	PLANT STARTUP & TEST	1,810,000				1,810,000
93 .	FIELD OFFICE ENGINEERING & SERVICE	16,000,000			1,920,000	17,920,000
9 .	TOTAL INDIRECT COSTS	73,525,000	1180000 MH	11,775,000	18,935,000	124,235,000
	TOTAL BASE COST	301,203,365	7871925 MH	92,183,176	70,610,760	463,997,301

5-10

TABLE 5.1-3

DIRECT CRAFT LABOR RATE SUMMARY
DOE-HTGR STEAMER STUDY
PORT ARTHUR, TEXAS
LABOR RATES - JANUARY 1978

		<u>Base Rate Incl. Rate Fringes (\$/hr)</u>
Asbestos Worker	R01	12.76
Boilermaker	R02	12.10
Bricklayer & Mason	R03	12.88
Carpenter	R04	10.97
Dock Builder	R05	10.97
Electrician	R06	12.65
Ironworker	R07	11.50
Laborer	R08	7.81
Millwright	R09	11.40
Operating Engineer	R10	11.73
Painter	R11	10.60
Pipefitter	R12	11.94
Plumber	R13	11.89
Roofer	R14	10.47
Sheet Metal Worker	R15	11.64
Teamster	R16	8.40

TABLE 5.1-4

ITEMS NOT INCLUDED IN COST ESTIMATE

1. Switchyard and transmission facility costs.
2. Owner's costs, including consultants, site selection, etc.
3. Waste disposal costs.
4. Fees and permits - federal, state and local.
5. State and local taxes.
6. Spare parts.
7. Nuclear liability and other insurance (except as noted).
8. Initial fuel loading.
9. Interest during construction.
10. Escalation.
11. Contingency.

TABLE 5.4-1

ANNUAL OPERATING AND MAINTENANCE COSTS
JANUARY 1978 DOLLARS

Staff	\$4,570,000
Maintenance Materials	
Fixed	1,734,000
Variable	0
Supplies and Expenses	
Fixed	3,558,000
	112,000
Insurance and Fees	400,000
Administrative and General	1,479,000
Total Fixed Costs	\$11,741,000
Total Variable Costs	112,000
Total Annual O&M Costs	\$11,853,000

Project Start

Figure 5.2-1

NOAK Project Schedule
1170 Mwt HTGR Steamer
Cogeneration Plant

Figure 5.2-2 (Sheet 2 of 2)

Construction Schedule
1170 MW HTGR Steamer
Cogeneration Plant

FIGURE 5.3-1 PERCENT OF CAPITAL COST CASH FLOW VS. TIME

SECTION 6.0

DESIGN ISSUES/TRADEOFFS, SAFETY AND LICENSABILITY

6.1 DESIGN ISSUES/TRADEOFFS

6.1.1 Reactor Core

The Steamer core (SC) uses the same fuel, fuel block design, core height, and reflector thicknesses as the HTGR-SC power plant core, and it is designed to operate at the same maximum power density (7.15 W/cm^3). The Steamer core neutron control and region flow control system components are also identical with those of the HTGR-SC core. The Steamer reactor core is reduced in diameter to provide approximately 50 percent of the thermal capacity of the core for the HTGR-SC power plant core. However, because of the geometrical layout of the hexagonal fuel blocks, the Steamer Core actually produces 52 percent of the HTGR-SC core power (1170 MW(t) versus 2240 MWt) when operated at the same power density. The core layout for the steamer is considered to have somewhat more favorable edge region control characteristics than that of the HTGR-SC core, resulting in more favorable core power peaking factors.

The design margins of the HTGR-SC steam generators (SGs) and helium circulators appear adequate to successfully accept the slightly increased coolant loop thermal capacity from the 1170-MW(t) core. The alternate approach of reducing the SC power density to approximately 6.85 W/cm^3 to attain exactly 50 percent of the HTGR-SC core power would maintain loop component capacity at the HTGR-SC design level, but would reduce the output of the Steamer plant.

6.1.2 Prestressed Concrete Reactor Vessel (PCRV)

Both the steamer and the HTGR-SC power plants utilize cylindrical multi-cavity PCRVs with the same maximum cavity pressure to contain the reactor core and primary and auxiliary coolant system components. For both plants, the core cavity is somewhat offset from the vertical centerline of the PCRV, and the coolant components are located in separate cavities beside the core. The same basic circumferential wire winding and linear vertical prestress system designs are utilized for both plants. The steamer's smaller core cavity diameter and reduction to two primary coolant loops instead of four results in a smaller PCRV diameter. Side wall and top and bottom head thicknesses of the PCRV are also less than in the HTGR-SC because of the reduced core cavity diameter, resulting in additional outside diameter and overall height reductions relative to the HTGR-SC PCRV.

The PCRV design is a variation of the side-mounted, multi-cavity vessels introduced in the British Hartlepool and Heysham AGR plants and of several designs developed in studies of large HTGR-SC plants. Several studies at General Atomic (GA) have demonstrated the advantages of off-setting

the reactor core from the geometric center of the periphery of the vessel in terms of vessel diameter and plant arrangement. This approach permits a reduction in vessel diameter relative to arrangements with the core cavity in the center of the vessel.

The offset core, multi-cavity PCRV configuration has been chosen for the large HTGR-SC program. No major impact on design confidence for this configuration relative to concentric core, multi-cavity PCRVs has been identified.

6.1.3 Core Auxiliary Cooling System (CACS)

The required heat removal capacity of the steamer auxiliary loops is smaller than that of the HTGR-SC power plant loops because of the smaller core. The auxiliary circulator design is the same for both the HTGR-SC and the steamer, with operating conditions being adjusted by speed control to suit the steamer core cooling requirements. The steamer CAHEs have shorter tube bundles than do the HTGR-SC power plant exchangers because of the reduced core cooling capacity. The cooling water system is also designed for the reduced cooling requirements.

Both plants utilize three auxiliary coolant loops in addition to the primary coolant loops to provide safe shutdown of the reactor. System design ground rules are the same. The design of the auxiliary helium circulator and helium shutoff valve in each auxiliary loop is identical for both plants. The CAHE utilizes the same basic design configuration, tube size, and tube pitch, but the tube bundle length is shorter than that of the HTGR-SC CAHE because of the reduced cooling duty requirement.

No engineering feasibility or technology development issues are expected to apply the HTGR-SC CACS design to the steamer.

6.2 SAFETY

6.2.1 Safety Features of the HTGR Steamer

The HTGR steamer plant design incorporates design features which have established a high degree of safety for the HTGR. The risk to the public from accidental releases of radioactivity is extremely low for the HTGR. The inherent safety characteristics of a reactor include many of the basic aspects of the design, such as low reactor power density, high core heat capacity, use of ceramic materials for the core and moderator, single-phase reactor coolant, and employment of a PCRV. These and other characteristics dictate the nature of the physical processes and mechanisms which determine the transport of fission products to the environment in the event of low-probability accidents. Other components of the design, such as the CACS and secondary containment structures, are classified as engineered safety features, since their safety function is to reduce the likelihood and mitigate the consequences of reactor accidents involving radioactivity release.

One of the important inherent safety features of the HTGR is the design and selection of materials for the reactor core. The large heat capacity associated with the core graphite, together with a low power density, provides for a slow rate of temperature rise in the unlikely event of a core accident. Thermal inertia introduces substantial time delays for mitigating actions to restore core cooling.

The ceramic materials (pyrolytic carbon and silicon carbide) used as fuel particle coatings maintain their integrity at temperatures well in excess of normal operating temperatures. During a postulated core heatup event, radionuclide release from the core does not begin until about ten hours following initiation and proceeds slowly thereafter. This time delay allows for radioactive decay of short-lived radionuclides before release, which significantly reduces the quantity of fission products that can be released from the core.

A second inherent safety feature of the HTGR is the use of chemically inert and thermodynamically stable helium as the primary coolant. Since the coolant is gaseous, a loss of coolant accident (LOCA) is a physical impossibility in the HTGR. The low stored energy and phase stability of the coolant eliminate several mechanisms which have been postulated to cause containment failure in the case of liquid-cooled reactors. For this reason, the containment building is able to contribute to the overall safety margin of the HTGR.

Another important safety feature of the HTGR is the design of the PCRV. Since the entire primary coolant system is enclosed within the PCRV, the integrity of the primary coolant system boundary is not heavily dependent on single structural elements. In the event of a core heatup, the inherent features of the core, discussed above, together with the PCRV design provide for relatively slow rates of radioactivity release from the PCRV

to the containment building. The large inner surface area of the PCRVR accommodates substantial quantities of condensible radionuclides (plate-out), which reduces the quantities of particulates and halogens that can be released from the PCRVR.

In addition to the inherent safety features of the HTGR steamer, a number of engineered safety feature (ESF) systems are provided which greatly mitigate the consequences of postulated accidents having a low probability of occurrence. These ESF systems are listed below.

6.2.1.1 Containment Building

The containment building minimizes the release to the environment of fission product activity following postulated accidents involving the primary coolant system. The containment building is designed to be leak-tight within specified limits and to withstand temperature and pressure transients of postulated accidents, including the design basis depressurization accident (DBDA).

6.2.1.2 Containment Air Purification and Cleanup System

To minimize the availability of fission products that might leak from the containment, the containment air purification and cleanup system removes and retains radioactive particulates and halogens present in the containment atmosphere that result from an accidental release of fission products from the reactor coolant system.

6.2.1.3 Containment Isolation Systems

The containment isolation system provides automatic containment isolation upon an increase in containment pressure or an increase in containment airborne activity beyond predetermined set points. The containment pressure protection system isolates all SGs upon an increase in containment pressure with normal primary coolant pressure.

6.2.1.4 Core Auxiliary Cooling System (CACS)

The function of the CACS is to provide an independent means of cooling the reactor core with the primary system pressurized or depressurized. The CACS cooling maintains all PCRVR component temperatures within safe limits.

The CACS consists of three cooling loops, each capable of removing 100 percent of the core residual and decay heat for safe cooldown from 102 percent of reactor steady-state power level under pressurized conditions. Two loops are required in the depressurized condition, each with 50 percent capability.

6.2.2 Siting Advantages

Analysis for HTGRs shows that the risks associated with locating an HTGR on an urban or industrial site are very low. From the standpoint of radiographical and environmental impact, the HTGR should have considerable siting flexibility.

It is believed that the atmosphere pollution impact associated with a nuclear plant may be much lower than that for a coal-fired plant because of the emission of pollutants such as SO₂ and particulates associated with coal-fired plants.

The total societal and economic impacts of emergency planning, testing, and implementation are beginning to receive considerable attention by regulatory bodies, including a Federal interagency task force. As a result, there may be new regulations and increased financial responsibility by plant owners for emergency plans, which could be translated into an advantage for the HTGR.

6.3 LICENSABILITY

An application has not been made for review of a two-loop plant of the size and design described herein. However, the 1170-MW(t) Steamer is a variation of a design that has undergone three NRC reviews at the Preliminary Safety Analysis Report (PSAR) stage. These have been the 3000-MW(t) Fulton Generating Station (Philadelphia Electric), the 2000-MW(t) Summit Power Station (Delmarva Power and Light), and GA's standard NSSS design for a 3000-MW(t) plant. Reviews of both the Summit and Fulton PSARs were carried to completion of NRC safety evaluation reports. The NRC review of the standard plant PSAR (GASSAR-6) was not completed, but a partial preliminary safety evaluation was prepared.

As a result of these reviews, a number of licensing issues have been raised by the NRC, and GA has responded or made plans to respond to these issues in several ways. It is believed that all issues raised can be resolved by relatively straight forward engineering and that no insurmountable obstacles to licensing are presently known or are likely to be revealed in the future.

SECTION 7.0

DESIGN VERIFICATION AND SUPPORT REQUIREMENTS

7.1 APPLICABILITY OF HTGR-SC PROGRAM TO STEAMER

The Steamer design is based on the HTGR-SC power plant design, and the design verification and support (DV&S) requirements will be covered in large part by the HTGR generic technology program. Since October 1977, GAC has been working with Gas Cooled Reactor Associates (GCRA), an organization of supporting electric utilities. The work has included an HTGR-SC specific program which has resulted in a 2240 MWt HTGR-SC design. This design is the basis for the 1170 MWt HTGR Steamer design. Work areas included in the HTGR generic technology and HTGR-SC programs with applicability to the Steamer are listed in Table 7.1-1. Objectives of the various work areas follow with comments made on significant and specific DV&S work required for the Steamer.

7.1.1 Safety and Reliability

The objectives of this work area are to ensure that (1) the HTGR-SC plant meets applicable licensing and safety criteria, (2) availability is optimized, and (3) results are effectively communicated.

Specific plant safety analysis reports (SARs) and responses to questions from the NRC or other licensing agencies will have to be prepared and defended for the Steamer. This information will include accident analyses of the core and all safety-related systems and components. The HTGR-SC analyses and safety reports will provide a firm base for development of the Steamer safety reports.

7.1.2 Systems Engineering

The objectives of this work area are to (1) establish and optimize the overall plant performance parameters, (2) develop the technical information and design criteria related to plant dynamics and acoustics/vibrations, (3) establish overall design for the primary coolant system, (4) develop design criteria for plant control and protection systems, (5) provide systems engineering support for utility-architect/engineer liaison, procedure development and site engineering, and (6) integrate the engineering design efforts.

The Steamer and HTGR-SC power plant have in common (1) identical design of the SGs, helium circulators, and helium shutoff valves, (2) identical primary coolant pressure and temperature conditions, except for differences in core and loop pressure drops, (3) identical core inlet conditions and core differential temperature t , and (4) identical steam conditions, except for SG pressure drop.

Steamer steady-state and transient performance analyses will be required, along with analyses to verify component performance with the slightly higher Steamer loop capacity. Core and component pressure drops will need to be analyzed in detail for the Steamer configuration. Flow conditions, helium circulator performance, and power requirements will need to be verified.

The Steamer loop arrangement is somewhat less symmetrical than that of larger HTGR-SCs and will require investigation of flow distribution and gas mixing (e.g., hot streaks) in both the upper and lower plenums. Flow distribution testing is to be carried out as part of the generic program. The program will examine mixing behavior in a general 6-loop arrangement, and limited testing will be done on a 4-loop arrangement. Similar tests can be included for the 2-loop Steamer at a later date. Mixing behavior with one loop shut down will also need to be investigated, if the plant is to operate at 50 percent power, so that higher availability may be achieved.

No other special primary coolant system design effort is expected for the Steamer beyond the HTGR-SC requirements.

The HTGR-SC CACS design is covered under the HTGR generic technology programs.

Analyses will be required to establish core cooling requirements for the Steamer core configuration in both pressurized and depressurized modes and to establish system performance under anticipated shutdown and emergency cooling conditions. The CAHE heat-duty requirements will need to be established, and analyses will need to be performed to verify the performance of the auxiliary helium circulator.

7.1.3 Structural Engineering

The main objectives of this program are to (1) perform the design development of the PCRV, liner/penetrations, thermal barrier, and reactor internals, along with supporting shielding design and seismic design, (2) resolve any outstanding licensing or safety issues, (3) perform supporting analysis and design review, (4) perform required design verification and support (DV&S) programs associated with the design of components that are unique to the steam cycle plant, and (5) provide technical and liaison support throughout the HTGR-SC program schedule.

Concrete stress distributions are somewhat more complex for the asymmetric cavity PCRV arrangement used in both the HTGR-SC and the Steamer, but the symmetric vessel analysis methods may be used for elastic response prediction and the establishment of bounds on time-dependent effects and ultimate strength. No need for additional methods has been identified to commit to this configuration in the HTGR-SC program. However, a model test of design adequacy is currently planned to facilitate licensing.

For the Steamer PCRV, it will be necessary only to develop design and analysis calculations, design and construction specifications, and structural plan, section and detail drawings. This is expected to be routine engineering, based on the HTGR-SC design and development programs.

7.1.4 Heat Exchanger Equipment

The objectives of this work area are to (1) develop an optimized SG design, (2) perform supporting analysis and specific DV&S test programs, and (3) provide appropriate coordination activities.

Analyses will be required for the Steamer to verify the performance of the SCs at the slightly higher duty required, relative to the HTGR-SC. The HTGR-SC CAHE design development, including analytical and design methods, is covered under the HTGR generic technology program. The HTGR-SC CAHE design will be used for the Steamer, except for a somewhat shorter tube bundle size consistent with the Steamer core cooling requirements.

7.1.5 Mechanical and Electrical Equipment

The objective of this work area is to provide technical and liaison support to the HTGR-SC program in applying the design of the mechanical and electrical equipment to the HTGR-SC plant. The Steamer is not expected to require any special effort in this area.

7.1.6 Alternate Design Studies

The objective of this work area is to consider commercial plant design alternatives to maintain a competitive position by keeping abreast of evolving utility preferences and regulatory climate. Any developments adopted for the HTGR-SC will also be applied to the Steamer.

7.1.7 Materials Technology

The main objective of this work area is to assure that information developed in the HTGR generic technology program and non-HTGR programs is effectively utilized for the HTGR-SC design. All information developed for the HTGR-SC will also apply to the Steamer.

7.1.8 Reactor Operating Experience

This area is included in the HTGR generic technology program plan.

7.1.9 Reactor Core Engineering

All DV&S work needed to justify design methods for core licensing and to verify that the core components will meet performance standards and design criteria is to be carried out under the HTGR generic technology program. This will apply to the Steamer as well as to the HTGR-SC. However, since the Steamer has a smaller core than the HTGR-SC with somewhat different

geometry in the outer control region, core performance analyses specific to the Steamer will be required.

The analyses will establish region peaking, intraregion tilt spectrum, fuel peak and operating temperatures, and temperature-sensitive fuel particle failure modes. Overall core safety margins and the impact on licensing requirements will also need to be evaluated.

The work in the core design area is divided into two tasks, which are briefly described below.

7.1.9.1 Reactor Core System

A comprehensive core performance evaluation will be carried out and documented. Spatial distribution and time histories of power distribution, heavy metal composition, core temperatures, fuel particle coating failure, fuel element strains and stresses, fission product release, and fission product distribution in the primary circuit will be determined. Steady-state and transient conditions will be investigated. The performance of the fuel elements in the core will be determined, and interfaces consistent with the fuel performance will be established. Both steady-state and transient conditions will be covered.

Core assembly drawings will be prepared which specify the arrangement of all core components. Design packages will also be prepared for the fuel elements, hexagonal reflector elements, plenum elements, neutron sources, control rods, and reserve shutdown material.

7.1.9.2 Project/Licensing Support

Plant SARs and responses to questions from the NRC or other licensing agencies will be prepared and defended. This information will include the accident analysis of the core and core components.

7.1.10 Fuel Material Development

This area is covered in the HTGR generic technology program plan and will apply to the Steamer as well as to the HTGR-SC.

7.1.11 Graphite Development

This area is covered in the HTGR generic technology program plan and will apply to the Steamer as well as to the HTGR-SC.

7.1.12 Fission Product/Coolant Chemistry

This area is covered in HTGR generic technology program plan and will apply to the Steamer as well as to the HTGR-SC.

7.1.13 Fresh Fuel Process Development

This area is covered in HTGR generic technology program plan and will apply to the Steamer as well as to the HTGR-SC.

7.1.14 Program Management and Support

Management methods and procedures developed for the HTGR-SC program will be generally applicable to the Steamer program.

7.1.15 Technology Center

An HTGR technology center will be established as part of the HTGR generic technology program plan and will provide support for the Steamer program as well as for the HTGR-SC and other HTGR programs.

7.1.16 Fort St. Vrain MEU Program

Testing of MEU fuel in the Fort St. Vrain reactor is to be conducted under the HTGR generic technology program plan and will apply to the Steamer as well as to the HTGR-SC.

7.1.17 BOP Engineering

The overall objective of this work is to aid in the design, licensing, construction, operation and maintenance of an economical and reliable large HTGR-SC power plant. More specifically, the initial objective is to develop conceptual BOP designs that will support evaluation and selection of the configuration for the 2240-MW(t) HTGR-SC plant. Following the selection of a concept, this task will prepare the documents and data necessary to support an architect/engineer in designing the BOP and integrating it into an overall plant design. It is also the intent to give necessary study and planning to such subjects as ISI, maintenance, plant availability, BOP aspects of licensing, prestartup testing, and operation. Much of the HTGR-SC work in this area will also apply to the Steamer.

7.2 Steamer DV&S Cost Estimate

The HTGR-SC program provides a strong base for the Steamer design, and additional design verification and support effort is expected to be minimal. A preliminary estimate of costs for additional design verification and support specific to the steamer NSSS is given in Table 7.2-1.

TABLE 7.1-1

HTGR-SC PROGRAM WORK AREAS APPLICABLE TO THE STEAMER COGENERATION PLANT

Work Areas	Partly Applicable	Fully Applicable
Safety and reliability	X	
Systems	X	
Structural engineering	X	
Heat exchanger equipment	X	
Mechanical and electrical equipment	X	
Alternate design and studies/ product improvement programs	X	
Materials technology		X
Reactor operating experience		X
Reactor core engineering	X	
Fuel materials development		X
Graphite development		X
Fission products/coolant chemistry		X
Fresh fuel process development		X
Program management and support	X	
Technology center		X
MEU for Fort St. Vrain	--	--
BOP engineering	--	--

TABLE 7.2-1

PRELIMINARY ESTIMATE OF STEAMER NSSS DESIGN VERIFICATION
AND SUPPORT COSTS*
(In 1978 Dollars)

Safety and reliability	$\$1.3 \times 10^6$
Systems engineering	$\$2.1 \times 10^6$
Structural engineering	$\$1.0 \times 10^6$
Heat exchanger design	$\$0.4 \times 10^6$
Mechanical and electrical design	$\$1.0 \times 10^6$
Materials engineering	--
Core and fuel development	--
Total	$\$5.8 \times 10^6$

*Incremental costs for specific steamer related tasks over and above the costs for the 2240 MWt HTGR-SC design development.