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GAS REACTOR INTERNATIONAL COOPERATIVE PROGRAM INTERIM REPORT

Safety and Licensing Evaluation of German Pebble Bed Reactor Concepts

September 1978

Work Performed Under Contract No. EN-77-C-02-4057

NUS Corporation
Rockville, Maryland

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SAFETY AND LICENSING EVALUATION OF GERMAN PEBBLE BED REACTOR CONCEPTS

Prepared by
NUS CORPORATION
Rockville, Maryland

For

GENERAL ELECTRIC

ADVANCED REACTOR SYSTEMS DEPARTMENT
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Prepared for
The U.S. Department of Energy
Contract No. EN-77-C-02-4057

September 1978

ABSTRACT

The Pebble Bed Gas Cooled Reactor, as developed in the Federal Republic of Germany, was reviewed from a United States Safety and Licensing perspective. The primary concepts considered were the steam cycle electric generating pebble bed (HTR-K) and the process heat pebble bed (PNP), although generic consideration of the direct cycle gas turbine pebble bed (HHT) was included. The study examines potential U.S. licensing issues and offers some suggestions as to required development areas.

By agreement this report has been reviewed by the cognizant Federal Republic of Germany industrial and laboratory operations and the resulting comments incorporated or noted herein.

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GENERAL ELECTRIC COMPANY PROLOGUE

The following report was prepared under contract to the Energy Systems Programs Department, General Electric Company, by the NUS Corporation of Rockville, Maryland. The purpose of the study was to have a technical company experienced in both U.S. and foreign licensing processes perform a critical review of German Pebble Bed Reactors (PBR) from a U.S. licensing perspective. It was, therefore, necessary that the selected company be closely involved with U.S. gas cooled reactor licensing experience. NUS met these criteria through previous technical and licensing consultation on the General Atomic HTGR for such organizations as the Electric Power Research Institute (EPRI), Southern California Edison, the German Ministry for Research and Technology (BMFT), and the Japanese Atomic Energy Research Institute (JAERI).

NUS was tasked with extrapolating the existing U.S. licensing experience to the various German PBR concepts, including the process heat plant (PNP) and the steam cycle electric plant (HTR-K). During the course of the study, the German PBR program was altered to incorporate the direct cycle gas turbine plant (HHT-K) in lieu of the HTR-K. As a result, NUS included comments on generic direct cycle gas turbine problems as applied to the HHT-K concept. It was considered important to include at least general comments on the formidable licensing tasks ahead for the HHT-K concept.

Due to administrative problems in gaining access to German data on the reference PNP and HTR-K designs, this report is not based upon a data bank completely consistent with that used in the report, "German Pebble Bed Reactor Design and Technology Review."[22]

Therefore, the reader is cautioned to consult the reference for specific technical details on the German designs. In those areas where NUS did not possess sufficient technical details prior to preparation of their report, they attempted to specify design criteria and generic problems that might be expected (based upon U.S. licensing activities for the HTGR).

As the result of the above considerations, the reader should understand that this report is intended to raise likely safety and licensing issues and identify areas requiring future development activities. NUS is approaching the study from the critical questioning perspective that the NRC would use. In some areas identified as concerns, more information on completed German R&D activities may indicate that the problems have been already solved. In other areas of concern, German R&D programs may already be planned to resolve the issues. Future Safety and Licensing activities at General Electric within the Gas Reactor International Cooperative Program will resolve these inconsistencies.

Considering the data base for the study, General Electric Company is in general agreement with the NUS conclusions of Section 12.0. Even for those areas where recent data indicate changes from that used by NUS, the general issues raised should be valid. It is felt that safety and licensing concerns do not present insurmountable problems for the development of advanced PBR concepts. Clearly there are topics, such as process heat generation, that do not have U.S. licensing precedent. However, by appropriate engineering efforts, these problems should be solvable, and are not unexpected given the early conceptual state of the designs.

NUS suggested that a U.S. sponsored PBR prototype plant, subject to NRC technical review, be considered. This idea appears to have real merit, given that the U.S. decides development of advanced PBR's for process heat is a desirable energy strategy. Licensing activities for such a prototype would form an excellent foundation for large commercial plant licensing.

By agreement the draft version of this report was reviewed by cognizant Federal Republic of Germany industrial and laboratory operations. The resulting comments included both specific corrections and general observations. The former were incorporated into the body of the text and the latter are discussed below.

The German reviewers concurred with the General Electric prologue statement to the effect that access to data during the course of the NUS study was limited. They expressed the common desire for the study to be updated to more adequately reflect the current level of German HTR knowledge. The General Electric Company agrees that future efforts within the International Cooperative Program must include more detailed licensability evaluations incorporating the expanding experience base of the FRG program. This report should, therefore, not be viewed as a final safety analysis, but rather as a preliminary review of the various FRG concepts that forms a basis for future discussion and investigation.

The German comments include some that could not be incorporated specifically into the report and must be evaluated further during future phases of the International Cooperative Program. The areas of concern include:

- *More complete delineation of the liscensing differences between the FRG and the U.S.A. with respect to external influences (earthquakes, aircraft impact, etc.)*
- *More complete discussions of the various design accidents of Chapter 10, including correlation to HTGR Safety Analysis Reports of General Atomic Co.*
- *A clearer correlation of process heat plant hazard analysis between conventional plants and PNP plants.*

1.0 INTRODUCTION

The purpose of this study is to review the Federal Republic of Germany (FRG) development programs in the area of Pebble Bed Reactors (PBR) and to determine which aspects of resultant commercial plant designs would be a concern within the U.S. licensing process. This study compliments and considers the results of other parallel studies being performed for the General Electric Company by NUS; namely, review of the FRG licensing process and technical safety criteria, review of High Temperature Reactor operating experience (outside of the U.S.), and review of international licensing regulations. The results of these other studies are presented in separate reports.

In compliance with the "umbrella agreement" established by the U.S. and FRG, this study is meant to be a preliminary evaluation of the potential for incorporating German PBR technology within the U.S. licensing framework. As such, it must necessarily consider information developed on a conceptual design basis and presented in a preliminary format. In addition, since the US/FRG confidentiality agreement related to this program was not effected until the advanced stages of this study, much of the information which is now available could not be incorporated in an orderly manner. Also, the information which has been documented is subject to significant future modification since major program decisions [e.g., relying on all steam electric generation (HTR-K) or direct cycle gas turbine (HHT-K)], and major system design decisions (e.g., fraction of electric generation for PNP concept, heat exchanger design, control rod design, et cetera) are now under review.

Because of the above considerations, systems and design concepts considered in the review have been documented within this report to a greater degree than might otherwise be appropriate. Also, considerations as to the licensability of these systems has been limited to more general technical concerns, in recognition that detailed investigations will be performed for specific areas during later stages of the DOE PBR program.

The first activity under this project involved a review of available PBR conceptual design information, principally involving the General Electric

SNPH and VHTR programs[18],[19] plus the KFA conceptual design report issued during 1974.[11] This information formed the basis for our interim report, and for a majority of our subsequent effort. Unfortunately, these reports did not provide detailed conceptual design information or design bases. After execution of the confidentiality agreement, a large amount of additional information has become available and has been incorporated in this study on a best-effort basis. This additional information suffers from the drawback that it was only available in German and, being highly technical in nature, was not easily translatable. Further design information may be expected through the US/FRG umbrella agreement and should be factored into future studies.

In preparing the outline for this report, the NRC Standard Format for HTGRs[21] was considered and adapted to the purpose of our review. Therefore, our outline bears a close similarity to that for General Atomic Safety Analysis Reports. This approach was considered to be most advisable since it forms the basis for future investigation, although conceptual design information is very limited, or not available, for some sections. In situations where insufficient information is available, an attempt has been made to discuss the principal licensing criteria which should be addressed by a given PBR design. This lack of information is particularly prevalent for the balance of plant design and for engineered safeguards features other than the core auxiliary cooling system.

In addition to a review of safety criteria, some preliminary calculations have been performed to verify that fundamental design parameters are within licensable limits. These calculations are presented for guidance only and are not meant to imply any design review function regarding acceptability of the PBR designs. Although they have been carefully checked, these calculations have not been subject to independent review or similar quality assurance procedures.

A large amount of information has been summarized in the tables presented in this report. These tables contain many useful reference comparisons between HTGR and PBR design concepts and between the PNP, HTR-K, and HHT-K conceptual designs. Pertinent references to these tables can be found throughout the report.

2.0 REACTOR SYSTEM

2.1 SUMMARY DESCRIPTION

This chapter addresses safety and licensing concerns for the PNP, HTR-K, and HHT-K nuclear fuel design, reactor core, reactivity control system, and other core related systems. It is primarily based on progress reports for current research and development programs under way in the Federal Republic of Germany (FRG) in supporting fuel and reactor technology[(1)] and proposed development of the high temperature reactor fuel cycle.[(2)]

For purposes of perspective, a summary tabulation of the principle mechanical, nuclear, thermal-fluid, and reactivity characteristics of the PNP, HTR-K, HHT-K systems is presented in Table 2-1 and compared with U.S. licensed HTGR systems. The PR-3000* core is a 3000 MWth, 2-zone pebble bed core with an average core height of 5.53 m and a core radius of 5.6 m. The thickness of the outer core zone is 1.0 m. The core contains approximately 3 million spherical fuel elements of 6 cm diameter irradiated to an average burnup of 100,000 MWd/MT. The average core power density for the reference uranium/thorium fuel cycle is 5.5 MW/m^3. The reactor core design is similar in many respects to the Thorium High Temperature Reactor (THTR) but with specific exceptions; the thermal rating of the unified PR-3000 reactor is about 4.0 times the rating of THTR. There is considerable variation in environmental conditions for the PR-3000 design applications as presented in Table 2-1. The limiting reactor outlet temperature of 950°C for the Process Nuclear Project (PNP) is that for which operational experience has been acquired in the AVR at Jülich. The experience with AVR spherical fuel elements after more than three years of operation at 950°C is reported as favorable,[(3)] although there are additional concerns as given in 2.2.1.1.

Alternative fuel cycles to the highly enriched U/Th fuel cycle are being analyzed and compared with the reference highly enriched uranium cycle as discussed in Section 2.2.9 of this report. As part of the International Fuel Cycle Evaluation (INFCE), a decision will be made in late 1978 on the fuel cycle selection, fuel element manufacture, testing, and possible recycling technologies to be adopted. In addition, immediate analyses of

*The PR-3000 notation is a generic title indicating a 3000 MWth pebble bed core.

TABLE 2-1

SUMMARY COMPARISON OF PNP, HTR-K, AND HHT-K NUCLEAR, THERMAL-FLUID, AND MECHANICAL CHARACTERISTICS WITH U.S. LICENSED HTGR SYSTEMS

		Peach Bottom	Fort St. Vrain	GASSAR-6	HTR-K Steam Cycle	HHT-K Direct Cycle	PNP/HKV Process Heat
1. General							
Capacity							
Net electrical output	MW	40	330	1159	1120	1240	118
Gross generation	MW	44.5	342	1176	--	--	--
Overall station net efficiency	%	34.6	39.2	38.6	37.3	41.3	Not applicable
Net heat rate	BTU/kw-hr	9810	8800	8851	Not available	Not available	Not available
Containment type		Steel	Atmospheric confinement	Reinforced Concrete/Steel	Reinforced concrete/steel and liner		
Number of main cooling loops		2	2	6	6	1 (2 gas ducts)	6
Number of emergency cooling loops		2	2	3	4 x 50%	3 x 100%	4
2. Reactor Core							
Reactor core output	MW(t)	115	842	3000	3000	3000	3000
Core dimensions	dia/ht, ft	9.16/7.5	19.6/15.6	27.7/20.8	36.7/ 18.2	36.7/18.2	36.7/18.14
NSS helium inventory	lbs	1000	8890	20,745	41,200	77,000	19,200
No. of fuel elements/columns		804/NA	1482/247	3944/493	3,000,000	3,000,000	3,000,000
Primary coolant flow	10^6 lbs/hr	0.492	3.39	10.84	10.48	11.97	7.06
Primary coolant inlet pressure	psig	305	688	725	870	1045	580
Avg. coolant temp, reactor inlet	^oF	650	762	606	511	855	572
Avg. coolant temp, reactor outlet	^oF	1380	1445	1392	1292	1562	1742
Core pressure drop	psi	3.2	8.4	11.3	10	10	8.5
Core orifices		NA	37 Variable	73 variable/ 18 fixed	Not applicable	Not applicable	Not applicable
Total initial neutron flux	nv	1.7×10^{14}	1.8×10^{14}	2.4×10^{14}	Not available	Not available	Not available
Avg. power density	KW(t)/liter	8.3	6.3	8.4	5.5	5.5	5.5

TABLE 2-1 (Continued)

		Peach Bottom	Fort St. Vrain	GASSAR-6	HTR-K Steam Cycle	HHT-K Direct Cycle	PNP/HKV Process Heat
Average heat flux	BTU/hr-ft^2	69,600	45,000	66,000	32,000	32,000	32,000
Maximum heat flux	BTU/hr-ft^2	110,650	140,000	172,000	95,000	95,000	95,000
Avg conversion ratio (equilibrium)		0.44	0.60	0.65	0.605	0.61	0.62
Maximum linear heat rating	KW/ft		3.8	6.3	Not applicable	Not applicable	Not applicable
3. Fuel and Thermal Data							
Fuel material		-Th/U-235, 93% enriched/U-233, recycle-			----Th/U-235, 93% enriched/U-233 recycle----		
Fuel form		Coated particles in graphite compacts/sleeves	Coated particles in cylindrical pitch-bonded fuel rods structurally maintained by hexagonal graphite blocks		BISO coated particles in spherical pitch-bonded matrix structurally maintained in molded graphite sphere		
Burnable poison material		Rh-103	B_4C in C	B_4C in C	Not applicable	Not applicable	Not applicable
Number of refueling regions, full/partial		Batch refueled	37	61/24	Not applicable	Not applicable	Not applicable
Elements, hexagonal across flats		N/A	14.7	14.17	Not applicable	Not applicable	Not applicable
Element length	in.	144	31.22	31.22	Not applicable**	Not applicable**	Not applicable**
Fuel rod diamter	in.	2.74 compact	0.491	0.624	Not applicable**	Not applicable**	Not applicable**
Permanent reflector thickness, top/bottom/side	in.	27/27/24	39.6/46.8/47.0	46.8/46.8/40.5	Not available	Not available	Not available
Replaceable reflector lifetime	yrs.	N/A	8	8	>25	>25	>25
Total quantity of U-235/TH, initial core	kg.	220/1450	882/19,458	1747/37,487	--------------.766/18.745--------------		
Total weight reactor graphite	10^6 kg.	0.17	1.44	2.19	Not available	Not available	Not available
Average fuel burnup	MWd/tonne	60,000	100,000	98,000	100,000	100,000	100,000

* OTTO fuel cycle
** Fuel spheres of 6 cm dia.
Δ U-235/U-233/TH at equilibrium

TABLE 2-1 (Continued)

		Peach Bottom	Fort St. Vrain	GASSAR-6	HTR-K Steam Cycle	HHT-K Direct Cycle	PNP/HKV Process Heat
Max. fuel centerline temp., short term	^{o}F	2430	2300	2559	1750	1850	2000
Average fuel temp.	^{o}F	1700	1500	1634	1238	Not available	7594
Average moderator temp.	^{o}F	1370	1370	1362	1202	Not available	1594
Isothermal temp. coef. at 1200^{o}K initial core, BOC	$p/^{o}C$	core 1 $-4.1x10^{-5}$ core 2 $-5.4x10^{-5}$	$-3.x10^{-5}$	$-4.5x10^{-5}$	$-3x10^{-5}$	$-3x10^{-5}$	$-3x10^{-5}$
Fuel reloading schedule, fraction core replaced each year		prototype	1/6	1/4	0.33	0.33	0.33
Fuel life, full power years		2.2	4.8	3.2	3.2	3.2	3.2
4. Reactor Vessel							
Type		Steel pressure vessel ASTM A212 Gr. B	--Prestressed concrete reactor vessel--		PCRV	PCRV	PCRV
Internal clearance, dia/ht	ft	14/35	31/75	37/47.3	53.5/50.6	53.5/50.6	Not available
Max. external dimensions dia/ht	ft	145/355	49/106	100.5/91.2	120.7/101.7	147.6/135.8	144.4/101.7
Min. PCRV wall thickness	ft	N/A	90	17.5	Not available	Not available	Not available
Normal working pressure	psig	355	688	710	870	1045 HP/	580
Liner thickness, core cavity/ penetrations	in	N/A	0.75/0.75	0.75/0.5	0.75/0.5	0.75/0.5	0.75/0.5
Liner temp., normal avg/hot spot	^{o}F	N/A	130/200	150/250	150/250	230/302	150/250
Thermal Barrier	^{o}F	N/A	Ceramic fiber blankets/ blocks covered by carbon steel or nickel alloy plates.		Ceramic fiber blanket covered by nickle alloy plates.	Not applicable (warm liner)	Ceramic fiber blanket covered by nickle alloy plates.
5. Helium Circulators (each)							
Number of circulators		2	4	6	6	Not applicable	6
Type		Horizontal single-stage centrifugal	Single-stage axial-flow compressor with driver		Vertical single stage centrifugal	Not applicable	Vertical, single stage centrifugal
Bearings		Oil lubricated	--Water lubricated--		Oil lubricated	Not applicable	Oil lubricated
Drive		Motor driven	Single-stage steam turbine		Motor driven	Not applicable	Motor driven
Speed	rpm	3200	9550	6750	3000	3000	3000
Steam flow, including bypass	10^{6} lbs/hr	N/A	1.395	1.332	Not applicable	Not applicable	Not applicable
Circulating capacity	10^{6} lbs/hr	0.246	0.8725	1.87	Not applicable	Not applicable	Not applicable
Compressor inlet temp.	^{o}F	626	742	721	510	95	Not available
Power	hp	2500	5200	16,270	9517	Not available	10,724

TABLE 2-1 (Continued)

		Peach Bottom	Fort St. Vrain	GASSAR-6	HTR-K Steam Cycle	HHT-K Direct Cycle	PNP/HKV Process Heat
6. Steam Generators (per module)							
No. of steam gener. modules		2	12	6	6	2 recuperators; 2 precoolers; 2 recoolers	6
Type		Forced recirculation	Once-through, helical coil with integral reheat; carbon steel, chrome-moly, Incoloy 625 and 800.		Once through, helical coil and SH straight tube with reheat	Straight tube, counter flow	Once through, helical coil and SH straight tube with reheat
Dimensions, ht/dia	ft/ft	30/8	25-7/5-6	69-9/12-8	84.3/13.1	108.3/22.0; 91.9/19.0	55.9/11.5
Heat transfer, main steam/reheat	10^6 BTU/hr	199/NA	209/34.7	1456/278	1720	Not applicable	1174
Bulk gas inlet temp.	^{o}F	1298	1427	1366	1292	1562	1375
Coolant mass flow	10^6 lbs/hr	0.246	0.284	1.843	1760	1.813	1.18
Superheater steam flow	10^6 lbs/hr	0.185	0.192	1.34	1.58	Not applicable	1.13
Superheater outlet, press./temp.	psig/^{o}F	1480/1005	2500/1000	2513/955	2538/959		1704/811
Reheater steam flow	10^6 lbs/hr	N/A	0.187	1.33		Not applicable	Not applicable
Reheater inlet press./temp.	psig/^{o}F	N/A	683/673	644/637		Not applicable	Not applicable
Reheater outlet press./temp.	psig/^{o}F	N/A	600/1002	586/1002		Not applicable	Not applicable
Feedwater, press./temp.	psig/^{o}F	1580/428	3100/403	2967/370	2828/365	Not applicable	2175/356
7. Reactivity Control							
Type		--Control rods and emergency shutdown canisters--			--------------Control rods and KLAK--------------		
Control rods		36/19 emergency	37 pair	73 pair	246	246	156
Emergency shutdown canisters		55 thermally released	37	73	Not applicable	Not applicable	KLAK
Drive (normal)		hydraulic	electric motor with cable and drum		198 Hydraulic or pneumatic or lift rod; 46 cable driven	Hydraulic or pneumatic or lift rod	Lift or rotary rod
Scram method rods		hydraulic/ electric	gravity		gravity/ pneumatic	gravity/ pneumatic	hydraulic/ pneumatic/

TABLE 2-1 (Continued)

		Peach Bottom	Fort St. Vrain	GASSAR-6	HTR-K Steam Cycle	HHT-K Direct Cycle	PNP/HKV Process Heat
Canisters			--------gravity--------		N.A.	N.A.	gravity
Equivalent control rod worth, all rods, hot	% k	24	23.1	25.8	33.4 (cold condition)		Not available
Scram insertion time	sec.	0.8	152	25	~ 100	~ 100	Not available
Minimum rod withdrawal time	min.	N/A	3	3.5	~ 100	~ 100	Not available
8. Core Auxiliary Cooling System							
Number of CACS circulators		Used with reactor shut down and depressurized to remove only the decay heat in core. Two redundant sets of cooling coils, at 3.5×10^6 BTU/hr each, around the steel reactor vessel. Reactor also utilizes natural circulation.	Use existing main circs.	3	4 x 50%	3 x 100%	4 x 50%
Compressor type			Water Turbine	Single-stage axial flow	Single stage, centrifugal	Single stage, centrifugal	Single stage, centrifugal
Drive type			Pelton wheel	Electric motor	Electric motor	Electric motor	Electric motor
Helium flow rate (each)	lbs/hr						
2) Depressurized PCRV with air ingress			34,200, 0 psig, 500°F	132,000	158.740	Not available	Not available
1) Pressurized PCRV			374,400, 545 psig, 500°F	147,000	237.316	Not available	Not available
Core auxiliary heat exchangers			Use steam generator	One helical tube bundle per CACS	U-tube bundle	Not available	Not available
Feedwater flow rate (each)	10^6 lbs/hr		N/A	653,000		Not available	Not available
1)					1.549		
2)					1.557		

the storage requirement for spent fuel elements during the market introduction phase have been initiated. Preparation of cost, safety, and risk analyses for fuel element long-term storage in retrievable surface and deep repositories have also commenced, as discussed in Section 2.2.8.

Vertical columns of graphite reactor blocks surround the active pebble bed core. The reflector blocks of the bottom and top reflector are of hexagonal geometry.* The outer blocks are larger. The coupled reflector is designed for permanency in the reactor cavity. These outer graphite blocks are surrounded in turn by a metal-clad, boronated shield, cooled by a bypass helium flow. The entire assembly of active core and reflector elements is mounted on graphite support blocks that form a floor which, in turn, is supported by graphite posts. The pebble bed core is continuously fueled by the additon of fuel spheres from the top of the PCRV head through 43 fuel entrance chutes. Fuel is continuously removed from the bottom of the pebble bed through 6 symmetrically arranged fuel exit chutes penetrating the PCRV bottom head and ducted to spent fuel storage cannisters outside the containment in a separate building.

In the HTR-K and HHT concept, the reactor is controlled by two redundant diverse shutdown systems consisting of 198 in-core rods and 48 reflector rods. One shutdown system consists of a portion of the absorber rods freely inserted into the pebble bed (in-core rods) and a portion of the absorber rods inserted in cylindrical holes in the side reflector (reflector rods). This shutdown system is sufficient to shutdown the reactor to hot, operating conditions. The second shutdown system is provided by the rods which are not reserved for the first shutdown system (in-core and reflector rods) and provides, with the first system, for the long-term shutdown of the plant. The second shutdown system is also utilized for power level control of the pebble bed core.

Other shutdown measures conceptually formulated to increase the availability of long-term shutdown include the addition of small absorber spheres (KLAK), a gadolinia absorber gas. The control rod drive assemblies are of three types and are housed in penetrations in the head of the PCRV above

*The side reflector consists of two cylindrical graphite walls.

the reactor cavity. The control rod drive mechanism designs are diverse. The rod drives for the initial shutdown system are actuated by pneumatic long-stroke pistons. The in-core rod drives for the second shutdown system are actuated by hydraulic long-stroke pistons. The reflector rod drives have an electrically driven mechanism which attaches to and supports the control rods by cables, but which allows the rods to enter holes in the reflector by gravity.

Flow of helium coolant is downward through the core. Flow passes from the upper core cavity region through the interstices and cusps formed by the spherical fuel elements and then through the core support blocks. Within the support blocks, flows are ducted into exit streams and directed to the hot outlet ducts from the lower plenum. The temperature of the exit stream is monitored by thermocouples in the support blocks in a number of regions. The mechanical and thermal-fluid arrangement varies depending upon the specific reactor application.

The core is supported from the bottom by graphite supporting columns on a bottom structure of graphite or metallic bases, specific designs for which differ depending upon the reactor application. The design of core lateral support provides for an arrangement which reduces the temperature in the location of the restraint structures to acceptable levels for use of metallic components.

Most of the structures to be described in this section are graphite or, in the case of the fuel, coated particles of fissile or fertile material contained in a graphite matrix. The mechanical, nuclear, and thermal-fluid design bases are presented in the following Sections, 2.2, 2.3, and 2.4.

2.2 MECHANICAL DESIGN

2.2.1 Fuel

2.2.1.1 Description

Each 6 cm fueled sphere is isostatically molded and consists of a 5 cm diameter core containing coated-particle fuel in a graphite matrix, with an outer fuel-free zone of 5 mm thickness. Identical graphite material is used for both the matrix and the fuel-free zone. The development and manufacture of these fuel spheres, which have been operated in the AVR reactor for more than 8 years without difficulty, has been described.[4] Table 2-2 lists the mechanical and thermal properties of the PR-3000 spherical fuel elements. The coated particles are subject to continuing development. At present, the reference design for PR-3000 application is given as

- $(U/Th)O_2$ kernel, 400 μm diameter, HTI-BISO coating (improved THTR particle)

Other potential fuel designs are also under consideration, including

- $(U/TH)O_2$ kernel, 500 μm diameter, LTI-TRISO coating
- UC_2 kernel, 200 μm, LTI-TRISO coating
- ThO_2 kernel, 500 μm, LTI-TRISO coating

Experimental evidence from AVR has demonstrated that reference fuel designs can perform adequately under PR-3000 conditions. However, ongoing US and FRG programs have indicated some concerns with potential fuel designs which are generic to both PBRs and HTGRs. There is evidence that the structurally preferred Low Temperature Isotropic (LTI) PyC coating typically used in BISO particles are susceptible to neutron-induced permeability to noble gaseous fission products such as Kr, Xe, ^3H, and to the solid fission products Cs and Ag.[5] This phenomenon is not relevant for PBR particle designs, however, because LTI coating is used only in confirmation with impermeable SiC layers. The effect has not been observed in AVR fuel since the BISO- or DUPLEX-coated particles in the Union Carbide Corporation, NUKEM, and HOBEG fuels have consisted of methane-derived, High Temperature, Isotropic PyC coatings. The LTI coatings are preferred for the PNP and HTR-K applications from the standpoint of PyC stress related considerations at high temperatures, provided the neutron-induced permeability problem can be overcome.

The fuel particle coatings serve as miniature pressure vessels for the containment of fission products. The buildup of fission gas pressure within the particles during irradiation, coupled with irradiation-induced density changes, possible migration of the fuel kernel, and creep in the coating, produce stresses and other conditions which can lead to coating rupture. The intact Pyrocarbon coatings serve as barriers to gaseous and nonmetallic fission products, while the silicon carbide layers in TRISO coatings serves as a barrier to certain rare earths and metals as well. Since the fission

TABLE 2-2

PROPERTIES OF SPHERICAL FUEL ELEMENTS FOR PR-3000

Heavy metal content	0.96 g ^{235}U, 10.2 g Th
Average burnup (heavy metal), MWd/MT	100,000
(% FIMA)	11
Average fast neutron dose (E>0.1 MeV), n/cm^2	4.3×10^{21}
Crushing strength**	s = 95/95/>1800 kg
Drop impact strength*** (standard test)	s = 95/99.99/>50 drops
Anisotropy factor of A3-matrix	1.1
Abrasion resistance (mg/hr per sphere)	<6
Thermal conductivity at 1000°C (cal/cm sec-°C)	>0.08
Corrosion rate at 1000°C (standard test, mg/cm^2-hr)	$\leq$1.5
Coated Particles	
Kernel diameter, μm	400
Kernel density, g/cm^3	9.5
Thickness of coatings, (buffer/sealing/ outer layer), μm	85/30/80
Density of coating, g/cm^3	1.0/1.6/1.85
Fuel material	$(U-Th)O_2$
Coolant gas temperature range, °C	300-960
Density of fuel spheres in core	0.64

** It must be ensured with 95% confidence that 95% of the elements have a crushing strength of more than 1700 kg.

*** It must be ensured with 95% confidence that 99.99% of the spheres will survive a 4 m free fall onto a closely packed bed of graphite spheres without fracturing or cracking.

product release from failed particles can be orders of magnitude greater than from intact particles, strong dependence is placed on the ability of the particle coatings to remain intact as the primary barriers to fission product release from HTGR fuels. Knowledge of the performance of these coatings under all reactor conditions is required to calculate potential fission product releases from the PR-3000 fuel.

For the TRISO coatings, the potential aspects of chemical failure of the coatings have been described.(6) The failure mechanisms in irradiated coated particles at temperature are well established and of concern for the PNP application, in particular. The current German development program of pebble bed fuel for the high temperature (950°C) PNP and potential HHT applications includes the following:

- Reduced susceptibility of particle kernels to thermal migration and chemical reaction with SiC coatings in TRISO particles.
- Improved fission product retention capability by reducing the in-service failure fraction.
- Improved fuel sphere isostatic molding processes to reduce the heavy metal contamination fraction from the present $\sim 10^{-4}$ to $\sim 10^{-5}$.

With BISO fuel, intact Pyrocarbon coatings continue to retain mobile fission gases and halogens, but metallic fission products such as barium, strontium, and silver are able to diffuse through these coatings and eventually enter the primary coolant. For the HHT-K, the release, transport, and deposition of the metallic fission products on turbomachinery blading appears to rule out the use of BISO-coated fuel particles for this application. Moreover, BISO coatings differ from TRISO coatings in stability since the TRISO silicon carbide layer is dimensionally stable under reactor operating conditions.

For all these reasons, it is particularly important that the fuel irradiation test program provide statistically significant data on reference fuel (kernel material, coating type, matrix formulation, fuel sphere molding, et cetera), particularly at temperatures above normal operating conditions. There should also be acceptable agreement between fuel analytical models used to establish fuel performance criteria and irradiation test data on reference fuel.

Notwithstanding the difficulty in analyzing fuel failure models and the insufficiency of irradiation data described in available literature, the excellent performance of coated fuel particles in the now decommissioned Dragon Reactor in England, the decommissioned Peach Bottom I Reactor in the United States, and the favorable fuel experience to date with the Fort St. Vrain Reactor, indicates that the basic fuel concept is sound and that these particles can be expected to perform well.

It is believed that the improvements from planned programs will be necessary to achieve current interim U.S. licensing criteria for HTGR fuels presented in NRC's NUREG-0111, "Evaluation of High-Temperature Gas Cooled Reactor Fuel Particle Coating Failure Models and Data," November, 1976. The possible use of uranium-thorium dioxide particles in the spherical fuel would constitute a change from U.S. HTGR fuel design and experience, and would be the subject of detailed NRC review prior to regulatory acceptance. In either event, extensive irradiation confirmation of the selected fuel particle type would be required as a requisite for NRC acceptance.

2.2.2 Upper, Radial, and Lower Reflectors

The reference fuel management scheme for the unified PR-3000 pebble bed core is the Once-Through-Then-Out (OTTO) strategy which is characterized by a skewing of the thermal and fast neutron flux towards the upper voided cavity above the core. The upper, inner radial graphite reflectors in this region are backed by large monolithic graphite blocks.[1] Of concern is the dimensional stability of the upper, inner radial reflector of the proposed near-isotropic graphite. Based on conservative projections of the dimensional growth of the inner upper reflector over the 40-year core life, stress levels are projected to exceed the strength of the graphite.[7] The proposed solution is to provide a reflector concept which allows for a degradation of an inner graphite layer of 7 cm thickness at the end of life. The replacement of the reflector or parts of the reflector is considered to be a possible contingency measure. While conservative and appropriate from safety considerations, the decision contributes to complexity in mechanical design and procedural (replacement) requirements later in core life.

Gilso-graphites are not expected to be commercially available for the PNP and HTR-K applications. Owing to domestic German sources, a coal-derived pitch coke is proposed for the SIGRI graphite instead. Another potential issue in the graphite reflector blocks involves tolerance stackup resulting from the large number of reflector elements in the core bottom and dimensional stability under the combined effects of high outlet gas temperatures and fast neutron fluence. Of potential concern is the development of openings in the reflector exceeding the diameter of a fuel sphere and the lodging of fuel elements between reflector blocks. Design provision is being made for a stiff-core lower reflector which is expected to mitigate the potential of this occurence. The design must accommodate imposed static fuel loads, thermal effects, neutron fluence effects, mechanical abrasion and wear effects, and seismic forces. Design validation and graphite material property documentation would be required for U.S. licensing properties. This requirement is discussed further in Section 2.2.6 below.

Since the structural integrity of the reflector elements is critical to the integrity of the core itself, the analytical approaches and allowable stresses will be reviewed in detail by the NRC staff. The use of two-dimensional finite element methods seems to be a reasonable approach. The proposed allowable stresses and other design criteria used in the design of the graphite reflectors (and core support structures), however, may be subject to more conservative revision by U.S. licensing authorities. The NRC has indicated that factors of safety used in the design should be more conservative than those specified in ASME Code Section II, Division I for graphite structures, and distinction should be made between primary and secondary stresses. The NRC is of the opinion that design criteria for graphite should not be established on the same basis as those for metallic reactor components since graphite is a brittle material, while steel is a ductile material, and the modes of failure and structural behavior of the two materials are totally different.[8]

In the United States, the General Atomic Company has proposed stress limits for HTGR structural graphite applications for the emergency and faulted design conditions as "no-loss-of-safety function." NUS foresees difficulty in developing these proposed limits for U.S. licensing (for both the HTGR and the PR-3000) since an acceptable definition of no-loss-of-safety function must be determined, together with a basis for providing a suitable margin for this definition. Other related areas of concern include a lack of definition of the ultimate strength for graphite and the applicability of oxidation rates on graphite structures of different sizes and geometrical configurations than those for which experimental data have been obtained. It is believed that the above issues would have to be satisfactorily resolved before U.S. licensing authorities would accept the pebble bed core reflector and core support structure design.

2.2.3 Fuel Exit Chutes

The PR-3000 pebble bed reactor is unique to U.S. licensing experience in that highly radioactive, hot fuel elements are withdrawn from the reactor core cavity through exit chutes into storage cannisters outside the PCRV. Fuel handling and spent fuel storage systems are discussed in Sections 2.2.7 and 2.2.8. The design criteria for the fuel exit chutes are not described in available literature. Structural considerations are as important in this extension of the primary coolant system boundary as are core-related structures to preclude the possibility of fuel blockage in an unaccessible region of the PCRV. The comments of the preceding section thus apply to the cylindrical graphite duct walls of the exit chutes. Presumably, AVR and THTR experience would be applicable for this application, but appropriate structural analysis would be required to meet United States licensing criteria. Of concern is the possible requirement for remote visual inspection of the fuel exit chutes to ensure their continued integrity throughout plant life. Based upon recent discussions between General Atomics Company and NRC, these concerns have been expressed for HTGRs, and actions to alleviate this situation are under consideration.

2.2.4 Thermal Shield

The thermal shield in the PR-3000 core is a complex safety-grade structure consisting of 1.2 m of internally cooled, graphite or carbon stone. A 10 cm annulus of the shield has a boron content of 1.5%, and a 40 cm gray cast metallic liner provides lateral support. The thermal shield is supported separately from the core on metallic supports extending from the PCRV bottom liner in the core cavity.(1) A 0.7 m annulus is provided between the outer thermal barrier and the inner PCRV liner for purposes of inspection and repair of both the outer thermal shield and the liner. Access and physical room permits the remote control inspection and repair of metallic parts, that is, the liner, the thermal barrier, the core support structure, and the radial and upper reflector supports. Work external to the thermal shield can be performed with the reactor fueled and shutdown.

The bottom thermal barrier consists of a 2 m thick graphite layer, a 35 cm thick layer having a boron content of 1.5%, and a 35 cm layer of gray cast metal. The boronated graphite layer in the thermal shield attenuates the thermal neutrons and the cast metal layer attenuates the gamma rays. The thermal shield also functions to provide radial and axial support for the inner graphite reflectors.

The properties and performance of graphite materials used in the thermal shield, the reflector elements, and the core support structure are very dependent upon the precursor materials and methods used in manufacture. There is no evidence at present that the in-service performance of graphite components can be quantitatively predicted from properties measured on nonirradiated graphite. To establish the means of ensuring that the core graphite will perform in a predictable and safe manner and that the commercial nuclear graphites developed by SIGRI will have the same characteristics as graphite presently being qualified in the United States for HTGR applications, an appropriate standard must be developed. To be acceptable to United States licensing authorities, the standard would require the graphite manufacturers to provide samples of, and qualifying information on, the raw materials used in manufacturing. The NRC position has been that it would not require proprietary information on graphite precursor materials or processing details as long as the sponsoring reactor manufacturer is able to show adequate proof of graphite reproducibility and performance under service conditions. German graphite standards should be reviewed to confirm that they achieve the United States licensing criteria.

2.2.5 Core Support Structure

The PR-3000 core support structure is conceptually similar to that developed for the THTR except for its larger dimensions (twice the core diameter and consequently a 4 times larger core cross section) and the higher reference gas outlet temperature. The inner bottom reflector consists of a near-isotropic reactor graphite. Account must be taken in the mechanical design to provide the necessary exit chutes and hot gas exit channels in the core support structure. The bottom surfaces facing the active core are azimuthally and radially sloped to permit fuel element flow toward the discharge chutes. The bottom core cavity geometry, itself, is subject to further analytical and experimental confirmation. Possible alternative designs are being considered, although the current reference design has 6 discharge chutes with graphite support columns. Thus, the design remains to be established in this important area.

An alternative design for the core support structure consists of layers of graphite ring segments containing diagonal coolant holes of 15 to 25 cm. The arrangement permits good mixing of the different gas streams resulting in a relatively uniform gas outlet discharge temperature profile. Pressure losses in the core support structure approximate 5% of the total circuit pressure loss. The conceptual design of the core support structure appears to offer good structural stability over the approximately 100 m^2 cross-sectional area of the core bottom.

The core support graphite blocks are laterally keyed and mounted on graphite supporting columns on a bottom structure of either graphite, carbon stone, or possibly metallic bases directly from the liner steel bottom. The core support structure is in a conceptual design stage. It is subject to appreciable lateral loads in the HHT-K due to pressure pulses under normal loop shutdowns and startups and loss of electric load transients. HHT-K pressure transients impose significantly larger loadings on all graphite core internals and require detailed examination before final design criteria of core-related graphite structures can be specified.

To meet U.S. licensing criteria, the PR-3000 reactor vessel internals consisting of the core support stucture, the inner and permanent reflectors, and the boronated graphite and cast metal shield of the thermal shield must meet the load combinations specified in ASME Code Section III, Division 2. The design criteria discussed in Section 2.2.2 for the reflectors also apply to the core support structure. Moreover, the in-service inspection requirements of ASME Code Section XI, Division 2, would be applicable for the support structure. The code requires the periodic determination of important structural characteristics of safety-grade graphite components. Increased availability is being provided for remote visual inspection in current HTGR designs. The method by which this would be accomplished in the present PR-3000 core support structure is not known. A possible method other than remote visual examination methods includes the feasibility of monitoring changes of strength of graphite with time or oxidation by using ultrasonic pulses, a technology under development by the Battelle Pacific Northwest Laboratories for the NRC.

2.2.6 Reactivity Control Systems

2.2.6.1 Conventional Control Rods

The number and type of control rods and drives differ for the PNP, HTR-K, and HHT-K applications. The number of control rod drives for the PNP is 156; for the HTR-K and HHT-K, the number is 246 of which 48 are of a mechanical cable drive with control absorbers located in the inner radial reflector. Various rod drives such as pneumatic, hydraulic, and rotary electric are being considered for the in-core rods. The HHT-K employs either hydraulic, pneumatic, or conventional lift rod drives.

For the PNP core, the required control for shutdown to hot equilibrium core from full load operation is 2.7% δk and requires 156 rods inserted to a depth of approximately 1/2 m into the pebble bed. During hot shutdown, the control rods can be moved into the pebble bed to a depth of 1.85 m without exceeding the accepted temperature limit of the metallic sheath surrounding the boronated graphite absorber compacts (1292°F). This depth is reported as sufficient for ensuring hot shutdown; to achieve further temperature reductions and a complete cold shutdown, the rod system must be moved into the pebble bed in several additional steps over a longer period of time until the control bank is fully inserted (4.5 m) into the pebble bed.

The maximum temperatures of the metallic sheath around the cylindrical conventional absorber rods have been computed as a function of rod insertion speed into the fuel bed array. For a limiting insertion rate of 10 cm/s, the temperature of the sheath exceeds 1652°F in approximately 45 seconds from actuation. For lower insertion rates of approximately 1 cm/s, the maximum temperature of the sheath would exceed 1292°F in approximately six minutes. While the calculations require elaboration as to underlying assumptions, the results indicate that an insertion drive velocity of 1 to 2 cm/s may be permissible with regard to allowable metallic sheath temperatures, provided the approximate 1300°F limit for austentic stainless steel could be exceeded for a short period of ten to fifteen minutes. The 156 rods for PNP are derived from a shutdown concept consisting of one reactor shutdown system and one diverse emergency shutdown system (KLAK), with an assumption of two worst adjacent inoperative rod drives.

A second problem unique to pebble bed reactors involves the insertion of control rods to core depths of approximately 4 to 4.5 m to assure the cold, protactinium-free shutdown of the reactor. The last 3 to 5 feet of the insertion causes the greatest problem since the necessary force increases approximately quadratically with penetration depth. At full insertion, the required force based on 1:6 scale model tests is approximately 1.3 times that for the THTR. From still other 1:10 scale model tests, rod insertion forces have been found dependent upon the number of previous insertions, i.e., the porosity of the pebble bed is reduced by repeated rod insertions. This circumstance has led to the development of a helically shaped control rod geometry (Section 2.2.6.2). In summary, the two issues - capability of rod insertion for long-term reactor shutdown and compatibility of absorber material properties with reactor environmental conditions - are believed to have significant safety and licensing implications.

2.2.6.2 Screw-Shaped Control Rods

A control absorber design variant under investigation is a screw-shaped absorber configuration which can be rotationally driven into the pebble bed. The porosity of the pebble bed is not decreased by this configuration and insertion scheme, permitting an unlimited number of insertions with this geometry.

Extensive design and validation test programs would be required to support licensing efforts of this design in the United States. While the absorbers could be designed to contain the equivalent quantity of reactivity control as the conventional cylindrical absorbers, the control rod mechanisms must be modified to provide the combined rotational and insertion forces required. Possible implications of the revised control rod drive, rod drive power, and PCRV penetration diameter should be established.

2.2.6.3 <u>KLAK Poison Spheres</u>

The KLAK poison spheres constitute an alternative reactor shutdown system to conventional control rods. The system is capable of providing approximately 20% δk of reactivity shutdown provided that the small spheres are distributed uniformly in the core. This reactivity shutdown would require approximately one KLAK sphere for each fuel element, or approximately 3,000,000 poison spheres occupying the interstices in the fuel bed array, based on a B_4C volume percentage of one percent. If twice this B_4C content were available, then, approximately half the 3,000,000 small poison spheres would suffice.

Retention of the poison spheres in the fuel bed array under seismic forces is of concern. Tests in FRG have shown that not more than 25% of the poison spheres would leave the core under unstated levels of seismic excitation. The balls are introduced through seven feeder positions which are normal rod drive mechanism penetration locations. Dimensions of the poison spheres are approximately 1 cm in diameter. While details are lacking, the KLAK alternate shutdown system appears to be the most pragmatic of alternative shutdown systems developed for the pebble bed core and is one for which there is U.S. HTGR licensing precedent (Fort St. Vrain).

2.2.6.4 <u>Alternative Shutdown Systems</u>

Two additional alternative shutdown systems are under investigation in FRG. One involves the injection of a gadolinia molecular gas having thermal neutron absorption properties into the helium coolant for reactivity shut-down purposes. Possible material effects with this gaseous absorber, the respective gaseous volumes required, bleed feed charging systems required, and the relative times for injection have not been reported. The concept may have potential but is in a conceptual stage at present. Additional development would be required prior to serious consideration.

2.2.7 Fuel Handling System

The PR-3000 fuel element insertion and withdrawal system permits a fuel element throughput rate of 2,654 spheres per day. There are 43 feed tubes arranged symmetrically in the PCRV top head through which fuel elements are dropped onto the fuel bed array within the core cavity. Six fuel exit chutes are provided in the PCRV bottom head for fuel withdrawal. Fuel elements are inserted and withdrawn within a 2 1/2 hour period each day. With the OTTO fuel management scheme, the axial pebble bed flow, which is intended to be uniform through the core at each radial and azimuthal position, is approximately 0.5 cm per day.

The PR-3000 core has a very low height-to-diameter proportion of approximately 0.5, and a large core diameter to pebble fuel diameter ratio of 187. To assure the even flow of fuel spheres, at least six pebble exit fuel chutes are required, based upon model testing of several design variants. The final selection of the configuration of the core bottom and the positioning and number of the exit chutes remains to be made. Therefore, the important assumption of uniformity of fuel bed motion throughout the core cavity remains to be verified. Moreover, it is known that the flow of spherical elements past a stationary wall (reflector) can result in irregularities in the flow pattern. Surface protrusions and niches in the reflector wall are proposed to preclude these effects. This solution remains to be verified in tests on a larger scale.

Thus, fuel sphere trajectories require additional experimental verification for the PR-3000 core. The pebble bed flow behavior must be incorporated into three-dimensional, multigroup, neutronic diffusion and depletion codes to ensure reliable predictions of core power distributions with life, lifetime reactivity effects, fuel temperatures, and uniformity of outlet gas temperatures. An accurate depletion model, which has been normalized to experimental data (AVR, THTR), is required to confirm that fuel and nonfueled core structures are operated within established limits of temperature, burnup, and fast fluence.

Full-scale modeling of the lower portion of the pebble bed (one-sixth of the core sector) would probably have to be performed to confirm geometric effects on pebble bed flow. The modeling would constitute a major development program. Moreover, the insertion procedure of dropping the spherical fuel elements from the entrance chutes onto the pebble bed within the core cavity -- a distance of approximately 13 ft. (4 m) -- would be unique to U.S. experience. Table 2-2 presents the specifications of drop impact strength for pebble bed fuel: a 95% confidence level that 99.99% of the fuel spheres will survive a 13.1 ft. (4 m) free fall onto a closely packed bed of graphite fuel spheres without fracturing and cracking. The U.S. Regulatory Commission could impose a more stringent standard, such as a 99% confidence level, which would significantly impact fuel production and test methods.

2.2.8 Spent Fuel Storage

The present conceptual fuel element storage capacity for the HTR-K, HHT-K, and PNP designs is six years (approximately 6,000,000 fuel spheres). The number of spent fuel element containers is in question: data in available references range from 164 to approximately 4,400. The storage capacity in the core refueling system is also ambiguous. Despite the ambiguity in the details of the PR-3000 spent fuel storage system, the overriding consideration is one of inadequate capacity. In the U.S., Away From Reactor (AFR) spent fuel storage for commercial reactors will not be available before the mid-1980's. Presently licensed commercial nuclear plants are now required to have provision for ten years of spent fuel storage capacity. Presumably, a comparable specification would be imposed upon a commercial pebble bed reactor system, whether for process heat or electric production purposes.

The increase in capacity by a factor of approximately two will impact significantly upon spent fuel storage design. Studies for the surface storage of spent fuel spheres are under way in FRG. Moreover, the underground storage in a salt-bed repository of spent AVR fuel spheres (100,000) is presently being implemented as part of a demonstration program. Definition of long-term spent fuel sphere storage requirements remains to be established in the FRG and the United States.

2.2.9 Alternative Fuel Cycles

Considerable attention has recently been focused on the possibility of modifying the fuel cycles of gas cooled reactors to reduce the risk of diversion of fissile material by governments or terrorist organizations for purposes of constructing nuclear weapons. Prominent among the possibilities considered is the use of "denatured" or medium-enrichment uranium in lieu of fully enriched material. These studies are the subject of the International Fuel Cycle Evaluation (INFCE) being conducted under international auspices. In the United States, it is reported that

1. The use of thorium and uranium of about 20% enrichment, or more, in HTGR's would improve both resource utilization and fuel cycle economics over the use of low-enrichment uranium alone.

2. For a nonrecycle mode of operation, significant reductions in U_3O_8 requirements can be achieved by optimizing the low-enriched uranium of low-enriched uranium-thorium cycle in the HTGR, and

3. The HTGR appears to have some inherent safeguard features not found in LWRs.

There are several strategies that use denatured fuel. Most involve the use of an energy park which

- Receives spent fuel from satellite power reactors
- Reprocesses the spent fuel
- Converts recovered Pu to U-233 in another reactor within the energy center (with a breeder or an HTGR, for example)
- Fabricates fuel elements with denatured U-233 for shipment to the satellite reactors outside the energy center.

The concept involves the use of fuel outside the energy park only of low-enrichment uranium and denatured U-233, and radioactive spent fuel. All safeguards-sensitive operations would take place within the energy center, which might be operated under some form of international auspice.

The particle configuration of HTGR fuel would also permit the use of separate low-enrichment particles and ThO_2 particles. The discharged fuel would then contain depleted low-enrichment uranium particles which contain Pu, and ThO_2 (fertile) particles which contain U-233. The former could be disposed of without reprocessing. The latter could be processed and the U-233 separately recovered for subsequent recycle. The low-enriched uranium cycle with or without recycle is the reference fuel system for LWRs, and it is believed that the HTGR could operate economically on this cycle also.

The option exists for recycling only the uranium or both the uranium and plutonium. The Pu-Th cycle could be used in energy centers to consume Pu and produce both electricity and U-233. It is conceivable that the best strategy for the long-term handling of Pu is to consume it in internationally safeguarded energy parks, thus reducing existing international stocks of this material. U-233 has the dual advantage of being denaturable and having a high associated radioactivity. As a result, the number of shielded fuel casks required to transport a given mass of U-233 recycle fuel would be much larger than for Pu-bearing mixed-oxide LWR recycle fuel. This would reduce the chances for diversion of significant quantities of the U-233 fissile inventory or facilitate its rediscovery if diverted, with attendant safeguard benefits.

Recently updated uranium and enrichment requirements for low-enrichment uranium and low-enrichment uranium-thorium fuel cycles for the HTGR have been reported in the United States.[9,10] From these studies, the following findings were made:

1. Low-enriched uranium and low-enriched uranium-thorium fuel cycles for a nonrecycle option require 15 to 20% less U_3O_8 than would the LWR.

2. Significant additional reductions in U_3O_8 and separative work (SWU) for HTGRs are possible with the recycling of the uranium discharged in the low-enriched uranium-thorium cycle.

3. Further reductions in the 30-year U_3O_8 requirements for HTGRs could be achieved if the U-233 could be recycled without denaturing.

In the FRG, the low-enrichment uranium and low-enriched uranium-thorium fuel cycles are being brought to a design stage comparable to the reference highly enriched uranium-thorium fuel cycle. Specific details of the differences in the low-enriched uranium and denatured thorium fuel cycle on out-of-core fuel management and in-core reactivity coefficients and control requirements remain to be established. With plutonium fuel systems in LWRs, additional movable control is required to maintain reactivity shutdown margins and compensate for increased reactivity coefficients. It is to be expected that comparable requirements may exist for U-233 fuel systems in a gas cooled reactor. Moreover, the characteristic fission product release fractions for the low-enriched uranium and low-enriched uranium-thorium fuel systems remain to be accurately established. It is known, for example, that appreciably higher Ag-110 m release is characteristic of low-enriched uranium fuel, which may pose significant additional difficulties for the HHT-K application as discussed in Sections 2.2.1.1 and 2.2.10.

2.2.10 Fuel Integrity and Fission Product Behavior

In high-temperature gas-cooled reactors, the major gamma-emitting fission products which plate out and restrict inspection, maintenance, and repair of components are Cs-137, Cs-134, and Ag-110m. It is these fission products which dictate the most stringent conditions for minimum release of the reactor fuel.

One measure of special importance for maintenance and inspection purposes is the Dose Constant (DC): the gamma dose rate which a one mCi point source provides at a distance of 1 cm in mrem/hr. These constants for the Ag and Cs isotopes are reported as follows:[16]

	$^{110}Ag_m$	^{134}Cs	^{137}Cs
$T_{1/2}$	253 days	2.1 days	30 years
DC	15.4	8.9	3.9

Thus, Ag-110m is shown to be more significant by a factor of approximately four than Cs-137 in the tabulation above. Whether this fission product is important after thirty years of reactor operation will depend on its long-term deposition characteristics.

Comparisons between the CS-137 release per year, as given by AVR and Peach Bottom-1 measurements and calculations, indicate that the AVR performance has remained well within the uncertainties of emission and transport for a small pebble bed reactor core with mixed uranium/thorium oxide, BISO-coated particles. The cesium release is dependent upon the type of coated particles employed (HTI versus LTI), their integrity, the fuel kernels utilized, and the extent of uranium contamination in the fuel-free graphite shell and matrix.

Mixed-oxide TRISO-coated particle designs, without any particle failure, yield Cs-137 release values which are lower by a factor of approximately seven when compared to mixed-oxide BISO particles with fractional uranium contamination of 2×10^{-5}. The lack of transport data for Ag precludes similar comparisons of the relative capabilities of the two coatings. If the SiC coating is as effective a barrier for Ag as it is for Cs, TRISO-coated particles could reduce the Ag release to acceptable values. This could be important if a slightly enriched uranium fuel cycle were selected since this fuel emits Ag more copiously.(16) Continued development of coated particle failure and release mechanisms is required for the PNP and HHT-K applications (950°C, 850°C reactor outlet temperature), respectively, since gaseous and solid fission product emission, transport and deposition in the primary system must be reduced to the lowest practicable level.

With the HHT-K, decontamination factors of 500 to 1000 are estimated to be required on turbomachinery surfaces to permit maintenance and repair on the basis of current state-of-art fuel technology. Decontamination solutions and procedures which are sufficiently effective to permit turbomachinery maintenance and repair may have a deleterious effect on machine lifetime. Therefore decontamination methods are still to be improved. In addition, improved fission product retention in HHT-K coated particle fuel appears a particularly important development objective.

The NRC interim failure release rate calculational model cited in Section 2.2.1 (NUREG-0111) is conservative, particularly in its assumptions as to failure rates for BISO particles based on U.S. irradiation data. The potential fission product release from the reference 3000 MWth fuel designs for both the reference highly enriched uranium/thorium, and, separately, slightly enriched uranium-thorium fuel cycles should be evaluated with the NRC model to assess safety and licensing issues.

2.3 NUCLEAR DESIGN

2.3.1 Design Bases

The PR-3000 nuclear design is based on the thorium-uranium fuel cycle in which uranium-235 enriched to 93 percent is used as the fissile material and thorium-232 is used as the fertile material. Table 2-1 summarizes the nuclear design parameters for the HTR-K, HHT-K, and PNP applications, and compares them with U.S. licensed HTGRs which employ a similar fuel cycle. In Table 2-1, the GASSAR-6 and Fulton Station Nuclear designs are identical.

The PR-3000 core is well represented neutronically as a semi-homogenous, graphite-moderated assembly where the helium coolant has no nuclear worth. The fissile and fertile fuels are zoned into two radially symmetric regions in the PR-3000 core. Control rod sequencing can be designed to supplement fuel zoning in achieving desirable power and fuel temperature distributions. Burnable poison spheres could be introduced if needed to help control overall local power distributions.

From a review of the available literature on the PR-3000 core, the following statements have been formulated as partial nuclear design criteria required for U.S. licensing purposes which could meet the objectives of General Design Criteria 10, 11, 12, 26, 27 and 28.

1. The power generation and fuel cycle objectives of the core design will be constrained by thermal and metallurgical limits.
2. The isothermal and fuel temperature coefficients shall be negative from room temperature to beyond 3000°K.
3. The core shall be designed such that axial xenon oscillations shall not occur.
4. Instrumentation shall be provided to detect any radial flux tilt or radial or azimuthal oscillations, and, should such conditions occur, these conditions would be correctable by appropriate control rod action.

5. The fuel in the core shall be appropriately zoned to minimize radial and axial gross and local power tilts and to maintain the power peaks within design limits throughout life, with due allowance for uncertainties of calculations, pebble bed movement, and uranium-235 loading.

6. Safe shutdown by primary control rods shall be achieved within an acceptable margin of negative reactivity which shall include allowances for uncertainties under any of the following conditions:

   (a) Indefinite shutdown at room temperature,

   (b) shutdown at hot, standby temperatures,

   (c) shutdown at hot, operating temperatures.

7. Core excess reactivity shall be designed to be compensated by primary control rods.

8. An alternative shutdown system shall provide reactivity control redundancy through a poison insertion mechanism which is actuated independently from the primary system of control rods.

For the shutdown margins specified in the suggested design bases above, further consideration should be given to the PR-3000 design which bases the excess reactivity at hot-standby temperatures on an approximate two week decay period for protactinium-233 (see Section 2.3.4). For the reserve shutdown system, the system should be capable of shutting down the reactor following an anticipated transient without scram (ATWS). It cannot be confirmed from a review of the PR-3000 core design(1) that the suggested design bases have been achieved. It is believed that comparable nuclear design bases would, however, be required to comply with U.S. General Design Criteria.

2.3.2 OTTO Fuel Management

The OTTO fuel management scheme skews the neutron flux to the upper portions of the core and voided cavity region above the fuel bed(11). A basic feature of the OTTO fuel management scheme is the mutual coupling between the core spatial power distributions and the movement or trajectory of the fuel spheres through the core. Fuel sphere temperature, specific power, and cumulative burnup are all influenced by individual fuel sphere trajectory.

As reported earlier, the trajectories of the fuel spheres require additional experimental verification of the PR-3000 core bottom configuration and exit geometry. The FRG neutronic model VSOP accounts not only for the time-dependent depletion of the core, but also for the spatial relocation of the fuel in the pebble bed array. The code reportedly incorporates necessary transport theory corrections to the neutron diffusion model for the voided cavity region, and is stated to account accurately for the reactivity effects of control rods in the upper reflector and cavity region. The transport theory corrections are mandatory due to limitations of diffusion theory in this region. Further confirmation of the accuracy of the absolute fast neutron flux, the thermal neutron flux, and the reactivity effects in this region is urgently required as discussed in Section 2.3.3.

2.3.3 Analytical Methods and Verification

Volume 2 of Reference 1 discusses the zero-power critical facility confirmation of power distributions of the PR-3000 mockup core with the OTTO fuel cycle. The fast neutron flux in the upper inner radial reflector, for example, is stated to be predicted to within 30% by different neutronic design models. The neutronic models, however, appear to be one- and two-dimensional transport theory models and not the two- and three-dimensional diffusion theory models with transport corrections (e.g., VSOP) which are used for normal design purposes.

The motion of fuel spheres in the pebble-bed array for a variety of radial reflector, bottom reflector and fuel exit geometries has been experimentally confirmed.[1] As previously stated, the insertion forces required for full insertion of conventional cylindrical control rods into a mockup PR-3000 fuel bed array has also been measured in 1:6 scale model tests. The reactivity effects of various control rod programs have been experimentally measured and neutronic models used for design confirmed. Reactivity coefficients for the cold, clean PR-3000 core have been measured and confirmed by analytical methods. The PR-3000 reactivity coefficients have not been measured at elevated temperatures, however, other than in the AVR reactor. Calculation of reactivity coefficients in the hot, equilibrium PR-3000 core is complicated due to the dependence of the coefficients on a number of effects as described in Section 2.3.4. Further experimental confirmation of the hot, equilibrium coefficients would be desirable. Documentation supporting the experimental confirmation of neutronic and reactivity coefficient effects would be required by U.S. licensing authorities. It is understood that the Los Alamos Scientific Laboratory has been investigating the neutronic performance of HTGRs for the NRC, including the PR-3000.

2.3.4 Reactivity Budget and Reactivity Coefficients

Reactivity coefficients quantify the inherent nuclear feedback that develops in response to a change in the neutron multiplication status, i.e., the reactivity of the core. Since reactivity coefficients are required to predict the core dynamic response to both anticipated transients and postulated accidents, these coefficients must be well understood. For use with postulated accidents, they must be specified in a conservative manner.

The response of the PR-3000 core to gross temperature changes is quantified by the isothermal temperature coefficient, which may be considered as the sum of the temperature coefficients of both the moderator and the fuel.

In this description, the fuel component would be comprised of only the Doppler effect and is computed from changes in the cross sections which occur in the resonance and epithermal energy ranges. The isothermal coefficient is determined by changes in cross sections at all energy levels. Therefore, the moderator coefficient would be the difference between the Doppler and isothermal coefficients. The computed coefficients depend on the fission product inventory in the core which varies with fuel depletion and control rod programming.

While computation of the reactivity coefficient is complicated due to these dependencies, the principle contributor to the coefficients is the increase in resonance absorption in thorium due to the Doppler effect. The moderator coefficient is smaller than the Doppler coefficient at operating temperature, but becomes positive at operating temperatures during the fuel cycle. The overall isothermal temperature coefficient at operating power levels, however, ranges from about -5×10^{-5} $\Delta\rho$ per degree Kelvin at the beginning of life to about -2×10^{-5} $\Delta\rho$ per degree Kelvin at the end of life. These values, if confirmed, are sufficiently negative to provide reasonable assurance that the PR-3000 core can be made inherently stable against positive reactivity transients from anticipated occurrences and postulated accidents, thus satisfying General Design Criterion 11. If difficulties should arise, the possibility exists for seeding the core with some material such as rhodium to guarantee a strongly negative Doppler reactivity over the full range of operating temperatures and transients throughout core life.

One assumption made in the PR-3000 reactivity coefficient calculations is that of a uniform temperature rise over the entire core. It would be expected, on the other hand, that local variations of temperature changes resulting from transients could occur in the large PR-3000 core. The spatial dependence of the isothermal reactivity coefficient should therefore be examined. The Doppler coefficients derived from resonance broadening of the fertile material have been calculated to vary from -2.4 to -3.3×10^{-5} $\Delta\rho$ per degree Kelvin for the thorium cycle, and between -4.0 and -4.5×10^{-5} $\Delta\rho$ per degree Kelvin for the low-enriched uranium cycle. For the equilibrium core, small positive values of the Doppler coefficient are reported, but the isothermal temperature coefficient remains negative.

The excess reactivity of the PR-3000 core is mechanically controlled by the movable control rods positioned in the upper reflector or core cavity region. The design excess reactivity is required to compensate for reactivity losses due to moderator and fuel temperature increases, fuel depletion, and poisoning by xenon, samarium, protactinium, and other fission products. Special consideration is necessary for protactinium-233, an intermediate isotope in the production of uranium-233 by neutron capture, and thorium-232. It decays to uranium-233 with a half life of 27.4 days and presents a number of neutron absorption cross sections that, like xenon-135, must be separately considered in the neutron analysis and control system design.

The total reactivity requirements for the PR-3000 core as reported[11] are tabulated below.

PR-3000 Reactivity Budget

	Reactivity Effect	Percent ΔK/K
1.	Long-term Shutdown (Xenon-135 decay, Protactinium-233 decay and uranium-233 build-in, temperature reduction to 20°C)	10.0
2.	Xenon override (100%/40%/100% Power profile)	2.3
3.	Most reactive rod pair stuckout	3.0
4.	Error in fuel loading	0.5
5.	Grid frequency regulation	0.5
6.	Reactivity contingency	0.5
7.	Fission product build-in	1.5
8.	Calculational uncertainty	1.6
	Subtotal	19.9

Thus the total control swing is approximately 20%, and the cold shutdown requirement is approximately one-half of this. The available control appears sufficient to shutdown the reactor with an acceptable margin for various conditions encountered during the fuel cycle. It could not be determined, however, what the minimum shutdown margin is at the most reactive time in core life with a shutdown to room temperature with a maximum-worth rod pair stuck out. In addition, further analysis appears required of the reactivity insertions possible from the ingress of water-steam under accident conditions for the PR-3000 core.

2.3.5 Xenon Stability

A potential detection and reactivity control issue with the PR-3000 core and OTTO fuel management is the sensitivity of the core to xenon instabilities. Studies have shown that damped radial and azimuthal xenon oscillations can be induced in the PR-3000 core under normal operating conditions. Detection schemes may include the positioning of the neutron detectors at the top of the cavity region in fuel entrance chutes to supplement the ex-vessel power detectors in the PCRV. The remedial control scheme involves the motion of control rods in the upper axial reflection region. Reportedly, a few inches of control rod insertion in the reflector are sufficient to provide radial power flattening and xenon control in the OTTO core.

The in-core detectors have not been defined with respect to either number, location, or performance requirements. The potential for xenon instability increases with reactor size, thermal neutron flux level, and reduced negative isothermal reactivity coefficients. The PR-3000 pebble bed height is approximately 5.5 meters, and this height tends to limit axial xenon instability. The PR-3000 core diameter is greater than 10 meters (11.2 m), however, and is therefore susceptible to radial and azimuthal oscillations.

Furthermore, it is believed that the initial fuel cycle on the OTTO management scheme would be somewhat less stable than the equilibrium core. While ex-vessel neutron detectors would be available for detecting region power tilts, regional monitoring of the hot-gas outlet temperatures may provide an alternative means of detecting regional power distribution mismatch. Further confirmation that the core can be designed such that divergent axial xenon oscillations do not occur is believed required. Further confirmation that radial and azimuthal oscillations which will occur can be reliably detected and controlled is also believed required.

2.3.6 In-Core Instrumentation

It is a U.S. Regulatory requirement that instrumentation be provided to detect any radial flux tilt or radial and azimuthal flux oscillation. It has been customary in the U.S. for the initial startup and power demonstration program of a new core design of a commercial reactor that the actual core power distributions be confirmed. It is expected that a similar requirement would apply for a first-of-kind reactor concept such as the PR-3000 core. At present, the lack of an in-core instrumentation capability for hot, low-power, power distribution confirmation in the PR-3000 core design is considered a potential deficiency which may have to be rectified.

2.4 THERMAL AND FLUID MECHANICAL DESIGN

2.4.1 Design Bases

Due to the high exit temperature of the helium coolant in the PR-3000 core, design bases must be provided that address the design limits of critical components throughout the primary coolant system as well as the fuel and core design limits. The core thermal and fluid mechanical design bases are intended to protect the integrity of

- The reactor primary coolant system boundary
- The fission product barriers within the fuel
- The safety grade reactor core structure.

The PCRV, its cavity liner enclosures, the steam generators and heat exchangers, the CACS heat exchanger surfaces, and portions of the main and CACS helium coolant blowers constitute the primary coolant system boundary. The upper radial and lower graphite support structures, the graphite fuel spheres, and the reflector elements define the coolant flow geometry of the pebble bed core. Fuel particle coatings are the primary fission product barriers of the core, and the fuel kernels and fuel sphere graphite matrix act as additional barriers to the escape fission products from the core.

U.S. Regulatory Requirements (General Design Criterion 10) require thermal damage limits for Normal, Upset, Emergency, and Faulted categories of plant conditions, such as those suggested in Table 2-3. The table defines each plant condition, suggests the amount of damage to be tolerated for each condition, and presents proposed quantitative thermal limits for the fuel, graphite components, and control rods including their metallic components. It also provides proposed design limits for heat exchangers and thermal barriers in terms of average and transient (hot-streak) temperatures of the core outlet flow where time-at-temperature transients are considered. The table also suggests explicit limits for the core support structure and all essential equipment necessary for safe shutdown of the reactor. The availability of the requisite information for the PR-3000 core and PNP, HTR-K, and HHT-K applications to confirm that the limits suggested in Table 2-3 can be achieved is lacking. It is believed that these data would have to be provided for U.S. licensing purposes. From available data,[1] however, the thermal bases may be deemed acceptable for preliminary design by U.S. licensing authorities provided that flexibility is retained in design choices to allow for the possible reduction of some temperature limits by U.S. authorities.

TABLE 2-3

SUGGESTED U.S. THERMAL DESIGN BASES FOR PR-3000

Category Condition	Definition	Description	Fuel	Graphite	Average Core Outlet[a]	Coolant Average Region Outlet[b]	Local Coolant Hot Streaks	Control Rod Cladding and Spine
Normal	Conditions occurring normal plant operation, start-up rated power load changes, shutdowns, and refueling[c]	No damage tolerated that requires reactor shutdown	Calculated number fuel particle coatings shall be limited such that the annual averaged value of circulating activity does not exceed a specified activity level.	4350°F	Steam generator limits: ~1420°F for steady state, ~1460°F for transients up to 15 min.	Steam generator limits: ~1520°F for steady state; ~1620°F for transients up to 24 hr; ~1770°F for transients up to 15 min.	Thermal barrier limits: (d) ~1700°F for mineral fibre wool; ~2140°F for ceramic blocks	~1600°F for steady state
Upset	Deviations from normal conditions which are expected with moderate frequency[c]							~2000°F for 1 hr. integrated over control rod lifetime
Emergency	Conditions having low probability of occurrence which are included to provide assurance that no gross loss of structural integrity will result[c]	Some repair to system may be required before restart	The reactor can be shut down to a safe condition with a small amount of fuel particle coating failure.	4530°F	CAHE[e] limit: ~1600°F for long-term operation		CAHE[e] limit: ~1680°F for long-term operation. Thermal barrier limits: (d) ~1800°F for 10 hr for mineral fibre; ~2500°F for 10 hr for ceramic blocks	~2000°F

TABLE 2-3 (Continued)

SUGGESTED U.S. THERMAL DESIGN BASES FOR PR-3000

Category Condition	Definition	Description	Fuel	Graphite	Average Core Outlet[a]	Coolant Average Region Outlet[b]	Local Coolant Hot Streaks	Control Rod Cladding and Spine
Faulted	Extremely low probability, postulated conditions whose consequences may be such that considerations of public safety may be involved[c]	Safe reactor shutdown and continued core cooling capability required		5430°F	CAHE limits: 1600°F for long-term operation with PCRV pressurized; ~1700°F for long-term operation with PCRV depressurized		CAHE limits: ~1680°F for for long-term operation with PCRV pressurized; ~1900°F for 4 hr with PCRV depressurized. Thermal barrier limits: (d) ~2000°F for 1 hr for mineral fibre; 3000°F for 1 hr for ceramic blocks	4300°F

(a) Including side reflector, and thermal shield bypasses.

(b) Including reflector control channel bypasses (HTR-K).

(c) ASME III, Paragraph NA 2110 NB 3113.

(d) Local coolant hot streaks are limited so that they do not exceed the local, continuous thermal barrier hot face surface temperature indicated.

(e) CAHE-Core Auxiliary Heat Exchanger.

2.4.2 Description of Analysis

The thermal and coolant flow design for the PR-3000 core is similar to that presently under evaluation in FRG for the THTR reactor. The movable, pebble bed fuel array of the PR-3000 core is, of course, quite unlike that of the prismatic fuel block design of the Fort St. Vrain and GASSAR-6 designs with which U.S. Regulatory authorities are familiar. The principal analysis aspects of the PR-3000 design that would appear to require review and evaluation (analytical models and assumptions) due to geometry differences with U.S. HTGRs include fuel-coolant geometry and pressure drop, potential for hot-streaks exiting from the core, cooling under depressurized conditions, thermal conduction within adjacent reflector elements, and consideration of laminar and transition flow regimes in the core regions.

The heat transfer and fluid flow characteristics for the PR-3000 and U.S. HTGRs vary greatly for Emergency and Faulted conditions from Normal conditions. Design areas reviewed by U.S. licensing authorities would include core temperature profiles, fluid flow parameters, flux tilt considerations, core coolant flow distribution, core pressure drops and fluid dynamic loads, flow transition regimes, thermal effects of operational transients, and uncertainties in estimates. It has not been possible to assess the PR-3000 core in these areas but it is believed such analyses would be required in support of U.S. licensing efforts. A summary of the principal issues in each of the eight areas is presented.

1. Temperature Profiles

   The principal uncertainties in the calculations of temperatures within the fuel elements and surface heat fluxes must be specified, including the conduction from the fueled matrix and individual fuel sphere power generation.

2. Fluid Flow Parameters

   The domain of transition flow ranges from a Reynolds number of 2000 for fully laminar flow to approximately 6000 for fully turbulent flow. Coefficients in the transition regime are usually found by linear interpolation between the upper limiting value in the laminar range and the lower limiting value in the turbulent range. Confirmation of these coefficients would have to include experiments for representative geometries, environmental conditions, and heat fluxes typical of those encountered under PR-3000 core service conditions. Since transition and laminar flows may be controlling in certain postulated transients and accidents, additional confirmatory research would probably be required in support of U.S. licensing efforts.

3. <u>Power Distribution Considerations</u>

Power generation within any core region varies over the annual fuel cycle life as a result of control rod motion, fuel depletion, and the buildup of a fission product inventory. This information, combined with the history of fast fluence and burnup of TRISO and BISO particles would provide the basis for calculating the fraction of failed fuel particle coatings. Knowledge of the failed coating fraction would then permit calculation of the total fission product release from the core as a function of life for comparison with design limits and experimental data.

4. <u>Core of Coolant Flow Distribution</u>

The analysis must describe the flow control through fueled and non-fueled (bypass) regions. The bypass flow fraction would be identified as a percentage of total core flow and apportioned to core, reflector, and thermal shield regions. The potential for cross (shunt) flows in the core would have to be established.

5. <u>Core Pressure Drops and Fluid Dynamic Loads</u>

Core pressure drops would have to be established and compared with experiment. Fluid dynamic loads would have to be computed for postulated reactor depressurization accidents on selected components of the primary coolant system. The analyses should include the bases for selection of the components, the analytical methodology, the consequences of the loads reported, and the experimental programs to confirm the analyses.

6. Flow Transition Data

The change from turbulent to laminar flow conditions under startup, shutdown, and low power operation for postulated abnormal conditions must be documented. Low helium flow rates that result in laminar flow conditions characteristically have high friction factors and reduced heat transfer coefficients. The potential for local flow stagnation exists under such circumstances. Additional experimental fluid mechanical research in this area may be required.

7. Thermal Effects of Operational Transients

The analytical methods for examining operational transients which may result in fuel temperature excursions due to increases in regional ratios of power to flow in the PR-3000 core would have to be documented.

8. Uncertainties in Estimates

The systematic and random errors in the thermal and fluid mechanical analysis would have to be presented. Systematic uncertainties would include measurement errors for region exit coolant temperature, core bypass flow (reflectors, thermal shield), and possible core cross flow. Random uncertainties would include manufacturing parameters, flow maldistribution in the inlet plenum, entrance and exit flow conditions, and material properties including thermal conductivity of fuel, nonfueled graphite, power distributions, and potential for graphite reflector dimensional changes affecting bypass flows.

2.4.3 Performance Criteria, Testing and Verification

The design bases previously presented in Section 2.3.1 specify Emergency and Faulted conditions as well as Normal and Upset conditions, and thermal and metallurgical limits of essential components in the primary coolant system as well as the core. Most of the information presented in Reference 1 and supporting documentation pertains to Normal and Upset Conditions for the core only. Justification of the PR-3000 core design is not considered sufficient in the areas of Emergency and Faulted Conditions, and on the basis of satisfying the thermal and metallurgical limits of essential components of the primary coolant system. Potential inadequacies include the possibility of flow reversal in the event of a postulated loss of flow incident, and a quantitative understanding of hot streaks of core outlet gas under a variety of normal, abnormal, and accident plant conditions.

2.4.4 Instrumentation Requirements

The PR-3000 core should be thermally instrumented to measure both inlet reactor coolant and regional reactor outlet gas temperatures in the core support structure. It appears that such thermal instrumentation could be provided without undue difficulty. Requirements for in-core neutron detection requirements are addressed in Section 2.3.6.

3.0 REACTOR COOLANT SYSTEM

3.1 SUMMARY DESCRIPTION

3.1.1 Schematic Flow Diagram

The process nuclear heat plant to hydrogenate lignite (HKV) consists of a gasification plant and a power plant. The reactor coolant system consists of the high temperature reactor, steam reformer (split tube heat exchanger), steam generator, gas circulator, reheat outlet system, and related gas ducting. These components are arranged within the prestressed concrete reactor vessel (PCRV). The reactor primary circuit consists of six parallel loops, each consisting of a series-connected steam reformer, steam generator, and circulator. The six pod cavities containing the steam reformers are arranged symmetrically around the central reactor cavity. The coolant circulators are mounted vertically below the steam reformer cavities. The remaining four pod cavities contain the cooler, circulators, and duct work of the reheat outlet systems.

The gas duct work connecting the reactor and steam reformer, steam reformer and steam generator, steam generator and circulator, and reactor-reheat outlet system are coaxial, horizontally arranged. The process gas ducting leading from the PCRV to the balance of plant outside containment is run through a horizontal concrete pipe chase beneath the PCRV for safety purposes. The PCRV penetrations for the feedwater and main steam lines for the steam generators are also bottom entry. The flow schematics are shown in drawings 3.3.1-1 and 3.3.1-2 of Reference 1. Briefly, of the 3000 MWth provided by the reactor, 692 MWth is allocated to the steam reformer, and 2,308 MWth is allocated to the steam generator. System temperatures, pressures, and flow rates are as shown in the schematic diagrams of Figure 3.3.1-1 in Reference 1.

The PNP nuclear plant to gasify soft coal (WKV) consists of a gasification plant and a power plant. The system consists of a He/He heat exchanger and steam generator, plus other components which are similar to the HKV plant.

The plant operational concept is as follows:(1)

- An annual load factor of at least 0.86 is expected.

- The gasification plant will be base loaded. The methane output of the gasification plant should be adjustable over the 75 to 100 percent range.
- The operational load variation of the gasification plant should be not greater than 10 percent per hour.
- The process gas temperature at the exit of the catalytic converter should be maintained at 810°C, plus or minus 10°C.
- The gasification plant offers the demand to the reactor plant; the steam reheat system receives no load from the gasification plant under normal operation.
- Under all normal operating conditions the process gas pressure (45 bar) should be higher than the pressure in the primary circuit (approximately 39 bar).
- In the event of the temporary nonavailability of the gasification plant, the reactor power level will be reduced to that consistent with electric production capabilities.
- Coolers in the converter of the gasification plant are sized to condense the steam effluent of the steam reformer, even during temporary unavailability of the gasification plant.

3.1.2 Elevation Drawing

Elevation and cross-sectional drawings have been reported.(1) Section 2.4 of this report summarizes the PCRV dimensional characteristics.

3.2 PRIMARY COOLANT PRESSURE BOUNDARY INTEGRITY

3.2.1 Design Criteria for Primary Coolant Pressure Boundary Components

Two different primary system arrangements are under consideration as a result of differing helium-to-helium heat exchanger concepts. One arrangement is based on four helium-to-helium heat exchangers per loop employing helical coils with gas distribution through a centrally located hot gas distributor duct. The other arrangement is based on two modularly constructed, U-tube, helium-to-helium heat exchangers per loop which are located in separate pod cavities. In this heat exchanger design, the helium inlet and outlet ducting is at the bottom with the helium circulator at the top of the cavity. The first arrangement is depicted in Figures 4.2.1-1 and 4.2.1-2 of Volume 1 of Reference 1. The second arrangement is depicted in Figures 4.2.1-3 and

4.2.1-4 of the same volume. In this section the design criteria of the PCPB components are presented without further reference to alternative plant arrangements.

(1) Hot Gas Ducts

The gas ducting is arranged so that all hot gas ducts are located internally and concentrically within the cold return duct, i.e., within the cold bypass flow of the cavities of the PCRV. Construction of the hot gas ducts has been reported.(1) Specific design criteria for the hot gas ducting is not in available references. NUS believes, however, appropriate design criteria for the hot ducting should include the following:(12)

- It should be removable and exchangeable.
- It should be capable of tolerating depressurization of the primary system at rates of approximately 10/bar/sec for a short time (HHT-K maximum depressurization rate).
- Materials with well-known long-term behavior (20 years or more) should be used.
- All metallic materials should be operated at as low a temperature as possible.
- Coaxial hot ducts should be surrounded by cold helium at higher pressure so that , in case of leakage, cold helium would flow into the hot system.
- Heat fluxes through the hot duct wall should be limited to approximately 140 to 170 KW/m^2 corresponding to a ΔT of about 50°C to preclude excessive thermal stresses in the duct wall.

(2) Helium-to-Helium Heat Exchanger for Steam Gasification (WKV)

Thermal criteria for the two helium-to-helium heat exchangers designs are presented in Tables 3-1 and 3-2. Table 3-1 presents the helical counter flow design and Table 3.2, the U-tube counterflow design. The tables incorporate NUS calculational checks of the thermal performance and heat exchanger effectiveness of these units. Agreement to within 3 percent of the reported heat loads was obtained.

The two sets of NUS calculated thermal performance values in Table 3-1 are based on two input data sets which differ only in outlet temperature

TABLE 3-1

3000 MWT PNP HELICAL He/He HEAT EXCHANGER

NUS HEAT TRANSFER SUMMARY
(950°C Reactor Outlet)

Assumed Parameters	Low Pressure Shell Side	High Pressure Tube Side
Reference	1	1
Mass flow, lb/hr	2.9365×10^5	2.8810×10^5
Inlet temperature, °F	1742	500
Outlet temperature, °F	572	1652
Inlet pressure, psia	580.13	609.14
Outlet pressure, psia	573.00	605.34
Helium specific heat, BTU/lb°F	1.25	1.25
Effective heat transfer surface, ft^2	41.7653×10^3	41.7653×10^3

Calculated Thermal Performance Values (per loop)	Reference Counter Flow		Alternate Cross Flow	
Outlet temperature, °F	1652 (tube side)	572 (shell side)	1652 (tube side)	572 (shell side)
Effectiveness, E, %	0.9275	0.9602	0.9275	0.9602
Thermal Conductance, UA, BTU/hr °F	4.1258×10^6	7.157×10^6	2.6506×10^7	6.7541×10^7
Over Heat Transfer Coefficient, U, BTU/hr ft^2 °F	98.79*	171.36**	98.79*	171.36**
Heat Transfer, Q, BTU/hr	4.1484×10^8	4.2947×10^8	4.1484×10^8	4.2947×10^8
Log Mean Temperature Difference, LMTD, °F	100.55	60.01	15.65	6.36
Effective heat transfer surface requirement, ft^2	41.7653×10^3	41.7653×10^3	26.827×10^4	39.414×10^4
* For U = 115 Btu/hr ft^2 °F, Effective heat transfer surface requirement, ft^2	35.8767×10^3	62.2348×10^3	23.0443×10^4	58.7313×10^4

TABLE 3-2

3000 MWT PNP U-TUBE He/He HEAT EXCHANGER

NUS HEAT TRANSFER SUMMARY

(950°C Reactor Outlet)

Assumed Parameters	Low Pressure Shell Side	High Pressure Tube Side
Reference	1	1
Mass flow, lb/hr	7.3412×10^4	7.246×10^4
Inlet temperature, °F	1742	500
Outlet temperature, °F	572	1652
Inlet pressure, psia	580.13	609.14
Outlet pressure, psia	573.00	605.34
Helium specific heat, BTU/lb°F	1.25	1.25
Effective heat transfer surface, ft^2	11.733×10^3	11.733×10^3
Calculated Thermal Performance Values (per loop)	Reference Counter Flow	
Outlet temperature, °F	1652 (tube side)	572 (shell side)
Effectiveness, E, %	0.9275	0.9587
Thermal Conductance, UA, BTU/hr °F	1.0721×10^6	1.8380×10^6
Overall Heat Transfer Coefficient, U, BTU/hr ft^2 °F	91.37*	156.65*
Heat Transfer, Q, BTU/hr	1.0434×10^8	1.0785×10^8
Log Mean Temperature Difference, LMTD, °F	97.32	58.68
Effective heat transfer surface requirement, ft^2	11.733×10^3	15.9826×10^3

* For U = 115 Btu/hr ft^2 °F, Effective heat transfer surface requirement, ft^2	9.322×10^3	15.9826×10^3

selection of the helical heat exchanger. In one column the shell side outlet temperature (572°F) is selected; in the other column the tube side outlet temperature is selected. The tube side outlet temperature selection yields a reasonable overall heat transfer coefficient, U, of ~100 BTU/hr ft^2 °F, for a helium-to-helium recuperating heat exchanger and an achievable, though high, effectiveness, E, of 0.93. The shell side outlet temperature selection yields an unrealistically high value of the effectiveness of 0.96 and overall heat transfer coefficient of ~170 BTU/hr ft^2 °F. Comparable NUS calculated thermal performance values are provided in Table 3-1 for an alternative cross-flow (unmixed fluids) helical heat exchanger for the identical input data sets. The calculated thermal conductance and effective heat transfer surface requirements for the alternative heat exchanger configuration (for the same Q) are shown to be significantly different as would be expected. For reference conditions, the effective heat transfer surface requirements appear consistent with assumed parameters.

The two sets of NUS calculated thermal performance values of Table 3-2 are similarly based on input data sets which differ only in outlet temperature selection as in Table 3-1. The tube side selection results in reasonable values of E and U, respectively. Comparable thermal performance data are calculated in Table 3-2 as in Table 3-1. For reference conditions, the effective heat transfer surface requirements appear consistent with the (assumed) parameters used.

The overall criterion of the helical unit is the design and development of modular, easily accessible heat exchangers, each arranged in separate cavities and having a thermal rating of 125 MWth. The active bundle consists of helically-configured pipes through which the secondary fluid flows. At the bottom ends, the helical tubes are mated radially into a hot central duct collector supported by the hot gas return duct. The helical units have a thermal rating greater than that of the U-tube units by a factor of 4, as a result of an equivalent increase in heat transfer area.

In the alternative arrangement, each circuit contains two adjoining 31.25 MWth heat exchangers, each containing eight U-tube modules. Each module, in turn, consists of one hot and two cold tube bundles for favorable space utilization and inspection and repair. The hot primary gas is divided into two heat exchanger distribution plena and flows through the inner coaxial tube into distribution rings. From the rings the gas is transported through an intermediate distributor to hot feed pipes, into the upper casing area of the hot branch, through the U-tube bundle, the bottom casing and exit cooled to 300°C, and into a central exit plenum. Design methods for compensation of differential thermal expansion of the hot and cold coaxial piping are not described.

It is believed that both heat exchangers would be designed to the equivalent of ASME Code, Section VIII, Division 2.

The hot ducting as described consists of internal insulation of ceramic material (carbon stone) and an inner connecting gas duct of graphite, with an appropriate outer bracing sheath. It appears from available literature that the conceptual design of the hot gas duct meets the suggested criteria above.

A design uncertainty is the bellows seal between adjacent hot gas duct work to accommodate thermal expansion. Sliding gaskets may be utilized in lieu of bellows. A disadvantage in comparison with bellows is the potential for fretting by sliding contact and the additional effort required before inspection. Design and development testing approaching full scale under service environmental conditions would be required for U.S. licensing purposes.

(3) Helium Stop Valves for Intermediate Loop of Steam Gasification Plant

Stop valves are provided to isolate each circuit in the event of heat exchanger fault conditions. For each of the intermediate circuits, two hot and cold stop valves are required. The valves are designed for 900°C and 40 bar differential pressure service conditions. Closing times are from five to thirty seconds. Conceptual valve design includes twin plate slide, ball valve, and coaxial valves. Industrial experience exists in the chemical and metallurgical industry with plate slides with similar temperatures and diameters but with lower differential pressures and seat leakage requirements.

Valves would have to be developed and demonstrated under appropriate environmental conditions for licensing purposes. The criteria for closing times would be dependent upon system transient analyses which remain to be performed.

(4) Steam Generator

The steam generators provide steam of high and intermediate pressure for the generation of electric power. In the steam gasification plant, high pressure superheated process steam is also provided to the gasifier and volatizer from the process steam end of the superheater. Heating surfaces of the preheater, evaporator, intermediate superheater and high pressure superheater, and the presuperheater for process steam, are combined into a single heat exchange unit for reasons of cost. The design has been described.[1] Table 3-3 summarizes the salient environmental conditions for the various regions of the steam generator and compares these with a comparable U.S. steam cycle HTGR unit. A check calculation by NUS of the thermal performance of the HTR-K and PNP steam generator is presented in Table 3-4. The overall heat transfer of the German design was confirmed to within two percent but agreement with the overall heat transfer coefficient and thermal conductance was less exact. The steam generator must be designed to the equivalent of the ASME Code, Section III, Division 2, and must meet the in-service inspection requirements of Section XI, Division 2. It cannot be ascertained from available design information whether the Code provisions could be achieved in the design.

(5) Helium Circulator

The helium circulator is a motor-driven, centrifugal, single-stage blower mounted vertically with the motor at the bottom for the PNP plant and at the top for the HTR-K plant. Somewhat different mass flow and pressure rise characteristics are provided for each as follows:

Parameter	PNP	HTR-K
Power Rating, MW	8.0	7.1
Mass Flow, lbs/hr, 10^6	1.175	1.746
Pressure Rise, psi	18.85	18.85
Control Method	Inlet throttling	Speed Control

Additional details are lacking.

The control system includes a variable frequency speed controller with each motor operated by an independent speed control system. The electro-mechanical components of the motors appear to be standard vertical motor items except for the end balls and cooling system which would have to be

TABLE 3-3

SUMMARY COMPARISON OF GAC MARK II-B AND PNP PR-3000 STEAM GENERATORS

	GAC MARK II-B	PNP PR-3000
1. Thermal Power/Unit, MWt	533	482
$Q \times 10^{-6}$, BTU/hr	1,820	1,646
2. He flow/unit, lb/sec	548.39	446.82
3. Inlet He Temperature, °F	1366	1292
4. Outlet He Temperature, °F	608	482
5. Inlet He Pressure, psi	772	566.95
6. Outlet He Pressure, psi	767	560.72
Pressure Drop, psi	5.6	6.2

	GAC SH2	GAC SH1	GAC Evap.	GAC Econ.	PNP SH2	PNP SH1	PNP Evap.	PNP Econ.
7. He Velocity, ft/sec	N/A	N/A	N/A	N/A	102.01	93.15	80.36	65.27
8. Heat Transfer Coefficient (HI) BTU/ft² hr°F	N/A	N/A	N/A	N/A	413.7	403.9	387.4	365.1
9. Inlet H_2O Pressure, psi	N/A	N/A	N/A	2850	2866.7	2991.4	3027.6	3230.6
10. Inlet H_2O Temperature, °F	N/A	N/A	N/A	400	914	689.2	690.8	356
11. FlowRate(H_2O), t/hr ($\times 10^{-3}$)	730.9	730.9	730.9	730.9	668.1	668.1	668.1	668.1
12. Outlet Steam/H_2O Pressure, psi	N/A	2501	N/A	N/A	2827.5	2886.7	2991.4	3027.6
13. Outlet Steam Temperature, °F	955	N/A	N/A	N/A	1004	914	689.2	690.8
14. H_2O Velocity, ft/sec	N/A	N/A	N/A	N/A	113.5	74.8	23.1	11.6
15. Heat Transfer Coefficient (H_2O), BTU/ft² hr°F	N/A	N/A	N/A	N/A	1339.9	1383.2	444.9	358.8
16. Overall Coefficient, BTU/ft² hr°F	N/A	N/A	N/A	N/A	223	232.1	292.5	274.6
17. Log Mean Temperature Difference, °F	N/A	N/A	N/A	N/A	333.7	343.8	135.5	129.2
18. Heat Transfer per Unit Surface, BTU/ft²	N/A	N/A	N/A	N/A	91,000	98,000	73,000	36,000
19. Heat Transfer Surface, ft² *	N/A	N/A (total = 33,527 ft²)	N/A	N/A	1495.4	6861.7	7067.2	23,122.2
20. Number of Tubes	360	360	360	360	275	275	275	275
21. Length of Tubes	N/A	N/A	N/A	N/A	19.61	90.0	92.73	303.33
22. Thermal Capacity, MW	N/A	N/A	N/A	N/A	29.46	154.45	110.82	196.27
23. Tube Diameter/Wall Thickness, Inches	1.00/.163	1.25/.191	.875/.110 1.00/.127	.875/.110	1.06/.142	1.06/.142	1.06/.142	1.06/.142
24. Tube EES Bundle Height, ft**	N/A	N/A (total = 32 ft-5 in.)	N/A	N/A	2.76	7.84	8.04	23.22

* Total Mk II-B EES main bundle effective heat transfer surface area is 33,527 ft² (excluding reheater) versus 38,546.5 ft² for PR-3000. (Mark II-B total heat transfer area with reheater is 38,546.5 ft²)

** Total Mk II-B tube bundle height (including reheater bundle is 32.42 ft versus 41.86 for PR-3000.

*** Average Mk II-B heat flux is 47,050 BTU/hr ft² versus 42,700 BTU/hr ft² for PR-3000.

NOTE: Data indicated N/A (not available) are GAC private data.

TABLE 3-4

3000 MWT PNP STEAM GENERATOR
NUS HEAT TRANSFER SUMMARY
(950° Reactor Outlet)

Assumed Parameters	Low Pressure Shell Side	High Pressure Tube Side
Reference	1	1
Mass flow, lb/hr	1.4965×10^9	1.6086×10^6
Inlet temperature, °F	356	1292
Outlet temperature, °F	1004	482
Inlet pressure, psia	3230.6	556.95
Outlet pressure, psia	2827.5	560.72
Coolant specific heat, BTU/lb °F	*	1.25
Effective heat transfer surface, ft^2	38.5465×10^3	38.5465×10^3

Calculated Thermal Performance Values (per loop)	Reference Counterflow
Effectiveness, E	0.8654
Thermal Conductance, UA, BTU/hr °F	4.0327×10^6
Overall Heat Transfer, Coefficient, U, BTU/hr ft^2 °F	174.41
Heat Transfer, Q, BTU/hr	1.6286×10^9
Log Mean Temperature Difference, LMTD, °F	403.88

* For Economizer and Evaporator C_p ~10.0; for SH2, c_p ~0.69; for SH1, c_p ~0.78.

especially designed for this application. Oil-lubricated ball bearings are used, and oil must be replenished during reactor operation so that radiation damage of the oil will not impair system performance. Oil vapor would be prevented from entering the coolant loop by means of a multistage labyrinth seal arrangement. A heat exchanger, presumably water-cooled, must be provided to maintain acceptable oil temperatures and ambient motor temperatures. The circulators must be designed to operate at all pressure levels from full helium inventory down to shutdown status and over a wide range of operating conditions. A qualification program would be required for U.S. licensing.

3.2.2 Overpressurization Protection

Generally, gas-cooled reactor plants provide overpressurization protection by means of the following systems: steam generator (or precooler and recooler for the HHT-K) isolation and dump system, main loop shutdown system, containment pressure protection system, core auxiliary heat exchanger isolation system, control rod withdrawal interlocks, and the prestressed concrete reactor vessel relief system. It has not been possible to confirm that all of these systems are incorporated in the PNP, HTR-K, and HHT-K plant designs, but there is no reason to expect that they, or an equivalent system, could not be incorporated. Steam generator isolation and dump systems are usually monitored by three or more moisture sensing instrument channels in each of the six main coolant loops. Signals are arranged in a two-out-of-three logic matrix, actuating one of two separate and independent trains of equipment, in turn actuating a dump valve and initiating a main loop shutdown. Ingress of water and steam into the primary coolant system must be prevented from exceeding a specified quantity (1,000 to 1,200 kg H_2O) based upon graphite corrosion criteria.

Main loop shutdown is usually initiated by any of a number of temperature exceeding specified limits, including main steam outlet temperature, main circulator helium outlet temperature, and reheat steam radiation indication. Core auxiliary heat exchanger systems are monitored by comparable sensors and instrumentation systems. The PCRV pressure relief system for the PNP and HTR-K plant designs appear to consist of two independent trains which provide overpressurization relief when specified maximum working pressures for either plant design are exceeded. The control rod withdrawal interlocks constitute conventional reactor protection systems. PNP, HTR-K, and HHT-K containment isolation systems are not defined in available reference material but could be designed to applicable U.S. regulatory criteria. Additional discussion is provided in Sections 3.4.3 and 3.4.4 on the HHT-K PCRV pressure relief system.

3.2.3 General Material Considerations

3.2.3.1 Metallic Materials

The design lifetime for the PNP, HTR-K, and HHT-K is approximately 300,000 hours. Primary system coolant boundary components must be designed for extended lifetimes to ensure reliability and to minimize maintenance and repair requirements. Selection of qualified structural alloy materials for these components is an essential requirement to achieve these goals. The long-term behavior of candidate alloys under representative reactor helium environments is being characterized and the environmental effects on material properties are being established quantitatively. Degradation mechanisms such as oxidation and carburization resulting from coolant impurity interactions are being established by correlating the occurrence of these effects with helium impurity concentrations and alloy composition.

The candidate alloys include high and low alloy steels, austentic stainless steels, high nickel or nickel-base super alloys, and possibly more advanced materials such as molybdenum or dispersion-strengthened alloys. For components operating above about 1652°F, creep or stress rupture properties are important. Other important properties may include short-term tensile or compression behavior, random high and low cycle (50 to 1,000 Hz) fatigue resistance, fracture toughness, thermal aging resistance, and helium impurity corrosion rates depending on the component operating environment.

Materials screening programs are in progress in the United States and Europe to select and evaluate candidate structural alloys for primary system components and ducting. This experimental screening and metallurgical evaluation program is backed by parallel, complementary screening tests in progress at the Central Institute for Industrial Research (CIIR) in Oslo, Norway, as the European High-Temperature Materials Program. The European program is under combined funding and direction by the U.S. Department of

Energy, the KFA-Jülich, and the European Economic Community, Petten, Holland. The U.S. and European programs have been described in Volume II of Reference 1 and elsewhere.[13, 14]

While most U.S. and European test data are for periods up to about 10,000 hr, data are required to at least 30,000 hr for reliable extrapolation to the design life of about 300,000 hr. The lack of these data at present constitutes the principal open issue regarding the projected performance of the candidate alloys under service conditions. The European materials research to date suggests that materials possessing the requisite properties up to 1562°F and test times of 30,000 hours do exist. However, for higher gas temperature applications and service times such as the PNP and HHT-K, it will be necessary to adopt special measures, such as intensified cooling for the turbine blading for the HHT-K. For the hot helium ducting, the use of ceramic insulation materials appears to be necessary.[15]

The metallic materials of primary system components, including heat exchangers which separate two working fluids, must be designed to the equivalent requirements of the ASME Code, Section III, Subsection NG. If materials not listed in Section III are employed, the design stress limits must be derived in a manner equivalent to those for Class 1 components. When creep is a factor, inelastic stress analysis calculational techniques equivalent to the ASME Code Case 1592 are required.

The European investigations have concentrated on eight alloys, of which two wrought alloys (Hastelloy X and Inconel 617) are included in the U.S. screening program. The eight include Hastelloy S, Hastelloy X, Inconel-586, Inconel-617, Incoloy 800H, Incoloy 802, G-24/24 Nb, and G-25/35 Nb. A number of these are in competition to remain as viable candidates, and the number of alloys being studied will be reduced as soon as sufficient material data for each are available.

For the HHT-K turbomachine, candidate alloys include Inconel 713 (low carbon) and a molybdenum nickel base alloy, TZM. The Inconel 713 LC is presently considered by the Europeans as the most suitable blading material for a helium turbine. The coefficients of thermal expansion at turbine inlet temperatures of the nickel-base and molybdenum base alloys (TZM) are quite different. The use of dual materials for turbine blading may present design problems from the point of view of tip clearance control. Turbine inlet temperatures would range from 1,562° to 1,742°F, whereas turbine outlet temperatures would be approximately 1,022°F. With TZM blading, the life of the turbomachine would not be dictated by the first-stage blade centrifugal stresses due to the very high creep rupture strength of TZM. At the reduced turbine exit temperatures, however, the ultimate tensile strength of TZM is less than the nickel-base alloys, making its application in the last few stages questionable in light of thermal transients associated with turbine outlet temperatures on loop shutdowns and loss of electric load. Further evaluation is in progress as to an appropriate choice for cooled and non-cooled turbine vanes and blading for the HHT-K.

3.2.3.2 Ceramic Materials

There are four basic differing uses of graphite in gas-cooled reactor design

- hexagonal replaceable reflector elements
- large semi-permanent reflector blocks
- core support blocks
- core support posts.

Design criteria have been described in Section 2.2. Generally, nuclear-grade graphites must have high strength, exhibit minimal dimensional change with irradiation, have low thermal expansivity, low elastic modulus, high thermal conductivity, and low impurity content. They should be readily machinable, available from multiple sources, commercially reproducible by grade, and relatively inexpensive. As a non-metal, graphite is a brittle material though less brittle than most ceramics. While stronger than most metals at HTGR temperatures, experience in its structural use, though encouraging, is limited.

The use of graphites for core safety-grade structural material represents a technical advancement beyond U.S. licensed reactor applications to date, including Ft. St. Vrain. The large body of available information on reactor-grade graphites is mostly on needle coke (an isotropic) and European Gilso-graphites (isotropic) at either lower temperatures or under less demanding conditions than proposed for the near-isotropic grades. In recognition of this situation, research programs are underway in many laboratories in Europe and the United States to acquire the necessary property data. Since fast neutron fluence is an important degradation phenomenon which must be accounted for in design, the data acquisition must include carefully characterized graphite specimen irradiations which are time consuming and expensive. A summary listed of HTGR graphites of different commercial grades proposed for the PNP, HTR-K, and HHT-K has been reported.[1]

3.2.4 Primary Coolant Pressure Boundary Leakage and Detection Systems

Primary coolant pressure boundary leakage and detection systems are not described in available literature. Conventional detection systems for commercial HTGR's include reduced system pressure (for systems other than HHT-K), audible sound levels, containment radiation monitors, and reheat steam radiation monitors. Internal primary coolant system pressure and moisture monitor detectors would signal failure of the steam-raising units in the HTR-K and PNP designs. Appropriate leakage and detection systems would have to be engineered and specified for U.S. licensing purposes.

3.2.5 In-Service Inspection Program

The primary coolant pressure boundary should be capable of in-service inspection to rules equivalent to the ASME Section XI, Division 2. (In the United States, the ASME Section XI, Division 2 Committee, which has included NRC and General Atomic Company participation, is presently inactive.) A number of issues remain to be resolved, including the type, number, and frequency of certain primary coolant pressure boundary in-service inspections. Section XI, Division 1, of the ASME Code for LWR's requires periodic volumetric surface, and visual, examination of all pressure-retaining wells, including the base material for at least one wall thickness beyond the edge of the weld. Similar periodic examinations are required of bolting and other critical components. It would be expected that comparable criteria would eventually be specified by the Division 2 Committee for HTGR's.

A comparison of the operating environments of LWR's and HTGR's shows that HTGR penetrations and closures would be operated under less severe conditions than those for pressurized light water reactors.

Parameter	HTGR	PWR
Operating pressure, psig	785	2,250
Operating temperature, °F	1,562 - 1,742	600
Total integrated fast neutron flux, n/cm^2 (E>1 MeV)	10^{17}	2.5×10^{19}
NDT shift due to irradiation, °F	0	300
Overpressure protection, tolerance and backpressure limit, percent	2	10
Completed vessel proof testing, pneumatic	1.15 x DP*	1.2 x DP*
Leakage	Continuous leakage monitoring during reactor operation	System leak test at each refueling outage

*DP - Design Pressure

The materials utilized for the PWR pressure vessels and HTGR liner and penetrations and closures are quite comparable. The HTGR features thinner plate and forgings, and smaller diameter bolts. Ultrasonic testing (100%) would be required for plate and forging materials, and tensile and fracture toughness qualification testing for base metal, weld, and heat-affected zone would be required. In fabrication practice and quality assurance, the methods used for the two reactor vessel types are believed similar. The exception is that pneumatic proof and leak testing of the HTGR would be performed at 1.15 times design pressure compared to 1.2 for PWR vessels.

The in-service inspection requirements of ASME Code, Section XI, Division 2, or equivalent, would extend to weldments of the pressure boundary and heat exchangers and steam generators of the HTR-K and PNP, and the precooler and recooler of the HHT-K. The core auxiliary heat exchangers would also be in-service inspected. Therefore, tube-to-tube sheet welds would have to be accessible for periodic inspection. The requirement for in-service inspection of transition welds in steam generator tubing has not been established. Accessibility to steam generator and heat exchanger primary system boundary weldments, structural supports, and central ducts where incorporated is also an important consideration in detailed component design.

3.3 THERMAL AND FLUID MECHANICAL SYSTEM DESIGN

3.3.1 Analytical Methods and Data Summary

A cogent summary of analytical methods and data for the thermal and fluid mechanical system design is not available. A summary comparison of the PNP, HTR-K and HHT-K principal mensuration data have been reported.(1) Table 2-1 presents certain of these data for comparison purposes with U.S. licensed HTGR systems. An overall comprehensive data summary for the PNP, HTR-K, and HHT-K systems is lacking.

Analytical methods used for design purposes with supporting empirical correlations and data are typically provided to U.S. licensing authorities as separate Licensing Topical Reports which support the detailed design in the standardized Safety Analysis Report. Typical licensing topical reports related to the nuclear, thermal, and fluid mechanical design provided to the NRC for the General Atomic Company steam cycle HTGR are listed in Table 3-5. Comparable topical reports in support of the PR-3000 core design and the various PNP, HTR-K, HHT-K plant applications would be required under U.S. licensing procedures.

Structures, systems, and components important to safety that must withstand the effects of a Safe Shutdown Earthquake and remain functional are classified as seismic Category I items. These plant features are those necessary to assure the integrity of the primary coolant system, the capability to shutdown the reactor and maintain it in a safe shutdown condition, or the capability to prevent or ameliorate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100. The NRC Regulations are set forth in General Design Criterion 2, and in Regulatory Guide 1.29, "Seismic Design Classification" (as applicable to HTGR nuclear plants) and industry standards. No designation of the equivalent of Category I structures for the PNP, HTR-K, and HHT-K designs is presented in available literature.

Moreover, the basis for U.S. licensing review of pressure-retaining components, such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves in fluid systems important to safety, would be compliance to design criteria and General Design Criterion 1, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, and to Regulatory

TABLE 3-5

SUMMARY OF GAC LICENSING TOPICAL REPORTS RELATED TO NUCLEAR, THERMAL AND FLUID MECHANICAL DESIGN BASES AND PERFORMANCE ASSESSMENT

Number	Topic	Status
LTR-1	Core Cooling Capability	Submitted to NRC for information
LTR-2	Nuclear Design Methods and Data	Submitted to NRC for information
LTR-3	Thermal Conductivity of Nuclear Graphite	Submitted to NRC for information
LTR-4	Afterheat Calculation	Approved by NRC
LTR-7	OXIDE-3 (Steam or Air Ingress)	Requires NRC action
LTR-9	Fuel Rod Thermal Conductivity	Approved by NRC
LTR-10	SORS (Transient Fission Product Release)	Requires NRC action
LTR-12	Stress Analysis Methods in Core Design	Submitted to NRC for information
LTR-13	CORCON (Core Heatup Transient)	Requires NRC action
LTR-15	Fuel Particle Behavior	Submitted to NRC for information
LTR-17	Core Thermal Design Methods	Requires NRC action
LTR-18	Core Power Distributions	Submitted to NRC for information
LTR-21	TAP Code	Submitted to NRC for information
LTR-	RECA Code	To be submitted to NRC in 1977
LTR-	Anticipated Transients Without Scram	To be submitted to NRC in 1978-79
LTR-	HTGR Materials (Metallic)	To be submitted to NRC in 1978-79
LTR-	Core Cavity Flow and Pressure Distributions	To be submitted to NRC in 1978-79

Guide 1.26 (as applicable to HTGR nuclear plants), and industry standards. Comparable delineation of FRG design criteria for pressure retaining components would have to be developed.

The General Design Criteria require that systems and components important to safety be protected from the effects of missiles, generated both from within the containment and external to the containment. In the case of the HHT-K, the criterion would extend to the effects of missiles generated by the turbomachinery within the primary coolant system boundary. The missiles to be considered, other than structural parts of the turbomachine for the HHT-K, would include various internally-generated missiles from pressurized components for the PNP and HTR-K. The criterion specifies that no significant missile could arise from the primary coolant system due to the stored energy contained within the system. While no comparable listing has been provided in available literature, it would appear that components with the potential for becoming missiles might include control rod drive mechanism assemblies, control rods, valve stems, valve bonnets, and other presusre retaining bolts, nuts, and casings. The requirements of protection of essential structures and vital equipment would have to be in accordance with the General Design Criteria 2 and 4. It is expected that acceptable missile protection could be achieved in the PNP, HTR-K, and HHT-K designs, although the latter would require the incorporation of turbomachinery disc catchers which could retain a fragmented turbine rotor at specified overspeed conditions (150%).

General Design Criterion 4 requires that structures, systems, and components important to safety be appropriately protected against the dynamic effects from postulated ruptures of high-fluid energy piping. The PNP, HTR-K, and HHT-K designs should be reviewed from the point of view of high energy fluid piping break locations, pipe break orientations, and break flow areas consistent with the criteria and level of protection of NRC Regulatory Guide 1.46 for piping inside containment.

The input seismic design response spectra to be applied in the design of specified seismic Category I structures, systems, and components should comply with Regulatory Guide 1.6, "Design Response Spectra for Nuclear Power Plants," and Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants." It appears that the seismic system and subsystem analysis procedures for the PNP, HTR-K, and HHT-K plants could be developed to provide an acceptable basis for the seismic design, with one possible exception: the fuel bed array and graphite structural supporting elements. It is believed that a pebble bed reactor core seismic program would be required for U.S. licensing purposes. Objectives of this program might include

- Basic response characteristics of the core
- Impact loading between adjacent fuel spheres and reflector elements
- Shear forces on keyed, interlocking graphite reflector and core support blocks which connect adjacent elements
- Displacements of various components, vertically, horizontally and rotationally (rocking)
- Dynamic loads acting on the permanent side reflector and core support structures
- Impact strength of the core components in terms of both failure under a single-load application and fatigue failure under repeated loads.

To provide these data, an extensive reactor seismic experimental and analytical program would be required. Difficulties with scale model testing with respect to similitude factors has been experienced in the United States by the General Atomic Company. Confirmatory seismic research at the Los Alamos Scientific Laboratory and the Brookhaven National Laboratory for HTGR systems is in progress. Los Alamos has advocated a highly flexible test facility that would aim to provide sufficient similitude that the distortion problems with earlier scale model testing performed by General Atomic would be overcome. Brookhaven is developing an alternate approach of verified computer modeling in which a number of detailed and related core seismic codes are checked against simple experiments to verify assumptions and equations in programming. These codes are then applied to full-sized structures. It is not yet clear whether scale model testing or verified computer models will offer the more successful bases for establishing the seismic integrity of the reactor core and supporting structural elements. There is no discussion, however, of this requirement for the PR-3000 core in available literature.

It is expected that the initial PR-3000 type plant would be considered a prototype design and would be instrumented for vibration analysis and test programs consistent with the requirements of a prototype reactor and in compliance with the NRC Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Pre-operations and Initial Startup Testing." With respect to vibration analysis, the NRC has requested of General Atomic a description of methods used to extrapolate fatigue data to 10^{12} cycles; conventional fatigue test data extend only to 10^{6} cycles. A comparable requirement, presumably, would be imposed on the PR-3000 core. Supporting documentation would have to be provided to demonstrate the structural adequacy of the reactor internals under the loadings that will result from flow-induced vibration.

Finally, the effects and consequences on reactor internals and other components within the primary coolant system boundary of the design basis depressurization accident would have to be developed. Flow velocities under rapid depressurization conditions for the PNP, HTR-K, and HHT-K appear to be based on a postulated depressurization break area of 1000 square centimeters. The resulting velocities within the primary coolant loop are comparable with normal operation values. The major effect of this accident has been determined by NRC to be a differential pressure loading on core structural components rather than dynamic effects from blowdown transients. The actual differential loadings sustained, particularly for the HHT-K structural components, is an issue in point. Appropriate analysis for all applications would have to be provided.

All Category I safety-related ASME Code Class 2 and 3 systems, components, and supports outside of the primary coolant system boundary must be designed to sustain normal loads, anticipated transients, dynamic events, and the operating basis earthquake, and the safe shutdown earthquake within design limits which are consistent with those in NRC Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category 1 Fluid System Components." When valves and pumps are tested for faulted conditions other than the design basis depressurization accident, the faulted condition loading combination should include the safe shutdown earthquake as well as the design basis depressurization accident. Similarly, the criteria used in developing the design and mounting of safety and relief valves of ASME Class 2 and 3 should provide adequate assurance that under discharging conditions the resulting stresses would not exceed the allowable design stress and strain limits for the materials of construction. Design and installation criteria of overpressure relief devices must conform with Regulatory Guide 1.67, "Installation of Overpressure Protective Devices."

3.3.2 Operating Restrictions on Circulators and CAHEs

Operating requirements on core auxiliary circulators or main circulators must include the power requirements to achieve some volumetric flow through the system under pressure equilibration with containment as a result of a complete depressurization event. The coolant must be assumed to be a mixture of air and helium after the depressurization, and bypass flow through inactive or blocked loop channels must be considered. Thus, the circulator has to operate over a wide range of stable flow conditions requiring essentially continuous speed adjustment. The rating of the core auxiliary heat exchanger will be based not only on the decay heat considerations (approximately 60 MWth, corresponding to 2 percent of rated reactor power), but also the potential for water-steam ingress under steam generator fault conditions and the requirement for rapid cooling of the core in order to limit the corrosion of the graphite structures. The cooling rates used in the thermal sizing of the core auxiliary heat exchangers are not specified in available literature. The afterheat removal loops, which are normally inactive during reactor operations, are thermally lightly loaded by bypass cold gas flow. Upon CACS actuation, the afterheat removal system must assume its full heat removal capability at a rate consistent with minimizing thermal shocks to the heat exchanger. Auxiliary circulator speed profiles for this purpose have not been specified. Operational restrictions on main primary system helium circulators are also not specified in available literature.

3.3.3 Temperature-Power Operating Regime

The normal production mode of the PNP plant is as follows. The gasification plant and electric production plant operate without reduction of bypass steam flow. Normal operation with reactor bypass for the steam electric plant would require that electrical energy for the gasification plant be taken from the power grid. For operation with reduced reactor outlet temperatures, operation of the electric power generation plant would continue, but the gasification plant would be in an inactive mode. Normal operation of the plant with all six primary circuit loops in operation would range from 75 to 100 percent of capacity. Continuous operation with five (N-1) primary loops is also possible.

Electric power generation is possible up to 75 percent peak xenon reactivity without a time limit. Normal startup of the reactor and the gas plant from cold iron conditions would require up to two weeks. If there were failures in gasification plant feed rates, the gasification plant could be switched to an inactive or dormant mode of operation. (For economic reasons this dormant mode should not last longer than approximately a week.) Startup of the gasification plant from the dormant operation would then require approximately 12 hours. Following a fast shutdown of the PR-3000 core from 100 percent power, it would be necessary to restart within one hour due to xenon reactivity effects. If this were not possible, a peak xenon period of about 24 hours would be sustained before continued operation were possible.

For the HTR-K, the temperature-power operating regime would be expected to be comparable with the steam cycle HTGR system of General Atomic. For the HHT-K, the temperature-power operating regime ranges from 30 to 100 percent power. For the HTR-K, the partial load range is 25 to 100 percent power, and for the PNP, from 75 to 100 percent power on the product side. The load requirements are as follows:

Load	HTR-K	HHT-K
Step Changes	$\pm$ 10%	$\pm$ 10%
Load Transients, percent per minute	$\pm$ 5%	$\pm$ 10%

For the HHT-K, the startup system should have the capability to operate for many hours to provide plant thermal conditioning during startup. Development of the low-speed control and external powering method are significant elements of the HHT-K technology, strongly affecting operational constraints, reliability, and cost. However, the startup system has not been reported. External powering requirements could be reduced by reducing helium inventory (system pressure) during turbomachine startup. At present, reduction to 40 to 50 percent of full load inventory is contemplated for the HHT-K. An overall temperature-power operating profile for the PNP, HTR-K, and HHT-K has not been presented.

3.3.4 Load Following Characteristics

The load following characteristics of the PNP system have not been described. For the HHT-K, load following is possible from 25 to 100 percent of rated power. The normal rated load change is 5 percent per minute, and the maximum step load change is 10 percent.

3.3.5 Transient Effects

For the PNP application, the principal transient results from interruption of the methane feed stock and water-steam to the reformer. An evaluation of this transient has not been provided. For the GT-HTGR*, the plant transients reported as yielding the major pressures and temperatures include

- Plant shutdown following the loss of electric load on all three loops
- Loss of electric load with an overspeed trip (due to control failure) of the turbomachine in one loop with the other two loops operating at constant speed and supplying electrical load
- Plant loss of all electrical load with subsequent trip of the turbomachine due to complete control failure
- Plant loss of electric load with overspeed trip of the turbomachine due to control failure
- Plant shutdown on scram from core over-temperature conditions resulting from rupture in the water lines to the precoolers and recoolers on both loops.

Typical transient parameters of interest include

- Peak transient cavity pressures
- Maximum pressurization rates in low pressure regions and maximum depressurization rates in high pressure regions
- Peak coolant and metal (turbine and heat exchanger) temperatures

*General Atomic High Temperature Gas Reactor-Direct Cycle Gas Turbine

- Loop mass flows and flow reversals
- Turbomachinery overspeed conditions.

The transient results indicate that peak depressurization phenomena of approximately -100 psia/s (plant shutdown due to loss of load in all loops), and peak helium temperature transient phenomena in the turbine outlet of approximately 100°F per second are representative transients for which design accommodation must be made. The core "spike" depressurization in the initial seconds of bypass valve opening may have the undesirable effect of temporary diversion of normal coolant flow to the active core region. The thermal transient results from the collapsing of the turbine pressure ratio, with turbine outlet helium temperature increased from 1160°F to over 1400°F within five seconds. Damage to the turbine may occur from this severe thermal transient. Transient effects for the HTR-K have not been reported.

3.3.6 Thermal and Fluid Mechanical Characteristics

The summary thermal and fluid mechanical characteristics for the PNP, HTR-K, and HHT-K reactor coolant systems are presented in Table 2-1 and discussed in Section 3.2.1.

3.4 PRESTRESSED CONCRETE REACTOR VESSEL

3.4.1 Summary Description

The principal features of the prestressed concrete reactor vessel for the PNP, HTR-K, and HHT-K have been described;[(1)] the salient features of these PCRs are presented in Table 3-6 and compared with licensed U.S. PCRV systems. The major code used for materials, design, fabrication, construction, and testing of PCRV structures is the ASME Section III Code, Division 2, Subsection CB. This Code, developed jointly by ASME and ACI with the active participation of NRC Staff, is considered acceptable by U.S. regulatory authorities. The principal differences in the PCRV designs for the PNP, HTR-K, and HHT-K include configuration, working pressures of the helium coolant, the number of cavities, and variation in pressure within the cavities. The working pressure and variation in pressure (HHT-K) effect both the external PCRV dimensions and method of prestress. Finally, for the HHT-K, the method of pressure relief and internal pressure equilibration is quite different from the HTR-K and PNP PCRV pressure relief system.

The PCRV liner for the PNP and HTR-K designs is provided with a layer of thermal insulation on its inner surface, and a cooling system on its outer surface. These two systems limit the temperature at the inside surface of the concrete to acceptable values. The thermal barrier prevents degradation of long-term concrete strength and reduces the severity of the thermal gradiant across the vessel wall. Liner design and performance requirements have been described,[1] and Table 3-6 summarizes the temperature performance of the thermal barrier. The PCRV liner for the HHT-K is of a new and unique design — a steel liner without thermal insulation — the warm liner concept. The warm liner is thus inspectable and repairable, a very desirable feature. The liner is backed by 20 to 30 centimeters of porous concrete with metallic cooling coils imbedded at this distance in the concrete and not on the liner wall.

3.4.2 Structural Materials

The major materials utilized in the construction of the PCRV are the concrete, the bonded reinforcing steel, steel imbedments, and hardware for the linear and circumferential prestressing systems. These materials can be designed to conform to the objectives of Subsection CB of the ASME Code, Section III, Division 2. Moreover, the construction, quality control, testing, and monitoring programs can be designed in accordance with the objectives of the Code. An in-service inspection program meeting the objectives of Section XI, Division 2, of the ASME Code would be required. The NRC has evaluated the effects of irradiation for U.S. PCRV materials and has concluded that no significant loss in strength of concrete would be expected. Similarly, the reinforcing steel and prestressing systems are protected by concrete, and no significant irradiation effects would be expected to occur. The effects of fatigue were also considered and found not to be a problem. Additional discussion of structural materials for the liner is presented in Sections 3.2.1 and 3.2.3.

3.4.3 Design Bases

The techniques that have been used in the analysis of the PCRV designs for the PNP, HTR-K, and HHT-K have employed computer programs, the validity of which have been confirmed by normalization of calculational data with that obtained from classical solutions and experimental model tests. Elastic

TABLE 3-6

SUMMARY CHARACTERISTICS OF THE PRESTRESSED CONCRETE
PRESSURE DESIGNS FOR THE FORT ST. VRAIN,
GAC STEAM CYCLE AND GAS TURBINE HTGR, AND EUROPEAN HHT-K, HTR-K, AND PNP

	Port St. Vrain Station	GAC SC-HTGR	HTR-K	GAC GT-HTGR	HHT-K	PNP
Power (MWe/MWt)	330/870	1320/3600	1120/3000	1200/3000*	1240/3000	300/3000
PCRV Type	Single cavity	Asymmetric multicavity	Symmetric multicavity	Symmetric multicavity	Asymmetric multicavity	Symmetric multicavity
Outside diameter (ft)	49	111.5	120.7	118.0	157.5	144.4
Height (ft)	106	89.0	101.7	110.5	135.8	101.7
Prestressing Systems	Longitudinal, circumferential and crosshead tendons	Longitudinal tendons and circumferential wire-wrap	Longitudinal tendons and circumferential wire-wrap	Vertical and diametral tendons and circumferential wire-wrap	Vertical and diametral tendons and circumferential wire-wrap	Longitudinal tendons and circumferential wire-wrap
Core Cavity						
Diameter (ft)	31	43.5		44.5	49.2	53.8
Height (ft)	75	47.3		47.3	49.2	55.8
Steam Generator						
Cavity (qty)	12	6	6	6	4	6
Diameter (ft)	3.3	15.3	15.7	19.5**	23.0**	15.7
Height (ft)	15.5	76.3	Not available	67.0	108.3	Not available
Auxiliary Loop						
Cavity (qty)	4	3	4	3	4	4
Diameter (ft)	3.3	8.9	9	8.9	9	9.8
Height (ft)	15.5	89	Not available	Not available	Not available	Not available
Turbomachine Cavity						
(qty)	--	--	--	3	1	5
Diameter (ft)	--	--	--	13.5	20	14.8
Length (ft)	--	--		52.0	106	72.2
Max. Cavity Pressure (psig)	845	835	870	1150	1045	580
Normal Liner Temperature						
Average (°F)	130	150	150	150	230	150
Hot Spot (°F)	200	250	250	250	302	250
Refueling Penetrations (qty)	37	109	****	94	****	****
Primary Coolant Leakage						
Design (%/yr)	1.0	1.0	Not available	Not available	Not available	Not available
Max. Allowable (%/yr)	14.4	3.6	Not available	Not available	Not available	Not available

*3-Loop Plant Configuration
**Maximum heat exchanger cavity diameter
***Maximun heat exchanger cavity diameter
****43 top fuel entrance chutes; 6 bottom fuel exit chutes
*****Reformer cavity for PNP

analyses have been performed to establish that stresses in the concrete, in the prestressing system, and in the reinforcement are within allowable limits under various postulated load combinations.[1] In addition, a visco-elastic or nonlinear analysis has been performed to establish that the performance of the PCRV designs and their response to time-dependent, long-term loadings meet design requirements. The ultimate load capacity of each PCRV design is to be established by a combination of further analysis and experimental model testing.

The linear prestressing system is essentially identical to that previously licensed for the Fort St. Vrain reactor in the United States. The circumferential prestressing system is based on wire winding techniques that have been successfully used in various industrial applications. The metallic materials and hardware to be used in PCRVs can be protected against corrosion.

The analytical techniques which have been utilized in the design of the thermal barriers are conventional in nature and involve both manual and computer methods. The tubes of the cooling system of the PNP and HTR-K are considered as part of the liner plate and analyzed in this manner. Both the thermal barrier and the cooling system are analyzed for normal and abnormal thermal and stress loads as required by the ASME Code.

3.4.4 Loading Characteristics

Analysis and model testing on PCRV structures performed to date have established that the PCRV as a whole deforms gradually and resists more than twice the specified maximum cavity pressure (MCP), while the top head can resist approximately three times the MCP with one MCP acting on the barrel portion of the structure.

The PCRV support, which may consist of radial or cylindrical walls is typically anchored by the vertical prestressing tendons to the containment base slab. The system can be designed so that all applicable load combinations specified in the ASME Code are achieved. In the PCRV liner, the tubes of the cooling system are not considered as strength contributors to the liner plate, either for the PNP and HTR-K or the HHT-K warm liner design.

As a result of many detailed stress analyses of multicavity PCRVs, the minimum PCRV outside diameter can be approximated from the cavity pressures and diameters by the following expression:[17]

$$D = \frac{(MCP_c D_c + 2MCP_{HX} D_{HX\ max}) F}{f\eta} + D_c + 2D_{HX\ max} + 2t$$

where,

D_c = Core cavity diameter, 53.81 (16.4m)

D_{HX} = Heat exchanger cavity diameter, 15.75 ft (4.8m)

MCP_{HX} = Maximum heat exchanger cavity pressure, 1044.24 psig (72 bar)

F = Safety factor, 1.1

f = Allowable compressive strength in concrete, 2,275 psig

η = Creep relaxation factor, 0.8

t = Precast panel thickness, 0

Similarly, the minimum PCRV height can be approximated by the sum of the core cavity height, H_c, plus the top and bottom head thicknesses. An expression developed to establish the top and bottom head thickness is:

$$H_t = \frac{MCP_c D_c}{4s}$$

where,

s = allowable shear stress $5\sqrt{f_{cua}}$ (f_{cua} = maximum concrete compressive stress, 6,500 psig)

D_c = 53.81 ft (16.4 m)

MCP_c = 1044.24 psig (72 bar)

Values for the PNP PCRV design are presented above as a basis for the following NUS check calculation.

The overall diameter for the PNP PCRV vessel from the above expression is 139.14 feet.

H_t is 34.85 ft for the PNP PCRV values, and overall PCRV height is

$$H = H_c + 2H_t = 55.77 + 69.69 = 125.46 \text{ feet.}$$

The values for overall PCRV diameter and height of 139.14 and 125.46 feet, respectively, compare with reported values of 141.08 feet and 101.71 feet. The computed height is approximately 19 percent less than that of the actual PNP PCRV height, an acceptable variation.

Comparable calculations have been made for the HHT-K PCRV with the following results. The computed overall PCRV diameter was 142.72 feet and the overall height 113.45 feet. These values compare with reported HHT-K PCRV diameter and height of 141.08 feet and 126.31 feet, respectively. The difference in height is explained by the fact that the HHT-K turbomachine cavitiy lies directly under the central core cavity, thus necessitating a bottom head which is thicker by approximately the cavity diameter (13.5 feet) than that for which the approximation is based. The general calculational agreement is indicative of common underlying PCRV design methodology of U.S. and European HTGR design organizations.

3.4.5 Summary

It is apparent that the analysis, design, and construction of the PCRV thermal barrier and liner cooling systems, and for anticipated loadings of the PCRV structure during its service life, could be in conformity with established criteria, codes, standards, and specifications acceptable to U.S. licensing authorities. Therefore, these components can be designed, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. In addition, criteria used in the analysis, design, and construction of the thermal barrier and cooling systems for the PNP and HTR-K applications appear to account for anticipated loadings and conditions that may be imposed, and it is believed that they would be acceptable to NRC. For the HHT-K PCRV structure and warm liner, considerable additional development and analysis remains to be performed for this design to meet applicable U.S. standards. There is no reason to believe that the concept would not meet applicable standards, codes, and specifications upon successful completion of planned development programs.

3.5 COMPONENT AND SUBSYSTEM DESIGN

3.5.1 Primary Coolant Circulators

The salient characteristics of the primary coolant circulators are summarized in Table 2-1. The primary circulator for the PNP and HTR-K is a single-stage centrifugal compressor with a series-connected, bladed diffuser. Details of the blower, the drive motor with cooler, and the shutoff control valve have been described.[1] The largest variable speed, motor driven circulator that has been designed for helium service and tested under simulated operational conditions is the 2.5 MWe THTR circulator. The scaling factor of approximately three involved in the extrapolation to the PNP and HTR-K designs requires a development and validation test program for this component. Technical features which would be evaluated by U.S. licensing authorities would include:

- compressor blade design
- compressor seal design
- critical speed operating margin (first flexural critical rpm);
- overspeed margin
- control and instrumentation systems
- bearing design; bearing seal design
- rotating member stress levels
- burst protection
- flow control; shutoff valve design
- high-pressure lubricating oil service auxiliaries

Other than validation of the design, no significant safety-related issues appear to be presented by the design.

For the HHT-K, primary coolant circulation is by means of the turbomachine axial compressor. Upon loop shutdown or loss of electric load, the compressor ratio collapses, and a phenomenon which is unique to licensed reactors occurs, namely, coolant mass flow reversal in the affected loop. If flow reversal were unacceptable to U.S. licensing authorities, design accommodation could be made by provision of an isolation or a check valve in the HHT-K design. The compressor (turbo machinery) would also be subject to appropriate design and validation test programs.

3.5.2 Heat Exchangers

3.5.2.1 Steam Generators

The salient characteristics of the HTR-K steam generator and a comparable steam cycle HTGR steam generator by General Atomic Company are summarized in Table 3-3. Tables 3-7 and 3-8 present NUS calculational checks of the HTR-K helical economizer-evaporator bundle (Table 3-7) and straight tube superheater bundle (Table 3-8) steam generator thermal performance. The

TABLE 3-7

HTR-K HELICAL ECONOMIZER - EVAPORATOR
STEAM GENERATOR THERMAL PERFORMANCE

Assumed Parameters	Helical Bundle Low Pressure Shell Side	Helical Bundle High Pressure Tube Side
Mass flow, lb/hr	1.175397×10^6	1.137302×10^6
Inlet temperature, °F	1277.6	356.0
Outlet temperature, °F	572.0	669.2
Pressure, psig	567.08	1707.03
Coolant specific heat, BTU/lb °F	1.25	8.0
Effective heat transfer surface, ft^2	15.25246×10^3	15.25246×10^3

Calculated Thermal Performance Values	Reference Counterflow
Effectiveness, E	0.7656
Heat Transfer, Q, BTU/hr	1.03671×10^9
Thermal Conductance, UA, BTU/hr °F	2.31034×10^6
Overall Heat Transfer Coefficient, U, BTU/ft^2 hr °F	151.51

TABLE 3-8

HTR-K STRAIGHT TUBE STEAM
GENERATOR THERMAL PERFORMANCE

Assumed Parameters	Straight Tube Bundle Low Pressure Shell Side	Straight Tube Bundle High Pressure Tube Side
Mass flow, lb/hr	1.175397×10^6	1.137302×10^6
Inlet temperature, °F	1472.0	669.2
Outlet temperature, °F	1277.6	1004.0 (1027.97)
Pressure, psig	567.08	1667.87
Coolant specific heat, BTU/lb °F	1.25	0.7
Effective heat transfer surface, ft^2	4.4131×10^3	4.4131×10^3

Calculated Thermal Performance Values	Reference Counterflow
Effectiveness, E	0.4469
Heat Transfer, Q, BTU/hr	2.8562×10^8
Thermal Conductance, UA, BTU/hr °F	5.4726×10^5
Overall Heat Transfer Coefficient, U, BTU/ft^2 hr °F	124.01

overall heat transfer, Q, was confirmed to within one percent for both bundles. (The high pressure, tubeside outlet temperature, however, was calculated to be 243.3°C in lieu of the reported 300°C for the helical bundle.) The calculated overall heat transfer coefficients for the helical and straight tube bundles also appear reasonable.

The HTR-K steam raising units consist of six identical once-through steam generators with integral superheaters and reheaters. Water in each steam generator is converted to superheated steam as it passes upward through the economizer, evaporator, and separate superheater sections of a helical tube bundle arranged in an annulus around a central duct containing a straight tube final superheating bundle. Helium from the core outlet plenum flows through a cross duct, and then through an outer gas shroud to the top of the steam generator module. Here the gas flow is reversed (180°), and the helium flows downward over the straight duct part of the superheater (Superheater I) and through the helical tube section containing the Superheater II, evaporator, economizer, and the reheater bundles.

The heat transfer correlation for the water side of steam generators is well known. The primary uncertainty in design methods and data is with the gas side (shell side) of the helical steam generators. The helical coils, the gas side pressure drop, and heat transfer are based on the data of Grimison. A friction factor correlation recommended by Grimison for in-line tube banks is used to calculate the pressure drops. The mass flow of the gas is based on a flow area which assumes an approximate 75 percent in-line and 25 percent staggered tube array. The gas side heat transfer is based on a modified form of the correlation proposed by Grimison based upon other experimental data. In the U.S., the NRC has confirmed this method of calculating the gas side heat transfer by the Oak Ridge National Laboratory for the General Atomic steam cycle HTGR.

At present, however, reliable analytical predictions for the helium flow distributions in large steam generator modules with 180° flow reversals cannot be made with confidence. It is believed that verification tests will be required by the NRC to confirm the analytical procedures. The operational testing of the THTR and Fort St. Vrain steam generator will provide additional confirmation of overall gas side heat transfer and pressure drop for helical bundles. The NRC would also probably require additional supporting information related to steam generator materials, potential helical vibrations, heat transfer data, fluid flow, and hydraulic stability. The General Design Criteria and in-service inspection requirements for the HTR-K steam generators are discussed in Section 3.2.1, and material considerations are discussed in Section 3.2.3.

3.5.2.2 Steam Reformers

The design of the steam reformer has been described.[1] The methane-water and steam mixture from the hydrogasification plant enters with an inlet temperature of approximately 330°C (626°F). The process gas flow is divided by the reformer piping and, upon heating and subsequent collection in the hot gas ducts, exits with a temperature of approximately 500°C (932°F). Throughout the heat transfer, the process gas is catalytically changed by the heat transfer; at the exit of the catalytic tubing, the process gas reaches a maximum temperature of approximately 810°C (1490°F). Upon collection by the hot duct, much of this heat is given up, and the process gas is reduced in temperature to an exit temperature of approximately 500°C.

Since the steam reformer is a primary coolant pressure boundary between dissimilar working fluids, it would be constructed to the requirements of ASME Code Section III, Division 2. The reformer would be in-service inspected to the requirements of Section XI, Division 2. Further discussion of design criteria is presented in Sections 3.2.1 and 3.2.3. The steam reformer is a critical heat exchanger for the PNP and would require a massive development and validation test program to confirm heat transfer characteristics, materials properties, and suitability of mechanical design (allowable stresses). The potential for gross tube failure propagation is a further critical concern.

3.5.2.3 Helium-to-Helium Heat Exchanger

The helium-to-helium heat exchangers are discussed in Section 3.2.

4.0 SAFETY RELATED STRUCTURES, SYSTEMS, AND COMPONENTS

The integrity of HTR structures, systems, and components under seismic loads is one of the most important safety and licensing issues facing the commercialization of both the HTR-SC and PNP/HTR-K plants.

Some of the specific issues which must be addressed within the U.S. licensing framework are

a. Trends in seismic testing philosophy from the original scale model testing to the testing of simulation models to verify analytical codes. It has been recognized that the difficulties of developing scaleup parameters from scale models made the applicability of the results of scale model testing to the real system questionable,

b. Need for development of analytical methods and computer codes that can perform seismic analysis independent of scaling laws,

c. More conservative stress limits for graphite components due to many uncertainties in analytical and testing methods,

d. Development of a long-range verification program to ensure the seismic adequacy of all Seismic Category I reactor components. This program should include

- Seismic classification of components,
- Acceptance criteria,
- Operating environmental conditions,
- Requirements for model testing or proof testing,
- Requirements for development of analytical methods and computer codes.

4.1 SEISMIC DESIGN

Although it is understood that the safety related structures, systems, and components of the PNP/HTR-K plant will be designed to withstand the most severe seismic disturbance postulated at the plant site, no detailed information is yet available on specific design criteria and bases. It is, however, necessary to consider the major criteria and bases in designing the plant. These necessary design criteria and bases are discussed below.

4.1.1 Seismic Classification

Structures, systems, and components important to safety are required to be designed to withstand the effects of SSE and remain functional and are classified as Seismic Category I items in accordance with the requirements set forth in General Design Criterion 2, and to Regulatory Guide 1.29. These plant features are those necessary to assure:

- the integrity of the primary system pressure boundary,
- the capability of safe shutdown of the reactor,
- or the capability to prevent or mitigate the consequences of accidents which could result in off-site doses greater than those specified in 10 CFR 100.

4.1.2 Seismic Design Input

The input seismic design response spectra (1/2 SSE and SSE) to be used in the design of Seismic Category I structures, systems, and components should comply with the requirements of Regulatory Guide 1.60 and 1.61.

4.1.3 Seismic System Analysis

Procedures for modeling, seismic soil-structure interaction, development of floor response spectra, torsional and overturning effects, and values of composite damping must be established.

4.1.4 Interface Requirements for Design of Balance-of-Plant (BOP)

Structural interface requirements in the following areas are required:

- information to establish at all support points the seismic response spectra envelopes for NSSS and BOP interfaces,
- information to establish envelopes of the seismic loads transmission between NSSS and BOP systems for Category I or Noncategory I system interfaces,
- information on mass and stiffness properties of NSSS to be coupled with seismic analysis model of BOP systems.

4.1.5 Seismic Qualification of Instruments and Electrical Equipment

Instrumentation and electrical components required to perform a safety function should be designed to meet Category I design criteria by

- establishing seismic requirements by system seismic analyses,
- incorporating these requirements into equipment specifications,
- and meeting these requirements either by appropriate analysis or by qualification testing.

A general program of seismic qualification of instruments and electrical equipment should be instituted in accordance with the requirements set forth in IEEE-344, 1975 and Regulatory Guide 1.100.

4.2 SYSTEM QUALITY GROUP OR SAFETY CLASS CLASSIFICATION

Fluid system pressure-retaining components important to safety are required to be designed, fabricated, erected, and tested to quality standards (or safety class classification) commensurate with the importance of the safety function to be performed, and are classified in accordance with the requirements set forth in General Criterion 1, the ASME Codes specified in Section 50.55a of 10 CFR 50, and to Regulatory Guide 1.26.

4.3 MISSILE PROTECTION DESIGN

Structures, systems, and components important to safety are required to be designed to withstand or be protected from the effects of various postulated internal or external missiles.

4.4 PROTECTION AGAINST EFFECTS OF PIPE RUPTURE

Protection of systems and components important to safety should be provided against the dynamic effects associated with pipe ruptures (pipe whip) and the resulting discharging fluid. (Reg. Guide 1.46).

5.0 ENGINEERED SAFETY FEATURES

5.1 CONTAINMENT SYSTEM

The functional requirements of the containment system in an HTR plant are to 1) provide a boundary against the leakage of radioactive materials and direct leakage of radiation for all postulated design basis events, 2) protect reactor from severe external conditions such as tornado and missiles, et cetera, and 3) maintain a backpressure to assure adequate core cooling by the core auxiliary cooling system (CACS) in the event of a postulated depressurization accident.

The containment system for an HTR consists of the following subsystems:

- Concrete containment structure
- Containment isolation system
- Containment atmosphere cleanup system
- Containment heat removal system
- Combustible gas control system

The containment systems must be designed as Seismic Category I and should satisfy all the requirements of an Engineered Safety Feature (ESF) of the plant as discussed below.

5.1.1 Concrete Containment Structure

This structure encloses the PCRV, the steam generators, process reformers, fuel loading and discharge systems, and various other equipment, and serves as an additional barrier for fission product release. The containment structure is a Category I reinforced concrete structure with steel liner for leaktightness.

The containment structure for the PNP and HTR-K plants is similar to the conventional dry containment structures employed in PWR and HTR-SC (HTR -- Steam Cycle) plants. The containment structure is to be designed to withstand the maximum pressure and temperature transients of a design basis accident which could result from a failure of the reactor coolant pressure boundary resulting in a discharge of energy from the PCRV into the containment. This design basis accident for the containment design should be defined and analyzed for the PNP/HTR-K Plants.

5.1.2 Containment Isolation System

The containment isolation system should be designed in accordance with the criteria GDC 16, 56, and 57 of the 10 CFR 50, Appendix A. A special consideration should be given to these criteria with regard to various process and steam lines penetrating the containment in an HTR-PH plant.

The primary function of the Containment Isolation System is to control the release of radioactivity from the containment following the design basis depressurization accident (DBDA) to be within the limits set forth in 10 CFR 100. The isolation system is designed to isolate the containment atmosphere from the external environment under all accident conditions by providing a protective barrier for each pipe penetrating the containment.

If a scheme of continuous purging of the containment atmosphere is employed to facilitate containment access during normal operation, special attention should be given to the design assurance that the purge valve will close following a postulated DBDA in sufficient time to limit the offsite doses, and that the effectiveness of the CACS operation is not degraded by a reduction in the containment backpressure.

5.1.3 Containment Atmosphere Cleanup System

The Containment Atmosphere Cleanup System is provided in HTR containment to remove iodine and other particulate fission products from the containment atmosphere to reduce offsite doses to within the limits set forth in 10 CFR 100 following a postulated accident. The functional requirement of the system is to recirculate the containment atmosphere continuously through the filter system following the release of primary system fission products into the containment.

In addition, the Containment Atmosphere Cleanup System in an HTR plant must include chemical process units to remove process gases from the containment in the event of an accident involving the process gas lines.

The system must withstand the initial high containment temperature and pressure following a PCRV depressurization accident, and demonstrate its efficiency for removal of methyl and elemental iodine (as well as other airborne fission products) in a helium-air atmosphere or in helium-nitrogen atmosphere if the containment is inerted with nitrogen.

5.1.4 Containment Heat Removal System

In a conventional HTR-SC plant, the Containment Heat Removal System does not require active components to remove heat from the containment atmosphere following a DBDA. The heat removal is achieved by the passive system consisting of the structures in the containment and the containment walls through which heat is conducted to the external atmosphere. However, because of the much higher temperature of the discharging primary helium in a PNP plant and consequent higher energy discharge into the containment, an active heat removal system may be required in an PNP/HTR-K containment, such as fan and cooling coil system. This system will recirculate the containment atmosphere to reduce post accident pressure and temperature inside the containment.

5.1.5 Combustible Gas Control System

The Combustible Gas Control System may be required to control the concentration of flammable gases such as hydrogen and carbon monoxide. The design of the system would be determined by source terms considered for release of hydrogen and carbon monoxide gases from the process heat module. Even if a Duplex Steam Reformer (DSR) were used in the process module, a design basis source term may have to be defined for the purpose of establishing design basis for the Combustible Gas Control System. In addition, for the source term from the steam ingress accident, the effectiveness of moisture monitoring devices, the amount of available moisture to be released into the core, and the action of the PCRV safety valves must be considered.

For the process gas production plant, the containment may be required to be inerted with nitrogen gas to preclude the possibility of flammable concentration of the combustible gases in the containment. Inerting of the containment atmosphere would preclude the adoption of a continuously purging containment system, which would permit access to the containment during normal operation. It is important to critically examine the design alternatives to determine if the inerting is necessary for the "pot boiler" type plant where all the process steam reformers and steam generators are located inside the PCRV cavity.

One of the major concerns in the HTR containment design is the analysis of containment responses following a postulated design basis depressurization accident (DBDA), including the peak pressure and temperature as well as the iodine removal functions of the containment atmosphere cleanup system.

The containment integrity and functional capability can be affected by two important blowdown phenomena, namely, the hot helium jet and plumes which affect the temperature of local containment boundaries and equipment inside the containment, and subsequent containment atmosphere mixing.

A general program of improving analytical models for evaluating the local thermal response of the containment and the degree of helium mixing is required, as well as verification test programs for these codes. In addition, a test program to confirm the functional capability of the containment atmosphere cleanup system in a DBDA environment is required.

5.1.6 Containment Backpressure Capability

Following depressurization of the reactor coolant system, sufficient backpressure must be maintained within the containment to provide a coolant density compatible with reactor coolant circulation and heat removal requirements of the core auxiliary cooling system.

5.2 CORE AUXILIARY COOLING SYSTEM (CACS) AFTERHEAT REMOVAL SYSTEM (NWA)

The Core Auxiliary Cooling System (CACS) or the Afterheat Removal System (NWA) is an engineered safety feature provided to assure safe cooldown of the core and to maintain adequate decay heat removal in the event main cooling loops become unavailable.

The necessary components for the CACS are

- Auxiliary helium circulators
- Auxiliary circulator service system
- Core auxiliary heat exchangers
- Core auxiliary cooling loop isolation system
- Auxiliary cooling water system
- Auxiliary service water system and associated heat sink
- Core auxiliary cooling actuation system.

There are several design alternatives with regard to the number of independent auxiliary cooling loops from two loops with 100% capacity per loop, three loops with 50% or 70% per loop capacities, to three-loops with 100% capacity per loop. The German PNP, PR-3000, proposes to utilize its main circulating loops for afterheat removal in both normal and emergency operations, rather than providing separate CACS loops for emergency operation. Even with the separate heat sinks and redundant emergency power sources for this scheme, the licensability of this design in the United States is very doubtful.

The current trend in LWR is either two or three loops with 100% capacity per loop in the United States and four 100% capacity loops in the Federal Republic of Germany. Purely from the standpoint of meeting the current NRC single failure criterion, the three loop 50% capacity CASC design seems to be adequate for an HTGR (GAC's HTGR-SC standard plant GASSAR-6 design).

However, the operational flexibility required and siting of the plant near population center that is required for a PNP plant may require three CACS loops with 100% capacity each.

One of the most critical design criteria for the CACS is the performance capability of the CACS circulator under reduced containment back pressure, or alternatively, the maintenance of sufficient containment back pressure to assure adequate core flow following a depressurization accident.

Another important consideration in the design of the CACS is the reliability of the auxiliary loop helium shutoff valves and the main loop shutoff valves. A safety class position indication device may be required for both valves.

5.2.1 Functional Design

Although no detailed design information is yet available, the Core Auxiliary Cooling System (CACS) or the Afterheat Removal System (NWA) of PNP and HTR-K plants will consist of four redundant loops, each with its own PCRV cavity. The auxiliary circulators are driven by electric motors and the auxiliary heat exchangers will be either helical tubes or U-tube design. The heat removal capacity will be 100% per loop for a post accident containment pressure of 3 bars during the design basis accident. The capacity per loop, however, will be at least 50% for the depressurized containment backpressure of one bar.

The feasibility of utilizing three of the six main loop steam generators as a passive heat sink to improve the overall availability of the afterheat removal system has been examined by the PNP project, but the high cost of qualifying these steam generators and associated ducts, valves, circulators, and support systems prompted the Project to recommend an alternate approach. The so called fuel "fast dump" system is one such design under consideration.

5.2.2 Design Basis

The design of the CACS (NWA) should be based on the assumption of PCRV depressurization and the loss of function of the primary means of core cooling. The manner in which these events affect the core and the environment in which the system will operate should be considered in the design.

The total amount of heat to be removed by the PNP CACS (NWA) has been found to be approximately 60 MJ/s, which corresponds to 2% of the rated reactor power.

5.2.3 System Design

The CACS (NWA) system design should meet the functional requirements established for the system from the safety analyses. In designing the system, several factors should be taken into account. The factors to be considered are

- Piping and instrumentation arrangement
- Equipment and component selection
- Applicable codes and standards
- Materials selection and compatibility
- Design pressure and temperature
- Coolant characteristics
- System protection provisions
- System reliability.

The four loop CACS (NWA) proposed for PNP/HTR-K plants will be completely independent. Each will consist of the hot and cold gas duct components located on the primary side of the helium/water heat exchanger, the helium/water heat exchanger, and the auxiliary helium circulator with the back pressure armature (shut-off valve).

The power for the electrically driven motor is fed from the plant emergency power sources. The secondary side of the heat exchanger cooling water system will depend upon the plant site characteristics, but a cooling water system with a water/air heat exchanger as the ultimate heat sink could be used.

All components exposed to the primary helium are located inside the PCRV. Those auxiliary systems not located inside the PCRV are protected from the effects of both the external and the internal missiles and pipe ruptures by means of physical separation of components or by locating them in reinforced concrete structures.

5.2.4 Performance Evaluation

Functional requirements of the system are based on safety analyses and tests in which the predicted effects of a spectrum of postulated events are considered. It is necessary to evaluate the system operational capability to assess the degree and the margin with which the system meets the functional requirements established. Such evaluation will provide the bases for any operational restrictions that might be necessary.

The PNP/HTR-K CACS (NWA) will be sized so that the system will have sufficient capacity to remove residual heat of 2% of the rated thermal output with a coolant flow rate of 2% of the normal flow rate.

A number of salient operational characteristics of the PNP/HTR-K CACS (NWA) are discussed below.

- The afterheat removal system (CACS/NWA) is inactive (standby) during normal reactor operation.
- A small amount of cold helium is continuously circulated through the loop in backward direction for cooling purposes.
- The CACS (NWA) heat exchangers are, therefore, continuously operated under a light load condition to transfer heat from the bypassed coolant in the primary side of the heat exchangers.

- The continuous operation of the CACS/NWA cooling water system enhances the availability of the cooling water system and reduces the standstill corrosion problem.
- The cooling water load is switched from light load to full load condition in case of a need for afterheat removal. The CACS/NWA auxiliary helium circulator speed will be gradually increased in such a way as to avoid an excessive thermal shock in the system components.
- Depending on the operating conditions, the hot helium enters the heat exchangers at a temperature of 600°C to about 1050°C and is cooled down to between 300°C to 200°C.

5.2.5 Inspection and Tests

The bases and means of the performance tests and periodic inspection of the system and the components should be established to enhance the reliability of the system. No detailed information on the PNP/HTR-K CACS (NWA) system is available.

5.3 CORE AUXILIARY COOLING SYSTEM (NWA) COMPONENTS AND SUPPORT SYSTEMS

5.3.1 CACS Helium/Water Heat Exchanger

Each unit of the PNP/HTR-K auxiliary heat exchangers consists of seven identical U-tube heat exchanger modules which are arranged next to each other in a circle within the PCRV cavity. The feedwater inlet of each module can be individually controlled. The water content of each module is approximately 500 kg.

The primary helium flows upward from the bottom to the top on the shell side of the U-tube heat exchanger, while the secondary system water flows downward inside the U-tube and then turns and flows out upward in the same direction as the helium flow. The shell side operating pressure is 35 bars which is approximately 5 bars below the tube side pressure.

The U-tube construction permits the inspection of an individual tube by means of the eddy-current test procedures. Any defective modules can be replaced individually.

The capacity of auxiliary heat exchangers is established conservatively as 60 MJ/s per loop at a pressure below 40 bars for normal afterheat removal operation. The system, however, is sized such that at least 50% of afterheat load per loop can be handled during the PCRV depressurization accident where the back pressure to the auxiliary circulator will decrease and the circulator efficiency will decrease accordingly.

5.3.2 CACS (NWA) Auxiliary Helium Circulator

PNP/HTR-K CACS (NWA) auxiliary helium circulators are designed such that the circulators operate under various operating conditions, including the PCRV depressurization accident condition.

The most conservative operating environment assumed for the PNP-HTR-K CACS (NWA) circulators is the depressurization accident in which the system pressure is assumed to drop to 1 bar and the coolant is a mixture of helium and air. In addition, bypass flows through the inactive (main loop) or defective loops have been considered.

The circulator is a single stage axial flow design. It is driven by a variable speed electric motor which is controlled by the use of a thyristor frequency inverter.

5.3.3 CACS (NWA) Circulator Service System

A service system, composed of several subsystems, is provided for the auxiliary circulators. These subsystems are

- Cooling system for motor
- Lubricating system
- Buffer helium system

No detailed design information on these subsystems for PNP-HTR-K plants is available.

5.3.4 CACS (NWA) Auxiliary Cooling Water System

A redundant, closed loop auxiliary cooling water system is provided for removing heat from the CACS helium/water heat exchangers. The heat transferred to this system is either removed by the nuclear service water system or directly to the ultimate heat sink of the plant via wet cooling towers or air blast heat exchangers. No detailed design information on this system is available for PNP/HTR-K plants.

5.3.5 Core Auxiliary Cooling Actuation System

An automatic actuation system is provided to start the core auxiliary cooling system following a reactor trip if main loop cooling is not available.

5.3.6 Qualification Testing, Surveillance, and Inservice Inspection

Several major components of the core auxiliary cooling system will require testing, surveillance, and inspection to demonstrate the levels of reliability.

5.4 CONTROL ROOM HABITABILITY SYSTEM

The control room habitability system is provided to assure the safety of control room occupants during an accident by providing radiological shielding, a control room emergency ventilation system, and other equipment to assure general habitability of the control room.

5.5 PCRV PENETRATION FLOW RESTRICTORS OR PENETRATION CLOSURES AND COAXIAL FLOW DUCTS

The function of the PCRV penetration flow restrictors and the coaxial duct flow limiters is to limit the consequences of a postulated PCRV depressurization accident caused by failure of either the penetration or the flow duct by restricting the free flow areas to a certain limited value, e.g., 1000 cm^2. As such, these restrictors and limiters are considered as engineered safety features.

An alternative to above designs may be a penetration closure designed as a pressure vessel in accordance with the requirements of ASME Section III, Class 1 structure similar to a pressure vessel of Light Water Reactor. A pressure vessel designed according to the rules of ASME code is not considered to fail due to its extremely low probability.

Therefore, a PCRV penetration and closure system design consistent with ASME Section III, Class 1 requirements will likewise have to be considered not failing. Such a design with bolted, domed flange cover and without the flow restrictor has been proposed by General Atomic Company for their HTGR-

Steam Cycle Lead Plant. The initial reaction of the ACRS to this new design seems to be favorable.

5.5.1 PCRV Penetration Closures

The PNP design for the PCRV penetration closures utilizes the conventional reinforced concrete lids. However, no detailed information is available on the design features of these closures.

5.5.2 Coaxial Helium Flow Ducts

A. Hydrogasification (HKV) Configuration

The helium flow ducts are arranged so that all hot helium pipes are located inside the cold return helium flow ducts in concentric configuration. In order to maintain a low outer pressure shell temperature, a thermal insulation system is necessary. The temperature differential between the inner and outer pipes is established by the outer cold gas temperature limit of 300°C. Ceramic fiber mats with metal supports will be used as insulators.

The coaxial helium flow ducts will have the following features:

- The hot gas pipe with insulation will be prefabricated for ease of installation inside the PCRV.
- The thermal expansion of the duct system will be compensated by means of a piston/ring sliding connection. This system will allow a relatively compact design and could accommodate large axial motions, as well as radial movements.
- No force is transmitted through this connection and, therefore, no bending moment is created at the junction with the graphite blocks of the core.
- This system consists of a sliding sleeve and a gasket mounted by means of a flange connection to the mounting plate.
- The mounting plate will be secured by anchor bolts to the core graphite blocks.

- The differential thermal expansion of steel and graphite will be compensated by a plate spring device.

- Elastic duct bracing devices will be used at the removable shell of the coaxial gas duct. The bracing device will consist of three separate elements.

B. Steam Gasification (WKV) Configuration

As in the case of the hydrogasification configuration, all hot helium pipes are located inside the cold return helium flow ducts. The duct consists of a bracing tube, an inside insulation of ceramic stones, and inside connecting gas duct of graphite. The ceramic stones are fastened to the bracing tube in such a manner as to minimize the effects of differential thermal expansion of the stones with respect to the bracing tube. The inner duct of graphite prevents direct contact of the hot gas with the ceramic stones and provides protection against vibration and erosion. Two different devices for compensating the thermal expansion of the bracing tube are being considered. The advantages and disadvantages of the two schemes are described below:

- Compensators - The advantage of the compensator is the rigidity of the construction with no sliding parts. The rigid compensator, however, has the disadvantage of being long and larger diameter compared to other devices such as a sliding gasket.

- Sliding Gasket - The sliding gasket device has the advantage of being smaller in dimension. The disadvantages of this device in comparison with a compensator are the possible fretting of the sliding components and increased difficulties in inspection.

A test program had been formulated for both the sliding seal devices and the compensators in conjunction with the PNP project. Successful completion of this type of qualification program would be required for 4.5 licensing procedures.

6.0 OTHER SAFETY RELATED SYSTEMS

6.1 STEAM GENERATOR/STEAM REFORMER ISOLATION AND DUMP SYSTEM

The major safety function of the steam generator/process reformer isolation and dump system is to minimize the frequency of lifting the PCRV relief valves as a result of large leaks in steam generator or steam reformer. Another important safety function is to limit the amount of water or process gas leakage into the core which could react with the graphite structures and fuel elements.

6.1.1 Intermediate Loop Shutoff Valve (WKV)

The function of the intermediate loop shutoff valves is to shut off the flow of hot helium in the intermediate loop and to isolate the primary system in the event of a failure of steam generator tubes. The shutoff valves are provided for both hot and cold legs of the primary circuit. Although no detailed design information on these valves is available, some of the main design features are listed below:

- The closing times are between 5 to 30 seconds.
- The shutoff valves are thermally insulated on the inside and rated for a temperature of 900°C.
- The valves are actuated by pressure equalization, as well as by 40 bar pressure differential.
- Three alternative type valves are being considered, namely, the twin plate slide type, the ball valve, and the coaxial valve.
- The primary concern for these shutoff valves is in achieving a low leakage rate. It is, however, considered to be a basically solvable engineering problem.

6.1.2 Steam/Water Dump System and Moisture Monitoring System

In order to mitigate the consequences of a steam/water ingress accident and to limit the potential steam-graphite and steam-fuel reactions, the Moisture Monitoring and Steam/Water Dump Systems are provided in an HTR.

The Moisture Monitoring System provides for a continuous sampling of the primary coolant and provides the moisture level indication together with a trip signal should the moisture level in the primary loop reach the trip setting. The system generates signals which actuate reactor trip, isolation, and dumping of the affected steam generator. The reactor trip function allows for the rapid cooling of the core and hence the termination of the steam-graphite reaction, while the isolation and dumping of the steam generator serve to limit the total amount of steam/water ingress into the PCRV.

The experiences to date have indicated that the real problem in the system is in the area of obtaining a detector with required reliability and performance for long-term operation. Although the Peach Bottom detectors have functioned satisfactorily, their service life has averaged approximately six weeks. The Peach Bottom detectors, Beckman electrolytic hydrometers, therefore are considered to be unsuited for a large HTGR application.

For Ft. St. Vrain, a rugged dewpoint monitor was developed and has been extensively tested at the plant. However, this system is complex and also expensive, and GAC is currently pursuing a course of finding a suitable commercial detector for large HTGR application. The Ft. St. Vrain system, however, meets the performance requirements and could possibly be utilized. These same concerns would apply to U.S. licensing review of the PNP/HTR-K concepts.

Based on U.S. HTGR experience, the performance criteria for an acceptable moisture monitor are

- Response time: less than 5 seconds,
- Accuracy of measurements: 0.1 - 4000 ppm range,
- Repeatability of measurements,
- Long-term reliability of the device.

To develop a moisture monitor that meets the above mentioned requirements will require a substantial amount of effort. No detailed design information is available for PNP/HTR-K plant.

6.1.3 Main Loop Shutoff Valves (HKV)

As discussed in Subsection 5.1, the main loop shutoff valves perform major safety functions in HTR. No information on these valves is available for the steam gasification plant, however.

6.2 REACTOR PLANT COMPONENT COOLING WATER SYSTEM

The reactor plant component cooling water system is provided to transfer heat from plant auxiliary systems and components during normal operation as well as during and after a design basis depressurization accident.

For the PNP/HTR-K plant, the component cooling water system transfers heat from the following auxiliary systems:

- PCRV liner cooling system
- Fuel storage cooling system
- Main loop circulator motor cooling system
- Helium purification system
- Reactor building ventilation system
- Other safety and auxiliary systems and components.

This system is a closed loop cooling system which forms an intermediate barrier between the plant components and the secondary cooling system. In order to localize the possible contamination of the cooling water system, the component cooling water system is divided into several closed loop partial cooling systems in PNP/HTR-K plants. Each partial cooling loop is provided with an isolating heat exchanger and with sufficient redundancy where required.

6.3 EMERGENCY POWER SUPPLY SYSTEMS

Various emergency power supply systems are provided for PNP/HTR-K plants in case the normal plant power supply is unavailable for those plant components which are important for safe shutdown of the plant, necessary to mitigate the consequences of a design basis depressurization accident, and required for other safety functions. These emergency power sources are

- Emergency diesel generators of 10 kV, 660 V, and 380/220 V for the loads which can stand a short period of power interruption after the loss of normal power supply

- DC power supplies of 200 V and 48 V and rotary converters for the loads which require uninterrupted power supply after the loss of normal plant power supply.

The emergency power supply requirements for the conventional block of the gas processing plant are relatively small compared to the reactor plant requirements. Emergency power is required mainly for a lubrication oil pump for the runout turbine condition, for protection and control equipment, and danger indicators.

6.3.1 Diesel Generator Emergency Power Supply System

In accordance with the requirements of General Design Criterion 17 of the Appendix A to DD CFR 50, the onsite emergency diesel generator set for a nuclear power plant should have sufficient capacity and capability to assure that integrity of reactor primary pressure boundary and containment, and other vital functions, are maintained in the event of postulated accidents.

The diesel generator emergency power supply system provided for PNP/HTR-K plants consists of two subsystems, the emergency power systems 1 and 2. These two subsystems together provide a four-fold redundancy with functional independence as well as spatial separation of the subsystems.

6.3.2 DC Emergency Power Supply System

The DC emergency power supply system for PNP/HTR-K plants consists of several subsystems of batteries and rectifiers. Since the rectifiers are powered from the emergency diesel generators, the batteries only have to bridge the time necessary to start the diesels in case of power loss, or to bridge only the time required to transfer to a reserve rectifier in case of rectifier failure. In addition, a 220 V battery through several rotary converters provides an uninterruptible 380/220 V three-phase power supply for control and measurement devices for process computers and essential lighting.

Recently, NRC has raised a concern over the availability of standby battery systems under emergency conditions. This concern is now being investigated by an NRC task force, the finding of which would also affect the PNP/HTR-K design.

6.4 PLANT FIRE PROTECTION SYSTEMS

The plant fire protection systems are provided to detect, extinguish, and mitigate the effects of fires by utilizing various automatically or manually actuated fire protection devices.

It is also required to physically separate all electrical equipment and circuits to preserve the independence of redundant equipment, as well as coating the electrical cables with fire resistant materials. As a result of the TVA Browns Ferry fire, NRC has promulgated considerably more stringent and comprehensive fire protection regulations (U.S. NRC Standard Review Plan 9.5.1) to mitigate the consequences of and/or prevent future fires. PNP/HTR-K designs licensed in the U.S. would definitely have to comply with these more detailed requirements.

6.5 INSTRUMENTATION AND CONTROL SYSTEMS

The various plant protection and control system devices require rigorous and complete qualification test programs.

6.6 DEMINERALIZED WATER MAKEUP AND STORAGE SYSTEM

High quality makeup water for various plant needs must be provided by the demineralized water makeup and storage system.

7.0 RADIOACTIVE WASTE MANAGEMENT

7.1 RADIOACTIVE WASTE SYSTEM

No detailed design information on gaseous, liquid, and solid radwaste systems for PNP/HTR-K is available. Performance requirements of these systems would have a direct impact on site suitability within the U.S. licensing procedure.

7.2 PROCESS AND EFFLUENT MONITORING SYSTEM

The process and effluent monitoring system is designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakages, monitor equipment performance, and monitor and control radioactivity levels in plant discharges into the environs. The system will also monitor the leakage of process gases from the process reforming systems.

8.0 RADIATION PROTECTION SYSTEM

8.1 DESIGN PROVISIONS FOR RADIATION PROTECTION

In order to keep the fission product release from the core into the containment as low as is reasonably achievable (ALARA) during normal operation, and to permit convenient routine maintenance with minimum exposure of maintenance personnel to the radiation, various design features are provided as follows:

- shielding
- maintenance scheduling to minimize the radiation exposure to workers
- design improvements to minimize equipment maintenance.

In compliance with 10 CFR 20, 10 CFR 50 Appendix I, and Regulatory Guide 8.8, the exposure of the plant personnel may be kept as low as is reasonably achievable by providing adequate physical shielding, plant layout, access control or other design features. However, these design requirements of assuring leak-tightness of PCRV and other shielding provisions should be balanced with plant cost considerations.

The major parameters that influence the design of shielding and plant layout are the uncertainties in fission product source terms generated under normal operation conditions such as circulating coolant activity, total plate-out activity in the primary circuit, and plateout distributions in the primary circuit. These uncertainties result in large conservative design margins for PCRV leak-tightness, shielding, containment isolation valves, and maintenance scheduling, and thus higher plant capital and operating costs. Elimination or reduction of these uncertainties is important in designing commercially viable PNP/HTR-K plants which are both economical and safe.

In order to facilitate the above-mentioned design requirements, the PNP/HTR-K plants will incorporate the following design features and operating procedures:

- Components located inside the PCRV cavities must be easy to inspect and repair.

- Accessibility and physical room permit the inspection and repair of the metal parts, such as the liner, isolation valves, core support structures, side and top supports by remote operation.
- Repairs on the inner graphite components can only be performed after core has been emptied.
- Maintenance and repair work outside the thermal shield can be carried out with the reactor shutdown without emptying the core.
- The reactor shielding will be designed to permit a repair crew with the necessary equipment to enter the core cavity in case of a failure of the remote operation equipment. The entry into the core cavity may also be possible under certain conditions after a few weeks of cooldown after reactor shutdown without emptying the core.

8.2 RADIATION SOURCE TERMS

No detailed information on the radiation source terms in PNP/HTR-K plants is available. However, the major parameters that influence the design of shielding and plant layout are the uncertainties in fission product source terms generated under normal operating conditions such as circulating coolant activity, total plateout activity in the primary circuit, and plateout distributions in the primary circuit components.

8.3 MAXIMUM HYPOTHETICAL FISSION PRODUCT RELEASE (MHFPR) AND OFFSITE DOSES

At the present time there are only "interim" criteria for fuel failure and fission product release for MHFPR for an HTGR in the United States (NUREG-0111). The development of fuel failure and fission product release criteria for the pebble bed HTR will be a significant licensing concern facing the PNP-HTR-K plants. This problem is particularly acute for these plants because of the close proximity of plant sites to industrial and population centers.

9.0 STEAM AND POWER CONVERSION SYSTEMS

9.1 HTR-K STEAM AND POWER CONVERSION SYSTEM

No detailed design information on the steam and power conversion system for HTR-K plant is available, except that the system will have the following features:

- Single-shaft steam turbine generator with steam reheating between IP and LP sections of the turbine
- Closed loop cooling of condensor with wet cooling tower
- Four feedwater heater stages
- Turbine driven feedwater pump.

9.2 PNP STEAM AND POWER CONVERSION SYSTEM

No detailed design information on the steam and power conversion system for PNP plant is available. Steam conditions available for power conversion are consistent with current technology and no licensing concerns beyond those normally encountered for light water reactors are anticipated in this area.

9.3 HHT-K POWER CONVERSION SYSTEM

No detailed design information on the direct cycle power conversion system for the HHT-K is available. Obviously a major design, engineering, and qualification program is necessary to develop licensable equipment suitable for commercial applications.

10.0 ACCIDENT ANALYSIS

10.1 DELINEATION OF POSTULATED ACCIDENTS AND ACCIDENT ANALYSES

The principal focus of PNP/HTR-K licensing actions will be on the development of postulated design basis accidents and delineation of accidents for the reactor and other systems capable of the release of radioactive materials. The design basis accidents are postulated and then analyzed to determine the upper limits of public consequences of a wide spectrum of accidents that are considered credible.

In selecting the reactor design basis accidents, four fundamental types of initiating events are to be considered. They are

- Reactivity insertion
- Steam/water and air ingress into the core
- Depressurization of the PCRV
- Loss of forced circulation.

Other design basis accidents are developed from the spectrum of events that could lead to radioactive release outside the containment building. In addition, for PNP plants, considerations in delineating credible accidents should also be given to those accidents involving the process reforming systems. Table 10-1 summarizes the classifications and identifies the engineered safety features provided to cope with the design basis accidents and those accidents that are precluded by design provisions. This summary is only representative and not meant to be inclusive.

10.2 REACTIVITY INSERTION ACCIDENTS

Among various transients caused by reactivity insertions such as loss of burnable poison, moisture ingress, sudden decrease in reactor temperature, and spurious withdrawal of control rod, the control rod withdrawal event may be considered as an enveloping reactivity insertion accident. It should be analyzed on the basis of withdrawing of the maximum worth rod at plant conditions (ranging from source level to full power), with the withdrawal motion terminated by reactor trip on signals from the plant protection system.

TABLE 10-1

CLASSIFICATION OF ACCIDENTS FOR PNP/HTR-K

CLASSIFICATION	REACTIVITY INSERTION	STEAM AND WATER INGRESS	DEPRESSURIZATION OF PRIMARY OR SECONDARY SYSTEM	LOSS OF FORCED CIRCULATION	RELEASE OF RADIOACTIVITY OUTSIDE CONTAINMENT
I. Accidents of lesser consequences than design basis accidents. These are representative and not inclusive.	• Loss of burnable poison • Moisture ingress • Decrease in reactor temperature • Control rod motion • Fuel discharge chute blockage	• Leakage from helium circulator bearing seal • Leakage from PCRV liner cooling system • Steam generator tube leakage • Process reformer tube leakage	• Slow depressurization • Rapid depressurization less than design basis depressurization accident • Steam line break inside the containment • Process gas line break inside the containment	• Temporary loss of main cooling system • Loss of feedwater	• Steam generator tube failure • Primary system instrument piping failure • Failure of gaseous radwaste system • Fuel handling and storage accidents • Release of radioactive liquid • Tritium leakage
II. Design basis accidents	• Spurious rod withdrawal terminated by protective action • Uncontrolled rod insertion terminated by protective action	• Steam generator tube or header rupture • Process reformer tube or header rupture	• Rapid depressurization rate determined by 1000 cm^2 area • Rupture of refueling tube • Rupture of steam pipe • Rupture of process gas pipe	• Sustained failure of normal core cooling • Total loss of core cooling (unrestrained core heatup).*	• Release of radioactivity due to design basis accidents
III. Engineered safety features limiting consequences of design basis accidents.	• Reactor protection system • Emergency shutdown system • Rapid fuel discharge system	• Steam generator dump and isolation system • Process reformer isolation system	• PCRV closure design • Gas duct flow limitors • Containment system • Inerting the process gas ducting • Process gas shut-off valves	• Core auxiliary cooling system • Afterheat removal by PCRV liner cooling system • Containment system	
IV. Accidents precluded by design provisions	• Control rod ejection	• Large moisture ingress combined with reactor depressurization or core heatup	• Depressurization area greater than 1000 cm? • Depressurization combined with containment failure	• Unrestrained core heatup in combination with containment failure	

*Total loss of core cooling (unrestrained core heatup) is the design basis for the maximum hypothetical fission products release used for reactor siting purposes.

Additional description of design of the PNP/HTR-K control rod drive mechanism will be required to justify the omission of control rod ejection accident from the consideration of reactivity insertion accidents.

10.3 STEAM AND WATER INGRESS ACCIDENTS

Ingress of moisture into the reactor is of concern for both chemical and physical reasons in PNP/HTR-K reactors. From a chemical standpoint, the reaction rate between graphite and water vapor becomes significant at temperatures greater than about 700°C. The product from this reaction is largely a mixture of hydrogen and carbon monoxide. For slow rate of moisture ingress, the major concern is with the long term corrosion of the graphite structures and fuel and not with the reaction products.

On the other hand, a high rate of ingress presents the problem of an increase in reactor pressure due to rapid graphite oxidation and rapid generation of the reaction product gases. These product gases pose potential dangers of combustion and explosion. Four different mechanisms by which moisture ingress accident may occur are: 1) steam generator tube rupture, 2) process reformer tube rupture, 3) violation of interface between primary helium and circulator bearing water system, and 4) PCRV liner cooling tube rupture.

Failure of the PCRV liner cooling tubes, circulator bearing malfunctions, and most steam generator and reformer leaks would result in relatively slow rates of ingress, which would be prevented from developing into a significant hazard by early detection and appropriate corrective actions. However, major ruptures of steam generator tubes and process reformer tubes are potentially of sufficient magnitude such that the rate of moisture ingress would require engineered safety features to mitigate the course of the accident.

10.3.1 Failure of Steam Generator Tubes

If the steam generator tubes fail and admit a large amount of steam and water mixture into the reactor, the primary system pressure will rise rapidly and the graphite structures will react chemically with moisture producing hydrogen and carbon monoxide gases.

The engineered safety features provided against this accident will have the following functions:

a. Specially designed moisture monitor will detect the pressure of excessive moisture in the system.

b. The reactor will be scrammed.

c. The defective steam generator will be isolated from its feedwater supply and its outlet steam path.

d. The content of the isolated steam generator is then dumped outside the PCRV to prevent further ingress of moisture into the reactor.

Because of the limited experience with the steam generator isolation and dump system, the design basis moisture ingress accident should be postulated on the basis of failure of the moisture monitor detection to dump the affected steam generator.

10.3.2 Failure of Steam Reformer Tubes

Two cases of the failure of steam reformer tubes accidents will be considered.

Case 1: Pressure on the Reformer Side Higher than the Primary Side

If the pressure on the reformer side is higher than the primary side, in the event of reformer tube failure the process gas (mixture of H_2, CO, CO_2, CH_4, H_2O) will ingress into the primary system of the reactor. This case, therefore, may be analyzed in the same way as a failure in the steam generator tube ruptures. In general, this accident can easily be mitigated by removing the impurities by means of the helium purification plant.

However, as in the case of steam generator tube ruptures, a design basis process tube rupture accident on the basis of failure of process gas shut-off valve should be defined and analyzed to evaluate the upper bound consequences of this accident.

Case 2: Pressure on the Reformer Side Lower than the Primary Side

If the pressure on the reformer side is lower than the reactor primary side, the primary helium will ingress into the process gas stream. The flow is limited by the flow area of the process reformer pigtails. The affected reformer will be isolated and the contents dumped into the containment where the hydrogen and carbon monoxide may be converted into H_2O and CO_2 by means of a catalytic converter. Therefore, this accident will be analyzed as an event of slow depressurization of the primary system.

An appropriate design basic accident for this event should be defined and analyzed, in order to evaluate the upper bound consequences of this accident.

10.4 INTERMEDIATE LOOP COMPONENT FAILURES

For PNP-WKV steam coal gasification plants, failures of intermediate loop components should be analyzed in addition to the primary loop component failures. Some of these intermediate loop component failures are discussed below.

A. Leakage in the Intermediate Heat Exchanger

Since the intermediate loop pressure is one bar higher than the primary loop, any failure of the intermediate heat exchanger tubes will result in ingress of intermediate loop helium into the primary loop.

The plant may continue to operate if the leakage rate is small and the primary helium purification plant can remove the impurities. For larger leaks the affected intermediate loop will be isolated by the intermediate loop shutoff valves after shutting off the associated circulator. The reactor may run with partial load or be shut down. A safety analysis should be performed on the basis of a single failure in the protective devices such as the shutoff valves to evaluate the upper bound consequences of the event.

B. Intermediate Loop Piping Ruptures Inside the Containment

The failure of intermediate piping inside the containment should be analyzed and the consequences evaluated to assess the capability of the containment structure and heat removal capabilities. Also, the event in which a failure of intermediate heat exchanger tubes occurs simultaneously with the rupture of intermediate helium piping should be analyzed.

C. Intermediate Loop Piping Rupture Outside the Containment

The failure of intermediate piping outside the containment should be analyzed and the environmental consequences evaluated.

10.5 PRIMARY SYSTEM DEPRESSURIZATION ACCIDENTS

Identification and analysis of a spectrum of primary system depressurization accidents ranging from the off-set rupture of a small instrument line to a postulated leak area of 1000 cm^2 (design basis depressurization) should be performed on the basis of a single failure in engineered safety features. Both slow and rapid depressurizations of the reactor result in release into the containment building of the inventory of radioactivity circulating with the helium coolant. With rapid depressurization the release could also include significant quantities of adsorbed and plated-out fission products and thus the hazards of the health and safety of the public should be evaluated.

10.6 MAIN STEAM LINE RUPTURE INSIDE THE CONTAINMENT

The effects of a rupture of the main steam line inside the containment should be analyzed to evaluate the consequences of temperature and pressure increase in the containment building.

10.7 RUPTURE OF PROCESS GAS COLLECTOR DUCTS INSIDE THE CONTAINMENT

The effects of a rupture in the process gas collector ducts inside the containment should be analyzed to evaluate the consequences of flammable gas concentration inside the containment building.

10.8 RUPTURE OF REFUELING TUBE ACCIDENTS

Failures of the fuel pebble refueling and discharge tube should be defined and analyzed to assess the effects of these accidents on the basis of a single failure criterion.

10.9 LOSS OF FORCED CIRCULATION

Various cases of partial and complete loss of forced circulation of coolant should be defined and analyzed.

10.11 MAXIMUM HYPOTHETICAL FISSION PRODUCT RELEASE

For the purpose of reactor plant siting evaluation, 10 CFR 100 requires the determination of the radiological consequences of a postulated fission product release accident that would result in potential hazards not exceeded by those from any of the accidents considered credible. This maximum hypothetical fission product release is based on an unrestricted core heatup accident resulting in failure of fuel particle coatings and transport of fission products from the fuel to the containment atmosphere without restraint by the primary coolant system pressure boundary.

10.12 OTHER ACCIDENTS

Other accidents, in addition to the accidents described in the preceding sections, should also be defined and analyzed. Some of these accidents are listed below:

a. Loss of spent fuel cooling

b. Anticipated transients without scram

c. Accidents caused by external events such as

- Earthquake
- Tornado
- Flood
- Aircraft crash
- Fire
- Chemical explosion, etc.

11.0 PROCESS HEAT PLANT SAFETY CONSIDERATIONS

The process streams resulting from PNP operation can either be utilized for chemical heat pipe or coal gasification applications. General criteria have been presented for design of the facility for utilization of this process gas.[1] These criteria include the following:

- Separation will exist between the nuclear plant and process heat plant.
- Process gas pipes into the reactor safety building will have nominal diameters of 500 mm.
- Quick closing isolation valves of a proven design will be utilized.
- No intermediate circuit is required.

Operation of the process heat plant is most likely not a safety concern in the sense that a failure can directly result in accidental radiological releases. However, it can be a concern from the standpoint of radiological contamination, plant transient performance, or missile generation.

11.1 RADIOLOGICAL CONTAMINATION

Performance of the duplex tube steam reformer (DSR) presupposes that its construction will prohibit gross failure propagations during faulted conditions and will maintain high decontamination factors during normal operation. These DSR concerns have been discussed previously in this report. As a minimum, a comprehensive qualification program would be required to verify these DSR performance requirements. However, even assuming these conditions are met, additional restrictions may be placed upon process heat plant operation. This is particularly true in the area of process steam monitoring, installation of HEPA filters, and mitigation of the consequences from gross DSR failure. The latter might be required, even though it is not mechanistically probable, as a measure to ensure a "defense in depth" concept. This latter concern does not have a precedent for evaluation since no NRC regulations exist regarding process heat applications. The only analagous situation now under consideration by NRC is the Midland Power Plant, being constructed by Consumers Power Company, which will supply process steam to the adjacent Dow Chemical Company facility. Midland employs a relatively conservative approach by incorporating a tertiary loop for steam supply to Dow which is subject to strict steam monitoring requirements. Additional investigations are necessary to determine what NRC requirements would prevail in this area of concern and the consequences of these requirements on PNP plant design.

11.2 PLANT TRANSIENT PERFORMANCE

Similar to the HHT-K secondary system, the PNP process may be subject to extreme pressure and temperature transients which could, in turn, impact on the nuclear plant performance during upset emergency conditions. Coal gasification technology is a relatively new field for which a very limited body of experience exists regarding safety analysis and system performance. The U.S. program for coal gasification has recently been receiving increased emphasis.

Coal gasification can be accomplished through an intermediate heat exchanger/steam generator via the steam gasification process or through a steam reformer/hydrogen separator via hydrogasification process. For high BTU coal gasification, which represents the economic incentive for process heat applications, U.S. development efforts have centered on steam gasification. The only hydrogasification process supported in the U.S. is the Hydrane process, considered a third generation type plant.

Along with the steam gasification program, DOE has initiated safety assurance studies to indicate failure mechanisms in the various processes.[20] Each process, in addition to a gasification step, contains a raw product gas upgrading step which can include methanation, CO to CO_2 shift, or additional oxidation. This upgrading step plus gasification defines the overall process flow which is subject to several failure mechanisms.

Preliminary safety analyses have been performed by utilizing the fault tree analysis method. The study has been limited to an evaluation of the gasifier section of the five principal high BTU steam gasification pilot plants being developed in the U.S., namely,

- HYGAS Steam-Oxygen Plant
- CO_2 Acceptor Plant
- BIGAS Plant

- SYNTHANE PLANT
- HYGAS Steam-Iron Plant.

These analyses only extended upstream or downstream from the gasifier section and consider the variation of a single process feed component, one at a time. Therefore, resultant safety analysis and transient performance evaluations are rather preliminary. However, since no specific safety analysis is available for PNP process designs these preliminary studies can provide a useful insight into the problems which may be encountered. A safety analysis summary for one representative U.S. plant is given below.

The HYGAS Steam-Oxygen Pilot Plant is typical of high BTU coal gasification plant concepts which are being developed in the U.S. Considering the substantial effort being placed in U.S. development programs, it seems reasonable to expect that a domestic PNP design might incorporate one of these plant concepts.

HYGAS coal gasification takes place in a single vertically oriented reactor system vessel (i.e., pressure vessel). A coal slurry product feed enters the top and high pressure oxygen and steam flows enter the bottom (typically at 1500 psig and 1200°F). Entering streams flow countercurrently through four principal reactor sections: coal drying section (CDS), low temperature reactor (LTR), high temperature reactor (HTR), and oxy-gasifier section (OGS). Coal slurry (ground coal suspended in light oil) enters the CDS where the oil and moisture are driven off within about fifteen minutes. The temperature of the fluidized bed in this section is nominally 600°F. Dried coal enters the LTR where the more volatile coal fraction reacts rapidly with product gases (about ten seconds). LTR temperatures are nominally 1100-1300°F. The unreacted coal (char) travels downward to the HTR where it is gasified by hydrogen and steam. The HTR fluidized bed nominally operates at a 1750°F and residence time is about twenty-five minutes. Unreacted char travels downward to the OGS where it reacts with oxygen and steam, residing about seven minutes at a nominal temperature of 1850°F. The remaining unreacted char is mixed with steam and exits as waste. The lower HTR and OGS sections are lined with refractory within the carbon steel reactor vessel. A startup heater, to initially raise product streams to gasification temperatures, is also contained within the refractory envelope.

A water jacket which surrounds the refractory, within the reactor vessel, is provided for cooling and maintained by forced circulation. A nitrogen jacket is provided within the reactor vessel, outside of all primary process sections and is maintained 1 psig above system pressure. A manually operated emergency shutdown system is provided, based on system instrumentation alarms (high/low temperature and pressure, flow blockage, etc.) which can shut off steam and oxygen flow and/or dump feed and product gases.

An initial safety analysis of this system has been limited to a mechanistic failure analysis of the HYGAS reactor using the fault tree approach. The preliminary fault tree resulting from this study[20] is given in Figure 11.1. As can be seen, the primary mechanisms of vessel failure are from mechanical failure, overpressure, and overtemperature. Corrosion and erosion are shown as two conditions which can contribute to a variety of failure paths.

Corrosion can be a concern in areas where dead space exists, especially when chemical impurities are present. Erosion is a further concern since coal slurry is known to be extremely abrasive and is capable of eroding hardened stainless steels.

For the HYGAS system, seven significant events (fault tree paths) were identified.

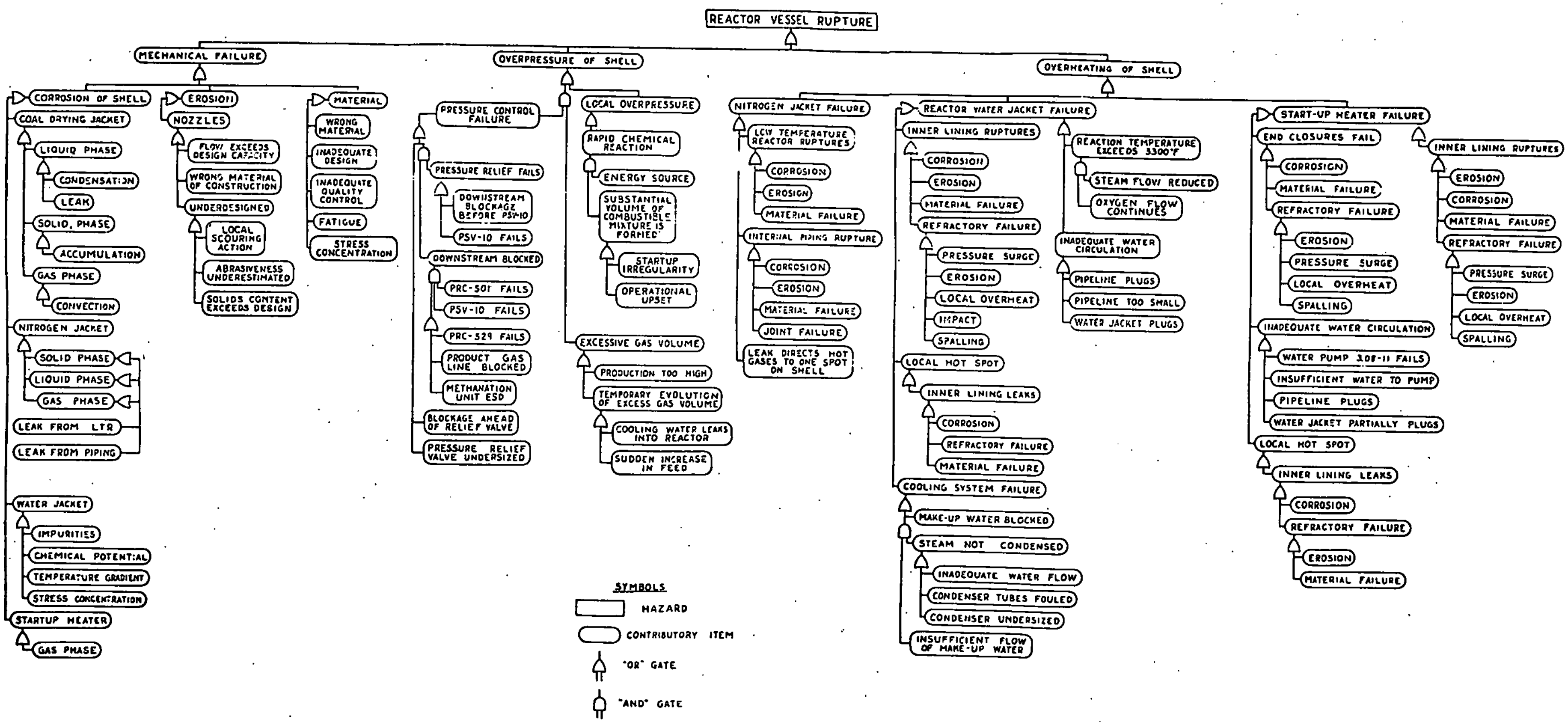
1. Corrosion in stagnant space surrounding the CDS.
2. Corrosion of gasifier shell.
3. Startup heater flame failure leading to explosive mixtures.
4. Loss of jacket cooling water.
5. Failure of jacket liner.
6. Startup heater overheating.
7. Steam blockage with full oxygen flow.

An investigation of process variable abnormal conditions led to the following system transient peak temperatures for vessel overheating. This is indicative of system conditions which must be accommodated for the HYGAS system.

- Char blockage in OGS feed - 2500°F
- Ash blockage in HTR feed - 2500°F
- Increased oxygen flow (20%) - 2000-2150°F

FIGURE 11-1

Fault Tree Analyses for HYGAS Coal Gasification Process [20]



- Increased oxygen flow (50%) - 2700°F
- Decreased steam flow (50%) - 3000°F
- Decreased steam flow (100%) - 3300°F

Simultaneous variation of more than one feed was not considered.

The conclusion which can be drawn from this study is that operating conditions and materials performance considerations of coal gasification facilities, while not inherently unsafe, do lend themselves to various credible accident conditions. Also, abnormal operation can lead to extreme fluctuations in process variable operating conditions. The consequences of these concerns must be evaluated for any PNP process design which would be licensed in the U.S. As a minimum, this evaluation should entail an investigation of primary failure mechanisms and system transient performance.

12.0 SUMMARY AND CONCLUSION

Pebble Bed Reactor concepts now under development in FRG have been reviewed and several concerns regarding the licensability of these concepts within the U.S. regulatory framework have been identified. In areas where system design or performance information was not available for PBR designs, the principal licensing criteria which should be met have been identified. This review has been based on PBR design material presented to us by General Electric Company - Energy Systems and Technology Division, NUS research material on related FRG programs, and information collected by NUS on U.S. HTGR activities.

It appears that the PNP and HTR-K conceptual designs are based on sound engineering principles and constitute a reasonable extrapolation to a commercial scale of AVR operating experience and THTR design experience. These designs appear to be in conformance with FRG regulatory procedures and guidelines and are consistent with accepted FRG practices for development of prototype nuclear reactor concepts. A limited check of some fundamental design parameters indicates they are within achievable engineering limits. Information regarding the HHT-K design concept was not sufficient to render a conclusive evaluation. However, based on general design information for direct cycle gas turbine nuclear plants, it appears that several fundamental engineering problems remain to be resolved. Principal among these are power conversion equipment maintenance and plant transient performance.

Based on the information available, it appears that systems safety analysis and materials performance validation for all PNP concepts are not sufficient in many areas to satisfy U.S. licensing requirements. These concerns may be resolved as more information and further testing and analyses become available. Nevertheless, there are presently several difficulties in licensing these concepts within the U.S. These difficulties arise primarily from either

1. a lack of detailed analysis and/or documentation regarding safety evaluation plus related design basis events and design criteria, or
2. a lack of sufficiently detailed qualification programs and/or full scale tests for major plant systems and components.

The concerns discussed in this report appear to be resolvable through continuing engineering development of PBR design concepts. In fact, many are mentioned in this report only because, based on the documentation available, insufficient information has been given describing conceptual designs. Other concerns are mentioned because, at the present moment, the PBR program is not yet to a stage where sufficient design details have been developed. These concerns are most likely not the result of fundamental engineering problems and resolution could be expected as the program matures, given a commensurate level of support. Further concerns such as materials performance at PBR design temperatures (reflector blocks, in particular) and U.S. seismic qualification of structures and large components may require design changes to achieve licensability; however, there is insufficient information at present to resolve these matters.

Considering the overall nature of the above concerns, it appears that a U.S. sponsored prototype facility, of conservative design, would be very helpful in providing the necessary experience to resolve these problems and could lead to the long term licensability of PBR concepts within the U.S. Based on our review of available information, a summary list of the principal safety and licensing areas of consideration is given in Table 12-1.

TABLE 12-1

PRINCIPAL SAFETY AND LICENSING
AREAS OF CONSIDERATION

Fuel Mechanical Design	Thermal & Fluid Mechanical System
Reflector Mechanical Design	Prestressed Concrete Reactor Vessel
Fuel Exit Chutes	Primary Coolant Circulators
Thermal Shield	Steam Generators
Core Support Structure	Steam Reformer
Reactivity Control Rods	Safety Related Structures, Systems & Components
KLAK Poison Spheres	Engineered Safety Features
Alternate Shutdown Systems	Additional Safety Related Systems
Fuel Handling System	Fire Protection
Spent Fuel Storage	Radioactive Waste Management
Alternate Fuel Cycle	Radiation Protection System
Fission Product Release	Steam and Power Conversion
Nuclear Design	Accident Analysis
Thermal Hydraulic Design	Process Heat Plant Operation
Primary Coolant Pressure Boundary	

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