HTGR-85-021

HTGR

CONTAINMENT VS CONFINEMENT TRADE STUDY SMALL HTGR PLANT **MASTER** PCRVR CONCEPT

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AUTHOR / CONTRACTOR

STONE & WEBSTER ENGINEERING CORP.

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UNDER SUBCONTRACT TO GAS-COOLED REACTOR ASSOCIATES
FOR THE DEPARTMENT OF ENERGY
CONTRACT DE-ACO3-78SFO2034

MARCH 1985

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CONTAINMENT VS CONFINEMENT

TRADE STUDY

SMALL HTGR PLANT

PCRV CONCEPT

Author: STONE & WEBSTER ENGINEERING CORP.

Issued By:
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MASTER

DOE Contract No. DE-AC03-78SF02034

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Summary Construction Schedule - 1260 MWt Confinement

LIST OF ACRONYMS

BONP	balance of nuclear plant
CB	control building
EAB	exclusion area boundary
GCRA	Gas-Cooled Reactor Associates
HTGR	high temperature gas reactor
HVAC	heating, ventilating, and air conditioning
LWA	limited work authorization
NSSS	nuclear steam supply system
O&M	operations and maintenance
PAG	protective action guideline
PCRV	prestressed concrete reactor vessel
PWR	pressurized water reactor
RB	reactor building
SSE	safe shutdown earthquake
SWEC	Stone & Webster Engineering Corporation

SECTION 1

INTRODUCTION AND SUMMARY

This trade study has been conducted to evaluate the differences between four different HTGR nuclear power plants. All of the plants use a pre-stressed concrete reactor vessel (PCRV) to house the core and steam generation equipment. The reactor uses LEU U/Th fuel in prismatic carbon blocks.

The HTGR power plants are rated at approximately 1200 MWt core power and utilize a steam turbine-generator power conversion cycle to produce electricity. The first plant concept is an 1170 MWt NSSS (1170 Ref NSSS) with forced circulation decay heat removal, utilizing a containment structure for the control of radio-nuclide release. The second and third plant concept is an 1170 MWt NSSS (1170 NSSS) with forced and natural circulation decay heat removal, utilizing a reinforced concrete building designed as a confinement or a reinforced concrete steel-lined pressure vessel to control radio-nuclide release. The fourth concept is a 1260 MWt NSSS (1260 NSSS) using a combination of forced and natural circulation concepts for decay heat removal with a confinement structure for the control of radio-nuclide release.

All plant concepts meet the utility/user requirements established for small HTGR plants. All plants will be evaluated with regard to their ability to produce safe, economical power to satisfy Goals 1, 2, and 3 of the HTGR program and by meeting the MUST criteria established in the concept evaluation plan.

Capital costs for each plant were evaluated on a differential cost basis. These costs were developed according to the "NUS" code of accounts as defined in the Cost Estimating and Control Procedure, HP-20901. Accounts that were identical in scope for all four plants were not used for the comparison. Table 1-1 is a list of capital cost accounts that were evaluated for each plant.

The PCRV concepts for the small HTGR plant were evaluated to determine the differences between concepts which utilize a containment or a confinement. All of the concepts meet the "MUST" criteria established for Goal 3, "Meet Power Generation Needs," and Goal 4, "Require Limited Development Costs."

The overall plant arrangement is not sensitive to either a containment or a confinement. The arrangement of reactor plant equipment differs between the containment and confinement concepts in that the containment concept uses two structures, the reactor containment and auxiliary building versus a single structure, the reactor building, for the confinement. These differences are described and illustrated in Section 2.

The reactor plant systems and structures differ considerably between the concepts with each concept representing an approach to meet Goal 1, "Provide Maximum Degree of Function Accomplishment." These differences are described in Section 3.

A dose analysis of the concepts showed that the 1170 Reference NSSS and the 1170 NSSS with a containment and the 1260 NSSS with a confinement were able to satisfy Goal 1 and meet all "MUST" Criteria for Goal 1. The 1170 NSSS with a confinement did not meet "MUST" Criterion IC, "Public Acceptance of Safety Characteristics."

A construction schedule was developed in Section 3 for the containment and confinement concepts. This schedule shows a 47-month schedule for a plant with a containment and 40-month schedule for a plant with a confinement.

Overall plant costs are presented in Section 6 and are:

1170 NSSS Confinement	\$844,000,000 (1917\$/kW)
1170 NSSS Containment	947,000,000 (2152\$/kW)
1170 Ref NSSS Containment	963,000,000 (2188\$/kW)
1260 NSSS Confinement	992,000,000 (2088\$/kW)

The estimated cost of product for these concepts are 4.2¢/kWh, 4.6¢/kWh, 4.6¢/kWh, and 4.5¢/kWh, respectively. All concepts meet the "MUST" criteria for Goal 2, "Provide Competitive Costs," for economic competitiveness with electricity generated using coal as a fuel.

Operations and maintenance (O&M) costs were evaluated for the concepts. It was established that O&M costs would be higher for the containment concepts due primarily to maintenance and tests associated with the containment and containment isolation systems.

This trade study has shown that the concepts are able to satisfy the "MUST" criteria, thereby meeting the goals of the HTGR program with the exception of the 1170 NSSS with confinement.

TABLE 1-1
COST ACCOUNTS

3-DIGIT ACCOUNT NO. -----	DESCRIPTION -----
21	STRUCTURES AND IMPROVEMENTS
212	REACTOR BUILDING
218T	ULTIMATE HEATSINK
218X	AUXILIARY BUILDING
218Z	EMERGENCY POWER BLDG.
22	REACTOR PLANT EQUIPMENT
220A	NSSS BASE SCOPE
221	REACTOR EQUIPMENT
222	MAIN HEAT TRANSPORT
223	SAFEGUARDS COOLING
225	NUCLEAR FUEL HANDLING
226	OTHER REACTOR PLANT EQPM'T
227	REACTOR PLANT INSTR/CONTROL
24	ELECTRIC PLANT EQUIPMENT
241	SWITCHGEAR
242	STATION SERVICE EQUIPMENT
243	SWITCHBOARDS
244	PROTECTIVE EQUIPMENT
245	ELEC STRUC & WIRING CTRLS
246	POWER & CONTROL WIRING
25	MISCELLANEOUS PLANT EQUP'T
251	TRANSPORTATION & LIFTING

SECTION 2

PLANT ARRANGEMENT

The HTGR Plant is comprised of a single reactor system that provides a source of steam for a single turbine-generator.

The reactor and the turbine generator have been placed in separate areas to permit construction of the associated structures and systems using nuclear and fossil power plant construction methods, respectively.

The reactor and its associated systems and structures, principally those parts of the plant that would be built to nuclear construction standards, are located within a double security fence. This area of the plant is designated as the nuclear area.

The turbine generator, its associated systems and structures, and all parts of the plant that would be designed and built to conventional fossil construction standards are located in a security area adjacent to the nuclear area. This area is designated as the power generation area. Cost savings are anticipated through the use of segregated design and construction activities.

A typical plant arrangement of the nuclear area is shown in the Plot Plan, Figure 2-1. This arrangement is for a containment concept. A confinement would differ in that the reactor containment and auxiliary building would be replaced with a single reactor building.

Nuclear Area

The nuclear area for those concepts using a containment contains the following major buildings and structures:

- Reactor Containment

- Auxiliary Building
- Control Building
- Fuel and Cask Handling Building
- Radwaste Building
- Emergency Power Supply Building
- Reactor Service Building
- Nuclear Service Water Cooling Tower
- Auxiliary Cooling Water Air-Cooled Heat Exchanger Structure
- Spent Fuel Cask Storage Area

The reactor containment is a steel-lined reinforced concrete, circular cylindrical structure capped by a hemispherical dome. It houses the PCRV, NSSS auxiliaries, and a polar crane.

Sufficient area has been provided around the PCRV to provide clearance for initial wire winding and rewinding of the circumferential prestressing tendons.

A polar crane is provided to service the PCRV and fuel handling components.

The auxiliary building is a rectangular, reinforced-concrete structure built adjacent to and around the containment. It houses the balance of nuclear plant (BONP) components. It is on a common mat with the containment which is located at grade elevation. The equipment layout for this building is also shown in Figures 2-2 through 2-5. A bridge crane is provided to service the fuel handling components. A 43-foot clear space above the refueling floor permits maintenance of the fuel handling machine, main

circulators, and auxiliary service cask. A reactor equipment service facility, with access from the refueling floor utilizing the overhead crane, has been provided.

A gas-tight passage through the mat connects the core cavity and the temporary fuel storage facility located at grade elevation in the auxiliary building, permitting the transfer of new and spent fuel in a helium environment. Spent fuel is transferred from the temporary fuel storage facility and, subsequently, loaded into storage casks and moved into the fuel and cask handling building. From there, the spent fuel storage casks are moved to the spent fuel storage area using a mobile transporter.

The auxiliary building structure also houses the heat transfer equipment and chimney for the natural-draft, air-cooled, heat exchangers.

The control building houses the control and electrical equipment required for plant operation, which includes the main control room, Class 1E switchgear, computer area, and battery room. A remote shutdown area, which is isolated from the control room and reactor building, is provided. The control building is a reinforced concrete structure and is located adjacent to the Reactor Containment.

The radwaste building houses the equipment and systems to receive and process all radioactive waste generation at the plant, with the exception of the spent fuel. The building is a single-story, composite steel and concrete structure. The areas in which radioactive wastes are stored are surrounded by concrete to provide radiation protection. Other areas are constructed with steel framing and metal siding.

The reactor service building is adjacent to the reactor and control buildings and houses the technical support center, a "hot" machine shop including an ultrasonic cleaning room, health physics, lockers, change rooms, and plant management and technical support offices.

The stand-by power supply building houses the two gas turbines and associated auxiliary and control equipment. This equipment is housed in a reinforced concrete structure. Separate buried fuel storage tanks have been provided for each system.

The Nuclear Area for the confinement concepts contains the following major buildings and structures:

- Reactor Building
- Control Building
- Fuel and Cask Handling Building
- Radwaste Building
- Emergency Power Supply Building
- Reactor Service Building
- Nuclear Service Water Cooling Tower
- Spent Fuel Cask Storage Area

The reactor building is a rectangular, reinforced concrete structure and is designed to function as a confinement boundary. It houses the NSSS and BONP components. The top of the reactor building mat is located at grade elevation. The equipment layout for this building is shown in Figures 2-6 through 2-10. The reactor building structure, in conjunction with the confinement ventilation system, provides a leakage barrier to limit fission product release to the atmosphere during normal operating and accident conditions.

Sufficient area has been provided around the PCRV to provide clearance for initial wire winding and rewinding of the circumferential prestressing tendons.

A bridge crane is provided to service the PCRV and fuel handling components. A 43-foot clear space above the refueling floor permits maintenance of the fuel handling machine, main circulators, and auxiliary service cask. A reactor equipment service facility with access from the refueling floor utilizing the overhead crane has been provided. Removable hatches will be provided in the roof to facilitate removal of the steam generators, using a mobile crane parked adjacent to the structure.

A gas-tight passage through the mat connects the core cavity and the temporary fuel storage facility located at grade elevation in the reactor building, permitting the transfer of new and spent fuel in a helium environment. Spent fuel is transferred from the temporary fuel storage facility and, subsequently, loaded into storage casks and moved into the fuel and cask handling building. From there, the spent fuel storage casks are moved to the spent fuel storage area using a mobile transporter.

The reactor building structure also houses the heat transfer equipment and flue for the natural-draft, air-cooled, heat exchangers.

The control building, radwaste building, reactor service building, and emergency power supply building are identical to those described previously.

The control building houses the control and electrical equipment required for plant operation, which includes the main control room, Class 1E switchgear, computer area, and battery room. A remote shutdown area, which is isolated from the control room and reactor building, is provided. The control building is a reinforced concrete structure and is located adjacent to the reactor building.

The radwaste building houses the equipment and systems to receive and process all radioactive waste generation at the plant, with the exception of the spent fuel. The building is a single-story, composite steel and concrete structure. The areas in which radioactive wastes are stored are surrounded by concrete to provide radiation protection. Other areas are constructed with steel framing and metal siding.

The reactor service building is adjacent to the reactor and control buildings and houses the technical support center, a "hot" machine shop including an ultrasonic cleaning room, health physics, lockers, change rooms, and plant management and technical support offices.

The emergency power supply building houses the two gas turbines and associated auxiliary and control equipment. This equipment is housed in a reinforced concrete structure. Separate buried fuel storage tanks have been provided for each system.

14884-DM-201D-1

NOTES:
1. SCALE: 1"=100'-0"

FIGURE 2-1

NOT TO BE USED FOR CONSTRUCTION

PLOT PLAN
1170 MW(t) CONTMT CONCEPT

PCRV CONCEPT
GAS COOLED REACTOR ASSOCIATES
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS

DRAWING NUMBER 14884-DM-201D-1

NO.	DESCRIPTION	CHG	DATE	NO.	DESCRIPTION	CHG	DATE	NO.	DESCRIPTION	CHG	DATE	NO.	DESCRIPTION	CHG	DATE
1				2				3				4			

REACTOR BUILDING			EQUIPMENT		REMARKS	
NO	DESCRIPTION	GROUP	NO	DESCRIPTION	GROUP	REMARKS
1	AUXILIARY HEAT EXCHANGER					
2	AUXILIARY HEAT EXCHANGER					
3	AUXILIARY HEAT EXCHANGER					
4	STEAM GENERATOR					
5	STEAM GENERATOR					
6						
7						
8						
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10	HEAT EXCHANGER					
11	HEAT EXCHANGER					
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22	AUXILIARY CIRCULATOR					
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26	HEAT EXCHANGER					
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REACTOR BUILDING			
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140	REACTOR BUILDING		

FIGURE 2-2

NOT TO BE USED FOR CONSTRUCTION

GENERAL ARRANGEMENT-REACTOR
BUILDING-EQUIPMENT LIST
CONTAINMENT CONCEPT 1170MMW

**PCRV CONCEPT
GAS COOLED REACTOR ASSOCIATES
SLOW & WHEELER ENGINEERING CORPORATION**
CHICAGO, ILL.

ORDERING INFORMATION
14884-EM-602U-1

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<p>CHARGE NO 14864 18</p> <p>CHARGE NO 14864 18</p>

NOT TO BE USED FOR CONSTRUCTION
GENERAL ARRANGEMENT REACTOR
BUILDING - PLANS
CONTAINMENT CONCEPT 1170MW(I)

**PCR CONCEPT
GAS COOLED REACTOR ASSOCIATES**
STONE & WEBSTER ENGINEERING CORPORATION

14 884-EM-602R-I

[illegible]

NOTES
1 SCALE 1"=10'
2 REFERENCE DRAWINGS

C-C
(FM-602R LK-5 LK-10)
(D-A)

PLAN EL 207-10"

FIGURE 2-4

NOT TO BE USED FOR CONSTRUCTION
GENERAL ARRANGEMENT - REACTOR
BUILDING - PLAN EL 207-10
CONTAINMENT CONCEPT 1170MW(1)

PCRV CONCEPT
GAS COOLED REACTOR ASSOCIATES
STONE & WHEELER ENGINEERING CORPORATION
BOSTON, MASS.
14 884-EM-6025-1
DESIGNED
BY

[illegible]

FIGURE 2-5

NOT TO BE USED FOR CONSTRUCTION
GENERAL ARRANGEMENT - REACTOR
BUILDING - SECTION A-A & B-B
CONTAINMENT CONCEPT (170MWU)

PERY CONCEPT
GAS COOLED REACTOR ASSOCIATES
ENGINEERING & CONSTRUCTION CORPORATION

14884-EM-602T-1

NO.	REVISION	DATE	BY	CHKD	APPD	DESCRIPTION
1						THIS DRAWING SUPERSEDES 14884-03-EM-602D-1
2						CHARGE NO 14884-03-EM-602D-1

PLAN
EL 124'-9"
REACTOR BLDG
1260MW(1) ANNULAR CORE

PLAN
EL 100'-6"
REACTOR BLDG
1260MW(1) ANNULAR CORE

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

REFERENCE DRAWINGS:

PLOT PLAN _____ D4-2018
GENERAL ARRANGEMENT REACTOR BUILDING
PLAN EL 100'-6" 124'-9" _____ E4-802L
PLAN EL 149'-0" 173'-3" _____ E4-802M
PLAN EL 197'-0" 221'-0" (377'-0") _____ E4-802N
SECTIONS 1-1 & 2-2 _____ E4-802P
EQUIPMENT LIST _____ E4-802Q

FIGURE 2-6
NOT TO BE USED FOR CONSTRUCTION

GENERAL ARRANGEMENT-REACTOR
BUILDING-PLAN-EL 100'-6" & 124'-9"
1260MW(1)-ANNULAR CORE

PCRV CONCEPT
GAS COOLED REACTOR ASSOCIATES
ATW & BENTON ENGINEERING CORPORATION
Project No. 14884-EM-602L-1

14884-EM-602L-1

NOTES:
1. SCALE: 1/4" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS: EM-6021

FIGURE 2-7

NOT TO BE USED FOR CONSTRUCTION
GENERAL ARRANGEMENT-REACTOR
BUILDING-PLAN-EL 149'-0" & 173'-3"
1260MW(1)-ANNULAR CORE
RCRV CONCEPT
GAS COOLED REACTOR ASSOCIATES
ATOMIC ENERGY RESEARCH ESTABLISHMENT
14884-EM-602M-1

NO.	REV.	DATE	BY	CHKD.	APP'D.	DESCRIPTION
1						ISSUED FOR CONSTRUCTION
2						REVISION
3						REVISION
4						REVISION
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PART PLAN
EL 277'-0"
REACTOR BLDG
1260 MW(t) ANNULAR CORE

NOTES: 1. 1" = 1'-0"
2. GENERAL NOTES & REFERENCE DWGS

FIGURE 2-8

NOT TO BE USED FOR CONSTRUCTION

GENERAL ARRANGEMENT-REACTOR
BUILDING-PLAN-EL 1976, 221-11, 277-0
1260 MW(t) - ANNULAR CORE

**PCRv CONCEPT
GAS COOLED REACTOR ASSOCIATES**

14884-EM-602N-1
SPECIALTY
EQUIPMENT
GROUP, INC.

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1-1
(EM-8011P-3,1,3)
(EM-8011P-3,1,3)

2-2
(EM-8011P-3,1,3)
(EM-8011P-3,1,3)

NOTES:
1 SCALE: 1/4"=1'-0"
2 GENERAL NOTES & REFERENCE DMS EM-601L

FIGURE 2-9

NOT TO BE USED FOR CONSTRUCTION
GENERAL ARRANGEMENTS REACTOR
BUILDING - SECTIONS 1-1 & 2-2
1260MW(1)-ANNULAR CORE
PCRV CONCEPT
GAS COOLED REACTOR ASSOCIATES
REACTOR & WASTE REPROCESSING SYSTEMS
14884-EM-602P-1

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REACTOR BUILDING			
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SECTION 3

DESCRIPTION OF REACTOR PLANT SYSTEMS

3.1 NUCLEAR STEAM SUPPLY SYSTEM

The HTGR power plant is characterized by a graphite-moderated, helium-cooled thermal reactor core within a multicavity PCRV. Table 3-1 outlines the major plant parameters for each reactor concept.

The fuel, concentrated in rod form, is located in vertical holes in hexagonal, cross-section, graphite elements. These graphite elements include vertical, full-length channels for coolant flow. Some of them have channels for insertion of reactivity control devices.

The 1170 MWt Nuclear Steam Supply System (NSSS) utilized for the containment and confinement concepts uses a reactor which has a cylindrical, prismatic-block core which is assembled from hexagonal, cross-section, graphite elements stacked eight high to form columns. A refueling region consists typically of seven columns clustered around a control rod location. There are 37 such regions, evenly spaced in a triangular array. Core power density is 5.8 W/cc.

The 1260 MWt NSSS utilized for the confinement concept uses a reactor which has an annular, prismatic core with hexagonal-shaped fuel elements. The fuel elements are stacked in columns. Six fuel element columns clustered around a central control rod column constitute a refueling region. There are 54 refueling regions. The active core is 10 elements high. Core power density is 3.9 W/cc.

Replaceable reflector blocks are positioned around the active core. These are solid graphite, hexagonal blocks of same size as the fuel elements.

Removal and replacement of the fuel block and replaceable reflector elements are accomplished from above by fuel handling system equipment. Some of this

equipment moves on a refueling structure in the upper plenum. This in-vessel refueling structure is in use only when the reactor is shut down and depressurized.

The core and reactivity control equipment, primary coolant, primary coolant circulators, steam generators, and decay heat removal equipment are all contained in a prestressed concrete reactor vessel (PCRV). The arrangement of this equipment in the PCRV is shown in Figure 3-1 for the 1170 MWt NSSS used with a containment, Figures 3-2A and B for the 1170 MWt NSSS used with a confinement, and Figures 3-3A and B for the 1260 MWt NSSS used with a confinement.

The PCRV for all reactor concepts is a vertical cylinder of varying dimensions. The core is located in a central cylindrical cavity, which is offset from the vertical centerline of the PCRV. On one side of the core cavity are two (1170) or four (1260) vertical cavities, each containing the steam generator and helium circulator for the primary coolant loops. The auxiliary heat exchanger and circulator (if used) are housed in cavities located on the other side of the core cavity. Horizontal prestressing is applied to the PCRV by circumferential wire winding, and vertical prestressing 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 (1130 psig), and it has a water-cooled steel liner with a thermal barrier on the gas-side surfaces.

The 1170 MWt NSSS used for the containment and confinement concepts has two heat transport loops comprising a steam generator, which is a once-through unit with helical coils in the economizer/evaporator section and straight tubes in the superheater section, and a main helium circulator which is a single-stage, water-lubricated, centrifugal compressor with variable-speed electric motor drive. Each circulator is provided with a shutoff valve to permit the reactor to operate with the heat transport loop out of service.

The 1260 MWt NSSS used for the confinement concept has four heat transport loops comprising a steam generator, which is a once-through unit with

helical coils and utilizes "down-hill" boiling characteristics and a main helium circulation which is a water-lubricated centrifugal compressor with variable-speed electric motor drive. Each circulator is provided with a shutoff valve to permit the reactor to operate with the heat transport loop out of service.

In addition to the two primary coolant loops, two auxiliary heat removal loops for the 1260 MWt NSSS and three auxiliary heat removal loops for the 1170 MWt NSSS are provided.

Forced-circulation loops in the 1260 and 1170 NSSS each contain a water-cooled heat exchanger, a helium circulator, and a helium shutoff valve. The natural-circulation loop in the 1170 NSSS for a confinement contains a water-cooled heat exchanger and a helium shutoff valve.

For a detailed discussion of decay heat removal methods see Section 3.3.

3.2 POWER CONVERSION CYCLE

The power conversion cycle for all three plant concepts is identical. An overall cycle diagram is shown in Figure 3-4. However, plant performance varies due to the different thermal ratings of the core. Key power conversion cycle parameters are presented in Table 3-2.

3.3 DECAY HEAT REMOVAL

The functional requirement of the decay heat removal system is to remove residual heat from the reactor core when the ability to cool the core with the heat transport loops is lost. This system is a backup system for the removal of decay heat from the reactor. It provides investment protection and controls radio-nuclide release from the core by limiting the temperature of the fuel and other components within the PCRV to acceptable values.

The approach to decay heat removal varies considerably between the three reactor plant concepts. Table 3-3 summarizes the methods used.

The decay heat removal method used by the reference 1170 NSSS with containment design uses forced circulation and consists of three separate, independent, and functionally identical, 50-percent capacity, cooling loops, as shown in Figure 3-5. Each loop comprises three heat transfer circuits: the primary coolant circuit, the water circuit, and the air circuit. Forced circulation is utilized in each circuit, which requires a reliable on-site power source. Helium is circulated through the core by the auxiliary circulator. The hot helium is then circulated past an auxiliary heat exchanger where the heat is transferred to a secondary coolant. The secondary coolant, which is pressurized water, is transported by a circulating pump to an air-cooled heat exchanger where heat is ultimately rejected to the atmosphere by fan-forced air flow over an extend surface heat exchanger. All equipment is designed and built to nuclear standards. The air-cooled heat exchangers are located in a concrete structure for tornado missile protection.

Each of the forced-circulation loops are capable of cooling the core during refueling, depressurized, and pressurized shutdown outages. When not operating in these modes, the forced-circulation loops operate in a standby mode, resulting in a small heat loss.

The decay heat removal method adopted by the 1170 NSSS with confinement consists of three separate and independent loops utilizing forced and natural circulation, as shown in Figure 3-6. Two of the loops are identical and are of the same design as those reported above for the 1170 NSSS with containment; however, each is sized for 100-percent decay heat removal capacity. The third loop is a dedicated natural-circulation loop which comprises three heat transfer circuits and is sized for 100-percent capacity decay heat removal with the reactor pressurized. The natural-circulation loop removes residual heat from the reactor core when neither the heat transport system loops nor the forced-circulation decay heat removal loops are available for this function. The natural-circulation loop is a dedicated backup heat removal system linking the core to the ultimate heat sink, using a natural-circulation chain of primary, secondary, and tertiary coolant flows. Pressurized primary coolant is required for the natural-

circulation loop to function effectively. This system provides the core cooling function to meet the investment protection goal. The secondary coolant piping, the natural-circulation air-cooled heat exchanger, and the associated natural-draft flues are designed and built to industrial construction standards. The components of the forced-circulation decay heat removal system are designed and built to nuclear construction standards. The air-cooled heat exchangers are located in a concrete structure for tornado missile protection.

The decay heat removal method adopted for 1260 NSSS with confinement consists of four independent loops utilizing forced and natural circulation, as shown in Figure 3-7. Two of the loops provide decay heat removal by forced circulation. These loops are identical and are of the same design as those reported above for the 1170 NSSS with containment. The other two loops constitute a decay heat removal method using the liner cooling system. Each loop can operate in a forced- or a natural-circulation mode, thus providing additional, independent, and diverse methods of decay heat removal.

Forced-circulation decay heat removal is provided to meet plant investment goals. Its function is to remove the decay heat from the core when the main loops are not available and to improve maintenance flexibility.

Forced-circulation decay heat removal is capable of cooling the core during shutdown outages when the reactor is pressurized and during refueling and maintenance when the reactor is depressurized. To satisfy the plant investment protection goals, each of the two shutdown cooling system loops removes 100 percent of the decay heat when the reactor is pressurized and 50 percent when the reactor is depressurized.

The components of the forced-circulation decay heat removal system that form a part of the primary coolant boundary are designed and built to nuclear construction standards. All other components, principally those that comprise the secondary heat transfer circuit, are designed and built to

industrial construction standards. In addition to the forced-circulation decay heat removal system, the liner cooling system is capable of providing decay heat removal from the core. The functional requirements for the liner cooling system are to maintain PCRV concrete temperature within design limits during normal operation and to remove decay heat from the reactor core in the event that the heat transport system and the forced-circulation decay heat removal system cannot provide this function.

Liner cooling decay heat removal is performed under depressurized reactor conditions or following an intentional depressurization to minimize convective heat transport of hot gases in the reactor system. The core decay heat is transported to the liner cooling system by heat conduction and radiation through a thin thermal barrier section in the core cavity.

The liner cooling system consists of two redundant, independent, cooling loops. Each loop is capable of removing 100 percent of the normal operation heat load or the decay heat load by forced circulation; both loops are needed to cool the liner and concrete by natural circulation (passive mode) during abnormal situations where loss of forced circulation has occurred.

During normal operations, cooling is accomplished by forced circulation of water around the liner cooling system loops, each of which is operating at 50-percent capacity. Heat removed from the liner and adjacent concrete is dissipated to the station cooling water system via a heat exchanger.

If loss of forced cooling in the heat transport system, shutdown cooling system, and liner cooling system has occurred, decay heat removal (depressurized mode) is accomplished by natural circulation in both liner cooling loops. The heat transferred from the liner and adjacent concrete to the water in the liner cooling tube system is transported by natural circulation to the natural-draft, air-cooled heat exchanger, where it is cooled. The cooled water is returned to the liner cooling tubes, thus completing a loop and creating natural circulation. The heat rejected from the natural-draft, air-cooled heat exchanger is discharged to the environment.

All components of the liner and cooling system that form a part of the primary coolant boundary standard, or are required to achieve natural-circulation decay heat removal, are designed and built to nuclear construction standards. All other components, principally those associated with forced-circulation heat removal, are designed and built to industrial construction standards.

3.4 REACTOR AUXILIARY SYSTEMS

The reactor auxiliary systems are generally similar for all concepts.

The reactor service equipment is identical for all concepts, except for the addition of a cask transporter for the fuel handling machine and auxiliary service cask for the HTGR concept with a containment.

The helium purification system is identical for all concepts except for the intentional depressurization capability which is provided for the 1260 NSSS.

The PCRV service system is identical for all concepts.

The helium storage and transfer system is identical for all concepts.

The cryogenic refrigeration system is identical for all concepts.

The reactor vessel liner cooling system differs for the 1260 NSSS in that it provides a decay heat removal function and has capability for natural-circulation heat transport modes. These differences have an inconsequential effect on the plant layout and have been included in the differential cost estimate.

The nuclear service water system and the reactor building closed-loop cooling water system differ for each of the three concepts. The differences are primarily associated with the components served and their functional requirements to meet Goal 1, and in the selection of the equipment used for the ultimate heat sink. The 1170 NSSS concepts use a missile-protected,

wet, mechanical-draft cooling tower and the 1260 NSSS concept uses a missile-protected, evaporative-cooled heat exchanger. These differences have a minor influence on plant layout and equipment arrangement and have been included in the differential cost estimate.

The radioactive waste processing system is identical for all concepts.

The helium purge and vent system is identical for all concepts.

3.5 FUEL HANDLING AND STORAGE

All three concepts utilize the same fuel handling and storage system. However, the 1260 NSSS requires additional spent fuel storage capacity due to larger spent fuel volumetric discharge rate due to the lower power density and due to the larger refueling fraction, approximately one-third core versus one-fourth core for the 1170 NSSS. These differences have been reflected in the building sizes and layouts and are included in the differential cost estimate.

3.6 INSTRUMENTATION AND CONTROL SYSTEM

The 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, 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 shutdown condition.

There is no basic difference in the plant's instrument and control systems for a confinement or a containment for all concepts.

3.7 PLANT ELECTRICAL SYSTEM

The plant electrical system transfers the electrical power generated by the plant to the high-voltage switchyard through the unit transformer, controls and meters electrical energy, and protects the power carrying components. It is the source of electrical power for the plant auxiliaries, including the plant control, protection, and surveillance equipment and the engineered safety features equipment.

In comparing a containment to a confinement, all of the subsystems are basically the same except plant cabling. Electrical cable entering a containment requires cable penetrations that must meet stringent environmental, mechanical, and leakage requirements, while similar penetration for a confinement have less stringent requirements. A fire stop, such as a Multi-Transit System, can be used for cables entering a confinement.

Furthermore, the concepts differ significantly in their requirements for standby power. The 1170 Ref NSSS with containment requires three standby power supplies, each rated at 2800 kW; the 1170 NSSS with confinement or with a containment requires two standby power supplies, each rated at 2800 kW; and the 1260 NSSS with a confinement requires two standby power supplies, each rated at 800 kW.

3.8 PLANT STRUCTURES

The reactor containment and auxiliary building for the 1170 NSSS share a common mat 63,100 mm (207 feet) by 45,700 mm (150 feet). Except for the grade level (i.e., the common mat) where the auxiliary building completely surrounds the containment, the containment is enclosed for approximately 60 percent of its circumference by the auxiliary building.

The reactor containment is a steel-lined, reinforced-concrete, circular cylinder capped by a hemispherical dome. The outside diameter of the cylinder is 38,100 mm (125 feet), and the concrete thicknesses are 1,500 mm (5 feet) for the walls and 760 mm (2.5 feet) for the dome. The height from top of the mat to the spring line of the hemispherical dome is approximately 50,300 mm (165 feet), and the overall containment height is 69,200 mm (227 feet). An equipment hatch, 6,700 mm (22 feet) in diameter, and a personnel hatch, 7 feet in diameter, exit into the auxiliary building from the containment building. The wall of the reactor containment has been thickened in the vicinity of these openings to provide additional strength to resist the tension forces resulting from the pressure due to the design basis events.

The walls and dome of the containment are reinforced with No. 18 reinforcing bars manufactured of A615 Grade 60 steel. Typically, the spacing of the vertical reinforcement is 300 mm (1 foot) center-to-center (measured horizontally along the wall center line) on the inside and outside faces, and the spacing of the circumferential reinforcement is 2 bars at 300 mm (1 foot) center-to-center (measured vertically) on the inside face and on the outside face. In addition, diagonal bars are provided at the outside face of the wall to resist seismic shear loads. The containment and auxiliary building mat is also reinforced with several layers of No. 18 bars. Both the containment and mat are constructed of concrete which has a 90-day strength of 34.5 MPa (5000 psi).

The design bases of the containment include both slow and rapid PCRV depressurization events, as well as seismic loads, tornado wind and missile loads, and others described in NRC Standard Review Plan 3.8.1. The design pressure for the containment is approximately 0.41 MPa (60 psi) and is based upon preliminary studies using data taken from the Fulton Station design for Philadelphia Electric Company.

A steel structure within the containment is located in the annular space between the PCRV and the containment wall. This structure has concrete slab decks at 4 elevations above the mat to support equipment housed within the

containment. In addition, the structure supports piping, HVAC ducts, and cable tray and conduit. This annular steel structure is subjected to seismic, thermal, dead and live loads, and extreme loads, including pipe rupture restraint forces and jet impingement loads. It is braced from the containment (to permit the transfer of seismic loads) with snubbers which will not permit large compressive stresses to occur in the structure during thermal heating due to the depressurization design events. The annular structure is constructed of A36 steel.

The auxiliary building is a five-story, reinforced-concrete structure housing all or parts of the following systems:

- Reactor Service System
- Core Auxiliary Cooling Water System
- Helium Purification System
- Fuel Handling and Refueling System
- Air Purification Units
- Main Steam and Feedwater Piping Isolation Valve Area
- Control Rod Storage
- Equipment Decontamination

All exterior walls and roof slabs are 610 mm (2 feet) thick to provide tornado missile protection for all safety-related components and systems housed within the building. All interior floor slabs are 1 foot thick (except where greater thickness is required for radiation protection) and are cast on sheet metal decking which is supported, in part, by structural steel framing.

The auxiliary building is designed to meet the requirements described in NRC Standard Review Plan 3.8.4. A shake space 152 mm (6 in.) wide is provided between the containment and the auxiliary building.

The reactor building for the 1170 NSSS and 1260 NSSS with confinement is a large, single-volume, reinforced-concrete structure with a 31.75 mm (1.25 in.) thick steel plate roof supported by steel trusses, which houses the PCRV and related reactor auxiliary equipment.

It is designed to withstand a safe shutdown earthquake (SSE) with a maximum horizontal ground acceleration of 0.3 g, as well as a design wind velocity of 177 km/hr (110 mph). During a tornado, the reactor building will maintain an operating environment and protect all safety-related components and systems from storm-generated missiles. Overpressure resulting from either tornado pressure drop or a design basis depressurization accident or pipe breaks is controlled with louvers and limited to 0.17 Bar (2.5 psig). Where required by site conditions, the reactor building is designed to resist the buoyancy forces and prevent water ingress from flooding. The reactor building houses all or parts of the following systems:

- The PCRV and Nuclear Steam Supply System Components
- Reactor Service System
- Auxiliary Cooling Water System
- Helium Purification System
- Fuel Handling and Fuel Storage Systems
- Air Purification Units
- Main Steam and Feedwater Piping and Valve Areas

- Control Rod Storage
- Hot Cell Facilities

The reactor building is 48,700 mm (160 feet) by 47,200 mm (155 feet) in plan and 175 feet high.

Outside of building dimensions, the reactor building for the 1260 NSSS is similar to that described above for the 1170 NSSS. The reactor building for the 1260 NSSS is 42,700 mm (140 feet) by 52,400 mm (172 feet) in plan and 200 feet in height.

Table 3-1

MAJOR PLANT PARAMETERS

<u>REACTOR</u>	<u>1170 NSSS</u>	<u>1170 REF NSSS</u>	<u>1260 NSSS</u>
Reactor rating	1170 MWt	1170 MWt	1260 MWt
Reactor enclosure	Confinement or Containment	Containment	Confinement
Reactor systems/no. of loops	1/2	1/2	1/4
Plant design life	40 yr	40 yr	40 yr
Design basis operation	80 percent capacity factor	80 percent capacity factor	80 percent capacity factor
NSSS			
Pressure vessel/dimensions	PCRV (87'-0" diam x 92'-6" high)	PCRV (82'-0" diam x 92'-0" high)	PCRV (94'-6" diam x 105'-9" high)
Fuel	LEU/Th	LEU/Th	LEU/Th
Fuel element	Prism	Prism	Prism
Core configuration	Cylindrical	Cylindrical	Annular
Decay heat removal	2 forced, natural heat removal	3 forced heat removal	2 forced heat removal plus forced and natural liner cooling
Helium pressure	72.3 Bar (1050 psia)	72.3 Bar (1050 psia)	72.3 Bar (1050 psia)
Refueling regions	37	37	54
Control rods	37 pairs	37 pairs	54 pairs
Reserve shutdown systems	37	37	54
Core power density	5.8 W/cc	5.8 W/cc	3.9 W/cc
Core inlet temperature	318°C (605°F)	318°C (605°F)	316°C (601°F)
Core outlet temperature	691°C (1276°F)	691°C (1276°F)	689°C (1272°F)
Refueling method	In-vessel by patch region	In-vessel by patch region	In-vessel by patch region
Steam generators	2/uphill boiling	2/uphill boiling	4/downhill boiling
Feedwater temperature/pressure	221°C (430°F)/236.5 Bar (3432 psia)	221°C (430°F)/236.5 Bar (3432 psia)	221°C (430°F)/206.7 Bar (3000 psia)
Steam temperature/pressure	541°C (1005°F)/173.3 Bar (2515 psia)	541°C (1005°F)/173.3 Bar (2515 psia)	541°C (1005°F)/173.3 Bar (2515 psia)
Circulators	2/electric driven	2/electric driven	4/electric driven

Table 3-2

KEY CYCLE PARAMETERS

Item		Parameter	
Core Power	MWt	1,170.	1,260.
Gross Power to Turbine	MWt	1,183.	1,266.
Turbine Exh Press	cm Hg	5.6	6.0
	(in. Hg)	(2.22)	(2.37)
Total Generation	kW	489,000.	520,000.
Auxiliary Power	kW	49,000.	45,000.
Feed & Cond Pump		16,900.	15,600.
Circulator		22,400.	19,000.
House Loads		9,700.	10,400.
Net Generation	kW	440,000.	475,000.
Gross Turbine Heat Rate	kcal/kWh	2.1	2.1
	(Btu/kWhr)	(8,260)	(8,302)
Net Station Heat Rate	kcal/kWh	2.3	2.3
	(Btu/kWhr)	(9,078)	(9,056)
Cycle Efficiency	%	37.6	37.7

Table 3-3

DECAY HEAT REMOVAL METHODS

Reactor Concept	Decay Heat Removal Method	Design Standards	Schematic
1. 1170 MWt HTGR Plant Reference Design	a. Auxiliary Cooling Forced Circulation	Nuclear	Fig. 3-5
2. 1170 MWt HTGR Plant Cylindrical Core Concept	a. Auxiliary Cooling Forced Circulation	Nuclear	Fig. 3-6
	b. Auxiliary Cooling Natural Circulation	Industrial	Fig. 3-6
3. 1260 MWt HTGR Plant Annular Core Concept	a. Shutdown Cooling System Forced Circulation	Industrial	Fig. 3-7
	b. Liner Cooling System Forced Circulation	Industrial	Fig. 3-7
	c. Natural Circulation Liner Cooling System	Nuclear	Fig. 3-7

[illegible]

FIGURE 3-2A PCRV TOP HEAD PLAN

- 1 THE PDRV IS DESIGNED TO THE REQUIREMENTS OF ASME BOLLER AND PRESSURE VESSEL CODE, SECTION II, DIV 2 - REF SA DWS 027784
- 2 MAXIMUM CAVITY PRESSURE FOR PDRV AND PENETRATION DESIGN = 130 PSIG

▶ CUTOUTS AND DIMENSIONS SHOWN IN THE PDRV SUPPORT RING AND STEAM GENERATOR PPE ROUTING AT THE BOTTOM OF PDRV MAY BE RELOCATED TO SUIT BOP DESIGN AND ACCESS.

- REFUELING PENETRATIONS - 84
- COLLUM SUPPORT FOR FHM AND ASC - 50
- PLENUM TRANSPORTER PENETRATIONS, PLUS COLLUM SUPPORTS FOR FHM AND ASC - 8
- HELIUM PURIFICATION HIGH TEMPERATURE ADSORBER LOCATIONS - 2
- HELIUM PURIFICATION HIGH TEMPERATURE FILTER LOCATIONS - 2

PLAN VIEW OF PCRV & NSSS COMPONENTS

PRELIMINARY

NSSS ARRANGEMENT PLAN VIEW

TITLE		PART NO.		DESCRIPTION		MATERIAL SPEC.	
PARTS LIST							
QUANTITY		UNIT		GA Technologies Inc. SAN DIEGO, CALIFORNIA			
END USER INTER-SEC. PROJ. PLANT				☐ NEW ☐ DUPL ☐ DESIGN			
REQ. QTY		SISTEM QTY		PROJECT NO.		DATE	
				042		12-30	
PRESS OF OTHERS SPEC. FOR THE DIMENSIONS ARE IN INCHES UNLESS OTHERWISE SPECIFIED DATE: 04-10-80 BY: 1-10-80				TITLE NSSS ARRANGEMENT 1260 MW(1) PCRV & COMPONENTS			
TYPICAL SECTION ANGLE 1/8"				1260 MW(1) PCRV & COMPONENTS E 32334 028028			
TYPICAL SURFACE 1/8"				1260 MW(1) PCRV & COMPONENTS E 32334 028028

FIGURE 3-3B

THIS DIMENSION MAY BE
RESOLVED BY USE OF
TRENCHES OR PITS FOR
REMOVAL AND ACCESS
TO TUBE SHEET (STEAM
OF STEAM GENERATOR

SECTION A-A

FIGURE 3-4 NSSS HEAT BALANCE DIAGRAM

1. MAIN LOOP HEAT
TRANSPORT SYSTEM (HTS)
FOUR 100% LOOPS
PRESSURIZED OR DEPRESSURIZED
PRIMARY-FORCED CIRCULATION
SECONDARY-FORCED CIRCULATION

2. AUXILIARY HEAT REMOVAL
SYSTEM
THREE 50% LOOPS, PRESSURIZED
OR DEPRESSURIZED
PRIMARY-FORCED CIRCULATION
SECONDARY-FORCED CIRCULATION

FIGURE 3-5

DECAY HEAT REMOVAL SYSTEMS
REFERENCE 1170 NSSS

1. MAIN LOOP HEAT TRANSPORT SYSTEM (HTS)
FOUR 100% LOOPS
PRESSURIZED OR DEPRESSURIZED
PRIMARY-FORCED CIRCULATION
SECONDARY-FORCED CIRCULATION

2. AUXILIARY HEAT REMOVAL SYSTEM
ONE 100% LOOP PRESSURIZED
PRIMARY-NATURAL CIRCULATION
SECONDARY-NATURAL CIRCULATION

3. AUXILIARY HEAT REMOVAL SYSTEM (SCS)
TWO 100% LOOPS, PRESSURIZED OR DEPRESSURIZED
PRIMARY-FORCED CIRCULATION
SECONDARY-FORCED CIRCULATION

FIGURE 3-6

DECAY HEAT REMOVAL SYSTEMS 1170 NSSS

1 MAIN LOOP HEAT
TRANSPORT SYSTEM (HTS)
FOUR 100% LOOPS
PRESSURIZED OR DEPRESSURIZED
PRIMARY-FORCED CIRCULATION
SECONDARY-FORCED CIRCULATION

2 SHUTDOWN COOLING
SYSTEM (SCS)
TWO 50% LOOPS PRESSURIZED OR
DEPRESSURIZED REFUELING AND
MAINTENANCE
PRIMARY-FORCED CIRCULATION
SECONDARY-FORCED CIRCULATION

3. LINER COOLING
SYSTEM (LCS)
TWO 100% LOOPS, DEPRESSURIZED
PRIMARY-CONDUCTION/RADIATION
SECONDARY-FORCED CIRCULATION
OR NATURAL CIRCULATION

FIGURE 3-7

DECAY HEAT REMOVAL SYSTEMS 1260 NSSS

SECTION 4

DOSE ANALYSIS

Dose analyses for three different cases of HTGR nuclear power plant designs were performed to determine radiological consequences with respect to the utility/user design requirement that no provisions are required for evacuation or sheltering of the public outside the plant exclusion area boundary, i.e., doses less than Protective Action Guideline (PAG) limits of 5 rem thyroid, 1 rem gamma-whole body.

Dose analyses of both the 1170 NSSS with confinement and containment designs for two hypothetical radiological release events were performed. The confinement design meets the PAG dose limits for Event 1 (rapid depressurization), but not for Event 2 (loss of core cooling). When a PWR type containment is used, and taking additional credit for source term reducing fission product physical retention mechanisms, dose limits are met for both events.

The Event 1 doses at the exclusion area boundary (EAB) of 425 m are 1.9 rem thyroid and 0.086 rem gamma-whole body, based on a controlled inleakage confinement design with no dedicated radiological elevated release point (stack). These doses are within the PAG limits.

However, for Event 2 with a 1.5 percent vol/day outleak rate from confinement and a ground level χ/Q , and no credit for natural fission product retention mechanisms other than radioactive decay, the thyroid dose at the EAB is 1776 rem. By taking credit for natural fission product retention mechanisms, in addition to radioactive decay, the thyroid dose is reduced to 61.0 rem. Neither case meets the PAG limits.

The doses with a PWR type containment were investigated for further reduction. For Event 1, with a containment leak rate of 0.1 percent vol/day, the thyroid dose is 0.0095 rem and the gamma-whole body dose is 2.20×10^{-4} rem. For Event 2, with a containment leak rate of 0.06 percent vol/day as determined by an accident specific thermal hydraulic analysis and taking credit for fission product natural retention as discussed above, the thyroid dose is 2.5 rem and the gamma-whole body dose is 0.0075 rem. These doses are well within the PAG limits.

Note that the radiological consequences for containment are the same for the 1170 NSSS and the 1170 Reference NSSS.

Dose analysis of the 1260 NSSS for the two hypothetical radiological release events shows that the confinement design meets the PAG dose limits for both Event 1 and Event 2.

The Event 1 doses at the EAB of 425 m are 1.3 rem thyroid and 0.34 rem gamma-whole body, based on a controlled inleakage confinement design with no dedicated radiological elevated release point (stack). These doses meet the PAG limits.

For Event 2 with a 1.5 percent vol/day outleak rate from confinement, and a ground level χ/Q , the thyroid dose at the EAB is 1.6 rem, and the gamma-whole body dose is 0.04 rem. These doses are well within the PAG limits.

It should be noted that by also taking credit for natural fission product depletion, in addition to radioactive decay, a 180 percent vol/day outleak rate from confinement would result in doses less than 5 rem thyroid, and 1 rem gamma-whole body.

In summary, the 1170 NSSS confinement design does not meet the utility/user design requirement of no evacuation or sheltering required during accidents. The 1170 NSSS containment design does meet the safety design requirements, when additional credit for source term reducing fission product physical retention mechanism is taken.

The 1260 NSSS confinement design does meet the safety design requirement without having to take credit for source term reduction mechanism.

SECTION 5

CONSTRUCTION SCHEDULE

This section presents the assumptions and basis for the schedule development, describes the construction sequence, and identifies and evaluates the critical path for the 1170 NSSS plant with containment, and presents a schedule comparison between the 1170 NSSS plant with containment and the 1170 NSSS and 1260 NSSS plants with confinement. The construction schedule details and analysis for the 1170 NSSS and 1260 NSSS plants with confinement can be found in their respective Concept Evaluation Reports. The plant design development is in the preconceptual phase, and use of assumptions and generic building enveloping was necessary and will be identified accordingly. The depth of analysis and conclusions drawn will be of a commensurate nature.

5.1 ASSUMPTIONS AND BASIS

The construction schedule development is based on the following parameters and assumptions:

- Site and design are pre-licensed; one-step licensing.
- There are no regulatory or political constraints.
- Equilibrium plant design - there are no design changes after start of construction.
- There are no cash flow or manpower constraints.
- Design information and component deliveries support construction required dates.
- Soil site with all other parameters are as specified in User/Utility Design Requirements.

- Schedule is summary level with activities shown on early start basis.
- Schedule is based on segregated construction approach and alternating 4-day and 10-hour work-shift concept.
- Activities from Authorization to Issue of Construction Permit (Start of Construction at month 0) run for 30 months.
- Site clearing, grading, and installation of temporary construction facilities and services are completed during the LWA period (prior to the Start of Construction at month 0). No excavation or foundation work for permanent plant structures is carried out during this period.
- Activities from hot function through power ascension testing are as defined by GCRA.
- Building/mechanical arrangement and quantities are based on conceptual design information and drawings, except as noted.
- Schedules for water treatment, auxiliary boiler, machine shop, warehouse, and miscellaneous yard structures are taken from SWEC reference designs.

5.2 DESCRIPTION OF CONSTRUCTION SEQUENCE

The construction sequence for the buildings for the 1170 NSSS plant with containment is the traditional series/parallel construction, beginning with civil work, followed in series by structural, and then mechanical, piping, and electrical running in parallel, except where physical constraints necessitate series activities.

Initial construction efforts focus on the reactor building (RB) with mat and containment floor liner being the primary activities during the first

9 months of construction. At this point, work begins on the vessel support and PCRV, running in parallel with, but independent of, the reactor containment liner and walls. Mechanical, piping, and electrical work begins in the lower elevations of the reactor building as soon as overhead structural work permits, running in parallel with construction of the PCRV and building structure, finally dominating the last 12 months of the RB schedule. Placement of the containment dome liner and completion of the containment and reactor building exterior concrete follows setting at the NSSS components into the PCRV and is independent of critical path mechanical and electrical activities ongoing within the containment.

Work on the turbine building begins early in the construction schedule and follows traditional sequencing. Structural work and condenser erection dominate the early months of construction, while turbine erection, mechanical, piping, and electrical dominate the later months.

Separate plant structures, such as the radwaste building, emergency power supply building, and yard structures, are shown with early start dates beginning when general excavation is complete at month 6.

Buildings contiguous with the reactor building begin as soon as work on the reactor building has progressed to the point where contiguous building construction will not have an adverse impact. The control, reactor service, and fuel/cask handling buildings begin as soon as the reactor building exterior wall has progressed several lifts above grade. The control building (started at month 13) must progress aggressively as it appears on the later portion of the critical path. The reactor service building (started in month 16) follows the control and reactor building exterior wall progress and has several months of float. The fuel/cask handling building (started in month 16) also follows the reactor building exterior wall progress and has considerable schedule float.

The turbine building and its contiguous structures are all structural-steel and metal-sided buildings and, except for foundations and access preferences, may progress independently. The water treatment and non-essential

switchgear buildings follow the turbine building foundations, beginning at month 5, and progress independently of the turbine building superstructure. Both buildings have zero float and are needed to support plant pre-op testing; however, neither is on the critical path. The auxiliary boiler and warehouse/shop begin at month 11, after a substantial portion of the turbine building superstructure is complete. The auxiliary boiler is needed to support plant pre-op testing and must progress as scheduled, while the warehouse/shop has considerable float and may be adjusted to suit the constructor.

5.3 SUMMARY OF SCHEDULE

The overall project duration for the 1170 NSSS plant with containment from authorization to completion of power ascension testing is 77 months. The construction duration for the plant, based on the present conceptual design, is 47 months from start of construction to completion of power ascension testing, as shown in the Summary Construction Schedule - Figure 5-1. The key project milestones are as follows:

<u>Milestone</u>	<u>Month</u>
Project Authorization	-30
Receive Construction Permit - Start Construction	0
Start PCRV	10
Start Reactor Building Mechanical, Piping, and Electrical Installation	16
Permanent Power Available/Start Plant Pre-Op Testing	21
Complete PCRV	27
Complete Turbine-Generator	25 1/2

Hot Functional Testing	38
Load Fuel	40
Complete Plant Warranty Run (Commercial Operation)	47

The critical path for this plant design is dominated by construction of the PCRV and the reactor containment building. The critical path originates with RB excavation, then runs through containment floor liner and fill slab placement. At that point, the critical path divides with one path running through the vessel support and PCRV, then into RB mechanical and piping. The parallel critical path runs through the containment liner, penetrations and hatches, to the containment wall and reactor building structure and, finally, through containment dome placement. At that point, the two paths combine and are joined by a parallel path in the control building running through RB and CB electrical installation and checkout testing. The critical path then follows plant pre-op testing through hot functional, load fuel, and rise to power testing and commercial operation.

The balance of the plant buildings have varying amounts of float and/or non-aggressive schedules and, therefore, little or no potential for becoming critical path activities except under the most adverse circumstances.

5.4 SCHEDULE COMPARISON

The construction schedule for the 1170 NSSS plant with containment is 8 months longer than the 1170 NSSS plant with confinement (see Summary Construction Schedule - Figure 5-2) and 7 months longer than the 1260 NSSS plant with confinement (see Summary Construction Schedule - Figure 5-3). The schedule growth is primarily due to the additional time required to construct the larger and more complex nuclear island mat and place the containment floor liner and fill slab, activities which must be completed prior to starting work on the PCRV. Additional schedule growth results from the containment liner erection and exterior wall construction restraining

reactor building and containment interior structural work, thus ultimately restraining the start of mechanical, piping, and electrical installation activities (see milestone comparison below).

MILESTONE COMPARISON

<u>Activity</u>	<u>1170 With Containment</u>	<u>1170 With Confinement</u>	<u>1260 With Confinement</u>
Start Const	0	0	0
Start RB Superstructures	10	4 1/4	4 1/4
Start PCRV	10 1/4	5 1/4	5 1/4
Start Mech, PPG and Electrical	16	10 1/4	10 1/4
Complete PCRV	27	22	23 1/4
Complete RB Superstructure	33	23 3/4	24 1/2
Hot Functional	38	30	31
Complete Mech, PPG and Electrical	39	31	32
Complete Power Ascension Testing	47	39	40

The structures abutting the RB (Control Reactor Services and Fuel Buildings) start later for the plant design with containment than with confinement; however, the lateness is approximately equal to the overall schedule growth and, therefore, the degree of schedule critically remains unchanged. Only the latter portion of the control building is on the critical path as it is for the plant designs with confinements; however, this building is more sensitive to RB and containment structural progress due to the close proximity of the structures and the increased complexity that results.

The outlying nuclear structures and the non-nuclear structures have the same construction starts and durations for all three plant designs. Due to the

increased schedule duration for the plant with containment, the schedules for each of these structures will realize increased float and reduced sensitivity to design and construction anomalies.

Overall, the construction schedule for the plant design with containment is longer by 7 to 8 months and more sensitive than the schedules for the other two plant designs (with confinement) because of the increase in both design complexity and bulk commodities for critical path structures.

1. Site and Design are pre-determined, one-time decisions
2. Input and Output are established on POLYMERIZATION CONSTRAINTS
3. Reaction and Temperature are input and output changes after input of
4. Input and Output are input and output changes after input of
5. Reaction and Temperature are input and output changes after input of
6. Reaction and Temperature are input and output changes after input of
7. Reaction and Temperature are input and output changes after input of
8. Reaction and Temperature are input and output changes after input of
9. Reaction and Temperature are input and output changes after input of
10. Reaction and Temperature are input and output changes after input of

COMMERCIAL
OPERATION

FIGURE 5-1
SUMMARY CONSTRUCTION SCHEDULE
1170 MW
NUCR PLANT
PCRV CONCEPT WITH CONTAINMENT
FEBRUARY, 1983
STONE & WEBSTER ENGINEERING CORP.
BOSTON, MA

LEGEND
CRITICAL PATH

FIGURE 5-2
SUMMARY CONSTRUCTION SCHEDULE

1170 MW
HTGR PLANT
PCRV CONCEPT
FEBRUARY, 1985

STONE & WEBSTER ENGINEERING CORP.
BOSTON, MA

CONSTRUCTION SCHEDULE ASSUMPTIONS AND BASIS

- CONFINEMENT STRUCTURE FOR REACTOR BUILDING
- SITE AND DESIGN ARE PRE-LICENSED; ONE-STEP LICENSING
- THERE ARE NO REGULATORY OR POLITICAL CONSTRAINTS
- EQUILIBRIUM PLANT DESIGN - THERE ARE NO DESIGN CHANGES AFTER START OF CONSTRUCTION
- THERE ARE NO CASH FLOW OR MANPOWER CONSTRAINTS
- DESIGN INFORMATION AND COMPONENT DELIVERIES SUPPORT CONSTRUCTION REQUIRED DATES
- SOIL SITE WITH ALL OTHER PARAMETERS ARE AS SPECIFIED IN USER/UTILITY DESIGN REQUIREMENTS
- SCHEDULE IS SUPPLY LEVEL WITH ACTIVITIES BEGIN ON EARLY START BASIS
- SCHEDULE IS BASED ON SEGREGATED CONSTRUCTION APPROACH AND ALTERNATING 8-HOUR, 10-HOUR WORK SHIFT CONCEPT
- ACTIVITIES FROM AUTHORIZATION TO ISSUE OF CONSTRUCTION PERMIT (START CONSTRUCTION - MONTH 0) RUN FOR 30 MONTHS AS PER LETTER SPEC 000-0100 DATED 1/15/85
- SITE CLEARING AND GRADING AND INSTALLATION OF TEMPORARY CONSTRUCTION FACILITIES AND SERVICES ARE COMPLETE DURING THE LMA PERIOD (PRIOR TO THE START OF CONSTRUCTION AT MONTH 0). NO EXCAVATION OR FOUNDATION WORK FOR PERMANENT PLANT STRUCTURES IS CARRIED OUT DURING THIS PERIOD. (REFER TO SPEC LETTER 000-0100 DATED 1/15/85, SUBJECT, ACTION ITEM NO 4 CAPITAL COST ESTIMATE DEVELOPMENT MEETING)
- ACTIVITIES FROM NOT FUNCTIONAL THROUGH POWER ASSEMBLY TESTING ARE AS DEFINED BY CCA
- BUILDING/MECHANICAL ARRANGEMENT AND QUANTITIES ARE BASED ON SMC "CONCEPTUAL DRAWINGS AND DESIGN INFORMATION EXCEPT AS NOTED"
- *SCHEDULES FOR WATER TREATMENT, AUXILIARY BOILER, MACHINE SHOP, WAREHOUSE, AND MISCELLANEOUS YARD ARE STRUCTURES TAKEN FROM SMC REFERENCE DESIGNS

SECTION 6

DIFFERENTIAL CAPITAL COST

An order-of-magnitude differential direct cost estimate has been prepared for each concept and is summarized in Table 6-1. Table 6-2 contains the capital cost summary to the three-digit account level for each concept. The order-of-magnitude estimate represents an equilibrium plant and incorporates savings achieved as a result of increased productivity associated with an alternating 4-day and 10-hour workweek where those techniques or work schedules may be effected. All costs are in present-day, January 1985, dollars.

Indirect costs (i.e., construction services and field engineering, engineering services and fees, owner's cost, and contingency) were calculated in accordance with the GCRA Economic Ground Rules, and applied as appropriate to the nuclear and fossil portions of the plant concept. The specific indirect rates used for the plant concepts are as follows:

	<u>1170</u>	<u>1260</u>
Construction Services and Field Engineering	25%	25%
Engineering and Home Office Services and Fees	15%	15%
Owner's Cost	10%	10%
Contingency	15%	20%

Allowance for funds used during construction is based upon 3.3 percent per year, calculated in accordance with GCRA Economic Ground Rules, and reflects 39-month, 40-month, and 47-month construction schedule for the 1170 NSSS with confinement, the 1260 NSSS with confinement, and the 1170 NSSS with containment as identified in Section 5, respectively.

TABLE 6-1
DIFFERENTIAL
DIRECT COST DETAIL
CONTAINMENT VS CONFINEMENT TRADE STUDY
HIGH TEMPERATURE GAS COOLED REACTOR
(JANUARY 1985 \$ MILLION)

3-DIGIT ACCOUNT NO.	DESCRIPTION	1260 MWL ANNULAR CORE PCRV CONFINEMENT			1170 MWL CYLINDRICAL CORE PCRV CONFINEMENT			1170 MWL REFERENCE DESIGN PCRV CONTAINMENT			1170 MWL CYLINDRICAL CORE PCRV CONTAINMENT		
		NSSS	BOMP	TOTAL	NSSS	BOMP	TOTAL	NSSS	BOMP	TOTAL	NSSS	BOMP	TOTAL
21	STRUCTURES AND IMPROVEMENTS												
212	REACTOR BUILDING	--	23.7	23.7	--	19.6	19.6	--	42.7	42.7	--	44.4	44.4
218T	ULTIMATE HEATSINK	--	8.1	8.1	--	13.5	13.5	--	16.9	16.9	--	13.5	13.5
218X	AUXILIARY BUILDING	--	--	0.0	--	--	0.0	--	15.6	15.6	--	16.3	16.3
218Z	EMERGENCY POWER BLDG.	--	1.4	1.4	--	1.4	1.4	--	2.0	2.0	--	1.4	1.4
	SUBTOTAL 21	0.0	33.2	33.2	0.0	34.5	34.5	0.0	77.2	77.2	0.0	75.6	75.6
22	REACTOR PLANT EQUIPMENT												
220A	NSSS BASE SCOPE	262.1	--	262.1	208.8	--	208.8	211.5	--	211.5	210.0	--	210.0
221	REACTOR EQUIPMENT	--	43.9	43.9	--	35.0	35.0	--	36.3	36.3	--	35.0	35.0
222	MAIN HEAT TRANSPORT	--	1.2	1.2	--	1.2	1.2	--	1.2	1.2	--	1.2	1.2
223	SAFEGUARDS COOLING	--	0.7	0.7	--	2.2	2.2	--	3.1	3.1	--	2.2	2.2
225	NUCLEAR FUEL HANDLING	--	4.2	4.2	--	4.2	4.2	--	4.2	4.2	--	4.2	4.2
226	OTHER REACTOR PLANT EQUIP'T	--	9.0	9.0	--	9.0	9.0	--	11.2	11.2	--	11.2	11.2
227	REACTOR PLANT INSTR/CONTROL	--	11.4	11.4	--	11.4	11.4	--	11.9	11.9	--	11.9	11.9
	SUBTOTAL 22	262.1	70.4	332.5	208.8	63.0	271.8	211.5	67.9	279.4	210.0	65.7	275.7
24	ELECTRIC PLANT EQUIPMENT												
241	SWITCHGEAR	--	0.9	0.9	--	0.9	0.9	--	1.5	1.5	--	1.0	1.0
242	STATION SERVICE EQUIPMENT	--	3.2	3.2	--	5.0	5.0	--	7.4	7.4	--	5.0	5.0
243	SWITCHBOARDS	--	0.1	0.1	--	0.1	0.1	--	0.1	0.1	--	0.1	0.1
244	PROTECTIVE EQUIPMENT	--	0.3	0.3	--	0.3	0.3	--	0.3	0.3	--	0.3	0.3
245	ELEC STRUC & WIRING CTRLS	--	2.1	2.1	--	2.1	2.1	--	3.1	3.1	--	3.1	3.1
246	POWER & CONTROL WIRING	--	4.1	4.1	--	4.2	4.2	--	6.5	6.5	--	6.5	6.5
	SUBTOTAL 24	0.0	10.7	10.7	0.0	12.6	12.6	0.0	18.9	18.9	0.0	16.0	16.0
25	MISCELLANEOUS PLANT EQUIP'T												
251	TRANSPORTATION & LIFTING	--	1.0	1.0	--	1.0	1.0	--	2.0	2.0	--	2.0	2.0
	SUBTOTAL 25	0.0	1.0	1.0	0.0	1.0	1.0	0.0	2.0	2.0	0.0	2.0	2.0
	TOTAL DIRECT COST	262.1	115.3	377.4	208.8	111.1	319.9	211.5	166.0	377.5	210.0	159.3	369.3

TABLE 6-2
ORDER OF MAGNITUDE CAPITAL COST SUMMARY
CONTAINMENT VS CONFINEMENT TRADE STUDY
HIGH TEMPERATURE GAS COOLED REACTOR

(JANUARY 1985 \$ MILLION)

DESCRIPTION	1260 MWt ANNULAR CORE PCRV CONFINEMENT				1170 MWt CYLINDRICAL CORE PCRV CONFINEMENT				1170 MWt REFERENCE DESIGN PCRV CONTAINMENT				1170 MWt CYLINDRICAL CORE PCRV CONTAINMENT			
	NSSS	BONP	CONV	TOTAL	NSSS	BONP	CONV	TOTAL	NSSS	BONP	CONV	TOTAL	NSSS	BONP	CONV	TOTAL
STRUCTURES AND IMPROVEMENTS	0.00	41.20	20.00	61.20	0.00	42.50	20.00	62.50	0.00	85.20	20.00	105.20	0.00	83.60	20.00	103.60
REACTOR PLANT EQUIPMENT	262.10	71.90	0.00	334.00	208.80	64.50	0.00	273.30	211.50	69.40	0.00	280.90	210.00	67.20	0.00	277.20
TURBINE PLANT EQUIPMENT	0.00	0.00	78.80	78.80	0.00	0.00	76.10	76.10	0.00	0.00	76.10	76.10	0.00	0.00	76.10	76.10
ELECTRIC PLANT EQUIPMENT	0.00	10.70	15.70	26.40	0.00	12.60	16.00	28.60	0.00	18.90	16.00	34.90	0.00	16.00	16.00	32.00
MISCELLANEOUS PLANT EQUIP'T	0.00	5.00	5.70	10.70	0.00	5.00	5.70	10.70	0.00	6.00	5.70	11.70	0.00	6.00	5.70	11.70
MAIN COND HEAT REJECTION	0.00	0.00	13.70	13.70	0.00	0.00	12.70	12.70	0.00	0.00	12.70	12.70	0.00	0.00	12.70	12.70
TOTAL DIRECT COST	262.10	128.80	133.90	524.80	208.80	124.60	130.50	463.90	211.50	179.50	130.50	521.50	210.00	172.80	130.50	513.30
CONSTR.SERVICE & FIELD ENB.	65.53	32.20	20.09	117.81	52.20	31.15	19.58	103.03	52.88	44.88	19.58	117.33	52.50	43.20	19.58	115.28
TOTAL FIELD COST	327.63	161.00	153.99	642.61	261.00	155.75	150.08	566.93	264.38	224.38	150.08	638.83	262.50	216.00	150.08	628.58
ENGINEERING SERVICES & FEE	49.14	24.15	12.32	85.61	39.15	23.36	12.01	74.62	39.66	33.66	12.01	85.32	39.38	32.40	12.01	83.78
OWNERS COST	37.68	18.52	16.63	72.82	30.02	17.91	16.21	64.13	30.40	25.80	16.21	72.41	30.19	24.84	16.21	71.24
TOTAL PLANT INVESTMENT	414.45	203.67	182.93	801.04	330.17	197.12	178.29	705.58	334.43	283.93	178.29	796.66	332.06	273.24	178.29	783.59
CONTINGENCY	82.89	40.73	18.29	141.92	49.52	29.57	17.83	96.92	50.17	42.59	17.83	110.58	49.81	40.99	17.83	108.62
TOTAL WITHOUT ALLOWANCE FOR FUNDS USED DURING CONSTRUCTION(AFUDC)	497.33	244.40	201.23	942.96	379.69	226.69	196.12	802.50	384.60	326.52	196.12	907.24	381.87	314.23	196.12	892.22
AFUDC	25.96	12.76	10.50	49.22	19.44	11.61	10.04	41.09	23.70	20.12	12.08	55.90	23.33	19.36	12.08	54.98
TOTAL CAPITAL EQUIPMENT	523.30	257.16	211.73	992.18	399.13	238.30	206.06	843.59	408.30	346.64	208.20	963.15	405.40	333.59	208.20	947.19
UNIT CAPITAL COST \$/KW				2089				1917				2189				2153

SECTION 7

OPERATION AND MAINTENANCE

Operating and maintenance costs have been developed for the small high temperature gas cooled reactor¹. These costs for a PCRV HTGR with a confinement were predicted to be $\$26.6 \times 10^6$ per year and would require an operating staff of 222 personnel. Since operating and maintenance costs have not been developed for a small PCRV HTGR with a containment, no detailed cost comparison of operating costs can be performed for PCRV concepts with a containment versus those with a confinement. However, it is expected that operating cost, as well as the plant staff of a PCRV HTGR with a containment, will be higher than a similar plant with confinement. This is attributable to the additional maintenance of the containment and related hardware.

Specifically, the containment will require additional work to maintain the steel liner and access hatches. Furthermore, the containment is subjected to periodic leak tests. These tests include a Type A test or integrated leak test. This test is conducted every 3 years and requires a test crew of approximately 8 people per shift (3 shifts per day). Of the 5 days needed to complete the test, 2 days will require suspension of all work in the containment and will extend the planned outage schedule for an equivalent duration. In addition to the Type A tests, the containment penetrations are subjected to Type B and C leak tests every 2 years. These tests require a test crew of 4 people per shift (3 shifts per day) and require approximately 90 days to complete.

In addition to the O&M expenses associated with testing, a work effort is required to maintain the containment and the penetration so that the tests will be successfully completed. The HTGR containment has approximately 198 penetrations, most of which have two isolation valves and a number of small

¹General Electric Company, "Interim Operating and Maintenance Report for Small High Temperature Gas Cooled Reactors, January 30, 1985.

diameter root valves for vents, drains, instruments, and testing. Those valves in severe service generally require pretest maintainence to restore leak rates to allowable limits. The extent of this effort has not been quantified at this time; however, it is expected that this will be the major contribution to the differential O&M costs between a containment and a confinement.

A number of other systems and features will also cause increased O&M costs for the containment plant. These include the extensive HVAC and chilled water system necessary to maintain an acceptable operating environment in the containment, along with the instrumentation and controls associated with the containment isolation system.

It can be concluded, therefore, that the O&M costs and the O&M staff for a PCRV HTGR with a containment will be higher than a similar plant with a confinement.