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U.S./FRG JOINT PEBBLE BED REACTOR EVALUATION

FINAL REPORT
FOR FISCAL YEAR 1977

by
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ABSTRACT

This is the fiscal year 1977 final report of work conducted by General Atomic Company (GA) as part of the U.S./FRG Joint Pebble Bed Reactor Evaluation for DOE/NRA/Reactor Programs. The scope of the program has been to develop technical, economic, and program information in support of decisions and planning by DOE regarding thermal gas-cooled reactor (TGR) development in the U.S. and cooperative international TGR programs. Particular emphasis has been placed on technical review of the large pebble bed high-temperature gas-cooled reactor (HTR) concept under development by the Federal Republic of Germany (FRG).

To make available information supportive of further DOE commitments for TGR development in the U.S. and for cooperative TGR programs with FRG, a matrix of decision input information has been prepared. This information is summarized in this report and includes comparison information for selected energy alternatives.

Technical review of the large pebble bed reactor concept emphasizes the core and related systems, considering that other HTR plant features are similar for either prismatic or pebble bed fuel cores. A review of the prestressed cast iron pressure vessel (PCIV) concept is reported along with safety and licensing considerations for potential U.S. licensing of an FRG-developed HTR plant.

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1. SUMMARY

1.1 SCOPE AND OBJECTIVES

The basic scope of this program is to develop technical, economic, and program information on TGRs needed by U.S. DOE staff for decisions and planning regarding TGR development in the U.S. and cooperative international TGR programs.

1.1.1 Background

The U.S. and FRG have been working together since early 1975 to develop an umbrella agreement which would provide for cooperation in the development of one or more gas reactor concepts. This agreement was executed during February 1977.

Representatives from both countries have agreed to examine several possible scenarios for cooperation in gas-cooled reactor development under which each country would take the lead for selected development programs. To analyze the various alternatives for cooperation, each country must develop an adequate understanding of the other's programs.

The German government is sponsoring a design effort at Hochtemperatur Reaktorbau GmbH (HRB) on an electric generating plant concept designated HTR-K. Design work in progress on direct-cycle versions under the HHT project also involves participation by Switzerland. The program for process heat applications is designated PNP.

Current design and development programs in the U.S. are sponsored both by DOE and private industry and performed primarily by GA. These programs include, for electrical generation, the steam-cycle HTGR Lead Plant and the conceptual design work on a gas turbine high-temperature gas-cooled reactor (GT-HTGR). Process heat applications studies in the U.S. have generally been designated VHTR.

In the FRG the principal objective has been to develop sufficient design and evaluation information to choose by mid-1977 one electricity generating plant concept for development. An industry recommendation was submitted to the German government and utility representatives in June 1977 recommending the direct-cycle Hochtemperaturreaktor-mit-Heliumturbine (HHT) for electricity generation. This recommendation was based upon the conclusions of a task force group involving the HTR-K (steam-cycle) project members and all of the HHT partners.

Work on this task related to HTRs under development in Germany principally involves review and evaluation of the large pebble bed core concept and certain elements of the HTR-Kraftwerk (HTR-K) design. Technical review of the HHT concept was not undertaken, as a direct cooperation has existed since 1974 between the GA GT-HTGR and the HHT project partners. GT-HTGR effort in the U.S. has been sponsored in part by DOE, and detailed technical data on the concept has been regularly reported through that project.

1.1.2 Objectives

Objectives have been to evaluate the pebble bed reactor power plants currently being designed in the FRG for:

1. Assessing the technical and economic potential for application in the U.S.

2. Providing technical and program information to DOE representatives for decisions regarding cooperative international programs.

3. Directly assisting in preparations by DOE staff for participation in U.S./FRG Ad Hoc Gas-Cooled Reactor committee meetings.

The evaluation during FY-77 covered the technology base, potential problems, and licensability in the U.S.

An additional major objective during FY-77 has been to develop TGR decision input information needed by DOE. The decision data can provide a basis for comparing HTR options with certain alternative electricity-producing and process energy systems. Both prismatic fuel and pebble bed HTRs are included for fuel cycle cases representing minimum U_3O_8 resource use, best economics, and best nonproliferation.

1.2 DECISION INPUT INFORMATION

To make available information supportive of further commitments by DOE regarding TGR development in the U.S. and, potentially, cooperative TGR programs with FRG, a matrix of decision input data has been prepared. The format, energy alternatives, and content were developed by the U.S. TGR working committee with input from the FRG through the U.S./FRG Ad Hoc Committee.

Information was compiled for the following selected alternative energy systems:

1. Coal
2. Light water reactor (LWR) (represented by pressurized water reactor).
3. Canadian deuterium-uranium (CANDU) (heavy water moderated reactor).
4. Light water breeder reactor (LWBR).
5. TGRs with prismatic or pebble bed fuel (steam-cycle HTGR, HTR-K; direct-cycle GT-HTGR, HHT; process heat VHTR, PNP).

These decision input data are summarized in Section 2 of this report for nine data categories. Summary tables and a brief discussion are included for each category.

Although it is not within the scope of this effort to weigh the various factors and reach quantitative overall evaluations, some general comments can be made in summary:

1. For several important categories of comparison, particularly resource utilization, safety, fuel cycle costs, proliferation/safeguards, environment, and unique capabilities, the HTR (whether prismatic or pebble fuel) offers advantages compared to alternative nuclear systems.

2. Coal and LWR options are commercially well established, with broad experience.

3. Only HTRs or coal can satisfy very high temperature process heat energy requirements to replace oil and gas fuels.

4. Environmental impact and mine/transport safety are major concerns in coal utilization.

1.3 PEBBLE BED HTR-K REVIEW

Primary emphasis in the technical review and evaluation has been on the large pebble bed reactor core and directly related systems. Other HTR primary circuit components and systems, although influenced, are not strongly dependent upon the fuel element concept, but rather on coolant conditions, performance requirements, risk, and economic factors. However, the prestressed cast iron reactor vessel (PCIV) concept initially studied as part of the HTR-K program was evaluated along with safety and licensing considerations for potential U.S. licensing of an FRG-developed plant concept.

Results of these reviews are presented in Section 3. Conclusions based on this work are that:

1. The pebble bed concept with on-line refueling and OTTO cycle offers the potential for very high helium temperature with relatively low maximum fuel temperature. Uncertainties in core performance increase with extrapolation to large core size.

2. Mechanical design requirements, i.e., reflector, core support, and control rod components, pose some of the most obvious problems of the large pebble bed reactor. However, potential solutions with backup approaches have been defined by the FRG participants for all identified problems. Development details and economic impact of ultimate solutions are yet to be determined.

3. Fuel cycle flexibility, resource consumption, and fuel cycle cost advantages for the pebble bed HTR are essentially the same as those for the prismatic fuel HTR.

4. The PCIV is inherently feasible and probably licensable in the U.S. but requires several years of development. Its advantages of structural redundancy and field erection are shared with the more familiar prestressed concrete reactor vessel (PCRV). Notable differences from the PCRV are that 1) prestressing forces in the tendons and circumferential windings are vessel temperature-dependent with minimum pressure capability in the cold condition, 2) fitting of the liner to the vessel is more difficult, requiring either grout placement or mechanical adjustment, and 3) large foundry capacity, not generally available in the U.S., is required.

5. Licensability review indicates that a plant licensed in FRG will be licensable in the U.S. with the exception that FRG does not require earthquake and accident loads to be combined in component stress evaluations. Reanalysis would be required for U.S. licensing. The necessity for design changes would depend on the analytical results.

6. To provide a basis for evaluation of the pebble bed concept under very high outlet gas temperature (950°C) operating conditions, conceptual core studies were performed for prismatic fuel cores operating at similar conditions. These results are reported in separate topical reports (Refs. 1-1 and 1-2).

REFERENCES

1-1. Baxter, A. M., and P. A. Iyer, "Core Design Study for an Advanced HTGR," General Atomic Company Report GA-A14571, October 1977.

1-2. Asmussen, K., and R. Rao, "Core Design Study of a Very High Temperature Reactor," General Atomic Company Report GA-A14586, October 1977.

2. DECISION INPUT INFORMATION

2.1 DISCUSSION

In partial support of U.S. DOE decisions regarding thermal gas reactor (TRG) development, and of US/FRG cooperation decisions, a matrix of comparable decision input data was prepared. The format and content for compilation of the information were developed by the U.S. TGR Working Committee with input from the FRG through the US/FRG Ad-Hoc Committee.

In West Germany, the choice was principally between TGR options for electricity generation. In the U.S., TGR options have also been compared with coal, LWR, CANDU, and LWBR. The German program has emphasized the pebble bed fuel concept whereas U.S. HTGR development has been based on prismatic fuel.

The prismatic and pebble bed fuel options for the three TGR plant applications, steam-electric, direct-cycle gas turbine-electric, and process heat, create six TGR possibilities. In addition, the three potential fuel-cycle approaches, best economics, best source utilization, or best nonproliferation/safeguards, create a total of 18 TGR possibilities. Fortunately, fuel cycle and plant design studies by GA have shown that current HTGR plant and core design concepts can accommodate any of the fuel cycle options under conditions sufficiently near optimal, so plant design changes are not necessary. Therefore, six HTR designs can be identified, with the three fuel cycle options for each.

The approach was to develop decision input data from available information. It was not within the scope of this study to develop new data, particularly for alternative (non-TGR) systems.

Major data categories are:

- Economics
- Safety
- Environment
- Proliferation/Safeguards
- Resource Utilization
- Development
- Unique Applications/Capabilities
- Maintainability
- Market

Within each of the above categories, key systems characteristics and parameters have been quantified to the extent available information permitted.

In several categories, particularly resource utilization, fuel cycle costs, safety, safeguards/proliferation, environment, and unique capabilities, the HTRs, whether using prismatic or pebble bed fuel, offer advantages relative to alternative nuclear systems. Differences between prismatic and pebble bed fuel HTRs lie primarily in development needs and possibly in reactor plant costs.

2.2 DATA SUMMARY

2.2.1 Economics

Table 2-1 compares capital and power (energy) costs for the systems considered, all adjusted by scaling to 3000 MW(t) rating. The values are based on equilibrium units and 1989 commercial operation. Capital cost estimates for the pebble bed reactor (PBR) were not available for inclusion. However, evaluation results to date suggest costs comparable to an equivalent HTGR or possibly higher costs for the PBR primary system.

Fuel cycle costs are summarized on Table 2-2 for the nuclear systems. Three potential fuel cycle requirements bases are evaluated, each with

TABLE 2-1
ECONOMIC COMPARISON - POWER GENERATING COSTS
(Dollars in Millions)

	Coal	LWR	CANDU	LWBR	SC-HTGR	Dry GT-HTGR	Binary GT-HTGR	VHTR	SC-PBR	Dry GT-PBR
Design Characteristics										
Thermal Power	3000	3000	3000	3000	3000	3000	3000	3000	3000	3000
Efficiency	40.0	32.5	30.0	33.5	39.6	39.6	46.9		∿40	42.7[a]
Electrical Output MW(e) Net	1200	975	900	1005	1188	1188	1407	Process Plant	∿1200	1281
Single or Twin	2	2	2	2	2	2	2	1	2	2
Capital Cost										
Unescalated ($7/77)	953.7[b]	1074.9	1193.1[c]	1074.9[c]	1219.2	1167.1	1412.9	814.5	(d)	(d)
Escalation to 7/89	965.3	1088.0	1207.7	1088.0	1234.1	1181.3	1430.1	824.4	(d)	(d)
Total Escalated ($7/89)	1919.0	2162.9	2400.8	2162.9	2453.3	2348.4	2843.0	1638.9		
Capital Power Cost										
In $/kW(e)	799.6	1109.2	1333.8	1076.1	1032.5	988.4	1010.3	546.3[e]	(d)	(d)
In mills/kW-hr	20.8	28.8	34.6	27.9	26.8	25.7	26.2	14.2[f]	(d)	(d)
Fuel Cycle Power Cost										
In mills/kW-hr	39.8[b]	19.9	17[c]	39.9[g]	15.7	15.7	13.2	6.2[f]	(h)	(h)
O&M										
In mills/kW-hr	4.8[b]	2.4	3.0	≥2.4	2.1	2.1	2.0	Not available	Not available	Not available
Total Power Cost (mills/kW-hr)	65.4	51.1	54.6	70.2	44.6	43.5	41.4	Not available	(d)	(d)

(a) 10°C heat rejection sink temperature.
(b) Ref. 2-1.
(c) Ref. 2-2.
(d) Not available; costs are probably greater than comparable HTGR.
(e) In $kW(t).
(f) In mills/kW-hr(t)
(g) Ratio to LWR from Ref. 2-2.
(h) Expected to be the same as comparable HTGR.

TABLE 2-2
FUEL CYCLE COST SUMMARY
(Costs Leveled Out Over 15 Years)

Cycle	Selection Basis: Best Economics		Best Resource		Best Nonproliferation	
	Recycle	Throwaway	Recycle	Throwaway	Recycle	Throwaway
HTGR SC/DC (39.6% ε)	15.7	19.6	17.5	19.6	18.7	20.9
HTGR DC Binary (46.9% ε)	13.2	16.5	14.8	16.5	15.7	17.6
HTGR Process Heat	(6.2)	(7.7)	(7.0)	(7.7)	(7.4)	(8.3)
PBR SC/DC (39.6% ε)	15.7	19.6	17.5	19.6	18.7	20.9
PBR DC Binary (46.9% ε)	13.2	16.5	14.8	16.5	15.7	17.6
PBR Process Heat	(6.2)	(7.7)	(7.0)	(7.7)	(7.4)	(8.3)
LWR (PWB) (33% ε)	19.4	21.6	$\sim$23 Est.	--	29[a]	21.6
CANDU (29% ε)	--	$\sim$17[b]	--	--	--	$\sim$17[b]
LWBR	--	--	--	--	--	--

Note: Steam cycle and direct cycle design fuel cycle costs in m/kW-hr(e); process heat design fuel cycle costs in m/kW-hr(t) listed in parentheses.

[a] LEU-Th PWR utilizing 20% ε U was calculated. Not included for nonproliferation since it required 93% U-235 topping in recycle uranium.

[b] 16 fuel cycle + 1 D_2O makeup.

recycle and throwaway options. Typically, economic penalties must be accepted for all nuclear systems with either maximum resource utilization or nonproliferation options.

Tables 2-3 and 2-4 summarize the economic groundrules and fuel cycle assumptions.

Economic benefits for HTGR steam cycle development have been estimated using the Hanford Engineering Development Laboratory (HEDL) ALPS model. Equivalent information for the other systems was not available. These results are presented in Table 2-5.

2.2.2 Safety

The material summarized herein has been compiled from the subject literature to provide information through which the safety of selected types of electrical generating plants can be compared. Most of the quantitative information comes from the WASH-1400 reactor safety study (RSS) of light water reactors and the Accident Initiation and Progression Analysis (AIPA) for the HTGR.

As a result of the probabilistic risk studies performed over the past few years to quantify nuclear reactor safety, it has become apparent that public risks from potential nuclear power plant accidents are determined by two major factors: engineering features and design detail, which tend to determine the likelihood of accidents; and inherent safety provided in the plant concept, which dictates the physical processes and the nature of radionuclide transport during the course of accidents and which affects the consequences should engineered safety features fail.

For design and operational features of other plants similar to those defined for the LWR or HTGR, the comparisons were extended to include at least qualitatively the most significant safety factors. For some of the safety issues raised little quantitative information could be found, and

TABLE 2-3
ECONOMIC GROUNDRULES

Unit Size:	
Twin Plant Nominal 3000 MW(t)	
Assume Equilibrium Units	
Assume 1989 Commercial Operation	
Financial Rates:	
Inflationary Escalation	6.0%
Discount Rate	10.2%
Core Working Capital, preirrad	7.8%
Core Working Capital, in core	15.6%
Core Working Capital, postirrad	15.6%
Plant Fixed Charge Rate	18.2%
Interest During Construction	10.2%
Fuel Values:	
U_3O_8 Base Price	\$38/lb in 1976
U_3O_8 Scarcity Escalation Rate	2%
Enrichment Base Price	\$103 in 1976
Tails Assay	0.3%
Time Parameters:	
Fuel costs are discounted to the startup year leveled out over 15 years	
Recycle of bred fuel starts in reload two	
Capital investment recovered over 30 years	
Capacity Factor:	
80%	

TABLE 2-4
FUEL CYCLE UNIT HANDLING COST ASSUMPTIONS[a]

	Recycle		
	HTGR	LWR (U or Th)	CANDU (U or Th)
Fresh Fabrication			
Initial Core	2920$/Block	133$/kg	53$/kg
Equilibrium Reload	3020$/Block	73$/kg	29$/kg
Refabrication	7480$/Block	260$/kg	260$/kg
Reprocessing	4035$/Block	280$/kg	280$/kg
Waste	2235$/Block	95$/kg	95$/kg
Shipping	1715$/Block	21$/kg	21$/kg

[a] Reprocessing unit cost assumed to be zero for throwaway cases. Waste and shipping costs included in throwaway costs.

- o LWBR handling costs ($/kg) same as LWR.
- o LWR-Th, CANDU and LWBR costs estimated.
- o Pebble bed fuel cycle costs estimated to be equal to prismatic HTGR costs for designs optimized for each selection basis.
- o Financial rates, fuel resource costs and time parameters given in Table 2-3.

TABLE 2-5
SUMMARY OF HTGR BENEFITS PREDICTED BY ALPS

Scenario	Nominal HTGR Benefits ($ Billions Discounted to 1977 at 4.5%/7.5%/10%)	Benefits with Capital Cost Variations	Variations in Capital Cost
No recycle	54.7/15/5.3	42.3/11/3.9	HTGR = 1.10 LWR
Uranium recycle in 2000, no HEU	63/16/5.1		
No restrictions in 2000	23/6/2.1	32.6/7.7/2.4 15.2/3.7/1.1 7/1.5/0.3	FBR = 1.75 LWR HTGR = 1.1 LWR HTGR = 1.2 LWR
	38/7.4/2.4 - U-233 symbiosis 34/9.5/3.5 - Pu symbiosis 27.6/7.3/2.5 - HTGR-GT at 0.9 LWR 31.9/8.3/2.8 - HTGR-GT at 0.8 LWR		
No restrictions in 2015	56/13.5/4.1		

for these issues only a general discussion can be provided. When quantitative information could be found, tabulations of the comparable parameters have been made.

Comparisons of TGRs and LWRs based on Probabilistic Risk Analysis (PRA) can be made considering results contained in the Reactor Safety Study (WASH 1400, Ref. 2-3) and the Accident Initiation and Progression Analysis (Ref. 2-4). Examples of comparable items are presented in Table 2-6. Comprehensive analysis of the other types of reactors is presently not available for review, but engineering judgement has been used to consider the relative merits of each design as compared to LWRs and HTGRs. A preliminary CANDU PRA presented in Ref. 2-5 indicates considerable similarities between CANDUs and LWRs in terms of total risk.

Nuclear power plant systems and components are designed to meet the most stringent standards found in any industry. The design basis includes consideration of man-made and natural hazards, as summarized by Table 2-7. These design features are verified prior to construction. Because these events at a power plant site are rare, however, only a few occurrences have actually demonstrated these features. In the 1971 earthquake centered in San Fernando, California, shock waves traveled 100 miles to the San Onofre power plant site. The ground accelerations were not significant enough to require reactor shutdown and the plant continued power production, although nearby fossil units designed to withstand forces 50% greater than the uniform building codes were damaged (Ref. 2-6). Nuclear reactors are designed to withstand earthquake magnitudes which would destroy the surrounding structures (Ref. 2-7).

A comparison of the various reactor vessel designs shows that the prestressed concrete reactor vessel (PCRV) used in TGRs offers some qualitative advantages over steel pressure vessels, even though both are designed to the same earthquake standards. Detailed analyses based on the total pressure vessel failure experience have been undertaken to investigate the potential for catastrophic disruptive failures in steel reactor pressure

TABLE 2-6
ENGINEERED AND INHERENT SAFETY, RELIABILITY, AND FUNCTIONABILITY

Parameter	Coal	LWR	CANDU	LWBR	TGR	
					SC-HTGR	PBR
Potential for core melts based on highest risk events	None	$\sim 5 \times 10^{-5}$/reactor year (Ref. 2-3)	Probably of the same magnitude as LWR	Probably of the same magnitude as LWR	Graphite does not melt; 2×10^{-6}/yr for core heatups	Graphite does not melt; probability of heatup occurrence similar to HTGR
Active engineered safety system functionability	Effectiveness of removal of hazardous materials is about 0.9 to 0.95% for stack releases during normal operation	Rapid response of ECCS required during many accident events. The likelihood of success is about 0.998 (Ref. 2-3). Full range of system response testing is not possible	Requirements similar to LWR	Requirements similar to LWR	Auxiliary cooling can be brought into operation within several hours without significant consequences. Likelihood of success high, 0.9997. Includes ability to restore normal cooling. Full range of required operating conditions can be simulated by testing	Design concept similar to HTGR
Passive safety systems such as the containment	None	Subject to damage which is correlated to various accident sequences. In PWR LOSP sequence the probability of containment failure is 0.2 (Ref. 2-3).	Similar to LWR	Similar to LWR	Dominant failure mode caused by equipment failures rather than a rapid rupture; hence the probability, like that calculated for a core heatup sequence, is low (AIPA)	Design similar to HTGR
Inherent features - coolant (see also Table 2-12	None	Requires sufficient cooling water at all times. Reintroduction of water at high temperatures can introduce reactions which increase accident consequences	Similar to LWR	Similar to LWR	Gas cooled reactors can be cooled by the normal coolant, helium, or at nominal graphite temperatures by air	
heat capacity		Heat capacity of core is much less than in graphite reactors; thus the rate of temperature increase following loss of cooling is rapid	Same as LWR	Same as LWR	High heat capacity of graphite core requires greater heat input to reach damage level temperature; hence for equivalent loss of cooling accidents a much longer time is provided to reestablish forced cooling	Approximately equal to HTGR

TABLE 2-6 (Continued)

Parameter	Coal	LWR	CANDU	LWBR	TGR: SC-HTGR	TGR: PBR
Retention characteristic of core	None	In accidents involving melting, metal reactions at high temperatures aid dispersion of materials from core	Same as LWR	Same as LWR	Quiescent behavior of graphite matrix during accidents acts to retain the fission products even when fuel coatings are broken	Similar to HTGR
Primary boundary	None	Primary coolant boundary failures are a significant contributor to core melt probability	Same as LWR	Same as LWR	PCRV penetration failures such as depressurization accidents are not significant contributors to core overheating conditions (AIPA)	Similar to HTGR
Radioactive inventory (see also Table 2-9)	Radioactive materials are only a small fraction of fuel mass	Fuel elements up to 4 years old remain in core	The use of natural uranium as a fuel limits maximum burnup; therefore the fuel is constantly replaced and radioactive inventory in the core is lessened	Increasing amounts of fuel generated in +4 years old fuel elements	Fuel elements up to 4 years old remain in core	Even though the residence time is limited to slightly over 3 years, the total radioactive inventory is thought to be equivalent to the HTGR-SC

TABLE 2-7

FACTORS AFFECTING THE RESPONSE TO THE NATURAL AND MANMADE HAZARDS SUCH AS EARTHQUAKES, TORNADOS, AND AIRCRAFT IMPACT

Parameter	Coal	LWR	CANDU	LWBR	TGR	
					SC-HTGR	PBR
Response to natural hazards:						
Earthquake	Designed to standards 50% more stringent than the uniform building code	All reactors times are designed to site-dependent factors which may be 10 to 100 times more stringent than those for any other structure in the local area. The protective barriers are:				
		Steel pressure vessel	Steel pressure vessels	Steel pressure vessel	PCRV - affords greatest protection	
Tornados	Same as earthquake	Containment designed to withstand the effects	Probably equivalent to U.S. standards	Containment designed to withstand effects	Containment designed to withstand effects	
Aircraft impacts	Same as earthquake	Same as tornados	Same as tornados	Same as tornados	Same as tornados	

vessels (Refs. 2-8 and 2-9). Steel pressure vessel integrity has been of primary interest in attempts to introduce LWRs into England, and a major investigation (Ref. 2-10) recommends requiring advanced methods of on-line surveillance for steel pressure vessels that would provide early indications of impending failures. Hypothesized disruptive failures are of concern in steel vessels because such events preclude normal and emergency cooling.

PCRVs, on the other hand, are designed with a steel liner supported by the prestressed concrete shell. The barrier diversity provides protection against major disruptive failures of the type hypothesized for the steel vessels. Even so, cracks, leaks, and holes up to the size of the largest penetration are considered in the design of safety systems such as outer containments. More important in terms of safety, however, is that core cooling can still be maintained even if the PCRV is breached (Ref. 2-11). Therefore, even though all reactors are designed to withstand a wide variety of possible events, the characteristics of the PCRV provide additional protection.

Table 2-8 shows the probability of containment failure associated with core overheating accidents. Only LWR and TGR (steam cycle) detailed probabilistic analyses are presently available. According to WASH 1400, Volume V, pages V-25 and V-26 (Ref. 2-3), the accident sequences that lead to some form of rapid containment failure (steam explosion, melt through, equipment failure, burning, or overpressure) have an occurrence probability of 6×10^{-5}/yr for PWRs and 3×10^{-5}/yr BWRs.

The representative event in the AIPA study (Ref. 2-4) for TGRs involves a core heatup with failure to activate the normal isolation system. There is only a weak correlation between the accident sequences and containment failure modes of gas cooled reactors. Summing the sequence probabilities involving containment failure for situations in which releases are possible according to the AIPA study (Vol. IV) results in a probability of less than 1×10^{-9}/yr.

TABLE 2-8
PROBABILITY OF CONTAINMENT FAILURE ASSOCIATED WITH CORE OVERHEATING ACCIDENTS

Parameter	Coal	LWR[a]	CANDU	LWBR	TGR: SC-HTGR[b]	TGR: PBR
Probability of containment failure						
Based on dominant risk event sequences	No containment	3×10^{-5}/yr BWR 6×10^{-5} PWR	Probably equivalent to LWR	Same as LWR	Less than 1×10^{-9}/yr	Equivalent to HTGR-SC
Containment failure likelihood in representative core heatup events	No containment	0.2 (vessel melt through) highly dependent on accident event	Probably equivalent to LWR	Same as LWR	Low - weakly dependent on event sequence	Equivalent to HTGR-SC

[a]Ref. 2-3, NTIS PB-248205, Vol. V, pp V-25, V-27, and V-39.

[b]Ref. 2-4, Vol. IV, p. A2-2.

Table 2-9 lists available core and plant radioactivity (hazardous material) inventories. The values are plant-site quantities only and do not include mine, mill or ultimate waste storage inventories.

Fractional release of certain fission products vs time is listed in Table 2-10 for an LWR and an HTGR (TGR). As noted, the assumption is that the LWBR and CANDU are similar to an LWR and that the pebble bed reactor is similar to the HTGR.

Table 2-11 presents information regarding the margin between operating conditions and failure limits of critical components and the minimum time needed to reach these limits should an accident occur. Each type of plant can be associated with many parameters that are closely related to plant safety. Of the more than 50 nuclear-related safety parameters, only two prominent factors are considered for evaluation in this section. One is decay heat generation rate. Reactors with on-line or continuous refueling can reduce the residence time of fission products in the reactor, thereby reducing the decay heat generation rate following shutdown. A second important parameter related in water reactors to breach of the primary barrier (i.e., fuel cladding) is the Departure from Nucleate Boiling Ratio (DNBR). It is defined as the ratio of burnout heat flux to actual heat flux, and it is dependent on the phase transformation characteristic of the water coolant. The emergency core cooling system (ECCS) in LWRs is designed to keep this ratio greater than 1.3 for the design basis accidents, thus establishing design requirements for system flows, pressures, and temperatures. Clad melting can occur shortly after the DNBR is exceeded in LWRs if backup cooling is not rapidly restored. In the case of TGRs no equivalent parameter exists, since helium does not change phase or cause a rapidly varying heat transfer coefficient with temperature transients. Following loss of forced cooling in a TGR, the first limits reached are internal PCRV component temperature limits resulting from either convective or subsequent forced circulation of high-temperature helium from the core.

The various coolants in a reactor system can have different influences on the course of an accident. In helium cooled reactors (HTGR and PBR)

TABLE 2-9
HAZARDOUS MATERIAL INVENTORY

Parameter	Coal	LWR	CANDU	LWBR	TGR	
					SC-HTGR	PBR
Radioactive Materials, Present in core	Not Applicable					
Noble Gases, Ci		1.2 + 09	Similar to LWR	Assumed Similar to LWR	1.2 + 09	Similar to HTGR
Halogens, Ci		8.7 + 08			8.1 + 08	
Others, Ci		8.7 + 09			9.0 + 09	
Total		1.1 + 10			1.1 + 10	
Radioactive Waste Materials Stored on the Plant Site	Not Applicable					
Gaseous Waste, Ci/a		Unknown	Unknown	Unknown	2×10^4	Assumed similar to HTGR
Solid Wastes, Ci/a					9×10^4	
Liquid Wastes, Ci/a					70	
Spent Fueld, Ci					2.6×10^8	
Nonradioactive Materials		--	--	--	--	--
Waste from Combustion (Fly Ash and Slag), Short tons/a	1×10^4 to 7×10^5					
Waste from Emission Abatement (Oxides of Sulfur and Nitrogen, Hydrocarbons, Aldehydes), Short Tons/a	2×10^5 to 5×10^5					

TABLE 2-10
FRACTION OF INITIAL RADIOACTIVITY RELEASED FROM THE CORE AND PRIMARY COOLANT SYSTEM TO THE CONTAINMENT BUILDING FOLLOWING CORE OVERHEATING ACCIDENT

Time (hr)	Kr-Xe		Iodines		Cesiums		Telluriums		Strontiums	
	LWR[a]	TGR[b]	LWR	TGR	LWR	TGR	LWR	TGR	LWR	TGR
1/60	0.03	--	0.04	--	0.13	--	0.001	--	--	--
1	0.9	--	0.9	--	0.83	--	0.15	--	0.10	--
4	1.0	--	1.0	--	1.0	--	1.0	--	0.11	--
10	1.0	0.02	1.0	3×10^{-5}	1.0	--	1.0	2×10^{-5}	0.11	--
30	1.0	0.04	1.0	1×10^{-3}	1.0	--	1.0	6×10^{-4}	0.11	--
100	1.0	0.13	1.0	5×10^{-3}	1.0	4×10^{-4}	1.0	3×10^{-3}	0.11	6×10^{-6}
200	1.0	0.16	1.0	6×10^{-3}	1.0	8×10^{-4}	1.0	3×10^{-3}	0.11	2×10^{-5}
720	1.0	0.17	1.0	6×10^{-3}	1.0	9×10^{-4}	1.0	3×10^{-3}	0.11	2×10^{-5}

[a] Also representative of LWBR and CANDU (Ref. 2-3, Appendix V).

[b] Representative of both HTGR-SC and PBR.

TABLE 2-11
MARGIN BETWEEN OPERATING CONDITIONS AND FAILURE LIMITS OF CRITICAL COMPONENTS AND MINIMUM TIME LAG BEFORE REACHING LIMITS IN THE EVENT OF ACCIDENTS

Parameter	Coal	LWR	CANDU	LWBR	TGR: HTGR-SC	TGR: PBR
Core inventory contributing to decay heat production	Not applicable	Fuel elements up to 4 years old	On-line refueling coupled with lower fuel burnups tends to reduce the decay heat generation rate	Breeding along with 4 year old elements makes decay heat generation rate slightly greater than for LWR	Approximately equivalent to the LWR except core thermal capacity is much greater	On-line refueling tends to reduce decay heat generation rate, but core thermal capacity somewhat lower than for HTGR
Limiting operating parameter	Probably boiler tube temperatures	Departure from nucleate boiling ratio	Departure from nucleate boiling ratio	Departure from nucleate boiling ratio	Helium temperature	Helium temperature
Margin between normal operating condition and limit of the critical parameter	Small	1.3 (typical PWRs, PSARs)	Probably the same as LWR	Probably the same as LWRs	Based on typical operating temperatures the ratio is $\left[\frac{2100 \pm 100 + 460}{\sim 1366 + 460}\right] \approx 1.4$	
Time required to reach limiting condition following accident event	Unknown	Less than 5 min	Somewhat longer time than for LWR because of lower decay heat generation rate	Less than for time LWRs because of slightly greater heat generation rate	About 3 hr	Similar to HTGR

the inherent properties of helium preclude unusual reactions with fuel coatings or other materials. In addition, since helium at the densities of operation is transparent to neutrons, no reactivity changes are caused by its presence or absence. Furthermore, even if the coolant escapes through a rupture in the main coolant system, circulating atmospheric pressure helium or air can provide cooling at the decay heat level. Thus, the helium coolant has no detrimental effects on the course of an accident (Ref. 2-4).

For both heavy and light water reactors, three factors influence the course of accidents. First, density changes can cause reactivity perturbations which can often shut off the chain reaction. However, reintroduction of cold water can cause a positive reactivity insertion, although the use of boron solutions as a poison complicates the effect of the water reactivity correlations. Second, during accidents involving loss of coolant, cooling water must be supplied very quickly to prevent clad melting. No other available medium (e.g., air) can sufficiently cool the core. The third problem involves potential reactions between water and cladding which result in explosive mixtures that may cause cladding failure along with breach of the containment or primary coolant boundary. Extreme care in design, maintenance, and operation is required to ensure that water coolant reactions do not increase the spread of radioactive materials during accident sequences (Ref. 2-12).

Table 2-12 summarizes these effects of primary coolant on the course of accidents.

Characteristics of radionuclide protection barriers are listed in Table 2-13. Even though the sequential barriers are functionally similar for LWRs and TGRs, there are significant differences in the materials and structures, as noted in the table. Table 2-14 compares the primary circuit activity levels for LWRs and TGRs.

TABLE 2-12
EFFECT OF PRIMARY COOLANT ON THE COURSE OF ACCIDENTS

<table>
<tr><th rowspan="2">Parameter</th><th rowspan="2">Coal</th><th rowspan="2">LWR</th><th rowspan="2">CANDU</th><th rowspan="2">LWBR</th><th colspan="2">TGR</th></tr>
<tr><th>SC-HTGR</th><th>PBR</th></tr>
<tr><td rowspan="4">Influence of primary coolant in the event of accidents</td><td rowspan="4">The hot gaseous combustion products can cause equipment damage by corrosion, but the failure consequence is primarily economic (e.g., equipment damage and plant outage)</td><td colspan="3">The water coolant is subject to metal/water reactions at high temperature. They liberate combustible gases whose reaction conditions may exceed containment design limits.</td><td colspan="2" rowspan="4">Helium inert gas has no violent reactions with other materials.

Loss of helium may allow air streams to enter primary circuit causing slow chemical reactions. Even so, reactivity perturbations are minimized because of continued neutron transparency.

Loss of helium does not prevent cooling at decay heat levels</td></tr>
<tr><td colspan="3">Primary coolant density changes affect reactivity</td></tr>
<tr><td>Positive or negative depending on boron concentration</td><td>Spectral shifts may cause reactivity anomalies</td><td>(Appear same as LWR)</td></tr>
<tr><td colspan="3">Plants are subject to uncontrolled core heatups and melting from decay heat generation when water coolant is not available</td></tr>
</table>

TABLE 2-13
CHARACTERIZATION OF THE PROTECTION BARRIERS

Parameter	Coal	LWR	CANDU	LWBR	TGR	
					SC-HTGR	PBR
Characterization of the protection barriers (radionuclide protection barriers)	Not applicable	The primary barrier to the release of fission products is the zircalloy clad materials encasing the reactor fuel pellets. The interior surfaces of the primary coolant system offer a retention sink through plateout effects. The primary coolant system is isolated from the secondary side by a steam generator. Fission product inventories within the primary coolant loop are controlled and maintained through the use of on-line primary coolant purification systems. The reactor vessel contains the primary coolant and reactor core and effectively isolates the fission products from the containment. The reactor vessel is surrounded by a containment which serves to limit the release of radioactive materials. Additionally, the containment building has atmosphere and liquid process systems to reduce free fission product inventories.			The primary barrier to the release of fission products is the ceramic multi-layer coating on the fuel particles. The fuel rod matrix and fuel element graphite offer large surface area sinks upon which certain fission products can be sorbed. The interior surfaces of the primary coolant system offer a retention sink through plateout effects. The primary coolant system is isolated from the secondary side by the steam generator. A portion of the fission products carried by the primary coolant gas is diverted to and removed by the helium purification system. The insulated water-cooled liner is an additional barrier preventing diffusion of fission products from the primary coolant circuit. The liner is backed by the PCRV concrete, which acts as a delay bed and sink for certain fission products. The containment provides the final barrier to environmental discharge of fission products. Additionally, the containment building has a cleanup system.	

TABLE 2-14
PRIMARY CIRCUIT ACTIVITY

Parameter	Coal	LWR	CANDU	LWBR	TGR	
					SC-HTGR	PBR
Primary Circuit Activity						
Design Levels, Ci	Not applicable	1.8×10^4 (Ref. 2-13)	Unknown	Similar to LWR	7.7×10^4	Similar to HTGR
Expected Levels, Ci	Not applicable	4.2×10^3 (Ref. 2-13)	Unknown	Similar to LWR	1.6×10^4	Similar to HTGR

2.2.3 Environment

Tables 2-15 through 2-17 compare the relative environmental effects of the five power generation systems. Table 2-15 contains data which describe the most important environmental aspects of the non-TGR types of electric power generating plants. The data presented in Table 2-16 delineate the effects of three different fuel cycles for both the steam cycle and direct cycle HTGR. No distinction is made between the environmental effects of the HTGR and the pebble bed TGR; since fuel-cycle parameters and plant performance are essentially the same, environmental effects of the two reactor types should be essentially the same.

Table 2-16 compares coal-fired and HTGR process heat plants that have been designed to produce synthetic natural gas (SNG). The other reactor types are not included in this comparison because they are not capable of operating at temperatures high enough to support this type of process.

2.2.4 Proliferation/Safeguards

Primary consideration in comparing weapons proliferation/safeguards characteristics of reactor and fuel cycle alternatives are the total fissile material stream quantities and the difficulty involved in separating out weapons grade material. Ideal systems would use dilute, difficult-to-separate fissile streams throughout the fuel cycle while retaining acceptable fuel cycle costs and uranium resource utilization.

Tables 2-18, 2-19, and 2-20 summarize the fissile mass flows, fuel cycle costs, and U_3O_8 consumption for the alternative fuel cycle constraints and reactors evaluated. Fissile material separation characteristics are indicated by a three letter code following each fissile mass flow value to be read as follows:

TABLE 2-15
ENVIRONMENTAL EFFECTS - NON-TGR

	Coal	Ref.	LWR	Ref.	CANDU	Ref.	LWBR	Ref.	LWBR - Prebreeder
Use of Natural Resources									
Land									
Site (acres)	525-1200 includes coal and ash storage	2-19 2-25	500 includes exclusion area	2-19	Approximately equal to LWR		Approximately equal to LWR		Approximately equal to LWR
Mining (Acres/yr)	100 to 400	2-14	20 to 50	2-14	16 to 40 throwaway cycle, 12 to 30 thorium cycle	2-22	Zero U_3O_8 requirement	2-2	37 to 92[c]
Waste Storage (acres/yr)	Included in 1 above	2-19 2-25	Tailings - 30 to 70	2-14	60% of LWR (18 to 42)	2-2	No Tailings		Tailings - 55 to 129[c]
Water (gpm)									
Plant Cooling Flow[a]	568,000[d]	2-16	744,000	2-16	870,000[e]		Approximately equal to LWR	2-2	Approximately equal to LWR
Plant Consumption[b]	10,500[d] (11,600 w/scrubbers)	2-15 2-16	13,800	2-16	16,100[e]		Approximately equal to LWR	2-2	Approximately equal to LWR
Enrichment Consumption	--		418[f]		Zero	2-2	Zero	2-2	686[c]
Effluents									
Chemical (Tons/yr)	$2.2 billion damage from releases	2-18							
Gaseous	SO_x - 0.35 x 15 x 10^4 NO_x - 3.5 x 10^4[g]	2-19	Equivalent to 45 MW(e) coal-fired plant	2-20	Total SO_x, NO_x and particulate = 220	2-2	Total SO_x, NO_x and particulate = 1.4 x 10^3	2-2	Equivalent to 74 MW(e) coal-fired plant[h]
Liquid	Acid = 0 to 2124 Iron = 0 to 2795 Silt = 0 to 35,523 Other = 1000	2-19	Silt = 2.4 x 10^4 Other - 1 to 5000	2-19	10% of LWR[i]		50% of LWR[i]		Approximately equal to LWR
CO_2	7.5 x 10.5 x 10^6	2-19	Equivalent to 45 MW(e) coal-fired plant	2-20	Zero		Zero		Equivalent to 74 MW(e) coal-fired plant[h]
Particulate	93 to 42,000[j]		Equivalent to 45 MW(e) coal-fired plant	2-20	Included in gaseous information	2-2	Included in gaseous information	2-2	Equivalent to 74 MW(e) coal-fired plant[h]
Radiological (Ci/yr)									
Gaseous	Mixed radium - 0.028 to 1, H^3 - unknown[k]		Kr^{85} - 4 x 10^5 I^{131} - 0.83 Rn-75 to 4800 U^3 - 1.8 x 10^4	2-20	Kr^{85} - 0 (No recycle) I^{131} - not available Rn-45 to 2900 H^3 - 10^5	2-22	Kr^{85} - 4 x 10^5 I^{131} - not available Rn-14 H^3 - 10^4[i]	2-2	Approximately equal to LWR
Liquid	Zero		2.1 U and daughters	2-20	H^3 - 17,000; other -3.2[i]		H^3 - 0; other 1.4[i]		Approximately equal to LWR
Rejected Heat (Btu/yr)	3.4 x 10^{13}[e]		4.4 x 10^{13}	2-17	5.1 x 10^{13}[e]		Approximately equal to LWR		Approximately equal to LWR

TABLE 2-15 (Continued)

	Coal	Ref.	LWR	Ref.	CANDU	Ref.	LWBR	Ref.	LWBR - Prebreeder
Waste									
Coal Ash, tons/yr	7.2 to 74 x 10^4 (g)		Equivalent to 45 MW(e) coal-fired plant (l)		Zero	2-2	Zero		Equivalent to 74 MW(e) coal-fired plant (h)
Uranium Tailings, tons/yr	—		7.1 to 10.0 x 10^4 (m)		4.2 to 6.0 x 10^4	2-2	1.4 to 2.0 x 10^4	2-2	1.3 to 1.8 x 10^5 (c)
Radioactive, solid, ft^3/yr	—		20,000 low level 70 high level	2-2	Approximately same as LWR	2-2	Approximately same as LWR	2-2	Approximately same as LWR
Coal Sludge, tons/yr	1.9 to 4.9 x 10^5	2-21	Equivalent to 45 MW(e) coal-fired plant (l)		Zero	2-2	Zero		Equivalent to 74 MW(e) coal-fired plant (h)

(a) Encompasses essentially all cooling requirements of the fuel cycle.

(b) Based on mechanical draft wet towers, high relative humidity.

(c) Based on GA calculations of U_3O_8 and SWU needs for LWR and LWBR prebreeder. Data were calculated using an equation similar to that of footnote f of Table 2-16.

(d) Based on the assumption that the efficiencies of both HTGR and coal-fired plants are 39%.

(e) Heat rejected by LWR = 4.4 x 10^{13} Btu/yr (Ref. 2-17). Other heat rejection values calculated using:

$$(\text{Heat rejected, other system}) = (\text{Heat rejected, LWR}) \times \frac{(1-\text{eff}_{\text{other}})}{(\text{eff}_{\text{other}})} \times \frac{(\text{eff}_{\text{LWR}})}{(1-\text{eff}_{\text{LWR}})}$$

where eff = efficiency of plant, eff_{HTGR} = 39%, and $\text{eff}_{\text{CANDU}}$ = 30% (Ref. 2-2).

(f) Calculated using the following equation:

$$\frac{(\text{Discharge to air during enrichment, LWR})}{(\text{Discharge to air during reactor operation, LWR})} \times (\text{Total consumption during LWR operation}) = (\text{Consumption during enrichment, LWR})$$

Discharge to air data taken from Ref. 2-17, Table S(A)-1, Option 6. Total consumption taken from Ref. 2-16. This equation was used in order to maximize compatibility of results, i.e., results based on Ref. 2-16 quantitative data.

(g) Lower limit release from Ref. 2-21. Upper limit release from Ref. 2-19.

(h) Based on GA calculations of SWU needs for LWR and LWBR prebreeder and an equation similar to that found in footnote f of Table 2-16.

(i) From Ref. 2-2, Table E-1 of Volume 3 and Table 5 of summary paper.

(j) Lower limit release from Table 6.5 of Ref. 2-21. Upper limit release from Ref. 2-19.

(k) Based on methodology used by M. Eisenbud and H. Petrow, in "Radioactivity in the Atmospheric Effluents of Power Plants that use Fossil Fuels," Science Magazine, April 1964.

Maximum permissible concentrations (MPC) taken from 10 CFR 20, App. B.

Isotope	MPC
$RA^{226,228}$	2 x 10^{-12}
Kr^{85}	3 x 10^{-7}
I^{131}	1 x 10^{-8}

Reference article states that 28 mci to 1 ci of mixed radium isotopes are emitted by coal-fired plant per year. Typical calculation:

$$(28 \text{ mci Ra}) \times \frac{(3 \times 10^{-7})}{(2 \times 10^{-12})} = 4.2 \times 10^6 \text{ mci } Kr^{85} \text{ equivalent.}$$

(l) From Ref. 2-20, based on assumption that all electricity used in the fuel cycle is supplied by coal-fired plants.

(m) Lower limit release from Ref. 2-19. Upper limit release from Ref. 2-20.

TABLE 2-16
ENVIRONMENTAL EFFECTS - SC-HTGR (Ref. 2-16)

	Best Nonproliferation	Ref.	Best Resource Utilization	Ref.	Best Economics	Ref.
Use of Resources						
Land (acres/yr)						
Site	500, including exclusion area	2-23	500, including exclusion area	2-23	500, including exclusion area	2-23
Mining	49[(c)]		15[(c)]		22[(c)]	
Waste Storage	55[(d)]		17[(d)]		25[(d)]	
Water (gpm)						
Plant Cooling Flow[(a)]	568,000	2-16	568,000	2-16	568,000	2-16
Plant Consumption[(b)]	10,500	2-16	10,500	2-16	10,500	2-16
Enrichment Consumption	585[(e)]		203[(e)]		293[(e)]	
Effluents (tons/yr)						
Chemical						
Gaseous (SO_x and NO_x)	Equivalent to 63 MW(e) coal plant[(f)]		Equivalent to 22 MW(e) coal plant[(f)]		Equivalent to 32 MW(e) coal plant[(f)]	
Liquid	97	2-24	97	2-24	97	2-24
CO_2	Equivalent to 63 MW(e) coal plant[(f)]		Equivalent to 22 MW(e) coal plant[(f)]		Equivalent to 32 MW(e) coal plant[(f)]	
Particulate	Equivalent to 63 MW(e) coal plant[(f)]		Equivalent to 22 MW(e) coal plant[(f)]		Equivalent to 32 MW(e) coal plant[(f)]	
Radiological						
Gaseous	KR^{85} - 5.3×10^5 I^{131} - 0.18 Rn - 500 H^3 - 3.3×10^4	2-24	Kr^{85} - 5.3×10^5 I^{131} - 0.18 Rn - 500 H^3 - 3.3×10^4	2-24	Kr^{85} - 5.3×10^5 I^{131} - 0.18 Rn - 500 H^3 - 3.3×10^4	
Liquid	2.0 uranium and daughters	2-24	2.0 uranium and daughters	2-24	2.0 uranium and daughters	
Rejected Heat (Btu/yr)	3.44×10^{13}[(g)]		3.44×10^{13}[(g)]		3.44×10^{13}[(g)]	
Waste						
Coal Ash, tons/yr	Equivalent to 63 MW(e) coal plant[(f)]		Equivalent to 22 MW(e) coal plant[(f)]		Equivalent to 32 MW(e) coal plant[(f)]	
Uranium Tailings, tons/yr	82,500[(h)]		82,500[(h)]		82,500[(h)]	
Radioactive, solid, ft^3/yr	20,000 low level 70 high level	2-2	20,000 low level 70 high level	2-2	20,000 low level 70 high level	2-2

TABLE 2-16 (Continued)

DC-HTGR (Ref. 2-16) - Dry Cooled, Binary Cycle Plant in Parentheses

	Best Nonproliferation	Ref.	Best Resource Utilization	Ref.	Best Economics	Ref.
Use of Resources						
Land (Acres/yr)	Same as SC-HTGR		Same as SC-HTGR		Same as SC-HTGR	
Water (gpm)						
Plant Cooling Flow[a]	0 (1.8×10^6)[i]		0 (1.8×10^6)[i]		0 (1.8×10^6)[i]	
Plant Consumption[b]	281 (11,300)[i]		281 (11,300)[i]		281 (11,300)[i]	
Enrichment Consumption	585 (585)[e]		203 (203)[e]		293 (293)[e]	
Effluents (tons/yr)						
Chemical	Same as SC-HTGR		Same as SC-HTGR		Same as SC-HTGR	
Radiological	Same as SC-HTGR		Same as SC-HTGR		Same as SC-HTGR	
Rejected Heat (Btu/yr)	4.2×10^{13}[i] (3.8×10^{13})		4.2×10^{13}[i] (3.8×10^{13})		4.2×10^{13}[i] (3.8×10^{13})	
Waste	Same as SC-HTGR		Same as SC-HTGR		Same as SC-HTGR	

(a) Encompasses essentially all of cooling requirements for the fuel cycle.

(b) Mechanical draft dry cooling tower, high relative humidity.

(c) Based on 3 MT of U_3O_8/acre and fuel requirement calculated at GA.

(d) Based on 1500 tons of tailings per acre.

(e) Calculated using the following equation:

$$\frac{(\text{SWU, HTGR})}{(\text{SWU, LWR})} \times (\text{Discharge to air during enrichment, LWR}) =$$

(Discharge to air during enrichment, HTGR)

SWUs were calculated at GA. Discharge to air during enrichment information was taken from Ref. 2-17.

(f) Assuming that essentially all electricity consumed in the fuel cycle is used by the enrichment plant and that all electricity is generated by coal, the equilvalent coal-fired plant size may be calculated using the following method and the fact (Ref. 2-20) that an LWR uses the equivalent of a 45 MW(e) plant. The equation used is:

$$\begin{array}{c}(45\ \text{MW(e) coal-fired plant})\\ \text{for LWR}\end{array} \times \frac{(\text{SWU, HTGR/yr})}{(\text{SWU, LWR/yr})} = (\_\text{MW(e) coal-fired plant for HTGR})$$

The separative work units for each reactor type have been calculated at GA. The SWU used for LWR = 70,000/yr.

(g) Heat rejected by LWR = 4.4×10^{13} Btu/yr (Ref. 2-17). Other heat rejection values calculated using:

$$(\text{Heat rejected, other system}) = (\text{Heat rejected, LWR}) \times \frac{(1-\text{eff}_{other})}{(\text{eff}_{other})} \times \frac{(\text{eff}_{LWR})}{(1-\text{eff}_{LWR})}$$

where eff = efficiency of plant

$\text{eff}_{HTGR} = 39\%$

$\text{eff}_{CANDU} = 30\%$ (Ref. 2-2)

(h) Based on U_3O_8 requirements calculated at GA and assumes 500 MT of tailings for every MT of U_3O_8 required. From Ref. 2-2, Table E-1, 180 MT of U_3O_8 yields 91,000 MT of tailings. Therefore, $\frac{91,000}{180} \approx 500$.

(i) Calculated at GA.

TABLE 2-17
ENVIRONMENTAL EFFECTS OF COAL-FIRED AND HTGR PROCESS HEAT PLANTS[a]
PRODUCING SYNTHETIC NATURAL GAS

	Coal	Nuclear
Resource Use		
Fuel and Feed (tons/yr)	Coal = 2.3×10^{7}	Coal = 1.21×10^{7} Uranium Ore = 3.2×10^{4}
Land (acres/yr)	2900 to 5600	877 to 3210
Solid Waste (tons/yr)	3.50×10^{6}	3.2×10^{6}[b]
Process Water Consumption (gpy)	7.0×10^{9}	4.5×10^{9}
Cooling Water (gpy)	3.3×10^{10}	1.8×10^{10}
Effluents (tons/yr)		
SO_x	2.1×10^{6}	7.6×10^{5}
NO_x	6.6×10^{6}	1.4×10^{5}
CnHm	4.6×10^{3}	15.5

(a) Each plant produces 6.13×10^{8} SCFD at 1074 Btu/SCF.
(b) Includes burnable char.

TABLE 2-18
FISSILE QUANTITIES AND SAFEGUARDS CHARACTERISTICS
LEU (0.2% TAILS)

	Recycle						Throwaway					
	FCC (mil/kW-hr)	30 Yr U_3O_8 [MT/GW(e)]	Fissile Material and Separation Requirements [kg/GW(e)] U-233 Sep	U-235 Sep	Pu-239 Sep	Ratio Wt FE/FISS[a]	FCC (mil/kW-hr)	30 Yr U_3O_8 [MT/GW(e)]	Fissile Material and Separation Requirements [kg/GW(e)] U-233 Sep	U-235 Sep	Pu-239 Sep	Ratio Wt FE/FISS
LWR	19.4	3690					21.6	5830				
Initial			0	1874 ICC	0	42			0	1874 ICC	0	42[a]
Reload, yr^{-1}			0	609 ICC	271 CCG	30			0	856 ICC	0	31[a]
Discharge, yr^{-1}			0	161 IHC	317 CHG	55			0	227 IHC	180 CHG	65[a]
Weapon Requirement, kg, metal				-	8					-	8	
LWBR - PBR	Unknown	5700 with U-235 recycle										
Initial			0	4300 ICC	o	26						
Reload, yr^{-1}			0	1500 ICC	0	24						
Discharge, yr^{-1}			300 CHR	800 IHC	100 CHG							
Weapon Requirement, kg/hm												
LWBR - BR												
CANDU		Unknown					17.0	4130				
Initial			0	980 ICC	0	Unknown				980 ICC	0	Unknown
Reload, yr^{-1}			0	530 ICC	360 CCG					931 ICC	0	
Discharge, yr^{-1}			0	230 ICC	360 CHG					<400 IHC	360 CHG	
Weapon requirement, kg/hm				-	6.5					-	6.5	
HTGR							Not available	4500				
Initial									0	1072 ICC		442
Reload, yr^{-1}									0	645 ICC		184
Discharge, yr^{-1}									0	66 IHC	45 CHG	1070
Weapon Requirement, kg/hm										~1000	14	

[a] Excludes structure.

TABLE 2-19
FISSILE QUANTITIES AND SAFEGUARDS CHARACTERISTICS
20% EU (DENATURED U CYCLE FOR HTGR) (0.2% TAILS)

	Recycle						Throwaway					
			Fissile Material and Separation Requirements [kg/GW(e)]						Fissile Material and Separation Requirements [kg/GW(e)]			
	FCC (mil/kW-hr)	30 Yr U_3O_8 [MT/GW(e)]	U-233 Sep	U-235 Sep	Pu-239 Sep	Ratio Wt FE/FISS	FCC (mil/kW-hr)	30 Yr U_3O_8 [MT/GW(e)]	U-233 Sep	U-235 Sep	Pu-239 Sep	Ratio Wt FE/FISS
LWR	Not available	3900										
Initial			0	2618 ICC	0	30[a]						
Reload, yr^{-1}			378 IHR	360[b] CCC 384[b] ICC	0	23[a]						
Discharge, yr^{-1}			386 IHR	314 IHC	80 CGH	37[a]						
Weapon Requirement, kg, metal			200	260 (15 HEU)	6.5							
LWBR - PBR												
LWBR - BR												
CANDU												
HTGR	18.7	3890					20.9	4570				
Initial			0	1737 ICC	0	272			0	1293 ICC	0	365
Reload, yr^{-1}			135 IHR	503 ICC	0	186			0	638 ICC	0	186
Discharge, yr^{-1}			135 IHR	87 IHC	36 CHG	459			77 IHR	61 IHC	27 CHG	716
Weapon Requirement, kg/hm			130	260	10				1000	1000	14	

[a] Excludes structure.
[b] 360 kg are 93% EU; total 744 kg U-235.

TABLE 2-20
FISSILE QUANTITIES AND SAFEGUARDS CHARACTERISTICS
93% EU (0.2% TAILS)

	Recycle						Throwaway					
			Fissile Material and Separation Requirements [kg/GW(e)]						Fissile Material and Separation Requirements [kg/GW(e)]			
	FCC (mil/kW-hr)	30 Yr U_3O_8 [MT/GW(e)]	U-233 Sep	U-235 Sep	Pu-239 Sep	Ratio Wt FE/FISS	FCC (mil/kW-hr)	30 Yr U_3O_8 [MT/GW(e)]	U-233 Sep	U-235 Sep	Pu-239 Sep	Ratio Wt FE/FISS
LWR	Unknown	2900										
Initial			0	2450 CCC	0	29[a]						
Reload, yr^{-1}			390 CHR	540 CCC	0	25[a]						
Discharge, yr^{-1}			370 CHR	200 CHC	0							
Weapon Requirement, kg,metal			8	17								
LWBR - PBR												
LWBR - BR	Unknown	0										
Initial			4000 CHR	0	0	14[a]						
Reload, yr^{-1}			2000 CHR	200 CHR	0	14[a]						
Discharge, yr^{-1}			2000 CHR	200 CHR	0							
Weapon Requirement, kg/hm			8	Mixed with U-233								
CANDU	Unknown	1019										
Initial			0	5000 CCC	0	Unknown						
Reload, yr^{-1}			2570 CHR	70 CCC	0							
Discharge, yr^{-1}			2570 CHR	Unknown	0							
Weapon Requirement, kg/hm			8	17								
HTGR	15.7	2170					19.6	4100				
Initial			0	1987 CCC	0	239			0	1137 CCC	0	418
Reload, yr^{-1}			321 CHR	273 CCC	0	199			0	565 CCC	0	209
Discharge, yr^{-1}			329 CHR	44 CHC	<1	317			133 CHR	29 CHC	<1	730
Weapon Requirement, kg/hm			8	17					8-13	17		

[a] Excludes structure.

Fissile Material Separation Requirements

Separation Method	Separation Conditions	Weapons Material Handling
C = Chemical	H = Hot (radioactive)	R = Remote (Shielded)
I = Isotopic	C = Cold (Contact)	G = Glove Box
		C = Contact

Total fissile dilution is indicated on the tables by the ratio of total fuel element (FE) weight to fissile material (FISS) weight for each alterantive.

2.2.5 Resource Utilization

Uranium resource utilization potential and fuel cycle flexibility of thermal reactor alternatives has become increasingly important with possible breeder delay and nonproliferation considerations. Tables 2-21, 2-22, and 2-23 list key fuel cycle parameters for the nuclear alternatives under the headings Best Resource, Best Economics, and Best Nonproliferation respectively.

Only the LWBR is included in the best resource comparison, as other fuel cycles are not appropriate for the concept. This system is characterized (Ref. 2-2) by large amounts of U_3O_8 needed for the (assumed LWR) prebreeders to develop the U-233 inventories for the self-sustaining (assuming low fuel recycle losses) breeders. More than 10 reactor years of prebreeder operation are required to provide the initial core inventory for a breeder reactor of the same size.

The pebble bed reactor mass balance and resource requirements for LEU throwaway and fully enriched self-generated U-233 recycle designs were based on the UOT and SFB cycles described by Teuchert, et al., (Ref. 2-26). Moreover, studies of comparable prismatic HTGR and pebble bed reactor cycles have shown that there is little difference in the resource utilization capabilities of the two thermal gas reactor types (Ref. 2-27).

TABLE 2-21
RESOURCE UTILIZATION - BEST RESOURCE

System Characteristics	Coal	LWR	HTGR[a]	PBR[a]	CANDU	LWBR[b,c]
Fuel Cycle Conversion Ratio						
Recycle		0.7	0.84	~0.84	~0.96	1.0
No Recycle		--	0.56	0.58	--	-- [0.8]
System Specific Power, kW(t)/kg		37	35	~45 @ CR=0.84	26	~27
Fuel Processing Loss, %		2.5%	2.5%	2.5%	2.5% (Est)	1.0%
Annual Specific Resource Requirement (0.3% Tails)						
MT U_3O_8/GW(e)		106	46	46=HTGR	12	0 [195]
kg SWU/GW(e) 10^3		78	34	34=HTGR	11	0 [128]
Cumulative 30 Year Specific Resource Requirement (0.3% Tails)						
MT U_3O_8/GW(e)		4180	2440	2440=HTGR	1762	0 [6900]
kg SWU/GW(e)		3100	1810	1810=HTGR	1560	0 [4500]
Discharge Fuel Composition, %						
U-235 in Feed U		20	62	~62	Unknown	Unknown [8.2]
U-233 + U-235 in Bred U		63	80	~80	~85 (Est)	~100 [~100]
Pu in U		--	0.5	~0.5	--	0 [1.2]
Fissile Pu in U		--	3	~3	--	~0 [unknown]

TABLE 2-21 (Continued)

System Characteristics	Coal	LWR	HTGR[a]	PBR[a]	CANDU	LWBR[b,c]
Possibilities of Further Conservation of U Resources						
a) Increased Conversion Ratio		Yes	Yes	Yes	Yes	Small
b) Symbiosis with FBR		Yes	Yes	Yes	Yes	Yes
Dev. Expenses and Risks Due to Transition to Higher Conversion						
a) Dev. Expense		Unknown	Small[d]	Unknown[e]	Large[f]	Unknown
b) Risks		Unknown	None	None	Unkonwn	Unknown
Flexibility of Fuel Cycle						
a) For Pu Use		Yes	Yes	Yes	Yes	NA
b) For LEU Use		NA	Yes	Yes	Yes	NA
c) For LEU-Th Use		Marginal	Yes	Yes	Yes	NA
d) For Changing Conversion Ratio		Yes	Yes	Yes	Yes	Small
e) For Thorium Use		Yes	Yes	Yes	Yes	Yes
Time for Cycle Transition, yr		$\sim$4	$\sim$5	$\sim$3-4 yr One fuel lifetime	$\sim$5	--
Problems Resulting from Transition		Unknown	None	None	Unkonwn	--

(a) Resource requirements for binary cycles in HTGR and PBR would be 16% lower than shown.

(b) Prebreeder information in brackets.

(c) LWBR considered only on best resource basis.

(d) ($\sim$5M$) for thinner coatings on fuel particles.

(e) Requires different ball fabrication process.

(f) Lower specific power.

TABLE 2-22
RESOURCE UTILIZATION - BEST ECONOMICS

System Characteristics	Coal	LWR	HTGR	PBR[a]	CANDU
Fuel Cycle Conversion Ratio	NA				
Recycle		0.60	0.76	0.65 (∿0.76)[b]	--
No Recycle		0.57	0.56	0.58	∿0.7
System Specific Power, kW(t)/kg		37	52	67 @ CR=0.65	23.4
Fuel Cycle Processing Loss, %		2.5%	2.5%	2.5	2.5 (Est)
Annual Specific Resource Requirement (0.3% Tails)					
MT U_3O_8/GW(e)		135	66	93 (∿66)[b]	133
kg SWU/GW(e) 10^3		70	49	82 (∿49)[b]	0
Cumulative 30 Year Specific Resource Requirement (0.3% Tails)					
MT U_3O_8/GW(e)		4580	2650	3100 (∿2650)[b]	5750
Kg SWU/GW(e) 10^3		2370	2000	2650 (∿2000)[b]	0
Discharge Fuel Composition, %					
U-235 in Feed U		0.9	40	18	<0.3
U-233 + U-235 in Bred U		--	73	80	--
Pu in U		1.5	<1	--	0.5
Pu Fissile in Pu		71	20	--	71

TABLE 2-22 (Continued)

System Characteristics	Coal	LWR	HTGR	PBR[a]	CANDU
Possibilities of Further Conservation of U Resources					
a) Increased Conversion Ratio		Yes	Yes	Yes	Yes
b) Symbiosis with FBR		Yes	Yes	Yes	Yes
Dev. Expenses and Risks Due to Transition to Higher Conversion					
a) Dev. Expense		Unknown	None	None	Large
b) Risks		Unknown	None	None	Unknown
Flexibility of Fuel Cycle					
a) For Pu Use		Yes	Yes	Yes	Yes
b) For LEU Use		Yes	Yes	Yes	Yes
c) For LEU-Th Use		Marginal	Yes	Yes	Yes
d) For Changing Conversion Ratio		Yes	Yes	Yes	Yes
e) For Thorium Use		Yes	Yes	Yes	Yes
Time for Cycle Transition, Yr		∿4	∿5	One fuel lifetime	2-3
Problems Resulting from Transition		Unknown	None	None	Unknown

(a) Resource requirements for binary cycles in HTGR and PBR would be 16% lower than shown above.

(b) PBR values from KFA data at CR=0.65. Estimated best economics at CR ∿0.76 as in HTGR.

TABLE 2-23
RESOURCE UTILIZATION - BEST NONPROLIFERATION

System Characteristics	Coal	LWR	HTGR[a]		PBR[a]	CANDU
Fuel Cycle Conversion Ratio						
Recycle		--	0.62		--	--
No Recycle		0.57	0.52		0.58	0.7
System Specific Power, kW(t)/kg		37	71 (Recycle)	95 (Throwaway)	89	23.4
Fuel Cycle Processing Loss, %		2.5	2.5		2.5	2.5 (Est)
Annual Specific Resource Requirement (0.3% Tails)			Recycle LEU	TA	TA	
MT U_3O_8/GW(e)		225	122	181	170	133
kgs SWUs per GWe (10^3) at 0.3% T		102	85	123	102	0
Cumulative Specific Resource Requirement (0.3% Tails)						
MT U_3O_8/GW(e)		7000	4030	5640	5200	5750
kgs SWU/GW(e) 10^3		3140	2795	3820	3100	0
Discharge Fuel Composition (%)						
U-235 in Feed U		0.9	4.1	3.5	1.5	<0.3
U-233 + U-235 in Bred U		--	85	84	--	--
Pu in U		1.0	2	2	1.1% Pu_f	0.5
Fissile Pu in Pu		71	59	51	Unknown	71

TABLE 2-23 (Continued)

System Characteristics	Coal	LWR	HTGR[a]	PBR[a]	CANDU
Possibilities of Further Conservation of U Resources					
a) Increased Conversion Ratio		Yes	Yes	Yes	Yes
b) Symbiosis with FBR		Yes	Yes	Yes	Yes
Dev. Expenses and Risks Due to Transition to Higher Conversion					
a) Dev. Expense		Unknown	Small	Small	Large
b) Risks		Unknown	None	None	Unknown
Flexibility of Fuel Cycle					
a) For Pu Use		NA	NA	NA	NA
b) For LEU Use		Yes	Yes	Yes	Yes
c) For LEU-Th Use		Marginal	Yes	Yes	Yes
d) For Changing Conversion Ratio		Yes	Yes	Yes	Yes
e) For Thorium Use		Marginal	Yes	Yes	Yes
Time for Cycle Transition, Yr		∿4	∿5	One fuel lifetime	2-3
Problems Resulting from Transition		Unknown	None	None	Unknown

(a) Resource Requirements for binary cycles in HTGR and PBR would be 16% lower than shown above.

HTGR fuel cycle data are taken from earlier General Atomic studies (Refs. 2-28 and 2-29) summarizing resource benefits, fuel cycle costs, and flexibility. Best nonproliferation cycles include throwaway and denatured U-233 recycle options.

Since the standard CANDU fuel cycle (Ref. 2-29) employs natural U with spent fuel stowaway, this cycle represents both best economics and best nonproliferation. For resource utilization, a U-Th cycle (Ref. 2-29) with recycle appears best.

A fully enriched U-Th cycle represents the best resource for the LWR (Ref. 2-30). For best nonproliferation, LEU throwaway has been chosen and for best economics, LEU-with-recycle values are tabulated (Ref. 2-28).

2.2.6 Development

Major development requirements, qualitative risk levels, and first unit (lead or demonstration plant) schedules are summarized by Table 2-24 and 2-25.

2.2.7 Unique Applications/Capabilities

In selecting systems for further development and commercial introduction, potential capabilities of the system to satisfy identified future needs must be considered in addition to the intended initial applications. Table 2-26 compares the alternative systems in terms of several applications and capabilities. In this comparison, the HTGRs offer the widest applications capabilities and greatest flexibility.

2.2.8 Maintainability

Total costs, personnel requirements, radiation expense, and plant downtime must be considered in evaluating and comparing maintainability characteristics. Table 2-27 summarizes such available information. For

TABLE 2-24
DEVELOPMENT REQUIREMENTS, THERMAL GAS REACTOR SYSTEMS

Decision Factors	Prismatic TGR			Pebble Bed TGR		
	Steam Cycle	Direct Cycle	Process Heat	HTK-K	HHT	PNP
Principal Development Requirements	FSV operation	He Turbomachine, control valves, PCRV model, vented thermal barrier, PCL loop demonstration	Heat exchanger, materials, thermal barrier	Control and secondary shutdown systems, side and top reflector solution, THTR operation, U.S. licensability (HHT and PNP same component development as prismatic requirements)		
Development Risk:						
Uneconomic Plant	Low	Low	Low	Low		
Cost and Schedule Overrun	Medium	Medium	Medium	Medium to high		
Possible Schedule						
Lead (Demonstration) Plant	1989	1992	1991	Not available		

TABLE 2-25
DEVELOPMENT REQUIREMENTS
NON-TGR SYSTEMS

Decision Factors	Coal	LWR	LWBR	CANDU
Principal Development Requirements	Confirm SO_x removal	Commercial	Confirm high conversion	Tritium reduction
			Fuel performance	U.S. licensability
Development Risk of:				
Uneconomic Plant	Medium	Commercial	High	Medium
Cost and Schedule Overrun	Commercial	Commercial	Medium	Low
Possible Schedule				
Lead Plant Startup	Commercial	Commercial	1992[a]	1990[a]

[a] Ref. 2-2.

TABLE 2-26
UNIQUE APPLICATIONS/CAPABILITIES

	Coal	LWR	LWBR	CANDU	HTGR Steam Cycle	HTGR Gas[a] Turbine	HTGR Process Heat	Pebble Bed Steam Cycle	Pebble Bed Gas[b] Turbine	Pebble Bed Process Heat
Electricity and Low Temperature Process Steam	Medium	Poor	Poor	Poor	Medium	Best	NA	Medium	Medium	NA
High Temperature Superheat Process Steam	Yes	No	No	No	Yes	NA	Yes	Yes	NA	Yes
High Temperature Process Heat	Yes	No	No	No	NA	NA	Yes	NA	NA	Yes
Increased Electricity Generation Efficiency Potential	Medium	Low	Low	Very Low	Medium	High	NA	Medium	Medium	NA
Dry-Cooled Electricity Generation Potential	Medium	Poor	Poor	Very Poor	Medium	Best	NA	Medium	Best	NA
Heat Storage Power Peaking Potential	Good	Medium	Poor	Poor	Good	--	NA	Good	--	NA
Component Technology Applicable to Either LMFBR or GCFR (Symbosis)	NA	No	No	No	Yes	Yes	Yes	Yes	Yes	Yes

Note: NA = Not applicable.

[a] Nonintercooled cycle.

[b] Intercooled cycle.

TABLE 2-27
MAINTAINABILITY

Parameter	Coal	LWR	LWBR	CANDU	HTGR-SC	PBR
Annual Man-Rem Dose Per Unit	Nil	505[2-32]	505[2-33]	500[2-34]	130(a)	(b)
Fuel Cycle and Waste Processing Annual Man-Rem Dose	Nil	480-520[2-35]	302-574[2-35]	(c)	(c)	(c)
Operation and Maintenance Exposure to Toxic Substances or Hazardous Environments (Fatalities per Unit-year)[2-36]	4	<1	(c)	(c)	(c)	(c)
Downtime for Refueling (days per unit-year)						
Predicted		20[2-37]	(c)		28[2-37]	
Experienced	(c)	40-78[2-38]	(c)	(d)	56(e)	(d)
Man-hours per Unit-Year for Maintenance[2-31]						
Forced Outages	11,817	5,846	(c)	(f)	(c)	(c)
Maintenance Outages	7,378	3,594	(c)	(g)	(c)	(g)
Planned Outages	14,702	4,080	(c)	(f)	(c)	(f)
Operation and Maintenance Cost (mills/kWh)	4.4[2-39] 2.0[2-40]	3.3[2-39] 1.7[2-40]	(c)	(g)	1.1[2-41]	(b)

Note: References in brackets []
Footnotes in parentheses ()

(a) General Atomic Company and Stone Webster Engineering Corporation unpublished studies. Breakdown is:

Refueling	20 man-rem/yr
Reactor operations and surveillance	20
NSS maintenance and ISI	20
BOP maintenance	50
Special maintenance	20
	130

(b) Unavailable, but probably similar to HTGR.

(c) Unavailable, but probably similar to LWR.

(d) On-line refueling.

(e) United Engineers & Constructors estimate.

(f) Unavailable, but probably less than LWR.

(g) Unavailable, but probably greater than LWR.

fossil-fueled units of over 600 MW(e) rating and for LWRs, a substantial data base is available (Ref. 2-31). Coal-fired plants probably present more problems than the average fossil unit, including oil-fired stations, because of pulverizer maintenance, furnace slagging, wet coal, and SO_x scrubber problems.

2.2.9 Market

Studies performed at HEDL using the ALPS code offer insight into potential market distribution of LWRs, HTGRs, and breeders under various fuel cycle, resource, and capital cost assumptions. Table 2-28 identifies the reactors and fuel cycles included. Results of these calculations are presented in Table 2-29. Approximate benefits relative to the base alternative can be noted by a comparison of total system costs among alternatives.

2.3 PLANT DESCRIPTIONS

2.3.1 Coal

Typical in size and design of coal plants for the Seventies, the Sherburne generating plant (Ref. 2-42) is a twin 680 MW(e) unit utilizing wet limestone flue scrubbers for emission control of SO_x and particulates. Construction was started on the plant in 1972 and commercial operation for the two units was scheduled for May 1976 and May 1977 respectively. The plant is located approximately 40 miles northwest of Minneapolis, MN.

2.3.2 Fuel Supply

The coal used by the plant is a subbituminous coal from the Coalstrip area of Montana (approximately 800 miles distant). The coal is delivered to the plant on unit trains at the rate of two 100-car trains per day. Each unit train has a capacity of 10,000 tons. Rail trackage accommodates two trains simultaneously on-site, one unloading and one waiting. The trains are unloaded at a rate of 3500 tons per hr (35 cars) providing an on-site turn-around of about 4 hr per train.

TABLE 2-28
DESCRIPTION OF REACTORS AND FUEL CYCLES

Reactor Type	Description	Introduction Date	Power Level [MW(e)]
LW-U5T	Low-enriched U-235/U-238 feed, throwaway fuel cycle	1981	1150
LW-U5L	Low-enriched U-235/U-238 feed	1969	1150
LW-UD	Denatured U-235+U-233 (self-recycle)/U-238 + Th-232 feed	2001	1300
LW-U3D	Denatured U-233/U-238 + Th-232 feed	2001	1300
LW-Pu	Pu/U-238 feed	2001	1150
FBR-U	Advanced oxide LMFBR, 0.300-in. pin, Pu/U-238 feed in core, U-238 in blanket; Br = 1.39, SP = 1.03	2001	800
FBR-Th	Advanced oxide LMFBR, Pu/U-238 feed in core, Th-232 in blanket; BR = 1.39, SP = 1.03	2001	800
HG-U5T	Denatured U-235/U-238 + Th-232 feed, throwaway fuel cycle; C/Th = 650, CR = 0.771/0.412	1991	1344
HG-U5L	Low-enriched U-235/U-238 feed	1991	1344
HG-U5D	Denatured U-235/U-238 + Th-232 feed, optimized for recycle	1991	1344

TABLE 2-28 (Continued)

Reactor Type	Description	Introduction Date	Power Level [MW(e)]
HG-U3D	Denatured U-233/U-238 + Th-232 feed	2001	1344
HG-U5H	High-enriched U-235/Th-232 feed; C/Th = 214/238, CR = 0.829/0.685	2001	1344
HG-U3	U-233/Th-232 feed; C/Th = 150, CR = 0.967/0.877	2001	1344
HG-Pu	Pu/Th-232 feed; C/Th = 650, CR = 0.502/0.627	2001	1344
DC-U5H	High-enriched U-235/Th-232 feed; C/Th = 214/238, CR = 0.829/0.685, direct cycle	2001	1344
DC-U3	U-233/Th-232 feed; C/Th = 150, CR = 0.967/0.877, direct cycle	2001	1344

TABLE 2-29
SUMMARY OF GAS-COOLED REACTOR BENEFIT STUDY RESULTS

Case	Fuel Recycle Option	Scenario	System Cost 1977 to 2037 ($ Billions at 4.5%/7.5%/10% Discount Rate)	Reactors Built [GW(e) from 1969 to 2037]/ Leveled Out Power Cost in 2007 (mills/kW-hr)			Cumulative U_3O_8 Consumed to 2037 (10^6 tons)	U_3O_8 Price in 2037 ($/lb)	Max. SWU to 2037 (10^6 SWU/yr)
				LWR	FBR	HTGR			
1	No recycle, no HEU	Base case	769.1/308.5/168.6	LW-U5T 1944/24.1 LW-U5L 315/25.1	--	--	7.5	270	165
2	Recycle of denatured U-233 in year 2000, no HEU	Base case	758.6/308.2/169.9	LW-U5T 222/23.7 LW-U5L 2038/23.4 LW-UD 0/23.4	--	--	6.7	270	160
3	Full recycle, use of HEU in year 2000	Base case	668.2/287.7/164.4	LW-U5L 1232/18.6 LW-Pu 600/15.9	FBR-U 427/14.8	--	4.2	175	65
4	No recycle, no HEU	HTGR	714.4/293.8/163.3	LW-U5T 471/23.7 LW-U5L 77/24.7	--	HG-U5T 0/21.6 HG-U5L 1712/21.6 HG-U5D 0/22.9	6.5	270	172
5	Recycle of denatured U-233 in year 2000, no HEU	HTGR	695.1/292.4/164.8	LW-U5T 304/21.7 LW-U5L 244/21.6 LW-UD 0/21.5	--	HG-U5L 812/19.5 HG-U5D 646/18.4 HG-U3D 255/18.4	5.5	203	140
6	Full recycle, use of HEU in year 2000	HTGR	645.5/281.6/162.3	LW-U5L 547/18.8 LW-Pu 341/14.2	FBR-U 131/14.8	HG-U5L 197/17.8 HG-U5D 211/17.1 HG-U3D 16/17.2 HG-U5H 423/16.6 HG-U3 293/15.2	4.1	169	30
7	Full recycle, use of HEU in year 2000	HTGR + GCFR	639.3/280.3/162.0	LW-U5L 547/18.2 LW-Pu 193/14.0	FBR-U 5/15.0 FBR-Th 354/13.4	HG-U5L 103/17.3 HG-U5D 204/16.7 HG-U3D 19/16.9 HG-U5H 245/15.4 HG-U3 239/15.2	3.5	152	68
8	Full recycle, use of HEU in year 2000	HTGR including HTGR + Pu	634.1/278.2/160.9	LW-U5L 547/18.2 LW-Pu 26/15.5	FBR-U 117/14.4	HG-U5L 201/17.6 HG-U5D 185/17.0 HG-U3D 14/17.0 HG-U5H 477/15.5 HG-U3 526/13.1 HG-Pu 167/12.9	4.0	169	85

TABLE 2-29 (Continued)

Case	Fuel Recycle Option	Scenario	System Cost 1977 to 2037 ($ Billions at 4.5%/7.5%/10% Discount Rate)	Reactors Built [GW(e) from 1969 to 2037]/ Leveled Out Power Cost in 2007 (mills/kW-hr)			Cumulative U_3O_8 Consumed to 2037 (10^6 tons)	U_3O_8 Price in 2037 ($/lb)	Max. SWU to 2037 (10^6 SWU/yr)
				LWR	FBR	HTGR			
				Capital Cost Sensitivities					
9	Full recycle, use of HEU in year 2000	Case 3 with FBR capital cost factor = 1.75	680.9/290.5/166.2	LW-U5L 1657/19.4 LW-Pu 603/15.8	FBR-U 0/21.5	--	5.1	203	103
10	Full recycle, use of HEU in year 2000	Case 6 with FBR capital cost factor = 1.75	648.3/282.8/162.8	LW-U5L 547/19.4 LW-Pu 336/14.0	FBR-U 0/22.0	HG-U5L 281/18.3 HG-U5D 244/17.5 HG-U3D 16/17.9 HG-U5H 401/16.7 HG-U3 433/16.7	4.3	186	88
11	Full recycle, use of HEU in year 2000	Case 6 with HTGR-GT intro- duced in year 2000 with capital cost factor = 0.9	640.5/280.4/161.9	LW-U5L 547/19.0 LW-Pu 344/13.7	FBR-U 63/14.6	HG-U5L 139/17.9 HG-U5D 192/17.2 HG-U3D 0/17.5 HG-U5H 7/15.5 HG-U3 23/15.5 DC-U5H 507/14.7 DC-U3 438/14.7	4.1	169	82
12	Full recycle, use of HEU in year 2000	Case 6 with HTGR-GT intro- duced in year 2000 with capital cost factor = 0.8	636.3/278.4/161.6	LW-U5L 547/19.2 LW-Pu 342/13.2	FBR-U 35/14.9	HG-U5L 136/18.0 HG-U5D 192/17.2 HG-U3D 0/17.6 HG-U5H 7/15.5 HG-U3 23/15.5 DC-U5H 521/14.0 DC-U3 454/13.9	4.2	169	83
13	Full recycle, use of HEU in year 2000	Case 6 with HTGR capital cost factor = 1.1	653.0/284.0/163.3	LW-U5L 547/18.7 LW-Pu 361/14.5	FBR-U 217/14.9	HG-U5L 167/18.5 HG-U5D 211/17.9 HG-U3D 15/18.0 HG-U5H 382/16.4 HG-U3 357/15.9	4.0	169	72
14	Full recycle, use of HEU in year 2000	Case 6 with HTGR capital cost factor = 1.2	661.2/286.2/164.1	LW-U5L 672/18.5 LW-Pu 422/14.9	FBR-U 253/14.5	HG-U5L 155/19.3 HG-U5D 105/18.7 HG-U3D 3/19.0 HG-U5H 362/17.2 HG-U3 288/17.2	4.0	169	70
15	No recycle, no HEU	Case 4 with HTGR capital cost factor = 1.1	726.8/297.6/164.7	LW-U5T 471/23.7 LW-U5L 77/24.7	--	HG-U5T 0/22.4 HG-U5L 1712/22.3 HG-U5D 0/23.7	6.5	270	172

TABLE 2-29 (Continued)

Case	Fuel Recycle Option	Scenario	System Cost 1977 to 2037 ($ Billions at 4.5%/7.5%/10% Discount Rate)	Reactors Built [GW(e) from 1969 to 2037]/ Leveled Out Power Cost in 2007 (mills/kW-hr)			Cumulative U_3O_8 Consumed to 2037 (10^6 tons)	U_3O_8 Price in 2037 ($/lb)	Max. SWU to 2037 (10^6 SWU/yr)
				LWR	FBR	HTGR			
				Denatured Fuel Cycles Phasing into Pu Recycle, FBRs, and Use of HEU					
16	Recycle of denatured U-233 in year 2000, no HEU	Case 2 with LW-Pu and FBR-U introduced in 2015	728/301.4/167.9	LW-U5T 342/21.9 LW-U5L 1311/21.8 LW-UD 0/21.8 LW-Pu 473/12.2	FBR-U 134/17.5	--	6.0	203	118
17	Recycle of denatured U-233 in year 2000, no HEU	Case 5 with LW-Pu, FBR-U, HG-U5H, HG-UD, and HG-Pu introduced in 2015	672.1/287.9/163.8	LW-U5T 132/21.4 LW-U5L 415/20.5 LW-UD 0/20.9 LW-Pu 48/14.0	FBR-U 167/17.0	HG-U5L 475/19.0 HG-U5D 519/18.1 HG-U3D 211/18.1 HG-U5H 63/16.6 HG-U3 77/16.7 HG-Pu 151/8.6	4.8	203	101

2.3.3 Plant Configuration

The Combustion Engineering controlled circulation steam generators are of the single reheat, balanced draft configurations. The furnace, of the centerwall type, has an overall width of 90 ft, depth of 43 ft and height of approximately 230 ft in each unit. The furnace walls are fabricated from welded panels with outer walls of 2-in. o.d. tubes. The entire unit is a Combustion Engineering standard design. The fuel firing system consists of seven elevations of pulverized coal nozzles arranged for tangential firing. Each elevation consists of eight tilting nozzles, ignited by an eddy-plate igniter using No. 2 fuel oil. The system is designed for NOx control by burner shaping and air draft control. Major plant parameters are summarized in Table 2-30.

2.3.4 Environment Controls

The primary system utilized in the control of air pollutants is the bank of wet limestone scrubbers for removal of SO_2 and particulates. Each unit will have 12 scrubber modules. During operation a limestone slurry is pumped from a reaction tank to underbed spray nozzles. The incoming gas contacts the sprayed slurry and then passes to the bed, which contains glass marbles. SO_2 and particulate matter are removed in the bed and then drained to the reaction tank where the necessary chemical reactions are completed and solids precipitated out. Drainage from the tank goes to the thickener where solid matter is settled out and the clarified water is made available for recirculation. The cleansed flue gas is passed through a de-mister for water droplet removal and is then heated to about 170°F for reduction of stack plume.

A single chimney is used for flue gas dispersion from the two units. The chimney height is 650 ft with a 32-1/2 ft liner diam.

Coal dust control at the point of unloading is accomplished by the use of a 120,000 cfm vacuum collection bag and filter system. Bottom ash, fly

TABLE 2-30
SHERBURNE COUNTY GENERATING PLANT
NORTHERN STATES POWER COMPANY

Maximum Generator Name Plate Rating Designed output	2 x 800,000 kVA @ 24,000 V 2 x 680 MW
Steam Generator, Combustion Engineering, controlled circulation, single reheat, balanced draft	
Primary steam flow	4,985,000 lb/hr 2640 psig 1007°F
Reheat steam flow	4,501,000 lb/hr 1005°F
Furnace dimensions	90 ft wide, 43 ft deep, 230 ft high
Draft system	2 x 5000 HP induced draft fans
Auxiliary Systems	
Flue gas scrubbers	12 per unit total power consumption 5000 HP 50% SO_2 removal, 99% particulate removal
Chimney	650 ft tall, 32-1/2 ft diam.
Coal Usage, Subbituminous	8300 Btu/lb, 0.8% sulfur ∿20,000 tons/day

ash, and scrubber sludge are disposed of in separate diked impound ponds. Five hundred acre ft are provided for bottom ash, and about 2500 acre ft for combined disposal of fly ash and scrubber sludge. Each pond is 50 ft deep and lined with clay soils to prevent exfiltration. The plant site can accommodate additional ponds to more than double the storage capacity.

2.3.5 Light Water Reactor

The light water reactor (LWR) is characterized in this evaluation by the Westinghouse Pressurized Water Reactor (PWR). The Westinghouse PWR is a standardized design in four thermal ratings as shown in Table 2-31. These nuclear steam supply system (NSSS) ratings are obtained by combining two steam generator sizes and two pump sizes in two, three, or four standard heat-transfer loops. The larger steam generators and pumps are combined with extra length fuel assemblies to obtain the higher four-loop rating.

Figure 2-1 shows a four-loop NSSS configuration.

The following is a summary description of a two-loop NSSS. The three- and four-loop NSSS are similar to the two-loop designs with identical reactor coolant loops and similar auxiliary supporting systems.

2.3.5.1 System/Component Descriptions

Reactor Coolant System

The reactor coolant system (RCS) consists of a reactor and two, three, or four closed-reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator. The RCS also contains an electrically heated pressurizer and certain auxiliary systems.

High-pressure water circulates through the reactor core to remove the heat generated by the nuclear reaction. The heated water leaves the reactor

TABLE 2-31
NSSS RATINGS

Thermal rating, MW(t)	1882	2785	3425	3817
No. of loops	2	3	4	4
Reactor vessel diam, in.	132	157	173	173
No. of fuel assemblies	121	157	193	193
Rod array	16x16	17x17	17x17	17x17
Fuel length, in.	144	144	144	164

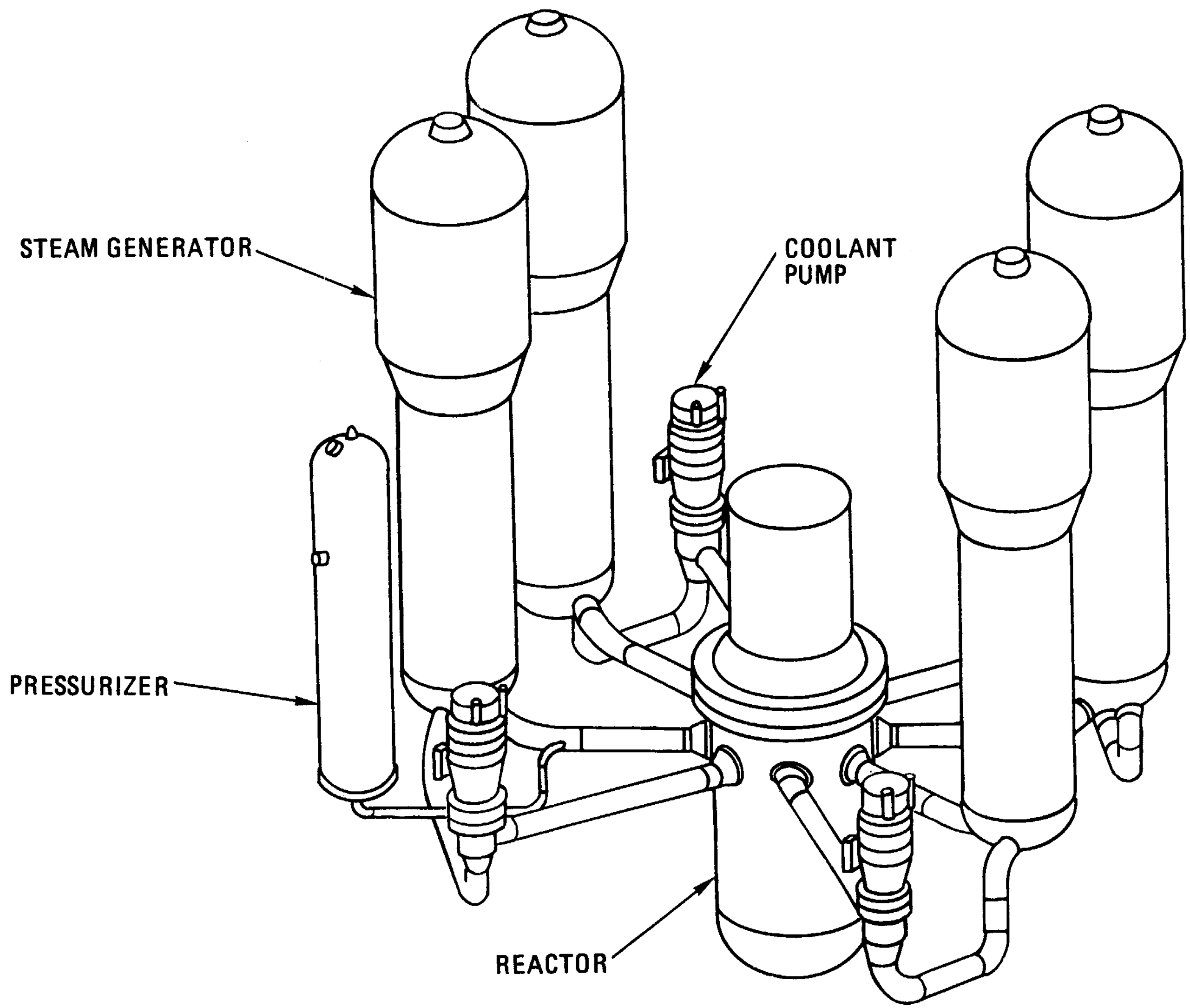

Fig. 2-1. Simplified diagram of a four loop NSSS

vessel and passes via the coolant loop piping to the steam generators. Here it gives up its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel.

Reactor Pressure Vessel

The reactor pressure vessel is cylindrical with a hemispherical bottom head and a flanged and gasketed removable upper head. The vessel contains the core, core support structures, control rod clusters, neutron shield pads, and other parts directly associated with the core. There are inlet and outlet nozzles between the head flange and the core. The vessel is designed and manufactured to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. Principal design parameters are listed in Table 2-32.

The body of the vessel is low-alloy carbon steel. Inside surfaces in contact with coolant are clad with a minimum of 1/8-in. austenitic stainless steel to minimize corrosion.

The vessel is supported by steel pads integral with the coolant nozzles. The pads rest on the steel base plates atop a support structure attached to the concrete foundation.

The core is housed below the nozzles and above the bottom hemispherical head. Shielding pads, attached to the lower core-support barrel assembly in high-flux regions, protect the vessel by attenuating neutron and gamma radiation and some of the fast neutrons that escape from the core, reducing gamma-heating-caused thermal stresses in the vessel.

The removable upper head of the vessel contains a bolting flange employing studs and nuts. Hydraulic tensioning of the studs permits uniform nut loading. An elongation gage is used to assure uniform loading. The optional rapid refueling system utilizes breech block studs for rapid head removal and attachment.

TABLE 2-32
REACTOR VESSEL DATA

Unit Size [MW(t)]	3817 Four-Loop	3425 Four-Loop	2785 Three-Loop	1882 Two-Loop
Inside diam. of shell	173 in.	173 in.	157 in.	132 in.
Overall height of assembled vessel, closure head, and nozzles	43 ft, 10 in.	43 ft, 10 in.	42 ft, 8 in.	39 ft, 1 in.
Radius from center of vessel to nozzle face				
Inlet	10 ft, 11 in.	10 ft, 11 in.	10 ft, 6 in.	9 ft, 7 in.
Outlet	10 ft , 3 in.	10 ft, 3 in.	10 ft, 3 in.	9 ft, 1 in.
Vessel material	Low-alloy steel	Low-alloy steel	Low-alloy steel	Low-alloy steel
Cladding material	Stainless steel	Stainless steel	Stainless steel	Stainless steel
Minimum cladding thickness	1/8 in.	1/8 in.	1/8 in.	1/8 in.
Operating pressure	2250 psia	2250 psia	2250 psia	2250 psia
Design pressure	2500 psia	2500 psia	2500 psia	2500 psia
Design temperature	650°F	650°F	650°F	650°F
CRDM housings	78	78	65	40

The control-rod drive mechanisms (CRDM) are on the upper vessel head. In-core flux measuring instrumentation is in the bottom head.

The thermal insulation surrounding the mechanism adapters and around the flange and studs on the top head must be removable. The insulation enclosure must be drip-proof and resistant to a 2% solution of boric acid. The chloride content of the insulation must be limited to prevent corroding of the stainless steel.

A lifting device handles the vessel head. The lower portion of this device has a platform for access to the control rod drive mechanisms.

2.3.5.2 Materials. The reactor pressure vessel shell and flanges are made of low-alloy steel of Type A 533 Grade B Class I and Type A 508 Class 2 for plate and forgings, respectively. These materials are used because of their strength properties, availability in required sizes and thicknesses, satisfactory service in a neutron and gamma field, and the capability of producing high quality weldments. The materials are also compatible with weld overlay cladding of stainless steels.

All surfaces of the reactor vessel in contact with reactor coolant are either cladded with or made from 300 series stainless steel or Inconel.

Type 304 stainless steel is used for the neutron shield pads, with the cobalt content controlled to a maximum of 0.20%.

2.3.5.3 Steam Generator. The Westinghouse steam generator is of a vertical U-tube design, in use since initial operation in the Yankee generating station in Row, Massachusetts in 1960.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel, and leaves the generator through another bottom nozzle. The inlet and outlet channels are separated by a partition plate. Manways and hand holes permit access to the tubes, the moisture separating equipment, and the primary channel head.

Centrifugal moisture-separators above the tube bundle remove most of the entrained water from the steam. Steam dryers are employed to increase the steam quality to 99.75% (0.25% moisture).

2.3.5.4 Coolant Pump. The standard coolant pump is a vertical, single-stage, shaft seal pump consisting of (from top to bottom) motor, seal assembly, and hydraulic unit. The pump has bottom suction and side discharge through a diffuser. The pump is single speed. A flywheel at the top of the pump increases the rotating inertia and pump coastdown if power is lost. Pawls and a ratchet inside the pump motor prevent its reversing.

Access to the shaft seal system for inspection and maintenance is through a removable spool piece in the pump shafting.

2.3.5.5 Pressurizer. The pressurizer maintains the coolant system pressure during steady-state operation and limits pressure changes during transients. During normal operation 60% of the volume is water and 40% is steam. Load decreases increase the water volume, and sprayers condense a fraction of the steam to reduce pressure. Load increases decrease the water volume, some water is flashed to steam, and immersion heaters limit the pressure reduction.

2.3.5.6 Control Rod Drive Mechanism. Full length control rod clusters are positioned by magnetic latch jack drive mechanisms on the reactor vessel head. These have proved rugged and reliable and are suitable for high-pressure, high-temperature water reactors.

The drive mechanism consists of five major components: 1) pressure housing, 2) operating coil stack, 3) internal latch assembly, 4) position indicator coil stack, and 5) control-rod cluster driveshaft.

All moving components of the mechanism are contained in a stainless steel pressure housing attached to a head adapter. The adapter is welded to the reactor vessel as an integral part of the vessel. The housing is completely free of mechanical seals and has no penetrations for hydraulic and electric lines.

2.3.5.7 Fuel Description, Current Design. The reactor contains Westinghouse rod cluster control fuel assemblies that form a core approximately cylindrical in shape. Pressurized, demineralized light water flows upward through the core, acting as both moderator and coolant.

Temperature coefficients of reactivity, neutron absorbing control rod clusters, and a neutron absorber (boron as boric acid) dissolved in the reactor coolant (chemical shim control) provide reactor control. Chemical shim control has been thoroughly proved and is used in all Westinghouse plants.

Rod cluster control assemblies are used for reactor startup and shutdown, for following load changes, and for controlling small transient changes in reactivity. For chemical shim control, a varying concentration of boric acid during core lifetime compensates for longer term reactivity changes resulting from fuel depletion and fission product accumulation.

All fuel assemblies in the reactor are mechanically identical, although the enrichment of fuel in U-235 within the core varies from region to region. For the initial core, fuel assemblies having the highest enrichment are the periphery of the core. Fuel assemblies with lower enrichments are intermixed in a checkerboard pattern in the central region of the core for a more uniform power distribution throughout the core.

The reactor is refueled on a modified out-in schedule. At refueling, certain fuel assemblies from the central region of the core are removed, fuel assemblies from the periphery of the core are mixed with the remaining fuel assemblies in the central region of the core in a pattern yielding the most uniform power distribution, and new fuel is installed around the core periphery.

The fuel rods, cold-worked, stress relieved Zircaloy-4 tubes, contain enriched uranium dioxide fuel pellets. The void volume within the tube is filled with helium and the rods are internally pressurized.

In those assemblies containing control elements, the individual absorber rods are inserted into the guide thimbles of the fuel assemblies. The long, slender absorber sections of the control rods are relatively free to conform to small misalignments. Tests have shown that the rods are very easily inserted and are not subject to binding. Rod cluster control, which is a standard feature, has been fully developed by Westinghouse.

Magnetic latch control rod drive mechanisms move the control rods. The latches are controlled by three magnetic cores and are so designed that the rod cluster control assembly is released upon a loss of power to the coils and trips the reactor by gravity.

Data and correlations obtained from critical experiments and operating reactors validate the calculational methods employed for the nuclear design of the reactor core. In the thermal-hydraulic design of the core, the maximum fuel and cladding temperatures both during normal reactor operation and at the maximum over-power are conservatively evaluated and are consistent with safe operating limitations. Detailed nuclear parameters are calculated for each phase of operation of the first core cycle and are compared with design limits to show that an adequate margin of safety exists.

The reactor core consists of fuel assemblies containing enriched uranium dioxide pellets in Zircaloy-4 tubes. The assemblies form a lattice that is nearly cylindrical.

The control elements are clusters of cylindrical absorber rods located in guide thimbles forming an integral part of the fuel assemblies. The absorber rods are raised or lowered in the thimbles to change core reactivity.

The individual, replaceable fuel assemblies are held firmly in position between a lower core plate and an upper plate. The core is surrounded by a form-fitting baffle which confines all but a small portion of the upward flow of reactor coolant within the fuel-bearing zone. Surrounding the baffle on the outside is a core barrel. A small amount of coolant is allowed to flow downward between the barrel and the baffle for cooling.

2.3.6 Steam Cycle HTGR

The steam cycle high-temperature gas-cooled reactor (SC-HTGR) (Ref. 2-43) is an advanced power plant now in the late conceptual design phases at GA. The plant described herein represents the current lead plant (LP) 3360 MW(t) reference design for GA's SC-HTGR. This plant can operate on highly enriched uranium/thorium fuel cycles as well as on fuel cycles optimized for resource conservation or proliferation-proof low-enriched uranium without plant modifications. Major plant parameters are listed in Table 2-33.

2.3.6.1 Basic Arrangement. The essential feature of the SC-HTGR is the prestressed concrete reactor vessel (PCRV). The PCRV contains the reactor core and the entire primary coolant system, including steam generators and helium circulators. The PCRV also serves as the primary coolant system pressure boundary and the biological shielding. The basic construction of the PCRV is shown in Figs. 2-2 and 2-3. The vessel consists of a central cylindrical cavity surrounded by thick concrete walls containing six larger and three smaller cylindrical cavities. The reactor core is within the vessel's central cavity. The six larger cavities contain the main helium circulators and the steam generators. The three smaller cavities house the auxiliary circulators and the core auxiliary heat exchangers, which comprise the major components of a core auxiliary cooling system (CACS).

The vessel cavities, interconnecting ducts, and penetrations are lined with a continuously welded carbon-steel membrane or liner. These leak-tight liners prevent the leakage of primary coolant. They also serve as internal forms for the concrete during construction of the PCRV.

TABLE 2-33
SC-HTGR REACTOR AND PLANT PARAMETERS

Core thermal rating, MW(t)	3360
Plant rating, MW(e)	1332
Net plant efficiency	40
Maximum helium temp., °C	716
Maximum helium press., bar	57
No. of primary loops	6
Core concept	Prismatic
Power density	7.0 w/cc
No. of control rod pairs	91
No. of power (shim) rods	91
Pressure vessel	
Type	PCRV
Configuration	Multicavity
Liner concept	Cool, continuously welded carbon-steel membrane (<66°C)
Thermal barrier	Ceramic fibrous and block insulation with cover plates, seal sheets, and multiple retention devices attached to the liner
Circulator concept	Single-stage steam driven (CRH) with single-stage axial flow compressor
Fuel handling system	Off-line hexagonal fuel element, loading and unloading
Auxiliary cooling system	Three loops, 100% pressurized and 50% unpressurized. Electrically driven circulators, pressurized water bayonet heat exchangers
Power conversion cycle	
Working fluid	Steam
Top temp., °C/press., bar	513/169
Reheat concept	Helium reheat
Reheat temp., °C/press., bar	539/40

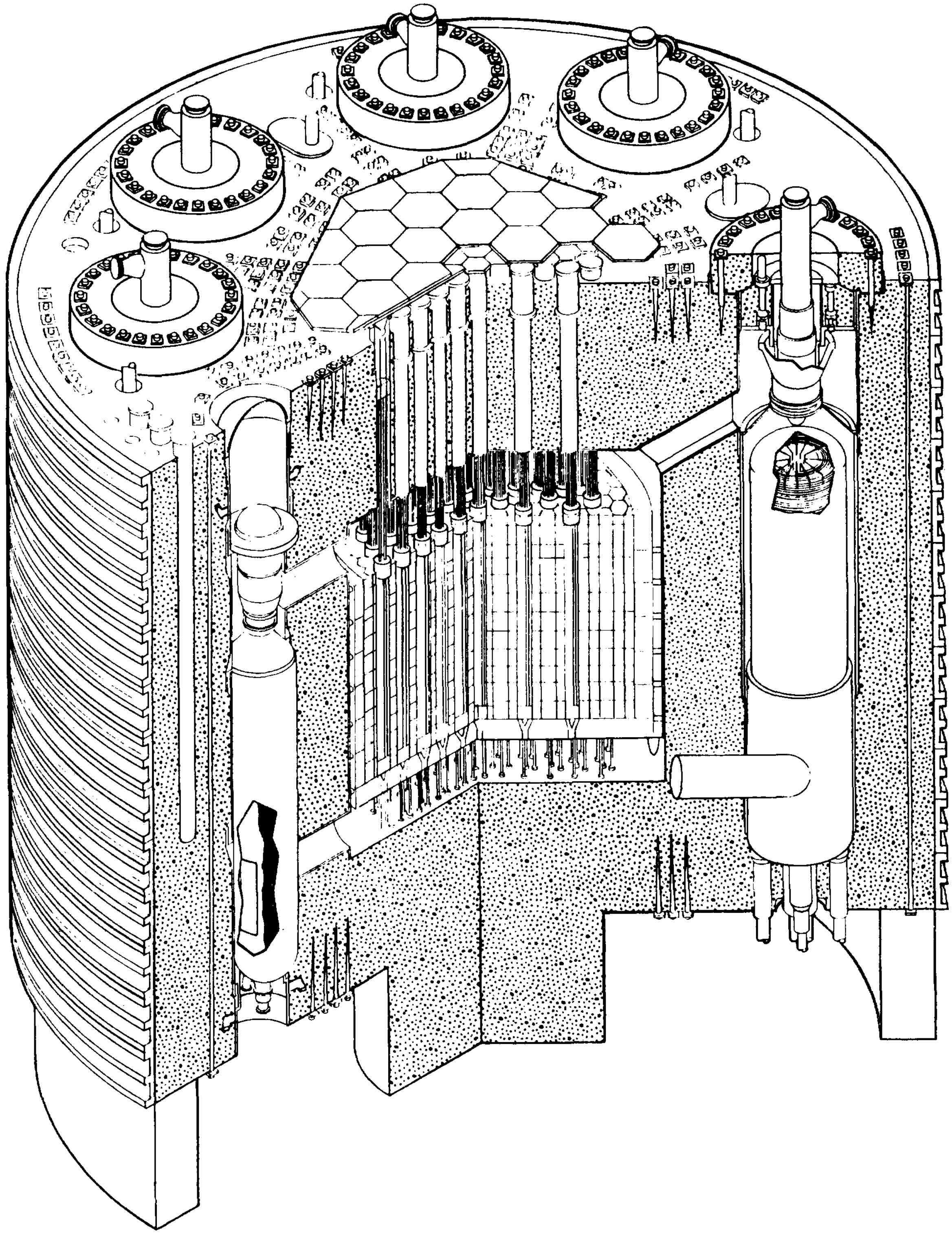

Fig. 2-2. PCRV for large HTGR

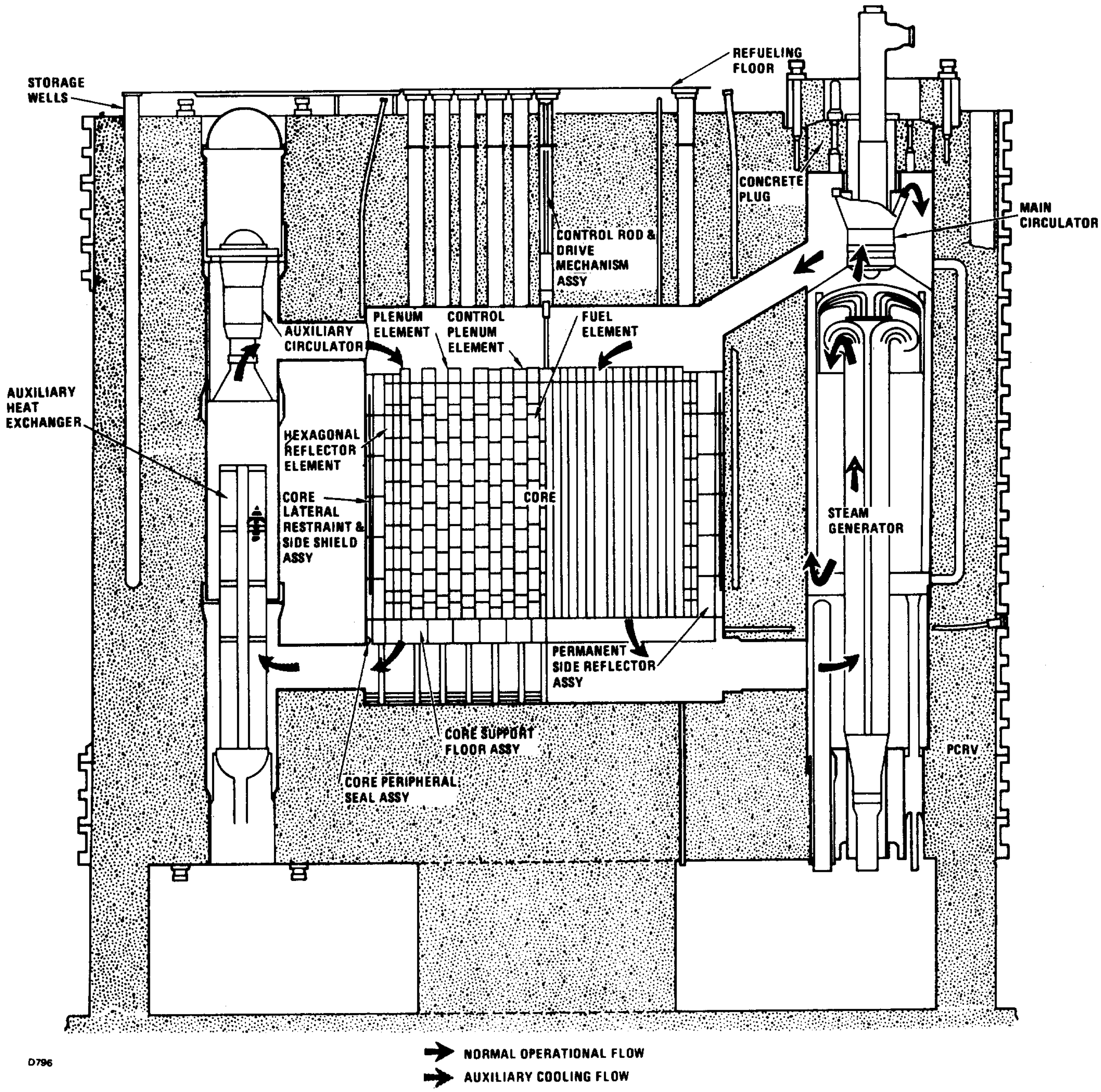

Fig. 2-3. NSSS internal components

The vessel is prestressed so that all major sections intersecting the interior cavities are subjected to a net compressive force under any pressure loading, including the maximum cavity pressure. A post-tensioning stressing method achieves a longitudinal prestress using high strength steel wires or tendons embedded in concrete. The radial prestress is accomplished by circumferential wire-winding.

Ceramic fibrous insulation blankets installed on the helium side of the PCRV liners protect the steel liner from excessive temperature. This thermal barrier is attached, compressed against the liner, and held in place by cover plates, seal sheets, and multiple retention devices attached to the liner.

The top head above the central cavity of the PCRV contains a number of cylindrical penetrations that house control-rod drives and provide access for refueling the core. This head also contains wells that house helium purification equipment, source range instrumentation, and neutron detectors, provide control-rod storage, and shield the reactor pressure relief system.

2.3.6.2 Fuel. The SC-HTGR employs a uranium/thorium fuel cycle, helium cooling, and graphite moderation. The reactor core consists of vertical columns of hexagonal graphite fuel moderator elements and graphite reflector blocks grouped into a cylindrical array and supported by a graphite core support structure. The core is divided into fuel regions, each typically consisting of a central control-rod element column and six surrounding fuel columns. The basic structural material of the core is conventional, nuclear-grade graphite machined in the form of hexagonal blocks. These blocks also serve as the moderator and heat-transfer medium between fuel and coolant.

The fuel materials in the initial core are 93% enriched uranium in carbide form and fertile thorium in oxide form. Initially, U-235 comprises the total fissile loading. Later in the fuel cycle, recycle U-233 may be included as a feed material, replacing much of the U-235. The uranium and thorium particles are coated with layers of pyrolytic carbon and bonded into rods that are loaded into the hexagonal graphite elements. The particle coatings provide the primary barrier for gaseous fission product retention. The fissile (U-235) fuel particles also have a SiC coating for additional retention of metallic fission products and easier separation from fertile material during recovery operations.

2.3.6.3 Control Rods and Drives. Reactivity control is accomplished by 182 control rods operated in pairs by 91 control-rod drives located in refueling penetrations in the top head of the PCRV. Each control rod consists of a series of absorber sections held together by a metal spine passing through the center of the assembly and arranged so that the sections can move in relation to one another. The control-rod drives have electrically-powered winches that raise and lower the control rods by flexible steel cables. Gravitational force acts to insert the control rods into the core following reactor trip. The 91 power (or shim) control rods, which are operated in banks arranged by fuel segment age, provide maneuvering control. The 91 containers of absorber granules, which can be gravity fed into special channels within each control-rod fuel column, provide reserve shutdown capability.

2.3.6.4 Variable Core Orificing. The distribution of coolant flow through the core is controlled by variable orifice valves. These valves regulate the inlet coolant flow to the 91 control regions on the basis of the average temperature at the outlet of these regions, and they ensure that the flow through each region is proportional to the power generated in that region. The orifice valves are adjustable when the system is pressurized and the plant is operating.

2.3.6.5 Helium Circulators. Each of the six primary coolant loops is equipped with a helium circulator. Each circulator unit consists of a single-stage axial flow helium compressor and a single-stage steam turbine drive. The steam turbine normally operates on cold reheat steam from the main turbine. Each circulator is equipped with a water-lubricated bearing system and a helium buffer seal system. The buffer seal system is designed to prevent outleakage of primary coolant.

2.3.6.6 Steam Generators. The main steam module is a once-through design consisting of a helical coil configuration in the economizer-evaporator-superheater (EES) region followed by a straight tube superheater (STSH) in the central core of the module. The reheater, in the lower portion of the steam generator, is of a radial flow design.

2.3.6.7 CACS. A core auxiliary cooling system (CACS) provides an independent means of removing reactor afterheat. This system consists of three independent cooling loops in three separate PCRV cavities. Each loop contains a heat exchanger, a helium circulator, and a helium shutoff valve. A cooling water system outside the PCRV supplies pressurized cooling water to the heat exchangers. The auxiliary circulators are electric-motor-driven axial flow compressors. The electric motor drives are variable speed units. The helium shutoff valves are on the inlet duct of the circulators. The auxiliary heat exchangers are a straight tube bayonet configuration.

2.3.6.8 Major Balance of Plant (BOP) Structures. The reactor containment building is a steel lined, reinforced prestressed concrete cylinder with a hemispherical dome and circular base mat. The building is designed to minimize radioactive fission product leakage and to maintain the minimum containment pressure for adequate operation of the CACS under design basis accident conditions. The design pressure is 58 psig.

The reactor service building and fuel storage building are reinforced concrete structures containing equipment for servicing the reactor, such as control-rod drive storage, radwaste system, fuel handling, fuel storage, inspection, and shipping equipment.

The main steam system conveys steam from the NSSS to the high-pressure (HP) turbine. From the HP turbine the cold reheat steam returns to the NSSS, where it drives the helium circulators. The steam is passed on to the reheater in the NSSS, after which the hot reheat is conveyed to the intermediate-pressure (IP) turbine. The exhaust steam moves to the two low-pressure (LP) turbines, which exhaust to one shell condenser. Some of the exhaust steam from the IP turbine is extracted to drive the steam generator feed pump turbines.

The power is converted by a single full-size, cross-compound, four-flow, 3600 rpm/1800 rpm turbine with 44.0-in. last stage blades. The generator terminal power is 1356.7 MW(e) at a turbine exhaust pressure of 2.5 in. HgA.

The reactor plant cooling water system (RPCWS), in conjunction with the nuclear service water system (NSWS), supplies cooling water to maintain the PCRV concrete temperature within prescribed limits and to provide for heat removal from other reactor plant equipment.

The core auxiliary cooling water system (CACWS) provides a closed-loop supply of cooling water to the core auxiliary heat exchangers (CAHEs) so that heat removed from the primary coolant may be rejected to the atmosphere. Three independent loops are provided, one for each CAHE, and any two are sufficient to cool the plant if the primary coolant system is depressurized; any one is sufficient if the primary coolant system is pressurized. Each loop of the CACWS contains an air-cooled heat exchanger with air flow supplied by six electric-motor-driven fans.

2.3.7 HTR-K, Large Pebble Bed HTR Steam Cycle Plant

The HTR-K is an advanced steam-cycle power plant undergoing conceptual definition in the Federal Republic of Germany. The principal difference between it and the GA 1332 MW(e) HTGR is incorporation of the German-developed pebble bed core. Other major technical departures from existing HTGR designs include proposed use of steam-to-steam reheat and electrically driven circulators. Plant parameters are listed in Table 2-34. Another difference between this plant and the SC-HTGR is the HTR-K's lower core power density of 5.5 MW/m^3.

Although during the early conceptual studies a prestressed cast iron reactor pressure vessel (PCIV) was considered, the concept selected for initial HTR-K development is based on the PCRV, with a primary circuit arrangement quite similar to the SC-HTGR's. However, because of the lower core power density and axially shorter (low height to diam. ratio of $\sim$0.5) core, this PCRV is somewhat larger than the counterpart SC-HTGR's.

TABLE 2-34
HTR-K REACTOR AND PLANT PARAMETERS[a]

Core thermal rating, MW(t)	3000
Plant rating, MW(e)	1200[a]
Net plant efficiency	40[a]
Maximum helium temp., °C	750[a]
Maximum helium pressure, bar	50[a]
No. of primary loops	6
Core concept	Pebble bed
Power density	5.5 w/cc
No. of control rods, in-core	198
No. of control rods, reflector	48
Pressure vessel	
Type	PCRV
Configuration	Multicavity
Liner concept	Cool, continuously welded carbon-steel membrane (<66°C)
Thermal barrier	Ceramic fibrous and block insulation with cover plates, seal sheets, and multiple retention devices attached to the liner
Circulator concept	Electric motor driven compressor
Fuel handling system	On-line sphere loading and unloading
Auxiliary cooling system	Four, 50% cooling loops. Electric drive circulators, pressurized water helical heat exchangers
Power conversion cycle	
Working fluid	Steam
Top temp., °C/press., bar	540/190
Reheat concept	Steam reheat
Reheat temp., °C/press., bar	Not available

[a] Parameters different for final plant design selected for decision evaluation in FRG. (See Section 3.2.1.2.)

Pebble bed cores, consisting essentially of randomly packed spherical fuel elements, are designed to be operated with continuous on-line fuel addition and removal. The core cavity arrangement is shown in Figs. 2-4 and 2-5. Six fuel discharge channels are provided. About 40 fuel input feeders are planned. The FRG-selected fuel management concept for all large pebble bed reactors is the once-through-then-out (OTTO) cycle. Two radial zones flatten the radial power shape, particularly near the side reflectors. A desirable characteristic of the OTTO cycle is that age-peaking (i.e., higher power generation in new fuel), occurs in the top of the core where helium coolant temperature is lowest.

The control and shutdown concept uses 198 in-core control rods and 48 reflector rods. For fast shutdown (scram), about 30 of the in-core rods and all 48 reflector rods will be inserted. Drives for the incore scram rods are long-stroke pneumatically activated cylinders whereas the reflector rods are electrically gravity activated. For the remaining ∿168 (long-term) shutdown rods, the concept is to use mechanically different long-stroke hydraulic cylinder actuation. In-core control rod insertion cores require high thrust actuators for the pebble bed concept. Alternative reserve shutdown systems involving granular or chemical deposition absorber introduction are under consideration should the two independent rod systems offer inadequate diversity.

The six primary system loops each have an integrated, insertable circulator unit using a constant speed electric-drive motor with variable inlet vane flow control. Bearings are lubricated by internal recirculation of lubricating oil from a self-contained integral oil reservoir.

The six steam generator modules are once-through up-hill boilers consisting of helical preheater, evaporator, and presuperheater sections with straight-tube final superheat bundles. Individual tubes can be plugged. The steam generators do not include reheater sections, as the plant uses a steam reheat cycle.

There are four auxiliary cooling loops, each with 50% decay heat removal capacity. Each loop is independent of the others and of other

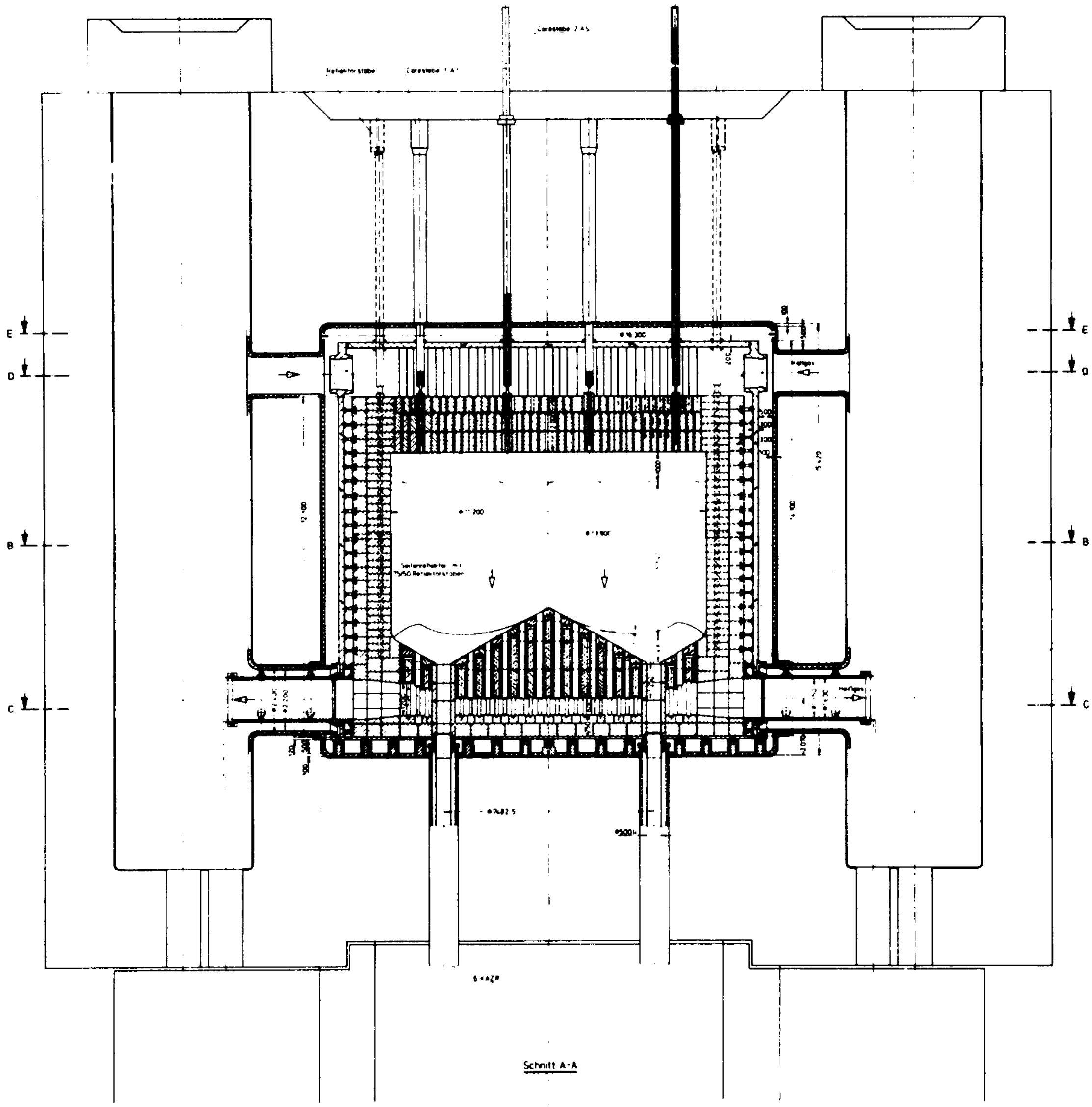

Fig. 2-4. HTR-K PCRV and reactor internals, elevation

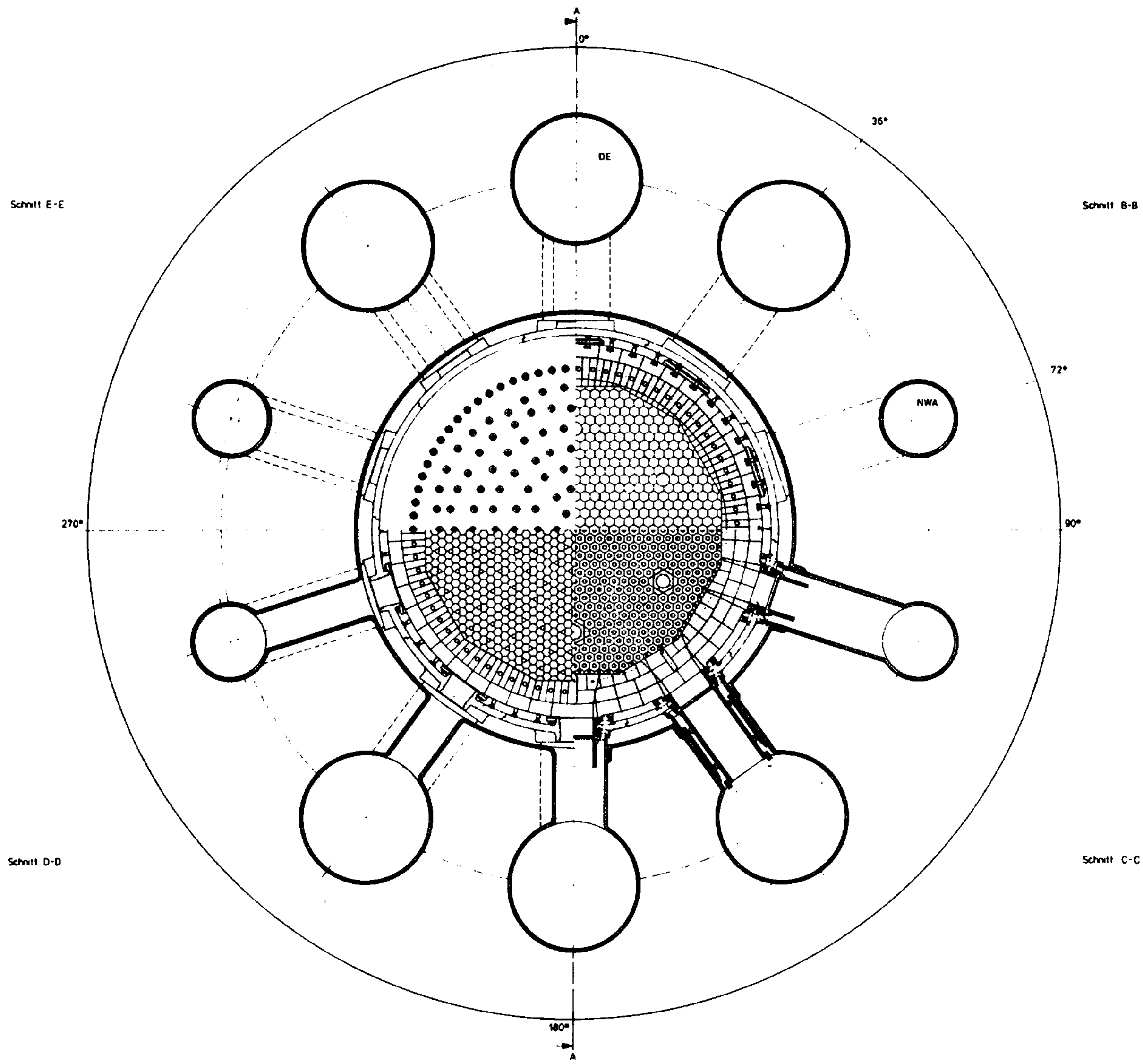

Fig. 2-5. HTR-K PCRV and reactor internals, plan view

operating systems. In addition, there are two emergency feedwater lines with closed-circuit systems that are independent of both the operating and auxiliary systems, including additional electricity supply for main circulators to provide emergency feed of three steam generators each.

2.3.8 Gas Turbine HTGR (GT-HTGR)

The dry-cooled twin 4000 MW(t) gas turbine HTGR (GT-HTGR) is under development at GA (Refs. 2-44 and 2-45). It comprises a graphite moderated core, fueled with fully enriched U-235 and cooled by helium that is circulated by four turbocompressors in separate loops within the primary coolant system. The core comprises rows of hexagonally stacked blocks of graphite containing coolant holes and fuel sticks. The fuel sticks hold coated fissile particles of UC and coated fertile particles of ThO_2 that have been bonded with pitch and fired. The core has a graphite reflector and is contained within a PCRV.

The power conversion loops within the PCRV are illustrated schematically in Fig. 2-6. A nonintercooled recuperative cycle has been chosen to enhance primary system simplicity and efficiencies. The power conversion loops are arranged within the PCRV as shown in Figs. 2-7 and 2-8. Each single-spool, two-bearing unit turbomachine sits inside a horizontal cavity below the centrally located core cavity. Helium is compressed to 1150 psia and then transferred via internal ducting to the tube side of a recuperative heat exchanger (recuperator), which is in a vertical cavity within the PCRV. The heated gas then enters the core cavity where it is further heated to 850°C (1562°F). The hot gas passes through a hot duct to the turbine where it expands to 476 psia, thus producing power, and then proceeds to the shell side of the recuperator. Heat is removed from the expanded helium to heat the high-pressure gas on the tube side. The helium leaves the shell side of the recuperator and passes to the vertical cavity containing the precooler, which is a water-cooled axial counterflow heat exchanger where heat is rejected from the power cycle. The helium then passes to the compressor to begin the cycle again.

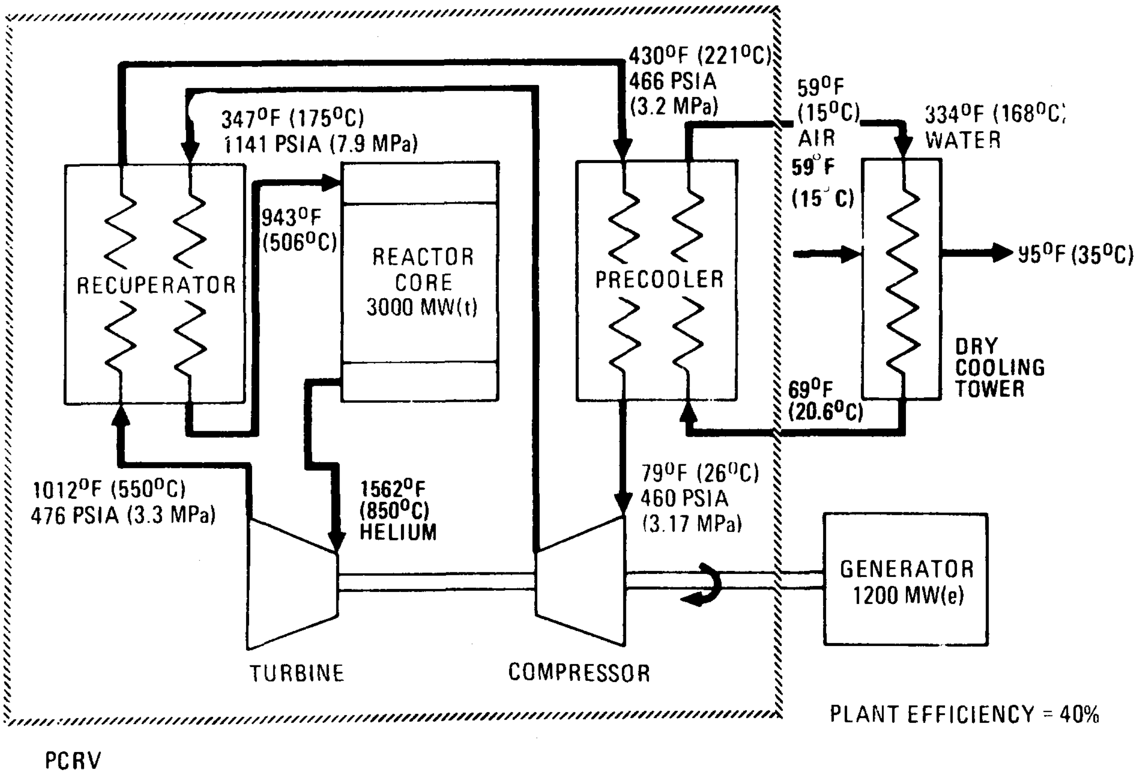

Fig. 2-6. Cycle diagram of GT-HTGR with dry cooling (ISO rating conditions, 1 MPa = 10 bar.)

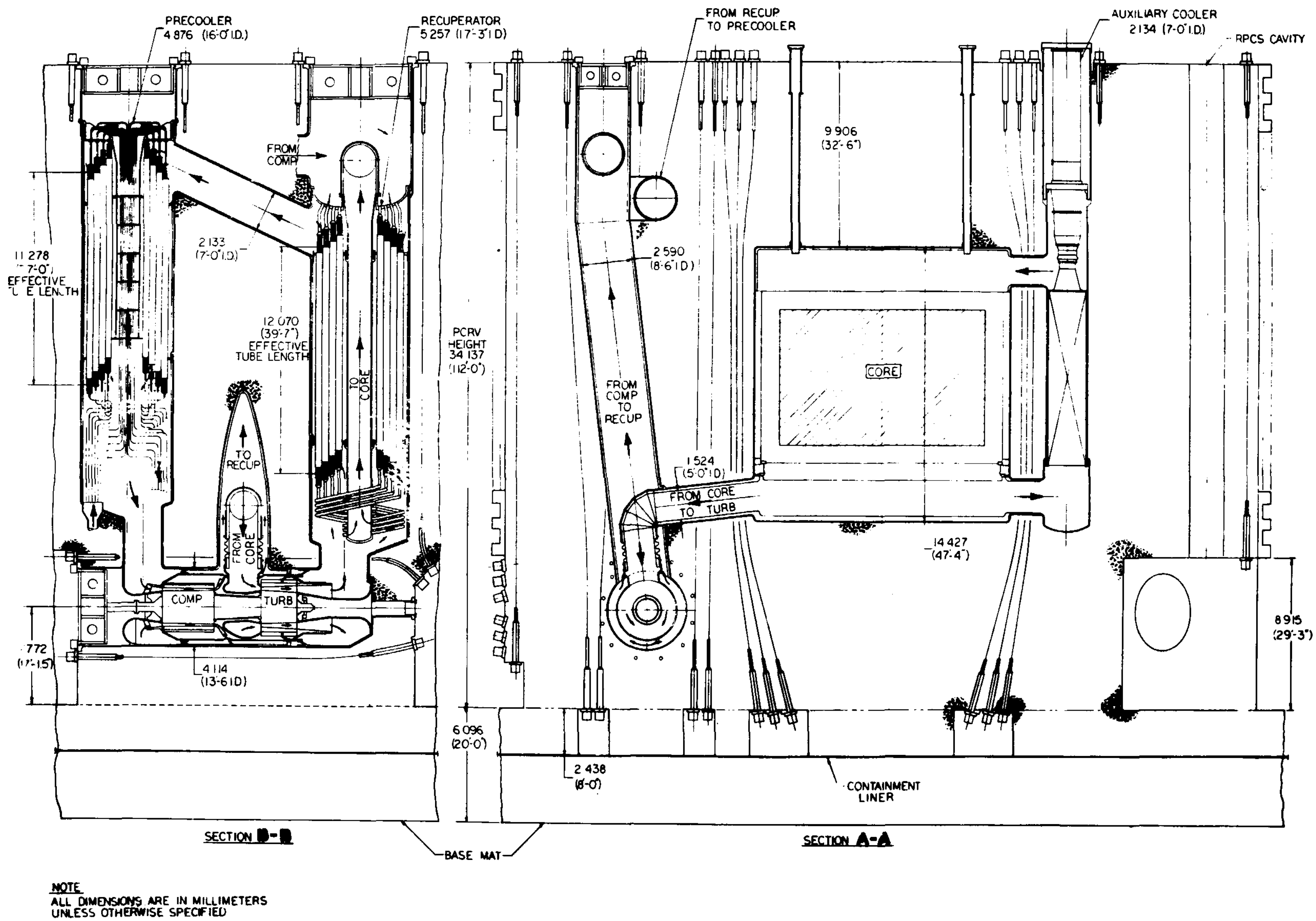

Fig. 2-7. Four-loop, 4000-MW(t), 850°C ROT, single-cycle, dry-cooled GT-HTGR (sheet 1 of 2)

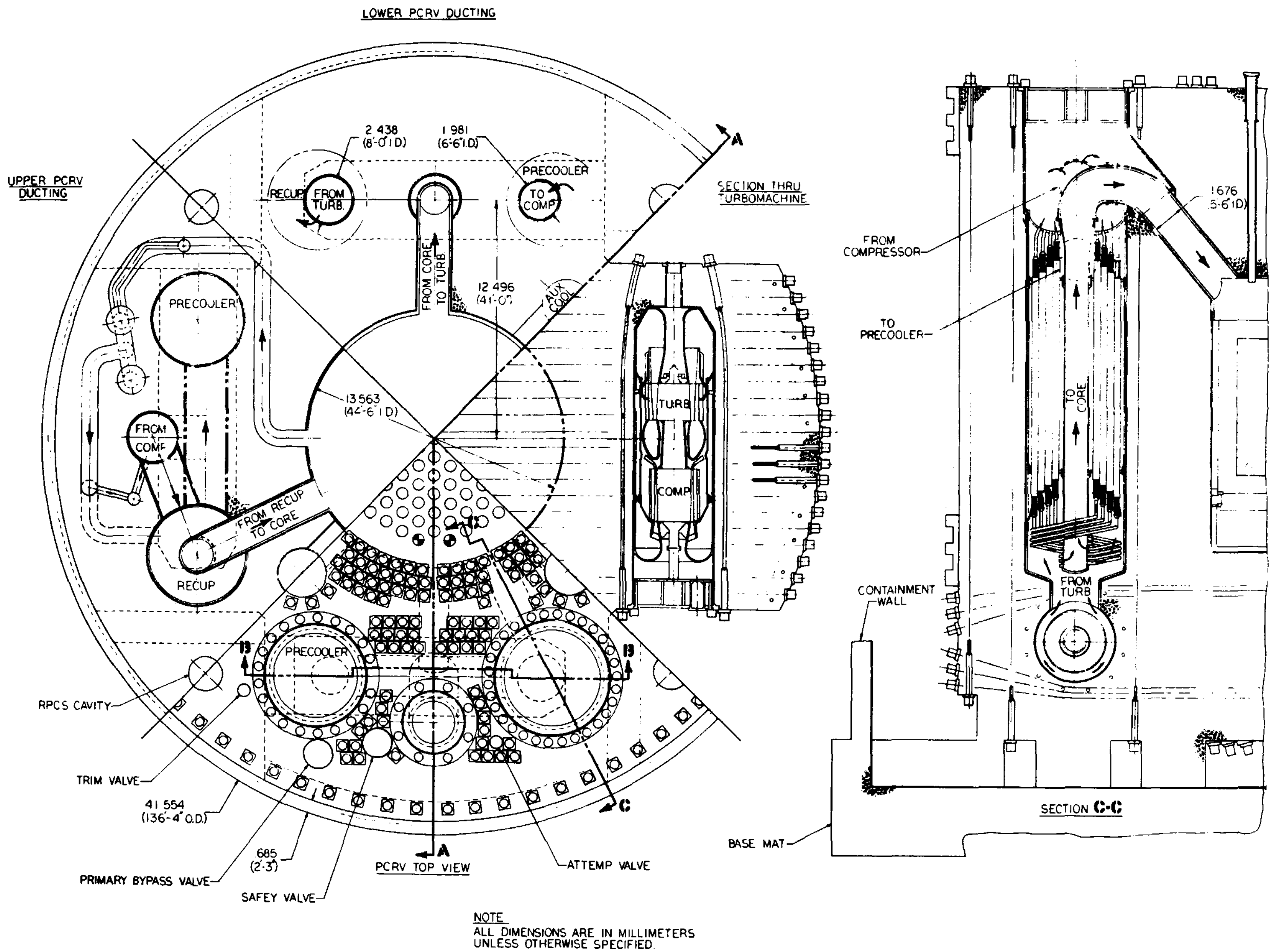

Fig. 2-8. Four-loop, 4000-MW(t), 850°C ROT, single-cycle, dry-cooled GT-HTGR (sheet 2 of 2)

The PCRV comprises a concrete monolith containing all four power conversion loops in an arrangement shown in Fig. 2-8. The PCRV has a circumferential prestressing system from the top of the vessel to a point below the core cavity but above the turbomachine cavities. Diametrical prestressing is used below that point. The axial prestressing throughout the PCRV uses a series of vertical tendons. Thermal barriers and water-cooled steel liners isolate the PCRV from the hot pressurized helium. The system parameters shown in Fig. 2-6 are for a single reactor unit of the twin plant; the performance parameters are given in Table 2-35.

2.3.9 Binary Cycle GT-HTGR

The wet-cooled twin 4000 MW(t) binary cycle GT-HTGR plant is being studied by GA for applications where cooling water is available. Higher efficiency and therefore reduced fuel cycle costs and uranium resource requirements are anticipated for this GT-HTGR option. The reactor uses a typical HTGR graphite moderated core, cooled by helium that is circulated by four turbocompressors in separate loops within the primary coolant system (essentially the same as the dry-cooled plant version).

The power conversion cycle diagram is shown in Fig. 2-9. The non-intercooled recuperative cycle is retained to provide heat rejection temperatures sufficiently high for an economic secondary power cycle.

The principal difference in the primary cycle between the dry-cooled plant and the binary wet-cooled plant is that ammonia working fluid passes through the precooler heat exchanger in the latter, thereby employing the primary cycle heat rejection heat exchanger as the boiler for the rankine secondary power cycle. The supercritical heated ammonia in the precooler is then expanded through twin dual flow ammonia turbines, driving additional electric generators to produce power. The ammonia is then condensed in a pressurized water-cooled shell and tube condenser and is pumped directly back to the precooler inlet. No feed heating is used.

Plant parameters for a binary cycle GT-HTGR are summarized in Table 2-36.

TABLE 2-35
GT-HTGR REACTOR AND PLANT PARAMETERS

Core thermal rating	4000 MW(t) (twin)
Plant rating	3164 MW(e)
Net plant efficiency	39.55%
Maximum helium temp.	850°C
Maximum helium pressure	79.3 bar
No. of primary loops	4
Core concept	Prismatic blocks
Power density	8.4 w/cc (7.1 w/cc in progress)
Pressure vessel	
Type	PCRV
Configuration	Side wall cavity type
Line concept	Water-cooled steel liner
Thermal barrier	Conventional design modified for high venting rates
Circulator concept	Power-producing gas turbine also aids as circulator
Fuel handling	Partial reloads with batch refueling
Power conversion cycle	Brayton (recuperated)
Working fluid	Helium 850°C/79.3 bar
Reheat concept	NA

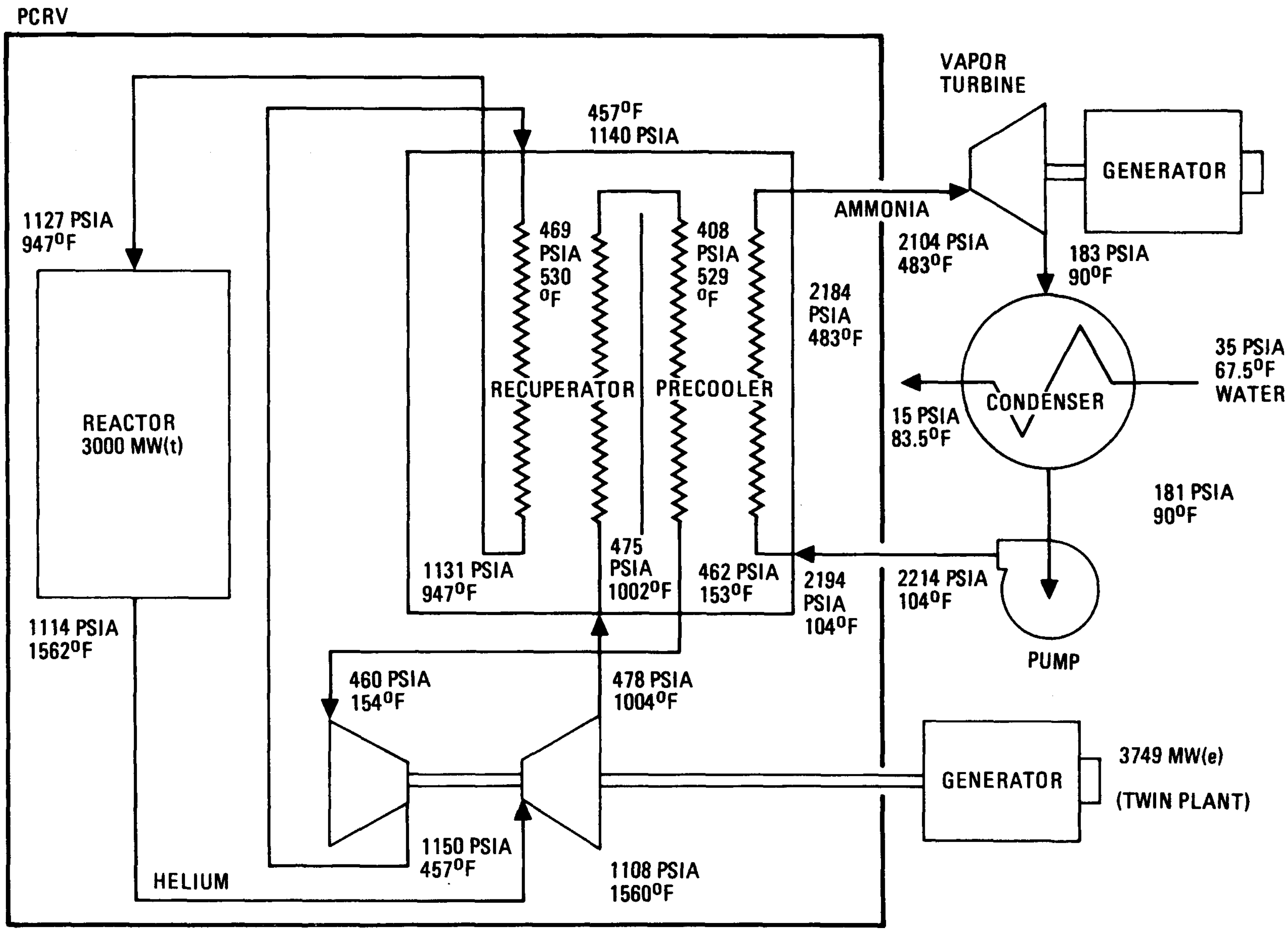

Fig. 2-9. Flow schematic of a binary cycle GT-HTGR

TABLE 2-36
BINARY CYCLE GT-HTGR REACTOR AND PLANT PARAMETERS

Core thermal rating	4000 MW(t) twin
Plant rating	3749 MW(e)
Plant efficiency	46.9%
Maximum helium temp.	850°C
Maximum helium pressure	79.3 bar
No. of primary loops	4
Core concept	Prismatic blocks
Power density	8.4 w/cc (7.1 w/cc in progress)
Pressure vessel	
Type	PCRV
Configuration	Side wall cavity type
Liner concept	Water-cooled steel liner
Thermal barrier	Conventional design modified for high venting rates
Circulator concept	Power-producing gas turbine also acts as circulator
Fuel handling	Partial reloads with batch refueling
Power conversion cycle	Recuperated Brayton primary cycle and supercritical Rankine secondary cycle
Working fluids	Helium 850°C/79.3 bar, Ammonia 250°C/145 bar

2.3.10 Process Heat VHTR

The VHTR designs for process heat are adaptations of the GA HTGR. The HTGR is a thermal reactor using helium as a coolant and having an all-ceramic core composed of thorium and uranium fuel with graphite as a moderator. This combination has enabled the HTGR to develop core outlet temperatures which are much higher than those of other reactor systems. The high temperatures permit steam-methane reforming by transporting the heat from the reactor through an intermediate helium loop to an outside reformer and steam generator. As in other current HTGR designs, all the primary system components are within a PCRV.

2.3.10.1 Intermediate Loop. A flow diagram of the intermediate loop arrangement is shown in Fig. 2-10. The primary reactor coolant, helium, carries heat from the HTGR core to an intermediate helium-helium heat exchanger (IHX). Helium entering the nuclear core at 773 K (932°F) is heated to 1256 K (1800°F). The hot helium is sent to the IHX, where it transfers heat to a secondary helium loop. The primary helium is compressed and returned to the nuclear core. The nuclear core, IHXs, and helium circulators are within a PCRV which contains and shields the primary coolant system. The overall layout of the PCRV and its dimensions are shown in Fig. 2-11.

The secondary helium carries heat to the reformers and steam generators. The hot helium at 1172 K (1650°F), which provides heat for the reforming reaction, is at 932 K (1218°F) as it leaves the reformer. The secondary helium transfers additional heat to the steam generators and is then compressed and returned to the PCRV.

Core designs for VHTR conditions are expected to employ fuel elements of similar design to those used in Fort St. Vrain, which contain a greater number of coolant channels and fuel rod holes than current large steam-cycle HTGR designs. Also, all fuel particles will include a silicon carbide layer for metallic fission product retention at the higher temperature conditions.

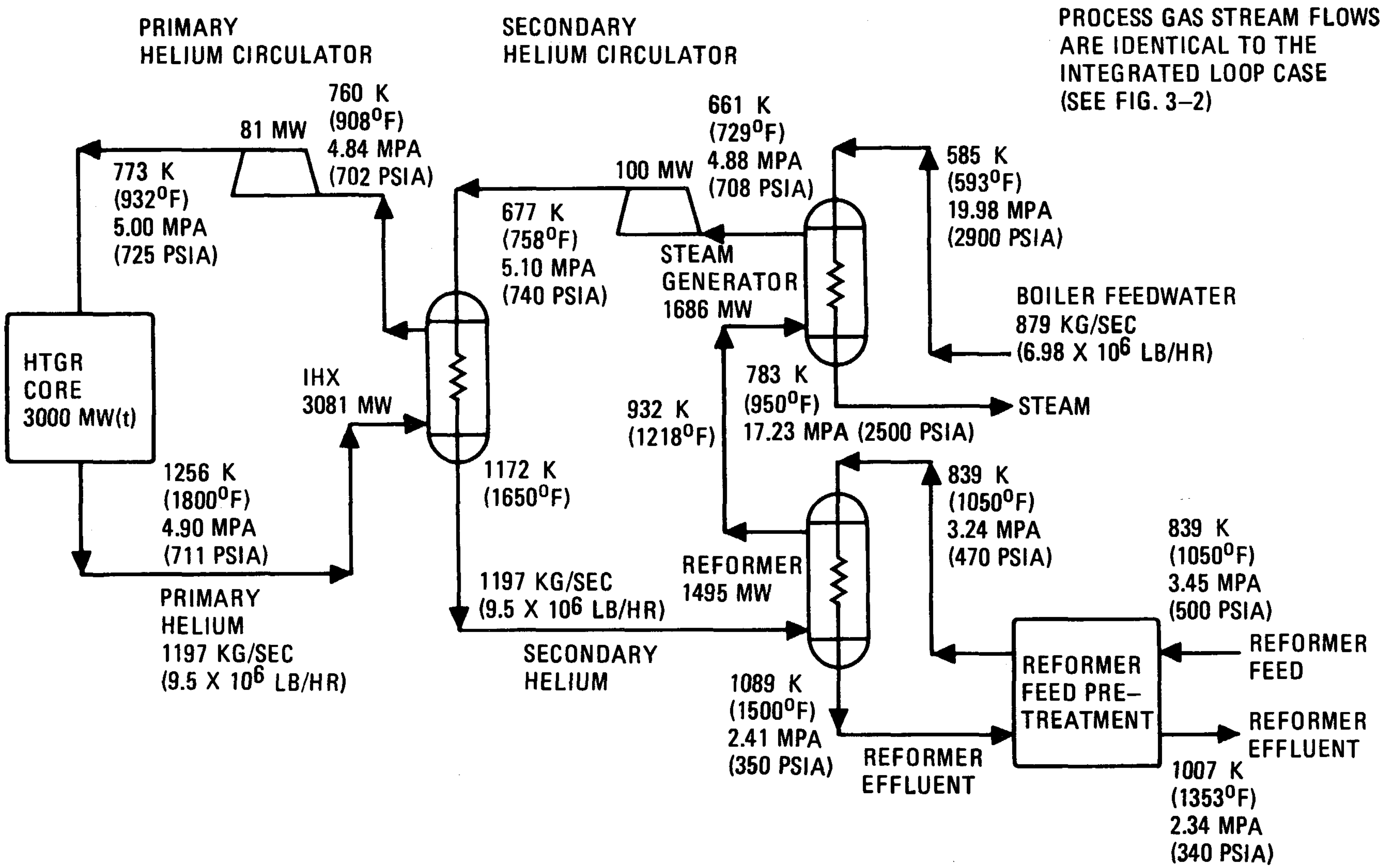

Fig. 2-10. Flow diagram, intermediate loop case

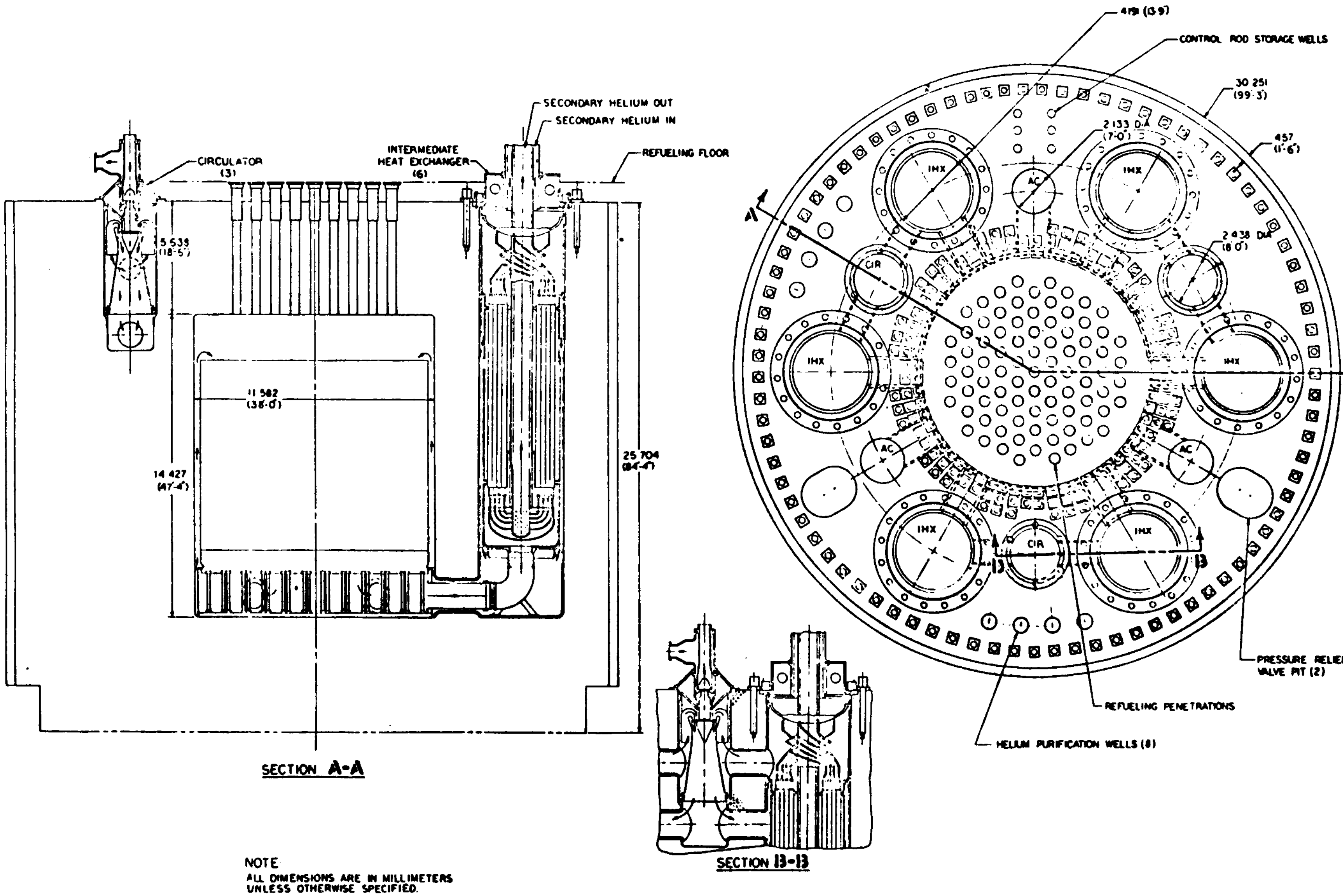

Fig. 2-11. Process heat reactor intermediate loop general arrangement

Most studies involving the use of the process heat VHTR have centered about hydrogen production for process uses. The process most studied has been the manufacturing of substitute natural gas; however, any process which uses hydrogen or a very high temperature, as previously described, can be coupled to the reactor. The integration of the process for making SNG is described here, and this process is the one for which comparisons will be made in later discussions.

Objectives of the process are to:

1. Process all types of coal (except anthracities).

2. Carry out all reactions with coal for fuel conversion using hydrogen.

3. Obtain adequately high hydrogen conversions without methanization.

4. Obtain high carbon conversion without oxygen.

Figure 2-12 presents a simplified version of the steps.

2.3.10.2 Process Conditions. The reformer feed at 839 K (1050°F) is passed through a feed pretreatment system to remove sulfur and adjust the feed composition. The treated process gas enters the reformer at 839 K (1050°F) and is heated to a maximum temperature of 1089 K (1500°F). The reformer effluent temperature of 1089 K (1500°F) was selected because at temperatures much below 1033 K (1400°F), plant performance is severely penalized, and temperatures above 1089 K (1500°F) approach the limitations of present high-temperature materials technology. The reformer pressure and the steam carbon mole ratio choices in the reformer feed were based on considerations of reformer design and performance and overall process heat and process requirements. About 75% of the feed carbon is converted to hydrogen and carbon monoxide by the reforming reaction

$$CH_4 + H_2O \rightarrow 3H_2 + CO \quad ,$$

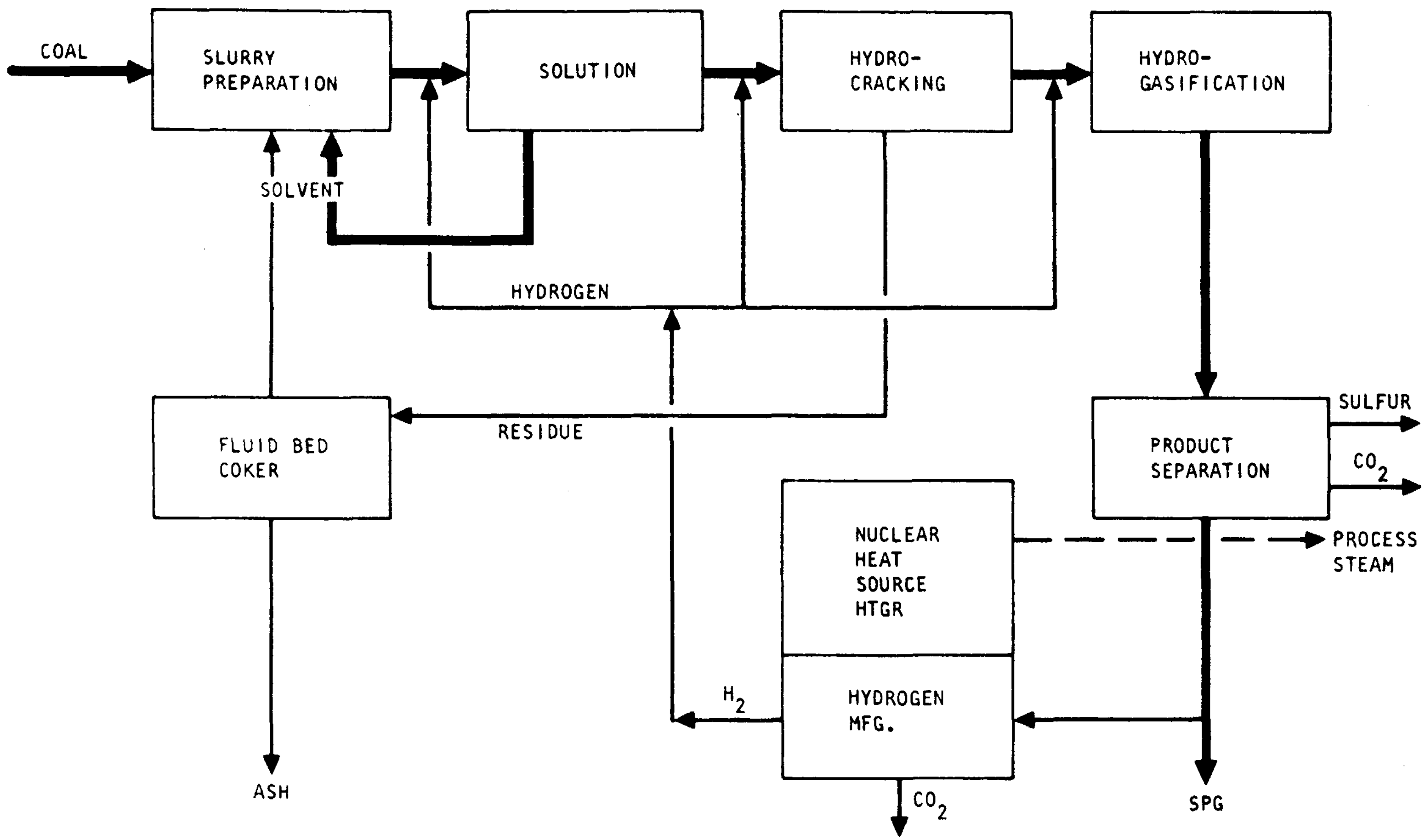

Fig. 2-12. Nuclear coal solution gasification process

and additional hydrogen is formed from the subsequent water gas shift reaction

$$H_2O + CO \rightarrow H_2 \; CO_2 \quad .$$

The reformer effluent is sent to the feed pretreatment system and then to the steam system for additional heat recovery. The integrated and intermediate loop arrangements produce 24.47 kg-mole/sec (1766 MMscfd) of reducing gas (H_2 + CO).

The entire process may be divided into four principal steps: coal preparation, coal solubilizing, hydrogasification, and hydrogen manufacturing.

Coal Preparation

In the coal preparation step, run-of-mine coal containing about 7% moisture is first pulverized to 60% through 200 mesh and then dried to a residual water content of 2%. The remaining water is removed in the slurry preparation step, where hot recycle solvent is added to the coal.

Coal Solubilizing

The coal solubilizing step converts the coal in the slurry into predominantly distillable liquid hydrocarbons. The coal slurry is pumped to a reactor in which the coal molecule is depolymerized in the presence of hydrogen and the product of the reaction separated into three principal streams. Light gases, along with hydrogen, are sent to a hydrocracking area, a proportion of the solvent is separated and returned to slurry preparation, and the remainder of the liquid is sent forward to solution hydrocracking.

For the purpose of this dicsussion, it has been assumed that solution hydrocracking will be carried out catalytically under previously demonstrated conditions. Most other processes involving hydrocracking have been directed toward the production of predominantly liquid products, thus minimizing the production of byproduct gas. These restrictions have limited the maximum temperature employed to between about 800°F (425°C) and 850°F (455°C). As a result, it has been necessary to use pressure above 2,000 psig (13.8 MPa) with highly active catalysts to obtain reasonable reactor kinetics. For this process, however, in which the primary product will be gas, these restrictions may not be necessary.

Solution hydrocracking converts the heavy coal monomers produced in the first step to predominantly distillable liquids, and the products are sent to a separation system. The remaining recycle solvent is separated from the residue of solution hydrocracking. Unconverted carbon, along with some heavy distillates, is sent to a fluid bed coker, and the distillable hydrocarbons are separated into two fractions, those boiling below 400°F (220°C) and those boiling above 400°F (220°C).

Although coal solubilizing was demonstrated many years ago, only recently has advanced catalyst technology and reactor design reduced pressure and catalyst costs to economical levels. Several organizations have announced plans to enter the field of coal liquefaction. A solvent refined coal (SRC) process has been developed by Pittsburg and Midway Company in conjunction with the Office of Coal Research. A 50 ton/day pilot plant is in operation at Tacoma, Washington. Gulf Oil Corporation has a 1 ton/day pilot plant in operation using the Catalytic Coal Liquefaction (CCL) Process.

In the current process, ash is separated by fluid bed coking. Using typical coal liquids, it may be possible to separate the ash by simpler and less expensive techniques.

Hydrogasification

The next process step is reaction of the coal liquids with hydrogen, or hydrogasification. The reaction of hydrogen with distillate petroleum products and whole crudes to produce mixtures of light hydrocarbon gases and aromatic fuels has been carried out in commercial-size equipment. The principal development problem will be to extend this technology to coal liquids.

Hydrogen Manufacture

The hydrogen required by the process is made from a portion of the product SPG by reaction with steam over conventional reforming catalyst. As shown in Fig. 2-12, nuclear energy provides all the heat and process steam required.

2.3.11 CANDU Reactor

The Candian Deuterium-Uranium (CANDU) Reactor System is principally distinguished by the use of natural uranium fuel, heavy water moderator, pressure tube containment, and on-line refueling. Development began with the small experimental ZEEP reactor in 1945. Reference 2-46 describes the reactor's development and Ref. 2-47 gives a brief history. CANDU development has been an independent Canadian effort.

The current development policy, as outlined in Ref. 2-48, calls for concentration on the following standard designs:

- Single 600 MW(e) Units
- 4 x 500 MW(e) stations
- 4 x 800 MW(e) stations
- 4 x 1250 MW(e) stations

Single units in the 500, 800 or 1250 MW(e) sizes could also be considered if there is sufficient utility interest. Long-term programs for plutonium and thorium utilization are also in progress.

2.3.11.1 Multiunit Stations. Ontario Hydro's policy has been to keep the size of individual units conservatively small and to obtain economics of scale by sequential construction of multiunit stations of large size. This policy, which started with fossil plants, was carried over into the nuclear plants, Pickering and Bruce.

The Pickering Station is described in Refs. 2-49 and 2-50 and the Bruce Station in Ref. 2-51. Pickering Stations A and B each have four reactors, with a gross output from each reactor of 540 MW(e).

Bruce Stations A and B are also being constructed with four reactors, each of 800 MW(e). The pressurized heavy water reactor (PHWR) design experience has been largely cumulative because of the similarity among reactor components and design approaches.

For comparison with other reactor types, it is assumed that U.S. utilities would not initially commit themselves to a large four-unit plant, such as the Ontario hydro units. Single or twin stations of the 600, 800, or 1250 MW(e) size appear to be more adaptable to U.S. commercial and licensing conditions.

2.3.11.2 Single Unit Stations. The most acceptable CANDU design for the U.S. market appears to be the single 600 MW(e) plant, four of which are under construction, including Gentilly-2. The Gentilly-2 general arrangement inside the containment is superficially similar to a PWR. The differences are primarily in the reactor. In this unit the heavy water moderator is contained in the shell side of a horizontal cylindrical vessel with flat heads and horizontal tubes. This vessel, called a calandria, is structurally similar to a horizontal return tube boiler but does not operate at any significant pressure or temperature. The moderator D_2O is circulated through an external heat exchanger for cooling. Supported by spacers inside each of the 380 calandria tubes are 103 mm i.d. zirconium-niobium alloy pressure tubes that contain the fuel. These small pressure tubes are the counterparts of the single large pressure vessel in LWRs. These tubes are capped at each end by alloy-steel end fittings and secured by rolled joints. The end fitting is closed by a closure plug and has a

D_2O feed pipe. The flow in the reactor is bidirectional in adjacent fuel channels. In both loops a steam generator and electrically driven pump are proved at each end of the reactor. The pumps have triple shaft seals and efficient leakage recovery. The operating shaft power is 5300 kW. The steam generators are vertical U-tube designs with integral pre-heating sections. The tubing is Incoloy 800 alloy, 0.652 o.d. by 0.044 in nominal wall thickness. Cobalt content is tightly controlled in all materials in contact with primary coolant to minimize radiation fields during maintenance. Primary coolant chemistry is carefully controlled and the purification system is sufficiently large to treat the entire primary system every 30 min.

The natural uranium fuel element is simple in construction, involving only six components. The short length minimizes distortion and the assemblies are easy to handle because of their relatively light weight. A typical fuel bundle for this size reactor contains about 18.5 kg(U), weighs about 21 kg and costs about $1000 including uranium. The nominal power rating of the outer elements is 540 W/cm. Fuel is loaded into and out of the reactor on power to maintain reactivity and power distribution. At equilibrium and 100% power, approximately 110 new fuel bundles per week are required.

Fuel channel operating conditions and fuel specifications for all CANDU plants are given in Table 2-37 (Ref. 2-51). Major plant parameters are listed in Table 2-38.

2.3.12 Light Water Breeder Demonstration Reactor

The light water breeder demonstration plant (LWBR) (Ref. 2-52) is being installed in the Shippingport Atomic Power Station. The two earlier Shippingport applications are PWR Core 1 (PWR-1) and PWR Core 2 (PWR-2).

The LWBR core, fueled with Th and U-233, will be installed in the existing Shippingport reactor vessel. A new reactor vessel closure head

TABLE 2-37
CANADIAN POWER REACTOR FUEL DESIGN AND OPERATING DATA

	Reactor and No. of Elements per Bundle						
	NPD 7	NPD 19	Douglas Point 19	Gentilly 1 BLW 18	Pickering A 28	Bruce A 37	600 MW 37
Elements							
Material	ZIRC-2	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4	ZIRC-4
Outside Diam., mm	25.4	15.25	15.22	19.74	15.19	13.08	13.08
Min. Cladding Thickness, mm	0.64	0.38	0.38	0.49	0.38	0.38	0.38
Bundles							
Length, mm	495.3	495.3	495.3	500.0	495.3	495.3	495.3
Max. Diam, mm	82.04	82.04	81.74	102.41	102.49	102.49	102.49
No. per Channel	9	9	12	10	12	13	12
Pressure Tube							
Min. Inside Diam., mm	82.55	82.55	82.55	103.56	103.38	103.38	103.38
Operating Conditions							
Coolant	D_2O	D_2O	D_2O	H_2O	D_2O	D_2O	D_2O
Nom. Inlet Pressure, MPa	7.9	7.9	10.16	6.32	9.6	10.2	11.09
Nom. Channel Power, MW	0.985	0.985	2.752	3.18	5.43	6.5	6.5
Exit Steam Quality, %	--	--	--	16.5	--	0.8/4.0	$\sim$2.55
Max. Mass Flow/Channel, kg/sec	6.6	6.6	12.6	11.2	23.88	23.81	23.94
Nom. Heat Ratings, kW/m	3.45	2.08	4.0	4.8	4.2	4.55	4.0
Max. Linear Element Power, kW/m	43.4	24.9	50.3	61.2	52.8	57.23	50.9
Max. Surface Heat Flux, kW/m^2	560.7	514.1	1070.0	986.5	1120.0	1393.0	1237.0
Nom. Bundle Power, kW	221.0	221.0	420.0	484.0	636.0	900.0	800.0
Avg. Discharge Bundle Burnup, MWh/kgU	156.0	156.0	190.0	168.0	170/185	196.0	180.0

TABLE 2-38
CANDU REACTOR AND PLANT PARAMETERS

Core thermal rating, MW(t)	2180
Plant rating, MW(e)	685/640
Net plant efficiency	29.4
Max. primary coolant temp., °C	312
Max. coolant pressure, bar	95
No. of primary loops	2
Core concept	Pressure tube calandria
Power density	
Pressure vessel	(a)
Type	None[a] (pressure tubes)
Configuration	Zr-Nb tube, 104 mm i.d. x 4 mm wall
Liner concept	--
Thermal barrier	Steel
Circulator (pump) concept	7050 hp centrifugal pumps
Fuel handling system	On-line
Auxiliary cooling system	Relief valves with containment pressure suppression
Power conversion cycle	
Working fluid	Saturated steam
Top Temp, °C/pressure, bar	258/45
Reheat concept	Steam to steam reheat combined with moisture separation between HP and LP sections
Estimated plant availability, % (Annual basis)	75% [avg for 4 x 500 MW(e) Pickering over 20 reactor-yr. Avg. for Pickering 1-2-3 over last 12 mo. was 93%]

[a] Reactor as a negative pressure concrete containment building.

and control drive mechanism will also be used. The LWBR core will be accommodated in the Shippingport plant without any major plant modification; the basic reactor plant and turbine generator plant will not be changed. However, although not specifically required for the LWBR, the heat exchanger portion of two steam generators in the reactor coolant system will be replaced prior to the installation of the LWBR. During PWR-2 operation, the tubing of these heat exchangers developed minor leaks. The replacement of the two heat exchangers will ensure continued reliable operation of the Shippingport plant. In addition, minor changes to some of the existing Shippingport auxiliary and support systems are required for compatibility with the operation of the LWBR core.

The Shippingport control and instrumentation systems will also be modified to support LWBR operation. New position indication and control equipment will be provided for the LWBR movable fuel control assemblies and associated mechanisms. The nuclear protection system is being changed to provide features appropriate to the LWBR core design. Other minor changes to the reactor plant instrumentation and control systems will also be made.

The major portion of the reactor plant is located below ground level in four large steel chambers connected to form the reactor plant container. The reactor plant container is surrounded by a thick-walled concrete enclosure that serves as radiation shielding.

The operation cycle for the Shippingport reactor and steam plant is shown schematically in Fig. 2-13. The cycle will not be changed for LWBR operation and is basically the same as for all operating PWRs.

The station consists of a pressurized water nuclear reactor and a turbine generator. The principal elements of the reactor plant are the reactor vessel containing the nuclear core, the pressurizer, and the four reactor coolant loops. Each reactor coolant loop contains four coolant stop valves, two in the inlet and two in the outlet piping of the reactor.

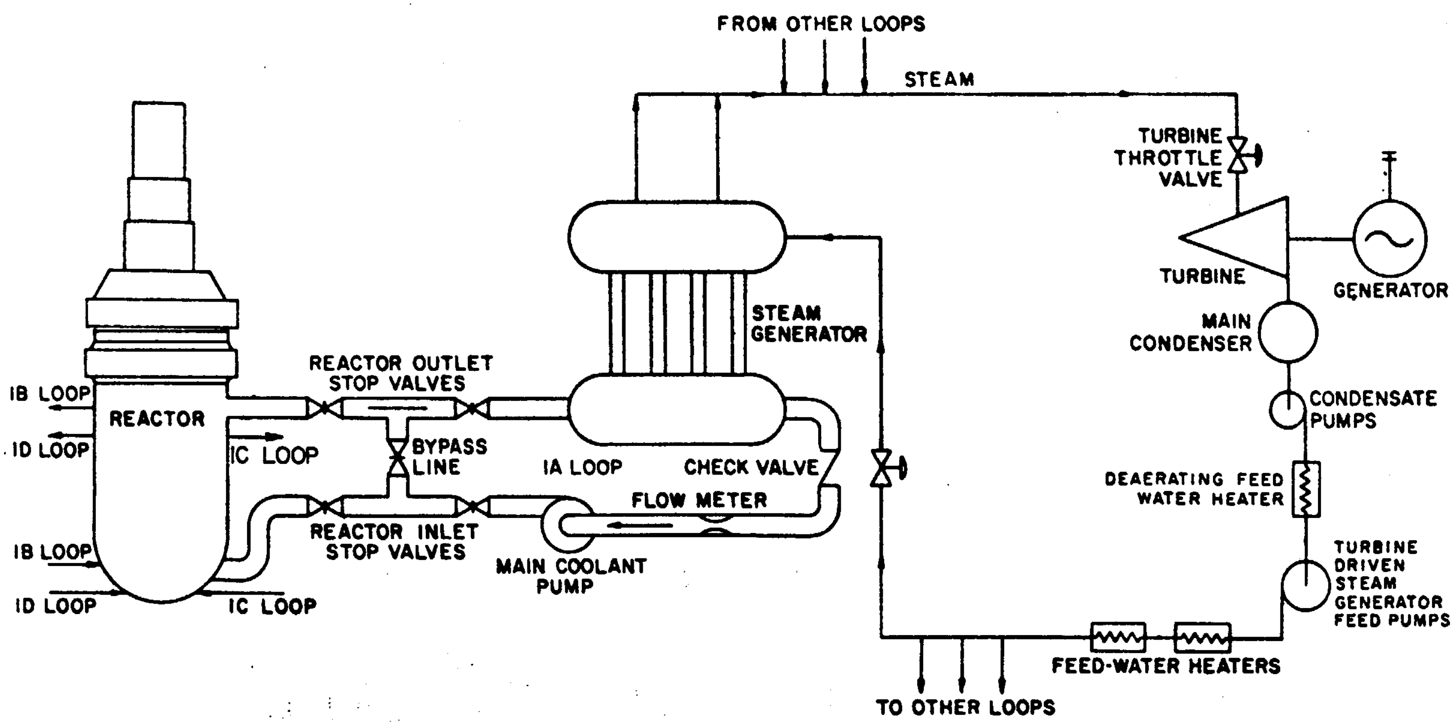

Fig. 2-13. Shippingport plant operating cycle with LWBR core

A vertical single-stage canned-motor centrifugal pump is installed in each reactor coolant loop. Each loop contains a horizontal steam generator consisting of a single heat exchanger, a separate steam drum, and interconnecting risers and downcomers. The reactor coolant pump circulates the coolant water that transfers the core heat from the reactor vessel to the loop steam generator. The heat is transferred from the reactor coolant through the tubes to the secondary side of the heat exchanger to generate steam. The steam produced in the steam generators is then carried by steam lines to the turbine generator unit to generate electrical power.

Figure 2-14 shows the arrangement of the new LWBR reactor core in the Shippingport pressure vessel. Twelve fuel modules, each hexagonal, will be placed into the vessel and structurally held in place. Each of these 12 hexagonal modules contains a central movable seed region surrounded by a stationary blanket region, as shown in Fig. 2-15. Control of the reactor is based upon changing the position of the movable seed assembly relative to the stationary blanket assembly to achieve the proper neutron balance.

Each of the 12 central movable seed regions is attached to and supported by a mechanism on the vessel head. The head provides the cover for the vessel and after installation of the core components will be bolted to the vessel by the ring of bolts shown in Fig. 2-16.

Surrounding the 12 hexagonal fuel modules in the LWBR is an annular region of 15 reflector blanket modules. These modules will not contain U-233 fuel initially, but instead will contain Th.

The LWBR fuel is in cylindrical ceramic pellets that are loaded into circular tubes capped and sealed by welding. The circular tubes are made of Zircaloy-4, which has heat-transfer and corrosion-resistance properties necessary for a reliable performance and a low probability for capturing neutrons. The fuel pellet itself has excellent properties; notably, it is corrosion resistant even if it comes into contact with high-temperature water, and it has a high melting temperature.

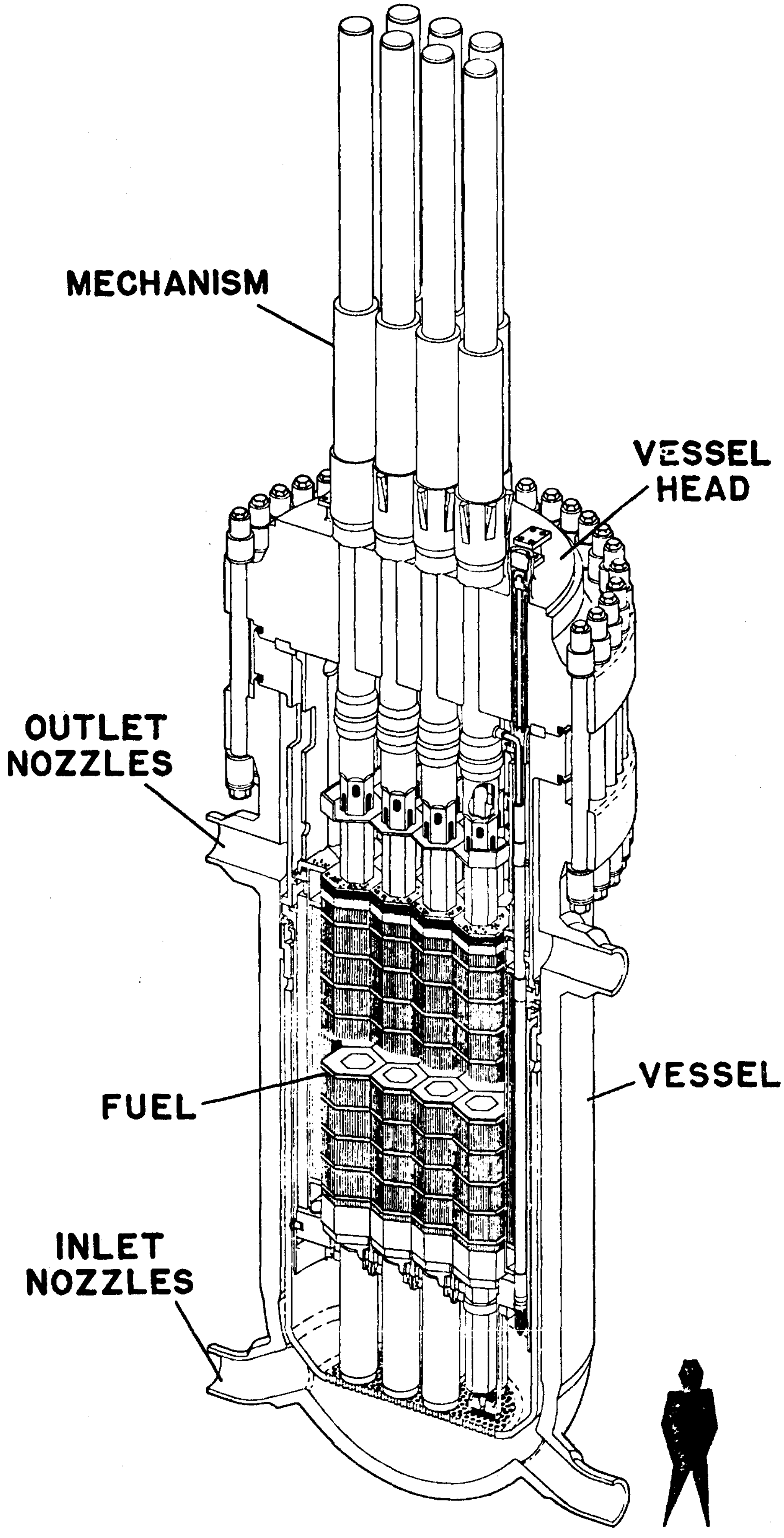

Fig. 2-14. LWBR core in the Shippingport vessel

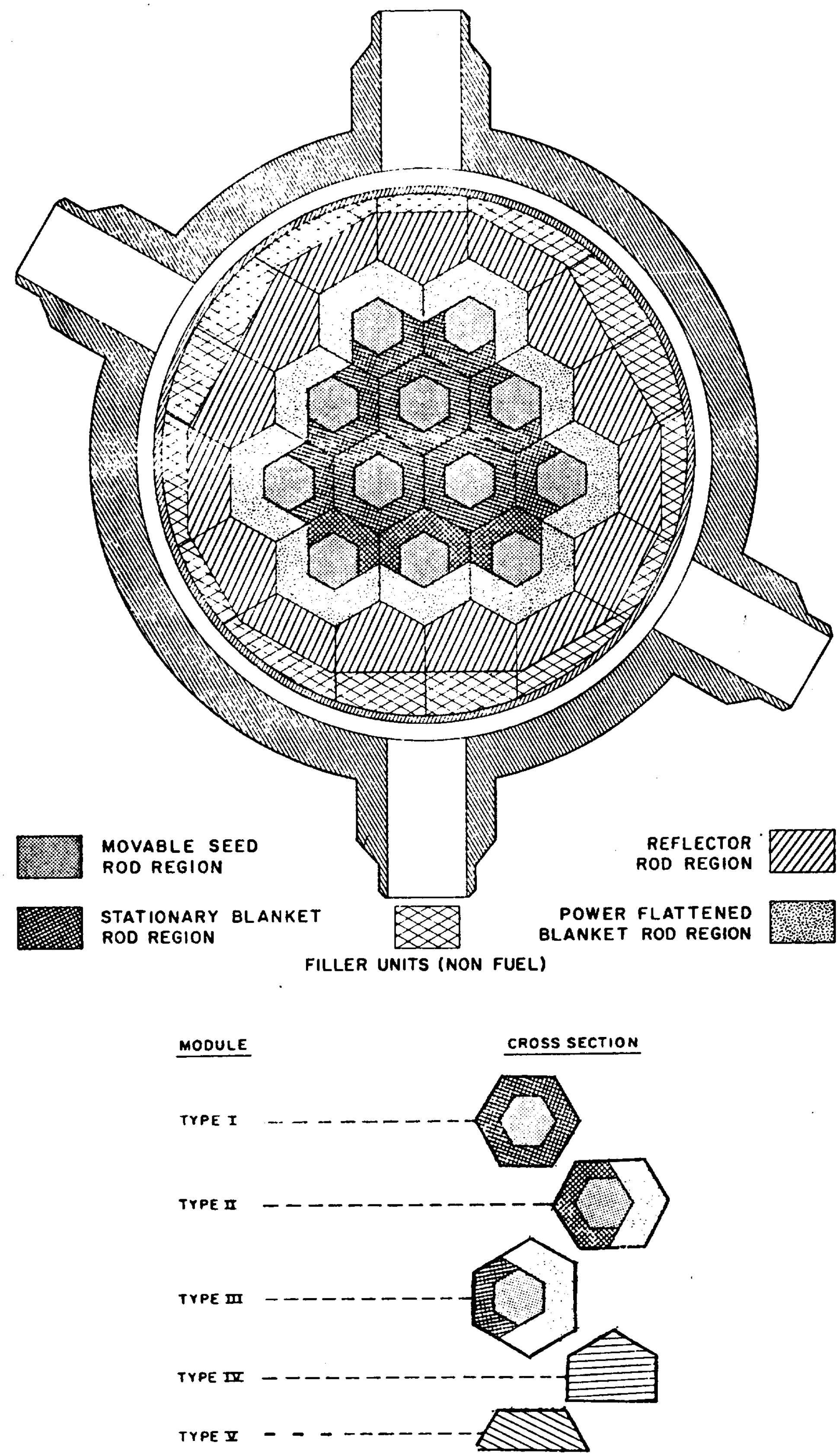

Fig. 2-15. LWBR core cross section

In the seed and blanket regions, the ful pellet contains a mixture of U-233 and Th both in oxide form, i.e., U-233 O_2 and ThO_2. In the seed region, the highest U-233 loading is about 6 wt-% of U-233 O_2 in the ThO_2 (Thoria-urania fuel pellets). In the blanket, the highest U-233 loading is about 3 wt-%.

For the LWBR core, the initial amount of U-233 has been obtained by the irradiation of Th in the Hanford and Savannah River Facilities of ERDA. The U-233 was shipped as uranyl nitrate hexahydrate solution from Hanford and as U-233 O_2 from Savannah River to the Oak Ridge National Laboratory, Oak Ridge, Tennessee. At Oak Ridge the U-233 was converted to U-233 O_2 powder for preparing thoria-urania fuel pellets for the LWBR.

The design of the LWBR evolved from the technology developed in PWR 1 and 2. The control mechanisms used in the LWBR are of the demonstrated basic design previously used for positioning the PWR control elements. However, in the LWBR the use of the PWR neutron-absorbing control elements for reactor control would interfere with the breeding process. Therefore, in the LWBR core the control mechanisms position seed fuel assemblies for reactor control. The movable seed fuel assemblies are slowly raised, inserting more fuel into the core, until the reactor is able to operate at power. The reactor is shut down by lowering the movable seed assemblies. When the seed asesmblies are in the lowest position, the reactor is shut down at the beginning of life and at any time throughout life. Similar movable fuel control systems ave been successfully employed for 20 yr in MTR and ETR materials testing reactors at the Idaho National Engineering Laboratory.

Major LWBR demonstration core and plant parameters are summarized in Table 2-39.

TABLE 2-39
MAJOR CORE AND PLANT PARAMETERS FOR THE LWBR CORE AT SHIPPINGPORT

Power Plant	
Gross electrical output, MW(e)	62[(a)]
Net station output, MW(e)	50[(a)]
Net station heat rate, Btu/kW-hr	13,450
Steam pressure	
Full load at generator, psia	744
No load at generator, psia	895
No. of loops	4
Reactor pressure drop, psi	69.2
Coolant piping. o.d., in.	18
Coolant piping, i.d., in.	15
Coolant velocity, main piping, ft/sec	35
Reactor Core	
Type	Pressurized light water, cooled and moderated seed and blanket
Total reactor heat output, MW(t)	204[(a)]
Total coolant flow rate, 10^6 lb/hr	30.6
Reactor coolant inlet temp. at 236.6 MW(t), °F	520
Reactor coolant outlet temp. at 236.6 MW(t), °F	542
Avg. coolant temp., nom. °F	531
Primary system pressure, nom., psia	2000
Nom. core height, including ThO_2 reflector, ft	10.0
Mean core diam., ft	7.5
Fuel loading (Th and U-233), metric tons	∿42
Lifetime, EFPH	15,000[(a)]
Fuel material	
Movable seed	U-233 O_2 - ThO_2; with ThO_2 end reflectors
Stationary blanket	U-233 O_2 - ThO_2; with ThO_2 end reflectors
Reflector blanket	ThO_2
Fuel cladding material	
Seed, blanket, and reflector	Zircaloy-4, low Hafnium

[(a)]These are the minimum expected performance values for the LWBR operation at Shippingport.

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3. PEBBLE BED HTR-K REVIEW

3.1 SUMMARY

During most of the DOE fiscal year 1977, principal HTR activities within the FRG have been directed toward definition and evaluation of HTR concepts, both for generating electricity and for nuclear process heat applications. For electricity generation, the dual-cycle HTR-K (steam-cycle) and the HHT (direct-cycle gas turbine) were compared and a number of plant configuration and component options evaluated for each. In all instances, the FRG designs are based on the pebble bed fuel concept. System and component designs for the GA prismatic fuel 1160 MW(e) SC-HTGR were used wherever applicable to minimize effort and to provide more developed cost estimates. [The 1160 MW(e) HTGR had previously been under active bid evaluation and initial licensing in FRG.] A groundrule established by the German Technical Ministry (BMFT) is that the electricity generating reactor systems and the process heat reactor concept should be common to the extent practical. Figure 3-1 outlines the range of variants and plant configurations for electricity generation.

During fiscal year 1977, two visits to FRG were made by GA technical teams (Refs. 3-1 and 3-2) to review the large pebble bed HTR design study results and evaluations with the FRG participants. These meetings, along with documents made available by the FRG organizations, provide the basis for the pebble bed reactor review results and descriptions reported herein. As previously stated, objectives are to evaluate the pebble bed reactor plants currently being designed in the FRG for purposes of:

1. Assessing the technical and economic potential for application in the U.S.

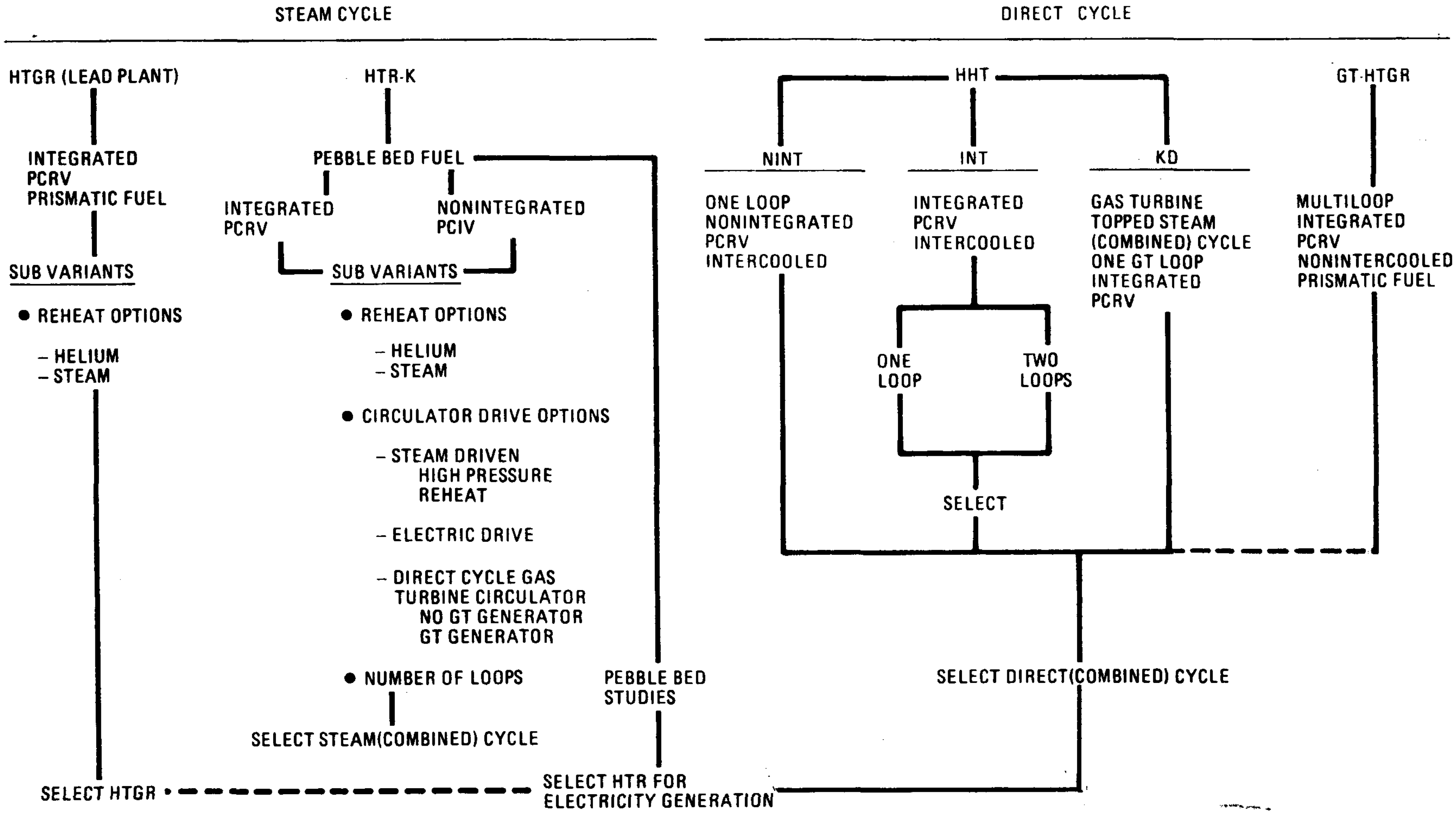

Fig. 3-1. HTR plant concepts and variants; FRG program to select an HTR concept for electricity generation [all variants ∿1200 MW(e)]

2. Providing DOE with technical and program information for decisions regarding cooperative U.S./FRG programs.

Although it is within the scope of this study to review FRG development program plans and economic comparisons of alternative concepts, information available for these evaluations was insufficient during FY-1977.

Primary emphasis in the technical review and evaluation has been on the large pebble bed reactor core and directly related systems. Generally, other primary circuit components and systems are not strongly dependent upon the fuel element concept, either pebble or prismatic, but rather on technical performance, risk, and economic considerations. However, the PCIV concept initially studied as part of the HTR-K plant definition was evaluated and the results of that review (Ref. 3-3) are summarized in this report along with safety and licensing considerations for potential U.S. licensing of an FRG pebble bed plant.

An important conclusion is that the inherent capability, safety, and fuel cycle advantages of the HTR concept can be realized with either a pebble bed-fueled core or prismatic fuel. The principal factors affecting this choice are in reactor design features, technical risks, and development requirements. The unique technical characteristics and design problems associated with the large pebble bed core are discussed below.

Among the various reactor arrangements, the pebble bed conceptually offers a combination of simplicity and high temperature performance potential when helium coolant and HTR graphite fuel systems are employed. Important advantages include the following:

1. Fuel elements are small, of simple spherical geometry, and can be integrally tested in readily provided test reactor facilities.

2. Core geometry definition requires only structural container boundary walls, not individual fuel element positioning restraints or spacers.

3. On-line refueling can be accomplished with relatively simple mechanisms and with only static channels penetrating the primary reactor vessel. However, singularizer mechanisms in the fuel discharge channels must be highly reliable.

4. The multiplicity of interconnected coolant passages serving each individual fuel sphere and throughout the core array tend to minimize blocked channel and hot channel effects.

5. When new fuel is continuously added uniformly over the top core surface and old fuel is similarly removed from the bottom of the core, the once-through-then-out (OTTO) fuel cycle produces the following benefits:

   - Age peaking occurs in the axial direction, resulting in a nearly ideal axial power shape for minimizing fuel temperature in a gas-cooled reactor, since power production is greatest where gas temperature is lowest and decreases as gas temperature increases through the core. Two radial zones are required, however, to flatten the radial power shape.

   - A characteristic of the axial flow-through cycle is that fast neutron exposure to the fuel elements is axially averaged, somewhat reducing the required maximum fuel element exposure while retaining the steep power shape ideal for gas-cooled reactors.

   - Fuel cycle options which result in high age peaking factors can be readily accommodated because of the axial push-through fuel management (e.g., lightly loaded throw-away cycles).

The above characteristics make this concept attractive. However, as organizations within the FRG recognize, there are also some significant engineering problems, especially for higher power density and larger size reactors.

3.1.1 Reflector Design

One principal difficulty is in reflector design. Because the reflectors provide structural containment of the pebble bed core, it is not practical to routinely replace adjacent reflector elements with the core in place. Core removal would involve a major outage with potential fuel reload difficulties. Therefore, removal is not anticipated; adjacent reflectors are expected to remain in place for the entire plant life. Even with the relatively low power density pebble bed core designs (5.5 MW/m^3), 30-yr fast neutron exposure to certain reflector elements is expected to reach $\sim 4 \times 10^{22}$ EDN. For expected operating temperatures, local break-away expansion of the graphite can occur to within radial depths of about 10 cm (HRB estimate). The current reference design solution is to configure adjacent reflector elements in the high fluence regions so that local cracking of the graphite can be accepted and to assure that any pieces falling off will not impede pebble flow but will either disintegrate or be sorted out by the normal fuel sphere handling devices. The fuel handling system includes capability for sorting and removal of failed sphere fragments.

For backup, the designs will provide for replacement capability of at least the upper one or two meters of adjacent side reflector. The approach is to conceptually define temporarily insertable dam equipment capable of exposing local sections of side reflector to a depth of about 2 m for inspection and replacement-in-principal capability.

3.1.2 Reactor Control

Another principal difficulty is associated with reactivity control, in that defined channels for the insertion and removal of control absorber cannot be practically provided in a pebble bed HTR core.

Pebble bed reactors up to possibly several 10s of MW(t) might be adequately controlled by absorber rods operating within channels in the side reflector blocks, similar to the AVR, 45 MW(t). However, for virtually any major reactor applications of economic interest, in-core control rods are required, at least for shutdown. Current German designs employ high thrust-force rod drives, requiring approximately 28 KN (6300 lb) for a single rod penetration to a depth of about 4 m when accompanied by ammonia injection to reduce friction between the graphite fuel spheres during insertion. Predicted force for insertion of a control rod to a depth of 5 m in THTR is over 8000 lb(f). Without ammonia, insertion force approaches 20,000 lb(f).

An optional rotating control rod design concept that avoids the need for ammonia and high axial thrust force is under consideration. A shallow, helical screw thread configuration with a pitch on the order of one fuel ball diameter is required on at least the leading portion of the rod. Upward displacement of fuel spheres surrounding the rotary rod has been observed in experimental studies.

To satisfy safety criteria for two independent shutdown systems, the reference HTR-K design employs a scram rod group (about 42 in-core and 24 side-reflector rods) capable of holding the core subcritical for at least 30 min. The remaining 156 in-core rods will have a different kind of drive mechanism for system independence and, combined with 24 additional reflector rods, will be capable of long-term shutdown.

Should a completely diverse reserve shutdown system be required, a concept using small absorber spheres (KLAC system) is under evaluation. The small spheres penetrate the bed but are sized so that some are trapped in certain interstices within the bed. Absorber retention during a seismic event is a concern, but adequate removal of these spheres for reactor startup following KLAC insertion is also in doubt.

Another conceptual possibility being investigated for reserve shutdown is the injection into the core of an aqueous gadolimium (Gd) compound. The Gd absorber should deposit on the fuel spheres in the upper core region.

Pulling rods and perhaps adding new fuel to achieve criticality provides for burnup of the Gd burnable poison as the plant is brought back to power.

3.1.3 Core Performance

Core performance also presents difficulties. Coolant flow orificing to compensate for radial power differences is not practical with the pebble bed core. Neither is it possible to provide in-core flux or temperature sensors for confirmation of reactor power or power-to-flow conditions within the active core. Azimuthal xenon oscillations may be difficult to detect.

Core thermal-flow limitations are mitigated to a large extent by axial push-through fuel management capability of the pebble bed reactor, and preliminary analyses confirm that local peak fuel temperature is lower in the pebble bed core than in a comparable prismatic core design. However, uncertainties remain, pending experience with large pebble bed core, as to whether power distribution and power-flow relationships can be adequately determined and monitored.

In the large pebble bed core designs, core height-to-diameter and power density are typically constrained by control rod insertion depth and side reflector dose respectively. Therefore, the inherently higher core pressure loss characteristic of the pebble bed may not be a significant limitation. The axial push-through fuel cycle allows a relatively large core temperature rise (and therefore, lower helium flow rate) without exceeding fuel temperature limitations.

3.1.4 Core Discharge

Complete core discharge is another area of concern. In the event that complete core unloading is required (e.g., for bottom reflector inspection or repair) reloading will require special equipment and procedures to avoid bottom reflector damage during initial reloading. Also, reloading of the removed fuel may not be practical with OTTO cycle fuel management.

3.1.5 Core Support

Finally, core support presents some difficulties. The core support and side reflector structures must contain the pebble bed core and resist weight (seismic), pressure drop, and control rod insertion forces throughout operating temperatures, without local gap accumulation which would allow fuel spheres (or large fragments) to become lodged between structural elements. Initial model tests of the current reference 3000 MW(t) design (similar to the THTR arrangement) indicate maximum possible gap accummulation of about 30 mm. Fuel element sphere diameter is 60 mm. However, the entire array will be under compression from radial spring-packs between the side reflector and thermal shield. An alternative conceptual design has been developed within the process heat (PNP) project for a support structure and side reflector system with less gap accumulation potential. However, the reference THTR design is being retained and model testing continues to qualify the concept for larger core dimensions.

Small shutdown absorber spheres (KLAC concept) may not be usable if support system gaps allow potential trapping of the absorber spheres between bottom reflector and support elements. KLAC spheres are on the order of 10 mm diameter.

Many of these technical problems with the pebble bed concept can be aggravated as core size (rating) increases. For example, up to a few 10s of MW(t) reflector rods might provide adequate shutdown margin. A short-term hot subcritical condition appears achievable for large OTTO cycle plants if side reflector rods are inserted and all in-core rods are brought through the top reflector and upper void to the top of the core. However, bed penetration is required for cold and long-term subcritical condition. Core support design, confidence of power/flow range, and core instrumentation capabilities are all thought to improve in smaller core sizes.

However, fast neutron exposure problems regarding the side and top reflector may not greatly improve as size is reduced. Power density and power distribution within the core primarily determine maximum reflector exposure. Assuming that the OTTO cycle is retained to achieve maximum outlet gas temperature capability, reflector dose is essentially defined by power density. Boundary flux suppression, unfueled ball curtain concepts, etc., generally do not appear attractive for reducing exposure; rather, these approaches appear less attractive as reactor size is reduced.

In summary, considering the foregoing advantages and technical uncertainties, the results of this review are that:

1. The pebble bed concept with on-line refueling and OTTO cycle offers the potential for very high helium temperature with relatively low maximum fuel temperature. Uncertainties in core performance increase with extrapolation to large core size.

2. Mechanical design requirements, i.e., reflector, core support, and control rod components, pose some of the most obvious problems of the large pebble bed reactor. However, potential solutions with backup approaches have been defined by the FRG participants for all identified problems. Development details and economic impacts of ultimate solutions are yet to be determined.

3. Fuel cycle flexibility, resource consumption, and fuel cycle cost advantages for the pebble bed HTR are essentially the same as those for the prismatic fuel HTR.

3.2 DUAL-CYCLE HTR-K PLANT

Dual-cycle HTR-K concept design studies were heavily emphasized in the FRG during 1976 and early 1977 to achieve definition of an advanced large pebble bed steam-cycle plant employing a PCIV. Expected benefits included lower cost, decreased erection time, elimination of liner thermal barrier over much of the interior liner surface (hot liner concept), and potentially easier decommissioning based on PCIV disassembly ease. As little work had been completed earlier on this concept, the project required emphasis to meet the limited schedule for the predecision phase.

A number of HTR-K configuration variants were studied, as summarized in Fig. 3-1. The favored PCIV configuration near the end of the study was a central vessel for the reactor core surrounded by ten satellite vessels, six for steam generators and four for auxiliary heat exchangers. The backup configuration was a larger single cavity PCIV, similar to the THTR configuration, with steam generators and auxiliary heat exchangers in an annular space between the core thermal shield and vessel liner. A multicavity PCRV similar to the GA design was carried as a backup for the HTGR-K and was the reference vessel for the process heat concept (PNP).

In early 1977, the PCRV backup design was chosen as the HTR-K concept decision candidate. PCIV evaluations indicated greater development risk, a delay of about two years of the first plant for PCIV development, and less than expected economic incentive, compared to the PCRV. A lower level of PCIV development is, however, continuing in FRG.

3.2.1 HTR-K Reference Design Description

3.2.1.1 Principles. The concept of an advanced high-temperature reactor with dual-cycle plant and spherical fuel elements for an electricity-generating plant is based on the following principles:

1. 3000 MW(t) reactor rated power.

2. Load following range of 100% - 25% rated power, related to thermal reactor power.

3. 40 yr service life, 280,000 full-load operating hr.

4. Reactor internals as well as thermal barrier and liner of the reactor pressure vessel exchangeable or repairable to avoid early

decommissioning of the reactor from unforeseen failures. In addition to the in-service inspections required by the authorities, greatest possible inspectability of reactor internals and thermal barrier to be achieved.

5. Safety design to follow a conservative interpretation of appropriate FRG criteria.

6. The reactor pressure vessel to be burst-safe.

7. The main components of the reactor plant to be in a reactor containment building.

3.2.1.2 Characteristics. Following is a description of each of the main plant components.

Plant

1. Net electric power, 1120 MW (wet tower cooling).
2. Net efficiency, 37.3%.

Reactor Core

1. Pebble bed reactor with OTTO-core.
2. Cylindrical core with 5.5 MW/m^3 power density.
3. U/Th-fuel cycle.
4. Two radial zone core.
5. Six fuel element discharge channels.
6. Hot-gas temperature 700°C,
7. Cold-gas temperature 260°C.
8. Dimensions: Diam., 11.2 m
Height, 5.5 m

Shutdown Concept

1. Two redundant diverse shutdown systems; 198 in-core rods and 48 reflector rods.

2. First shutdown system consisting of part of the absorber rods to be freely inserted into the pebble bed (in-core rods) and part of the absorber rods traveling in bore holes in the side reflector (reflector rods); this shutdown system serves for scram.

3. Second shutdown system consisting of rods which are not reserved for the first shutdown system (in-core and reflector rods); this system serves for long-term shutdown and power control.

4. Emergency measures required to increase the availability of long-term shutdown (rod repair, small absorber spheres, or absorber gas) to be further determined.

Primary Circuit

1. Coolant gas flowing downward through the core; system pressure 60 bar.

2. Six main loops with coordinated steam generator-circulator units.

3. Four separate independent core auxiliary cooling systems.

Secondary Circuit

1. Single-shaft turbomachines with steam reheating between IP section and LP section.

2. Closed-circuit cooling of condensate with wet cooling tower.

3. Three-stage LP preheating, single-stage HP-preheating.

4. Feed water pump drive by turbine.

Reactor Pressure Vessel

1. Prestressed concrete reactor vessel in an integrated multicavity design similar to GA HTGR.

2. Liner with thermal barrier on inner surface.

3. Prestressed concrete plugs as vessel closures of steam generator cavities.

4. Inner dimensions of vessel ensuring clearance for dismantling of thermal barrier and liner repair.

5. Vessel foundations not prestressed with reactor containment building.

6. Two depressurization systems for overpressure protection.

7. Dimensions: Core cavity diam., 16.3 m
   Core cavity height, 15.4 m
   PCRV outside diam., 37.4 m
   PCRV outside height, 31.6 m

Reactor Internals

1. Hot gas duct and cold gas duct separated.

2. Side reflector designed for reactor service life, local breakup on inner surface accepted.

3. All reflector internals exchangeable or repairable in principle.

Control Rods

1. Structural design of in-core rod drives for first shutdown system: pneumatic long-stroke pistons.

2. Structural design of in-core rod drives for second shutdown system: hydraulic long-stroke pistons.

3. Structural design of reflector rod drives: electric/gravity.

Main Circulators

1. Six integrated insertable units with constant speed motor drive, each above the steam generator (6,6 MW per rotor circulator shaft power).

2. Oil-lubricated bearings; oil supply by service system integrated in the circulator unit.

3. Flow control by inlet vane control.

Steam Generator

1. Six integrated units in separate reactor cavities with up-hill boiling; steam conditions 175 bar at 515°C to turbine.

2. Helicoils for preheater, evaporator, and presuperheater.

3. Straight-tube bundle for final superheater.

4. Feed water supply/steam delivery on the lower end over tube sheets.

5. Each tube plugged individually.

Afterheat Removal Concept

1. Concept comprises an auxiliary cooling system and an emergency feed water system.

2. Four auxiliary cooling loops (4 x 50%) independent of each other and of the operating systems.

3. Two emergency feed water lines with closed-circuit systems independent of the operating systems and the auxiliary cooling systems and with additional electricity supply of the main circulators for emergency feed of three steam generators each.

Reactor Containment Building

1. Support structure: solid steel concrete foundation plate, horizontally prestressed cylindrical concrete shell; prestressing by individual prestressing cables.

2. Tightness ensured by a steel liner directly applied to the concrete.

3.2.2 PCIV Concept Review

The PCIV concept was developed by the German company Siempelkamp about ten years ago in close cooperation with the nuclear research center in Jülich. Siempelkamp is a foundry specializing in large castings for heavy machinery, turbine, ship yards, and other applications. To date very few prestressed cast iron pressure vessels have actually been built or ordered.

A PCIV is built of cast iron elements kept under compression by vertical and circumferential prestress tendons. The elements are relatively large, the largest dimension being typically 10 to 12 ft, and although they are hollow, they often weigh as much as 30 to 40 tons. The elements are not made in any standard size, but are specifically designed for each pressure vessel. In a simple cylindrical single-cavity vessel with few and small penetrations, the elements will be relatively simple geometrically, and only a few types of elements will be needed. In a typical nuclear vessel for gas-cooled reactors, however, and particularly in a multicavity configuration, many types of elements of complex geometry will be required, because of the large penetrations and other irregularities.

The elements are shipped to the site and assembled, using a system of interlocking keys and keyways. To avoid an uneven stress distribution, the pieces must fit together accurately, so machining to close tolerances is necessary.

The vertical prestress cables are routed in the continuous channels formed by the hollow elements, whereas the circumferential prestress is wrapped on the outside of the vessel. The prestressing system is similar to that used in a prestressed concrete reactor vessel (PCRV).

Like the PCRV, the PCIV's internal cavities and all penetrations have a continuous steel liner for leak tightness. To compensate for tolerances, it is possible to: 1) allow an initial gap between the liner and PCIV that would be subsequently pressure grouted, or 2) install the liner with both the final longitudinal and circumferential joints left unwelded, fit the liner to the PCIV, and complete the final welds. For the proposed HTR-K hot liner design, the second method has been considered. The liner can be anchored to the PCIV by T-bars or similar anchors that slide into slots in the cast iron segments during assembly. As in a PCRV, the liners will be under compressive strain, and may require anchorage to prevent buckling. Since there is no shrinkage creep in cast iron, however, the liner compression in a PCIV will generally be considerably less than for a PCRV. For the hot liner HTR-K concept, designs having no anchoring have been under evaluation.

Several types of thermal insulation and cooling systems have been proposed, depending on the intended use of the vessel. For an LWR, a thermal barrier with water-conducting cooling tubes in front of the liner has been proposed. For a gas-cooled reactor the thermal barrier is in front of the liner and the cooling water behind it, the water running in tubes that may or may not be welded to the liner or in the channels formed by the hollow cast iron segments.

No aspect of the vessel and its prestressing system is inherently infeasible. The concept appears to be fundamentally acceptable, with nothing to prevent PCIVs of the proposed type being built and licensed for nculear application in the United States. However, in one important aspect the prestressing concept deviates from that used in a prestressed concrete vessel. In a PCRV the stress in the wires is at its maximum immediately after the prestressing operation, and decreases slightly but gradually through operating life, whereas in the proposed PCIV most of the prestressing force is applied through the thermal expansion of the cast iron elements. Each cool-down reduces the stress to the small initial value, resulting in cyclic loads in the prestressing wires and their anchoring system.

Applying most of the prestress through thermal expansion presents additional problems, namely the inability to accurately predict and control the prestressing level because of uncertainties and variations in temperatures and in the coefficient of thermal expansion. Moreover, maintaining a prescribed relationship between coolant pressure and vessel temperature which will impose operational restrictions becomes essential. A rise in pressure without increasing temperature can cause internal pressure loads exceeding the prestress forces, thus causing vessel failure. Another problem is that of initial pressure testing, as the vessel must be heated to design temperatures to conduct a design pressure test.

According to the available information, the only PCIV yet built is a test vessel constructed and tested by Siempelkamp in the early 1970s. The vessel was designed as a model in 1 to 7.5 scale of a cast iron equivalent of the prestressed concrete vessel for the 300 MW THTR. The test vessel is about ten feet tall and has an outside diameter of about 8 ft. The single cavity vessel consists of 37 cast iron elements and has 6 penetrations in each of the top heads and sidewalls. A rubber liner provided leak tightness during the hydrostatic test. The vessel was tested to up to 2.3 times the operating pressure, and the test appears to have been successful, presenting satisfactory correlation between test results and analytical predictions. However, the test only demonstrated the principle of a PCIV as a pressure retaining structure, and did not deal with such important design features in nuclear application as the liner, cooling system, and thermal barrier.

A more detailed concept evaluation of the PCIV is reported in Ref. 3-3.

For the same THTR project a PCIV helium storage vessel of approximately the same size as the model above has been designed and licensed. Fabrication is scheduled to commence in the near future. The vessel is designed for a very high pressure (about 3000 psi) but only moderate temperature (about 120°F). A much larger PCIV has been conceptually designed as burst protection (possibly to be required in Germany) for an LWR planned for the Schmehausen site. The burst protection vessel is built around but separated from the reactor vessel to prevent a catastrophic failure through rupture.

A more detailed concept evaluation of the PCIV is reported in Ref. 3-4.

3.3 LARGE PEBBLE BED CORE

3.3.1 Arrangement

The large pebble bed reactor is characterized by a relatively lower power density, a somewhat larger diameter, and an axially-shorter core than those for current prismatic designs.

The core is contained by the side and bottom reflector structures, which must support the combined weight, pressure drop, and control rod insertion forces. Vertical support posts in the lower plenum support individual hexagonal bottom reflector elements, and spring-pack assemblies reacting against the cylindrical thermal shield provide radial force on the side and bottom reflectors.

To provide for the flow of fuel elements, the bottom reflector thickness is varied in the 3000 MW(t) pebble bed reactor to achieve six conical surfaces, each with an apex at one of the six fuel discharge channels. These six channels are 60° apart, and are radial outward from the core center at about two-thirds of the active core radius. Coolant holes at the bottom reflector-core interface are relieved with a nearly spherical radius trepan to avoid corner chipping of the graphite by the spherical fuel elements. Figure 2-5 shows an elevation section of the large pebble bed core reflector and support structures. Figure 2-6 is a multiple section plan view.

A cutaway of the smaller 750 MW(t) THTR reactor now under construction at Uentrop, Germany, is shown in Fig. 3-2. In this plant, a single fuel discharge channel is at the core center. Graphite top reflector elements are suspended from the top head of the pressure vessel by metal rods. Neutron exposure of these rods is quite high (about 10^{21} nvt), requiring a special alloy, designated X10 CrNiTi 189 within FRG.

3.3.2 Fuel Elements

Fuel element spheres are 60 mm in diameter. A mixture of PyC coated $(U,Th)O_2$ fuel particles and graphite matrix powder is pressed into a spherical shape of about 50 mm diam, which is then covered by a 5 mm thick layer of unfueled graphite matrix. Figure 3-3 illustrates fuel element features.

Fuel element performance limits for the pebble fuel listed on Table 3-1 have been established primarily on the basis of irradiation experience with full-size fuel elements. Calculated stress limits have not been used to establish the basis for limiting power, temperature, and fluence, for

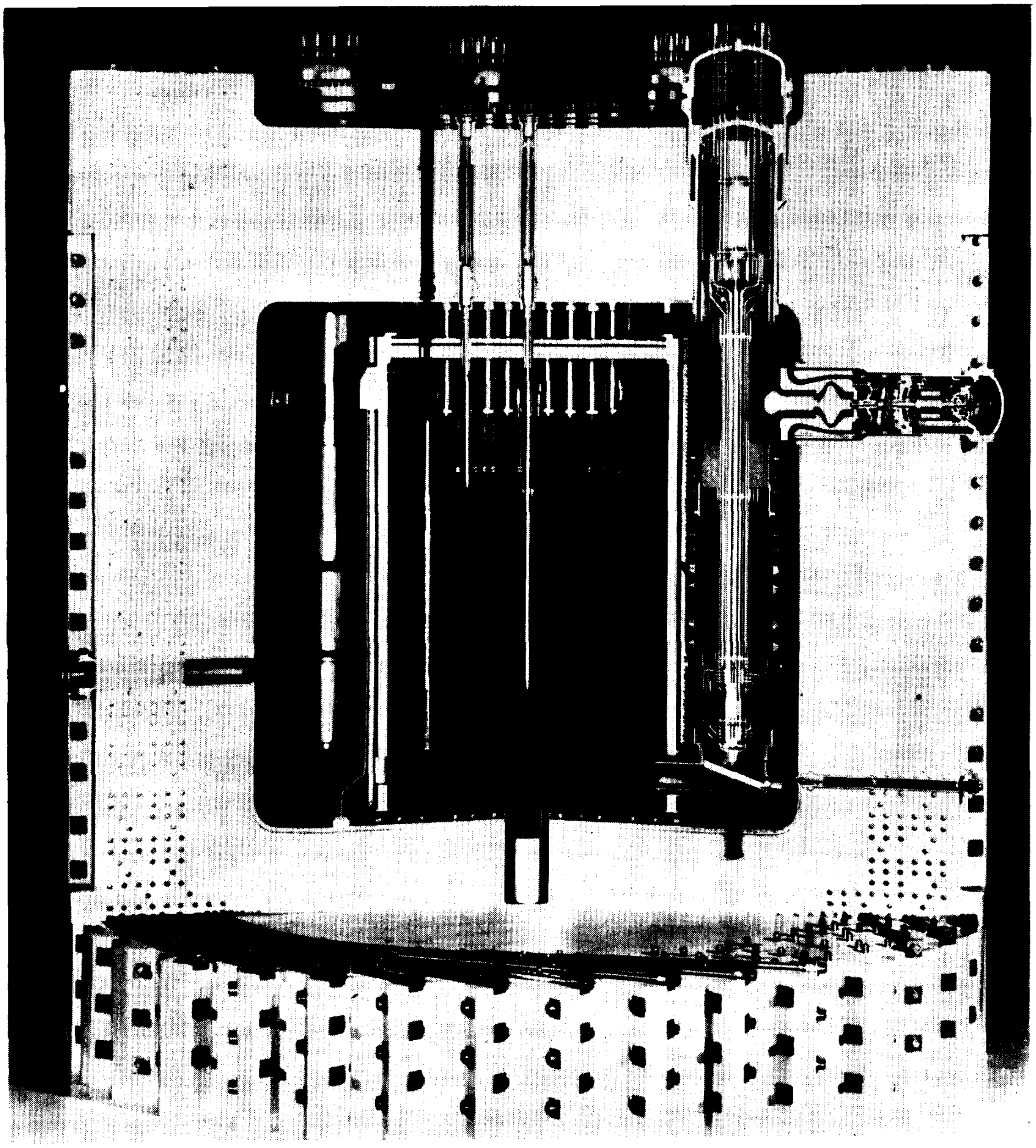

Fig. 3-2. THTR elevation X-section

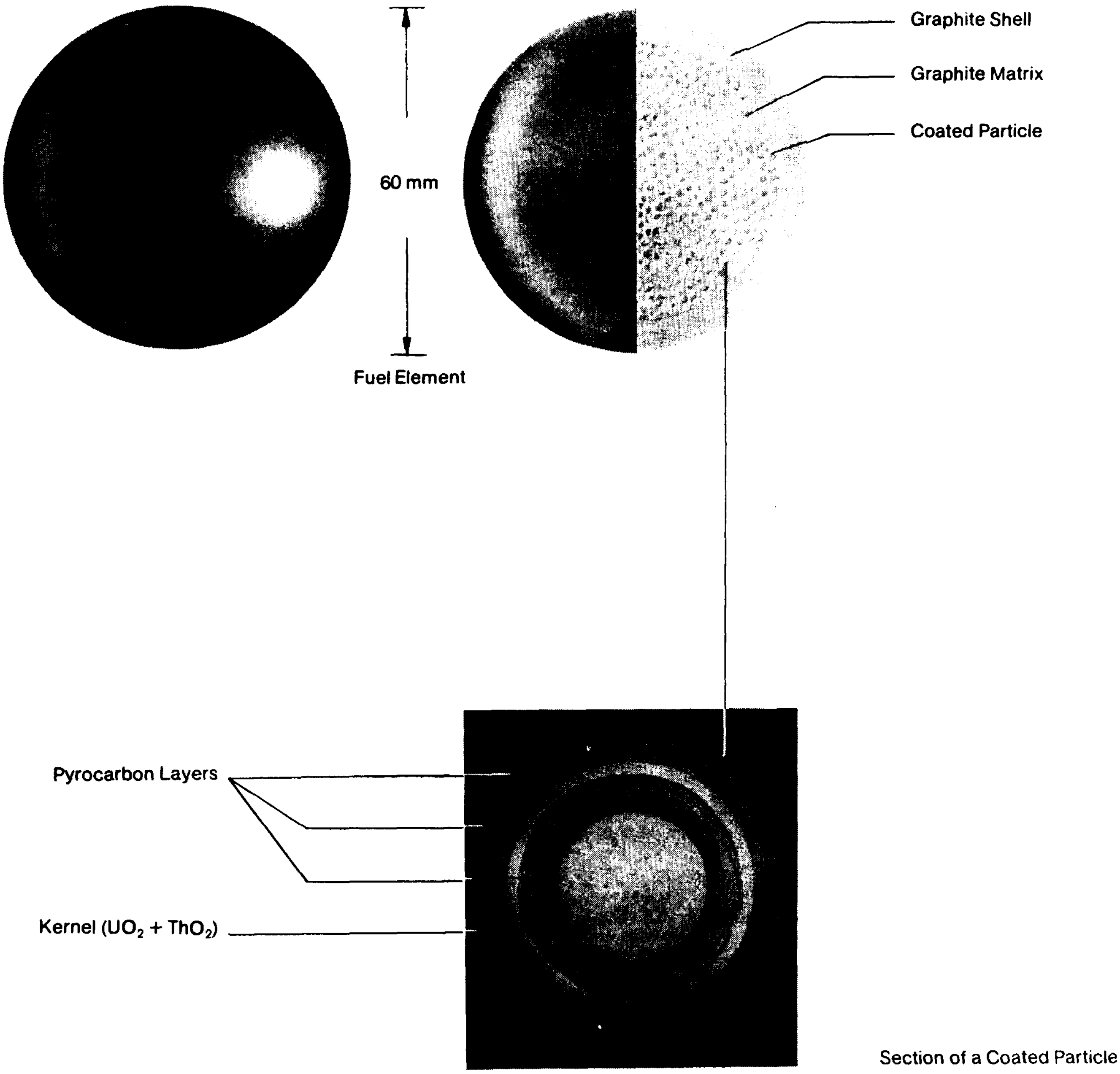

Fig. 3-3. Spherical fuel element

TABLE 3-1
FUEL ELEMENT LIMITS

Tested Limits	Limiting Factors	PNP Design Values
Maximum Center Temp, 1250°C	Fuel particle performance	1020°C
Fuel Ball Surface Temp, 1050°C	Corrosion rate limit	$\sim$1000°C
Power, 4.1 kw/ball	Some test experience at 4.5 kw/ball	3.0 kw/ball
Fast Neutron Exposure, 6.3 x 10^{21} nvt (>0.1 MeV)	--	4.5 x 10^{21} nvt
Fuel Burnup, 14.1% FIMA	Single particle fuel	10.7%
Heavy Metal Loading, 11.2 gm/ball	Maximum production experience ($\sim$20 gm/ball development experience, $\sim$40 vol/0)	11.4 gm/ball

example, since successful irradiation tests have been conducted at conditions more severe than reactor conditions. However, analytical stress and graphite damage models have been developed for the fuel elements, and correlation with experience is being undertaken. By comparison, large reflector blocks cannot be readily tested in full-scale and in this instance analytical results provide the design basis information.

3.4 PEBBLE BED CORE EXPERIMENTAL PROGRAMS

3.4.1 Fuel Sphere Flow in Core

The importance of satisfactorily uniform sphere flow in the core of a pebble bed reactor has been recognized from the beginning of the pebble bed program. To date, no verified or accepted analytical means can predict pebble flow behavior. Therefore, HRB has constructed at its Jülich laboratory experimental equipment with which flow patterns in the pebble bed can be measured.

Three characteristics of pebble flow of primary interest are residence time of the individual spheres, void fraction of the bed, and flow pattern through the core. The effects of several parameters on each of these characteristics have been investigated by HRB.

A direct method of determining the movement of the balls through the core is the use of glass spheres in a transparent container. When the voids between the spheres are filled with a fluid of the same refractive index as the glass, groups of colored spheres can easily be observed as they travel down through the core. This method has the advantage of giving a direct three-dimensional picture of the flow patterns, but parameter variations in ball density and friction are difficult to achieve. Therefore, this method has been abandoned. Nevertheless, the results of the tests indicated that the flow patterns were laminar and that it would be possible to operate the core with radial zones.

To obtain more experimental flexibility, scale models of the pebble bed reactor core designs with ball removal machinery at the bottoms were constructed. Into these beds can be placed layers or individual groups of colored test spheres. After the test spheres are loaded, a portion of the core can be circulated, more test spheres added, another portion circulated, and so on until the test spheres begin to exit. The core can then be unloaded from the top and the positions of the test spheres recorded. In this way, measurements of flow paths and ball dispersion can be made. Continued normal circulation of the core allows determination of residence times.

The equipment includes a one-sixth scale model of a 3000 MW(t) core with six discharge hoppers and a one-half scale THTR core arrangement with one discharge hopper. During early experiments, an unexpected effect was observed. When the pebble bed was circulated slowly for a long time, the spheres near the container wall began to assume a systematic order resembling crystallization (Ref. 3-5). Continued circulation of the bed caused a spreading of the crystallization toward the interior of the core. This crystallization is highly undesirable because of the change in void fraction and a decrease in the relative flow velocity of the balls in the crystalline structure. To prevent it perturbations were introduced in the walls of the container. Early models used vertical grooves; later experiments showed that concave areas in the sides of the container work just as well. These side perturbations precluded the formation of crystalline patterns and allowed the ball flow and voidage to be more nearly uniform.

A very important result of the ball flow work is the conclusion that scale has no effect on the ball flow patterns. Some other important results and aspects of the experimental work are described below.

Flow Patterns and Residence Times

The flow path of a ball from top of reactor to outlet may vary up to three ball diameters, but is most often much narrower (Fig. 3-4). When six discharge hoppers are used, the residence time of spheres varies only

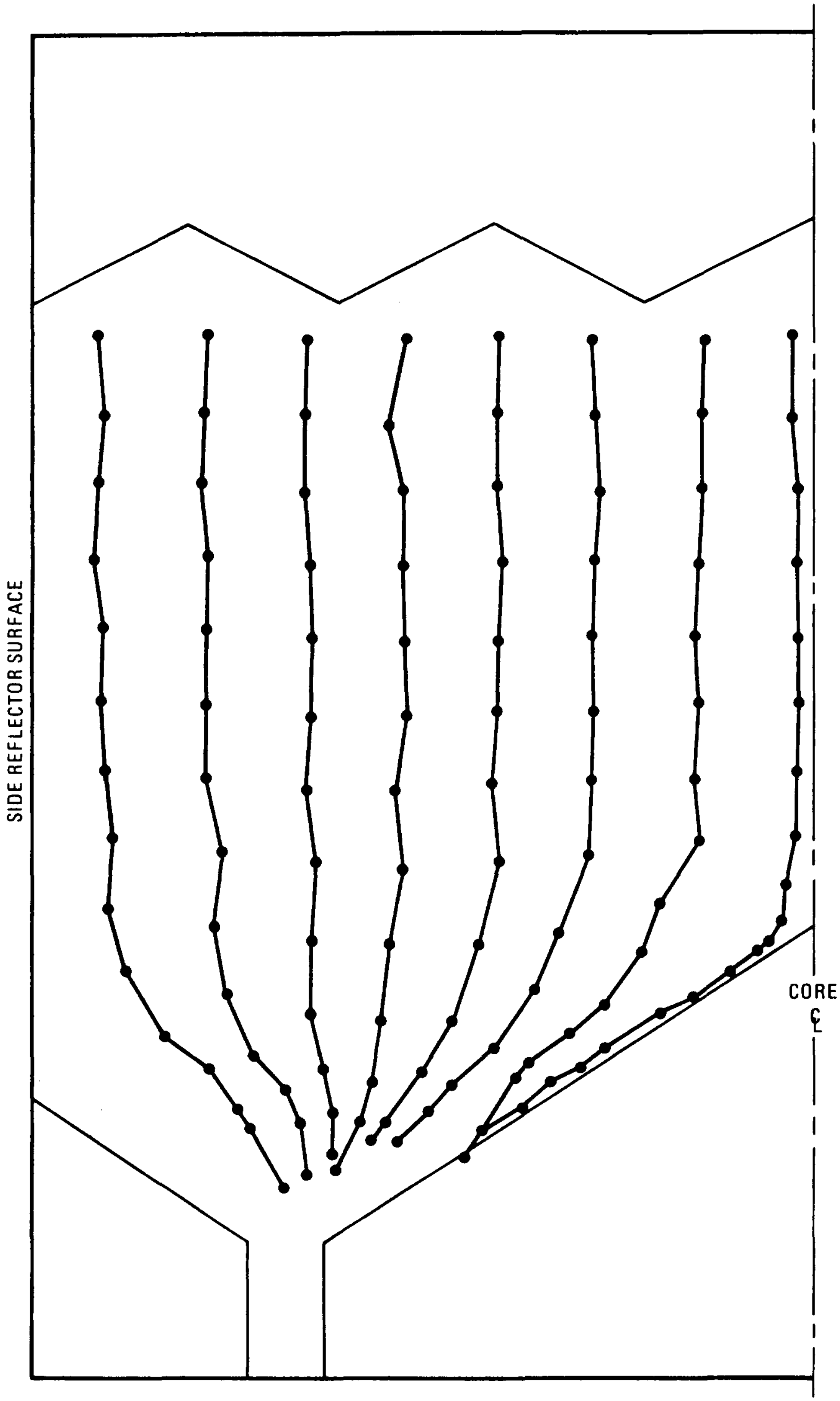

Fig. 3-4. Ball flow paths for six discharge tubes at midpoint between discharge channels (30°)

by a factor of approximately 1.5 (Figs. 3-5 through 3-7). Ball velocities are very uniform for 4.0 to 4.5 m into the core. Beyond this they vary, but power density in the bottom of the core is so low that the small variation is of little consequence.

The effects of the following parameters were investigated before the final design configuration was selected: 1) radius versus sharp angle transition from core cylinder to discharge cone, 2) angle of discharge cone, 3) bed height to diameter ratio, 4) bed to ball diameter ratio, 5) discharge hopper to ball diameter ratio, and 5) specific weight of test spheres. As friction becomes greater, the ball flow becomes more uneven. Even though HRB has not performed experiments in hot helium (which will increase friction by a factor of approximately 3), experience suggests that the effects will be second order.

Void Fraction

The void fraction of a pebble bed lies between that of cubic and hexagonal packing arrangements. It is approximately 0.38 to 0.39. Near the wall the void fraction is slightly higher, allowing more coolant gas flow. Core pressure drop is proportional to the third power of the void fraction. Therefore, small variations can cause marked changes in core flow pattern. This is a major reason for avoiding crystallization of the bed.

The void fraction must not vary significantly during operation as variation will change the neutron leakage and therefore core reactivity. Operational experience at the AVR indicates that the void fraction remains very constant during circulation of the bed. The void fraction increases from top to bottom in the pebble bed (Fig. 3-8). The larger voids near the bottom do not collapse unless the bed is disturbed (for example, by the insertion of rods).

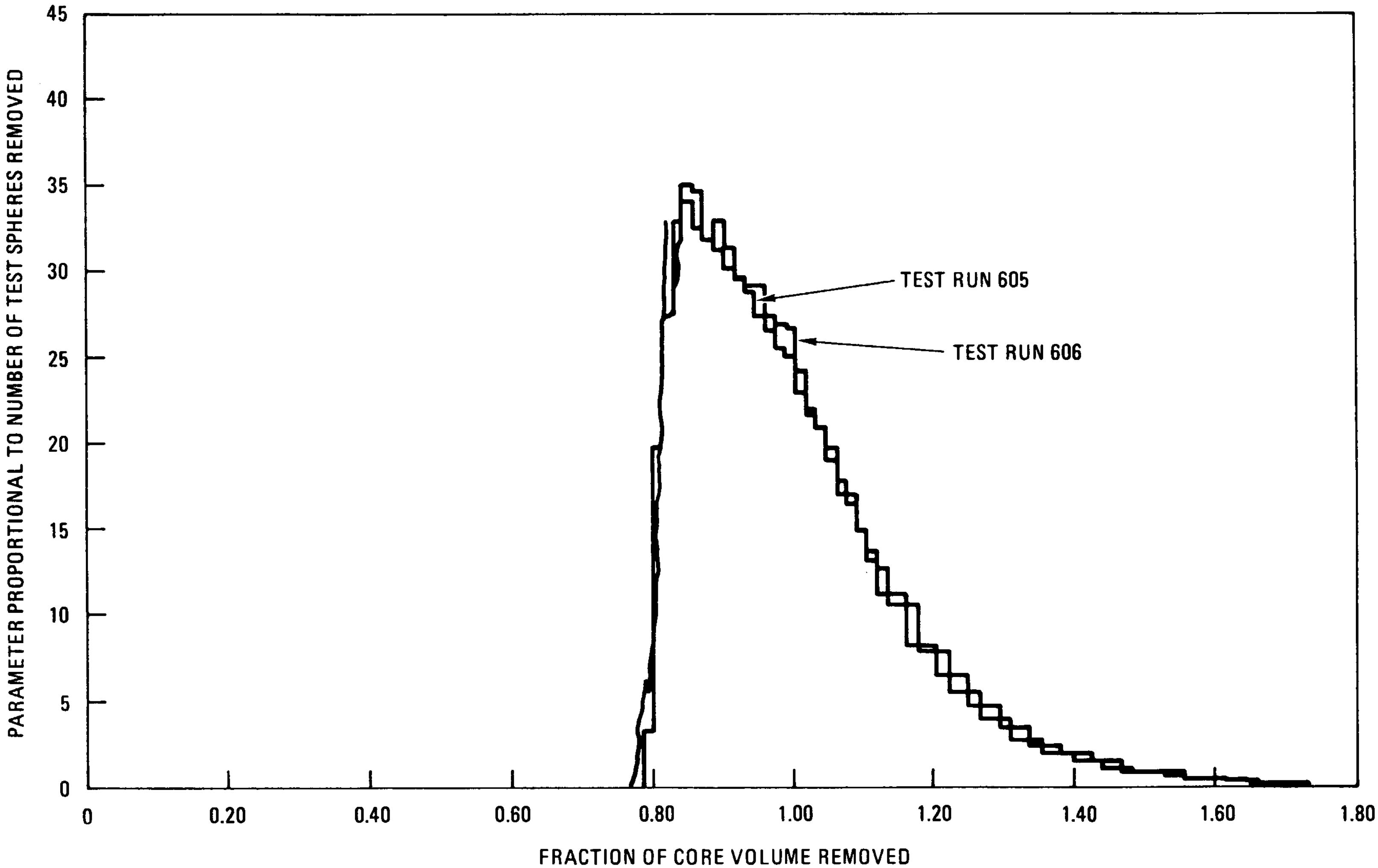

Fig. 3-5. Fuel element residence spectrum HTR-P-3000 (six discharge tubes)

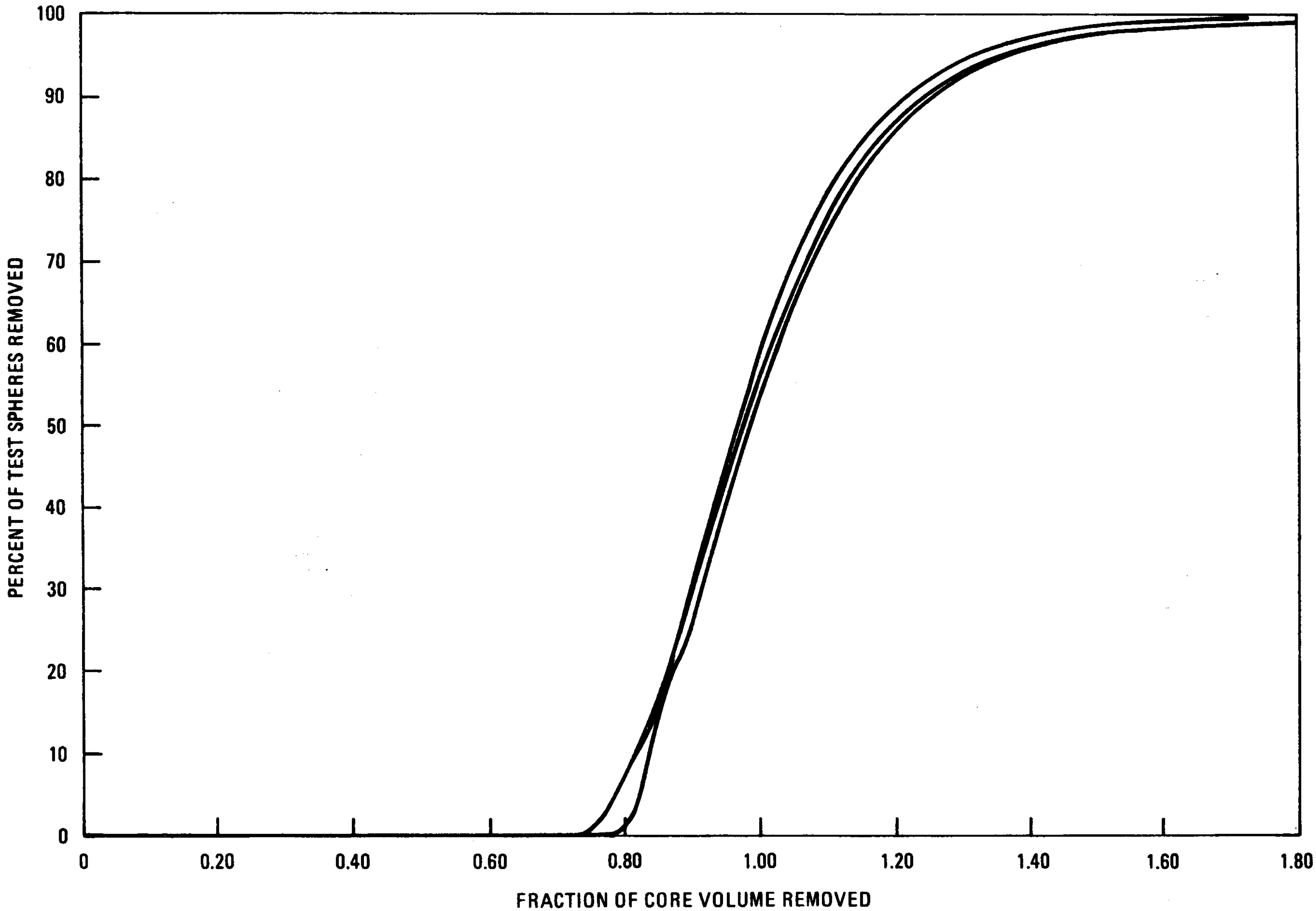

Fig. 3-6. Integral fuel element residence spectrum, 1:6 HTR-P-3000 (six discharge tubes)

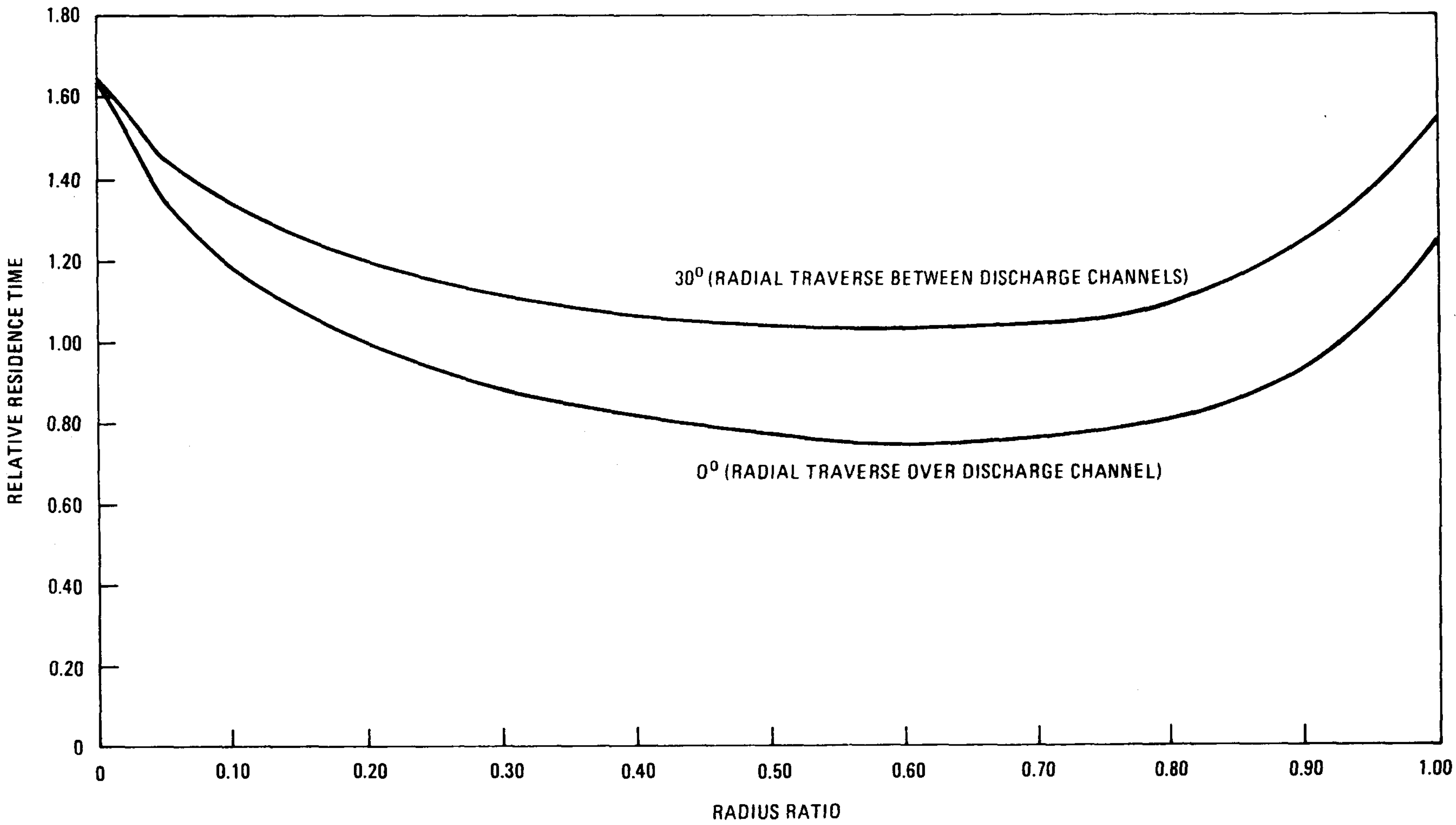

Fig. 3-7. Diametral sphere flow variation, 1:6 PNP-3000 - model (six discharge tubes)

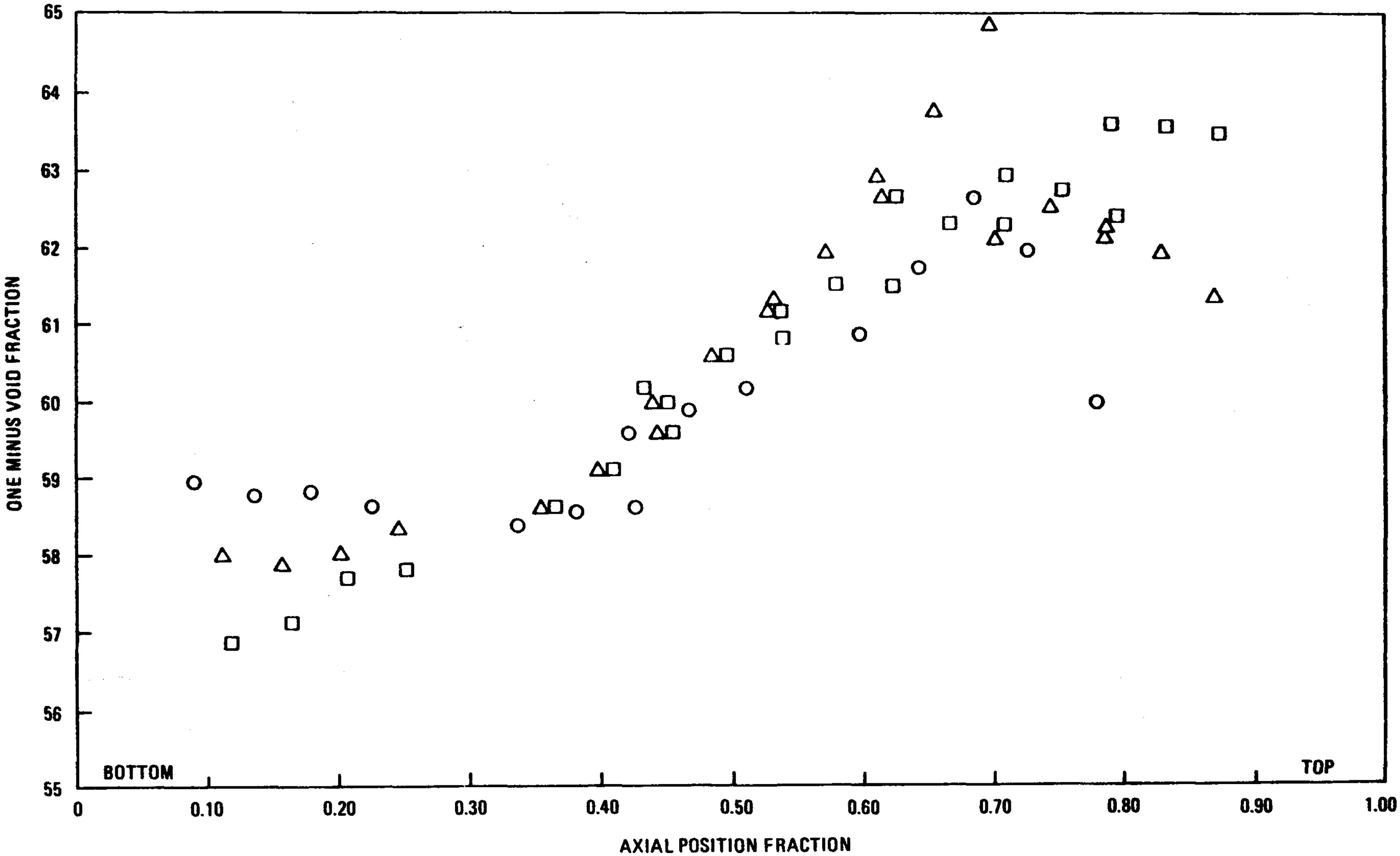

Fig. 3-8. Graphite spheres, equilibrium conditions, 1:6 PNP-3000 - core

3.4.2 Control Rod Insertion

The experimental apparatuses at HRB, Julich (HRB facilities at KFA), allow the measurement of the forces on rods in both early one-sixth and current one-half scale models. The one-sixth model results were used with the following scaling equation to predict the results of the one-half scale tests:

$$F = F_m \left(\frac{M}{M_m}\right)^3 \left(\frac{\gamma}{\gamma_m}\right)\left(\frac{\mu}{\mu_m}\right)$$

where F = calculated force

F_m = force measured in model

M/M_m = ratio of untested model dimensions to tested model dimensions

γ/γ_m = ratio of density of untested balls to tested balls, including the effect of core Δp, i.e.,

$$\gamma = \gamma_{graphite} + \frac{\Delta p}{(1-V)H}$$

where Δp = pressure drop across core

V = void fraction

H = core height

μ/μ_m = ratio of untested model friction coefficient to friction coefficient of tested model. (μ for a hot helium atmosphere is 2 to 3.5 times greater than for air)

This equation was developed from the theory of similitude. Its use gave predicted results for the one-half scale tests, based on one-sixth scale measured results that were within 3% of the measured results for the one-half scale tests.

German experiments have concluded that the results of these scale tests will allow confident prediction of rod forces for the large HTR-K.

Some of the extrapolated results for rod bank scrams for a 3000 MW(t) core are shown in Figs. 3-9 and 3-10. The multiple curves in each figure represent successive insertions with no circulation of the balls between tests. These figures reveal that 1) the rod forces are higher near the edge of the core than near the middle and 2) as the rods are scrammed repeatedly, the forces rise. This rise is caused by the nonuniform distribution of voids along the axis of the core, as shown in Fig. 3-8. The lack of uniformity is caused by virtual doming deep in the core. As the rods are inserted repeatedly, this doming is destroyed, the bed becomes more packed, and the rod insertion forces increase.

Because of expense and difficulty, the Germans have done no large scale multiple rod experiments under reactor conditions. Their work has been based on reduced scale models and the scaling equation discussed above.

3.4.3 Small Ball Shutdown System

Experiments to assess the feasibility of dumping small boronated balls into the core as a second independent and diverse shutdown system are currently in progress. In addition, tests to determine the effects of seismic vibration on the distribution of small balls are being conducted.

3.4.4 Fuel Handling System

The fuel handling machinery at HRB, Jülich, was used to develop the THTR fuel transport and separation techniques. Similar systems are proposed for the HTR-K except that fuel elements will pass through the core only one time. Also, much useful information is available from the operation of the AVR fuel handling system. The apparatus at Jülich has been used to perform tests to measure the effects of carbon dust on ball movement and to determine the collection points of the dust within the apparatus.

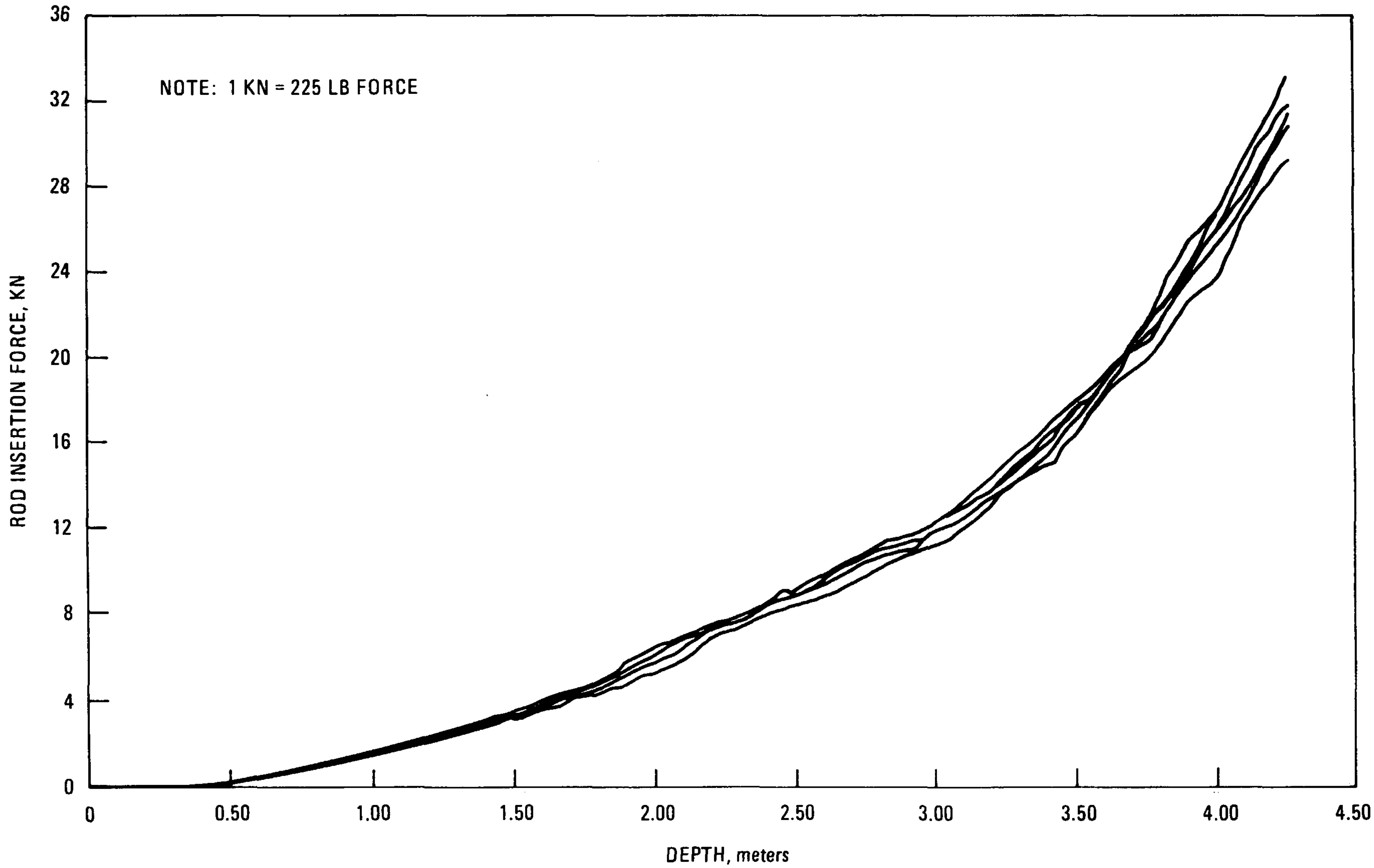

Fig. 3-9. Control rod insertion force in HTR-P-3000; 198 control rods (banked) with NH_3 and ΔP = 0.06 bar. Position: 016 (near reflector); test run: 207

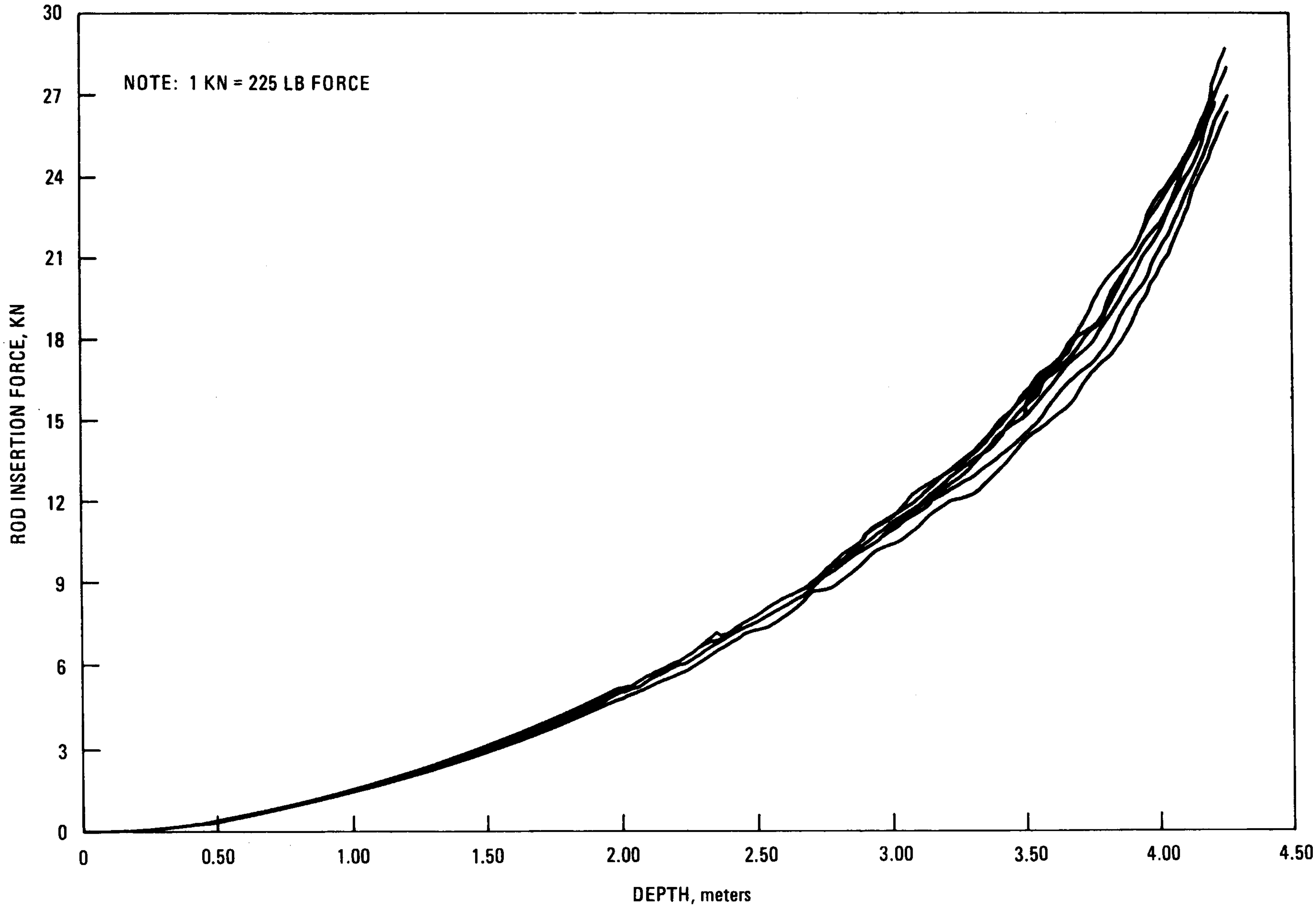

Fig. 3-10. Control rod insertion force in 198 control rods (banked) with NH_3 and ΔP = 0.06 bar. Position: 104 (near core center); test run: 207

3.4.5 Nuclear Design

As described earlier, design basis fuel management is the once through, or OTTO, fuel cycle, in which the fuel spheres are introduced at the top of the core and flow slowly through the core to be discharged as spent fuel (Ref. 3-6). To flatten the radial power shape, two radial zones are used, with somewhat higher loading in an annular zone of about 1 m in radial thickness adjacent to the side reflector. Table 3-2 summarizes fuel loadings for typical high enriched U (HEU) core design.

Table 3-3 lists selected core nuclear parameters for the large core with OTTO cycle.

Figure 3-11 shows equilibrium axial power shapes (unrodded) for three radial locations. Radial power distributions and unrodded, equilibrium conditions for several axial heights are presented in Fig. 3-12.

German analyses indicate that the large pebble bed core probably will not have axial xenon instabilities. Azimuthally, however, the possibility of oscillations being self-damping is marginal.

Currently, the control and shutdown reference design calls for 198 cylindrical in-core rods and 48 reflector rods, as shown on Table 3-3. Of these, about 30 in-core and probably most of the reflector rods will be used for fast shutdown. The remainder, using a different drive mechanism to avoid systematic problems, will be employed for long-term shutdown. Alternative schemes including rotating control rods and small absorber ball diverse shutdown systems continue under study at FRG.

3.4.6 Critical Assembly

The KAHTER experimental facility is a small, well-instrumented pebble bed critical assembly being used to verify the calculational methods for the reactor physics analysis of pebble bed cores. The facility is at KFA, Jülich.

TABLE 3-2
HEAVY METAL LOADING SUMMARY (REF. 3-7)
3000 MW(t) PEBBLE BED CORE

Loading, gHm Per Fuel Sphere	Center Zone	Outer Zone	Core Average
Thorium	10.34	10.34	10.34
U-235	0.78	0.964	0.841
U Total	0.837	1.035	0.903
Total HM	11.177	11.375	11.243

TABLE 3-3
EQUILIBRIUM CORE PARAMETERS (REF. 3-7)
3000 MW(t) PEBBLE BED CORE, OTTO CYCLE, HEU

Power Density	5.5 MW/m^3
Diam	11.2 m
Nominal Height	5.5 m
Fuel Spheres	60 mm diam
Avg Packing Fraction	0.61
No. In-Core Control Rods	198
No. Side Reflector Control Rods	48
C/Th	355
Moderation Ratio	10,000
Fuel Residence Time (Avg)	1161 days
Conversion Ratio	0.62
k_{eff}	1.0032
Avg Burnup	99,900 MW d/t
Avg FIMA	10.6%
Maximum Fuel Element Power	3.01 kW/element
Maximum/Avg F.E. Power	3.1
Maximum Fast Fluence, >0.1 MeV	
Fuel Element (Lifetime)	4.43×10^{21}
Top Reflector (Annual)	1.565×10^{21}
Side Reflector (Annual)	1.681×10^{21}
Bottom Reflector (Annual)	0.116×10^{21}

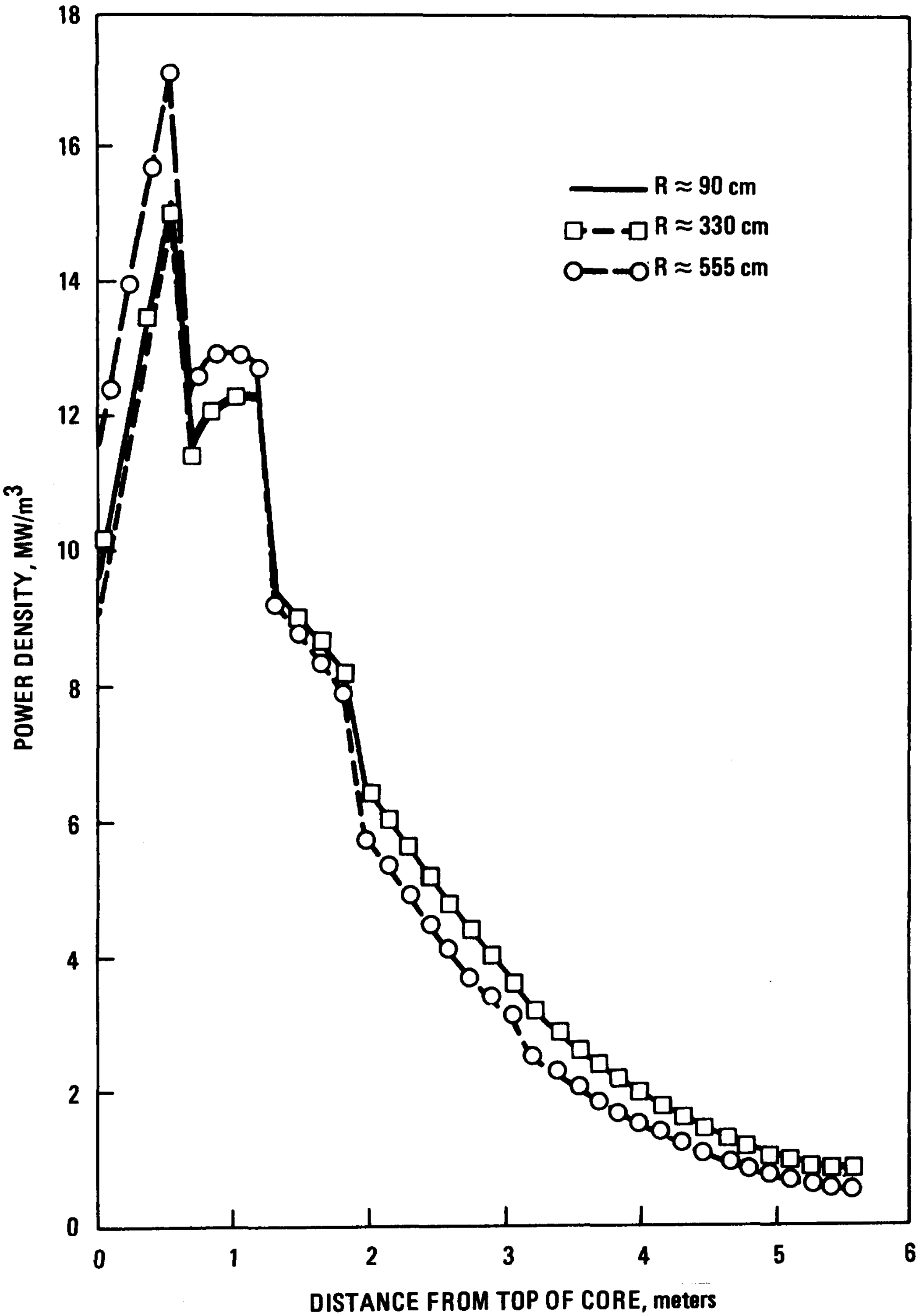

Fig. 3-11. Axial power distribution, 3000 MW(t) pebble bed core, unrodded; average power density 5.5 MW/m^3

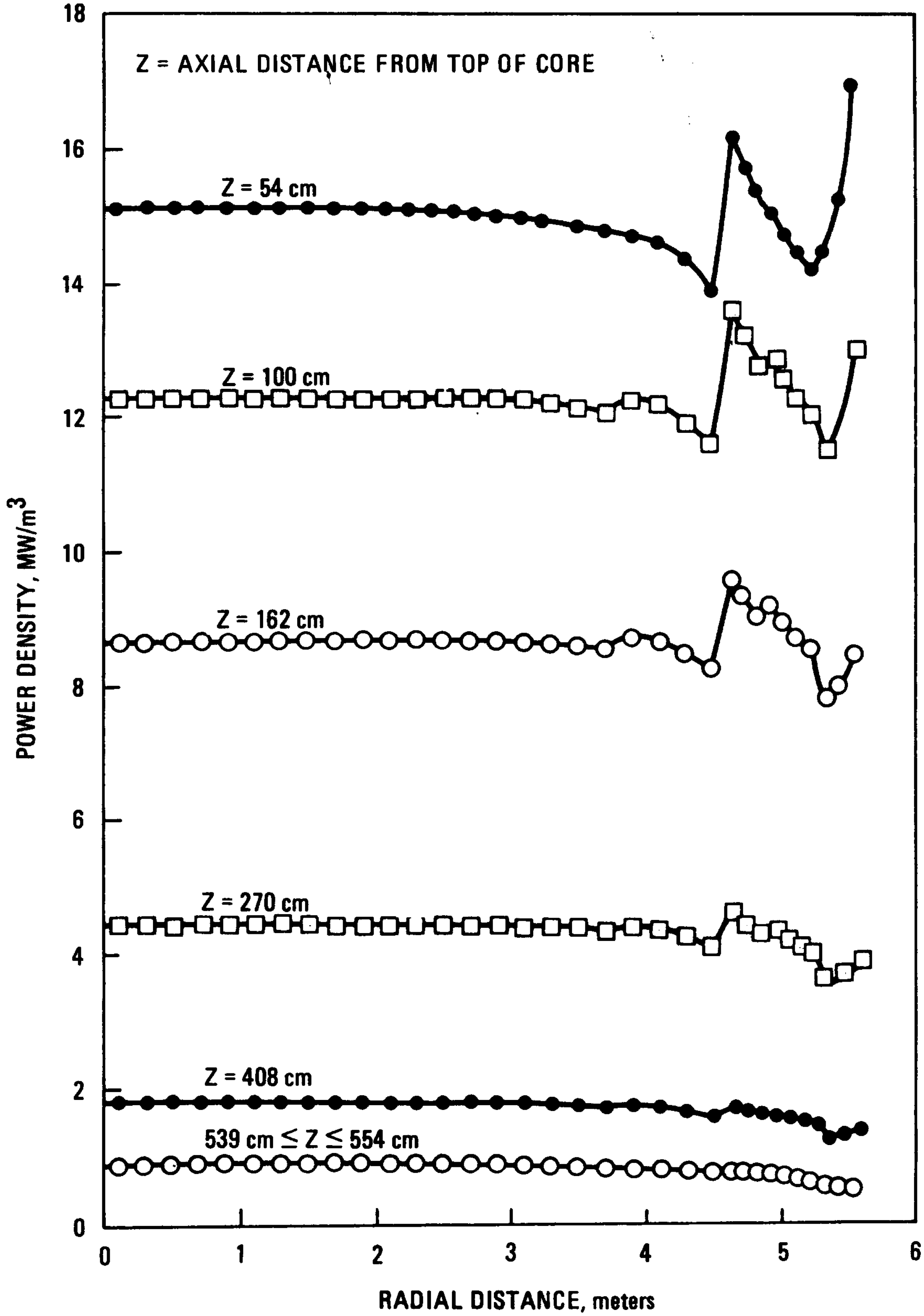

Fig. 3-12. Radial power distribution, 3000 MW(t) pebble bed core, unrodded; average power density 5.5 MW/m^3

Eight critical experiments were described during the February 1977 GA team visit. For the six relevant to the current core design, criticality was predicted within a range of ±0.3% Δk. The remaining two experiments showed that the methods used to calculate burnable poison worths needed improvement, although the errors were less than 0.9% Δk. Some of the experiments were designed to test ability to predict the worth of control rods inside the pebble bed and in the radial reflector. These experiments demonstrated that rod worths could be predicted to within 10%. Flux mapping measurements were also compared with calculations. For the instances studied, the discrepancy between experiment and calculation was generally within $\sim$5% near the core-reflector boundary.

During the next year a series of experiments is planned to assess the ability of calculational methods to predict parameters in cores using an OTTO fuel cycle. KAHTER is being loaded so that the flux profile will be skewed similarly to an OTTO cycle and a number of measurements, criticality, flux distributions, and rod worths, can be made. Rod worths in the upper void will be measured. This program will further test their calculational methods.

3.4.7 Core Performance

Although standardization is not complete in the various FRG organizations, representative flow and heat transfer correlations used for pebble bed performance analyses are available in the literature. Specifically, correlations used by GHT in the COBRA-IIIC (Ref. 3-8) code are given below.

1. Friction pressure drop: high Reynolds number - Barthels correlation, Ref. 3-9; low Reynolds numbers - Jeschar correlation, Ref. 3-10.

2. Heat transfer between fuel element and helium: Jeschar correlation, Ref. 3-11.

3. Radial heat transfer due to conduction and mixing - Zehner-Schluender correlation, Ref. 3-12.

Results using the COBRA-III C code for analysis of a typical large pebble bed core have been reported by Gysler, et al. (INTERATOM) (Ref. 3-13). Figures 3-13 and 3-14 show temperature and flow distributions in the axial and radial directions respectively for 960°C outlet gas temperature conditions.

Hot spot estimates by GHT staff indicate +90°C for systematic factors and +50°C from a combination of statistical uncertainties for a total of +140°C over nominal PNP maximum fuel temperature. Details of this work are not available for elaboration in this report.

3.4.8 Fission Product Release

Pebble bed core designs employing the OTTO cycle are characterized by an axially flat fuel temperature profile caused by the steeply decreasing axial power shape. With the moderate radial peaking predicted by FRG participants, peak fuel temperatures and temperature gradients are below threshold levels for temperature-activated fuel particle failure. Thus gaseous fission product release is controlled by initial tramp uranium contamination outside of the PyC particle coatings and by burnup-dependent failure of initially defective particles, with current fuels having $\sim 2 \times 10^{-4}$ fraction contamination and $\sim 1 \times 10^{-3}$ in-service failure respectively. Resulting fractional release (release to birth ratio) of Kr 85 m should be about 5×10^{-5} R/B for the large pebble bed core.

Metallic fission products are retained by metal carbide coatings such as silicon carbide on the fuel particles, but they are capable of diffusive release through pyrocarbon coatings, particularly cesium and silver nuclides. Diffusive transport through PyC, however, depends to some degree on the PyC structure and to a great degree upon temperature. Therefore, metallic release from fuel elements containing fuel with an SiC layer (or SiC alloyed PyC) on all particles will depend mostly on particle-coating failure. With only PyC coatings, release will be mostly temperature- and time-dependent,

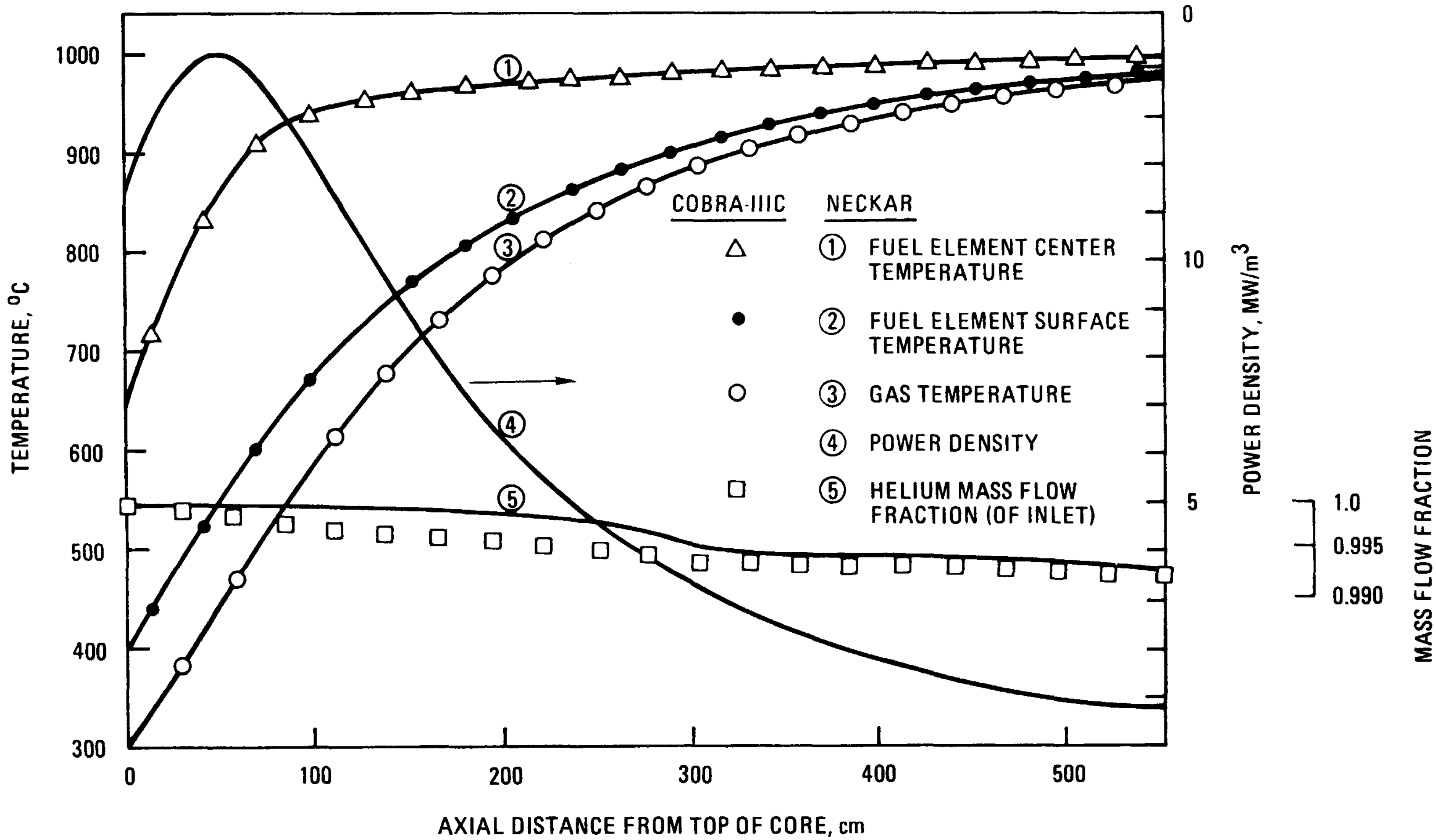

Fig. 3-13. Axial distributions of temperature, helium mass flow, and power density in center region of core, 3000 MW(t) PNP

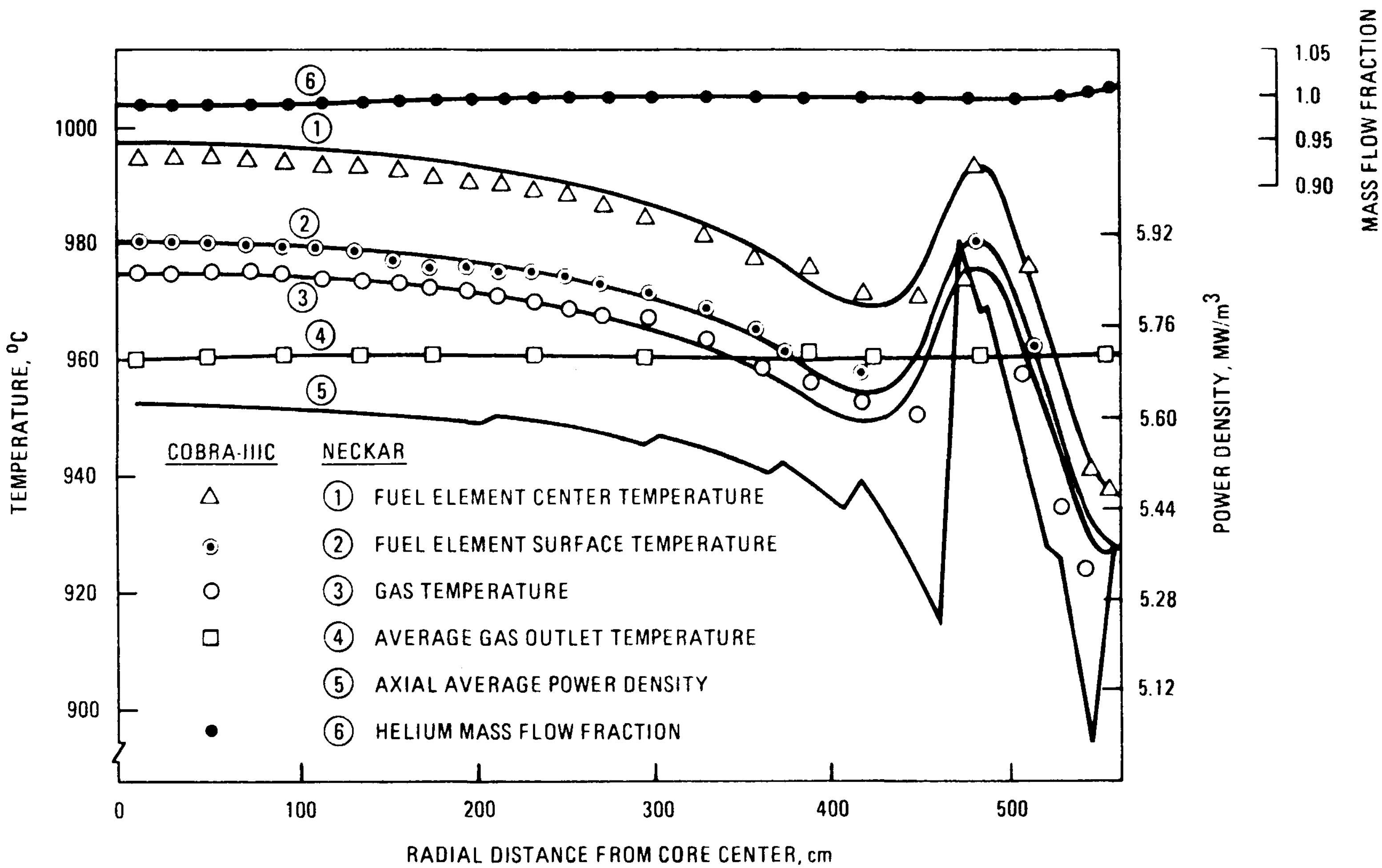

Fig. 3-14. Radial distribution of temperatures and helium mass flow at core outlet and radial distribution of axial average power density, 3000 MW(t) PNP

although a significantly lower diffusion rate is observed for high-temperature methane PyC (HII) as compared to lower deposition temperature propylene coatings (LTI).

However, for the pebble bed concept, peak fuel and fuel element surface temperatures predicted by FRG organizations just reach the threshold of the temperature range where cesium release is predicted to increase very rapidly with temperature. Also, the more expensive HTI PyC coatings are normally assumed for PNP core cases without SiC-coated fuel. Predictions (Ref. 3-14) of cesium 137 release for the 3000 MW(t) PNP conditions of 300°C core inlet with 950°C outlet helium temperature are:

Particle Coating	Net Core Release
All PyC (HTI)	0.25 Ci/MW-yr
All SiC	0.16 Ci/MW-yr

For PyC, if local power distributions or power/flow conditions within the large pebble bed core are found to be significantly less favorable than currently predicted, a significant increase in predicted cesium release could be anticipated. With the SiC fuel, there will be much less sensitivity to these uncertainties. For comparison, predicted cesium 137 release with all SiC-coated fuel is estimated to be essentially the same, 0.18 Ci/MW-yr, from a prismatically fueled core for the same conditions of power density, helium temperatures and fuel residence time.

3.5 FUEL CYCLE

The initial phase of the fuel cycle evaluation of pebble bed reactor designs involved a literature search and review and the preparation of a fuel management information package that could generally aid in understanding how the important fuel management variables affect the resource utilization and costs of both pebble bed and prismatic block HTGR designs.

A comprehensive report (Ref. 3-15) on pebble bed fuel cycle characteristics was reviewed and results from that report form the basis for comparing GA calculations of pebble bed designs to those performed at KFA.

Some preliminary calculations have been performed for two Th/U-235 cycles described in Ref. 3-15. The preliminary results from these calculations show that GA predictions of U-235 feed requirements agree to within 6% with those of KFA.

The results have also been compared to an equivalent prismatic block design with annual refueling. This comparison shows that there is little difference in the uranium requirements for annually-fueled prismatic designs and continuously-fueled pebble bed designs. This latter result is caused by the fuel savings resulting from on-line fueling in the pebble bed being essentially balanced by the fuel savings ensuing from the composition and lower neutron leakage characteristic of the prismatic block design.

3.5.1 Objectives of Fuel Cycle Evaluation

The basic objectives of the fuel cycle evaluation are to:

1. Verify the German prediction of pebble bed mass balances for each of the three cycles considered, i.e.,
   - Low enriched U cycle.
   - Th/U-235 converter.
   - Th/U-233 near-breeder.
2. Understand the basic fuel management differences between the pebble bed and comparable prismatic designs for each cycle considereu.

3.5.2 Mass Balance Verification and Comparisons

For a given C/HM ratio and power density, there are three differences between prismatic and pebble bed designs that may affect the mass flow requirements. These are:

1. Neutron leakage
2. Refueling frequency
3. Neutron cross sections

This analysis effort has been concerned with understanding the effects of differences caused by refueling frequency and neutron leakage. Depletion calculations aimed at understanding these effects have been performed for the Th/U-235 cycle designated as KFA Case 4011.

The nine-group microscopic cross sections (5 fast, 4 thermal) based on the prismatic block design were used for studying these effects for both reactor types. Using the same microscopic cross section sets to study both the effects of refueling frequency and neutron leakage is justified; it would not be expected to appreciably bias the conclusions.

For the multi-group set used it is expected that the actual cross section differences between the two designs will be very small, with the possible exception of those for the resonance absorber nuclides such as Th-232 and U-238.

3.5.3 Cycles for Evaluation

Cycles selected will be:

KFA No. (Ref. 3-15)	1013	1213	4011	4021	9022
Power Density, w/cc	9.0	9.0	9.0	5.0	5.0
Cycle	LEU	LEU	Th/U-235	Th/U-235	Th/U-233
C/HM	360	360	245	245	110

Cycle Length, EFPDs	627	815	907	1633	1267
Fuel Element Type	Shell	Shell	Ball	Ball	Ball
Burnup, MWD/T	100,000	130,000	100,000	100,000	40,000

3.5.4 Neutron Leakage Effects

Comparison of the quoted pebble bed leakage values from Ref. 3-15 and those calculated at GA for prismatic block designs indicates that the pebble bed design has appreciably higher neutron leakage rates. Such a comparison is shown in Fig. 3-15, in which the neutron leakage rates for both designs are plotted as a function of the core power density. The pebble bed values taken from Ref. 3-15, as indicated in Fig. 3-15, involve a change in the active core height with change from a power density of 9 w/cc to 5 w/cc.

For the same power density and L/D ratio, the pebble bed leakage rates should be approximately 15% higher because of the higher void content in that design. From simple neutron leakage estimates based on geometric buckling calculations the neutron leakage in the pebble bed at a power density of 9 w/cc should be about 0.053 compared to 0.040 for the prismatic design. However, the actual values are higher, primarily because of the presence of control rods in the upper reflector and the fact that the upper reflector is above the top of the active core.

3.5.5 Refueling Frequency - Comparisons for KFA Case 4011

The analysis of the effect of refueling frequency was based on depletion calculations performed for the Th/U-235 cycle designated as KFA Case 4011. Calculations were also performed for an equivalent prismatic design with annual refueling. The following four cases were analyzed.

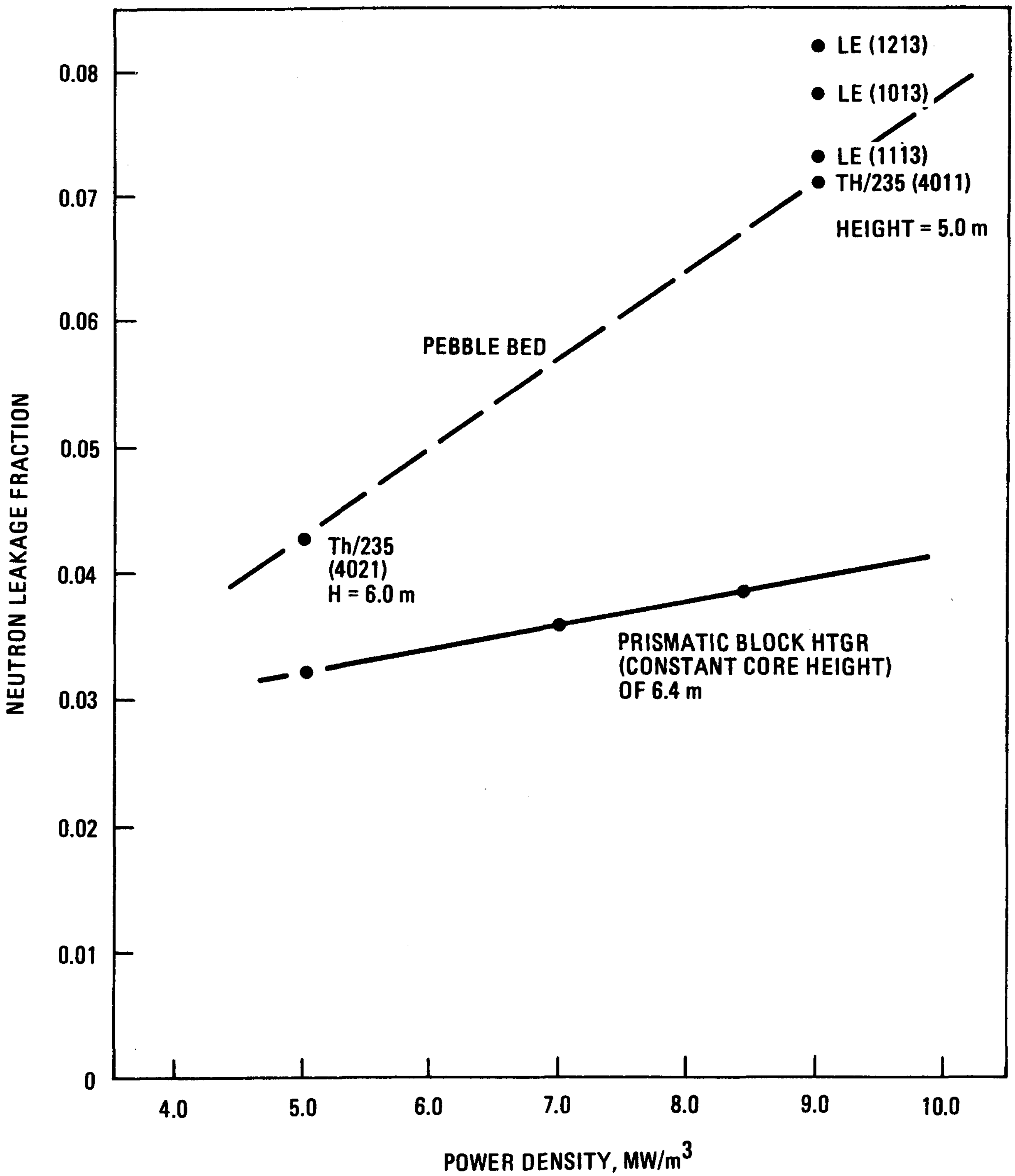

Fig. 3-15. Comparison of neutron leakage fraction at varying power densities for HTGR prismatic block and pebble bed designs

Case	Description	Neutron Leakage	Refueling Frequency
A	Pebble bed with core composition of KFA 4011	0.071	Annual
B	Same	Same	Semiannual
C	Same	Same	On-Line
D	Prismatic with same C/HM and power density as KFA 4011	0.04	Annual

From a comparison of A, B, and C, the effect of the refueling interval can be determined. The effect of core composition and neutron leakage can be determined by comparing A and D. Case C can be compared to the published KFA results for Case 4011 from Ref. 3-15. This latter case can in turn be compared to Case D for the comparison of KFA-predicted pebble bed results to the comparable annually-refueled prismatic design predicted by GA.

The comparison of the reactor mass balances for these cases is shown in Table 3-4. The U-235 feed requirement as well as the U-235 and U-233 reactor inventory values for each case is shown in Fig. 3-16 as a function of the refueling frequency.

Several important conclusions can be drawn from the data shown:

1. On-line refueling reduces the U-235 feed requirement by about 18% relative to annual refueling.

2. The lower neutron leakage in the prismatic block design results in about a 12% reduction in the U-235 feed requirement relative to a comparable pebble bed design.

3. The GA prediction of KFA Case 4011 agrees to within 6% with the KFA calculation.

4. The U-235 feed requirement predicted by KFA for Case 4011 is nearly the same as the GA prediction for the annually-refueled prismatic design.

TABLE 3-4
COMPARISON OF MASS FLOW REQUIREMENTS FOR PEBBLE BED (KFA 4011) AND COMPARABLE PRISMATIC DESIGN

	KFA 4011 On-Line	Case A GA 4011 Annual	Case C GA 4011 On-Line	Case D Prismatic Annual
Residence Time, Full Power Days	907	907	907	907
Conversion Ratio	0.583	0.52	0.60	0.594
HM Charged/Full Power Day				
Heavy Metal (kg/day)	29.58	30.63	30.05	34.00
U-235 (kg/day)	2.38	2.77	2.24	2.45
Discharged/Full Power Day				
Heavy Metal (kg/day)	26.37	27.55	27.01	30.94
U-233 (kg/day)	0.619	0.694	0.649	0.723
U-235 (kg/day)	0.172	0.303	0.141	0.209
Pu-239 (kg/day)	0.0017	0.0048	0.0042	0.0040
Pu-241 (kg/day)	0.0011	0.0023	0.0020	0.002

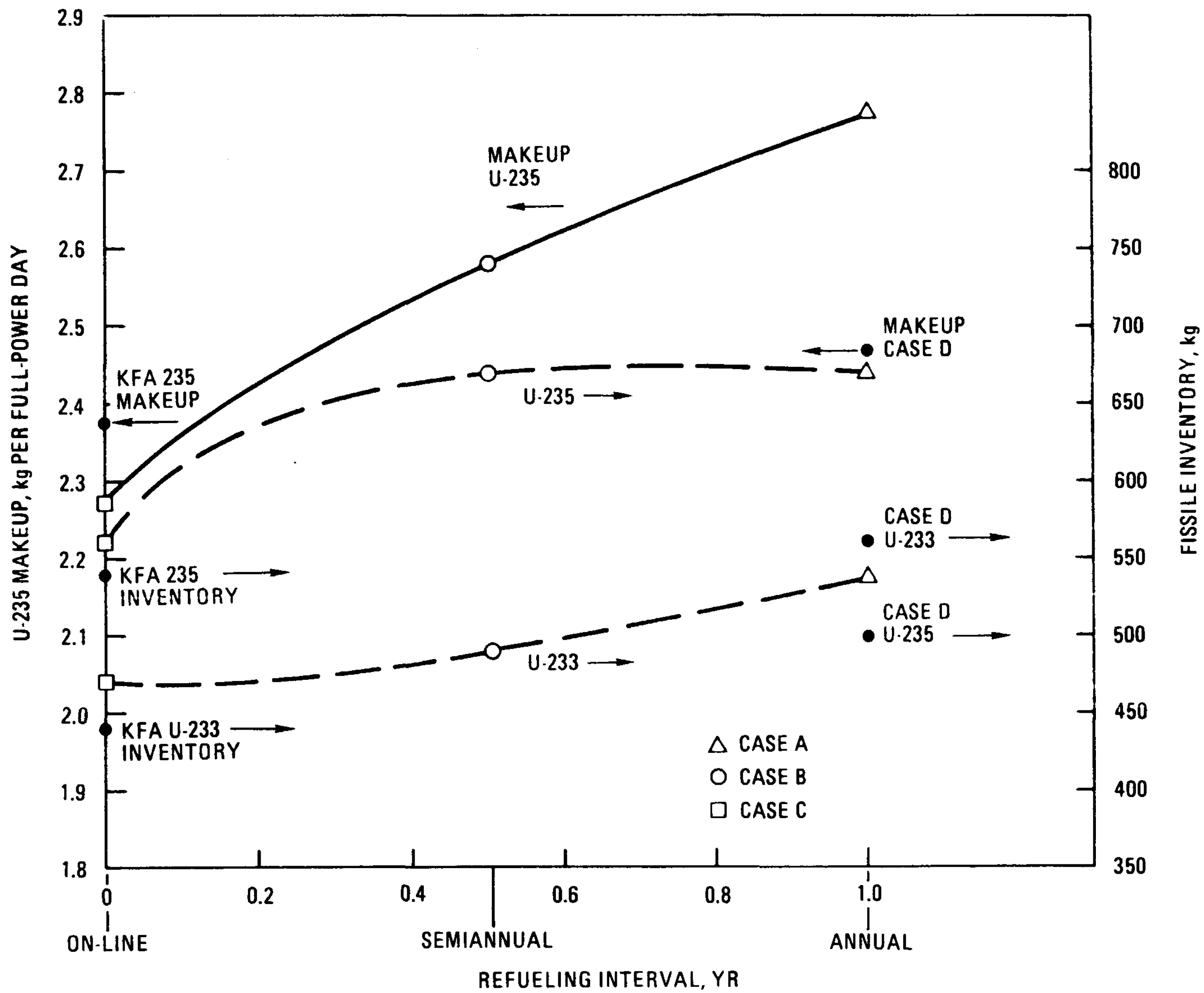

Fig. 3-16. Fissile makeup and fissile inventory versus refueling frequency, based on KFA 4011

Additionally, concerning the GA prediction of Case 4011, the thorium is more homogeneously distributed in the pebble bed ball configuration than it is in the prismatic block fuel pins; i.e., it is less lumped. The use of thorium cross sections based on the more lumped configuration will lead to an underestimate of the thorium reaction rate and, for a nonrecycle assumption, to an underestimate of the U-235 feed requirement. This effect, although not yet quantified, will result in a higher U-235 makeup requirement and may improve the agreement with the KFA result.

These results indicate that the fuel utilization capabilities of the two designs are nearly equal. Similar results have been obtained for a Th/U-235 cycle at lower power density (Case 4021) and for a low enriched uranium cycle (Case 1013), as discussed below.

3.5.6 Comparisons for Reduced Power Density - KFA Case 4021

The second cycle investigated, KFA Case 4021, is similar to Case 4011 except it is a larger core of lower power density (5 w/cc) and has a lower neutron leakage rate (Fig. 3-15). The pebble bed configuration with on-line refueling has not been calculated, but an equivalent annually-refueled prismatic design has been calculated and compared to the KFA pebble bed prediction. The mass balance comparison for the two designs is summarized in Table 3-5.

The results show that at the reduced power density the prismatic design requires about 5% less U-235 feed than does the pebble bed.

3.5.7 Comparison for Low-Enriched Cycle - KFA Case 1013

A detailed comparison of exactly comparable cycles operating on the low enriched uranium cycle has not been performed. Results of calculations of cycles similar to KFA 1013 indicate that there will also be little difference between prismatic and pebble bed resource utilization in that cycle. This comparison is shown in Table 3-6 for an HTGR design and two similar pebble bed designs.

TABLE 3-5
COMPARISON OF MASS FLOW REQUIREMENTS FOR PEBBLE BED (KFA 4021) AND COMPARABLE PRISMATIC DESIGN

	Pebble Bed Case 4021[a]	Prismatic Block Design
Power Density, w/cc	5.0	5.0
Residence Time (Full Power Days)	1633	1633
Neutron Leakage, %	4.26	3.30
Refueling Frequency	On-Line	Annual
Conversion Ratio	0.625	0.661
C/HM Ratio	245	245
Fissile Material in Core, kg		
U-233	813	902
U-235	701	758
Pu-239	3	4
Pu-241	2	2
Charged per Full Power Day		
Heavy Metal, kg/d	29.34	33.48
U-233, kg/d	--	--
U-235, kg/d	2.11	2.01
Discharged per Full Power Day		
Heavy Metal, kg/d	26.13	30.33
U-233, kg/d	0.583	0.657
U-235, kg/d	0.122	0.109
Pu-239, kg/d	0.002	0.003
Pu-241, kg/d	0.001	0.002

[a] From Ref. 3-15.

TABLE 3-6
LEU FUEL CYCLE CHARACTERISTICS, PEBBLE VS PRISMATIC

	HTGR LEU C	Pebble 1 (KFA 1013)	Pebble 2
Thermal Power, MW(t)	3000	3000	1000
Electrical Power, MW(e)	1200	1200	400
Efficiency, %	40	40	40
Power Density, kW/l	8.4	9.0	8.0
Capacity Factor, %	80	80	80
Fuel Lifetime, yr	3	2.15	2.90
Reload Frequency, yr	1	On-Line	On-Line
Burnup, GWD/tonne	106.2	100.1	115.0
Carbon/Fertile Atom Ratio	350	355	355
Conversion Ratio	0.52	0.549	0.54
Initial Enrichment, %	5.4	—	—
Avg. Reload Enrichment, %	10.4	9.71	10.9
Avg. Discharge Enrichment, %	2.3	2.2	—
Equilibrium Cycle Loadings, kg/yr			
Loaded			
U Total	8278.0	8689.9	2453.8
U-235	860.4	835.1	267.5
Discharged			
U Total	7190.0	—	—
U-235	166.3	169.9	—
Pu Total	150.8	∿157	—
Pu Fissile	90.7	94.0	—
Total Heavy Metal	7340.7	7740.9	—
Annual Uranium Requirements			
Gross			
U_3O_8, ST/MW(e)	0.200	0.194	0.185
SWU, MT/MW(e)	0.137	0.131	0.128
Credit in Spent Fuel			
U_3O_8, ST/MW(e)	0.050	0.053	—
SWU, ST/MW(e)	0.028	0.026	—
Net			
U_3O_8, ST/MW(e)	0.15	0.141	—
SWU, ST/MW(e)	0.109	0.105	—

3.5.8 Near Breeders and Potential Net-Breeding

One of the advantages of the HTGR over alternative designs is the low fissile makeup required. This is primarily because of the conversion of fertile Th-232 to fissile U-233 during exposure. Recycle of residual bred U-233 from previous discharges further reduces fresh fissile requirements.

Since the conversion ratio (fissile produced/fissile consumed) for the HTGR can be made slightly greater than unity, the potential exists for a design which breeds. Such a design would, in the asymptotic condition, have zero makeup requirements.

In general, the conversion ratio is increased by enlarging the thorium loading through reducing either the carbon-to-thorium (C/Th) ratio or the power density. However, increasing the thorium loading requires increasing the fissile inventory and results in the economic penalty of increased fuel and handling costs. Added reprocessing and refabrication losses negate part of the reduction in resource consumption resulting from increased conversion.

To quantify these effects, near-breeding HTGRs have been investigated to determine the characteristics which minimize fissile consumption and to identify breeders possibly requiring no net fissile makeup. Similar studies have been performed for the pebble bed concept (Ref. 3-16).

Two key parameters varied to investigate near-breeding conditions are power density and C/Th ratio. Constraints applied in the selection of cases were:

1. 3 w/cc for minimim power density, reflecting minimum exposure and limiting energy costs.

2. 60 as minimum C/Th ratio, limited by available fuel volume.

These compare with 7.14 w/cc and C/Th = 180 for current steam-cycle HTGR plant design and 5.5 w/cc with C/Th = 355 for the current PNP pebble bed core.

A number of base cases were analyzed with varied power densities and C/Th ratios based on the following consumptions:

1. Recycle of all uranium with 2-1/2% total losses in reprocessing and fabrication.

2. Four year residence time.

3. Makeup feed from HTGR discharge (71% U-233, 20% U-234, 7% U-235, and 2% U-236), based on average fuel composition in the reprocessing pool.

4. 3360 MW(t) with 70% capacity factor and annual cycle.

5. No credit taken for possible breeding in a fertile blanket.

Following the calculation of base cases, the first four assumptions were modified to estimate effects on fissile consumption.

A necessary requirement for breeding is a conversion ratio greater than 1.0. This in itself is not a sufficient condition for breeding, since two major effects, reprocessing and fabrication losses and increasing fissile inventory caused by U-236 buildup, are not incorporated in the conversion ratio definition. With this qualification, the conversion ratio is an important criterion in selecting cases with breeding potential. Tabulating conversion ratio against power density and C/Th ratio allows estimating the conditions for a conversion ratio of 1.0. Some representative conditions are:

1. 4.7 w/cc; C/Th = 60
2. 4.0 w/cc; C/Th = 72

3. 3.4 w/cc; C/Th = 90

4. 3.0 w/cc; C/Th = 105

Several additional cases with conversion ratios greater than unity were also identified. These include, at a conversion ratio of about 1.01, a 3 w/cc and C/Th = 90 case and a 4 w/cc, C/Th = 60 case. A 3 w/cc case with C/Th = 60 has a conversion ratio of approximately 1.032. Figure 3-17 illustrates the effect of varied thorium loadings on conversion ratio.

After application of the conversion ratio criterion, the asymptotic fissile makeup requirement was used to determine if any design was actually breeding (zero makeup). Under the conditions stated above, the lowest fissile makeup requirement is 56 kg per reload. This minimum occurs at 3 w/cc with the C/Th ratio approximately 105. Significantly, this minimum occurs for a case with a conversion ratio of 1.00, rather than at the maximum conversion ratio of the cases considered. Makeup requirements are presented in Fig. 3-18.

Further analysis demonstrates why the increase in conversion ratio to CR = 1.032 for the extreme case of 3 w/cc, C/Th - 60 does not lead to minimum makeup requirements. For two cases at 3 w/cc, the first with C/Th = 60, the second with C/Th = 105, in general the fissile makeup requirements has three components:

1. A gain of fissile material proportional to the excess of the conversion ratio above unity, available for recycled reloads.

2. A loss of recycled material in reprocessing and fabrication.

3. An increase in fissile inventory to compensate for U-236 buildup.

The conversion ratio of 1.032 in the C/Th = 60 case produces about 34 kg of fissile fuel annually. The fissile discharge to recycle is nearly 4500 kg per reload, with 114 kg in reprocessing and fabrication losses. The inventory increase from U-236 is about 1 kg/yr. A makeup requirement of 114 + 1 - 34 = 81 kg results.

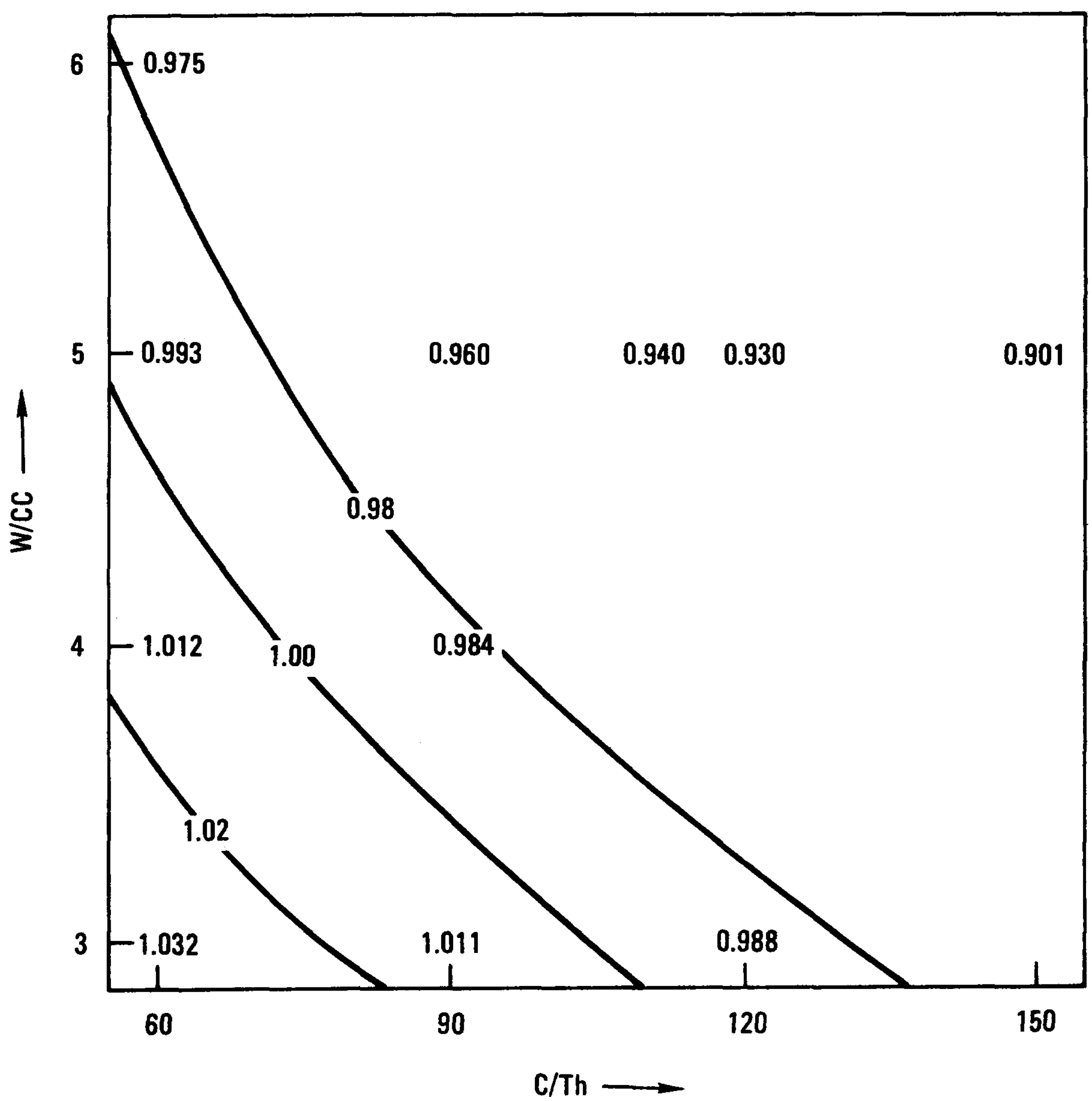

Fig. 3-17. Average conversion ratio dependence on power density and metal loadings (4-yr residence times)

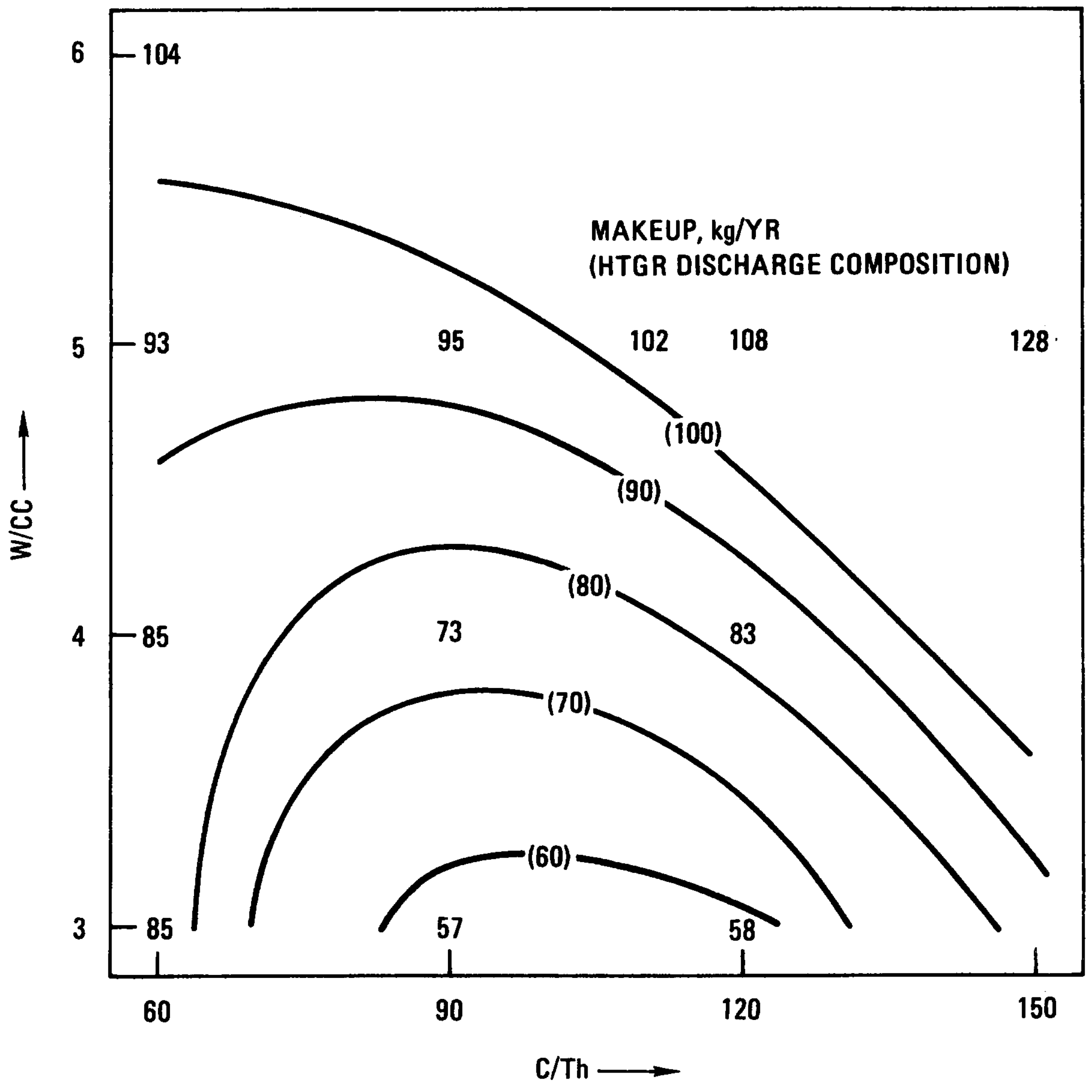

Fig. 3-18. Asymptotic annual fissile makeup (kg/yr) requirements for 4-yr cycles assuming 2-1/2% losses

For the C/Th = 105 case, the fissile discharge is only 1800 kg with losses of 46 kg per reload. The inventory increase is 10 kg/yr. Since the conversion ratio is 1.00, the annual fissile makeup requirement is 46 + 10 = 56 kg.

These results demonstrate that in near-breeding cases with 2-1/2% out-of-core losses, the most significant trend is that the losses from increased inventory dominate the gain from added conversion ratio as thorium loading increases. Base case requirements are presented in Fig. 3-19.

Since recycle losses dominate the near-breeding cases, the effect of decreasing those losses was analyzed. The assumption of 99% recycle (1% loss) of all fissile discharge is still insufficient for zero makeup in any case. For the highest conversion ratio (3 w/cc, C/Th = 60, Cr = 1.032), 99.3% recycle (0.7% loss) is the minimum for zero makeup.

These results indicate that no true breeder can be found given the original assumptions. Consequently, the task is to identify conditions leading to minimum fissile consumption, assuming 2-1/2% loss out-of-core.

The base case conditions include four-yr residence time. Variation of this parameter has conflicting effects on makeup requirements. Decreasing residence time improves the conversion ratio but increases fissile losses by reprocessing more fuel each cycle. Increasing residence time provides a larger average fission product inventory, requiring a greater fissile inventory and decreased conversion while reducing recycle loss. Since recycle losses dominate in near-breeding cases, longer residence times tend to reduce makeup requirements. Figures 3-20 and 3-21 demonstrate the effects of varying residence time from several base case conditions.

The 30-yr net fissile requirements were used as an indicator of fissile consumption. This is the 30-yr fissile requirement (initial core plus reload makeup) less the early core fissile inventory, as reprocessed. (This accounts for fission product buildup but does not credit inventory increase from U-236 buildup.) On this basis, a six-yr residence has an advantage.

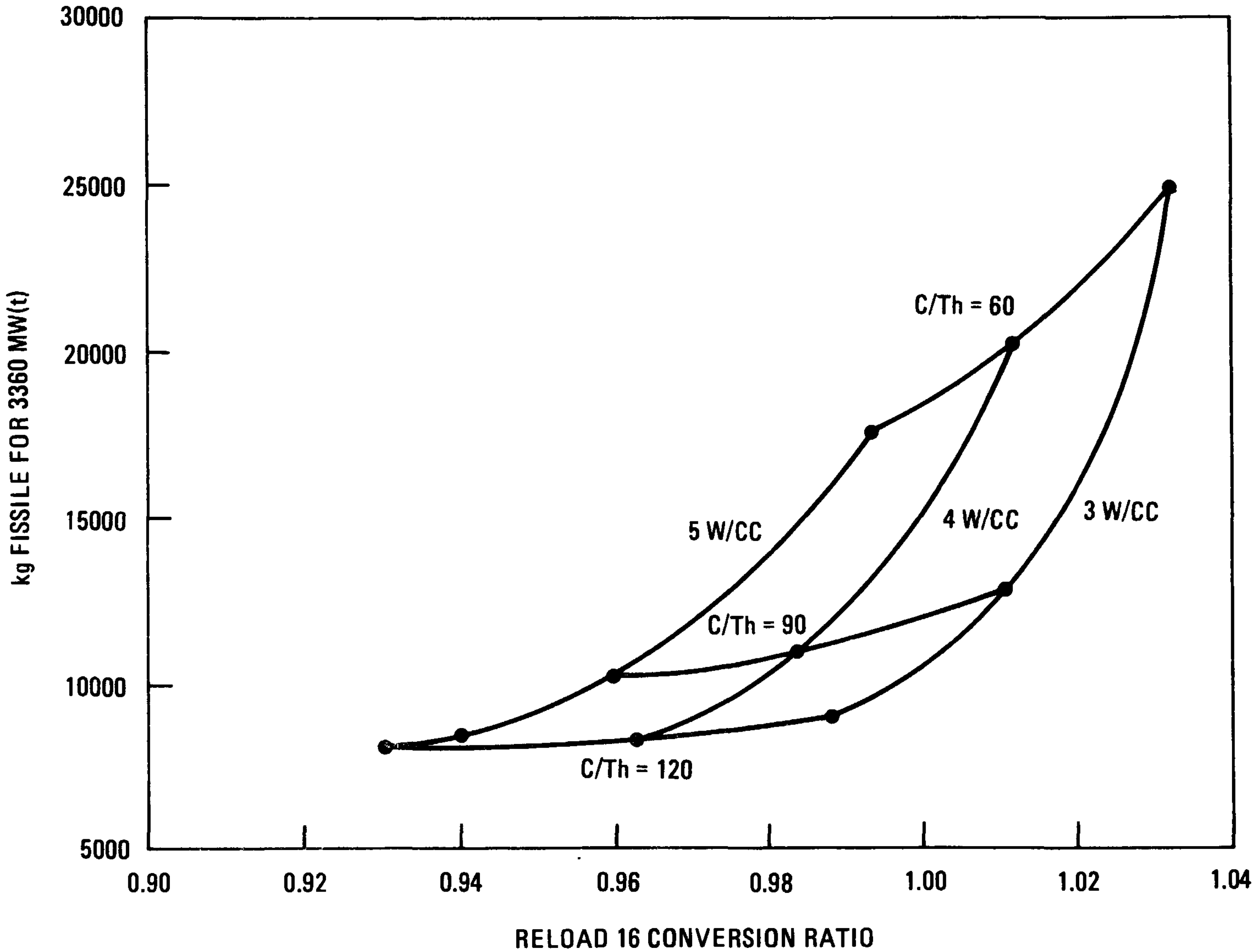

Fig. 3-19. 30-yr fissile requirements versus conversion ratio

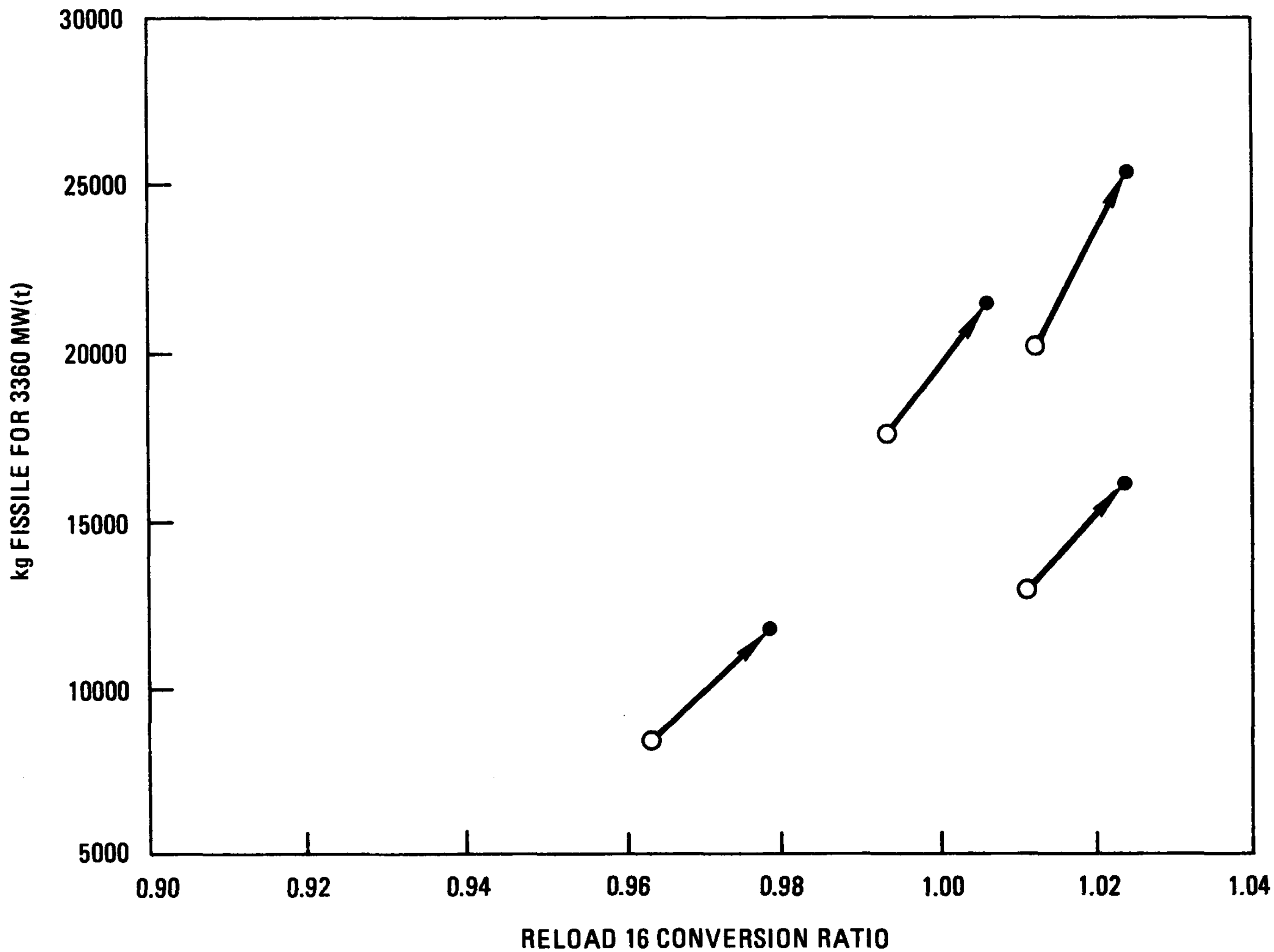

Fig. 3-20. 30-yr fissile requirements versus conversion ratio (effect of going from 4-yr$^{(0)}$ to 2-yr$^{(x)}$ residence)

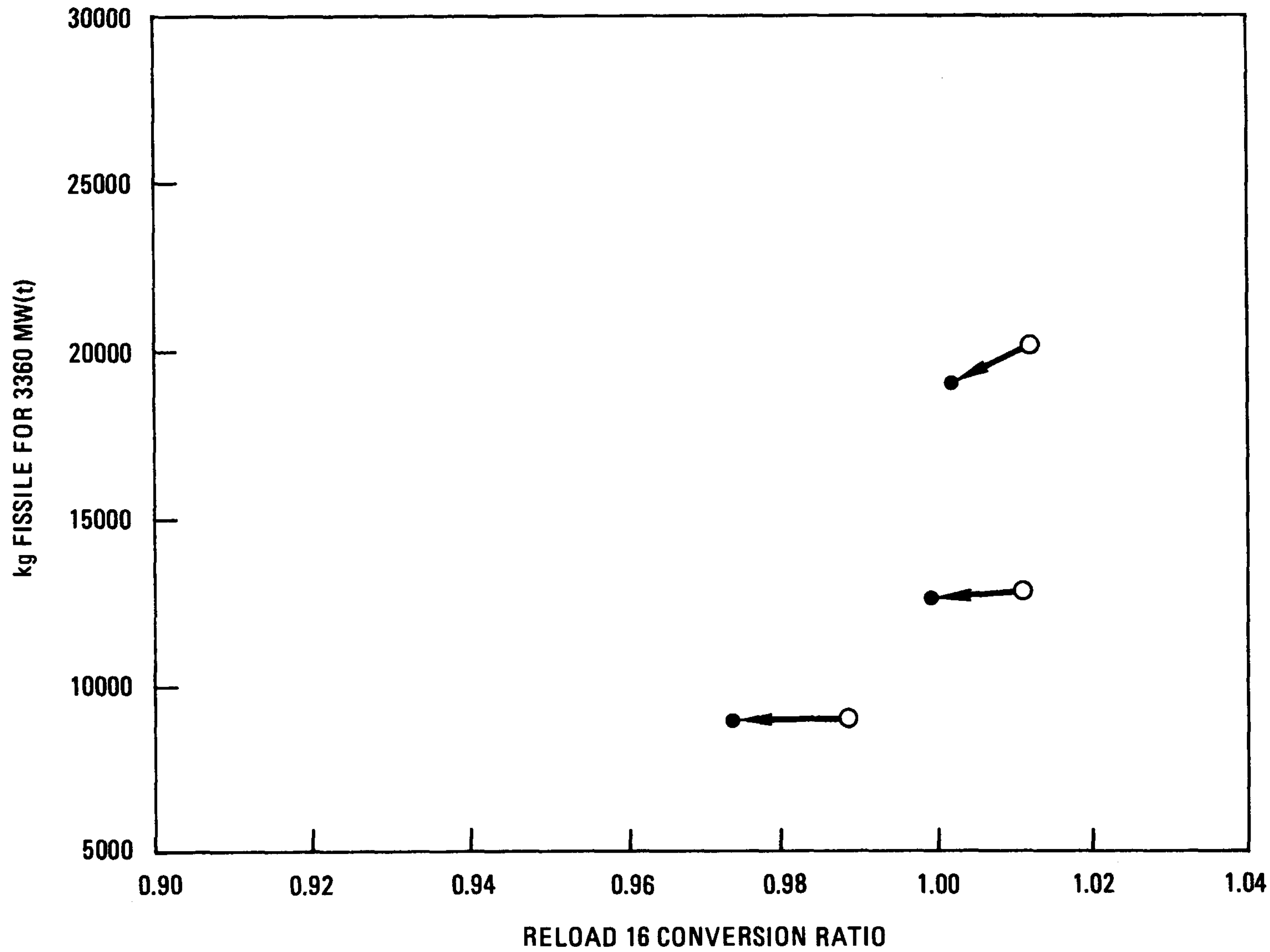

Fig. 3-21. 30-yr fissile requirements versus conversion ratio (effect of going from 4-yr (O) to 2-yr (x) residence)

The potential contribution of a fertile blanket may be estimated by noting that axial leakage is 2% and radial leakage is 0.6% to 1.0% (as power density varies from 3 w/cc to 5 w/cc). For $\eta = 2.2$, assuming 50% leaked neutrons captured in the blanket, an axial blanket adds (0.5)(0.02)(2.2) = 0.022 to the conversion ratio, or 21.8 kg/yr. The radial blanket adds $(0.5)\left(\frac{\text{power density}}{7\ \text{w/cc}}\right)(0.015)(2.2) = 0.007$ at 3 w/cc or 0.012 at 5 w/cc, providing 7.0 kg/yr at 3 w/cc to 11.7 kg fissile annually at 5 w/cc for recycle. Although this contribution could decrease makeup requirements, minimum consumption conditions would not shift. In addition, this fissile increase is not free, since the effect of the core reflector is reduced, requiring a somewhat larger core fissile inventory.

A significant result is that, assuming 2-1/2% losses, six-yr residence time is preferred, decreasing the cost penalty associated with minimizing fissile consumption. A second result is that whereas the highest conversion ratio is at a C/Th ratio of 60, which might be attained only with a new particle design, the lowest fissile consumption cases have C/Th ratios near that of the current SC-HTGR plant design. Third, because buildup of U-236 causes a fissile inventory increase, little or no long-term advantage is gained by using feed particularly enriched in U-233, such as that from a GCFR blanket or an HTGR with no recycle. Figure 3-22 demonstrates the small advantage for one case. Finally, with decreased recycling losses, minimum net fissile requirements would be characterized by reducing C/Th ratio.

To summarize the preliminary results of this study, the cycle with the lowest net consumption appears to be 3 w/cc with C/Th in the range of 90 to 120 with a six-yr residence time. The longer residence time has two advantages. First, fissile losses in reprocessing and refabrication are decreased, with little penalty in reduced conversion ratio. Second, fuel cycle costs are reduced. The fuel cycle costs for some four-yr cases are given in Fig. 3-23 as an indication of the cost penalty associated with some high conversion systems.

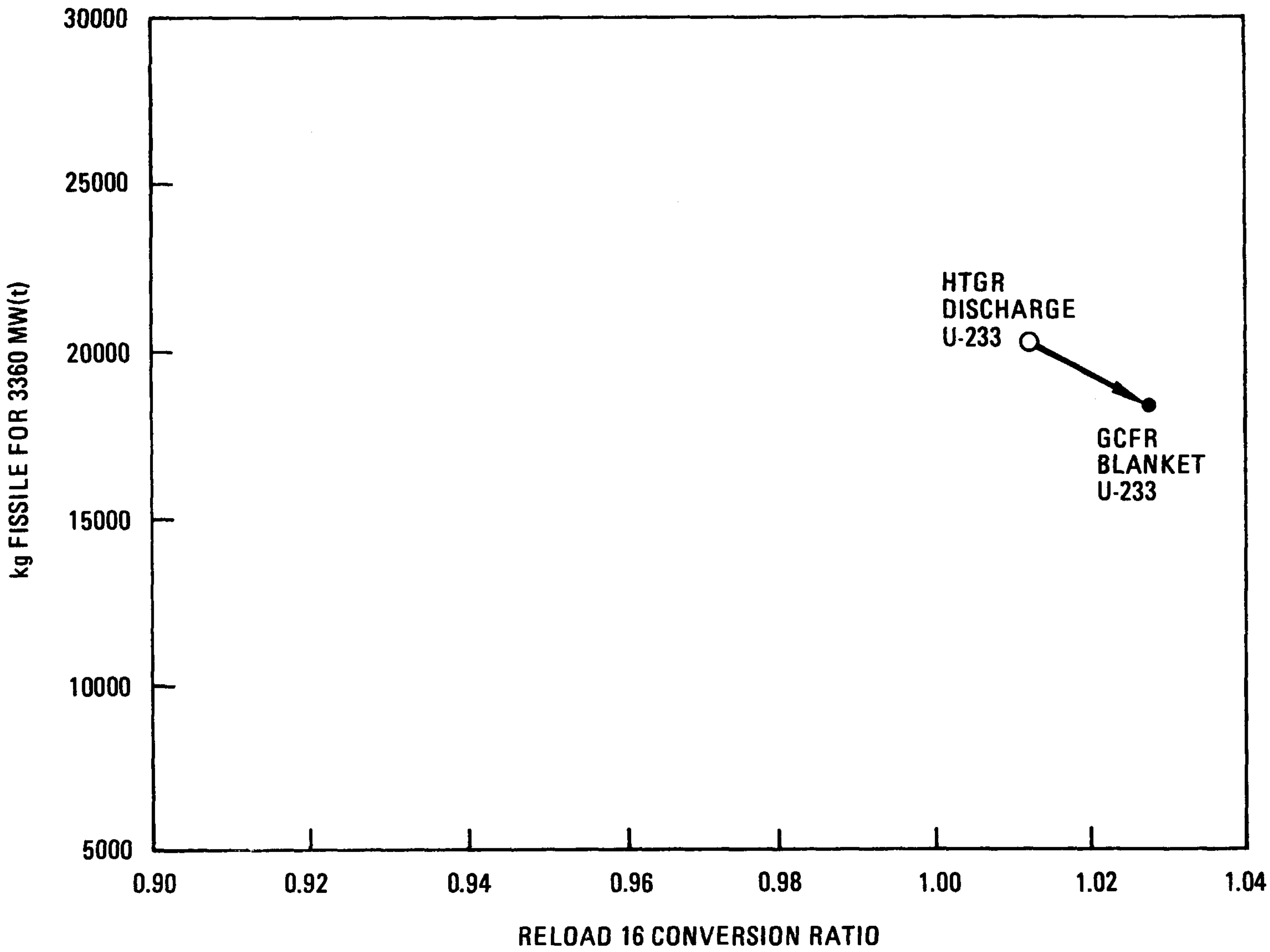

Fig. 3-22. 30-yr fissile requirements versus conversion ratio (effect of changing U-233 enrichment in feed)

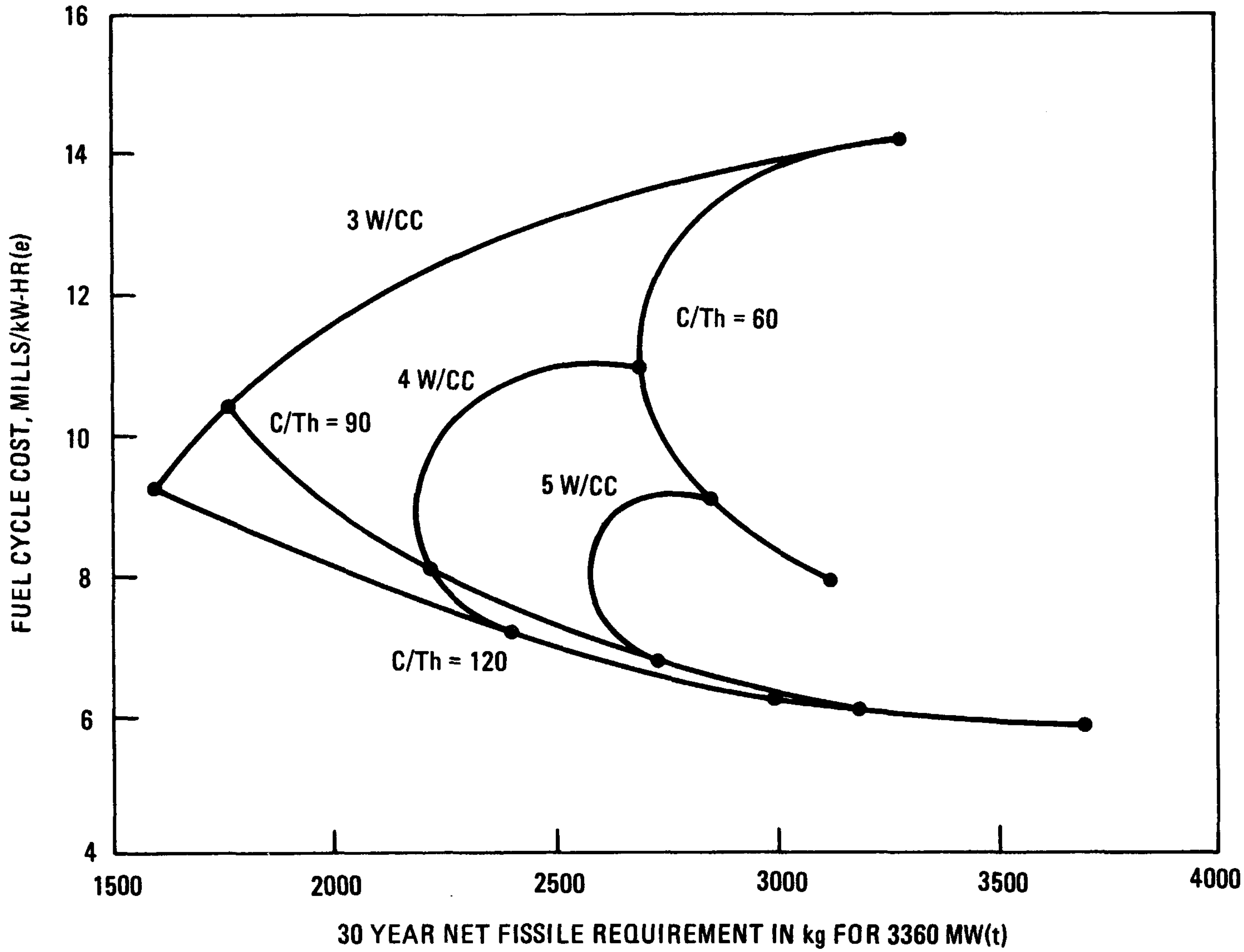

Fig. 3-23. 15-yr leveled-out fuel cycle cost versus 30-yr net requirement (all 4-yr cases)

3.6 SAFETY AND LICENSING

3.6.1 Licensability of the Pebble Bed Reactor (HTR-K) in the U.S.

During the period covered by this report, FRG efforts on the HTR-K included conceptual design and decisions input evaluations. Detailed safety analyses were not within the scope of either aspect. Therefore this study was confined to the evaluation of concept descriptions, outline drawings, and preliminary numerical data.

An essential part of the information base for formulating options concerning U.S. licensability of a German design is a knowledge of the similarities and differences between the U.S. licensing criteria and the criteria under which the design was developed. The first portion of this evaluation therefore is a survey of the most important aspects of the U.S. and German safety and licensing requirements; it will provide a familiarity with most of the basic requirements of the two countries helpful in understanding the conclusions regarding the licensability of the various design characteristics.

The remainder of the licensability study is a series of evaluations of specific design characteristics and systems. In each instance, the characteristic or system is evaluated in terms of its acceptability under the U.S. criteria and a conclusion is drawn as to its licensability. For aspects of the design which do not meet U.S. criteria, design changes or documented analyses which might make the design acceptable are discussed.

3.6.2 Comparison of U.S. and German Safety/Licensing Requirements

The basic safety/licensing philosophy in the U.S. has been termed "defense in depth." This philosophy dictates designs as inherently safe as practically achievable, multiple barriers against the release of radioactivity to the environment, and a multiplicity of safety devices to protect the public and plant from equipment failure, human error, and natural phenomena. The German philosophy is essentially the same. With few exceptions, differences in application of this philosophy are only of degree, with the German requirements being at least as stringent as those imposed in the U.S.

The Germans have not yet developed a set of codes, standards, and regulatory guides or a standard format for safety analysis reports and standard review plans as comprehensive as those in the U.S. However, a major effort for the development of standards is under way. Much of this work draws heavily on similar U.S. Standards efforts.

As evidenced by the safe operation of their nuclear power plants, both the U.S. and Germany have developed effective procedures for judging the safety of proposed designs. Because the safety records of reactor operation in the two countries are excellent, neither regulatory group can be persuaded to deviate from its present procedures without well documented evidence that the change will not increase public risk. Their reluctance to alter the present applications of philosophy is an important similarity between the two regulatory bodies.

Aspects of the safety/licensing evaluations carried out in Germany and the U.S. that are discussed below have been selected because they are major factors in the determination of system and component performance requirements.

3.6.2.1 General Requirements. Both the U.S. and German regulatory bodies have published general design criteria for nuclear power plants. These are the least detailed of the regulatory requirements but establish the basic design philosophy. The U.S. requirements are found in 10 CFR 50, Appendix A, "General Design Criteria." The German counterpart was published in 1974 by the Federal Ministry of the Interior (BMI) and the Reactor Safety Advisory Commission (RSK).

3.6.2.2 Safety/Licensing Requirements Documents.

Mechanical Stresses and Loading Combinations

The stress limits and loading combinations which must be used in the analysis of component response to accident situations are better defined in the U.S. than in Germany. Much effort has been expended in the U.S. to develop a set of rules for combining loads and determining the allowable stress levels for accidents. The NRC Regulatory Guides and the ASME Code are statements of the U.S. requirements. There are as yet no written stress criteria to guide German designers. The acceptability of each calculated stress is assessed based on engineering judgment and accepted practice.

Evidence indicates that the Germans may design the HTR-K plant to the U.S. ASME criteria. The THTR, which was designed using the German approach, has been analyzed using the ASME Code. The results of this analysis showed that the ASME Code was satisfied if the vessel life was decreased slightly and a few operating procedures altered. This is an indication of the compatibility of the design methods and criteria used in the two countries.

Until the German design work is performed using the ASME criteria and guidance from the NRC regulatory guides, it can be expected that the NRC will require both identification of the design differences that arise from application of different criteria and demonstration of the ability of the design to meet U.S. criteria.

Earthquakes

Appendix A of 10 CFR 100, "Reactor Site Criteria," specifies two different earthquake magnitudes that must be considered for design in the U.S. The smaller of these is the Operating Basis Earthquake (OBE) and the larger is the Safe Shutdown Earthquake (SSE). The German Institute for Reactor Safety and other German nuclear experts have defined two different earthquake levels for consideration in design, the Design Basis Earthquake (DBE) and the larger magnitude Safe Shutdown Earthquake (SSE). The two sets of definitions of earthquakes are strikingly similar. The OBE and DBE are defined to be reasonably expected occurrences during the life of the plant. The safety-related portions of the plant must be designed so that the stresses to which they are subjected during the DBE/OBE can be accommodated with wide margins. The magnitude of the SSE is defined so as not to be expected during the life of the plant. Both U.S. and German criteria for safety-related component response to the SSE require that these components maintain the ability to perform the required safety functions so that shutdown and cooling of the reactor can be initiated and maintained without exceeding the allowable stress and dose limits. Specific U.S. procedures for determining the magnitude of the SSE are given in 10 CFR 100; in Germany the statistically calculated occurrence rate of the SSE must not exceed that of the Design Basis Accident.* In addition, the Germans have adopted the NRC earthquake design spectra presented in Regulatory Guide 1.60 for both the horizontal and vertical components of ground acceleration.

The most significant difference between the U.S. and German loading combinations used in stress analysis is related to earthquakes. Germany is in an area of relatively low seismic activity, and German designs are not required to combine earthquakes with the loading combinations used for accident analysis. On the other hand, NRC rules require that earthquake loads be added to the other loads which result from accidents.

Should the licensing of a German reactor system design be attempted in the U.S., analysis will have to demonstrate the adequacy of the German systems under the more severe loading combinations required in the U.S.

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*Because the German criteria were written for LWRs, it is assumed that the Design Basis Accident is the Loss of Coolant Accident.

Radiation Dose to the Public

United States limits for public exposure to radiation for normal operation and accidents are given in 10 CFR 20, "Standards for Protection Against Radiation," Appendix I, to 10 CFR 50, the numerical guides for as-low-as-reasonably-achievable, and 10 CFR 100, "Reactor Site Criteria." The U.S. limits for release of radioactive materials in liquids and gaseous effluents are specified in 10 CFR 20 and 10 CFR 50.

German limits are given in documents published by the Institute of Reactor Safety (IRS), the Federal Ministry for Internal Affairs (BMI), the Reactor Safety Advisory Committee (RSA), and working groups of the various nuclear bodies.

A comparison of the limits for whole body and thyroid exposures shows that the German limits are lower for the accident situations but slightly higher for normal operations:

Permissible Dose Per Reactor

	Whole Body	Thyroid
Normal Operation		
Germany	30 mrem/yr-gas 30 mrem/yr-liquid	90 mrem/yr
U.S.	5 mrem/yr-gas 3 mrem/yr-liquid	15 mrem/yr
Accidents		
Germany	5 rem/event	15 rem/event
U.S.*	25 rem/event	300 rem/event

*These limits are reduced to 20 rem and 150 rem at the Preliminary Safety Analysis Report stage of the licensing procedure.

Unlike the practice in the U.S. and in addition to the absolute limits given above, the Germans (IRS) require that the accident doses be made as low as practicable. What this means in terms of hardware is difficult to predict, because the Germans have no criteria by which to judge the practicality of various means of dose reduction.

The difference between the allowable doses for normal operation in the two countries is not anticipated to be a major barrier to licensing a German design in the U.S. Calculated doses are very site-dependent, and it is not expected that the doses from the HTR-K will exceed those from the HTGR.

3.6.2.3 Safety Analysis Ground Rules.

Single Failure

Current U.S. safety criteria including several General Design Criteria, Regulatory Guides, and numerous industry documents require that safety-related systems be designed to be single-failure-proof. This means that a system required to function in response to an accident must be capable of performing its safety function assuming an active or passive failure of one of its components. For the most part, the single failures assumed in system analyses have been limited to failures of active components. Well defined criteria on where and when passive failures have to be assumed have not been published by the NRC.

The German single failure criteria are the same as those of the U.S. for active components. Also, the German criteria specifically require that each possible accident sequence be evaluated to determine if imposition of an arbitrary passive failure is warranted. As a result of these evaluations, the Germans have deemed it necessary to include the assumption of a passive failure in many accident sequences. The more conservative German approach to the application of single failure will facilitate licensing of the HTR-K design in the U.S.

Systems Under Repair

All German safety analysis is done assuming that one loop of a required safety system is out of service for repair. This assumption, in combination with the single failure criteria, necessitates additional redundancy in safety-related systems, so that an accident may be accommodated with the assumed loss of two loops (one out for repair and another eliminated by a single failure). This combination of capacity degradation from repair and arbitrary failure has been termed the N minus two (N-2) criterion. There is no similar criterion in the U.S. The additional redundancy required by the N-2 criterion would facilitate U.S. licensability.

Consequential Failures

Accident propogation assumptions concerning the consequential failure of equipment due to the dynamic and environmental effects of accidents are the same in both the U.S. and Germany. Safety equipment must be protected against missiles, whipping pipes, jet impingement, etc., and hardened against the effects of hostile accident environments.

The German guideline for consequential system piping failures is that any system piping that experiences accident loads in excess of those experienced during normal operation and that is not inspectable during service can be assumed to fail in an accident situation. Enforcement of this general rule is tempered in some applications. For example, some reheater welds that are in compression during certain accidents do not have to be assumed to fail during those accidents. In other equipment such as the high stress areas near the edges of steam generator tube sheets and core auxiliary heat exchanger (CAHE) tubes and tube sheets, sufficient in-service inspection (including a pressure test for the CAHE) must be instituted prior to the exclusion of such components from the assumption of consequential failure.

The U.S. rules on consequential failures have evolved through a case by case approach, resulting in in-service inspection requirements less strict than those in Germany. The German mandate for the increased in-service inspection can only increase confidence in safe reactor operation and should enhance licensability in the U.S.

Plant Protection System (PPS) Failures

Engineering practices in the U.S. have led to the development of PPS designs with diverse backups for initiation of many safety-related system actions. In Germany, diverse PPS backups are a requirement.* Where diverse backup signals cannot be developed for a protection system action, the system must be upgraded with more detectors. The specifications for upgraded systems and the precise meaning of diverse have not been made clear to the German designers for all cases. When detailed requirements are delineated, they may cause design problems for the moisture monitoring system and the radiation detectors in the hot reheat line.

For the purposes of German safety analysis, the plant must be capable of recovering safely from any type of accident, assuming failure of the primary PPS initiating signal. In the U.S., the PPS functions are evaluated on a system by system basis and there is no general rule requiring a diverse backup for each PPS-initiated action.

If the PPS system for the HTR-K can meet the German diversity criteria, it should be licensable in the U.S.

Credit for Nonsafety-Related Equipment

The NRC has not clearly defined rules for assessing the availability of nonsafety-related systems during accident situations. However, the HTGR is designed to be capable of safe recovery from any credible accident

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*Safety-related interlocks such as the one which prevents simultaneous closure of both pressure vessel relief trains and the one which prohibits the simultaneous withdrawal of two or more control rods are excluded from this diversity requirement.

using safety-related equipment only. German nuclear review groups, in particular, Technical Safety Control Services (TUV), have been explicit in their requirements and generally allow no credit for safety-related systems.

The similarity between the U.S. and German design and analysis philosophy for the use of nonsafety-related equipment precludes U.S. licensing problems.

3.6.2.4 Accident Sequences.

Initiating Events

The set of initiating events selected for analysis in Germany is similar to that analyzed in the U.S. In both countries, a spectrum of internal events ranging from anticipated transients to the low probability large breaks in a PCIV/PCRV penetration is investigated. Because of the cooperation between German and American engineers during the development of the HTGR in the U.S., many of the accidents assumed for the German design are identical to those for the HTGR. However, there are a few differences. For example, a minor difference in selection of pipe break locations exists because of the guidance given U.S. designers by Regulatory Guide 1.46. This guide limits the locations at which pipe failures have to be assumed to those of highest stress or cumulative usage. No such guidance has been given by German authorities. Consequently, they may assume pipe breaks at any location.

In both Germany and the U.S., the assumption of catastrophic failures in components such as steam generators may be precluded by in-service inspection. However, even though such techniques are judged to be sufficient to reduce the probability of large failures to levels which do not warrant consideration, the probable smaller breaks in these components must be assumed.

Also considered for initiating events in the U.S. and Germany are various external events such as natural phenomena, sabotage, explosion, and airplane crash. Natural phenomena include winds, floods, and storms. There are site-specific factors which must be defined for each proposed reactor installation. Explosions from activities near nuclear plants are also very site-specific and are evaluated in both countries.

The U.S. authorities have approved protection against saboteurs primarily through administrative controls which reduce the opportunity for industrial sabotage and by guards at the perimeter of the plant. Some of the administrative controls employed are monitoring and separation of vital equipment and restrictions of access to vital areas of the plant. One of the German proposals and the more recent U.S. proposals for plant security emphasize more physical protection of the inner portions of the plant site, such as the reactor building and emergency equipment. In the German plan, the reactor building design and security system would ensure that forced entry is difficult and time consuming; redundant emergency equipment would be physically separated and in some cases bunkered. In addition, the Germans are considering the institution of secret provisions for plant surveillance. Information regarding the German development of criteria for sabotage protection is very difficult to obtain because this TUV-sponsored work is kept secret.

Airplane crash at sites in Germany is much more likely than at U.S. sites. To date, selected U.S. sites have a probability of impact by heavy aircraft so low that it need not be considered in design. However, the density of air traffic in Germany is high enough to dictate that the design of the reactor containment building, control room, and other safety-related structures be capable of protecting vital systems and components from damage from aircraft. The design basis for aircraft crash is a Phantom II jet traveling at 215 m/sec. The allowable damage from such a crash is breech of the integrity of the containment, but no failure of the primary coolant system boundary or safety-related shutdown and cooling equipment. Considerations related to airplane impact forces and crash debris also determine separation and bunkering requirements for safety-related equipment outside the containment.

Combination of Initiating Events and Failures

Some of the most important aspects of the safety analysis of any accident situation are the assumptions concerning the availability of systems which respond to signals generated by the abnormal event. The HTGR is so designed that any initiating event may be accommodated assuming the loss of all systems not designed to operate in the post-accident environment, all systems which are disabled by other aspects of the accident, and all nonsafety-related systems. In addition, a single active component failure is assumed in the systems required to respond to the accident. For accidents requiring reactor trip, the rod with the highest worth is assumed to stick in the withdrawn position and off-site power is assumed to be lost.

The required assumptions for German safety analysis include those used for the HTGR and the following:

1. Passive (or active) failure in systems which respond to the accident.

2. Failure of the primary reactor protection system initiation signal.

3. One train of a redundant safety-related system out of service for repair.

The U.S. and German general rules listed above are sometimes altered for specific accident sequences. For example, the German authorities do not require that a single failure be postulated for plant response to an airplane crash. In some cases deviation from the general rules may be justified by quantitative or qualitative evaluations of the likelihood of occurrence of a given accident sequence. To date, this technique has been used sporadically in the U.S. and efforts to develop more exact and consistent methods of accident sequence probability evaluations are currently under way in both countries. A great advance in this area was accomplished by the work performed for WASH 1400, the Reactor Safety Study.

3.6.2.5 Example of an Accident Sequence Required for Analysis. As indicated in the discussion of combining initiating events and failures, the German ground rules for safety analysis will probably require the assumption of the unavilability of more safety-related equipment than will the U.S. rules. The following example illustrates the differences between U.S. and German analyses.

Design Basis Depressurization Accident (DBDA)

The HTGR sequence has a nonmechanistic 100 sq in. hole in a PCRV penetration, reactor scram (with most reactive rod not moving) on low PCRV pressure or the backup high containment pressure signal, loss of main loop cooling, loss of off-site power, and a single active failure preventing operation of one CACS loop. In addition, the load combination used in the stress analysis includes loads from a postulated SSE along with the normal loads and accident loads.

The HTR-K sequence contains all the features listed above except for the treatment of the SSE, and adds the following: consequential 90 lbm/sec leak in a steam generator, failure of moisture monitor system to dump the contents of the leaking steam generator (thereby allowing the entire contents to empty into the core), and one CACS loop out of service for repair.

The German authorities believe that the assumption of a consequential steam generator leak is justified because it has not been demonstrated that the stresses which result from the DBDA are low enough to ensure that the steam generator will not fail. The failure of the moisture monitors is justified because they are not designed to function in the rapidly changing abnormal flows and pressures that develop during a DBDA.

General characteristics of the German safety analysis and the requirements to which they lead are highlighted by the example discussed above. The Germans are quite concerned about consequential failures and tend to be very conservative in the assumptions relating to them. The addition of one more CACS loop to the U.S. design is clearly mandatory because of the N-2 criterion.

Impact on Licensability

The foregoing discussion indicates that there is only one aspect of German safety analysis which will present a licensing problem in the U.S. In all other respects the German rules are at least as conservative as those in the U.S. The unacceptable aspect is the exclusion of earthquake loads from the calculation of stresses from accidents. That the earthquake loads are not included in the analysis does not necessarily mean that the HTR-K design is inadequate but does mean that reanalysis must be performed prior to acceptance of the design by the NRC.

3.6.3 Requirements for Particular Systems

3.6.3.1 Shutdown Systems. General Design Criteria 26 and 27 from Appendix A of 10 CFR 50 require that U.S. reactors have two independent reactivity control systems of different design principles and that one of these systems be capable of holding the reactor core subcritical under cold conditions. In addition, the systems must ensure that reactivity changes under normal operation, anticipated operational occurrences, and accident conditions do not cause the design limits appropriate for each situation to be exceeded, with allowance made for malfunctions such as stuck rods. The Federal Republic of Germany's guidelines which deal with requirements for shutdown systems are essentially the same as General Design Criteria 26 and 27.

Although the written requirements for both countries are primarily directed toward LWRs, the requirement for two independent and diverse shutdown systems for all types of reactors seems inescapable. To meet the diversity requirement, LWRs use a combination of rods and liquid poison injection; the GA HTGR design has been licensed in the U.S. with one shutdown system using rods and a second employing boron pellets. Some of the proposals for the HTR-K shutdown systems have included the combination of a rod system with either the injection of a neutron poison or the insertion of small boron balls.

Several uncertainties concerning the use of these systems have been identified:

1. Insertion of rods deep enough into the pebble bed core for cold shutdown is difficult because of the high friction which exists in the dry helium atmosphere. However, predicted forces are within acceptable limits if ammonia is injected into the core during control rod insertion. Estimated corrosion from ammonia decomposition products is small according to FRG designs.

2. The introduction of BF_3 gas into the core has been considered and abandoned because it may lead to a corrosion problem from the fluorine created by dissociation and because boron removal and core restart may be difficult. However, the main concern and cause for elimination is the doubtful effectiveness of this system if there is a depressurization accident. Another concept whereby an aqueous solution of gadolinium acetate is sprayed into the core promises to be effective even after depressurization and is currently being investigated.

3. The insertion of small boron balls which fit between the fuel balls requires proof that a distribution which ensures shutdown will be achieved and requires devising a practical way for removing the poison balls. Also, during the design development there was concern about ensuring that a subcritical configuration would be maintained during seismic excitation of the pebble bed. Preliminary experiments indicate that this system will shut down the core adequately and that changes in distribution from seismic excitation will be acceptable.

However, the current reference shutdown system employs only control rods pushed into the core and reflector. In this scheme the rods are divided into two groups, one for hot shutdown and the other for cold shutdown. Each group of rods is inserted by a drive mechanism of a different operating principle, one hydraulic and the other pneumatic. The Germans are investigating the use of an electric-motor-driven, threaded rod which is screwed into the pebble bed. This type of rod reduces the forces on both the rods and balls.

For typical LWR and HTGR reactors the rods experience only small structural loads during insertion. Thus even under loss of flow transients, control rod stability during insertion is not a concern. However, particularly for the process heat pebble bed core, this may not be the case under loss of flow conditions. The combination of high core temperature and large structural forces on the control rods requires a step insertion procedure. Initially the rods may be inserted only a short distance to achieve hot subcritical; then after initiation of auxiliary cooling and partial core cooldown, ammonia will be injected and a further increment of rod insertion accomplished. This procedure continues until the rods are fully inserted.

Because the insertion of rods directly into the core has the potential to damage both the rods and fuel balls, it has been proposed that the number of rod insertions be minimized by reducing power by means other than scram. Power may be reduced in one of two ways, insertion of a small amount of negative reactivity via neutron absorber material or reduction of coolant flow so that the core temperature rise inserts negative reactivity via the temperature coefficient. Under equilibrium conditions, the HTR-K is designed to operate with only a few rods in the upper portion of the core. Control is maintained by reflector rods, rods in the upper plenum, or rods inserted only a short distance into the core. In the event of positive reactivity insertion accidents, some of these rods can be driven in to counteract the accident and reduce core power to a low level. For accidents other than reactivity insertion, power may be lowered by the reduction of coolant flow and the subsequent temperature rise. In either type of transient, the insertion of rods directly into the core is avoided, thereby increasing the reliability of the rod systems by decreasing the number of stress cycles. In addition, if the cause of the accident is quickly removed, this return to power promises to be simpler and less time consuming than the pulling of scrammed rods. Two types of accidents for which power reduction is not acceptable are moisture ingress and loss of forced cooling (including depressurization). In these cases, the reactor will be scrammed. Both of these events are of low probability and should not subject the rods and core to a high number of scram cycles.

The concept of not automatically scramming the reactor for many events which have in the past been handled by scram is of particular interest for licensing. In other core configurations a scram does not subject the core and rods to the severe stresses experienced in the pebble bed. Perhaps for this reason reactor vendors have accepted the inconvenience of scram and have made no concerted effort to avoid it. Recent ATWS studies conducted for the GA HTGR have contributed to understanding the inherent capabilities of power control using the negative temperature coëfficient for gas-cooled graphite reactors. Based on this knowledge, plant protection employing the concept of power reduction by means other than scram may be feasible and licensable for gas-cooled reactors in the U.S. Of course, the licensability of this concept will have to be based on analyses which demonstrate that the power reductions are adequate to meet appropriate limits for each event.

The shutdown system which the Germans have chosen for further development is the one employing hydraulic and pneumatic rods. Before this system can be deemed acceptable, sufficient diversity must be considered, along with potential for rod ejection accidents and reliability of the NH_3 injection system. Should a backup shutdown solution be needed, various organizations within Germany are working on screw rods, poison injection, and the introduction of small boron balls.

Because the German and U.S. guidelines for reactivity control systems are essentially identical, the shutdown system which is finally adopted for the HTR-K will probably meet the U.S. licensing requirements.

3.6.3.2 Core Instrumentation. Evidence that the U.S. regulatory bodies are interested in closely monitoring the performance of the core is given by General Design Criteria 12 and 13, which state:

> "The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions which exceed specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed." (GDC 12 - Suppression of Reactor Power Oscillations)

> "Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences and for accident conditions as appropriate to ensure adequate safety, including those variables and systems that can affect the fission process, the integrity of the core, the reactor coolant pressure boundary. . . ." (GDC 13 - Instrumentation and Control)

The intent of these GDCs is met by U.S. reactors through the use of in-core instrumentation which measures flux and temperature. For both LWRs and HTGRs neutron flux intruments which move within the core calibrate permanent flux intruments either in the core or in structures around the core. In PWRs and HTGRs the core outlet temperature is measured by thermocouples in the outlets of the fuel assemblies or refueling regions. These instruments allow the monitoring of power and temperature distributions throughout the core so that safe operation can be ensured. Although these systems are not classified as safety systems, NRC concern about their reliability has been clearly indicated by the requirement that HTGR core outlet thermocouples be redundant and separated.

The slowly flowing pebbles and homogeneous structure of the pebble-bed core make it impractical to use in-core instruments to measure core performance. Therefore, the designers of the HTR-K have developed alternative means for measuring core performance.

The German solution to the temperature measurement problem in the HTR-K is to mount thermocouples in the bottom reflector. Even though there are no core coolant channels to direct flow to these detectors, the flow patterns through the pebble bed reportedly assure that a local power anomaly in as few as 100 fuel balls can be detected. KFA, is currently developing a thermocouple which can be calibrated, in situ, by a noise thermometer. A system with this capability should meet all regulatory requirements if the expected detection sensitivity can be clearly supported.

For flux measurement, the Germans propose detectors in both the upper and side reflectors. The upper reflector instruments are two to four m from the highest power region of the core and are accessible for calibration. German calculations indicate that the system can perform the necessary safety functions associated with power transients as well as detect xenon oscillations during normal operation. A major problem in the calculation of the capabilities of this system is the modeling of the smoothing effects of the upper plenum void. Prior to this system's gaining NRC acceptance, all analytical calculations will undergo close NRC scrutiny, and predicted system performance may have to be verified experimentally.

Judging from the written criteria and current practice, the NRC is keenly interested in ensuring that reactors have adequate core instrumentation, and it will be difficult to license the HTR-K in the U.S. without temperature and flux monitoring systems which the NRC feels are equivalent to those of the HTGR.

3.6.3.3 Reactor Vessel. At the outset of this pebble bed reactor study, the Germans were proposing the use of a PCIV instead of the more conventional PCRV. The PCIV concept includes an uninsulated metal liner cooled by primary helium coolant at the temperature of the core inlet. This scheme is significantly different from that employed in the HTGR PCRV. In the latter design, the liner is cooled by a system of water pipes in the concrete vessel, outside but next to the liner, and the liner is well insulated from direct contact with the primary helium coolant.

Advantages claimed for the PCIV are relative ease of in-service inspection and repairability of the liner. Both of these considerations are of importance in Germany because of the interests of regulatory and utility groups. Equivalent levels of interest have not been shown in the U.S.

A possible disadvantage of the PCIV concept is that it may make the liner more vulnerable to damage during interruptions of coolant flow. Loss-of-cooling events will have to be analyzed to determine the temperatures to which the liner is driven and the time available to restore cooling with the CACS before unacceptable damage occurs.

In general, the PCIV concept does not appear to be unlicensable in the U.S. However, an uninsulated liner, cooled by primary helium flow, would require a great deal of design and development. Detailed definition of the liner cooling flow paths and analyses of use of CACS for cooling in the event of accidents would have to be completed before this highly important safety-related structure could be proved to perform satisfactorily under all postulated conditions. This work should be considered a prerequisite to licensability in the U.S.

German evaluations of development work necessary to finalize a practical, licensable PCIV indicated a two to three year schedule penalty with increased technical risk. Therefore, the PCRV with a liner cooling system was chosen as the reference design for the HTR-K plant.

3.6.3.4 Reflector. Graphite that receives high fast neutron exposure at high temperature exhibits the phenomenon of shrinkage followed by expansion. This behavior is a special problem for the upper two m of the side reflector of the HTR-K, where the high power tilt toward the top of the core causes a high fast-neutron dose. Doses of concern extend to a depth of about ten cm into the reflector. The calculated doses indicate that the graphite behavior cannot be confidently predicted; i.e., the doses are in a range in which the graphite failure mechanism is not well understood. Some of the solutions for increasing the useful life of the reflector are:

1. Developing new types of graphite which can take the dose without failure.

2. Limiting the graphite failure to tolerable amount.

Graphite development programs are under way in Germany, but the final results of the work will require several years. Therefore, the reflector design which has been selected for further development work is a limited failure concept in which slots are cut in the face of the reflector so that pieces of graphite which become weakened and fall off will have a size which will not disrupt the ball flow and will not jam the ball removal mechanism. Some designers feel that the loose graphite chunks will be ground up by the motion of the pebbles and thus have essentially no effect on ball motion. However, if this is the case, the introduction of graphite dust into the system may be come a significant problem.

All parties working on the problem agree that the final core and vessel designs should include provisions for reflector replacement and in-service inspection. The NRC has accepted replaceability for the HTGR reflector blocks which receive high fast-neutron doses. Therefore, the provision for side reflector replacement in the HTR-K should make any design concept licensable in the U.S. if it can accommodate in-service inspection.

The HTR-K reflector problems are not confined to the side reflector. The top reflector is also in an area which could receive a high fast-neutron dose. However, during normal operation control rods will be inserted into the upper reflector, somewhat reducing the dose to this graphite. The detrimental effects are also reduced because this graphite operates at a lower temperature than that at the sides. A second aspect of the top reflector dose problem concerns the tensile rods by which the reflector blocks are suspended. GA calculations had indicated a potential problem from the radiation doses experienced by these metallic components, should they be constructed from materials covered by the ASME code. Discussions with the HRB engineers indicated their agreement with GA's calculated doses, but the metal they propose to use for the tensile members is a German nickel alloy which can withstand the high radiation. This material is currently not covered by the ASME code and its use in the U.S. would be predicated on its acceptance by ASME or NRC.

3.6.4 Quantitative Safety Analyses

3.6.4.1 Maximum Hypothetical Fission Product Release (MHFPR). The physical and operational characteristics of the HTR-K which differ from those of the HTGR may lead to differences in the doses resulting from postulated accidents. To investigate this possible variation in dose, the most severe accident postulated in the U.S., the MHFPR, was evaluated for both reactor types.

The accident sequence analyzed is that used for the siting dose calculations required by 10 CFR 100. The combination of failures postulated for the analysis has an extremely low probability of occurrence but is selected because it gives an upper bound for all accidents with reasonable probabilities of occurrence.

The sequence is:

1. A rupture in the primary system pressure boundary which causes a rapid depressurization of the system.

2. Total failure of all systems supplying forced circulation of cooling gas through the core.

The core undergoes an unrestricted heatup, which causes failure of the fuel particles and subsequent release of the contained radioactivity. For the HTGR, this accident has been analyzed in detail. The fuel failure as a function of temperature, the diffusion of radioactive nuclides through the graphite matrix, and the transport of the nuclides from the reactor vessel to the containment have been modeled in the analysis. Similar models for the pebble bed core and fuel are not available to GA at this time. Therefore, a simplified, conservative analysis of the MHFPR for the HTR-K was performed. The results of this simplified analysis are much higher than those which would be obtained from an analysis similar to that used for the HTGR. In order to judge the degree of conservatism introduced by the simplified analysis and to obtain results for the HTGR which could be compared to those for the HTR-K, the HTGR was also evaluated using the simplified model.

The simplified analyses employed a one-node, adiabatic core heat-up calculation. The average fuel-temperature-versus-time results from these analyses were then used to predict the fuel failure rate, based on known HTGR BISO-fuel particle behavior. At this stage, the lack of data on spherical fuel element activity release characteristics was overcome by assuming no holdup in the fuel and no holdup in the reactor vessel. In essence, as soon as the fuel failed, its contained activity appeared in the containment. The containment model, meteorology, site boundary distance, and low population zone distance were assumed to be equal to those employed in dose calculations for current HTGR design work.

The results of the simplified calculations show that the HTGR and HTR-K doses are essentially equal except for the 30-day thyroid dose. For this dose, the HTR-K exposure is lower by a factor of approximately three. This difference arises because the OTTO fuel cycle used in the HTR-K has mean fuel residence times of approximately three years, whereas the reference lead plant HTGR fuel residence time is on the order of four years. As a result of the longer residence time (which is not an inherent characteristic, but an economic choice), the HTGR fuel contains a higher inventory of fission products and therefore causes a faster core heatup following shutdown. The more rapid temperature rise causes the volatile fission products (such as iodine) to be released sooner from the HTGR core and results in higher doses. Table 3-7 gives the results of three analyses:

1. The simplified HTR-K analysis.
2. The simplified HTGR analysis.
3. The highly sophisticated, state-of-the-art HTGR analysis.

TABLE 3-7
3000 MW(t) MHFPR DOSE SUMMARY

	HTGR		HTR-K	
	Simplified	State-of-the-Art	Simplified	Expected, State-of-the-Art
0 to 2-hr Doses[(a)] (rem)				
Whole Body Gamma	0.04	0.0005	0.04	0.0005
Thyroid	3	0.05	3	0.05
Bone	5	0.0002	5	0.0002
0 to 30-day Doses[(b)] (rem)				
Whole Body Gamma	4.8	1	4.4	0.9
Thyroid	270	60	96	21
Bone	30	2	26	2

[(a)] Exclusion Area Boundary = 425 m.

[(b)] Low Population Zone = 1600 m.

The fourth set of data included in Table 3-7 is the HTR-K doses that are expected to result from an analysis similar to the state-of-the-art HTGR analysis. These doses were calculated by multiplying the simplified HTR-K results by the ratio of the state-of-the-art to the simplified HTGR results.

It is important to note that the results of the simplified analyses for both reactor types are below the 10 CFR 100 limits, even though the analysis methods are very conservative.

3.6.4.2 CACS Design. The analysis of CACS designs in the U.S. and FRG are performed using the RECCA (HTGR) and THERMIX (HTR-K) codes. Because of differences between the codes, a direct comparison of their results may not give an accurate indication of any differences that may exist between the CACS systems of the two reactor plants. The best measure of the advantage that one design might have over the other would be a comparison of the results of detailed analyses of both reactors using the same code. In the absence of such analyses, the CACS evaluation in this study has been limited to consideration of the major factors which affect its design.

The CACS designers must prevent reverse flow through the core from natural convection to ensure that the core can be adequately cooled. Although CACS design is dependent on many factors, three which have great impact are the physical configuration of the core, the core temperature and power distributions, and the system pressure. The flatter the power and temperature distributions and the lower the system pressure (less dense coolant) are, the less likely the phenomenon of reverse natural convection flow. Also, a taller core with a well defined flow channel is more conducive to natural convection.

Comparison of the two reactor systems on the basis of these factors has led to the following conclusions:

1. The apparent temperature profile advantage of the OTTO cycle HTR-K because of its flatter radial power distribution and hence

flatter temperature distribution is offset by the variable flow control orifices of the HTGR. These orifices control the distribution of cooling flow delivered to the core and permit operator control over the core temperature profile during both normal operation and CACS cooldown. Thus, the inherent uniform HTR-K temperature distribution has been essentially duplicated in the HTGR by an engineered control feature.

2. If the final HTR-K design employs low system pressure (40 bar),* an advantage over the HTGR prismatic core (50 bar) will result from decreased thermal buoyancy effects in the less dense coolant. Also, buoyancy effects will be less pronounced in the HTR-K because of the shorter core height.

In addition to reverse flow, CACS designers are interested in the rate at which the core heats up after shutdown. As previously discussed, the decay heat of the HTR-K is lower (approximately 15%) than that of the HTGR because of the difference in average fuel residence times in the core. The other major core characteristic which affects the rate of core temperature rise is core heat capacity per unit of power generated. This parameter may be expressed to a close approximation in kilograms of graphite per megawatt. Shown in Table 3-8 are the values of this parameter and other related parameters for four different reactors. The results show that the low power density of the pebble bed core is completely negated by its relatively high void fraction. The heat capacities of the large reactors are nearly identical, and the small Fort St. Vrain (FSV) HTGR has a significant advantage over the THTR. Combining the effects of decay heat and heat capacity indicates that the large HTGR should heat up approximately 15% faster than the large HTR-K, whereas the THTR should heat up approximately 8% faster than the FSV HTGR.

*Final design increased to 60 bar.

TABLE 3-8
SPECIFIC HEAT OF HTGR AND HTR-K

	HTR-K - THTR	HTGR - FSV	Large HTR-K	Large HTGR
Thermal Power, MW	750	840	3000	3200
Power Density, MW/m^3	6.0	6.3	5.5	6.8
Void Fraction, %	39	21	39	∿24
Graphite Density, kg/m^3	1067	1387	1067	∿1330
Graphite Density, kg/MW	178	220	194	∿196
Normalized Power Density, $MW/10^3$ kg graphite	5.6	4.54	5.15	∿5.11

3.6.5 Conclusions Related to HTR-K Licensability in the U.S.*

The HTR-K large plant design has not progressed to the stage at which details for all systems have been developed. In spite of this limitation, the review of design work being performed in the FRG and licensing practices of the German regulatory bodies has led to the following conclusions:

1. The basic safety criteria and general regulations of Germany are for the most part very similar to those found in the U.S. Code of Federal Regulations, Title 10.

2. The allowable public radiation doses for normal operations are higher in Germany than in the U.S. However, there is no reason to expect that the HTR-K will not be capable of meeting the U.S. criteria.

3. The greatest potential for HTR-K licensing problems in the U.S. stems from the difference in loading combinations used for accident stress analysis. The Germans do not include earthquake loads in the stress analysis for accidents, whereas the inclusion of earthquake loads is a U.S. requirement. In general, the German design practices lead to margins which are comparable to those inherent in designs based on the U.S. ASME Boiler and Pressure Vessel Code. This comparability of margins plus evidence that the Germans may adopt design practices similar to those associated with the U.S. ASME Code promises to minimize the licensing problems related to the differences in loading combinations.

4. The German safety analysis assumptions concerning equipment availability, postulated failures, consequential failures, and initiating events are at least as conservative as those in the U.S. A reactor design which satisfies the German safety analysis requirements should have no U.S. licensing problems.

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*References 3-17 through 3-19.

5. In general, it can be expected that the HTR-K design which satisfies all German criteria for an operating license will be acceptable under U.S. criteria with little or no change.

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