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MODULAR PEBBLE-BED REACTOR REFORMING
PLANT DESIGN FOR PROCESS HEAT

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September 1982

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UNITED STATES DEPARTMENT OF ENERGY

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ADVANCED REACTOR SYSTEMS DEPARTMENT • GENERAL ELECTRIC COMPANY
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GENERAL **ELECTRIC**

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TABLE OF CONTENTS

	<u>Page</u>
Abstract	vii
1.0 Introduction and Summary	1-1
1.1 General	1-1
1.2 Design Options	1-2
2.0 Requirements	2-1
2.1 General	2-1
2.2 Overall Plant	2-1
2.3 Core Design	2-3
2.4 Reactor Vessel - Equipment Vessel	2-3
2.5 Refueling Concept	2-4
2.6 Decay Heat Removal	2-5
2.7 Operation Logic - Sequence	2-6
2.8 Reformer	2-6
2.9 Steam Generator	2-6
2.10 NHS/BOP Layout Criteria	2-7
3.0 Principal Plant Parameters	3-1
3.1 Heat Balance	3-1
3.2 Principal Plant Operating Parameters	3-1
4.0 Mechanical Systems Design	4-1
4.1 System Description	4-1
4.2 Components	4-2
4.2.1 Reactor Vessel	4-2
4.2.2 Component Vessel	4-3
4.2.3 Reformer	4-3
4.2.4 Steam Generator	4-4
4.2.5 Circulator	4-7
4.2.6 Control Rod and Control Rod Drives	4-7
4.3 Fuel Handling System	4-8
4.3.1 New Fuel	4-8
4.3.2 Recycle Fuel	4-9
4.3.3 Spent Fuel	4-10
4.3.4 References	4-11
4.4 Vessel Supports	4-11
5.0 Reactor System Design	5-1
5.1 General Design Characteristics	5-1
5.2 Core Design Specifications	5-1
5.3 Reactivity Control	5-3
5.4 Core Analysis Model and Methods	5-5
5.5 Core Performance	5-7
5.6 References	5-15
6.0 Shutdown Heat Removal	6-1
6.1 General	6-1
6.2 Heat Removal Analysis	6-2
6.3 References	6-5

Table of Contents (continued)

	<u>Page</u>
7.0 Maintenance and In-Service Inspection Requirements	7-1
7.1 Introduction	7-1
7.2 Maintenance, Inspection and Availability	7-3
7.2.1 Maintenance and ISI of Nuclear Heat Source (NHS) Components	7-5
7.2.1.1 Reformer	7-5
7.2.1.2 Steam Generator	7-10
7.2.1.3 Fuel Handling System	7-16
7.2.1.4 Control Rod and C. R. Drives	7-17
7.2.1.5 Circulator	7-18
7.2.1.6 Cavity Cooling System (CCS)	7-19
7.2.1.7 Instrumentation	7-19
7.2.1.8 Helium Purification System	7-19
7.3 In-Service Inspection (ISI) Requirements	7-19
7.3.1 Primary Coolant Boundary	7-20
7.3.2 Reactor Internals	7-20
7.4 References	7-21
8.0 Preliminary Safety and Licensing Evaluation	8-1
8.1 Introduction	8-1
8.2 Safety and Licensing Issues and Their Evaluation	8-2
8.2.1 Fission Product Release in Core Heatup Accidents	8-2
8.2.2 Xenon Instability	8-6
8.2.3 Reactor Shutdown System	8-9
8.2.4 Applicability of Document No. 950396	8-9
8.3 Conclusions and Recommendations	8-10
8.4 References	8-11
9.0 Identification of Technical Issues	9-1
9.1 Core Design	9-1
9.2 Burnup Measurement	9-2
9.3 Fission Products Release Model	9-2
9.4 Neutronics Modeling Approximations	9-3
9.5 Control Availability	9-3
9.6 Vessel Supports	9-4
9.7 Hot Restart	9-4
9.8 Reactor Vessel Temperature	9-5
9.9 ISI of Reactor Internals and Support Structure	9-5
9.10 Reactor Removal as an Assembly	9-5

APPENDIX

A	Review of Modular HTGR Vessel Support Concepts for the Side-by-Side Configuration	A-1
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LIST OF TABLES

	<u>Page</u>
3-1 Principal Plant Operating Parameters	3-2
4-1 Reformer Design and Operating Parameters	4-5
4-2 Steam Generator Design Parameters	4-6
5-1 Design Specifications for the Modular Pebble Bed Core	5-9
5-2 Reactivity Requirements for the Modular Pebble Bed Core	5-10
5-3 Performance Parameters for the Modular Pebble Bed Core	5-11
5-4 Fuel Inventory and Mass Balance at Equilibrium	5-12
6-1 Maximum Transient Fuel Temperature for the Pebble Bed Reactor	6-6
7-1 Scheduled Outage Time Estimate Four Year Interval	7-4
7-2 Remove and Replace Reformer Catalyst Time	7-6
7-3 Remove and Replace Reformer	7-12
7-4 Steam Generator Tube Plugging	7-15
8-1 Parameters Affecting Fission Product Release	8-4
8-2 OTTO Cycle Design Options	8-8

LIST OF FIGURES

	<u>Page</u>
3-1 Pebble Bed HTGR-R Modular Plant Heat Balance	3-3
4-1 Modular Pebble Bed Reactor Assembly	4-13
4-2 Reformer/Steam Generator Assembly	4-14
4-3 Modular Pebble Bed Reactor Cross Section	4-15
4-4 Modular Pebble Bed Reactor - Installation	4-16
5-1 RZ Computational Model for the Modular Pebble Bed Core	5-13
5-2 Axial Profiles of Temperatures for the Modular Pebble Bed Core	5-14
6-1 An R-Z Calculational Model for the Transient Thermal-Hydraulic Calculations of Pebble Bed Reactors	6-7
6-2 Transient Temperatures for the Modular Pebble Bed Reactor System --Pressurized, OTTO	6-8
6-3 Transient Temperatures for the Modular Pebble Bed Reactor System --Pressurized, Recirculation	6-9
6-4 Transient Temperatures for the Modular Pebble Bed Reactor System --Depressurized, OTTO	6-10
6-5 Transient Temperatures for the Modular Pebble Bed Reactor System --Depressurized, Recirculation	6-11
7-1 Shielded Work Platform, Reformer Catalyst Removal and Replacement	7-9
7-2 Reformer Removal	7-11
8-1 Average Axial Fuel Burnup in MPBR	8-7
9-1 Modular Pebble-Bed Reactor Assembly -- (Horizontal Section)	9-7
9-2 Modular Pebble-Bed Reactor Assembly -- (Vertical Section)	9-8
A-1 Twelve Radial Shear Legs - Upper Level	A-12
A-2 Upper Level Radial Beams and U-Shaped Coaxial Pipe	A-13
A-3 General Arrangement of Conceptual Supports	A-14
A-4 Vessel Support Restraint Structure	A-17
A-5 Reactor Vessel Support Concept	A-18
A-6 Component Vessel Support Concept	A-19
A-7 Multiple Supports of Smaller Size	A-21

ABSTRACT

This report describes a preliminary design study of a Modular Pebble-Bed Reactor System Reforming (MPB-R) Plant. The system uses one pressure vessel for the reactor and a second pressure vessel for the components, i.e., reformer, steam generator and coolant circulator. The two vessels are connected by coaxial pipes in an arrangement known as the "side-by-side" (SBS). The goal of the study is to gain an understanding of this particular system and to identify any technical issues that must be resolved for its application to a modular reformer plant.

The basic conditions for the MPB-R were selected in common with those of the current study of the MRS-R in-line prismatic fuel concept, specifically, the module core power of 250 MWt, average core power density of 4.1 w/cc, low enriched uranium (LEU) fuel with a U-235 content of 20% homogeneously mixed with thorium, and a target burnup of 80,000 MWD/MT.

Study results include the pebble-bed core neutronics and thermal-hydraulic calculations. Core characteristics for both the "once-through-then-out" (OTTO) and recirculation of fuel sphere refueling schemes were developed. The plant heat balance was calculated with 55% of core power allotted to the reformer.

The major attribute sought by the SBS arrangement is the thermal isolation of components by preventing natural circulation after the occurrence of events such as loss of BOP cooling and subsequent reactor trips. By preventing coolant flow, the reformer steam generator and vessels are not subjected to thermal shocks which could shorten component lifetimes. Decay heat removal, after loss of normal cooling system events, is accomplished by heat conduction and radiation through the reactor vessel to the surrounding structure.

Some major technical issues which require early resolution are:

Control Availability - Although the general consensus is that reactivity control can be accomplished with reflector rods, it can not be confirmed without carrying out the control worth calculations.

Vessel Supports - The proposed method of supporting the vessels in the SBS arrangement with coaxial piping is not a proven concept. The concept appears feasible, but much work is needed to prove its viability.

Hot Restart - During the decay heat passive removal period, the core fuel sphere temperatures remain slightly above normal operating while the components in the component vessel cool down. Restarting with a hot core and cold components, in a reasonable time period, may not be feasible.

In-Service Inspection (ISI) - At present the ISI of the reactor internals and core support structure can not be accomplished from the fuel side without unloading the core. If this must be done every two to four years the outage time would negate the availability advantage made possible by "on-line" refueling.

1.0 INTRODUCTION AND SUMMARY

1.1 General

A preliminary design study of a Modular Pebble Bed Reactor System Reformer plant (MPB-R) has been undertaken using the side-by-side arrangement. The goal of this study is to gain an understanding of this particular system and to identify any technical issues that must be resolved for application to a modular reformer plant. This report presents the preconceptual design of the MPB-R plant which will be updated to include the work completed at the close of FY82.

The study progress results reported herein include the pebble bed core neutronics and thermal-hydraulic calculations. The core design characteristics for both the OTTO and recirculation fuel ball refueling schemes were developed. The basic conditions for the MPB-R were selected in common with those of the current study of the MRS-R in-line prismatic concept, specifically, module core power of 250 MWt, average core power density of 4.1 w/cc, low enriched uranium (LEU) fuel with a U-235 content of 20% homogeneously mixed with thorium, and a target burnup of 80,000 MWD/MT.

The plant heat balance was calculated with 55% of core power allocated to the reformer. Each module system produces 147 MWt of pipeline energy, 11.5 MWe net electrical with an overall plant heat utilization efficiency of 63.4%. The components were sized to these conditions and preliminary layouts were completed.

For a side-by-side arrangement, the major attribute sought is thermal isolation of components in which natural circulation is prevented after the occurrence of events such as loss of BOP cooling and subsequent reactor trips. By preventing helium flow, the steam generators, reformers, and vessels are not subjected to thermal shocks which could shorten component lifetimes. Decay heat removal after loss of normal cooling system events is accomplished passively by heat conduction and radiation through the reactor vessel to the surrounding structure.

1.2 Design Options

The core design option under consideration consists of the once-through-then-out (OTTO) pebble bed fuel cycle and the recirculation fuel cycle. The relative simplicity of the OTTO refueling scheme cycle is a desirable goal but the core thermal conditions are compromised by higher axial peak to average fuel temperatures. Characteristics of a core with the OTTO cycle, and with recirculating fuel pebbles (up to approximately ten times per pebble) are presented. The selection is open pending more detailed study.

Fuel handling systems have been identified for both type of fuel cycles. Refueling for both fuel cycle schemes is accomplished during on-line plant operation.

The component arrangement consists of a pebble bed core, top mounted control rods, axial and radial reflectors all located in a single reactor vessel. The RV is connected to a component vessel by a concentric cross duct. The component vessel contains the reformer, helical coil steam generator, circulator, and flow partitions.

2.0 REQUIREMENTS

2.1 General

2.1.1 Functional design approach and technical requirements shall be the same or equivalent to those used in developing the reference monolithic 1170 MW_t DC 950°C PH Plant as modified by this document.

2.1.2 Conform to GCRA Functional Specification - HTGR Modular Reactor System, draft copy dated December 4, 1981.

2.1.3 Conform to the monolithic HTGR-PH 1170 MW_t Nuclear Safety Plant Specification, Document No. 905396, issue one.

2.1.4 C-E Modular Reactor Ground Rules, Letter PAT-81-068, December 2, 1981.

2.1.5 GA Modular HTGR-PH Core Design Ground Rules, GA Memo CNE-236-55-81.

2.1.6 Bechtel Modular Reactor Systems BOP Ground Rules BL-128, January 25, 1982.

2.1.7 Shutdown heat removal via heat dissipation through the reactor vessel to redundant safety-class cooling systems. All primary system components shall be designed to accommodate two loss of BOP heatup transients per year.

2.2 Overall Plant

2.2.1 HTGR Direct Cycle reforming to produce both synthesis gas for distribution to thermo chemical pipeline (TCP) and generation of electricity.

2.2.2 Modular side-by-side integral reactor heat source to include the core, reformer, steam generator, circulator, control devices, and flow divider devices all within the reactor vessel enclosure.

2.2.3 Seismic Design Requirements

OBE = 0.15g

SSE = 0.30g

2.2.4 Reactor Module Thermal Capacity 250 MW_t

2.2.5 Aggregate Plant Thermal Capacity 2000 MW_t

2.2.6 Availability Goal:

- Average Capacity Factor 95%

- Minimum Capacity Factor 50%

2.2.7 Total plant to be constructed and operational within a minimum time cycle consistent with overall cost objectives. The construction time goal shall be four (4) years from the start of limited work authorization.

2.3 Core Design

2.3.1 Core Fuel Configuration - Pebble Bed

2.3.2 Coolant - Pressurized Helium, Down Flow Through the Core

2.3.3 Reactor Outlet Temperature - 950°C (1742°F)

2.3.4 Reactor Inlet Temperature - 400°C (742°F)

2.3.5 Control rod location in the reflector only, and a slump-proof Reserve Shutdown System (RSS) shall be incorporated as part of the control system.

2.3.6 Fuel Cycle

2.3.6.1 Low Enrichment Uranium (LEU)

2.3.6.2 Choice between One-Through-Then-Out (OTTO) and Recirculation to be determined.

2.3.7 Fuel Design Requirements

2.3.7.1 80,000 MWD/MT Burnup

2.3.7.2 Weight of a heavy metal in each fuel ball shall be limited based on water ingress reactivity safety limit.

2.3.7.3 Fuel Ball Diameter - 60 mm

2.3.7.4 Fuel Kernel Coating:

TRISO (two external layers of pyrolytic graphite plus one intermediate layer of silicon carbide)

2.3.7.5 Maximum Fuel Temperature: Steady state - 1260°C (2300°F)

2.4 Reactor Vessel - Equipment Vessel

2.4.1 Design Temperature

< 454°C (850°F)

< 371°C (700°F) Goal

2.4.2 Design Pressure - 900 psig, consistent with ASME rules for relief valves for peak pressure reached in an isolated modular reactor system reactor scram heatup transient.

2.4.3 Materials of Construction - TBD

2.4.4 Flange connection at upper head of reactor vessel.

2.4.5 Top mounted control rods and drives

2.4.6 Insulated against transients (if necessary), but designed to dissipate decay heat through the reactor vessel to the reactor cavity.

2.4.7 Ability to remove axial and radial reflectors through the top of the vessel with the vessel head removed.

2.4.8 Reactor vessel supported rigidly, equipment vessel supported with ability for lateral movement to accommodate thermal expansion of the cross duct.

2.4.9 Ability to remove the reformer tube bundle through the top of the equipment vessel and remove the circulator and drive motor assembly from the bottom of the vessel.

2.4.10 Cross duct outer surface shall not have a bellows or any surface discontinuities. Inner duct fitted with expansion duct and insulated to minimize heat transfer between flowing helium streams.

2.4.11 Transportation considerations shall not be a limiting factor in reactor vessel or equipment vessel design sizes.

2.5 Refueling Concept

2.5.1 On-line Refueling. Gravity flow of fuel balls through lower fuel discharge tube.

- 2.5.2 Fresh fuel balls introduced at top of core.
- 2.5.3 Refueling system to serve recirculation or once through then out OTTO fuel strategy - TBD.
- 2.5.4 Two-zone core refueling system.
- 2.6 Decay Heat Removal
 - 2.6.1 Requirements (consistent with NRC/GA requirements)
 - a. GDC 34 (Pressurized)
 - Fuel, pressure boundary and component design limits are maintained within upset (PC-2) conditions following any initiating event and single active failure. (Peak core temperature <1600°C).
 - No investment loss.
 - b. GDC 35 (Loss of Coolant-depressurized)
 - No fuel clad damage which interferes with effective core cooling.
 - Off-site releases within NRC criteria.
 - Investment loss (faulted primary loop condition) accepted.
 - c. Meet 10CFR50 (Paragraph 50.46) ECCS Criteria.
 - 2.6.2 Preconceptual Configuration/Approach
 - a. One Main Cooling System connected to Non-Hardened (Non-Safety Class) BOP is used for normal shutdown.
 - One circulator with non-safety grade pony motor
 - Single steam generator unit per module
 - Circulator check or isolation valve - TBD.
 - b. Heat Dissipation through Reactor Vessel to Redundant Safety-Grade Reactor Cavity Cooling Systems used for Off Normal Conditions.

- Redundant cooling coils on biological shield or module containment and/or safety-grade H&V system.

2.6.3 Off-normal decay heat removal shall not require operation of circulators (for pressurized or loss of pressure cases).

2.7 Operation Logic - Sequence

2.7.1 Pressurized Decay Heat Removal

- BOP intact. Forced circulation (Main Motor) through main BOP cooling system (MCS).
- BOP intact. Forced circulation (Pony Motor) to (MCS).
- Loss of BOP. Natural circulation in core region. Reactor vessel coolant isolated from component vessel coolant by absence of thermal driving head and closed crossover valve. Radial convection, radiation, and conduction heat transfer from core through RV to cavity cooling system.

2.7.2 Depressurized Decay Heat Removal

- BOP intact. Forced circulation (Main Motor) to MCS.
- BOP intact. Forced circulation (Pony Motor) to MCS.
- Loss of BOP. Radial convection, radiation, and conduction from core through RV to cavity cooling system.

2.8 Reformer

2.8.1 Type - Single wall bayonet tube

2.8.2 Material of construction - Inconel 617 tubing

2.8.3 Steam/Methane ratio - 3.0 to 1.

2.9 Steam Generator

2.9.1 Type - Coiled tube

2.9.2 Material of construction - open

2.10 NHS/BOP Layout Criteria

2.10.1 Site conditions - same as for monolithic plant

2.10.2 Down flow through core

2.10.3 Top mounted control rod drive system

2.10.4 Reformer located above circulator, removable without cutting process gas piping.

2.10.5 Top section of component vessel (containing reformer and steam generator) to be removable.

2.10.6 Single circulator and steam generator coil per reactor module

2.10.7 One-line refueling

2.10.8 Refueling, maintenance/in-service inspection, repair or accidents in one module shall not prevent/interrupt operation of other modules. Sufficient module independence to be provided to allow any one module to be shut down and taken out-of-service while the other modules continue to operate.

2.10.9 Sufficient redundancy and interconnections to achieve 95% average plant capacity.

- final number of reformer trains and extent of redundancy to be determined based on capital cost and capacity factor requirements. For initial design, assume one train per 250 MW_t NHS with redundant active components.
- one steam turbine with bypass system designed to maintain steam flow to process trains during extended TG outages.

- auxiliary condenser to allow plant operation during turbine

2.10.10 Layout Basis for Cost Estimating

Assume NHS modules are installed four at a time - four with the initial plant, and another four when the plant is taken to the final aggregate size. However, eight modules may be constructed at one time if this is the least cost option.

2.10.11 Plant layout arranged such that the final aggregate plant will accommodate eight independent 250 MW_t modular reactors, with each set of four located near the centralized facilities shared by all eight reactors, (control room, fuel handling, fuel storage, change rooms, maintenance facilities).

2.10.12 Plant arranged such that construction of the later add-on reactor modules (four units) can be conducted without affecting operation of the earlier units.

2.10.13 Initially, each reactor to have shared decay heat removal systems, with the option for using separate systems later, if safety criteria can not be satisfied using shared systems.

2.10.14 A confinement type reactor building, with filtered exhaust and a metal enclosed maintenance bay, to be used for current design approach.

2.10.15 Degree of leak-tightness and exhaust air cleanup system requirements for the confinement structure to be based on postulated design basis (site suitability source term) fission product release from a single modular reactor to the space within the confinement and the requirement that the other modules remain operational.

2.10.16 Reactor Confinement

- Region surrounding reactor vessel to be capable of atmosphere composition and pressure control, i.e., inertable during power operation and controllable between + TBD psig and - TBD inches of water gauge.
- When primary coolant boundary is open, confinement to be maintained such that all fission products and radioactive materials are prevented from causing more than TBD % of the radiation levels allowed by applicable codes and regulations.

2.10.17 The following NHS auxiliary and plant support systems to be shared by all eight NHS modules, unless the capital cost of providing full capacity initially is not justified, due to a long time span between the first four and the final four NHS modules; or unless the availability goal requires more than one system:

- Reactor plant cooling water
- Fuel handling and storage cooling water
- Radioactive waste
- Helium supply/storage
- Nitrogen
- Decontamination/maintenance
- Nuclear service water
- Circulating water (main plant cooling tower)
- Essential chilled water
- Plant raw water treatment
- Instrumentation and control
- Plant perimeter fencing and security
- Fire security

2.10.18 Turbine generator plant support systems to be provided on an as-needed basis.

2.10.19 A redundant reactor cavity cooling system (passive to the extent possible) to be provided to maintain the normal temperature of the concrete supporting structures at, or below, 150°F. Accident design temperatures allowed to go up to (TBD).

2.10.20 All gas/water/steam piping penetrating the reactor containment (or confinement) boundary to meet the isolation criteria of Appendix A of 10CFR50. (Assumed to also apply to HTGR-HTPH.)

2.10.21 Conceptual design of NHS and BOP equipment to facilitate use of pre-fabricated modules to minimize on-site fabrication.

3.0 PRINCIPAL PLANT PARAMETERS

3.1 Heat Balance

The heat balance for the Modular Pebble Bed Reformer Plant is shown in Figure 3-1. The overall plant conditions are similar to those of the modular prismatic plant under study except the core inlet temperature has been reduced to 752°F (compared to 797°F for the prismatic) and the crossover temperature (helium leaving the reformer) is 1184°F (compared to 1214°F for the prismatic).

3.2 Principal Operating Parameters

Table 3-1 presents the principal plant operating parameters. The plant will have eight modules for an aggregate size of 2000 MWt.

TABLE 3-1. PRINCIPAL PLANT OPERATING PARAMETERS

Plant Size, MWt	2000
Number of NHS Modules	8
NHS	
Core Power, MWt	250
Core Outlet Temperature, °C (°F)	950 (1742)
Core Inlet Temperature, °C (°F)	400 (742)
Helium Pressure, bar (psia)	46 (667)
Circulator Pumping, MWt	3.7
Helium Flow, kg/sec (lb/hr)	87.4 (693,900)
Primary Pressure Drop, bar (psi)	1.17 (17)
Reformer System	
Power Split, %	55
Reformer Load, MWt	139.5
Reformer Pressure, bar	49
Steam/Methane Ratio	3.0/1
Helium Crossover Temperature, °C (°F)	640 (1184)
Process Gas Inlet Temperature, °C (°F)	537 (1000)
Process Gas Outlet Temperature, °C (°F)	637 (1180)
Process Gas Flow, kg/sec (lb/hr)	68 (540,000)
Methane Conversion, %	60
Steam-Electric System	
Steam Generator Total Load, MWt	904
Throttle Pressure, bar (psia)	165 (2,400)
Throttle Temperature, °C (°F)	565 (1,050)
Feedwater Temperature, °C (°F)	260 (500)
Throttle Flow, kg/sec (lb/hr)	283 (2.25x10 ⁶)
Gross Power Output, MWe	204
Condenser Load, MWt	376
Pipeline	
Pumping Power, MW	48
Pressure, bar (psia)	76 (1,100)

PEBBLE-BED HTGR-R MODULAR PLANT HEAT BALANCE

FIGURE 3-1

PIPELINE ENERGY	147 Mwt
GROSS POWER	25.8 MWe
NET POWER	11.5 MWe
PLANT EFFICIENCY	63.4%

4.0 MECHANICAL SYSTEMS DESIGN

4.1 System Description

The modular pebble-bed reactor (MPBR) system consists of the fuel spheres core, surrounded by graphite top, bottom and radial reflectors, all housed in a cylindrical steel core barrel. The core barrel rests on a steel support in the reactor pressure vessel as shown in Figure 4-1. Thirty-six gravity assisted control rods are located in the top of the reactor vessel, with the control rod drives mounted outside the vessel head. Thirty of the control rods are inserted in the radial reflectors and six are inserted in columns of graphite which are extensions of the radial reflector into the core region.

Helium coolant flows down through the core to an annulus (hot gas plenum) in the bottom reflectors. As the coolant passes through the core its temperature is increased from 752°F to 1741°F and the pressure decreases from 656 psia to 650 psia. From the annulus the helium flows through the inner pipe, of coaxial pipes, which connects the core barrel to the shroud in the component pressure vessel. The helium then flows upward through the reformer, leaving the reformer at 1148°F and 641 psia. Upon exiting the reformer the helium flow turns and moves downward through the helical coiled steam generator, and leaves the steam generator at 735°F and 639 psia.

From the steam generator the helium flows through ducts down to the helium circulator which is located in the center bottom of the component pressure vessel. The circulator drive is located outside the bottom closure of the vessel. The circulator boosts the temperature of the helium to 752°F and the pressure to 659 psia. The helium is returned into the component pressure vessel outside the shroud and then flows through the annulus between the inner and outer coaxial piping. The outer pipe connects the component pressure vessel to the reactor pressure vessel.

Helium enters the reactor vessel below the bottom of the core support plate. Holes in the support plate are aligned with holes provided

for all the control rods and with coolant passage holes surrounding the control rod holes in the radial reflectors. The helium flows up through all the holes to a plenum in the top reflectors to complete the flow circuit. (See Figures 4-1 and 4-2.)

In the reformer, a mixture of methane and steam enters at 1000°F and 718 psia, which is converted to syngas leaving the reformer at 1180°F and 655 psia.

In the steam generator, feedwater enters at 500°F and 2900 psia and steam leaves the steam generator at 1050°F and 2400 psia.

All gas, water and steam pipes connected to components in the component pressure vessel have guard pipes and appropriate thicknesses of thermal insulation. Likewise, the inner coaxial pipe is insulated and equipped with expansion components to allow for movement due to changes in temperature. The two pressure vessels and the outer coaxial pipe are not insulated which allows in-service inspection (ISI) of all the pressure boundary outer surfaces to be readily accomplished.

The helium circulator is equipped with a shut-off valve which is closed for temperature excursions that may occur in the reactor core. This prevents thermal shock of the reformer and steam generator. Decay heat from the core is transmitted by conduction, radiation, and convection to the reactor pressure vessel and then from the vessel by convection and radiation to redundant sets of cooling coils attached to the walls of the reactor vessel cavity. Water for the cooling coils is supplied from cooling towers by redundant pumps for each circuit.

4.2 Components

4.2.1 Reactor Vessel

The reactor vessel of the Modular Pebble Bed Reactor (MPB-R) is made of SA-387, GR.22, CL-2 (2-1/4Cr-1Mo) steel. It has a 251-inch outer diameter with a nominal wall thickness of 6.50 inches. The overall height of the reactor vessel is about 68 feet. The reactor vessel houses the core,

top, bottom and radial reflectors and insulation in a 1 inch thick cylindrical core barrel. The core barrel is centered, radially, in the reactor vessel so that there is a 7 inch annulus of stagnant helium between the core barrel and the reactor vessel. (See Figure 4-1).

Penetrations in the reactor vessel are as follows:

- a. Thirty-six (36) in the head for control rod drives.
- b. Seven (7) in the side, above the core, for the insertion of new and recycled fuel spheres.
- c. One (1) in the bottom center for discharging fuel spheres.
- d. One (1) very large opening for the co-axial pipes that connect the reactor vessel to the component vessel, located in the side of the reactor vessel above the core support and below the core. The large structural pipe connecting the two vessels is discussed in more detail in Section 4.4 below.
- e. Other penetrations may be added to facilitate and expedite in-service inspection of the inside surfaces of the reactor vessel, the core barrel and the core support structure. The best location for these penetrations and the number of penetrations is TBD.

4.2.2 Component Vessel

4.2.3 Reformer

The reformer is positioned in the components vessel as shown in Figure 4-2. Table 4-1 provides the reformer design and operating parameters. The reformer design of this side-by-side arrangement differs from the reformer of an in-line arrangement in the allowable tube length. The side-by-side reformer tube lengths are not restricted to accommodate the refueling access space. The side-by-side reformer tubes are thus about 4 meters longer and less in number. The tube bundle diameter is also considerably smaller.

The reformer configuration is a bayonet tube design with center return pigtail and monolithic type of catalyst located in the annular volume. Hot helium enters at the bottom of the tube bundle and flows in a cross flow pattern in the tube bundle up to the bottom of the tube sheet.

Helium exits at the top of the bundle and makes a 180° turn and flows downward through the steam generator tube bundle.

4.2.4 Steam Generator

The steam generator is a helical coil tube configuration once-through flow type of unit. The coils are located outside of and circle the reformer tube outer shroud cylinder

Feedwater flows into the tube bundle at the lower tubesheet progresses upward and out at the upper tubesheet. There are 130 tubes arranged in parallel. Table 4-2 provides the design details.

TABLE 4-1
REFORMER DESIGN AND OPERATING PARAMETERS

REFORMER DESIGN

Heat Transferred	139.5 MWt
Material	Inconel 617
Reformer Tube, Type	Single Wall
Reformer Tube, O.D.	64.5 mm (2.549 in.)
Reformer Tube, I.D.	49.3mm (1.948 in.)
Active Tube Length	14.0 m (553 in.)
Number of Tubes	769
Pitch Factor	1.24
Baffled Regions	4
Bundle Diameter	2.4 m (95 in.)
Pigtail O.D.	22 mm (0.866 in.)
Pigtail I.D.	20 mm (0.787 in.)

OPERATING PARAMETERS

Primary Helium	
Flow	87.4 kg/sec (693,900 lb/hr)
Inlet Temperature	1222°K (1739°F)
Outlet Temperature	913°K (1184°F)
Inlet Pressure	44.8 bar (650 psia)
Outlet Pressure	44.2 bar (641 psia)

Process Gas	
Flow	68.1 kg/sec (540,000 lb/hr)
Inlet Temperature	811°K (1000°F)
Outlet Temperature	911°K (1180°F)
Inlet Pressure	50 bar (725 psia)
Outlet Pressure	45 bar (655 psia)

Steam-to-Methane Ratio 3.0/1

Methane Conversion 60%

PROCESS GAS MOLE FRACTION

Reformer Inlet	
CH ₄	25%
H ₂ O	75%
Reformer Outlet	
CH ₄	7.5%
H ₂ O	40.3%
CO	6.3%
H ₂	40.5%

TABLE 4-2
STEAM GENERATOR
DESIGN PARAMETERS

Number of Tubes	130
Concentric Rows of Tubes	6
Tube Length, ft	282
Tube Dia/Wall, in	0.875/0.146
Bundle Height, ft	17.0
Bundle Annular Thickness, in	13.76
Transverse pitch, in	1.96
Longitudinal pitch, in	1.33

4.2.5 Circulator

The circulator is located at the bottom of the component vessel. A two stage axial flow design is selected to minimize radial space requirements. The circulator will have a water cooled and lubricated bearing with a helium water type of buffered seal. The seal interface forms the pressure boundary and has a back facing arrangement to provide a seal when the circulator motor and bearing/seal cartridge is removed. The drive motor will not be a class 1E motor since a passive non-circulatory mode of off-normal conditions decay heat removal is used.

The required circulation power to the fluid is 3.7 MW at a helium flow of 87.4 kg/sec and pressure drop of 1.17 bar at motor/drive assumed efficiency of 88%, the motor size is 6000 hp.

Coupled with the circulator will be a shutoff valve for the major purpose of achieving rapid isolation to protect the components and vessels from thermal shock.

4.2.6 Control Rod and Control Rod Drives

There are thirty-six (36) control rods and control rod drives in the MPB-R. Thirty of the control rods are in a hexagon pattern, 5 in a line, on each side of the hexagon shaped core. They move up and down in holes provided in hexagon shaped reflector blocks. Six columns of graphite, in each corner of the hexagon shaped core, also contain a control rod. The columns could be enlarged to accept 2 control rods each, if needed. (See Figure 4-3).

The control rods in the reflectors are used to radially shape the power output of the core and when fully inserted bring the core to hot shutdown. The control rods in the columns are used to bring the core to cold shutdown. Total travel provided for the control rods is 70-80% of the core height, i.e., 19-22 feet.

The control rod drives are designed on the same principle as those used in the Fort St. Vrain HTGR. The control rod is attached to a flexible steel cable which is wound on a drum. The drum is gear driven through a miter-box by a drive-shaft. The drive-shaft penetrates the reactor vessel head where double sets of seals are used to prevent helium coolant leakage to the containment building. The drive-shaft is connected to an electromagnetic clutch which is driven by an electric stepping motor. In the case of SCRAM or power failure the clutch instantaneously disengages and the control rod falls to its fully inserted position by gravity.

4.3 Fuel Handling System

The fuel handling system comprises three parts: 1) new fuel handling, including insertion of fuel spheres into the core; 2) recycle fuel handling, which includes inspection for broken spheres, measurement of burnup and reloading in the core; and 3) spent fuel handling, which includes storage, packaging and shipment off site.

There are 367,920 fuel spheres in the core of each MPBR. The fuel spheres are recycled 10 times in a two-year period or once in every 73 days. To achieve an equilibrium core, 10% of the fuel spheres with the highest burnup are replaced with new spheres beginning with the sixth recycle. The replaced (spent) fuel spheres are collected in canisters for storage and later packaged for shipment. Storage and shipping are accomplished in a centrally located Reactor Service Building.

4.3.1 New Fuel

New fuel is unpacked in a glove box in a cell at a high enough elevation that transfer from the glove box to the core is accomplished by the spheres flowing through a 2.5 inch pipe by gravity. After visual inspection, the spheres are automatically counted and registered in the fuel handling computer as the spheres flow through a counter and valves to a hopper. Each hopper serves four MPBR's and holds one day's supply of fuel spheres (2,016).

Much of the technology for the Fuel Handling System proposed for the Modular Pebble Bed Reactor (MPBR) is taken from the system developed and operated by Arbeitsgemeinschaft Versuchs-Reaktor (AVR) in West Germany. The AVR fuel handling system has been expanded for use in the Thorium-Hochtemperatur-Reaktor (THTR), which is currently scheduled to go into operation in the near future. Due to the limited size of the MPBR core, some simplifications and a reduction in the number of components planned for use in the THTR are possible. The AVR and THTR fuel handling systems have been reported in several international publications, one of the best being Reference 4.3.4-1.

Each hopper is connected by 2.5-inch pipes with counters and valves to four distributors, one at each MPBR. The distributors are located near and above the MPBR vessels.

Six 2.5-inch pipes connect the bottom of the distributor to equally spaced loading positions about the periphery of the core. Each of these feed pipes has a counter and valves. The valve nearest the core is a maintenance valve that is only closed when performing maintenance on other valves, counters or the distributor. The six feed pipes are purged of air and pressurized with about 5 psia helium greater than the helium coolant pressure at the top of the core. This prevents contamination of the feed pipes by contaminated coolant.

Beginning with the sixth recycle of fuel spheres, new fuel spheres are inserted into the core at the rate of 504 spheres per day. They are inserted 6 at a time, one from each feed pipe, every 10 minutes over a period of 14 hours. The quantity of 504 fuel spheres is 10% of the total quantity of spheres recycled daily for each MPBR. By replacing 10% with new fuel starting with the 6th recycle an equilibrium core is obtained at the end of 10 recycles.

4.3.2 Recycle Fuel

A quantity of fuel is discharged, one sphere at a time, from the bottom of the core. The fuel flows by gravity through valves, a counter and

a damaged sphere separator. Damaged or broken spheres are removed from the recycle system and collected in a container. Undamaged spheres are routed through a burnup measurement detector. Dependent upon the degree of burnup the sphere is routed to either a pneumatic elevator or flows by gravity to the spent fuel collection cell. Spent fuel is loaded into containers for storage. Pressurized helium is used by the pneumatic elevator to return fuel spheres to the top center of the core via a 2.5 inch pipe equipped with a counter and valves. Two valves, one nearest the bottom of the core and one nearest the top of the core in the recycle pipes are for maintenance. They are only closed when maintaining a component in the recycle system or when removing and replacing a damaged sphere container.

All controls of the singulizer, damaged sphere separator, burnup detector, pneumatic elevator and all valves are located behind biological shield walls. All controls are automatically actuated by the fuel handling system computer for nominal operations. All controls are equipped with manual-override and their behind shield walls locations permits repair or replacement without radiation exposure to workers.

Fuel spheres are recycled at the rate of 5,040 per day. The proposed timing is one sphere every 10 seconds for a period of 14 hours. This allows 10 hours per day for moving spent and damaged fuel containers to storage and for equipment calibration and repairs.

4.3.3 Spent Fuel

When a spent fuel sphere is detected it is routed through the pneumatic elevator to the spent fuel line. This line is a 2.5 inch pipe with a counter and valves. It is connected to a storage container in the spent fuel collection cell. The collection cell has a helium atmosphere at ambient temperature.

When a storage container is full, it is disconnected from the spent fuel line and moved to the central storage facility. Decay heat removal during storage may be forced air or by submerging the container body in water. The capacity of a storage container is 6,050 spheres, which is

10% of the fuel removed from the core in a 12 day period. Disconnecting the container from the spent fuel line and moving it, out of the cell through an air-lock, to the storage facility is accomplished by remotely operated equipment.

After a suitable period of decay storage the spent fuel containers are transferred to a packaging and shipping cell. The spent fuel spheres are transferred from the storage containers to shipping canisters. The canisters are sealed to be the primary containment of a double containment shipping system. The canisters are then loaded into a shipping cask. The cask is closed, sealed, seal tested and cleaned of contaminants, if needed, and removed from the Reactor Service Building.

4.3.4 References

- 4.3.4.1 Hennings, U., "Fuel Handling System For Core Elements of a Pebble-Bed Reactor," Brown Boveri/Krupp, Mannheim, Germany, Nuclear Applications Technology, Vol. 7, October 1969, page 334.

4.4 Vessel Supports

The mechanical system presented in this Section includes the reactor pressure vessel and the component (reformer and steam generator) pressure vessel, located side-by-side with coaxial interconnecting piping. For this arrangement to respond to thermal movements and to accommodate seismic forces, in a straight forward reliable way, the two pressure vessels and piping must be supported in the same horizontal plane at the centerline of the interconnecting piping. In addition, since the reactor is of primary safety concern, it is desirable to solidly tie the reactor pressure vessel to the stiff and massive concrete building structure to minimize seismic induced loads.

Support of the reactor and component pressure vessels and the interconnecting coaxial piping is accomplished by the use of a fixed-rigid support (vessel to concrete connection) of the reactor vessel, opposite the interconnecting piping, in the plane of the piping centerline. This support

carries its proportional share of the total weight of the reactor vessel, component vessel and the coaxial piping. The remainder of the weight is supported by a fixed - sliding connection, on the component pressure vessel, opposite the interconnecting piping, in the plane of the piping centerline as shown in Figure 4-4. Also in the same plane as the interconnecting piping centerline, 120° on each side of the fixed-rigid and fixed-sliding supports are seismic restraints (snubbers) which allow for both vertical and horizontal thermal movements. Additional seismic restraints are located at the center-of-gravity of each vessel and near the top of each vessel.

The interconnecting coaxial piping must be structurally designed to act as the primary pressure boundary, as a thermal ram to move the component pressure vessel and as a seismic restraint for the component pressure vessel. Thus, the outer pipe diameter and wall thickness will be determined by the most severe combinations of requirements resulting from service as a pressure vessel, a column and a beam.

The vessel support system described above may be feasible, however, it needs to be evaluated by analytical analysis. The loads must be determined and the supports and snubbers sized. The diameter and wall thickness of the outer coaxial pipe must be determined. Fabrication and installation capabilities need to be considered. This vessel support system is a technical issue, which requires more study to insure the success of the side-by-side MPBR.

MODULAR PEBBLE-BED REACTOR ASSEMBLY

FIGURE 4-1

REFORMER/STEAM GENERATOR ASSEMBLY
MODULAR PEBBLE BED REACTOR

FIGURE 4-2
4-14

MODULAR PEBBLE-BED REACTOR CROSS-SECTION

FIGURE 4-3

MODULAR PEBBLE BED REACTOR - INSTALLATION

FIGURE 4-4

5.0 Reactor System Design

5.1 General Design Characteristics

In assessing the characteristics of the Modular Pebble Bed Reactor (MPBR) System many of the key design parameters were essentially fixed in order to make use of the extensive design experience in the Federal Republic of Germany. For instance, the choice of heavy metal loading was influenced by the consideration of the water ingress accident. At 7 g of heavy metal per ball, the maximum reactivity due to water ingress was limited to less than about 1.0% which is within the general envelope of the other anticipated accidents for the modular system. Another example is the choice of core height-to-diameter ratio. At a ratio of approximately 3.2, there is evidence that adequate shutdown margin can be provided by the control in the side reflector and the nodal extension of the reflector into the core region. The recirculation of the exposed fuel balls was intended to flatten the axial power distribution. This is especially important in the case of vertical in-line reactor vessel alignment in which the maximum fuel temperature is located at the top of the core and is directly influenced by the power density at that position. For the side-by-side reactor vessel alignment, the recirculation of the exposed fuel balls reduces the axial peak-to-average power ratio, and consequently, has a significant impact on the maximum fuel temperatures during the core heat-up transients. However, ball recirculation complicates the design of fuel handling system. In particular, a reliable system is required to analyze and sort the exposed fuel balls being discharged from the core bottom, circulate the fuel balls that exceed the specified maximum burnup to storage, and recirculate the remaining balls back to the top of the core. It is noted that these requirements can be eliminated by utilizing the Once-Through-Then-Out (OTTO) fuel cycle scheme.

Several other factors were also taken into consideration in specifying the core design parameters. For instance, the choice of power density at 4.1 w/cc was made to provide a meaningful comparison between the MPBR core

and General Atomic prismatic modular system, although a lower power density would be desirable for reducing the xenon absorption and decay heat source.

5.2 Core Design Specifications

The preconceptual MPBR core design was chosen to be a 250 Mwt system with the following key features:

1. Average helium inlet and exit temperatures of 300°C and 950°C, respectively;
2. A heavy metal loading of 7 g of mixed thorium and low enriched uranium (LEU) fuel per feed ball with the U-235 enrichment in the uranium heavy metal specified to be 20%; and
3. A fuel management scheme based on the recirculation of each fuel element a total of ten times for a total fuel exposure of 80000 MWD/MT. A simultaneous analysis of the OTTO fuel cycle was also performed to provide a base for assessing the advantages and disadvantages of the two approaches.

A listing of all the principal core design parameters for the MPBR is given in Table 5-1. The cylindrical core cavity is formed by graphite blocks which act as neutron reflectors. The top reflector contains penetrations for the inlet tubes of the fuel handling system. The bottom reflector has penetrations for the fuel exit tubes and for the flow of the coolant exit gas. Control rods are located in the side reflectors and in the nodal extensions of reflectors into the core.

Two radial fissile enrichment zones were specified to flatten the radial power distribution, and consequently, to flatten the helium outlet temperature profile and reduce the maximum fuel temperature. At equilibrium

of the recirculation core, one tenth of the discharged fuel balls at the bottom of the core (i.e., those that achieved the desired exposure after recirculation) were discriminated and transferred to storage. The remaining fuel balls were recirculated to the top of the core. Fresh fuel balls were mixed with the recirculated balls in the outer enrichment zone to provide the appropriate radial enrichment splits. The feed enrichment requirement was found to be 8.30% in order to keep the core critical at the equilibrium cycle. For the OTTO core, the disloaded fuel balls at the core bottom were discharged from the core and transferred to storage. The feed enrichments for this fuel cycle were found to be 8.19 and 10.74% for the inner and outer enrichment zone, respectively. The utilization of two enrichment zones complicates the fuel handling procedures. However, flattening the radial power is deemed necessary to ensure a small variation in the peak and the average helium exit temperature.

The target burnup was specified to be 80,000 MWD/MT corresponding to an average fuel residence time of approximately 740 days. The choice of high burnup system was intended to maximize the in-site utilization of the bred U-233 and therefore reduce the requirements for gross uranium ore.

5.3 Reactivity Control

The reactivity requirements for the long-term shutdown of the preconceptual MPBR core are given in Table 5-2. The results, which were based on the recirculation core, are of preliminary nature and some of the results were extrapolated from German analyses for the modular pebble bed core. More detailed analyses and careful evaluations are needed for the final conceptual design.

The maximum assumed accident in Table 5-2 is due to the ingress of about 500 kg of water in the core. It was found that the water ingress reactivity increased initially in direct proportion to the amount of water in the core, reached a maximum of 1.1% Δk at 500 kg of water, and then decreased gradually. The results were consistent with those found in the Germany for a heavy metal loading of 7 grams per ball.

The shutdown requirements include a running-in period reserve of approximately 0.6% Δk for the first core, and a load following capability for overcoming about 2.2% Δk due to burnup effects and xenon buildup while operating at reduced power. The excess reactivity in Table 5-2 also provides a margin for local power control, including the control of xenon induced power oscillations. The subcriticality requirements were assumed to be 0.3% Δk .

The principal long-term reactivity effects include the temperature defect, Xe-135 decay, and the decay of Pa-233 and Np-239 in going from hot operations at nominal power to cold shutdown. The temperature defect result is based on a direct calculation using the cross section data set generated at room temperature. The reactivity effects of Xe-135, Pa-233, and Np-239 decay were obtained by computing the effective neutron multiplication factors from direct diffusion calculations which were carried out with the nuclide density distributions after decay (for instance, the concentrations of Pa-233 were added to U-233 in the case of Pa-233 decay), and then comparing with the reference k-effective. The results from the first-order perturbation analyses also showed fairly good agreement with the direct calculation for these long-term reactivity effects.

The total reactivity requirement for the preconceptual MPBR core, including a 10 percent uncertainty factor for the total long-term effect, was computed to be 14.0% Δk .

The reactivity control for the preconceptual modular core consists of two independent systems. The primary system contains 30 reflector rods located around the periphery of the active core. The secondary system contains six reflector rods located at the noses that extend into the active core. A layout of the control system is shown in Figure 4-3. The primary control system serves as both the power control system and shutdown system; and the secondary control system is used for shutdown only. The power control includes a load following capability over a range of 100-40-100% of full power, and the capability for the local adjustment of the power distribution including the suppression of xenon induced power oscillations.

The shutdown requirements include both a fast and long-term shutdown capability. The fast shutdown system is designed to hold the core subcritical for a short period of time while it remains near operating temperature. Thus, the reactor is capable of a fast restart. For long-term shutdown, the control system must hold the reactor subcritical at ambient temperatures for an indefinite period of time. It is noted that the fast shutdown can be accomplished with the primary system alone. For the long-term shutdown, both the primary and secondary systems are needed.

By limiting the core diameter to less than 3 meters and utilizing two enrichment zones and the noses that extend into the active core, it is felt that reactivity control can be accomplished with reflector rods without any active control measure in the top reflector and active core. However, control worth calculations have not yet been completed and continuing efforts are needed to estimate the effectiveness of reflector rods.

5.4 Core Analysis Model and Methods

The calculational model for the preconceptual MPBR core was simplified to give an R-Z representation of the core layout, as shown in Figure 5-1. This model neglects the conical shape of the core at the bottom, and includes a pseudo absorber cross section in the side reflector to take into account the control system in that region. Substitution of a diluted graphite reflector in place of the perforated upper reflector and void region was also incorporated in the calculational model. The core is subdivided into two cylindrical zones corresponding to different enrichments. Fuel balls were assumed to move step-wise down. For the OTTO core, the difference in ball velocities was factored in by using six radial channels with finer axial divisions being used for the outer channels. However, only two radial channels with equal axial divisions were used for the recirculation core in order to reduce the computational cost.

The burnup calculations for the OTTO core were carried out with the normal pebble bed calculational scheme. That is, fresh fuel balls were loaded into the core at the top of the core, burned for a specified period of time, moved downward to a lower axial position, and finally discharged

from the core bottom. For the recirculation core, however, a different approach was taken to simulate the recirculation of the exposed fuel balls. Because the fuel balls were circulated through the reactor core a total of 10 times, it was necessary to have at least 10 distinct fuel batches in each axial layer to keep track the history of fuel ball passage. In this analysis, a total number of five batches was used for each axial layer in each radial enrichment zone with the fresh fuel, which is identified as batch 55 in Figure 5-1, introduced at the outer core zone. All fuel balls were to pass through the inner and outer core zones alternately to reach an average burnup of 80,000 MWD/MT and were finally discharged from the core at batch position 50. Thus, the flow of fuel balls can be illustrated by the following batch sequence:

55 (fresh fuel)	→60→.....→100,
100→1→6→.....	→46,
46→54→59→.....	→99,
99→2→7→.....	→47,
47→53→58→.....	→98,
98→3→8→.....	→48,
48→52→57→.....	→97,
97→4→9→.....	→49,
49→51→56→.....	→96,
96→5→10→.....	→50 (discharged fuel).

Due to the large number of burnup steps needed to perform the fuel shuffling scheme listed above, the computational cost is relatively expensive compared to a simple OTTO fuel cycle. It is recommended that convergency criteria for the diffusion calculations be loosened and a good initial isotopic concentration for each fuel batch be estimated before performing the burnup and diffusion calculations.

The analysis of the pebble bed core was carried out by utilizing the linked spectrum, neutronics, burnup, fuel management, and thermal hydraulics modules in the V.S.O.P. Code System (Reference 5-1). Flux solution calculations were performed for the R-Z model by means of a synthesis approach in

which one-dimensional diffusion computations in four energy groups were carried out in the radial and axial dimensions, and then linked together. The synthesized flux solution was then utilized for the burnup calculations in all core regions. The feed enrichments were adjusted and the flux solution and burnup calculations were repeated until a critical core with an acceptable radial power peaking was achieved for the equilibrium fuel cycle.

The calculations of the radial and axial power peaking factors were carried out by utilizing the BOLD VENTURE Code System (Reference 5-2). The isotopic concentration and cross section data for the equilibrium cycle were generated by the VSOP Code system and linked to the BOLD VENTURE Code System through interface codes. The radial and axial peak-to-average power ratios were computed from the two-dimensional flux distribution and the regionwise fissile fuel concentrations.

5.5 Core Performance

Neutronics calculations were carried out for both the OTTO core and the recirculation core to study the impact of recirculating the exposed fuel balls on the core performance parameters of the preconceptual MPBR. The key performance parameters are listed in Table 5-3. The results indicated that with the exception of the axial power distribution the recirculation of exposed fuel balls had only a small impact on the neutronic performance parameters. For the recirculation case, the axial peak-to-average power ratio was found to be 1.78 at the axial position of about one fifth of the core height from the top of the active core. The OTTO core had a similar asymmetric axial power profile. However, the peak power was significantly pronounced with a peak-to-average ratio of 4.51 which occurred at a core position of about one tenth of the core height from the top of the active core. Both power profiles are shown in Figure 5-2a. The discontinuities in power profile were due to the use of a fixed number of axial regions in the burnup calculations that resulted in the discontinuities of the fissile enrichments. These discontinuities can be eliminated by using a much larger number of axial zones, although the computational costs would increase accordingly.

The impact of axial power profile on the steady-state core temperature distributions can be seen in Figure 5-2b. For the recirculation core, the fuel temperatures at the ball center followed the profile of coolant temperature closely and the maximum fuel temperature was located at the coolant exit. In contrast, the differences between the fuel and coolant temperatures at the coolant inlet are substantially larger for the OTTO core due to relatively higher power densities, and the maximum fuel temperature was found to be located at the upper core zone, instead of at the coolant exit. By taking into account the radial power peaking factors as well as the batch peaking factors (i.e., the ratio of the maximum batch power to the average batch power that is non-negligible in the case of ball recirculation), the maximum steady state fuel temperatures were computed to be 1098°C and 1041°C for the recirculation core and the OTTO core, respectively. Both temperatures were within the design limit of 1250°C for steady state operation.

The impact of the axial power peaking on the maximum fuel temperature during the core heat-up transient has also been investigated for the modular pebble bed core. The results will be presented in Section 6.

A breakdown of fuel inventory at the equilibrium fuel cycle is listed in Table 5-4 for both the recirculation core and the OTTO core. Also listed are the requirements for the supply fuel and the content of the discharged fuel at the equilibrium fuel cycle. The fuel cycle cost was computed to be 6.648 mill/kwhe for the recirculation core. For the OTTO core, the fuel cycle cost was computed to be 7.026 mill/kwhe, which is slightly higher than that of the recirculation core.

TABLE 5-1
Design Specifications for the Modular Pebble Bed Core

<u>Parameters</u>	<u>Recirculation</u>	<u>OTTO</u>
Thermal power, MW	250	
Power density, MW/m ³	4.1	
Target burnup, MWD/MT	80000	
Heating of Helium, °C	300-950	
Helium inlet pressure, bar	50	
Plant efficiency, MWe/MWt	0.4	
Core height, m	9.2638	
Core diameter, m	2.8949	
Core height-to diameter ratio	3.2	
Fueling zones	Inner/Outer	
Number of flow channels in model	1/1	4/2
Channel radii, m	1.024/1.448	0.591/0.836/1.024/ 1.182/1.321/1.448
Channel relative residence time	10/10	10/10/10/10/11/15
Zone number for fresh feed balls	2	1 and 2
Feed enrichment, %	8.30	8.19/10.74
Percent U-235 in feed ball uranium	20	
Number of fuel ball circulation times	10	1
Reflectors	Top/Bottom/Radial	
Thickness, m	1.0/1.0/1.0	
Graphite density, g/cc	0.8/1.6/1.6	
Control poison Σ_a	0.0/0.0/1.0-4	
Fuel ball design		
Diameter, cm	6.0	
Graphite shell thickness, cm	0.5	
Graphite density, g/cc	1.70	
Coated particle design		
Diameter, μ m	200	
Density UO ₂ /ThO ₂ , g/cc	10.8/9.9	
Heavy metal loading, g/ball	7.0	
Coatings	C/C/SiC/C	
Thickness, μ m	100/30/30/40	
Density, g/cc	1.0/1.9/3.2/1.8	

TABLE 5-2

Reactivity Requirements for the Modular Pebble Bed Core

<u>LONG-TERM SHUTDOWN SYSTEM</u>		<u>Reactivity (%Δk)</u>
Maximum Accident-Water Ingress		1.1
Excess Reactivity for Load Following (100/40/100% Power) and Running-in Period Reserve		2.8
Requirements for Subcriticality		0.3
Long-Term Effects		
Temperature Defect	3.9	
Decay of Xe-135	2.9	
Decay of Pa-233 and Np-239	2.1	
Total		8.9
Uncertainty (10% of total long-term effects)		0.9
<u>Total Reactivity to be Controlled by Long-Term Shutdown System</u>		14.0

TABLE 5-3
Performance Parameters for the Modular Pebble Bed Core

<u>Parameters</u>	<u>Recirculation</u>	<u>OTTO</u>
K-effective	0.9980	0.9987
Average fissile enrichment $N_{fiss}/N_{HM}, \%$	4.38	3.98
Average fuel residence time, day	736.7	720.7
Average burnup, MWD/MT	79368	76749
Conversion ratio	0.444	0.412
Source neutrons/fissile absorption	2.030	2.009
Capture/fission in fissile material	0.236	0.246
Xe-override (100-40%), Δk	-0.022	-0.028
Maximum fast neutron dose ($>10^5$ eV), $10^{21}/\text{cm}^2$		
Disloaded pebble	2.03	1.94
Radial reflector per year	0.77	1.76
Top reflector per year	0.66	1.97
Bottom reflector per year	0.17	0.01
Power peaking factor		
Radial	1.169	1.124
Axial	1.782	4.509
Core maximum	2.071	5.028
Maximum power per pebble, kw/ball	1.86	3.82
Maximum helium temperature, °C	1060	998
Maximum fuel temperature, °C	1098	1041
Average fuel temperature, °C	775	910
Fuel cycle cost at equilibrium, mill/kwhe	6.648	7.026

TABLE 5-4
Fuel Inventory and Mass Balance at Equilibrium

	<u>Recirculation</u>	<u>OTTO</u>
Fuel inventory, kg/GWt		
U-233	58.47	75.83
U-235	307.45	246.96
Pu-239	13.61	13.87
Pu-241	3.93	6.14
Th-232	5214.23	4846.90
U-238	3028.99	3262.51
Pa-233	11.89	10.82
U-234	6.96	12.39
U-236	74.71	97.73
Np-239	0.70	0.78
Pu-240	8.03	10.92
Pu-242	3.09	6.52
Np-237	3.09	4.58
Total fissile	383.46	342.80
Total heavy metal	8735.15	8595.95
Fuel supply - discharge, kg/GWdt		
U-233	0.0 - 0.1405	0.0 - 0.1362
U-235	1.0413 - 0.1097	1.1440 - 0.2118
Pu-239	0.0 - 0.0213	0.0 - 0.0185
Pu-241	0.0 - 0.0086	0.0 - 0.0093
Th-232	7.2453 - 6.9112	7.0273 - 6.7521
U-238	4.2184 - 4.0067	4.6344 - 4.4216
U-236	0.0 - 0.1451	0.0 - 0.1535
Pu-240	0.0 - 0.0152	0.0 - 0.0169
Pu-242	0.0 - 0.0117	0.0 - 0.0122

FIGURE 5-1
RZ Computational Model for the Modular Pebble Bed Core

FIGURE 5-2
Axial Profiles of Temperature Distributions at Core Center and of Radially-Averaged Power Density Distribution for the Modular Pebble Bed Core

5.6 References

- 5-1 E. Teuchert, U. Hansen, and K. A. Hass, "V.S.O.P. - Computer Code System for Reactor Physics and Fuel Cycle Simulation," Jul - 1649 (1980).
- 5-2 D. R. Vondy, T. B. Fowler, and G. W. Cunningham III, "The BOLD VENTURE Computation System for Nuclear Reactor Core Analysis, Version III," ORNL-5711 (1981).

6.0 Shutdown Heat Removal

6.1 General

The analysis of the shutdown heat removal system was carried out to investigate the inherent safety of the modular pebble bed reactor. The principal objective of this analysis was to determine the fuel temperature distribution as a function of time following the loss of forced convection cooling transient. This information will be utilized in ongoing studies to determine fission product release rates and, thereby, provide a base for assessing the impact of the transient on the continued plant operations. By utilizing high heat capacity of the pebble bed core in conjunction with a relatively low power density and a small core diameter, it was deemed possible to design an inherently safe reactor module with a power rating of approximately 250 MWt.

In the heat-up transient following the loss of forced convection cooling of the core, the decay heat is removed by the heat transfer process radially outward from the core. That is, decay heat from the reactor is removed by inherent measures such as conduction and radiation to the vessel wall and then to redundant coolers outside the pressure vessel. In the case of a pressurized transient, the natural convection process in the pebble bed core provides for the redistribution of heat into the upper core region as well as for the rapid radial heat transport to the core/side reflector interface. On the other hand, the natural convection process can not make any significant contribution to heat transport in the case of a depressurized transient. Therefore, decay heat in the active core is transported outward only by conduction and radiation processes. Thus, the maximum fuel temperature in the loss of flow accident for a depressurized transient would be higher than for a pressurized transient.

6.2 Heat Removal Analysis

The analyses of decay heat removal were carried out by utilizing the THERMIX-2 code (Reference 6-1). THERMIX-2 was developed at KFA Laboratory in the Federal Republic of Germany for the two-dimensional steady state and transient thermal-hydraulics analysis of pebble bed reactors, and was made available to GE-ARSD through the U.S. - Germany Umbrella Agreements. The THERMIX-2 code consists of two major modules - THERMIX and KONVEK. The THERMIX module is used for the calculation of conduction and radiation processes. The KONVEK module is used to compute the fluid flow and temperature through the porous solid structure. The heat exchange between the solid and fluid is calculated in KONVEK according to empirical correlations. Coupled THERMIX/KONVEK calculations were carried out for the thermal-hydraulics analyses of the preconceptual modular core to obtain the temperature distributions for the steady state condition and for the pressurized transient. However, only the THERMIX module was utilized for the depressurized transient since convective heat transport was assumed to be insignificant.

The calculational model for the thermal-hydraulics analysis of the preconceptual modular pebble bed reactor was simplified to give an R-Z representation of the reactor layout, as shown in Figure 6-1. In the radial direction, the pebble bed core was surrounded by layers of side reflectors, carbon brick, core barrel, stagnant helium, and reactor pressure vessel. Surface coolers were located outside the pressure vessel and became the ultimate heat sink during the core heat-up transients. A constant temperature of 50°C was assumed for the surface coolers. In the axial direction. The reactor core was modeled by layers of helium, reflector, helium, carbon brick, core barrel, and carbon brick above the active core; and layers of reflector, helium, carbon brick, helium, and carbon brick below the active core. In general, the axial heat transport process was less significant than the radial heat transport process during the heat-up transients due to the location of main heat sink. Therefore, any inadequacy in modeling the regions above and below pebble core would not have a significant impact on the calculated maximum fuel temperatures.

The temperature distributions during heat-up transients were determined in two steps: (1) a steady-state calculation was performed with forced convection of helium coolant to obtain the initial temperature distributions, and (2) transient calculations were performed without forced convection utilizing a time-dependent decay power distribution which was computed as the heat source in the reactor core. The two-dimensional steady-state power distributions were obtained from neutronics calculations and used as input data to THERMIX-2. Time-dependent decay heat distributions were calculated from the input power density distribution and the reduction factors computed by THERMIX-2. Both pressurized and depressurized transients were analyzed for the recirculation core and the OTTO core.

The results of shutdown heat removal analysis are presented in Figures 6-2 and 6-3 for a pressurized transient and in Figures 6-4 and 6-5 for a depressurized transient. Transient temperatures were plotted at the core location where the maximum fuel temperature occurs; and at the bottom surface of the top reflector, the bottom of the active core, and the pressure vessel wall. Average transient temperatures were also determined for the active core. In a pressurized transient, the maximum fuel temperature was found to be 1620°C after shutdown time of 3 hours for the OTTO core, and 1291°C after 5.5 hours for the recirculation core. Both temperatures were found to be at the core centerline and about 93 cm below the top of the ball fill. The difference in the maximum fuel temperature can be attributed to the difference in the axial peak-to-average power ratio as described in Section 5. It is noted that the transient temperatures of the top reflector were very high and closely followed the peak fuel temperatures. The results indicated that the natural convection process in the pebble bed core provided for the redistribution of decay heat into the upper core region when the reactor remained pressurized.

In a depressurized transient, the maximum fuel temperature was found to be 2088°C after shutdown time of 13 hours for the OTTO core, and located at the core centerline about 93 cm below the top of the ball fill. For the recirculation core, it was found to be 1644°C after 27 hours and located at the core centerline about 278 cm below the ball fill. The

difference in maximum fuel temperature between the OTTO and recirculation core was again attributed to the difference in the axial power peaking factor. It is noted that the maximum temperature in the top reflector region was limited to approximately 1200 and 700°C for the OTTO and recirculation core, respectively. This is due to the absence of natural convection so that axial heat transport is carried out only by the radiation process in the void region and by conduction in the solid.

The maximum temperatures of the reactor pressure vessel for both the pressurized and depressurized transients were found to be less than 400°C. It was felt that, by limiting the vessel temperature to less than 400°C, the integrity of pressure vessel can be maintained and the replacement of pressure vessel would not be necessary even after the occurrence of the worst accident.

A summary of the results of the decay heat removal analysis is given in Table 6-1. In general, the OTTO core yields higher fuel temperatures than the recirculation core due to higher axial peak-to-average power ratio. Also, higher fuel temperatures were found in a depressurized transient than in a pressurized transient because of lack of natural convection to transport the decay heat to active core boundaries. In all cases, the maximum fuel temperatures were found to be within the design limit of 2200°C as specified in the preliminary design requirements and guidelines for the modular pebble bed reactor (Reference 6-2).

TABLE 6-1
Maximum Transient Fuel Temperatures for the
Modular Pebble Bed Reactor

	Fuel Temperature (°C)	Shutdown Time (hr)	Axial Location Below Ball Fill (cm)
<u>Pressurized Transient</u>			
OTTO	1620	3.0	93
Recirculation	1291	5.5	93
<u>Depressurized Transient</u>			
OTTO	2088	13.0	93
Recirculation	1644	27.0	278

FIGURE 6-1
AN R-Z CALCULATIONAL MODEL FOR THE TRANSIENT THERMAL-HYDRAULIC
CALCULATIONS OF PEBBLE BED REACTORS

FIGURE 6-2
TRANSIENT TEMPERATURES FOR THE MODULAR PEBBLE BED
REACTOR SYSTEM -- PRESSURIZED, OTTO

FIGURE 6-3
TRANSIENT TEMPERATURES FOR THE MODULAR PEBBLE BED
REACTOR SYSTEM -- PRESSURIZED, RECIRCULATION

FIGURE 6-4
TRANSIENT TEMPERATURES FOR THE MODULAR PEBBLE BED
REACTOR SYSTEM -- DEPRESSURIZED, OTTO

FIGURE 6-5
TRANSIENT TEMPERATURES FOR THE MODULAR PEBBLE BED
REACTOR SYSTEM -- DEPRESSURIZED, RECIRCULATION

6.3 References

- 6-1 K. Petersen, H. Barthels, and R. Schulten, "Natural-Convection After-heat Removal from the Pebble Bed HTGR," Trans. Am. Nucl. Soc. 24, 304 (1976).
- 6-2 G. R. Pflasterer, "Modular Pebble Bed Reactor Design Requirements and Guidelines," Letter to W. L. Von Flue, XL-898-20025, dated January 29, 1982, General Electric Company.

7.0 MAINTENANCE AND IN-SERVICE INSPECTION (ISI) REQUIREMENTS

7.1 Introduction

A preliminary assessment of the maintenance and in-service inspection (ISI) requirements for the Modular Pebble Bed Reactor (MPBR) reforming system has been completed. The specific design studied uses a steel reactor vessel and a separate steel component vessel. The two vessels are connected by coaxial pipes in an arrangement commonly called "side-by-side." A detailed description of the MPBR and the nuclear heat source (NHS) components are presented in Section 4.0.

The objective of this report is to describe the maintenance and ISI requirements that have been identified and to briefly present the approach to be used and the times required for maintenance and ISI of the MPBR. Primarily, the report addresses the scheduled maintenance and ISI of the NHS components. Maintenance requirements for the balance-of-plant (BOP) components are the same as for the HTGR reforming plant, which has been evaluated by Bechtel Group Incorporated (BGI) and reported in Reference 1.

In assess the MPBR maintenance and ISI requirements, paramount consideration was given to:

- o Code of Federal Regulations, which requires that nuclear plant operators and maintenance personnel be exposed to only radiation levels that are "as low as reasonably achievable (ALARA)."
- o American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Division 2, "Rules for In-Service Inspection and Testing of Components of Gas-Cooled Plants."

Other considerations are:

- o Accomplish maintenance and inspections in minimum time periods.
- o Accomplish as many maintenance and inspection activities in parallel as possible.

In addition to the above, features that would improve and/or expedite maintenance and inspection were identified for inclusion in the NHS component and plant arrangement designs.

The approach followed in making the assessment was to first assemble pertinent background data and operating experience of existing pebble bed reactors: the German AVR and THTR plants. Some components used in Peach Bottom I and Fort St. Vrain are also used in the MPBR and their maintenance and ISI was also considered.

Due to basic design differences between these plants and the MPBR, it is required that maintenance and inspection methods and procedures be different than those used in the past. The conceptual design of new maintenance equipment and procedures indicate improvements in efficiency and reliability and reduced time requirements; however, the new equipment and procedures introduce uncertainties in the maintenance time estimates. For this reason, the maintenance times presented in this report are estimated times for the "Nth" MPBR plant and they are probably optimistic for the first plant.

Whenever possible, direct contact (hands-on) maintenance procedures are preferred. Often this can be accomplished by using temporary shielding of steel or lead. Radiation levels can be reduced by a factor of 10 by using 1.6 inches of steel or 0.9 inch of lead shielding. Whenever the radiation exposure can be reduced to <10 mrem/hour, direct contact maintenance is employed. Handling of core components, helium circulator

stator and rotor, control rods and removal of a reformer require remotely operated equipment and shielded transfer casks.

At this time, the radiation levels about the MPBR vessel and the component vessel have not been determined. However, due to the distance of separation of the components from the core, radiation due to activation and to streaming should be virtually eliminated and radiation due to plate-out on the components reduced by a small amount. Therefore maintenance and ISI of the reformer, steam generator and circulator can use the same type of equipment and procedures as planned for the modular HTGR, as reported in Reference 2.

Much of the maintenance and ISI activities are the same for the MPBR as for the modular HTGR. The major difference in maintenance and ISI is refueling the core. The modular HTGR uses batch (whole core) refueling on a four-year interval; whereas, the MPBR uses "on-line" refueling, i.e., a small percentage of the total number of fuel elements (spheres) are removed from the core and replaced on a daily basis. For this reason, maintenance and ISI activities are not scheduled to be in parallel with refueling, but are scheduled to coincide with replacing catalyst in the reformer, which is scheduled for every four years, as shown in Table 7-1.

7.2 Maintenance, Inspection and Availability

Since normal operation of the modular pebble-bed reactor-reforming (MPBR-R) plant provides for on-line refueling, scheduled outages are "keyed" to the replacement of catalyst in the reformer instead of batch refueling as used for the prismatic core of the MRS-R plant. It is assumed that the

TABLE 7-1
SCHEDULED OUTAGE TIME ESTIMATE
FOUR-YEAR INTERVAL

<u>Activity</u>	<u>Hours</u> ⁽¹⁾
1. Shutdown Reactor	20.5 ⁽²⁾
2. Depressurize Reactor	2.5
3. Preparation (put ISI and maintenance equipment in place)	24.0
4. Remove and Replace Reformer Catalyst	297.0
In parallel with 4:	
Service: Control Rods & Drives	
Helium Circulator	
ISI: Coolant Boundary	
Penetration Seals	
Steam Generator	
Core Barrel - Non-fuel Side	
Core Support - Non-fuel Side	
Instrumentation	
5. Replace High Temperature Filter/Absorber	24.0
6. Cleanup (Close all penetration, test seals, and remove all ISI and maintenance equipment)	24.0
7. Pressurize Reactor, Steam Generator and Reformer	3.0
8. Start-up Reactor	<u>20.0</u> ⁽³⁾
TOTAL	415.0

(1) Includes 80% efficiency factor for equipment plus 80% efficiency for labor.

(2) Includes 8-hour dwell for catalyst oxidation.

(3) Includes 6-hour dwell for catalyst reduction.

reformer catalyst has a life of four years so as much maintenance and ISI activities as possible are scheduled on four-year intervals.

7.2.1 Maintenance and ISI of Nuclear Heat Source (NHS) Components

These are the components in contact with the reactor core and/or the helium coolant plus the reactor vessel cavity cooling system.

7.2.1.1 Reformer

The reformer is made of 769 catalyst tubes, 64.5 mm O.D. - 49.3 mm I.D. x 14 M long. The tube bundle is surrounded by a shroud. Helium coolant flows up around the outside of the tubes out the top of the shroud and then down through the steam generator. The catalyst tubes are welded to a tubesheet that is installed and sealed to the component pressure vessel. Process gas enters on top of the catalyst tube sheet and flows down through the catalyst and then up through a "pigtail" tube located in the center of the catalyst. The pigtails are fastened to a tube sheet above the catalyst tube sheet which serves as the bottom of the syngas plenum. The plenum has a lid which prevents the process gas from mixing with the syngas. The syngas flows from the plenum through a pipe which connects the plenum to the component pressure vessel and then through the syngas outlet nozzle.

7.2.1.1.1 Reformer Catalyst Replacement

The reformer catalyst is scheduled for replacement every four years. The overall time required is 342 hours. A detailed procedure for catalyst replacement is presented in Table 7-2. The radiation level above the catalyst tube sheet is expected to be 160 mrem/hour, maximum. To reduce this level for contact maintenance, the use of a shielded work platform is planned (see Figure 7-1).

The reformer design includes a 10 percent margin for tube failures and tubes are tested and maybe plugged during each catalyst replacement. Time for this activity is included in Table 7-2.

TABLE 7-2
REMOVE & REPLACE REFORMER CATALYST
TIME⁽³⁾ (HOURS)

NOTES:

- 1) 100 Spare Pigtail and Catalyst Assemblies are ready for insertion.
- 2) In parallel with activities 9 through 13, separate spent catalyst from pigtails and assemble fresh catalyst to pigtails.
- 3) Includes 80% efficiency factor.
- 4) Estimated maximum number of failures.
- 5) See Figure 7-1.

		MPBR
		1 Reformer
		<u>Tubes=769x14 m lg.</u>
<u>Activity</u>		
1. Shut down reactor (1742°F to 242°F at 120°F/Hr plus 8 hours for catalyst oxidation)		20.5
2. Depressurize reactor (680 psi to - 3 in. H ₂ O) and the reformer		2.5
3. Purge reformer of nitrogen and back-fill with air		3.0
4. Remove vessel head		4.0
5. Remove process gas plenum lid		3.0
6. Disconnect process gas pipe		0.3
7. Remove pigtail tubesheet (unfasten pigtails)		28.0
8. Install shielded work platform		8.0
9. Remove pigtails with catalyst	}	186 ⁽⁵⁾
10. Vacuum reformer tubes		
11. Vacuum test 50% of reformer tubes and tube-to-tubesheet welds (ISI)		
12. Plug ⁽⁴⁾ leaking tubes and/or repair tube-to-tubesheet welds, inspect & test		
13. Insert pigtails with fresh catalyst		

TABLE 7-2 (cont'd.)

Figure 1. REMOVE & REPLACE REFORMER CATALYST

	(H O U R S)																								
<u>Activity</u>	<u>0</u>	<u>10</u>	<u>20</u>	<u>30</u>	<u>40</u>	<u>50</u>	<u>60</u>	<u>70</u>	<u>80</u>	<u>90</u>	<u>100</u>	<u>110</u>	<u>120</u>	<u>130</u>	<u>140</u>	<u>150</u>	<u>160</u>	<u>170</u>	<u>180</u>	<u>190</u>					
No of Tubes = N = 769 x 14 m lg.	<u>250 MWt MODULAR REACTOR</u>																								
9. Remove Pigtails (Nx8.4)											<u>108</u>														
10. Vacuum Reformer Tubes (Nx8.4)	<u>8</u>												<u>116</u>												
11. Test Tubes (N/2x6)				<u>16</u>												<u>66</u>									
12. Plug Tubes (3x6 hrs.)									<u>74</u>		<u>92</u>														
13. Replace Pigtails (Nx12.6)				<u>24</u>												<u>186</u>									

TABLE 7-2 (cont'd.)
REMOVE & REPLACE REFORMER CATALYST
TIME⁽³⁾ (HOURS)

	MPBR
	1 Reformer
	<u>Tubes=769x14 m lg.</u>
14. Remove shielded work platform	8.0
15. Replace pigtail tubesheet, fasten pigtails and test seals	42.0
16. Connect process gas pipe, test seals	0.5
17. Replace process gas plenum lid, test seals	4.5
18. Replace vessel head, test seals	6.0
19. Purge the reformer of air and back-fill with nitrogen	3.0
20. Pressurize the reactor and reformer (-3 in. H ₂ O to 680 psi)	2.7
21. Start up (from 190°F to 1742°F at 120°F/hr plus 6 hours for catalyst reduction)	<u>20.0</u>
Total Hours	342.0
Days	14.25

SHIELDED WORK PLATFORM REFORMER CATALYST REMOVAL AND REPLACEMENT

FIGURE 7-1

Time is also included in the table for inspection of the catalyst tubes using an eddy current probe. At present, it is recommended that one-half the tubes be inspected during each refueling outage.

7.2.1.1.2 Reformer Removal and Replacement

Due to the high temperatures ($>1700^{\circ}\text{F}$) imposed on the reformer and lack of empirical data as to failure rates, it is assumed that all reformers will be replaced once during the life of a MPBR-R plant. The replacement of a reformer requires that the core be unloaded of radioactive fuel and the coolant flow stopped. The reformer, after being unfastened from the vessel, is raised through an isolation valve into a shielded transfer cask (see Figure 7-2) and moved to a decontamination facility. After cleaning, it is refurbished or dismantled for disposal as radioactive solid waste. The shielded transfer cask is used to insert a replacement reformer into the vessel. The procedure to remove and replace a reformer is presented in Table 7-3; the activity is estimated to take 508 hours.

7.2.1.2 Steam Generator (S/G)

The S/G is constructed as an integral part of the component pressure vessel. The feedwater inlet tube sheet and the steam outlet tube sheet are welded into nozzles which are in turn welded into the side of the vessel. Radiation levels on the outside of these tube sheets are expected to be 100 mrem/hour, maximum, hence temporary shielding will be required for testing tubes and for tube plugging. It is recommended that 50% of the S/G tubes be tested by eddy current probe during every outage for replacement of reformer catalyst.

It is possible that all steam generator tube leaks may occur between scheduled outages and an unscheduled outage will be necessary for finding the leak(s) and plugging the tube(s). Table 7-4 presents a time estimate for a forced outage to plug steam generator tubes. The time for eddy current probe test and tube plug welding is based on experience of B&W

REFORMER REMOVAL

FIGURE 7-2
7-11

TABLE 7-3
REMOVE AND REPLACE REFORMER

ASSUMPTIONS:

- 1) Direct contact of installation fasteners is possible.
- 2) Core is unloaded and coolant flow is stopped during the time the reformer position is vacant.
- 3) A replacement reformer is available and ready for installation.

	MPBR 1 Reformer Tubes=769x1.4 m 1g. Time (Hours)
1. Shut down reactor. (Reduce coolant temperature from 1742°F to 242°F at 120°F/hr. plus 8 hours for catalyst reduction)	20.5
2. Depressurize reactor (680 psi to -3 in. H ₂ O) and the reformer.	2.5
3. Prepare to unload the core. (Set fuel handling system for rapid discharge).	4.0
4. Unload the core.	205.0
In parallel with 4, prepare the reformer for removal:	
a. Purge the reformer of process gas and backfill with air.	
b. Remove vessel head.	
c. Remove reformer process gas plenum lid.	
d. Disconnect process gas pipe.	
e. Remove tubesheet to vessel fasteners.	
f. Attach lifting yoke.	
g. Install spool piece - test seal.	
h. Install isolation valve - test seal.	
i. Position and seal transfer cask to isolation valve.	
5. Shut off He circulator	0.3
6. Purge the transfer cask, valve spaces and space above the reformer tubesheet of air and backfill with helium	3.0
7. Raise the reformer into the transfer cask	0.5
8. Close the transfer cask valve and close the isolation valve	0.5
9. Disconnect transfer cask from the isolation valve	0.5
10. Move the transfer cask with reformer to repair facility	1.0

TABLE 7-3 (cont'd.)
REMOVE AND REPLACE REFORMER

	<u>Modular 250 MWt</u>
11. Open transfer cask valve	0.3
12. Lower the reformer into the decontamination station	0.5
13. Move the transfer cask to the new reformer	0.5
14. Attach lifting yoke to new reformer	1.5
15. Raise reformer into transfer cask and close cask valve	0.8
16. Move the transfer cask to the vessel, fasten and seal it to the isolation valve	2.0
17. Purge the transfer cask and valve spaces of air and backfill with helium	2.0
18. Open the transfer cask valve and open the isolation valve	0.5
19. Lower the new reformer into the vessel	0.8
20. Release the lifting yoke from the crane	0.2
21. Disengage the transfer cask from the isolation valve and move it to storage	1.5
22. Disengage the isolation valve from the spool piece and move it to storage	1.0

TABLE 7-3 (cont'd.)
REMOVE AND REPLACE REFORMER

	<u>Modular 250 MWt</u>
23. Disengage the spool piece from the vessel and move it to storage	1.0
24. Disengage the lifting yoke from the reformer and move it to storage	0.8
25. Secure the reformer to the vessel with fasteners and test the seals	16.0
26. Start up He circulator	0.3
27. Reload the core	205.0
In parallel with 27, complete installation of the reformer.	
a. Connect process gas pipe - test seals	
b. Replace the process gas plenum lid - test seals	
c. Replace the vessel head - test seals.	
d. Purge the reformer of air and backfill with nitrogen.	
28. Prepare the reactor for startup. (Set fuel handling system for normal operation.)	4.0
29. Pressurize the reactor and the reformer to 680 psi.	3.0
30. Check out and test reactor instrumentation.	8.0
31. Start up reactor. (Raise temperature from 92°F to 1742°F at 120°F/hr plus 6 hours for catalyst reduction.)	<u>20.5</u>
Total Hours	508.0
Days	21.1

TABLE 7-4
STEAM GENERATOR TUBE PLUGGING
MRS-R

<u>Activity</u>	<u>Time (Hours)</u>
1. Shut down reactor, initial dry out	10.7
2. Depressurize	2.5
3. Open component vessel cavity	4.0
4. Put access equipment in place (ladders, scaffles, etc.)	4.0
5. Health physics check	2.0
6. Put maintenance equipment in place (shielding, tube leak detector, pipe uncoupling tools, etc.)	4.0
7. Uncouple and remove 2 pipe spools	4.0
8. Test 130 tubes using eddy current probe (15 min. per tube - 282 ft. long)	41.0
9. Put plug welding equipment in place	4.0
10. Plug tubes (assume 3 tubes and 6 plugs)	6.0
11. Inspect and test plug welds (repair if needed)	6.0
12. Remove plug welding and inspection equipment	4.0
13. Replace pipe spools (test seals)	6.0
14. Remove maintenance equipment	4.0
15. Remove access equipment	4.0
16. Close component vessel cavity	4.0
17. Pressurize reactor	3.0
18. Start up reactor (hold at various temperatures to dry out core)	16.8
	<hr/>
Total	130.0
Contingency (15%)	20.0
Efficiency Factor (70%)	<u>64.0</u>
Hours	214.0
Days	8.92

and C-E for inspection and tube plugging of steam generators used in their PWRs.

7.2.1.3 Fuel Handling System

The fuel handling system comprises three parts: (1) new fuel handling, including insertion of new fuel spheres into the core; (2) recycle fuel handling, which includes inspection for broken spheres, measurement of burn-up and reloading into the core; and (3) spent fuel handling, which includes storage, packaging and shipment off site. A detailed description of each part of the fuel handling system is given in Section 4.3.

7.2.1.3.1 Fuel Handling System Maintenance

Since all components of the fuel handling system are exercised daily, there are no scheduled outages for maintenance of this system. One valve in each fuel feed line and in the one fuel discharge line nearest the core are safety (or repair) valves. This valve is normally "open" and it is closed whenever a component in a line needs to be repaired or replaced.

Both fixed and temporary shielding are used to limit radiation exposure to workers when servicing components in the fuel handling system. Normally the fuel handling system works 14 hours per day, which allows 10 hours per day for maintenance activities. This can be extended to a 20-hour period without interfering with the daily rate of fuel flow into and out of the core. If a repair outage greater than 20 hours is required, the system can run for 24 hours per day when returned to operation until the core equilibrium conditions are restored. For example, the equilibrium conditions of the core can be re-established following a down time of seven days by running the fuel handling system 24 hours per day for 10 days.

The most frequent maintenance requirements of the fuel handling system is the repair of ball valve actuators and the replacement of the valves. Since all the operating (non-safety) valves can be repaired or replaced during reactor operation, these activities have no effect on plant availability.

If a safety valve must be replaced, the reactor must be shut down and depressurized. A glove box-type air lock must be built about the valve and the valve is removed and replaced by working through glove ports. For replacement of the safety valve in the fuel discharge line, the air lock must include shielding to protect the workman from radiation. The air lock has a helium atmosphere at a pressure slightly greater than the coolant in the core. It is estimated that during the 40-year life of the MPBR, the safety valve of the discharge line will need to be replaced once and two of the seven feed line safety valves will also need to be replaced. The time required would be 108 hours for the discharge line valve and 68 hours for each feed line valve, as reported in Reference 3.

7.2.1.3.2 Fuel Handling System ISI

Since all components of the fuel handling system are actuated daily, except for the safety valves, they are continuously subjected to inspection. All seals are buffered and continuously monitored. The safety valves are actuated to full "closed" and pressure tested every three months as required by the ASME Code, Section XI, Division 2. The fuel handling lines connections to the reactor pressure vessel are volumetrically inspected 100% in every 10-year (or less) time period.

7.2.1.4 Control Rod and Control Rod Drives

Control rod and control rod drives are described in Section 4.2.6. It is estimated that the life of all control rods will be eight years, so one half will be replaced every four years in parallel with replacing the reformer catalyst. The control rod and drive is designed to be removed and replaced as a common assembly. The equipment required will be almost identical to the equipment used for Fort St. Vrain control rod removal and replacement, i.e., an isolation valve and a shielded transfer cask. The reactor must be shutdown and depressurized. The time required is five to eight hours per control rod and drive assembly as reported in Reference 3.

By replacing 50% of the control rod and drive assemblies every eight years, the ISI requirements are automatically satisfied. The mounting

connection seals are tested upon installation and continuously monitored during reactor operation. The control rod drives are tested by actuation every three months. The control rod penetration nozzles are visually inspected at frequent intervals.

7.2.1.5 Circulator

The coolant circulator is described in Section 4.2.5. It is mounted to a penetration in the bottom center of the component pressure vessel. The circulator consists of a two-stage rotor and stator assembly, including seals, a pressurized water bearing, and a variable-speed electric drive motor.

Radiation levels on the outside of the stator installation flange and at the midpoint of the motor should be equal or less than the circulator used for the MRS plant, i.e., 16 and 3 mrem/hr., respectively, hence removal and replacement of either the bearing and seal cartridge or the motor can be accomplished by contact maintenance without supplementary shielding.

The high pressure outer water seal of the water bearing and seal cartridge has an expected life of four to six years. It should be replaced every four years in parallel with replacing the reformer catalyst. It can be removed and replaced in less than one day. Electric motors of this size have a life of 12 to 15 years and it too can be removed and replaced in less than one day.

The stator and rotor assemblies are designed for 40-year life. Hopefully, neither will be removed during the life of the plant. To remove a stator and/or rotor assembly will require a shielded transfer cask and a remotely-operated removal/insertion machine. The reactor will have to be shut down and depressurized. The time required would be about 210 hours total: 30 hours each for shutdown and startup and 150 hours to replace the circulator components.

ISI of the circulator will be accomplished during every shutdown for replacement of the reformer catalyst. Water leakages of the bearing and

seal will be continuously monitored and the reactor shutdown for circulator repair if leakage exceeds a predetermined acceptable levels.

7.2.1.6 Cavity Cooling System (CCS)

There are no scheduled outages for this system. The system is tested during every shutdown of the reactor. The system includes three separate sets of cooling coils per module and three hardened heat sinks per set of four modules. Each cooling circuit uses three 50% redundant pumps. The only maintenance anticipated is the replacement of pumps and leaking valves or the repair of a pipe break. These repairs will be very infrequent and they can be made during scheduled reactor outages or during reactor operations.

7.2.1.7 Instrumentation

All instrumentation is sufficiently redundant that failures should rarely require a forced outage. There is sufficient time during scheduled outages. to repair and/or replace all anticipated failed instrumentation.

7.2.1.8 Helium Purification System

The helium purification system comprises small diameter piping, valves, three 50% compressors and a high temperature filter absorber. The filter absorber is replaced every four years as well as repair or replacement of valves and compressors. The total time required is anticipated to be 24 hours.

7.3 In-Service Inspection (ISI) Requirements

At present, there are no existing ASME or other codes or NRC guidelines directly applicable to the gas-cooled pebble-bed reactor. "On-line" refueling precludes ISI of the reactor internals and core support structure from the fuel side. Inspection of the inside surface of the core barrel and core support plate will require the removal of all fuel, top,

bottom and radial reflectors and thermal insulation. To accomplish this inspection on the schedule(s) required by the ASME Code Section XI, Division 2 (100% per 10-year period) would require three or four outages of such long duration that all availability benefits afforded by on-line refueling are negated. The only appropriate course of action appears to be the forming of a task force to write a new Division to ASME Section XI, which would be applicable to pebble-bed reactors.

7.3.1 Primary Coolant Boundary

All outer surfaces, including welds of the reactor vessel, component vessel and outer pipe of the coaxial pipes connecting the vessels are readily accessible for inspection. All seals of all penetrations are buffered with helium and continuously monitored as are the vessel to head flanges and the seals between the process gas and coolant and between the water/steam and coolant.

7.3.2 Reactor Internals

Inspect ports are available for visual and volumetric inspection of the outer surfaces of the core barrel and core support structure. The inner surfaces cannot be inspected as explained in Section 7.3, above.

A visual inspection of the inside surfaces of the top and upper radial reflectors above the fuel bed may be possible using a periscope and air-lock mounted to the fuel feed lines. At present, it is planned that, if these reflectors become damaged due to excessive temperatures or neutron flux, the fuel would be unloaded and the rest of the core would be removed as one assembly. The exact procedure and the equipment for performing this operation requires a more definitive reactor design and it is recommended as a technical issue to be addressed in future study.

7.4 References

- 7-1 Letter Report, T. M. Sweeney, BGI to L. D. Mears, GCRA, Subject: "GRCA/HTGR Project - Job 14508, Monolithic/Modular HTGR Availability Study Update BL-196," dated August 20, 1982.

- 7-2 C. R. Davis, "A Preliminary Assessment of the Maintenance and In-Service Inspection Requirements for Modular Reactor Systems," G. E. Report No. XL-848-00568/P2773, dated July 1982.

- 7-3 ORNL-5694, "Comparative Evaluation of Pebble Bed and Prismatic Fuel at High Temperature Gas Cooled Reactors," dated January 1981, pp. 3-97 through 3-112.

8.0 PRELIMINARY SAFETY AND LICENSING EVALUATION

8.1 Introduction

A preliminary evaluation of the inherent safety and safety features of the modular pebble-bed reactor (MPBR) concept has been conducted. The evaluation identified safety issues which are common between the MPBR and the modular reactor system (MRS) to take advantage of previous work on the MRS. Safety issues peculiar to the MPBR have been given special attention. The specific issues covered include:

- a. Fission product release for the recirculation and OTTO fuel cycles
- b. Xenon instability
- c. Reactor shutdown system
- d. Applicability of GDC and the monolithic HTGR-PH 1170 MWt Nuclear Safety Plant Specification, Document No. 950396, Issue One.

These issues are evaluated in Section 8.2. The evaluation resulted in the following main conclusions:

- a. The recirculation fuel cycle has a definite safety advantage over both the OTTO cycle and MRS in terms of fission product release in a core heatup accident. It was estimated that the fission product release of a recirculation cycle is at least 7 times less than either of the other concepts.
- b. There is no significant difference in the fission product release in a core heatup accident between the OTTO cycle and the MRS.

- c. There are safety and economic advantages in reducing the active core height of the OTTO cycle to about half its current height. Since almost all of the fuel burnup occurs in the top half of the core, the bottom half is virtually spent fuel which is being stored in the reactor vessel. Removal of the spent fuel in a separate fuel tank will result in reduction of the reactor vessel height, control rod length, side reflectors and shield, and decay heat load in the reactor vessel. The helium outlet temperature penalty for reducing the core height to half its current value is about 20°C.
- d. On a worst case (licensing and decay heat removal design) basis, both the OTTO and recirculation MPBR have a better safety margin in terms of the maximum core fission product inventory than the MRS. The maximum fission product inventory for the MPBR concepts is about half that of the MRS concept. This inventory will be less by another factor of 2 if the bottom half of the active core is removed in the OTTO cycle.

Section 8.2 contains other conclusions as appropriate for the safety and licensing issues identified. Section 8.3 provides a list of recommendations to enhance the safety and licensability of the MPBR concepts.

8.2 Safety and Licensing Issues and Their Evaluation

8.2.1 Fission Product Release in Core Heatup Accidents

Fission product release from TRISO-coated particles depends on the following parameters:

- a. fuel temperature history
- b. thermal power

c. burnup

Fuel failure and fission product diffusion depend on the fuel temperature. The inventory of short-lived fission products is roughly proportional to the thermal power. The inventory of long-lived fission products is roughly proportional to the burnup.

Reference 8.1 contains an analysis of the MRS core heatup accidents. The reference gives estimates of core temperatures and fission product release in these accidents. Core temperatures only have been established for the MPBR heatup accidents in Section 6. These temperatures, thermal power, and burnup have been used for this preliminary evaluation to obtain a rough estimate of the fission product release in the MPBR heatup accidents as discussed below.

The fuel failure fraction obtained from curve fitting of test data has been given by GA by the following expressions:

$$f = 1.397 \times 10^{-5} \exp(.0042T); T \leq 2050^{\circ}\text{C} \quad (1)$$

$$\text{or } \frac{dT}{dt} > 15^{\circ}\text{C/hr}$$

$$= 1.397 \times 10^{-5} \exp(.0042\{T + 17.5t\}); T > 2050^{\circ}\text{C} \quad (2)$$

$$\text{and } \frac{dT}{dt} \leq 15^{\circ}\text{C/hr}$$

The exponential dependence on the temperature implies that the failure fraction will be primarily contributed by the high-temperature region in the core. Table 8.1 contains the maximum average core temperatures and the maximum temperature at the core axial location where the temperature is greater than at any other location. Since the spatial distribution of the core temperature in the course of a core heatup accident has not been available, the following approximate expression has been developed to estimate the MPBR fuel failure fraction from the MRS fuel failure fraction

Table 8-1

Parameters Affecting Fission Product Release

	MPBR		MRS
	Recirculation Cycle	Otto Cycle	
Maximum Core Temp, $T_{\max}$ ($^{\circ}\text{C}$)	1644	2088	2093
Location (cm from Top)	278	93	@ mid plane
Time (hr)	27	13	40
Maximum Average Core Temp, T_{av} ($^{\circ}\text{C}$)	1050	1070	1371
Time (hr)	20	5	30
Power Peaking Factor at Maximum Temp	2	5	3
Maximum Burnup at Maximum Temp (fraction of end of life)	.48	.45	1

$$f_{\text{MPBR}} \leq f_{\text{MRS}} \frac{[T_{\text{av}}/T_{\text{max}}]_{\text{MPBR}}}{[T_{\text{av}}/T_{\text{max}}]_{\text{MRS}}} \exp(.0042 \Delta T), \quad (3)$$

where

$$\Delta T = [T_{\text{max}}]_{\text{MPBR}} - [T_{\text{max}}]_{\text{MRS}} \text{ } ^\circ\text{C}$$

Equation (3) overestimates the MPBR failed fuel fraction. Applying the equation for T_{max} and T_{av} of Table 8.1 gives:

$$f_{\text{MPBR}}(\text{recirculation cycle}) \leq .15 f_{\text{MRS}} \quad (4)$$

$$f_{\text{MPBR}}(\text{OTTO cycle}) \leq .77 f_{\text{MRS}} \quad (5)$$

In terms of failed fuel fraction, therefore, the MPBR-recirculation cycle is better than the MPBR-OTTO cycle. Both cycles are better than the MRS.

Using the peaking factors and burnups of Table 8.1, the following estimates of the activity released in curies, A, have been obtained;

$$\begin{aligned} A_{\text{MPBR}}(\text{recirculation cycle}) &\leq .1 A_{\text{MRS}} && \text{for short-lived fission products} \\ &\leq .072 A_{\text{MRS}} && \text{for long-lived fission product max release} \\ &\leq .15 A_{\text{MRS}} && \text{for long-lived fission product expected release} \\ A_{\text{MPBR}}(\text{OTTO cycle}) &\leq 1.28 A_{\text{MRS}} && \text{for short-lived fission products} \\ &\leq .347 A_{\text{MRS}} && \text{for long-lived fission product max release} \\ &\leq .62 A_{\text{MRS}} && \text{for long-lived fission product expected release} \end{aligned}$$

The above results show a definite safety advantage of the MPBR recirculation cycle over both the OTTO cycle and the MRS concept. The primary reason for this advantage is the less-peaked power distribution which leads to lower temperatures in the course of the core heatup accident. In contrast, the OTTO cycle results in a power distribution which is peaked in the top half of the active core. As seen from Figure 8.1, almost all of the fuel burnup takes place in the top half of the core in the OTTO cycle. The lower half contains virtually spent fuel. This leads to a reduced heat loss through the reactor vessel side walls but more heat is lost through the top reflector. It seems that these two effects cancel each other to the extent that core temperatures during a heatup accident are comparable to those of the MRS.

The above results indicate that the safety of the MPBR-OTTO cycle and MRS are comparable. However, the fact that the bottom half of the active core is spent fuel which is left to decay for about one year offers an advantage over the MRS and suggests some design options which have economic and safety merits. Two such options are to use burnable poison to flatten the axial power distribution or to cut the height of the active core to half its current value. Table 8.2 contains these options and their pros and cons.

8.2.2 Xenon Instability

The analysis and implications of Xe instability in the MPBR could differ from those in the MRS. While the question of instability in the MRS is primarily related to control rod motion, analysis for the MPBR should also include the pebble motion. As pebbles move down the reactor vessel, they see different neutron flux levels, which results in local Xe and flux oscillations. Through its life in the reactor vessel, the pebbles will be subjected to these oscillations and those resulting from control rod motion. Whether the reactor system will reach a stable state, and whether these oscillations will result in significant fuel failure need to be investigated.

Figure 8-1 - Average Axial Fuel Burnup in MPBR

Table 8-2

OTTO Cycle Design Options

Option	Advantages	Disadvantages
Leave as is	Decay of spent fuel for one year	
Use burnable poison	<ul style="list-style-type: none"> - Flattened axial power - Reduced fuel failure - Reduced fission product release 	<ul style="list-style-type: none"> - Reduced neutron economy - Potential mishaps with burnable poison control
Reduce Active Core height by half its current value	<ul style="list-style-type: none"> - Reduced vessel size - Reduced control rod length - Reduced side reflectors and shield - Reduced decay heat load in vessel 	<ul style="list-style-type: none"> - Reduced outlet helium temperature by 20°C - Redesign interface with component vessel - Reduced heat transfer surface across the vessel walls - Higher top and bottom reflector temperatures in core heatup accidents

8.2.3 Reactor Shutdown System

The reactivity control primary and secondary systems will be designed to meet GDC 20 through 29 of 10CFR50, Appendix A. Section 5.3 provides an estimate of 14.0% Δk of the reactivity requirements. This should be interpreted as the combined control system capability per GDC 27. GDC 26 states that "one of the systems shall be capable of holding the reactor core subcritical under cold conditions." GDC 26 applies for normal operation including anticipated operational occurrences, planned and normal power changes including Xe burnout. Notice that no postulated accident conditions are required for cold shutdown. A possible split of the reactivity requirement which satisfies both GDC 26 and 27 is:

Primary: Load Following and Running In	2.8% ΔK
Subcriticality	.3% ΔK
Postulated Accident (covers stuck-rod requirement)	1.1% ΔK
Total	4.2% ΔK
Secondary: Long Term Effects	8.9% ΔK
Requirements for Subcriticality	.3% ΔK
Uncertainty (10% of long term effects)	.9% ΔK
Total	10.1% ΔK

With this split of reactivity requirements, the secondary system only is needed for long term (cold) shutdown, and both systems satisfy GDC 26 and 27. As stated in Section 5, the feasibility of designing the primary and secondary systems to meet these requirements is yet to be determined.

8.2.4 Applicability of Document No. 950396

Applicability of the monolithic HTGR-PH 1170 MWt Nuclear Safety Plant Specification, Document No. 950396, Issue One, to the MRS concept has been evaluated in Reference 8.2. The main conclusions of Reference 8.2 apply for the MPBR; namely, safety class assignment and design basis

accidents (DBA) of the monolithic HTGR do not necessarily apply for the modular systems but have to be developed on a case by case basis.

8.3 Conclusions and Recommendations

The main conclusions of this work is that the MPBR has enough inherent safety merits to warrant further development. Specifically, the extremely low fuel failure fraction in the recirculation cycle and the decay for one year of spent fuel in the OTTO cycle provides attractive alternatives.

The preliminary nature of this report should not be overlooked. Investigation in the following areas will be required to have a better understanding of safety and licensing issues and to enhance the plant safety:

- a. evaluation of the economics and safety of using burnable poison or shorter active core in the OTTO cycle
- b. reliability and safety evaluation of the fuel handling system
- c. safety evaluation of seismic events
- d. evaluation of fission product release from the pebbles under steam and oxidizing conditions
- e. evaluation of Xe instability
- f. evaluation of plant and reforming stability
- g. evaluation of potential locations for in-vessel instrumentation
- h. evaluation of the public and investment risk.

8.4 REFERENCES

- 8.1 Core Heat Up Studies for the 250 MWt VHTR Modular Reactor System, GA Document No. 906364, February 1982.
- 8.2 K. A. El-Sheikh, "Safety and Licensing Approach of the MRS-R Concept," GE, XL-848-20547/P2773, June 1982.

9.0 IDENTIFICATION OF TECHNICAL ISSUES

Several technical issues have been identified that must be addressed and resolved in order to assess the viability and economics of the MPB-R side-by-side system. These are discussed below.

9.1 Core Design

There are a number of technical issues in the area of core design which will require additional investigations. The most important of these issues include the following.

9.1.1 Additional studies are required to determine the trade-offs between the use of recirculated fuel and the OTTO fuel cycle. In particular, the information regarding the characteristics of a burnup measurement device must be obtained in order to determine whether the fuel ball recirculation scheme is feasible.

9.1.2 Studies are required to determine the sensitivity of the important core performance parameters to the modeling approximations for the radial reflector, axial reflector and core axial void region.

9.1.3 Studies are required to determine if the poison reactivity in the reflector control rods (including the rods in the reflector inserts which extend into the core), is sufficient to satisfy the shutdown reactivity requirements.

9.1.4 An investigation must be carried out to determine the fission product release rates for a loss of coolant transient and, thus, provide a base for assessing the impact of the transient on the continued plant operations.

The issues outlined above are briefly discussed in the remainder of this section.

9.2 Burnup Measurement

The fuel balls are recirculated in the pebble bed core to flatten the axial power profile. The principal disadvantages of this fuel management scheme is the requirement for a fuel handling system to discriminate and sort the exposed balls. Thus, fuel balls which exceed the specified maximum burnup are discharged to storage, and the remaining balls are recirculated back to the top of the core. The design of a reliable fuel sorting mechanism is essential in order to make the recirculation scheme feasible.

For a pebble bed core with a thermal power of 250 MW, a burnup of 80,000 MWD/MT, a heavy metal loading of 7 grams per ball, and a fuel management scheme based upon the circulation of fuel balls a total of 10 times, the flow rate of fuel balls that pass through the sorting device is approximately 3.1 balls per minute, or one ball per 20 seconds with continuous operation. At this rate, it is uncertain whether the fuel burnup measurement based upon the Cs-137 activity using gamma spectrometer is feasible. Thus, it may be necessary to reduce the number of fuel ball passages in the core or even use the OTTO fuel cycle scheme to alleviate the problems of sorting out the used fuel balls. That, in turn, would increase the axial power peaking and, subsequently, the transient fuel temperatures.

It appears from technical literature that another burnup measurement device that differs from the gamma spectrometer has been developed in Germany. It is speculated that the device is a small reactor which injects neutron source into the fuel balls. Fuel exposure is determined by detecting the change of neutron emission rate after neutrons pass through the fuel balls. It is believed that burnup measurement based upon the neutron emission technique is faster than gamma spectrometry. In any event, we need to know what the burnup detector is, how it works, and how fast can it work before we select the ball recirculation scheme as the reference design.

9.3 Fission Products Release Model

In the area of shutdown heat removal, the major concern and uncertainty has been in the modeling of fission products and actinides release as

a function of fuel temperature. The amount of fission products release during accidents can be estimated by integrating the temperature-dependent fission product release rate and the time-dependent fuel temperature distribution in the pebble bed core after the shutdown of the reactor. Unfortunately, a large discrepancy in the release rate has been found between the German model and General Atomic's model. A careful assessment of fission product release model is needed to properly estimate the amount of fission product release during accidents.

9.4 Neutronics Modeling Approximations

Earlier GE pebble bed studies and German studies have indicated that the axial power profiles in a pebble bed core were significantly influenced by the modeling of the top reflector region, including the representation of the upper void region above the ball fill. For normal pebble bed reactors in which the control rods are located in the upper reflector region, the calculational model incorporates the substitution of a diluted graphite reflector in place of the perforated upper reflector and void region, and the control in the top reflector and void region is factored into the model by including a psuedo absorber cross section in the neutronics calculations. This allows a treatment which has been adjusted by means of a comparison with two-dimensional transport calculations. For the modular pebble core in which the control rods are only present in the side reflector region, the calibration and adjustment of the psuedo absorber cross section by comparing with two or three-dimensional transport calculations are still not available. Additional studies are needed in order to determine the sensitivity of the axial power distribution to the model representation.

9.5 Control Availability

The question of control availability must be addressed to ensure that the proposed design of shutdown systems can meet the control requirements with a proper margin. Although the general consensus is that reactivity control can be accomplished with reflector rods by limiting the core diameter to less than 3 meters and by utilizing two enrichment zones and the

inserts that extend into the active core, it can not be confirmed without carrying out the control worth calculations.

An alternate core configuration, which uses eight graphite columns (inserts) in the fuel region, for 16 control rods, as shown in Figures 9-1 and 9-2, is being considered. This design of 2 control rods in each of the eight columns plus 16 control rods equally spaced, radially, in the reflectors, approaches the reactivity control used for the modular prismatic core which uses 19 core control rods and 12 reflector control rods.

In order to maintain the same core volume in the alternate design, the equivalent diameter of the core is increased to 3.3 m (130 in.) to accommodate the columns, without increasing the core height.

Control worth calculations of this design will also be accomplished in the next phase of the MPBR design development.

9.6 Vessel Supports

As noted in Section 4.5 above, the proposed method of supporting the side-by-side arrangement of reactor and component pressure vessels with the coaxial piping is not a proven concept. Although the concept appears feasible, much work is needed to prove its feasibility. Analytical analysis is needed to determine the size and thickness of the outer coaxial pipe, the size and shape of the reactor vessel fixed-rigid support and the component vessel fixed-sliding support. Likewise the capacity of the snubbers for both vessels and their locations must be determined.

The fabrication of the vessel's supports must be considered and the capability of installation of the support points to be aligned to the centerline of the coaxial pipes must also be considered. Questions such as the following must be answered:

- a. How much misalignment of the support points to the coaxial pipe centerline can be tolerated?

- b. Whatever the allowable misalignment, can it be held in fabrication and in installation?
- c. Does the degree of precision in the alignment required make the concept impractical or excessively expensive?

An independent analysis of the vessel supports has been completed by Bechtel Group Incorporated (BGI) which is included in this report as Appendix A. BGI concludes that the vessels can be supported on columns with spherical ends, which rotate as movement occurs due to temperature changes. This concept has been used before, for example, the graphite core support columns in Fort St. Vrain.

During the next phase of the MPBR design development, it will be necessary to size and locate snubbers, design the reactor and component vessel cavities and the building, so that seismic analysis of the vessel supports and vessel intervals can be accomplished.

9.7 Hot Restart

Upon shutdown involving loss of BOP systems, the passive decay heat removal method employed by the side-by-side system effectively thermally isolates the components from the reactor vessel. During the passive heat removal period, the core fuel ball temperatures remain at or slightly above normal operating while the components in the component vessel cool down. In order to restart without waiting a very long time for the core to cool down during the passive cooling mode, it is necessary to restart the system. Restarting with a hot core and cold components and balance of system could not only negate the thermal transient advantage of the side-by-side arrangement, but may not be feasible in a reasonable time frame.

Study of this situation is necessary involving a thermal-hydraulic computer code to model and analyze system conditions. In conjunction, methods to offset the hot restart conditions should be devised if the situation cannot be mitigated by operational measures.

9.8 Reactor Vessel Temperature

During the passive decay heat removal mode, natural circulation of helium in the pebble bed core volume takes place. In the circulation route, the upper axial reflector region will continually be exposed to the hottest helium. This is a rather drastic change in temperature since the down flow pebble bed core is ordinarily exposed to core inlet temperatures. It remains to be seen whether or not the upper reflector/upper head region can withstand prolonged periods of this temperature soak or if some kind of an auxiliary cooling system is required.

9.9 ISI of Reactor Internals and Support Structure

Need to establish criteria for the inspection of the reactor internals and support structure which is acceptable for licensing the reactor, which does not require the removal of the pebble-bed fuel. The inspection method and procedure must be able to verify the integrity of the core barrel and core support structure with access for inspection limited to the outer surfaces only of these components.

9.10 Reactor Removal as an Assembly

Features need to be included in the reactor design that permits the reactor assembly (minus fuel) to be removed and replaced from the reactor vessel. Equipment and procedures to perform this operation need to be designed and cost estimated. A means of inspecting reflector elements for deterioration and the degree of deterioration permissible before the reactor assembly must be replaced needs to be determined.

FIGURE 9-1

MODULAR PEBBLE-BED REACTOR (HORIZONTAL SECTION)

82-476-01

APPENDIX A

REVIEW OF
MODULAR HTGR VESSEL SUPPORT
CONCEPTS FOR THE SIDE-BY-SIDE CONFIGURATION

PREPARED BY
BECHTEL GROUP INC.
San Francisco, California
Job No. 14508

September 1982

APPENDIX A

Table Of Contents

	<u>Page</u>
1.0 Introduction and Summary	
1.1 General	1-1
1.2 Review Procedure	1-1
1.3 Summary	1-2
2.0 Erection Procedure	1-2
3.0 Loading and Design Criteria	1-5
4.0 Alternate Support Concepts	1-8
5.0 GHT Support Concept	1-8
6.0 Roller Support Scheme Development	1-9
7.0 Summary of Technical Issues	1-12

APPENDIX A

List Of Figures

	<u>Page</u>
Figure 1: Twelve Radial Shear Legs-Upper Level	1-9
Figure 2: Upper Level Radial Beams And U-Shaped Coaxial Pipe	1-10
Figure 3: General Arrangement of Conceptual Supports	1-11
Figure 4: Vessel Support Restraint Structure	1-14
Figure 5: Reactor Vessel Support Concept	1-15
Figure 6: Component Vessel Support Concept	1-16
Figure 7: Multiple Supports Of Smaller Size	1-18

APPENDIX A

1.0 Introduction

1.1 General

A 1981 report "High Temperature Modular System for the Generation of Process Heat" was issued by Gesellschaft fuer Hochtemperatur-reaktor-Technik mbH (GHT) and was the resource document of this design review. In addition the July 1982 report "Modular Pebble-Bed Reactor Reforming Plant Design for Process Heat" by General Electric Company was provided for information. The GHT report indicated that the reactor and application (component) vessels are each supported at 3 points at the level of the connecting coaxial pressure pipe. The purpose of this Bechtel review was to examine the implications of this conceptual support scheme, review its technical feasibility, and identify associated technical issues. The work also included suggesting alternate support schemes if appropriate.

1.2 Review Procedure

The likely erection procedure for these vessels was first considered in case this would significantly influence the location and type of vessel supports. Next loading criteria were developed on an approximate basis so that support concepts would be examined relative to realistic load magnitudes. The important load case after erection involves the horizontal and vertical seismic forces acting on these large, heavy vessels. Some alternate support concepts were idealized during the process of analyzing ways to support these vessels when they are subjected to large thermal changes and these are briefly reported. Finally the GHT concept was evaluated and a system modification was developed and is described in this report. This modification arose from the need to resist seismic forces as well as handle large thermal movements of these vessels.

APPENDIX A

1.3 Summary

Review of the vessel erection procedure established the difficulties inherent in handling these vessels as a coupled pair. The Bechtel review by heavy rigging specialists indicated a clear preference for lifting and setting these vessels separately and then completing the coaxial pipe connector after placement. This review of erection procedure also showed that the support concepts are relatively unaffected by the vessel erection method.

Seismic loading magnitudes were postulated and this indicated that the supports must be capable of providing resistance to uplift forces. The approximate magnitude of lateral forces also guided the conceptual development of the lateral restraint systems that must be considered.

The GHT concept of anchoring the reactor vessel and causing the component vessel to slide due to thermal changes was found to be conceptually acceptable. This review recommends that more than three supports be considered under each vessel and that sliding supports be replaced by mechanical systems providing rolling contact. This approach is developed and described in this report as a recommended modification worthy of further design development to better establish support system details and materials.

2.0 Erection Procedure

For the purpose of this review the pressure vessels were assumed to have internal structures removed or minimized so that weights, for lifting and handling, were approximately as follows:

Reactor vessel	1000 tons
Component vessel	700 tons

APPENDIX A

Although these weights are quite heavy they are well within the scope of contemporary heavy-lift rigging technology.

Conventionally, pressure vessels of similar configuration are raised from the horizontal to the vertical position and erected into place utilizing a tilt-fixture mounted on the bottom end of the vessel to carry a portion of the vessel weight. The top is raised upright by means of lifting a pivotal strong-back attached to the vessel's upper end (head) flange surface. Once the vessel is vertical, the lower tilt-fixture assembly is removed, preparing the vessel for installation.

In the case of pressure vessels without a flange surface to bolt the lifting attachment to, the vessel is supplied with two side trunnions to which slings or links are attached to be used for raising the vessel upright and transferring to the vessel's foundation supports.

These conventional methods would still be applicable here, and these methods of rigging the vessels should have no impact on the method or location of vessel supports.

2.1 Erection of Combined Vessels

Review by rigging specialists of the problems of lifting the vessels connected by the coaxial pipe resulted in the conclusion that this is not practical. The problem does not lie with the combined weight (which approaches 2000 tons) since present day crane and rigging technology could produce the equipment necessary to handle such loads. Rather the problem is that in order to be able

APPENDIX A

to handle the two vessels as a combined unit without inducing any torsional or shearing stresses to the coaxial piping while transporting, upending and erecting them, both vessels would have to be securely mounted to a shipping/handling frame. This framework, in addition to carrying the load of the vessels, must be rigid enough to absorb the bending, torsional, and other stresses induced by handling the vessels. This shipping frame must have attachments for the upending and erecting operation, such as a tilt-fixture for the lower end and a lifting truss for the top end. The different weights, location of center of gravity, diameter and length of vessels, all make the design requirement of the shipping/handling frame much more complicated. It is roughly estimated this hardware could add five hundred tons to the weight of the vessels.

Additionally, setting two heavy vessels from a common "hook" with correct orientation and elevation and proper percentage of bearing area on the vessel supports, could be a source of construction alignment and final set problems. Consequently, even though setting the vessels separately gives rise to needs for welding and stress-relieving the coaxial pipe in-situ, this new demand would seem to be a preferred trade-off to the difficulty of erecting the vessels as a combined pair.

2.2 General Erection Procedures

Understanding that the Reactor Building has not been detailed at this conceptual stage, and that an overhead crane is not yet specified for such a building, then the following list of alternative rigging approaches is recommended:

- a) The vessels are erected using a temporary elevated runway gantry crane, with its main runway girders supported by temporary columns inside and

APPENDIX A

outside of the Reactor Building, allowing the trolley hoist to pick and upend the vessels outside the Reactor Building and transfer them inside to their permanent supports.

- b) The vessels are erected under a stationary construction hoist (gallows frame) located over the center lines of the vessel's permanent position. A secondary system is required to bring the vessels into the building and under the hook in a horizontal configuration, ready for upending.
- c) The vessels are erected from outside the Reactor Building by the up-and-over approach. The rigging can be performed utilizing a very large twin-boom luffing derrick or by using a Transi-Lift, a heavy capacity derrick mounted on crawler transporters for added mobility.

Any of the above schemes are feasible for handling the projected loads, and they all have various advantages and disadvantages which must be assessed when vessels and buildings are in a more detailed state of conceptual development.

3.0 Loading and Design Criteria

3.1 System Dead Loads (D)

For the purpose of this study the operating weights of the vessels were given by G. E. as follows:

Reactor vessel = 1370 tons (2741 kips)

Component vessel = 826 tons (1652 kips)

3.2 System Temperature Conditions (T)

Reviews of the available reports suggested that the vessels would operate at a maximum temperature of 800°F. This was therefore adopted as the temperature range to consider for estimating changes in positions of supports and vessels. (This implicitly assumed the coldest cavity temperature to arbitrarily be 0°F for this review.)

APPENDIX A

3.3 Seismic Loads (E)

It was necessary to make an estimate of seismic loads on this system in order to make a realistic evaluation of support concepts. Since no seismic spectra at the reactor supports have been developed, some reasonable assumptions were used to derive seismic loads.

3.3.1 Equipment Design Spectra

The pressure vessels are seismic Category I equipment and must be qualified for seismic forces. The specified ground or site excitation is transmitted and amplified up through the various floor levels of the building. The pressure vessels, and other equipment, respond to these excitations applied at their supports. The general design procedure is to develop acceleration response spectra at various locations in a given building due to the specified input ground motion. Idealized equipment design spectra have been developed from various Bechtel nuclear power plant jobs. This has also been studied extensively in the LMFBR Program for typical nuclear island structures. These resources were used along with the following assumptions:

- o Use response spectrum curves corresponding to the safe shutdown earthquake (SSE) conditions
- o Support the vessels at conventional elevations in a reinforced concrete building
- o Use site ground accelerations of 0.3g's
- o Use equipment damping 3% of critical
- o Assume vessels and supports provide combined frequencies in the peak spectrum response range

By this method it was found reasonable to use the following levels of acceleration for the vessel seismic response calculations:

Horizontal 3g's

Vertical $\pm$ 2g's

APPENDIX A

3.4 Load Combinations

Various other loadings occur in prototype design situations which are not specifiable here. To approximately account for vertical effects due to such unspecified loads, the dead loads were increased by 20%. For example, the maximum down load for the reactor vessel, in combination with seismic, is estimated by:

$$1.2(2741) + 2(2741) = 8770 \text{ kips}$$

The loads are accounted and combined in this manner to give the following table.

	Support Loads (kips)	
	Reactor Vessel	Equipment Vessel
Horizontal	8223	4956
Vertical down	8770	5286
Vertical upward	2193	1322

3.5 Design Criteria

Nominal steel yield of 36 ksi was assumed to begin this work but was discounted to 30 ksi to recognize the high-temperature service conditions. This was sufficient for this preliminary work. Steel shear yield stress was assumed as 15 ksi. Concrete criteria were not involved since the work remained concerned with steel support components.

APPENDIX A

4.0 Alternate Support Concepts

In the process of concept evaluation the overall system behavior was assessed independently of the GHT scheme. The support schemes shown in Figures 1 and 2 were postulated where the vessels are supported at or near the top. These schemes use many radial supports that are constrained to slide radially and thereby keep each vessel on center during temperature changes. This location for supports provides a vertical distance to the coaxial pipe which expands to push the lower ends of the vessels apart. To reduce this action, the coaxial pipe was postulated as a large U in plan. In addition to increasing reactor vessel stresses these schemes fail to satisfy GHT concerns that the coaxial pipe be as stress-free as possible since differential thermal motions can occur at that level as the vessels expand. Furthermore a long, curved coaxial pipe is impractical for its specialized function.

Accordingly the GHT approach of keeping supports at the level of the coaxial pipe was confirmed. This removes differential thermal displacements of the vessels as a threat to stressing the coaxial pipe.

5.0 GHT Support Concept

As was stated earlier, the arrangement of the supports in the GHT system, at the level of the coaxial pipe, protects the pipe from differential thermal movements of the vessels. The coaxial pipe acts as a thermal ram to move the equipment vessel over sliding supports. As long as those supports exhibit low friction then the stresses in the coaxial pipe will remain small. This also requires flexibility in all pipes connecting into the equipment vessel so that thermal translation can freely take place. Figure 3 shows the GHT system of supports but modified in the following manner.

APPENDIX A

Figure 1: Twelve Radial Shear Legs-Upper Level

APPENDIX A

Figure 2: Upper Level Radial Beams And U-Shaped Coaxial Pipe

Figure 3: General Arrangement Of Conceptual Supports

APPENDIX A

There was concern from this review that the sliding, low-friction-type supports may deteriorate with time in the high-temperature, high-radiation environment supporting the vessels. Accordingly structural details were developed for an alternative mechanical rolling-type system that would provide very low-friction for thermal movements and would be quite stable in the given environment.

In particular the two vessels were postulated to be carried as in the GHT arrangement (3 supports per vessel) but using long struts with spherical bearing caps. The spherical diameter is chosen equal to the length of the strut. This allows the supported point to move in any direction under varying loads and temperatures. The actual spherical diameter, hence length of struts, would be determined by stress analyses and careful review of candidate steels for the service conditions. This study assumes struts are 5 feet long and thus provide a maximum roll of 3° under a 3 inch translation of the vessel due to temperature changes.

6.0 Roller Support Scheme Development

The system presented here was conceptualized for the estimated reactor vessel loads and was generally particularized for the lighter component vessel.

Compression (down) loads are resisted by steel box-section struts, with spherical end caps, each strut being 15 inches square on the outside and using 2" plate material. Struts to carry the lighter equipment vessel would figure at about 10 inches square. However, these sections will increase in size when more detailed design work accounts for the rolling eccentricities of the vessel loads.

The component vessel support located directly opposite to the reactor vessel anchored support, along the coaxial pipe axis, can utilize cylindrical caps on the struts. The cylinder diameter will be the strut length, similar to the spherical

APPENDIX A

cases. This is because motion is constrained along the axis of the coaxial pipe. This study does not show details of the restraints needed to prevent motion tangential to the equipment vessel.

Each vessel requires 3 stub cantilever members to react in the strut supports. These members are sized in this review at about 2 feet wide by 4 feet deep box sections using 2 inch plate webs and flanges.

The problem of restraining seismic uplift forces is solved in this scheme by utilizing similar struts above the vessel cantilever beams and providing an overall steel restraint structure anchored by tie rods into the concrete. This overall scheme is shown in Figure 4 and, for the case of 3 supports per vessel, provides a restraint structure about 18 feet high. It is an unresolved technical issue in this review as to the way in which these restraint structures should influence the cavity design and shape. It is recommended that the upper struts be precompressed to a value at least as great as the design uplift tension. This may be done by incorporating hydraulic packing pads in the tops of the restraint structures and then designing a wedge system to carry the precompression when the jacks are removed.

Figure 5 and 6 are provided to show how the conceptual support structures relate to the size of the individual vessels.

6.1 Seismic Snubbers

The support concept presented here also assumed that horizontal seismic forces would be taken by a ring of snubbers arranged in a circle around each vessel, near the center of gravity of the vessel. In practical design a certain amount of lateral force can be expected at the supports and snubbers may be placed there accordingly.

Figure 4: Vessel Support Restraint Structure

Figure 5: Reactor Vessel Support Concept

Figure 6: Component Vessel Support Concept

APPENDIX A

The size and arrangement of snubbers will have an influence on the size and shape of the cavity. For example, considering the reactor vessel forces, if groups of snubbers were positioned at 15° intervals around the circle, then each group would need to provide about 540 kips of reaction. This was calculated for both vessels for 15° and 10° spacings, assuming full lateral load absorption by the snubbers. Force capacity required at each location is tabulated below.

	Required Capacity (kips)	
	10° Spacing	15° Spacing
Reactor Vessel	360 k	540 k
Equipment Vessel	217 k	326 k

This an unresolved technical issue as to how sufficient snubber restraint can be provided. This will involve a study of snubber sizes and capacities, attachment requirements, and requirements for cavity size and shape.

6.2 Other Support Arrangement

The review work described here has been based on the GHT arrangement of 3 supports per vessel. This gives rise to the maximum load condition per support. Increasing the number of supports, as in Figure 7, will possibly reduce the loading on each support. However, it may be necessary to increase the number of supports if the lateral seismic forces cannot be resisted in a way that prevents additional vertical forces appearing at the supports. In addition,

Figure 7: Multiple Supports Of Smaller Size

APPENDIX A

the big loads transferred by the vessel cantilever members may have an important influence on the vessel designs. This may be improved by using more supports to get smaller support loadings. These factors are noted in the following, concluding section.

7.0 Summary of Technical Issues

Technical issues identified in this review of the GHT support scheme are listed here for consideration in future studies. They are presented without discussion.

- o Selection of high-strength steel for the spherical bearing caps to meet the service environment
- o Determination of length of struts (or spherical diameter) depends on steel selection and stress analyses
- o Definition of conceptual equipment response spectra for design seismic forces for vessel intervals and vessels supports
- o Development of snubber sizes, geometry and 'fit' into cavity, and best arrangements around vessel
- o Cavity shape and size relative to inserting vessels with support cantilevers; support system requirements; snubber system requirements
- o Seismic response of base-supported, coupled pressure vessels due to specified seismic inputs
- o Cavity liner plate design and cooling requirements related to cavity shape and sizing including support arrangements.