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GAS REACTOR INTERNATIONAL COOPERATIVE PROGRAM INTERIM REPORT

German Pebble Bed Reactor Design and Technology Review

September 1978

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

General Electric Company
Advanced Reactor Systems Department
Sunnyvale, California

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GAS REACTOR INTERNATIONAL COOPERATIVE PROGRAM
INTERIM REPORT

GERMAN PEBBLE BED REACTOR
DESIGN AND TECHNOLOGY REVIEW

GENERAL ELECTRIC

ADVANCED REACTOR SYSTEMS DEPARTMENT
310 DeGUIGNE DRIVE
SUNNYVALE, CALIFORNIA 94086

September 1978

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

ABSTRACT

This report describes and evaluates several gas-cooled reactor plant concepts under development within the Federal Republic of Germany (FRG). The concepts, based upon the use of a proven Pebble Bed Reactor (PBR) fuel element design, include nuclear heat generation for chemical processes and electrical power generation. Processes under consideration for the nuclear process heat plant (PNP) include hydrogasification of coal, steam gasification of coal, combined process, and long-distance chemical heat transportation. The electric plant emphasized in the report is the steam turbine cycle (HTR-K), although the gas turbine cycle (HHT) is also discussed. The study, performed by the General Electric Company, is a detailed description and evaluation of the nuclear portion of the various plants. The chemical plants of the various PNP concepts were outside the scope of the assigned work. The general conclusions are that the PBR technology is sound and that the HTR-K and PNP plant concepts appear to be achievable through appropriate continuing development programs, most of which are either under way or planned.

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

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SECTION 1

INTRODUCTION

1.1 BACKGROUND

Among tasks assigned to General Electric (GE) Company by the U.S. Department of Energy (DOE) was the requirement that GE perform a conceptual design review, a technology evaluation, and a review of proposed development plans for two specified West German gas-cooled reactor concepts:

- HTR-K: An electricity generating gas-cooled reactor that utilizes a pebble bed core and a conventional steam turbine secondary cycle.

- PNP: A process heat gas-cooled reactor that uses a pebble bed core and several alternative advanced chemical plants for heat utilization.

1.1.1 WORK STATEMENTS

The GE activities are part of an ongoing three-year US/FRG joint evaluation to provide the following:

Conceptual Design Review - After appropriate arrangements have been made through the BMFT, General Electric shall interface with KFA and other FRG organizations who may be involved with the PNP and HTR-K projects in order to develop and provide to DOE a description and understanding of the conceptual designs. In FY 1977 this activity is not anticipated to involve direct U.S. contributions to the FRG, HTR-K and PNP projects, but rather to assure that the U.S. industry and U.S. Government representatives on the Ad Hoc GCR Committee have an adequate understanding of the design and design bases which evolve from the projects. One purpose of this task is that the contractor, in future years, provide input to and directly influence the design of the PNP and HTR-K as they evolve, to the extent dictated by the results of the FY 1977 effort.

Technology Evaluation - General Electric shall, in conjunction with the FRG participants in the PNP and HTR-K projects, evaluate the technology base as it now exists for the PNP and HTR-K plants and identify in detail the technical problems which must be resolved in order to develop and commercialize the PNP and HTR-K systems. This assessment shall be done in sufficient detail to allow DOE representatives on the Ad Hoc GCR Committee to be fully informed regarding the status of the technology and the magnitude of the technical problems which must be resolved.

Development Plan - General Electric shall, in conjunction with the FRG organizations in the HTR-K and PNP projects, review and evaluate the proposed FRG development plans for the HTR-K and PNP. This review shall include consideration of cost, schedule, proposed scope of the development program, facilities requirements, and the proposed strategy for bringing the HTR-K and PNP into commercialization.

1.1.2 GERMAN PROGRAM ALTERATION

The work efforts described above were complicated by changes in German program emphasis occurring during FY'77. Two of the German industrial participants (HRB and BBC) proposed that the electricity generating plant selected for national development be the direct cycle helium turbine plant (HHT) powered by a pebble bed reactor, instead of the HTR-K specified in the DOE work statement. Work on the HTR-K project in Germany was essentially terminated in July 1977. A final decision by the German Government and electric utilities will be made during 1978 whether to pursue HTR-K or HHT.

Consequences of this change in emphasis included a lack of detailed information being made available on the HTR-K plant and an increased interest in the HHT concept. Therefore, a more detailed analysis of the PNP designs than of the HTR-K designs was performed. Also, a general description and generic evaluation of the HHT are provided in Section 6.

A description and evaluation of the bases for the apparent German decision to pursue the HHT, instead of the HTR-K, is beyond the scope of this topical report.

1.1.3 SOURCES OF INFORMATION

In order to review and evaluate the HTR-K and PNP plant concepts, three primary sources of information were utilized. The "PNP Project Status Report for the End of the Concept Phase" (1) provided much of the fundamental data on the PNP. The secondary sources of information were the numerous unpublished handouts and survey documents generated by the PNP and HTR-K projects. Finally, a two-week evaluation trip to Germany resulted in significant technical data being received in the form of viewgraphs and meeting notes. The latter two sources provided the basis for the HTR-K discussions.

One other important source of information should be acknowledged. Dr. K. Kugeler of the Nuclear Research Center (KFA) Institute for Reactor Development in Jülich, West Germany, spent two weeks at GE and provided major assistance in the form of updated information and technical review of report drafts.

1.1.4 PNP CHEMICAL PLANT EXCLUSION

A Confidentiality Agreement between the General Electric Company and the German participants implemented the overall agreement between the United States and the Federal Republic of Germany for gas reactor technology exchange. The Confidentiality Agreement specifically excluded access to technical information on the PNP chemical plants (i.e., gasifiers, gas cleanup systems, etc.). As a consequence, the PNP coverage of this report addresses only systems and components associated with the primary loop.

1.2 SUMMARY

The remainder of this topical report, with the exception of Section 6 on the HHT concept, examines the two German concepts (HTR-K and PNP) for utilizing the high temperature characteristics of the pebble bed gas-cooled reactor. The emphasis is nuclear plant design. Economics, fuel cycles, marketing and safety/licensing are discussed in depth in other reports.[20,21,22,23,&24]

Section 2 provides plant descriptions and overall evaluations for the HTR-K and the PNP concepts. The evaluations include descriptions of the technology bases and the proposed German development plans. Table 1-1 provides some principal features of the steam cycle electric plant (HTR-K) and three alternate process nuclear heat (PNP) plants under consideration. The sole output of the HTR-K plant is electrical power. The PNP plants produce methane, frequently called synthetic natural gas (SNG), and some surplus electrical power, using either coal hydrogasification (HKV), steam gasification (WKV), or a combination of HKV and WKV. The PNP can also be used for chemical heat pipe energy transport systems. All the plants employ a large 3000 MWth pebble bed reactor utilizing the Once-Through-Then-Out (OTTO) fuel cycle. In each case, the core is cooled by helium in a primary system composed of six parallel circuits, or loops. Each circuit contains, in addition to the core, the major components listed in the table, along with necessary interconnecting gas ducts.

Section 3 addresses the nuclear reactor design, including the core and related equipment, while Section 4 examines the primary system components in more depth. Both sections incorporate descriptions and specific component evaluations.

Section 5 looks at several auxiliary systems, such as the Fuel Handling System and the After-Heat Removal System and provides evaluations specific to those systems.

Section 6 describes the general character of the HHT concept and lists some of the generic issues associated with its development. Section 7 is a tabulation of major references.

TABLE 1-1

HTR-K AND PNP SUMMARY DATA

	Units	HTR-K	PNP, HKV	PNP, WKV	PNP, COMBO
Power	MWth	3000	3000	3000	3000
Output	-	electricity	SNG & electricity	SNG & electricity	SNG & electricity
Design Life	years	40	40	40	40
Fuel Cycle	-	OTTO	OTTO	OTTO	OTTO
Helium Temperature In/Out	^{o}C	260/700	300/950	300/950	300/950
Helium Pressure	bars	60	40	40	40
Helium Flow Rate	kg/s	1320	890	890	890
Core Height/Radius	m	5.5/5.6	5.5/5.6	5.5/5.6	5.5/5.6
Number of Primary Circuits	-	6	6	6	6
Primary Circuit Components	-	• steam gen. • circulator	• steam gen. • circulator • reformer	• circulator • intermediate heat exchanger	• circulator • intermediate heat exchanger

1.3 LIST OF ABRREVIATIONS

Table 1-2 is a listing of some of the important abbreviations used in this Topical Report. Note that the company abbreviations appear on the figures reproduced from German originals. The company responsible for the particular figure is circled.

TABLE 1-2

LIST OF ABBREVIATIONS

AVR	Arbeitsgemeinschaft Versuchsreaktor, GmbH; Experimental Reactor Consortium
BBC	Brown, Boveri, & Cie (Switzerland)
EVA	Einzelspaltrohr Versuchsanlage; Single Reformer Tube Test Facility
FRG	Federal Republic of Germany
GA	General Atomic Company (U.S.A.)
GHT	Gesellschaft für Hochtemperaturreaktor Technik mbH; Company for High Temperature Reactor Engineering
HHT	Hochtemperatur Reaktor mit Helium Turbine; High-Temperature Reactor with Helium Turbine (Direct Cycle Helium Turbine Plant with PBR)
HKV	Hydrierende Kohlevergasung; Hydrogasification of Coal PNP
HTGR	High-Temperature Gas-Cooled Reactor; (GA Steam Cycle Prismatic Design)
HTR-K	Hochtemperatur Reaktor-Kernkraftwerke; High-Temperature Reactor Nuclear Power Plant (FRG Steam Cycle Pebble Bed Design)
HRB	Hochtemperatur Reaktorbau, GmbH; High-Temperature Reactor Construction Company
KFA	Kernforschungsanlage, GmbH; Nuclear Research Center
KLAK	Kleine Absorberkugeln; Small Absorber Balls
NWA	Nachwärmeabfuhr System After-Heat Removal System
OTTO	Once-Through-Then-Out (Pebble Bed Fuel Management Scheme)
PBR	Pebble Bed Reactor
PNP	Prototypanlage Nukleare Prozesswärme; Prototype Nuclear Process Heat Plant
THTR	Thorium High-Temperature Reactor
WKV	Wasserdampf Kohlevergasung Steam Gasification of Coal PNP Plant

SECTION 2

OVERALL PLANT (DESCRIPTION AND EVALUATION)

As discussed in Section 1, this report is a design/technology evaluation of several German Pebble Bed Reactor concepts, specifically the electricity production steam cycle version (HTR-K) and the process heat production version (PNP), although some coverage of the gas turbine concept is provided in Section 6. This section describes the general characteristics of the HTR-K and the PNP, provides an overall evaluation, and briefly reviews the design conformance with safety criteria. Detailed discussions of components and systems is included in Sections 3, 4, and 5.

2.1 ELECTRICITY PRODUCTION PLANT (HTR-K)

2.1.1 GENERAL DATA

HRB has continued to develop the Pebble Bed concept and until recently was designing a 3000 MWth reactor, the HTR-K (Hochtemperatur Reaktor-Kernkraftwerke). Like the AVR and THTR, the HTR-K is a Pebble Bed Reactor and generates electricity with a standard steam cycle. The six-loop primary circuit with steam generators, circulators, and the core is housed in a prestressed concrete reactor vessel (PCRV).

A listing of the HTR-K characteristic data is provided on Table 2-1. The depth of GE analysis of the HTR-K concept during FY-77 was limited by the amount of technical data acquired from the German participants.

TABLE 2-1

HTR-K CHARACTERISTIC DATA

OVERALL PLANT

Primary Circuit

Reactor thermal power	3000 MW
Reactor coolant	Helium
average inlet temperature	260°C
average outlet temperature	700°C
pressure after circulator	60 bar
Total mass flow	1320 kg/s
Number of loops	6

Steam Plant

Power from steam generators	3025 MW
Total steam flow rate	1206 kg/s
Conditions at high pressure turbine inlet	
temperature	512°C
pressure	168.5 bar
Conditions at tap for reheat	
temperature	192°C
pressure	13.2 bar
Condenser pressure	0.09 bar
Gross electricity production	1190 MW
Hotel load	70 MW
Net electricity production	1120 MW
Net thermal efficiency	37.3%

Table 2-1 (Continued)

COMPONENTS

Reactor Pressure Vessel

Type	PCRV
Construction	Multiple cavity
Outside dimension	
height	31.6 m
diameter	37.4 m
Core cavity	
height	15.4 m
diameter	16.3
Steam generator cavity	
number	6
diameter	5.05/4.45 m
After-heat removal system cavities	
number	4
diameter	3.06/2.24 m
Pressure Relief Cavities	
number	6
diameter	2.50 m
Operating pressure	60 bar
Design pressure	66 bar

Main Circulator

Number	6
Type	Radial
Drive	Electric motor
Helium conditions at exhaust	
temperature	260°C
pressure	60 bar

Table 2-1 (Continued)

Flow rate	220 kg/s
Pressure rise	1.32 bar
Power (at motor terminals)	7.06 MW
Flow control	Throttling
Steam Generator	
Number of steam generators	6
Type of bundle	helical tube/straight tube
Thermal power output during normal operation	504.12 MW
Primary side:	
medium	helium
pressure	60 bar
maximum inlet temperature	700°C
outlet temperature	262°C
mass flow rate	221.7 kg/sec
Secondary side:	
live-steam pressure	175 bar
live-steam temperature	515°C
boiler feed-water temperature	185°C
mass flow rate	201 kg/s
Heating surface	3,959 m^2
Heat exchanger tube dimensions:	
Straight tube bundle (Incoloy 800)	25 x 4.8 mm
Helical tube bundle (10 Cr Mo 910)	22 x 3.45 mm

Table 2-1 (Continued)

NWA Heat Exchanger	
Design	U-tube bundles built as a box
Thermal power	31.6 MW
Primary side	
medium	76% He, 24% air
pressure	1.4 bar
Maximum inlet temperature	880°C
Outlet temperature	230°C
Flow rate	11.6 kg/s
Secondary side	
medium	H_2O
pressure	36 bar
inlet temperature	110°C
outlet temperature	148°C
flow rate	200 kg/s
Heat surface	920 m^2
NWA Circulator	
Type	Axial
Drive	Electric motor
Inlet temperature	207°C
Outlet pressure	1.4 bar
Flow rate	11.6 kg/s
Pressure rise	0.078 bar
Power at motor terminals	875 kW
Flow control	Speed

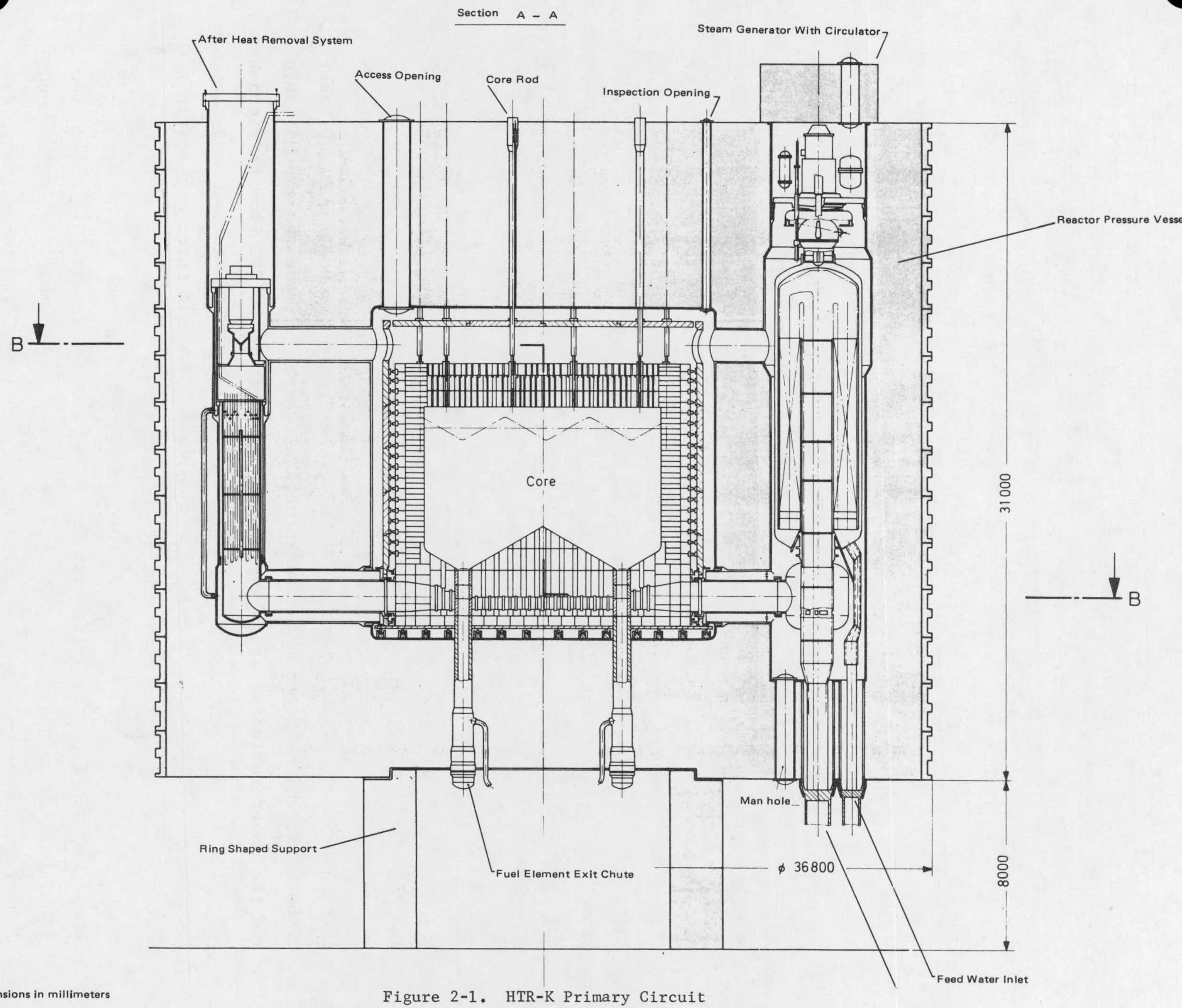
2.1.2 REACTOR CORE

The reactor, shown in Figure 2-1, has a cylindrical core, 11.2 m in diameter and 5.54 m in height, which contains approximately 3 x 10^6 spherical fuel elements. The fuel elements consist of coated particles of mixed Th-U oxide contained in a graphite matrix 5 cm in diameter. This matrix is surrounded by a fuel-free graphite shell 0.5 cm thick. In order to flatten the radial neutron flux profile, the core has two regions of different fissile loadings: an inner cylindrical region and an outer annular region which contains the fuel elements with a slightly higher fissile content. The fuel is burned in a once-through-then-out (OTTO) cycle and achieves an average burnup of 100,000 MWd/MT. In the OTTO cycle, the fuel elements enter at the top of the core and pass through only once with a residence time of approximately three years. After removal from the core, the fuel is not reloaded.

The core cavity is formed by graphite blocks which act as the neutron reflector. The top reflector contains penetrations for control rods and the inlet tubes of the fuel handling system, while the bottom reflector has penetrations for the fuel exit tubes and holes for the flow of cooling gas. Surrounding the side reflector graphite is a cast-iron thermal shield which supports the side reflector. Another cast-iron thermal shield is located above the top reflector.

2.1.3 PRIMARY CIRCUIT

The primary circuit has six loops (see Figure 2-1), each containing a circulator and a steam generator. The core is cooled by helium at a pressure of 60 bars, which enters the core through a chamber between the thermal shield and the upper reflector at an average temperature of 260°C. From this chamber, the main flow path is down through the upper reflector and control rods into the core. After flowing through the core, the gas exits through holes in the bottom reflector to the lower mixing chamber at an average temperature of 700°C. The hot gas then flows through the hot gas duct to the steam generator, where it transfers heat to the secondary circuit. It enters the circulator, which drives the gas into the upper gas chamber. A small amount of helium is diverted from the circulator to cool the side thermal shields and the hot gas ducts.



Note: All dimensions in millimeters

Figure 2-1. HTR-K Primary Circuit

The primary circulators are integrated slide-in units with electric drives and are installed vertically above the steam generators. During normal operation, each circulator requires 7.1 MW and drives 220 kg of cold helium per second with a pressure rise of 1.32 bars. Two methods of flow control were considered, variable-speed motors or a constant-speed motor with inlet control. The current reference design is inlet control.

The steam generators consist of a straight tube superheater, a helical tube evaporator, and a helical tube economizer. The heat load of each unit is 504 MWth. On the primary side, 700°C helium flowing at 220 kg/s enters a chamber below the steam generator and flows upward through the superheating section. The helium turns and passes down over the helical coils, where it is cooled to approximately 260°C. It then turns and flows upward to the circulator inlet. On the secondary side, the feedwater enters at 180°C, and superheated steam exits at 515°C and 175 bars.

The After-heat removal system, shown in Figure 2-1, is designed to remove 2% of the reactor's normal thermal power under depressurized conditions. It consists of four redundant loops, each capable of removing 1% of reactor power. The components inside the PCRV are the auxiliary circulators and heat exchangers.

2.1.4 CONTAINMENT STRUCTURES

2.1.4.1 Prestressed Concrete Reactor Vessel (PCRV)

The HTR-K is an integrated system in which the entire primary circuit (the core, steam generators, circulators, and the after heat removal components) is contained in a prestressed concrete reactor vessel (PCRV), as illustrated in Figure 2-1. The PCRV contains ten cavities, six with a steam generator and circulator and four for the after-heat removal components. In addition, there is a large central cavity for the reactor core. The positions and relative sizes of these cavities are illustrated in Figure 2-2.

Since concrete has much greater load carrying capability in compression than in tension, the concrete in the PCRV is prestressed to remain in compression under all expected operating or transient conditions. The vertical prestressing is provided by tendons, while the circumferential prestressing is done with wound cables.

A steel liner is attached to the walls of all the PCRV cavities as a gas leakage barrier. It is insulated on the inside with fiber insulation and cooled by water flow in pipes welded to it on the concrete side. During normal operation, the insulation and cooling maintain the concrete temperature at 66°C.

2.1.4.2 Reactor Protective Building

The reactor protective building surrounds the PCRV, as well as the fuel handling system and reactor maintenance equipment. It consists of a massive, reinforced concrete base plate and a prestressed concrete cylindrical shell which is designed to withstand aircraft impact and the pressure waves resulting from gas cloud explosions. The building is sealed by a steel liner located on the inside surface of the concrete. An access hatch to the reactor auxiliary building has been provided for transport of materials.

2.1.4.3 Reactor Auxiliary Building

The reactor auxiliary building is constructed from reinforced concrete and is designed to the same specifications as the reactor protective building. It contains storage areas for fresh and spent fuel elements. Besides these storage areas, the building has storage for control rods and hot cells for handling radioactive materials. Two rail connections to the reactor protective building are used for the transport of the fuel element carts.

2.1.5 CONTROL AND INSTRUMENTATION

2.1.5.1 Reactivity Control

The HTR-K reactivity control system contains 198 control rods which operate in the top reflector or in the core itself (core rods) and 48 control rods which travel in channels in the side reflector (reflector rods). These rods are split into two independent control systems, one containing 42 core rods and 24 reflector rods and the other containing

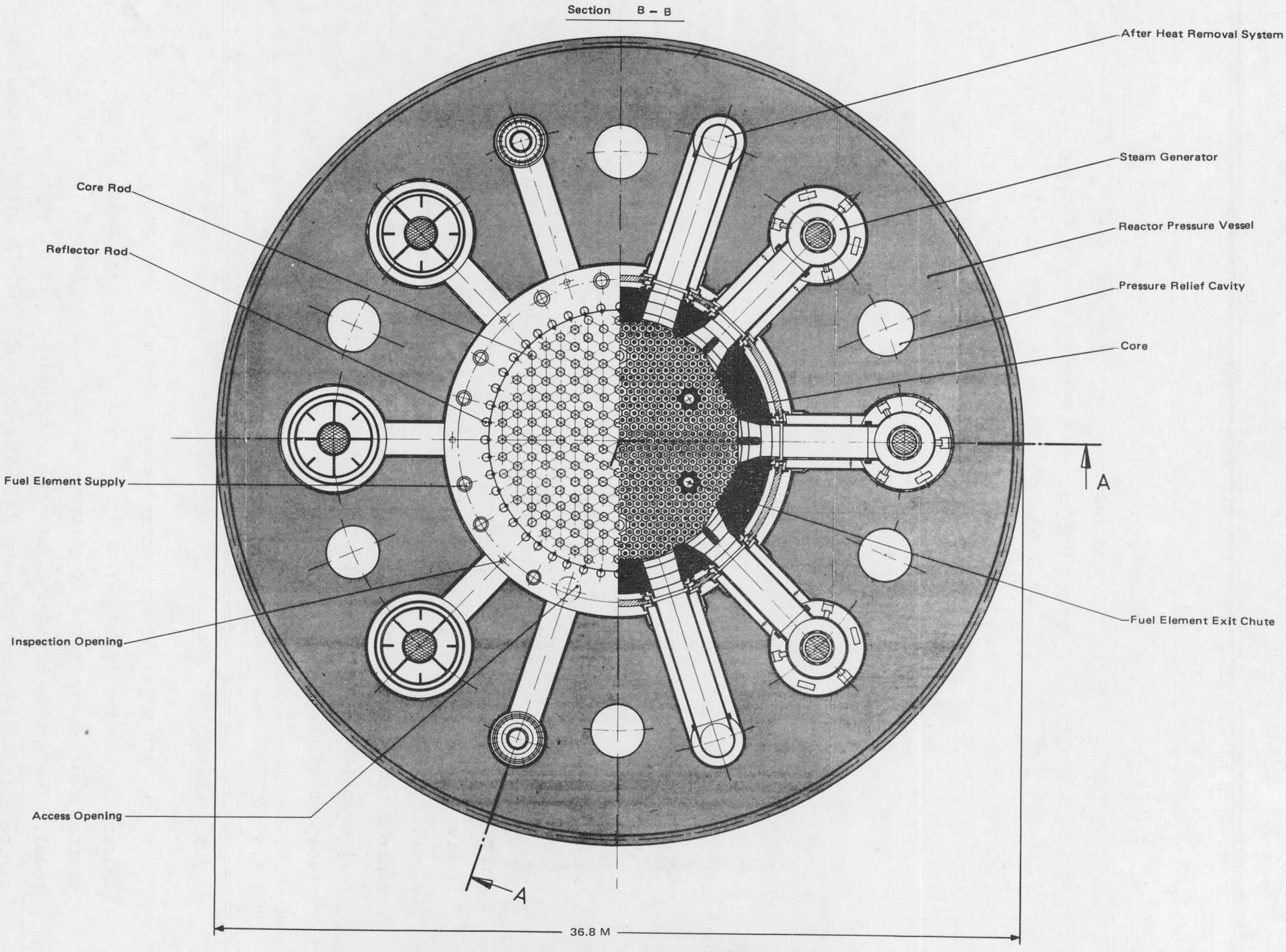


Figure 2-2. PCRV Horizontal Section (HTR-K)

the remaining 156 core rods and 24 reflector rods.

The first system (42 core rods, 24 reflector rods) provides fast shutdown capability from all normal operating or accident conditions and is designed to maintain hot subcriticality during the short period of time when xenon override is possible. If the reactor cannot be restarted quickly, the xenon content of the core reaches the level where the reactor cannot be restarted. Under these conditions, the reactor must be held subcritical as it cools down and temperature effects introduce positive reactivity. The second system is then activated to bring the reactor to cold subcriticality. In addition, the second system is used for load following and power distribution control. It also compensates for uncertainties in the initial core loading and during the transition to an equilibrium core.

A backup design for this control system is the PNP reference control system which uses control rods and small absorbing balls (KLAK). This system is discussed in Section 2.2.

2.1.5.2 Nuclear Instrumentation

Design of a nuclear instrumentation system for a large PBR has proven to be problematic, due to the difficulty involved in using in-core detectors. Hence, a design which uses ex-core detectors is being pursued. Presently, the system utilizes detectors between the side reflector and thermal shield to measure the leakage through the side reflector and provide an indication of the axial power distribution. The radial power distribution is monitored by fast flux detectors located in the upper reflector. The fast flux is measured because the thermal flux in the upper reflector is not representative of the power distribution due to the effect of the empty space between the core and the top reflector.

2.1.6 STEAM PLANT

The HTR-K steam plant, shown in Figure 2-3, consists of a single-shaft turbine generator with steam reheat and four stages of feedwater heating. Steam leaves the steam generators at 175 bars and 515°C and

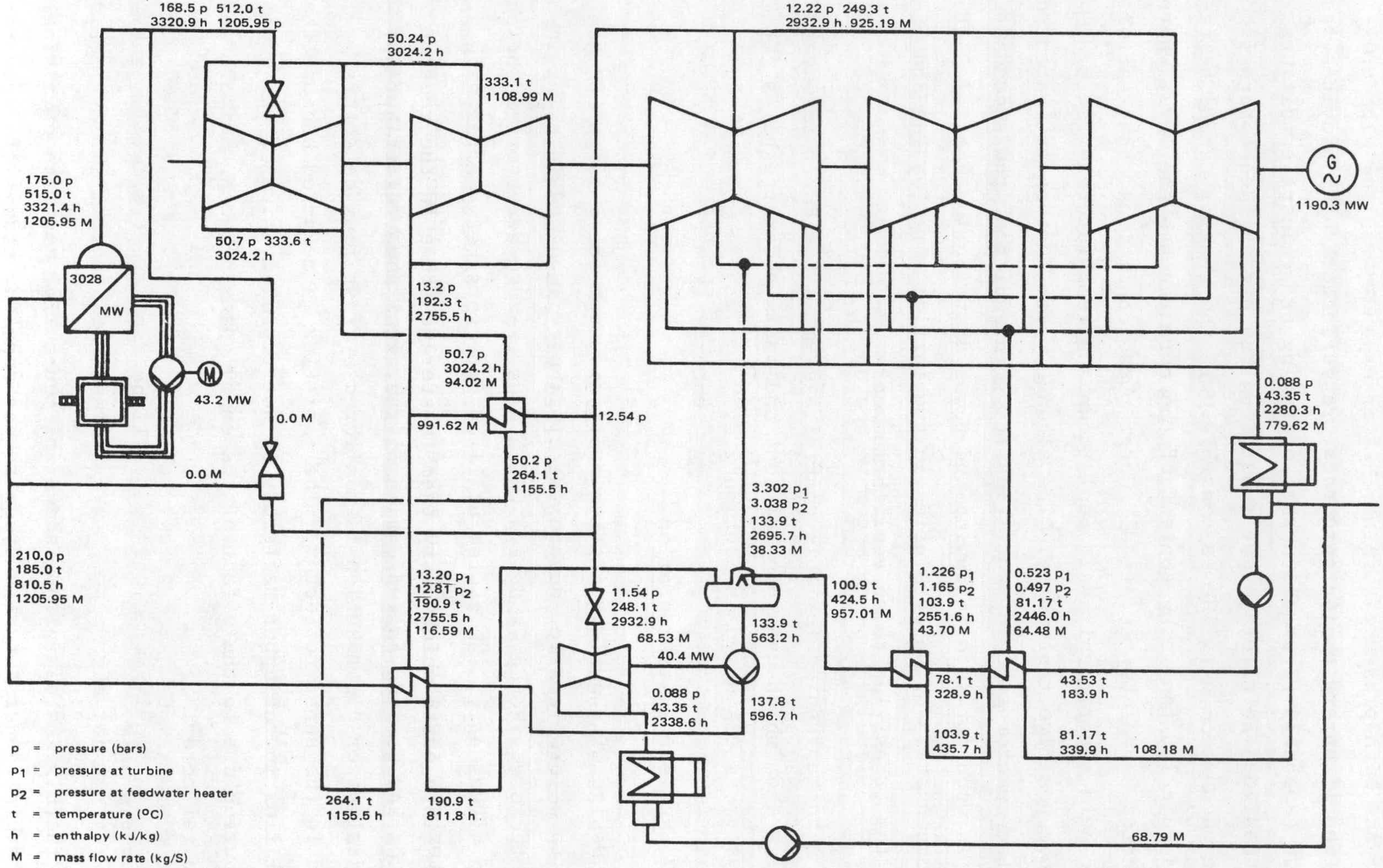


Figure 2-3. HTR-K Steam Plant Schematic

travels to the high-pressure turbine. After expanding in the high-pressure turbine, it enters the intermediate pressure turbine at 50 bars and 333°C. From this turbine, the steam goes to the reheater, where it is heated to 250°C at 12.5 bars by steam from the high-pressure turbine. Most of the reheated steam is distributed to the three low-pressure turbines, where it expands and exits to the condenser at 0.088 bars pressure. Part goes to power the main feed pump turbines. The water from the condenser passes through three low- and one high-pressure feedwater heaters before entering the steam generator at 255 bars and 198°C. Condenser waste heat rejection is to a wet cooling tower.

2.1.7 FUEL HANDLING

The fuel loading system, shown in the upper portion of Figure 2-4, consists of three separate, similar systems - one for the outer zone and two for the inner zone. Fresh fuel elements travel from loading containers above the PCRV to the proper loading position over the core. The fuel then enters the core through inlet tubes which penetrate the PCRV top slab.

After passing through the core, the fuel elements exit through the discharge tubes in the bottom reflector and travel to a damaged element separator. There, under primary system pressure, damaged and undamaged fuel elements are separated mechanically and then travel to separate containers. When the containers become full, they are depressurized, and the spent fuel elements are emptied into carts which carry them to storage areas in the reactor auxiliary building. The fuel removal system is shown in the lower portion of Figure 2-4.

2.2 PROCESS NUCLEAR HEAT PLANT (PNP)

2.2.1 GENERAL DATA

There are two PNP concepts that will be described in detail, the hydrogasification (or HKV) plant and the steam gasification (or WKV) plant. General data on HKV and WKV is provided on Table 2-2. A third concept under development is the combined process, described in Section 2.2.4.

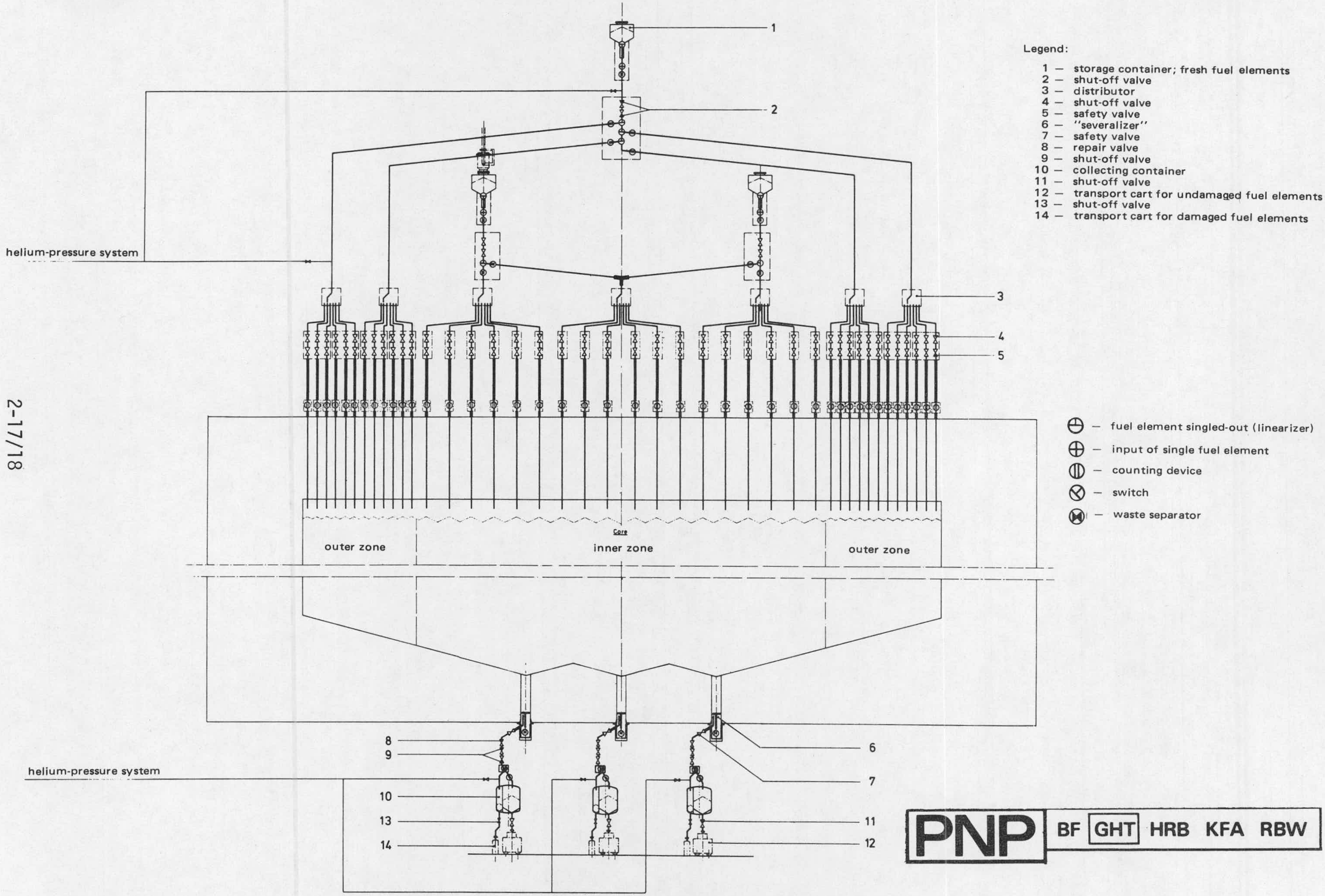


Figure 2-4. Pebble Bed Fuel Handling System Schematic.

TABLE 2-2

PNP CHARACTERISTIC DATA

OVERALL PLANT	Units	HKV	WKV
Primary Circuit			
Reactor thermal power	MW	3000	
Reactor coolant		Helium	
average inlet temperature	^{o}C	300	
average outlet temperature	^{o}C	950	
Pressure after circulator	bar	40	
Total mass flow rate	kg/s	888	
Number of loops		6	
Steam Plant			
Steam generator output	MW	2308	2033
Reheat		none	steam
Feedwater temperature	^{o}C	180	150
Live steam temperature (before turbine)	^{o}C	535	535
Live steam pressure (before turbine)	bar	110	180
Flow rate of live steam	kg/s	862	646
Intermediate pressure tapping			
use		methane reforming	process steam
steam pressure	bar	50	44
steam temperature	^{o}C	418	328.5
mass flow rate	kg/s	278.7	97.5

Table 2-2 (Cont'd)

	Units	HKV	WKV
Low-pressure tapping			not applicable
use		coal drying	
steam pressure	bar	5	
steam temperature	^oC	170	
mass flow rate	kg/s	151	
Low-pressure steam generation		not applicable	
source			waste heat from gas plant
steam pressure	bar		6
steam temperature	^oC		159
mass flow rate	kg/s		346.2
Condenser pressure	bar	0.12	0.12
Gross electrical output	MW	589	927
Net electrical output	MW	118	637
Electrical power for HTR	MW	47	60
Electrical power for intermediate circuit	MW	not applicable	120
Electric power for steam plant	MW	30	50
Electric power for gasification plant	MW	394	60
<u>Gasification Plant</u>			
Coal usage	t/h	2317 (soft 26%C)	418 (hard 80%C)
Heat value			
high (H_h)	kcal/kg	2524	not available
low (H_L)	kcal/kg	2059	6873

Table 2-2 (Cont'd)

	Units	HKV	WKV
CH_4 production	m^3/h	368000	239000
CH_4 for reforming	kg/s	64.1	not applicable
Waste coke	t/h	285.7	"
Tar and oil produced	t/h	not applicable	70.5
Load range	%	100-75	100-75
Efficiency of the complete plant			
with H_h	%	72	not available
with H_L	%	68	58.8
COMPONENTS			
Reactor Pressure Vessel			
Type		PCRV	
Construction		multiple cavity	
Operating pressure	bar	40 to 42	
Design pressure	bar	46.2	
Concrete temperature	^oC	66	
Overall dimensions			
height	m	31	35
diameter	m	44	46
Wall thickness			
top	m	7.5	9
bottom	m	6.5	9
side	m	12.6	

Table 2-2 (Cont'd)

	Units	HKV	WKV
Core cavity			
height	m	17	17
diameter	m	16.4	18
Cavities for components:			
Steam reformer			not applicable
number		6	
diameter	m	4.8	
He/He heat exchanger		not applicable	
number			24/12*
diameter	m		3.25/5.5*
Hot gas distributor		"	
number			6
diameter	m		4
Steam generator			not applicable
number		6	
diameter	m	4	
Process gas pipeline			not applicable
number		6	
diameter	m	1.8	
After-heat removal system			
number		4	4
diameter	m	3	3.8

*First entry is for helical design; second entry is for U-tube design.

Table 2-2 (Cont'd)

	Units	HKV	WKV
Circulator			
Number		6	
Type		radial	
Drive		electric motor	
Helium temperature (at exhaust)	°C	300	
Helium pressure (at exhaust)	bar	40	
Flow rate	kg/s	148	
Pressure rise	bar	1.3	
Power at motor terminals	MW	8	
Flow control		throttling	
Steam Reformer			not applicable
number			
type		counter flow with inside return	
Power derived from helium	MW_t	115.3	
Heat regained from reformed gas within the steam reformer	MW_t	21.5	
Total power for reforming	MW_t	136.8	
Helium mass flow rate	kg/s	148	
Helium inlet temperature	°C	950	
Helium outlet temperature	°C	800	
Process gas composition		$4\ H_2O + CH_4$	
Process gas flow rate	kg/s	58	
Process gas temperature entering the containment	°C	330	
Process gas inlet temperature to catalyst	°C	500	
Process gas maximum temperature in catalyst	°C	810	
Process gas maximum reaction temperature	°C	800	
Process gas outlet temperature from inner return tube	°C	680	
Process gas temperature leaving the containment	°C	510	

Table 2-2 (Cont'd)

	Unit	HKV	WKV
Process gas inlet pressure	bar	44	
Process gas outlet pressure	bar	40	
Number of reformer tubes		313	
Reformer tube ID/OD	mm	130/150	
Length of reformer tube	m	12.2	
Catalyst form		Raschig Rings (10x10mm)	
Return Tube			
outer diameter	mm	30	
inside diameter	mm	26.8	
total length	m	13.5	
He/He Heat Exchanger		not applicable	
Construction		Helical Counter Flow	U-Tube Counter Flow
Number		24	12
Thermal power	MW	125	250
Flow rate	kg/s	37/36.3+	74/73+
Inlet temperature	°C	950/240+	950/240+
Exit temperature	°C	300/900	300/900
Operating pressure	bar	40/42	40/42
Design pressure	bar	45	45
Number of tubes		1900	5840
Size of tubes	mm	22.4x2.25	18x1.8
Shell diameter	m	2.4/2.85	not available

+The second value is the secondary side.

Table 2-2 (Cont'd)

	Units	HKV	WKV
Steam Generator			
Construction		Helical & straight tube bundles	Not available
Flow		parallel in straight, counter in helical section	counter
Thermal power per unit	MW	385	Not available
Total thermal power	MW	2803	2033
Helium flow rate	kg/s	148.1	Not available
inlet temperature	°C	800	687
outlet temperature	°C	300	240
Water/steam flow rate	kg/s	143.3	Not available
inlet temperature	°C	180	150
outlet temperature	°C	540	540
outlet pressure	bar	115	185
After-Heat Removal (NWA) System			
Design		4x50%	
NWA circulator			
type		axial	
drive		electric motor	
helium flow rate	kg/s	8.6	
He temperature	°C	250	
He pressure after blower	bar	minimum of 1	
power at motor terminals	KW	616	
flow control		speed	

Table 2-2 (Cont'd)

	Units	HKV	WKV
NWA heat exchanger			
design		U-tube, box type	
thermal power	MW	27	
primary side data			
-medium		helium	
-inlet temperature	°C	1000*	
-outlet temperature	°C	250	
-minimum pressure	bar	1	
-flow rate	kg/s	8.6	
secondary side data			
-medium		H_2O	
-inlet temperature	°C	60	
-outlet temperature	°C	140	
-pressure	bar	not available	
-flow rate	kg/s	80	

*Note that the NWA system sees this temperature only during accident conditions.

2.2.2 HYDROGASIFICATION (HKV) PLANT

In the hydrogasification process, hydrogen is used to convert the coal to methane. In theory, nearly any kind of coal can be used, but the process has been found to work well with soft coal containing a relatively high fraction of volatile constituents. Hydrogasification processes are described by the following equations:

$$C + 2H_2 = CH_4 \quad ; -20.6 \text{ kcal/mole, exothermic} \qquad (2\text{-}1)$$

$$CH_4 + H_2O(v) = CO + 3H_2; \quad 49.0 \text{ kcal/mole, endothermic} \qquad (2\text{-}2)$$

$$CO + H_2O(v) = CO_2 + H_2 \quad ; -9.9 \text{ kcal/mole, exothermic} \qquad (2\text{-}3)$$

Equation 2-1 describes the hydrogasification of coal. Since the overall process starts with only coal, water, and heat, the hydrogen needed for gasification has to be produced elsewhere within the plant. This is accomplished by catalytically reforming some of the product methane to hydrogen as described by Equation 2-2. The hydrogen production can be increased further by using the shift reaction of Equation 2-3 to react steam and carbon monoxide to produce carbon dioxide and additional hydrogen. It is not necessary that the reactions occur one after another in separate components or in the order given above. It is possible that some of these conversions will occur simultaneously in one component. It is preferable that the hydrogasification reaction occur at high pressure and relatively low temperature to obtain improved methane formation rates. Pressures in the range of 40 to 100 bars are adequate and, in this case, a design pressure of 80 bars has been selected. A design temperature of 850°C has been chosen, again to improve the methane formation rate, even though the chemical equilibrium of the reaction is somewhat adversely affected by high temperatures. Considering the entire process plant, these values are best overall.

The kinetics and operational aspects of steam reforming (Equation 2-2) have been known for several decades and have been applied on a large industrial scale. To obtain high hydrogen output rates, the catalytic reforming reaction (using a nickel-base catalyst) should take place at low pressure and high temperature. However, to reduce the cost of subsequent compression work currently in conventional applications, it is generally advantageous to use somewhat higher pressure, such as 30 bars. Nevertheless, in this nuclear application, the desire to limit pressure differential stresses on reformer tubing and to reduce the likelihood of leakage out of the primary helium system has resulted in a pressure range of 40 to 50 bars.

The shift reaction described by Equation 2-3 is also based on well-established technology and industrial experience. The reaction normally takes place at temperatures lower than those of the other two reactions (300-400°C), and a pressure of about 40 bars is used.

The overall gasification facility consists of the reactor plant, steam plant, electrical plant, fuel handling facility, and chemical process plant.* The manner in which the individual plants are related is illustrated in Figure 2-5, and an overall energy flow diagram is given in Figure 2-6. The reactor provides heat for (1) coal drying, (2) chemical processing, (3) make-up heat losses, and (4) producing steam. Energy is also brought into the overall system in the form of the feed coal. As shown in Figure 2-6, the reactor and coal represent an equivalent energy input of 9736 MWth (reactor, 3000 MWth; coal, 6736 MWth). The methane output, the heating value of the residual coke, and the net electrical power surplus all are forms of output energy (methane, 4179 MWth; coke, 2346 MWth; electricity, 118 MWe, totalling 6643 MW). An index of overall plant performance, η_h, can be determined by dividing the energy of the reactor and coal into the energy of the coke, methane, and electricity:

$$\eta_h = (6643/9736)100 = 68\% \qquad (2\text{-}4)$$

This value represents the ratio of useful energy output to the total energy input of the overall plant.

2.2.2.1 Reactor Plant

The reactor plant consists of the 3000 MWth pebble bed reactor and six parallel helium heat transfer circuits, each containing a circulator, a steam generator, a steam reformer, and necessary gas ducting. All six circuits are located within the integrated prestressed concrete reactor vessel (PCRV).

*Data on the chemical process plant is not presented, since it is not within the scope of the General Electric/German Confidentiality Agreement.

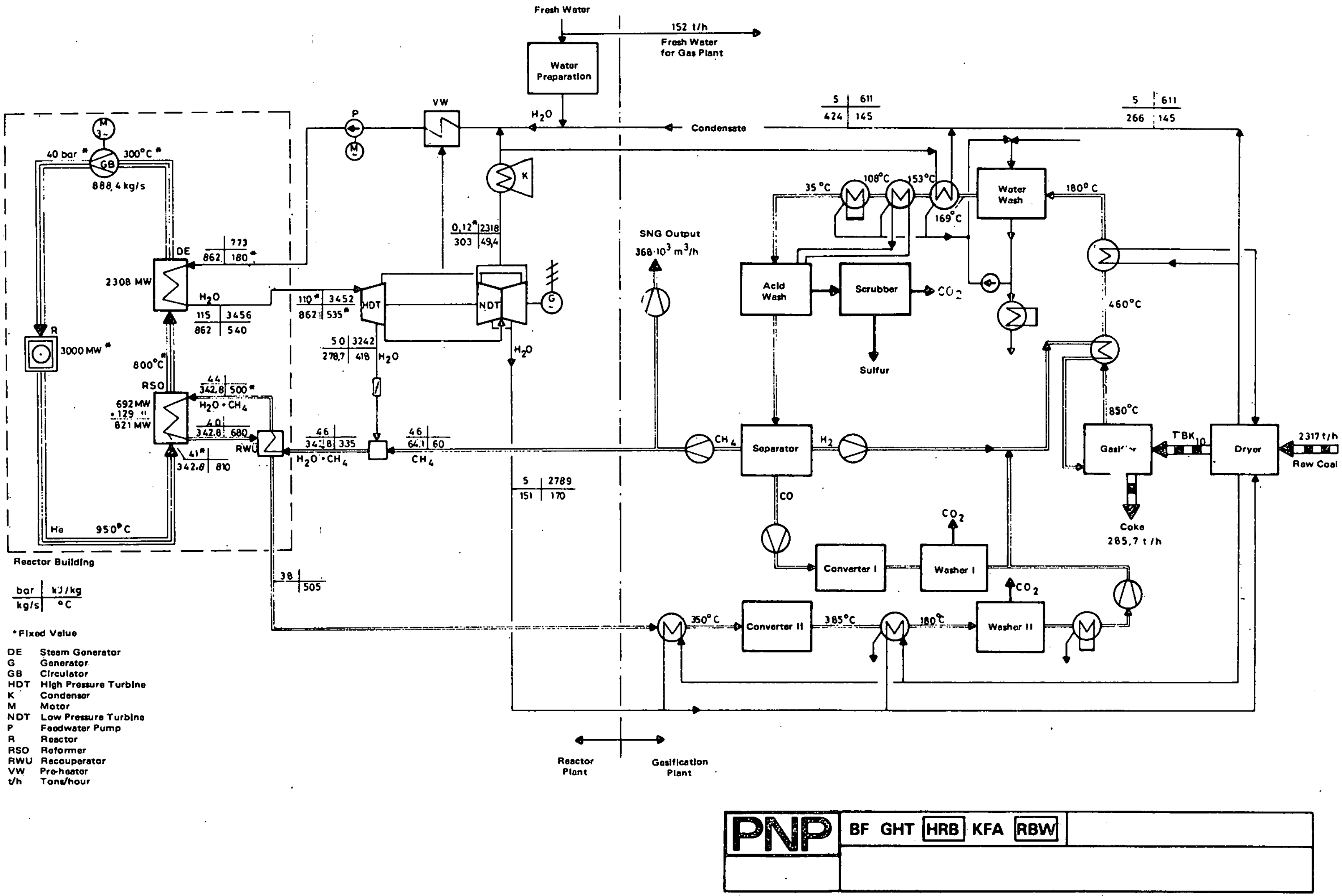
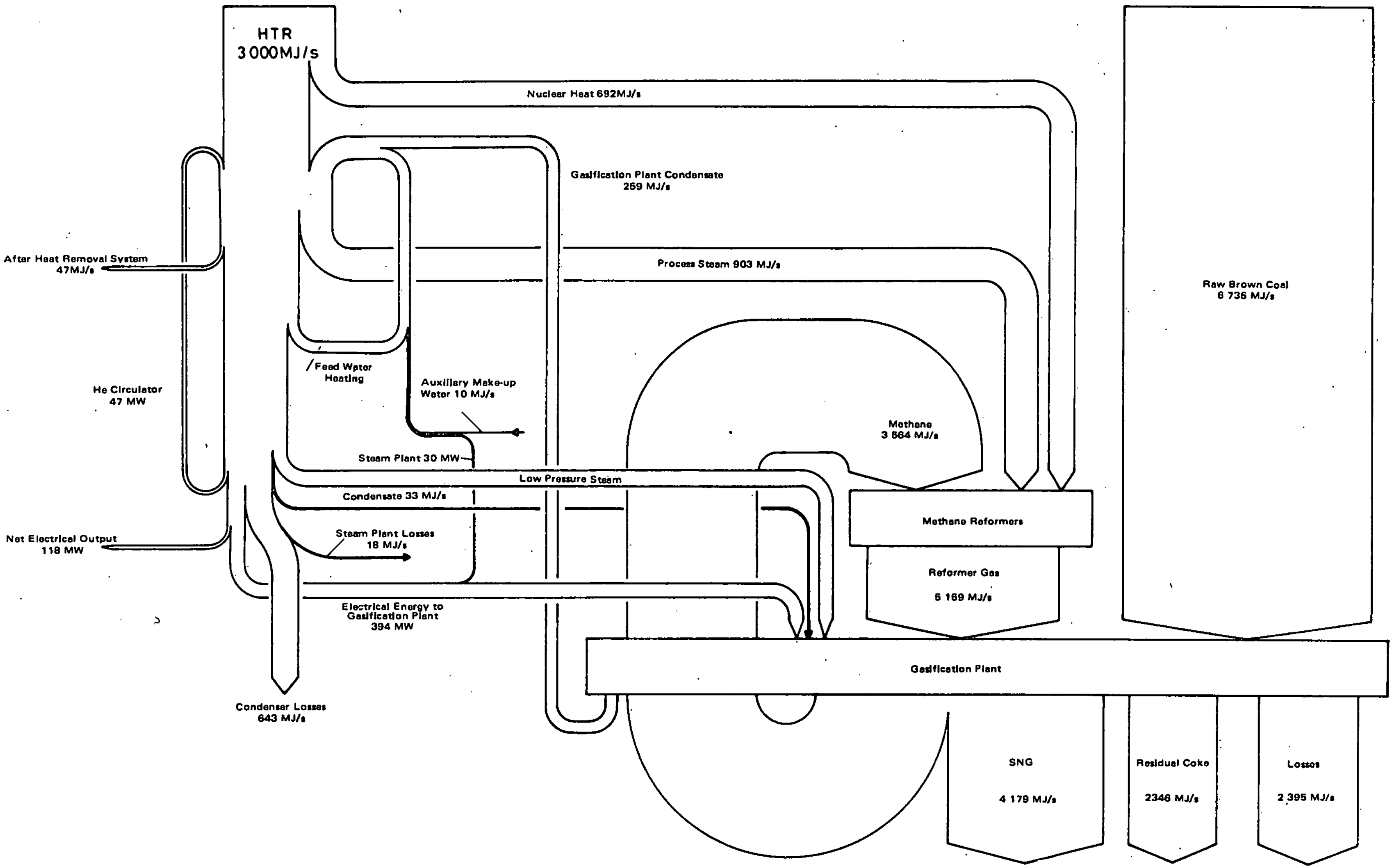


Figure 2-5. Hydrogasification Plant Schematic



PNP BF GHT HRB KFA RBW

Figure 2-6. Hydrogasification Plant Energy Balance

2.2.2.1.1 Reactor Core

The reactor, shown in Figure 2-7, has a cylindrical core 11.2 meters in diameter and 5.54 meters in height, which contains approximately 3 million spherical fuel elements. As in the HTR-K electrical plant, the fuel elements consist of coated mixed Th-U oxide particles contained in a graphite matrix 5 cm in diameter. Around this matrix is a fuel-free graphite shell 0.5 cm thick. To flatten the radial power profile, the core has two regions of different fissile loadings, an inner cylindrical region surrounded by an outer annular region containing fuel elements with a slightly higher fissile loading. The fuel is burned in a once-through-then-out (OTTO) cycle and achieves an average depletion of 100,000 MWd/tonne.

The core cavity is formed by interconnected graphite blocks which act as the neutron reflector. The top reflector contains penetrations for control rods and the inlet tubes of the fuel handling system. The bottom reflector has penetrations for the fuel exit tubes and for the flow of cooling gas. Surrounding this graphite structure is a metallic thermal shield which supports the side reflector.

2.2.2.1.2 Primary Circuit

The primary circuit has six parallel helium circuits, each containing a steam reformer with integral recuperator, a steam generator, and a helium circulator. In each circuit, these components are connected in series with appropriate ducting as shown in Figure 2-7. The arrangement of the six helium circuits in relation to one another and the reactor is illustrated in Figure 2-8.

The reactor core is cooled by helium at a pressure of 40 bars. The helium enters the reactor at a temperature of 300°C near the bottom of the reactor. It flows around the core container moving generally upward and bathes the thermal shield and the PCRV (Figure 2-7). Once above the core, the flow is turned downward through the upper reflector and thermal shield and passes downward through the pebble bed. Upon reaching the core bottom, the gas, now at an average temperature of 950°C, exits the core through holes in the lower reflector and enters a mixing chamber below the core. From here the gas flows radially to the reactor circumference, where it enters the six ducts leading to the steam reformers.

The helium ducts between the reactor and steam reformers are horizontal and coaxial (Section 4.3.1). Hot reactor outlet helium flows in the inner cylindrical duct, while cooler helium returning to the reactor flows in outer annular duct. The flow rate in an individual circuit is 148 kg/s with gas velocities of approximately 21 m/s and 61 m/s for the cold and hot flow streams, respectively. To reduce stresses in the hot gas duct and undesirable heat transfer between the two gas streams, insulation is used within the coaxial duct.

The steam reformers illustrated in Figure 2-7 consist of vertically mounted, straight-tube exchangers. Access ports are provided at the top of the PCRV. The net heat load for each reformer is 115 MWth. On the primary side, helium at 950°C enters a plenum at the bottom and then passes vertically to circumferential dump ports near the top of the unit. The helium emerges at an average temperature of 800°C and proceeds to the steam generator. On the secondary side the process gas enters the PCRV and is ducted upward along the outside of the steam reformer to near the top of the PCRV, where it enters the steam reformer. There it passes downward through a top-mounted recuperator integral to the steam reformer. The process gas flows downward in the catalyst bed then makes a 180° bend and returns to the top of the unit inside tubes surrounded by the catalyst bed. Upon reaching the top of the reformer, the process gas again passes through the recuperator, leaves the unit, is ducted downward, and exits the PCRV at the bottom. The process gas enters the reformer at 46 bars and approximately 335°C, and leaves at 38 bars and a temperature of approximately 505°C.

The steam generators are also shown in Figure 2-7. The heat load on each unit is 385 MWth. The primary-side helium enters near the top and flows downward on the outside of the vertical superheater bundle and then over the evaporator and economizer helical tube bundle to exit finally near the bottom of the unit. The helium enters at 800°C and leaves at 300°C. Feedwater enters the PCRV at the bottom of the steam generator and passes upward inside the helical economizer/evaporator tube bundle. The steam produced in the evaporator then moves up the vertical length of the superheater to the top of the unit. There the tubing turns and passes straight downward to the outlet steam

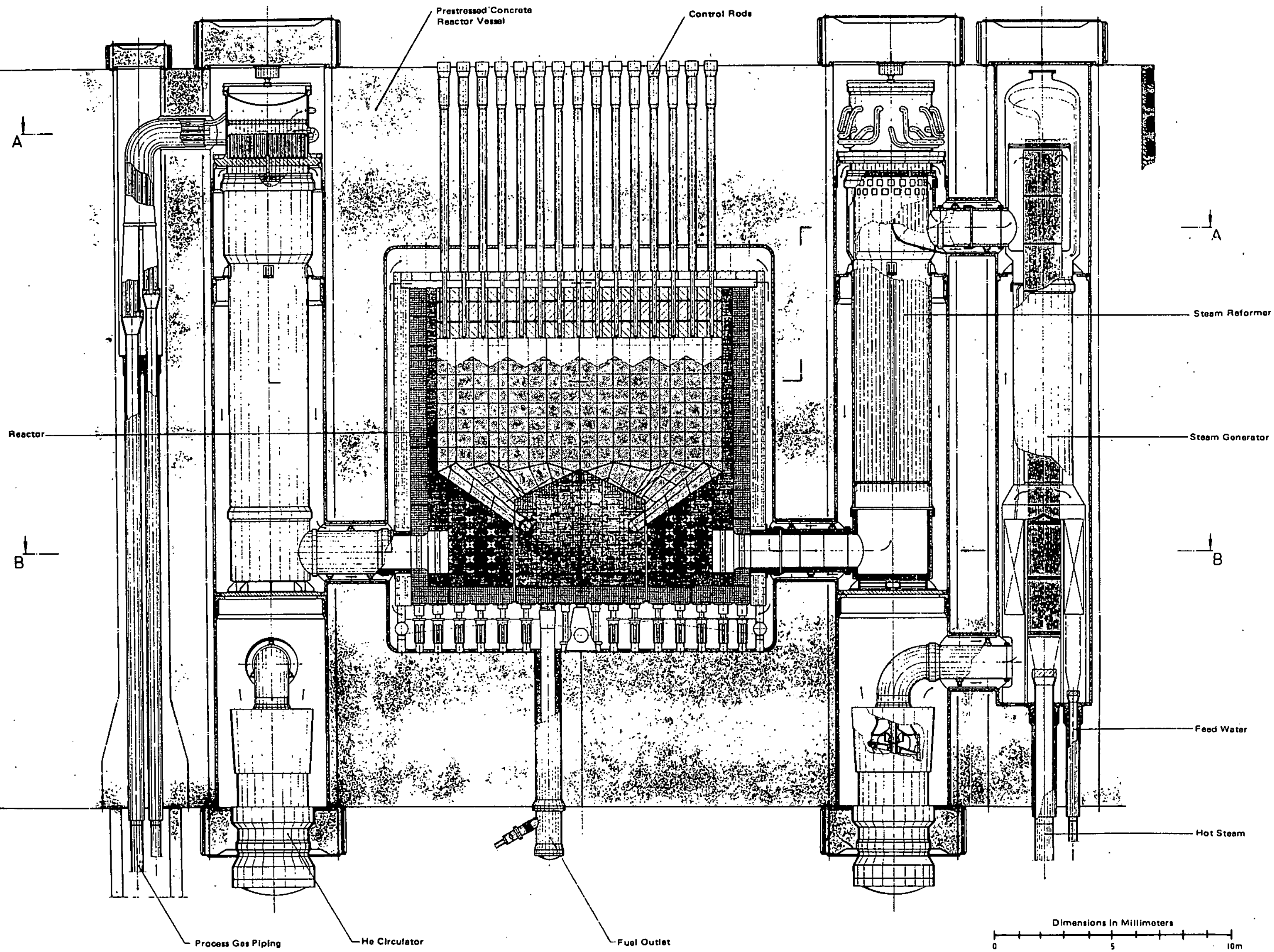


Figure 2-7. Hydrogasification Plant Primary Circuit

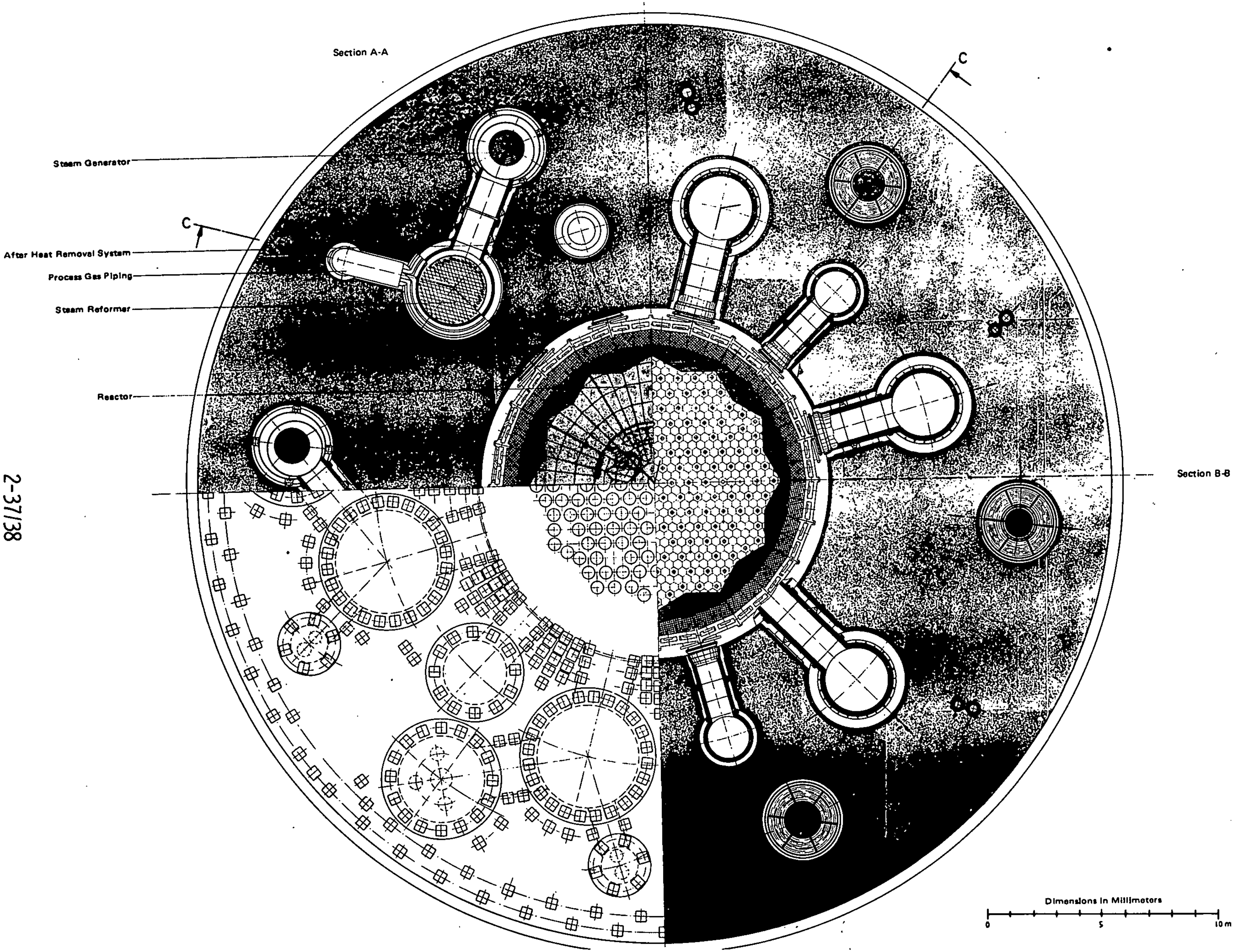


Figure 2-8. Plan View of HKV Reactor Plant

header and leaves the PCRV at the bottom. The feedwater enters the unit at 180°C, and steam exits at 540°C and 115 bars. Access to the unit for servicing can be gained from the bottom of the PCRV.

The helium circulators are bottom-mounted units located below the reformers. As shown in Figure 2-7, they can be accessed for servicing from below the PCRV. The totally enclosed circulators are electrically driven, and each requires about 8 MW to circulate the 148 kg of 300°C helium per second with a pressure rise of 1.3 bars. The present method of flow control is by using constant-speed motors with variable inlet control for load change and shutdown.

The after-heat removal (Nach Warme Abfuhr, NWA) system is designed to remove 2% (60 MWth) of normal-rated reactor power under depressurized conditions equal to 1 bar containment pressure. The system consists of four separate loops, each conservatively sized to remove 1% of rated reactor power. Each NWA loop contains a variable-speed, electrically driven helium circulator and a helium-to-water heat exchanger, both located within the PCRV. The circulators are single-stage axial flow units capable of circulating 80 m^3/s. The heat exchangers are of the U-tube type with the cooling water flowing inside the tubes (refer to Section 5.3).

2.2.2.1.3 Containment Structures

The PCRV planned for the PNP hydrogasification plant is an integral upright cylinder of prestressed concrete generally similar to the one used on the HTR-K electrical plant. The PCRV is illustrated in Figure 2-9 along with principle dimensions. It accommodates the entire primary system (core, steam generators, steam reformers, helium circulators, and after-heat removal system) within the concrete structure. Thus, in addition to the large central cavity for the reactor core, the PCRV contains 16 major cavities: 12 for the steam generators and steam reformers and 4 for the decay heat removal loops. Other smaller openings are provided for fuel handling equipment, instrumentation, control rods, and piping.

The vessel is prestressed vertically and circumferentially by using steel cables and banding, respectively. The internal design pressure for the vessel is 46.2 bars, 10% greater than the maximum expected operational pressure of 42 bars and 15% greater than the nominal operating pressure of 40 bars. A steel liner is attached to the walls of all PCRV cavities as a gas leakage barrier. It is insulated on the inside with a fiber insulation and is cooled by water flowing in pipes welded to the cavity liners. This cooling flow, together with the internal insulation, is sufficient to limit concrete temperatures to 66°C during normal plant operation. Analysis of PCRV cooling and overall design is continuing.

The reactor protective building surrounds the PCRV, as well as the fuel handling system and reactor maintenance equipment. It consists of a massive, reinforced concrete base plate and a horizontally prestressed concrete cylindrical shell. It is sealed by a steel liner attached directly inside the concrete. An access air lock to the reactor auxiliary building has been provided for transferring materials. The inside height of the building (80 meters) was determined not only by the dimensions of the PCRV and fuel handling facilities but also by the crane lift-height necessary for removal of the steam generators and/or the steam reformers. Within the 54 m inside diameter of the building, the PCRV is located eccentrically (to one side) to provide possible work space within the building in the event that it becomes necessary to remove either a steam generator or reformer.

The reactor auxiliary building is constructed from reinforced concrete and contains storage areas for fresh and spent fuel elements. Besides these storage areas, the building has storage for control rods and hot cells for handling radioactive materials. Rail connections to the reactor protective building are used for transferring fuel elements carts.

2.2.2.1.4 Control and Instrumentation

Analysis of the reactivity control system and the number of rods required is still continuing. However, the current primary reference reactivity control system employs 156 control rods. They are electrically driven rotating rods accessible from the top of the PCRV. As they are driven downward into the core, the rods rotate to minimize disturbance of the pebble bed. Reference

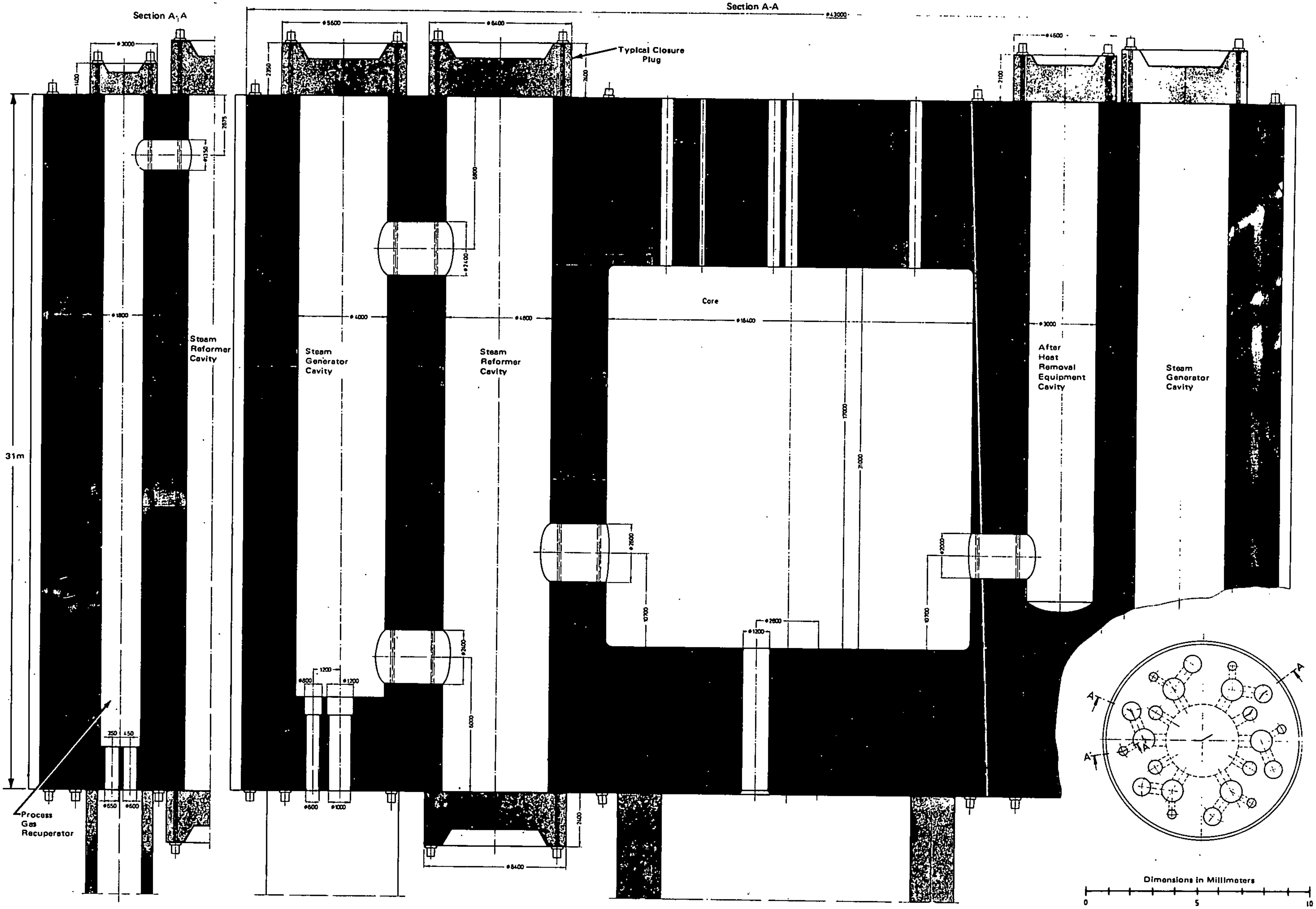


Figure 2-9. Hydrogasification Plant PCRV

design speed is 2 cm/s and rod motion is stopped periodically during insertion in order to maintain acceptable temperatures in the rod. The high (950°C) operating temperature of the PNP plant makes this stepwise insertion necessary. The rods contain boron carbide as the absorber material.

A secondary reactivity control system is provided because of German licensing requirements and uses small absorber balls containing boron carbide. These KLAK (Kleine Absorber Kugeln) balls are approximately one-sixth the size of the fuel elements and would be dropped directly into the fuel element bed from overhead containers within the PCRV. Approximately one small absorber ball would be required for each fuel element. Tests demonstrate that the KLAK balls will move downward into the interstitial spaces in the pebble bed, remain in place, and shut down the reactor.

The design of the nuclear instrumentation system is currently being developed. In the current design, the instruments are placed between the side reflector and thermal shield to measure the flux. Power-range instruments would be placed circumferentially around the core at four equally spaced locations. At each of the four locations, six detectors would be vertically spaced along the height of the core; their individual currents would be indicative of the axial distribution and would be added to provide a total signal representing thermal flux at each of the four circumferential locations. The radial power distribution is monitored using fast flux detectors in the upper reflector. Fast flux instruments are used, since the thermal flux in the upper reflector is not representative of the power because of the effects of the vacant space between the top of the core and the upper reflector.

Source and intermediate-range instruments are provided for reactor startup but are repositioned during power operation to minimize exposure to high temperature.

2.2.2.2 Steam Plant

Steam is produced for coal drying, the hydrogasification process, and the production of electricity in a turbine-generator set; the plant is depicted in Figure 2-10. At design conditions steam is produced at a rate of 862 kg/s and a pressure of 110 bars and 535°C. From the steam generator the steam passes directly to the turbine, which is a double-flow single-shaft condensing type unit. It uses a single high-pressure casing and a separate low-pressure casing. The electrical rating is 589 MWe and the shaft speed is 3000 rpm. Two process steam extractions are taken from the high-pressure turbine unit and provide steam to the reformer at 418°C and 50 bars. Five steam extractions are used for regenerative heating of the feedwater. The turbine is equipped with protective devices, and, in the event it is shutdown (such as an over-speed trip), steam can be throttled and ducted to an auxiliary condenser. This condenser as well as the condenser on the low-pressure turbine is equipped with three condensate pumps taking suction in the conductor hot wells. Each pump is capable of handling 50% of the condenser condensate flow. The condensate pumps discharge to a header leading to a common feedwater purification system and subsequently to the feedwater preheaters. Electrically driven feedpumps (two at 50% capacity) circulate the flow through a final preheater to the steam generator, which it enters at 180°C. Cooling water is supplied to the condenser by a closed-loop system utilizing a single dry cooling tower.

2.2.2.3 Electrical Plant

The turbine is directly coupled to a two-pole generator. Principal design data are summarized below. The synchronous generator is excited by means of a shunt excitation system.

Generator Data	
Rating	660,000 kVA
Power Factor	0.8
Terminal Voltage	21000 ± 5%, V
Excitation	Stationary Thyristors
Step-Up Transformer	
Number	2
Type	Three-phase
Rating	400,000 kVA
Power Systems Network Transformer	
Number	1
Type	Three-phase
Rating	650,000 kVA

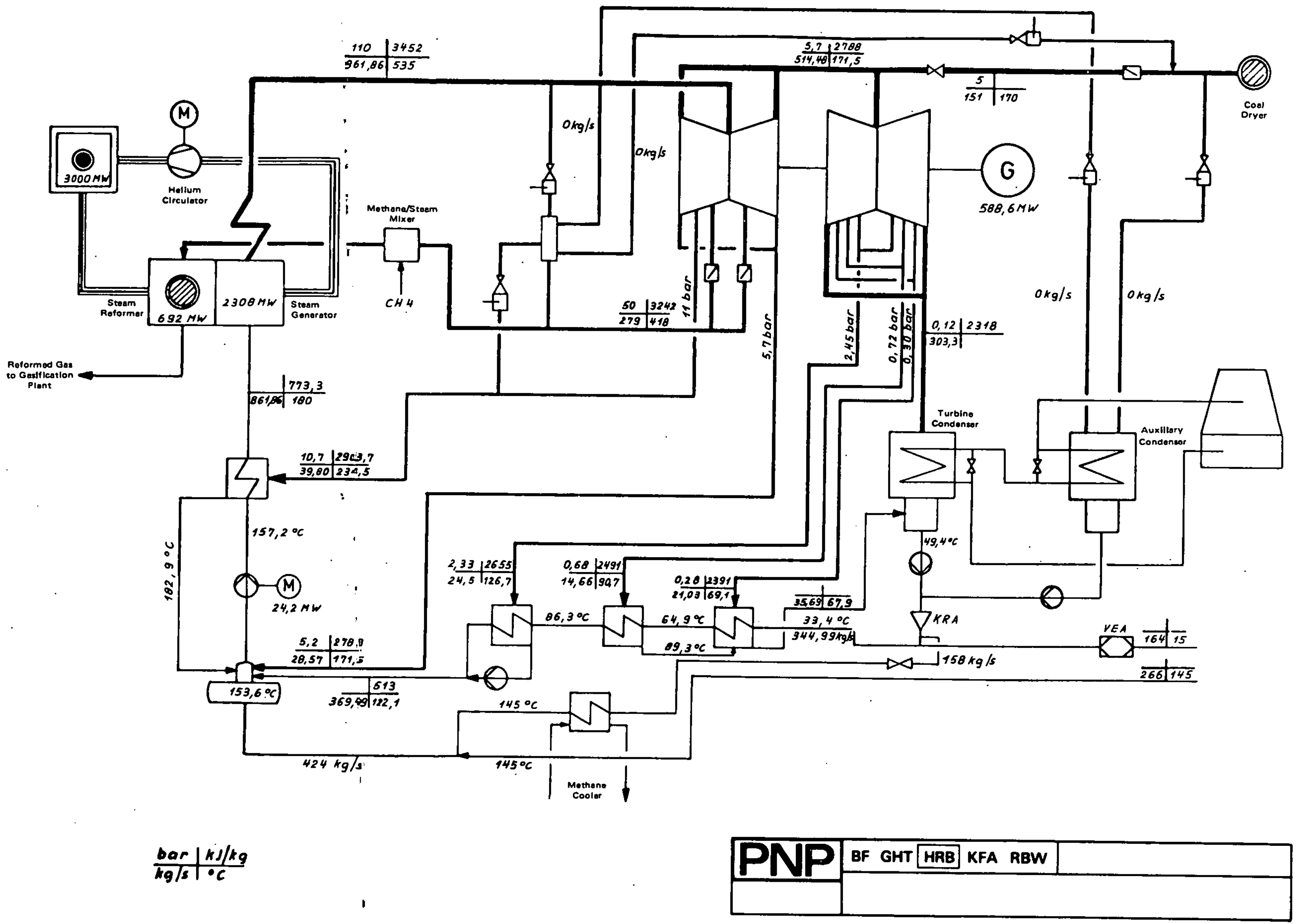


Figure 2-10. Steam Plant for the Hydrogasification Plant

2.2.2.4 Fuel Handling System

The fuel handling system is basically the same as that previously discussed for the HTR-K electric plant (Section 2.1.7). Its operating principles are the same as those that have been demonstrated on the operational AVR plant, and which will also be used at the large THTR plant under construction in the Federal Republic of Germany.

2.2.3 STEAM GASIFICATION

In this process, steam is used to convert coal to gas. The following reactions are the basis for the process:

$$2C + 2H_2O = 2CO + 2H_2; + 56.8 \text{ kcal/2 moles C, endothermic} \quad (2\text{-}5)$$

$$CO + H_2O = CO_2 + H_2; - 9.9 \text{ kcal/mole, exothermic} \quad (2\text{-}6)$$

$$CO + 3H_2 = CH_4 + H_2O; - 49.0 \text{ kcal/mole, exothermic} \quad (2\text{-}7)$$

Equation 2-5 describes the gasification reaction of coal and steam. The process is carried out at high temperature, in the neighborhood of 700 to 800°C depending on the type of coal used. Equation 2-6 is the carbon monoxide shift reaction and has the same function as previously discussed in Section 2.2.2. This reaction is normally performed in the 300 to 400°C temperature range. Finally, Equation 2-7 is the methanation reaction and results in the formation of the product gas; it is normally run at lower temperatures, of about 300°C. In future plants high temperatures of up to perhaps 650°C will be available for this reaction, which will allow more efficient use of process heat.

For steam gasification, like hydrogasification, the overall facility consists of the reactor plant, steam plant, electrical plant, fuel handling facility, and chemical process plant. However, unlike the hydrogasification plant, an intermediate helium circuit is used to transfer heat from the reactor and primary circuit to the remainder of the plant. Thus, the intermediate circuit separates the reactor plant from the rest of the gasification equipment. This separation avoids bringing large quantities of coal into the PCRV and allows greater (nuclear/chemical) plant separation distances. However, the complexity and cost of the overall plant may be increased. Figure 2-11 illustrates the overall plant and the manner in which the intermediate loop is incorporated. As before, the reactor provides all the process heat used in the plant as well as steam to the turbine generator for plant electrical loads. An overall energy flow diagram is given in Figure 2-12. Input energy is introduced by the reactor (3000 MWth) and the coal (3482 MWth). Useful output energy exists in the residual tar and oil (771 MWth), the produce methane gas (2677 MWth), and the net electrical surplus power (637 MWe). The overall plant performance index, η_s, becomes

$$\eta_s = 100(771 + 2677 + 637)/(3000 + 3482) = 63\%$$

This value represents the ratio of useful output energy to the total energy input to the plant.

2.2.3.1 Reactor Plant

The reactor plant consists of the 3000 MWth pebble bed reactor and six primary helium heat transfer circuits connected in parallel. Each circuit contains a circulator, intermediate heat exchangers, and necessary gas ducting. All six circuits are located within the integral prestressed concrete reactor vessel. Presently two reactor plant configurations are under consideration, each reflecting a different intermediate heat exchanger design. Since evaluation of the two designs is currently in progress, major features of both alternates are summarized in the following discussion.

2.2.3.1.1 Reactor Core

Figures 2-13 thru 2-16 illustrate the two reactor configurations under consideration. The core designs for both alternates are equivalent and are basically the same as the hydrogasification plant described in Section 2.2.2.1.1.

2.2.3.1.2 Primary Circuit

Both reactor designs have six primary helium circuits, each with a circulator and intermediate heat exchangers. The arrangement of the six primary circuits within the PCRV is shown in Figures 2-13 through 2-16.

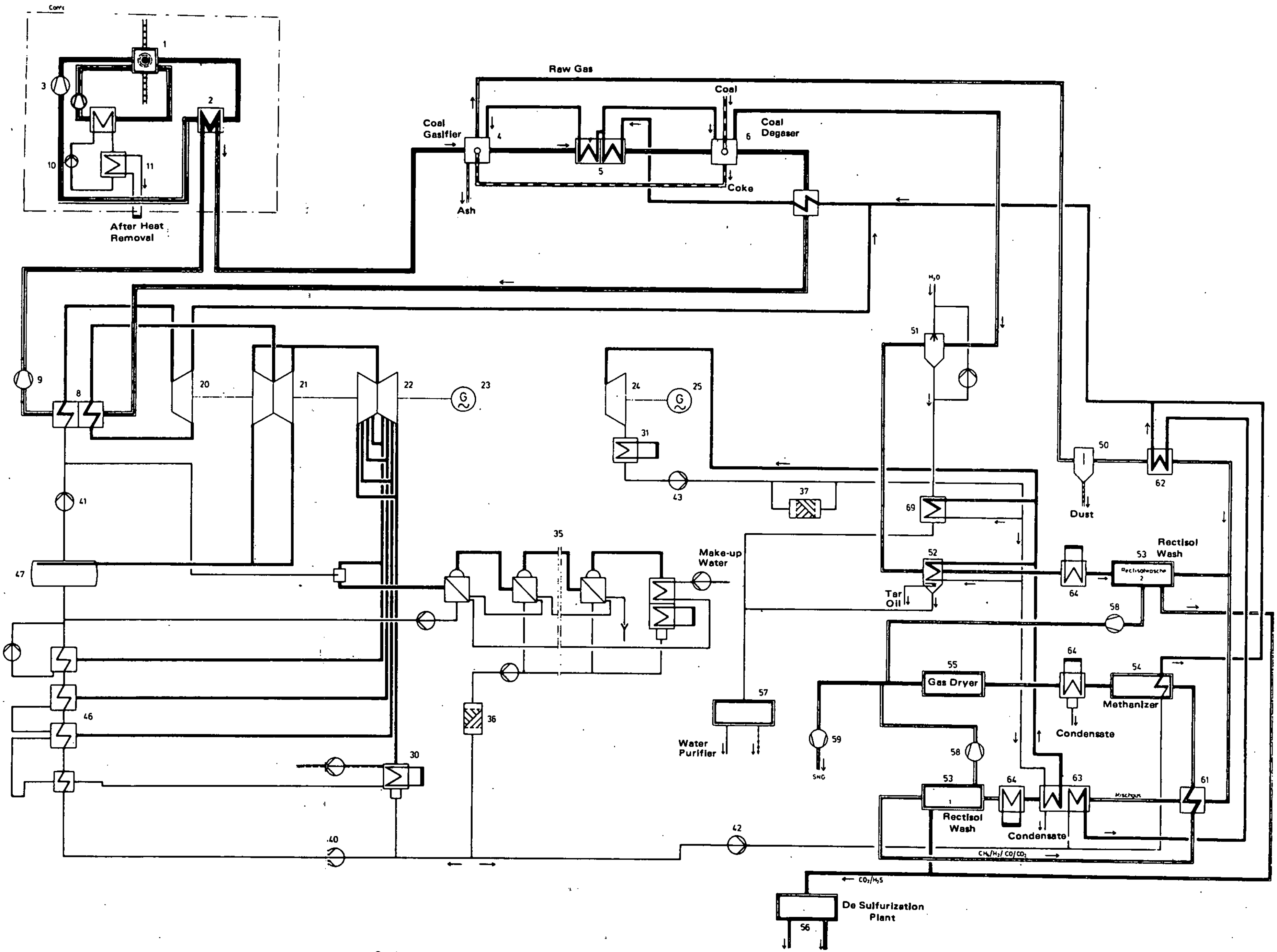


Figure 2-11. Steam Gasification Plant Schematic

Legend for Figure 2-11
Steam Gasification of Coal

Reactor Installation with Intermediate Circuit

Main Circuits

1. High-Temperature Reactor (Pebble Bed)
2. He/He Heat Exchanger
3. He-Primary Circuit Blower
4. Gasifier
5. Process Steam High Superheater
6. Degaser
7. Process Steam Superheater
8. Steam Generator with Intermediate Superheater
9. He-Intermediate Circuit Blower

Auxiliary Circuits

10. Reheat Exhaust Intermediate Circuit
11. Reheat Exhaust Cooling Circuit
12. Oxidation Catalyzer
13. He-Recouperation Heat Exchanger
14. Activated Charcoal Filter with Cooler
15. Molecular Screen
16. Low-Temperature Absorber

Steam Power Installation

Drive Units

20. HD-High Pressure Turbine
21. MD-Middle Pressure Turbine
22. ND-Low Pressure Turbine
23. Generator
24. Exhaust Steam Turbine (GE)
25. Generator

Condensation and Water Purifying

30. Main Turbine Condenser
31. Exhaust Steam Turbine Condenser
35. Thermo Water Purifier
36. Distillate Desalting
37. Exhaust Steam Condensation Cleaning

Condensation and Feedwater Circuits

40. Main Condenser Pump
41. Main Feed Pump
42. Process Steam Feed Pump
43. Exhaust Steam Condensation Pump
46. ND-Low Pressure Preheater
47. Feedwater Container

Gas Generating Plant

Gas Cleaning Plant and Gas Compression

50. Raw Gas Dust Remover
51. Gas Cleaning (wash)
52. Oil and Tar Removal
53. Rectisol Wash
54. Methanization
55. Gas Drying
56. Desulfurization Plant
57. Waste Water Purifying
58. Product Gas Compressor (Condenser)
59. SNG-Compressor (Condenser)

Gas Heat Exchanger with Reheat Recovery

61. Gas Recouperation Heat-Exchanger
62. Raw Gas Cooler/Process Steam Superheater
63. Mixed Gas Cooler/Process and ND-Low Pressure Steam Generator
69. Wash Water Cooler with ND-Low Pressure Steam Generator

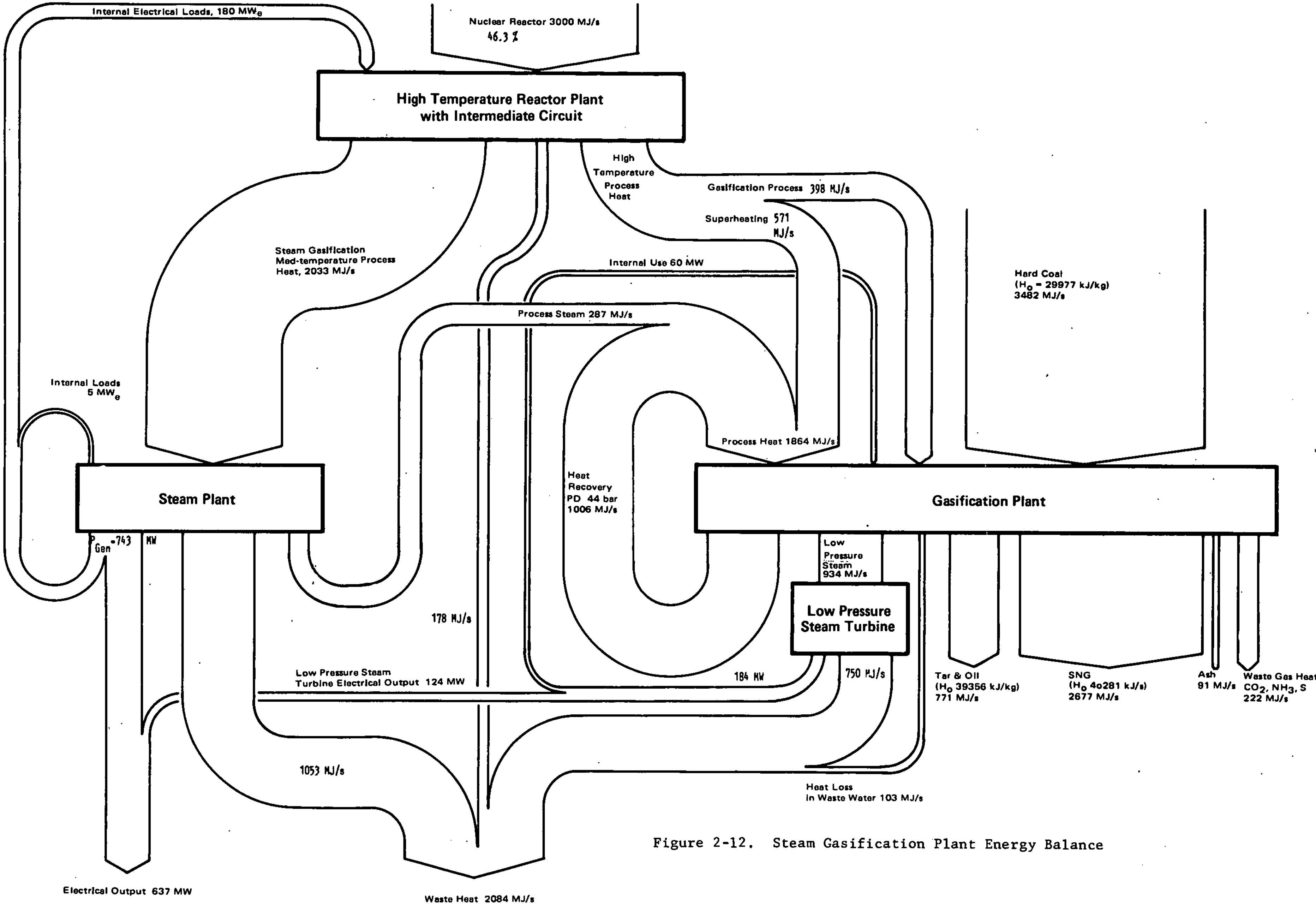


Figure 2-12. Steam Gasification Plant Energy Balance

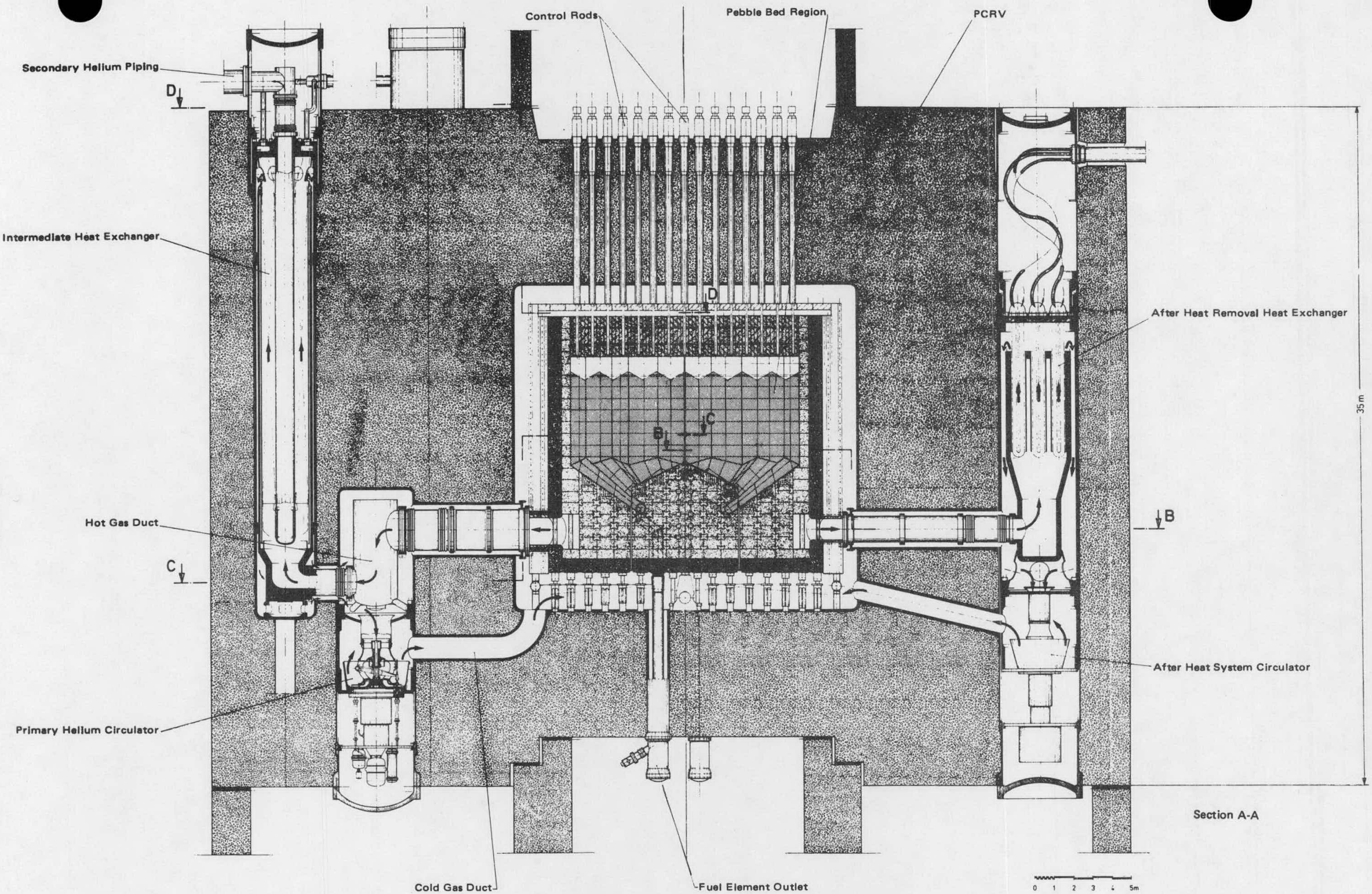


Figure 2-13. WKV Primary Circuit (Alternative A), Vertical View

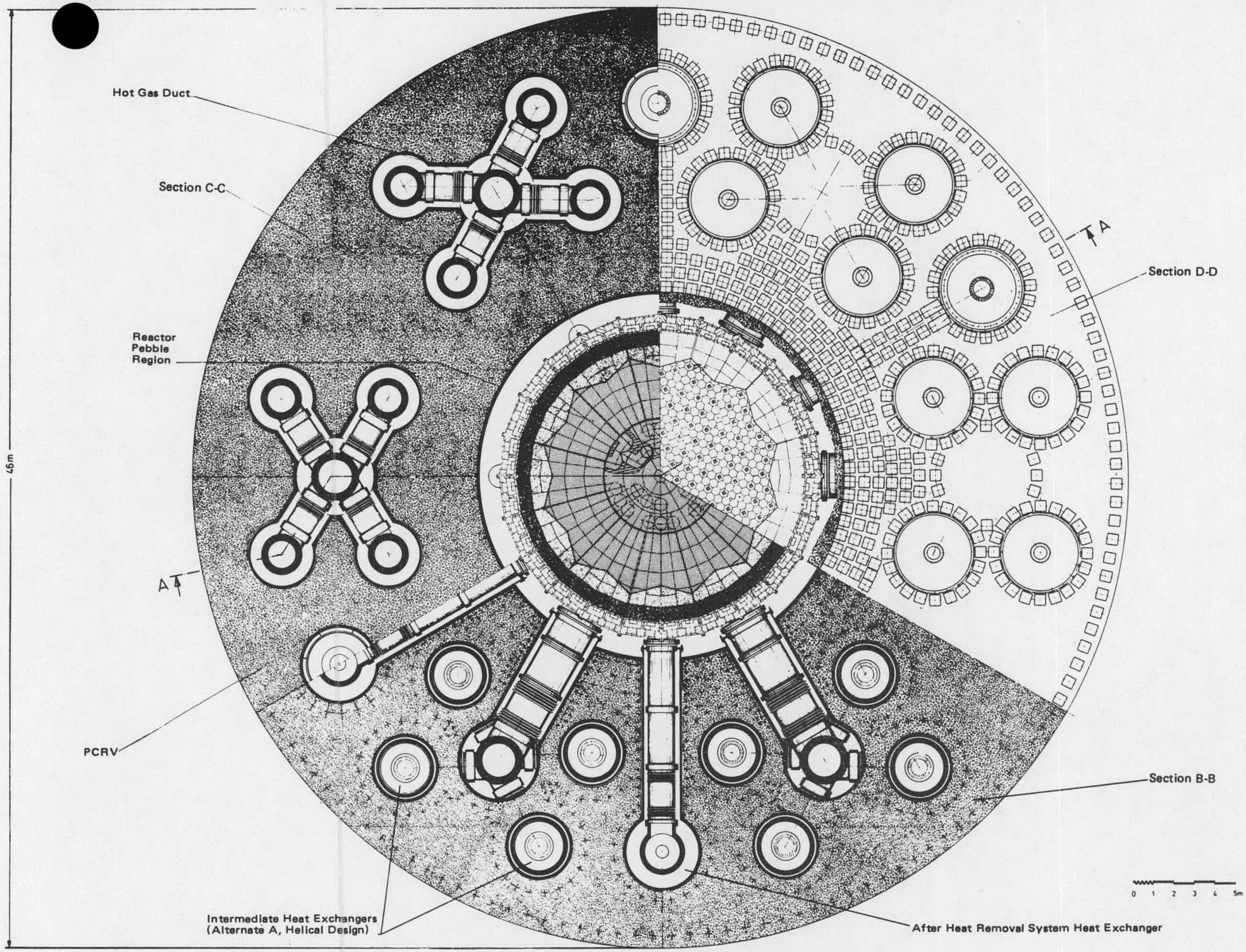


Figure 2-14. WKV Primary Circuit (Alternative A), Horizontal View

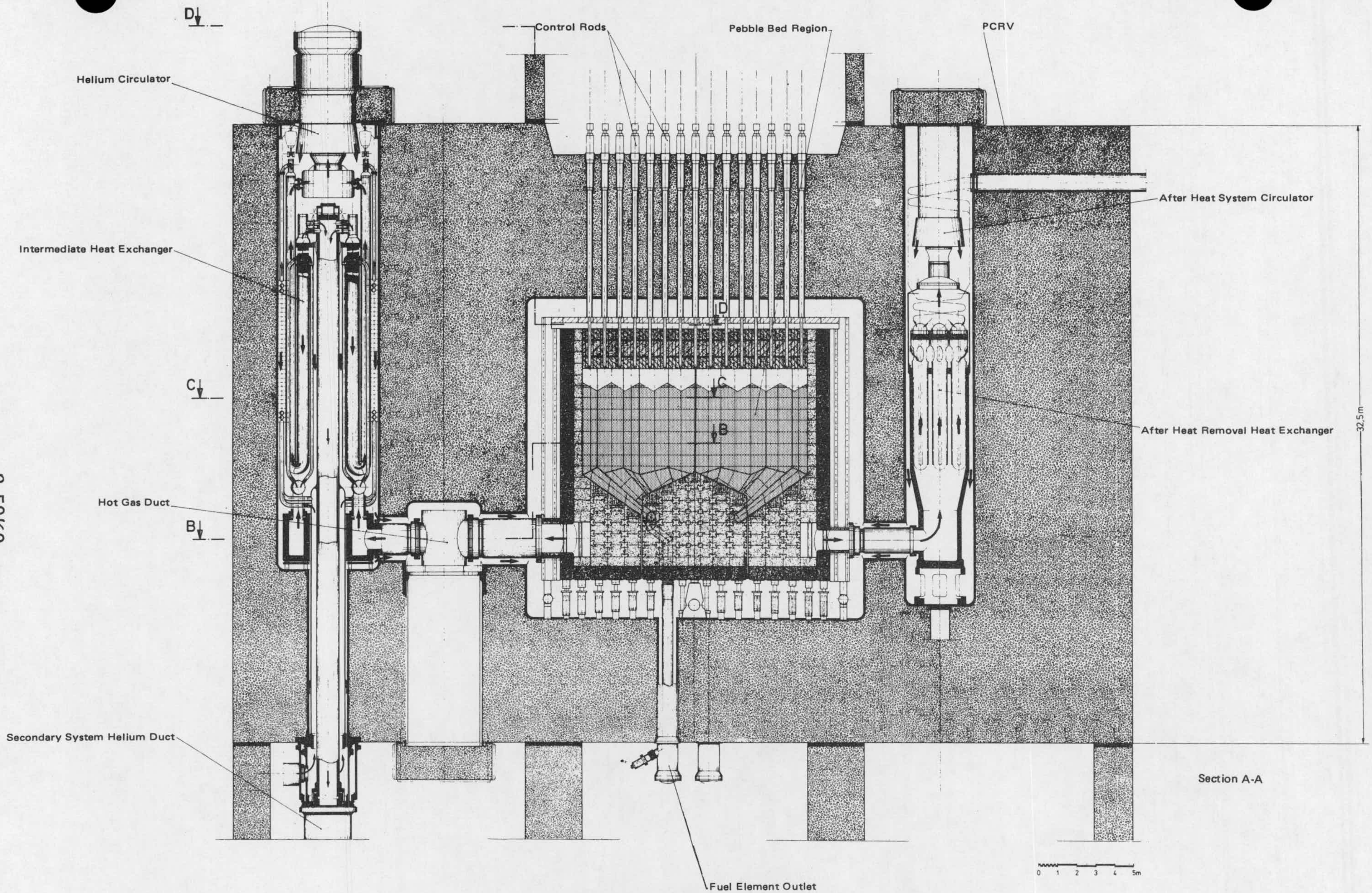


Figure 2-15. WKV Primary Circuit (Alternative B), Vertical View

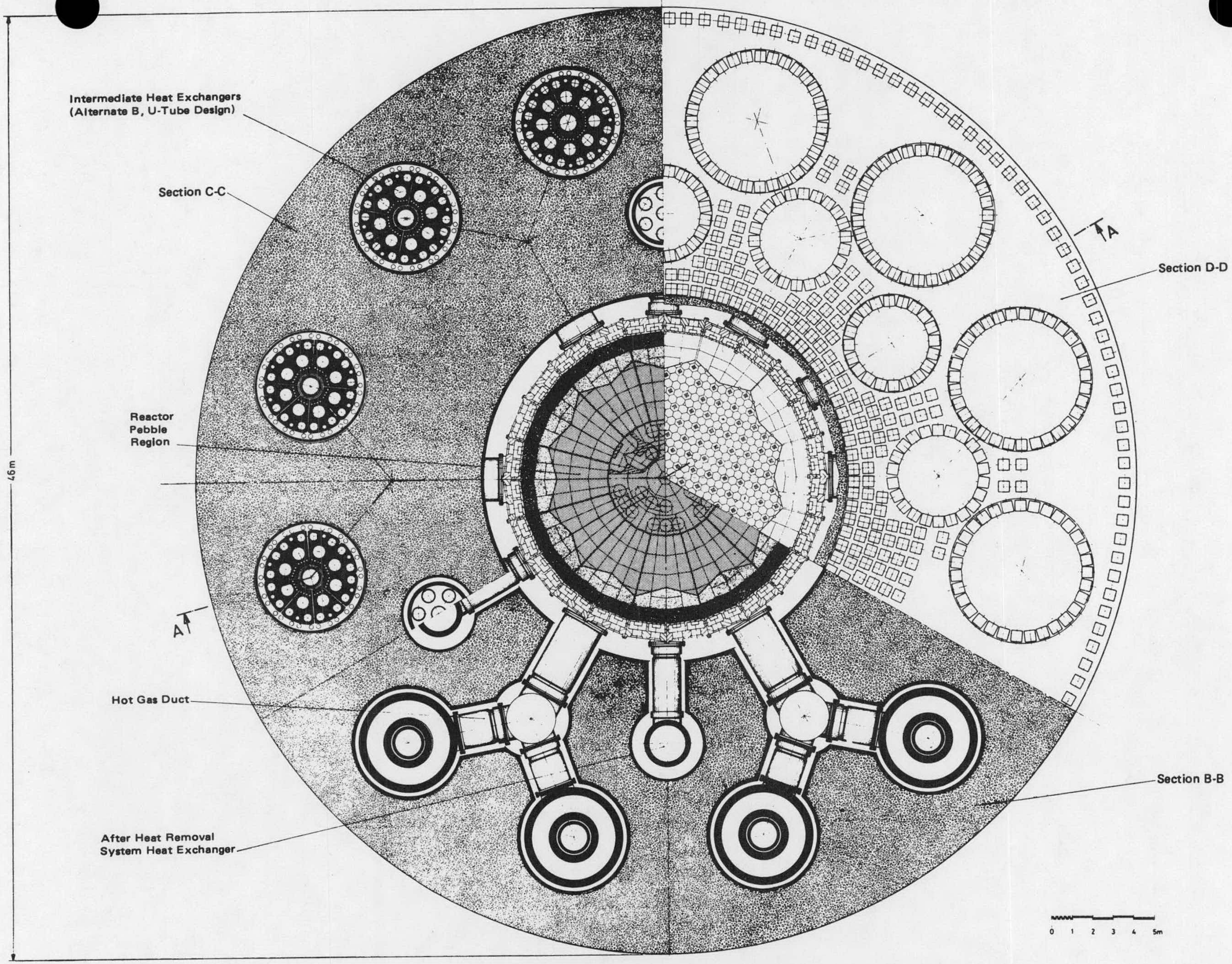


Figure 2-16. WKV Primary Circuit (Alternative B), Horizontal View

In both cases the reactor core is cooled by helium at a pressure of 40 bars. In Arrangement A (Figures 2-13 and 2-14), 300°C helium enters the reactor at the bottom from a noncoaxial circulator return duct. From the inlet plenum at the bottom of the core, the helium flows upward along the sides of the reactor cooling the thermal shield in its passage. Once above the core, the flow is turned downward through the upper thermal shield and reflector and passes down through the core pebble bed. Upon reaching the core bottom, the gas, now at an average temperature of 950°C, passes through the lower reflector. The gas is collected in an annular exit plenum located at the circumference of the graphite core container and at an axial elevation opposite the six exit ducts. The horizontal exit ducts containing hot helium are separated from the PCRV liner by an annular gas space in which cool (300°C) helium is bled from the reactor inlet plenum. Since this helium flow bypasses the heated region of the core, its flow rate is deliberately kept small using flow restrictors. After leaving the reactor and entering one of the six hot gas ducts, the flow in each loop enters a distribution chamber and is routed to one of four parallel connected intermediate heat exchangers in the case of Alternate A (Figure 2-14), or to one of two parallel connected intermediate heat exchangers in Alternate B (Figure 2-16). The number of heat exchangers per loop is in part determined by the flow and heat transfer areas of the two heat exchanger designs and the necessity of obtaining a large helium temperature drop across each unit. This latter concern is important to ensure that the PCRV liners and concrete are not exposed to excessive gas temperatures.

The two intermediate heat exchanger (IHX) designs under consideration have been prepared with the objective of facilitating IHX installation, testing, and servicing. Alternate A (Figures 2-13 and 2-14) uses helical construction with the primary gas entering at 950°C near the bottom and passing upward on the outside of the helically wound tube bundle. Near the top, the primary stream passes through flow ports into an annular flow space around the outside of the unit; it then flows downward and leaves at 300°C near the bottom to be sent to the primary circulator. The 240°C secondary helium stream enters at the top of the IHX and flows downward inside the helical tube bundle. Near the bottom it emerges from the tube bundle into a central cavity, where it is turned and directed upward to the IHX outlet, where the gas temperature is 900°C. Each IHX has a rated heat load of 125 MWth. They are installed and serviced from the top of the PCRV.

In Alternate B (Figures 2-15 and 2-16) a U-tube IHX is used with two parallel connected heat exchangers in each of the six primary helium circuits. In this alternative hot gas (950^{o}C) from the reactor enters a distribution chamber; there is one such chamber in each of the six circuits. The chamber divides the flow between the two IHX units. Primary gas from the reactor enters the IHX annular inlet plenum at the bottom and is routed to one of eight U-tube bundle heat exchanger modules. The legs of the U-tube are of unequal length (more like a J rather than U) to accommodate thermal expansion and the manner in which the tube bundle is supported. The primary gas is directed part way up the axial length and made to enter the end of the short length of the J-tube bundle. It flows on the shell side of the bundle downward, turns, and flows up the long leg to the top of the IHX unit, where the integrally mounted primary gas circulator is located. Primary gas temperature at the circulator is 300^{o}C. The circulator discharge flows downward over the outside of the heat exchanger internals to the bottom of the unit. There the flow streams from the eight modules are collected and ducted to a common point to be combined with the helium returning from the other heat exchanger in each primary circuit before being routed back to the reactor. The secondary helium flow stream, initially at 240^{o}C, passes up through the bottom of the PCRV into an inlet plenum, where it is routed by 32 feed lines to the top of the long legs of the 8 J-tube modules. It flows down, turns, and then flows upward as it passes through the J-tube bundle; upon emerging from the tube bundle the secondary helium, now at a temperature of 900^{o}C, turns and flows straight down a central cavity containing the combined discharges of all eight modules. The gas exits the PCRV at the bottom, and access to the IHX for servicing is from the top of the PCRV. While it is necessary to first remove the top-mounted circulator to gain IHX access, it is thought that inspection of the U-tubes will be facilitated, since both ends of the bundles would then be exposed for examination. The heat load for each of the heat exchangers is 250 MWth.

The helium circulators in the primary circuit are electrically driven, draw approximately 8 MWe, and circulate 148 kg of 300^{o}C helium per second. In Alternate A, using the helically wound intermediate heat exchanger (Figure 2-9), the primary circulator is mounted at the bottom of the PCRV, separate from the IHX. In Alternate B, using the U-tube intermediate heat exchanger, the primary circulator is located at the top of PCRV above the IHX. The method of flow control is by using constant speed motors with variable inlet control for load change and shutdown.

The after-heat removal system is designed to remove two percent (60 MWth) of normal rated reactor power under depressurized conditions equal to one bar containment pressure. The system consists of four separate loops, each conservatively sized to remove 1% of rated reactor power. Each decay heat loop contains a variable-speed, electrically driven helium circulator and a helium-to-water heat exchanger--all located within the PCRV. The circulators are single-stage axial flow units mounted below the NWA system heat exchanger (Section 5.3).

2.2.3.1.3 Secondary Circuit

The secondary circuit passes through the coal gasifier, the process steam superheater (final), the raw coal degasifier, the process steam reheater (intermediate), turbine steam generator and re-heater, and the secondary helium circulator (Figure 2-11). In the present design, secondary helium leaves the PCRV through double isolation valves and is transported to the product gas generation building. There it enters the coal gasifier at 900°C and emerges at 836°C. In the gasifier, devolatilized coal from the raw coal degasifier is reacted with steam to produce process gas, which is then piped to other chemical plant equipment for further processing before release for final distribution. Upon leaving the gasifier, the helium passes to the high-temperature steam superheater. This unit provides high-temperature process steam for the coal gasifier and degasifier. The helium emerges at a temperature of approximately 774°C and proceeds to the raw coal degasifier. There heat from the helium is used to vaporize volatile constituents in the coal. The helium leaves at 750°C and passes to the superheater region of the steam generator, where it provides an initial stage of superheating for the process steam and also reheats part of the high-pressure turbine exhaust. Before proceeding to the secondary circulator, the helium passes through the steam generator region. Helium enters the electrically driven circulator at 240°C. The secondary helium circuit has an operating pressure of 42 bars (2 bars greater than the primary circuit) and a helium flow rate of approximately 146 kg/s. The method of flow control has not been established, but mechanical throttling and a by-pass arrangement are among those being considered. Of all the major components in the secondary helium circuit, only the IHX is located within the PCRV.

2.2.3.1.4 Containment Structures

The PCRV planned for the PNP steam gasification plant is an integral upright cylinder of prestressed concrete, which, except for the number of cavities, is similar to the one used on the PNP hydrogasification plant. Cavities within the PCRV house the reactor core, intermediate heat exchangers, and primary helium circulators. As can be seen from Figure 2-14, there are 24 cavities needed for the helical IHX units, six more somewhat smaller cavities for the circulators and distribution chambers, as well as four for the decay heat removal system heat exchangers and circulators. Thus, counting the central reactor core cavity, there are 35 major PCRV cavities for design Alternate A. For design Alternate B, which uses fewer of the U-tube-type IHX units, the total number of cavities is 23 including those for the six flow distribution chambers. When compared to the hydrogasification plant PCRV, the greater number of equipment cavities used in the steam gasification plant increases the required size of the PCRV. The diameter increases from 44 to 46 m and the height changes from 31 to 35 m. The larger dimensions apply to the steam gasification plant PCRV for both IHX designs. The other PCRV information regarding design pressure and cooling previously given for the PNP hydrogasification plant (Section 2.2.2.1.3) is applicable to the steam gasification plant as well.

The previous hydrogasification plant information on the reactor and auxiliary structures is basically applicable to the steam gasification plant with few dimensional changes to reflect differences in PCRV design.

2.2.3.1.5 Control and Instrumentation

Refer to Section 2.2.2.1.4 for control and instrumentation data applicable to both the hydrogasification and steam gasification PNP plants.

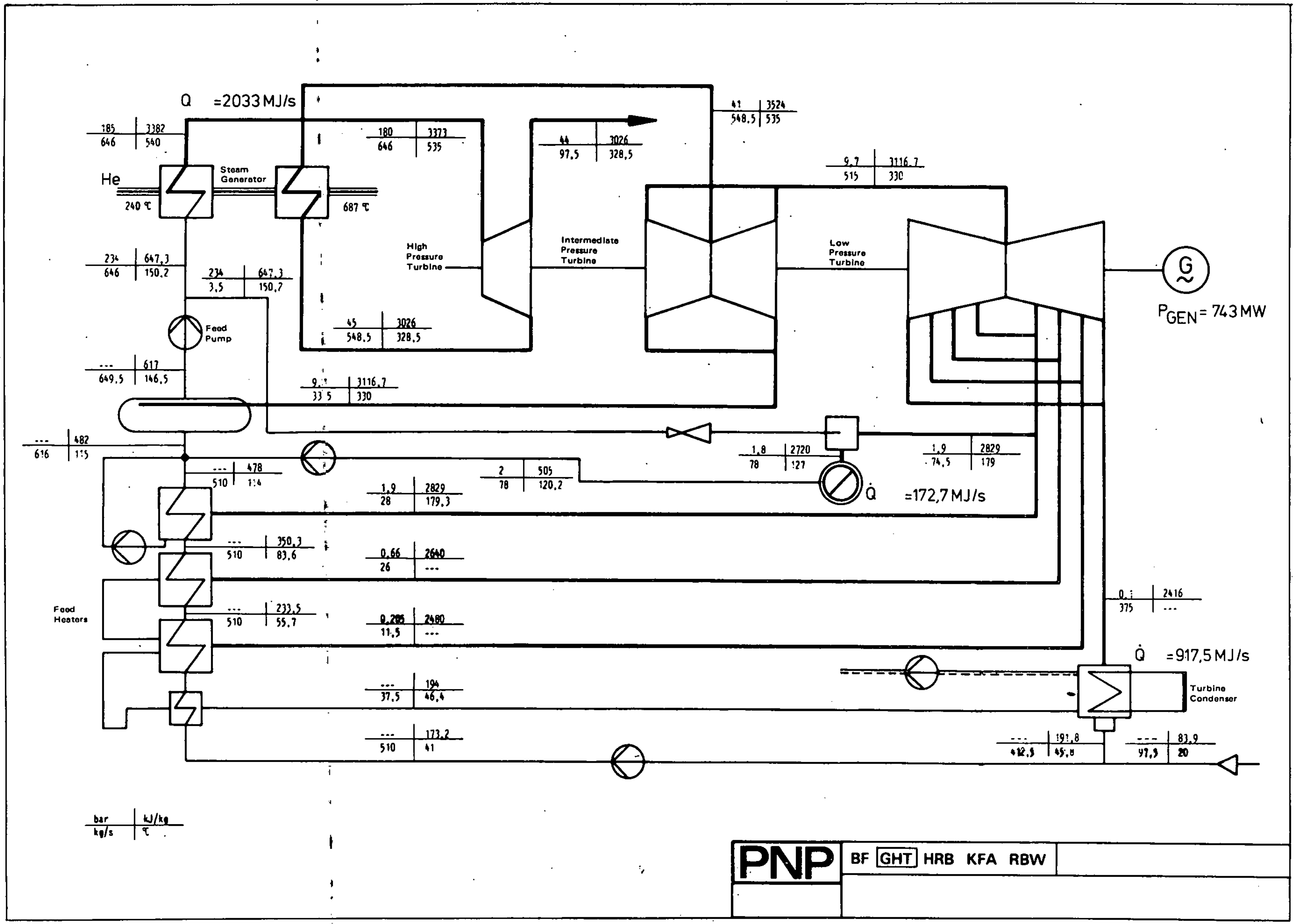


Figure 2-17. Steam Plant for the WKV Plant

2.2.3.2 Steam Plant

Steam is produced for raw coal degasification, the steam gasification process, and the production of electricity in a turbine-generator set; the steam plant is depicted in Figure 2-17. At design conditions steam is produced at a rate of 646 kg/s, a pressure of 185 bars, and a temperature of 540°C. From the steam generator the steam passes directly to the turbine, which is a three-pressure stage, single-shaft condensing-type unit. The electrical rating is 743 MWe. Steam discharged from the high pressure units is divided into two streams, one of 97.5 kg/s is fed to the first and second process superheaters to provide chemical process steam. The remaining 548.5 kg/s is reheated to 535°C before being sent to the intermediate pressure turbine unit. Four turbine steam extractions are used for regenerative heating of the feedwater, which is returned to the steam generator at 150°C. Waste heat is dissipated to the environment principally through the use of dry-type cooling towers.

In the chemical process plant, low-pressure steam of 6 bars and 159°C is obtained through waste heat extraction. This steam is expanded in a separate turbine-generator at a rate of 364.2 kg/s, and an additional 184 MWe are obtained. Of the total generating capacity of 927 MWe, 290 MWe are consumed within the plant (Figure 2-12), leaving 637 MWe available for external distribution.

2.2.3.3 Electric Plant

The electric plant supplies power for internal facility use with any excess being distributed externally through the normal power grid network. The conceptual arrangement of the electrical plant has been established, but additional engineering analyses are required and are in progress to define the specific design. The basic concept is to connect the main turbine-generator set to the external power distribution network through two parallel transformers and to connect the smaller waste heat turbine-generator set to the external network through a separate power transformer. The power produced in the main generator at 21,000 V is fed through the parallel connected double-winding, three-phase isolation transformers. Either one of the parallel connected power transformers can be selected to supply power to the PNP plant during startup and shutdown operations. During normal PNP plant operation, in-house power needs are taken from generator feeder busses and distributed through six 3-winding transformers to the six sections of the 10 kV main switchgear installation. One transformer winding feeds an intermediate circuit blower drive motor. The primary circulator drive motors are powered from corresponding 10 kV main switchgear installation. Emergency power supplies in the form of standby diesel generations and dc power sources are planned for emergency plant conditions.

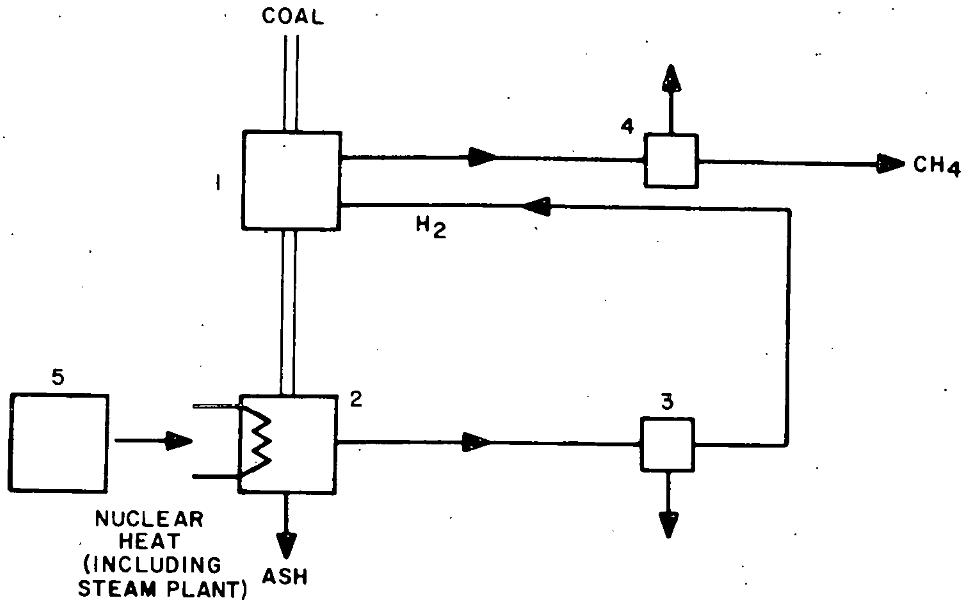
2.2.3.4 Fuel Handling System

Section 2.1.7 contains information descriptive of the fuel handling system.

2.2.4 COMBINED GASIFICATION PROCESS CYCLE

Initial work has been performed to develop the conceptual design of a combined-cycle plant using both the steam and hydrogasification processes. One of the principle advantages of such a plant is its potentially improved performance over either of the steam or hydrogasification processes. Such performance improvements are possible because the combined-cycle plant would virtually eliminate the discharge of residual coke, tars, and oils. Instead, discharges would be consumed within the plant to increase methane production.

In the combined-cycle plant, as it is currently planned, the feed coal would be fed to the hydrogasification and steam gasification processes in series. The principles of the combined-cycle process using hard coal are shown in Figure 2-18. The coal is first fed to the hydrogasifier. There nearly 50% of the coal would be converted in an exothermic reaction using purified hydrogen taken as a product of the steam gasification stage. The product gas leaving the hydrogasifier is used to preheat the incoming hydrogen, and is then cooled down and cleaned in water-washing equipment. Next, the H_2S and CO_2 are taken out. The product gas from the hydrogasifier then enters a low-temperature separation stage and is separated into H_2, CO, and CH_4. The methane is then distributed as the product of the plant.



1 – Hydrogasification

2 – Steam gasificaiton

3 – Gas purification

4 – Gas purification

5 – Nuclear reactor including heat exchange system
(including steam plant)

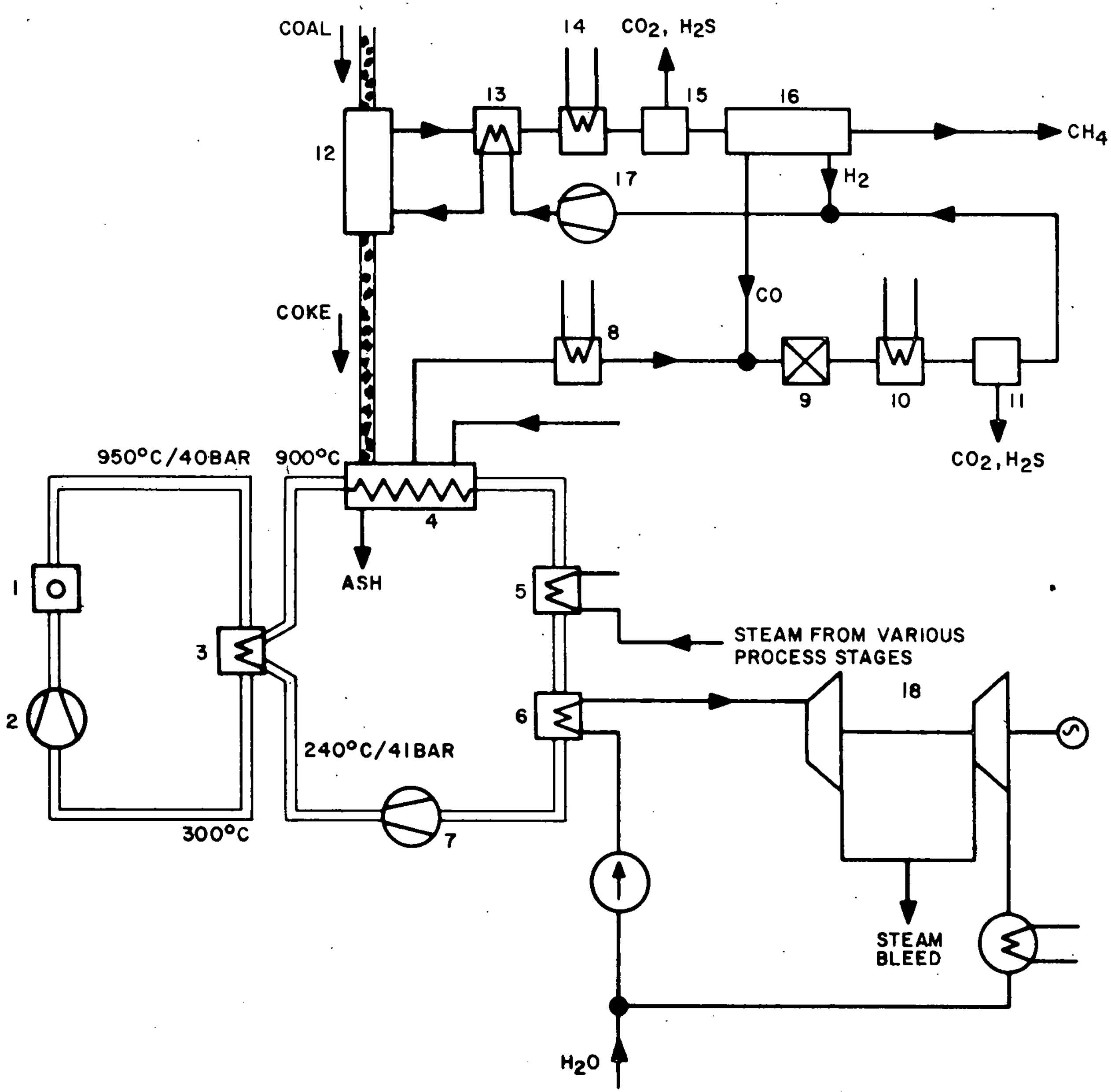
Reactions:

Hydrogasification	$C + 2H_2 \rightarrow CH_4$	(900°C)
Steam gasification	$C + H_2O \rightarrow CO + H_2$	(800°C)
Shift conversion (in 3)	$CO + H_2O \rightarrow CO_2 + H_2$	(400°C)
Net Process Reaction	$2C + 2H_2O \rightarrow CO_2 + CH_4$	

Figure 2-18. Principles of the Combined Gasification Process

The residual coke from the hydrogasification process is introduced as feed to the steam gasifier together with hot steam. The product gas leaving the steam gasifier is cooled and passes a shift converter in which the CO is converted to CO_2 and hydrogen. This shift converter uses gas from the steam-gasification step and the CO stream coming from the low-temperature gas separation stage. After cooling and passing through gas purification equipment to remove H_2S and CO_2, the hydrogen is compressed and used as feed for the hydrogasification stage. The H_2 coming from the low-temperature separation stage is also mixed with this steam. In the intermediate helium circuit which contains the IHX, there is a superheater for steam and a steam generator. The steam from the steam generator is used to operate the turbine-generator set. The feed for the superheater is steam available at the various stages of the overall process. A schematic flow diagram of the overall process is shown in Figure 2-19.

Table 2-3 presents a comparison of the steam, hydro-, and combined-cycle gasification processes. Note that, in addition to the data on coke, tar, and oil production, the methane production rate is significantly improved. Further, because the hydrogen produced in the steam-gasification state is consumed in the hydrogasification process, there is no need for a second methanation reactor. While the process shown in the preceeding figures is based on using hard coal, in theory any type of coal could be used. Because of its potentially improved performance, the combined-cycle process will receive further study as part of the German Process Heat Program.



1) Nuclear reactor
2) Helium circulator
3) IHX
4) Steam gasifier
5) Steam superheater
6) Steam generator
7) Helium circulator (intermediate circuit)
8) Cooler
9) Shift converter
10) Cooler
11) Gas purification
12) Hydrogasification
13) Hydrogen preheater
14) Cooler
15) Gas purification
16) Low temperature separation stage
17) Hydrogen compressor
18) Steam turbine plant

Figure 2-19. Combined Gasification Process Schematic

TABLE 2-3

COAL GASIFICATION COMPARISON DATA

		Hydrogasification of Lignite	Steam Gasification of Hard Coal	Combination of Hydro and Steam Gasification of Hard Coal*
Primary System Data				
Reactor thermal power	MWth	3000	3000	3000
Helium temperatures	°C	950/300	950/300	950/300
Helium pressure	bars	40	40	40
Secondary System Data				
Helium temperatures	°C	-	900/240	900/240
Helium pressure	bars	41	41	41
Chemical Process System Data				
Gasification temperatures	°C	850	786	900/793
Surplus of electrical power	MWe	118	637	300
Coal input	tons/hr	2317	418	780
Methane production	$10^3 m^3$/hr	368	239	550
Coke production	tons/hr	286	-	-
Tars/oil production	tons/hr	-	71	-

* Values are approximate pending design finalization

2.3 PNP AND HTR-K OVERALL EVALUATIONS

This section will provide evaluations of the HTR-K and PNP concepts described in Sections 2.1 and 2.2. The evaluations have been based on the following considerations: the status of technological development (i.e., the technology base), the proposed development plans, and the detailed component and systems evaluations of Sections 3, 4, and 5.

2.3.1 HTR-K TECHNOLOGY BASE

Gas-cooled reactors using steam cycles for electric generation have been built and operated over the past twenty years in a number of countries, including the United Kingdom, France, the United States, and the Federal Republic of Germany. The experience gained in some European gas-cooled reactors is discussed in the report "Construction and Operating Experience of Selected European Gas-Cooled Reactors."[23] That report demonstrates that much of the work done for the Dragon reactor and the gas-cooled reactors of the UK is relevant to advanced gas-cooled reactors. That report also describes the specific pebble bed experience gained through the operation of the AVR reactor and the construction and licensing of the THTR pebble bed prototype plant.

Gas-cooled reactor construction and operating experience within the United States is considerably less extensive than in Europe, although the Peach Bottom reactor and the Fort St. Vrain prismatic reactor have generated much important fuel and component data. The large prismatic designs, although not actually built, went through part of the U.S. licensing process before the applications were cancelled. Significant gas-cooled reactor licensing precedents were established and much fundamental engineering work was conducted. This experience is generally applicable to large pebble bed plants.

The technology base is summarized in Table 2-4. Much of this experience is generally applicable to the HTR-K, such as graphite technology, fuel technology, PCRV technology, etc., regardless of configuration differences between previous plants and HTR-K. Other portions of the experience base are specifically applicable to HTR-K, such as that obtained from AVR and THTR. These include

TABLE 2-4

HTR-K TECHNOLOGY BASE SUMMARY

Plant	Thermal Power (MWt)	Location	Experience Type	Primary Coolant	Startup Date	Fuel Type
Dragon	20	U.K.	Const. & Operation	Helium	1965	Rods
Peach Bottom	115	USA	Const. & Operation	Helium	1967	Rods
AVR	46	FRG	Const. & Operation	Helium	1967	Pebble Bed
Fort St. Vrain	842	USA	Const. & Startup	Helium	1976	Prismatic
THTR	750	FRG	Construc- tion	Helium	1981	Pebble Bed
AGRs	135-1690	UK	Const. & Operation	CO_2	1956-1979	Rods
GCRs	105-1450	France	Const. & Operation	CO_2	1959-1972	Rods
HTGR Studies	2000-3000	USA	Licensing & Studies	Helium	-	Prismatic

pebble bed fuel, fuel handling, transient analysis of THTR, licensing precedents, etc. Reference No. 24 and Sections 3 and 4 of this report should be consulted for details on AVR and THTR experience; however, a brief summary is provided in Table 2-5.

TABLE 2-5

HTR-K TECHNOLOGY BASE-PEBBLE BED SUMMARY

	AVR	THTR
Power (MW_{th})	46	750
Circulators:		
Type	Electric	Electric
Size (MW)	0.128	2.5
Steam Generators:		
Type	Involuted	Helical
Helium Flow	18 kg/s	296 kg/s
Control Rods:		
Type	Not Pebble Bed Inserted	Pebble Bed & Reflector Inserted
Drive	Electric	Electric & Pneumatic
Reactor Vessel:		
Type	Steel	PCRV-Large Cavity
Pebble Bed Core:		
Power Density	2.2 MW/m^3	6 MW/m^3
Refuelling	Continuous	Continuous
Outlet Temp.	850-950°C	750°C

2.3.2 HTR-K DEVELOPMENT PLAN

As discussed in Sections 1 and 5, the German program for development of an electricity generating pebble bed plant has apparently been shifted from a plan based on the HTR-K to one based upon the direct-cycle gas turbine concept (HHT). As a result, there is no longer a development plan specifically for the steam cycle HTR-K.

Even though the HTR-K activities have been at least temporarily cancelled, component development will not be completely lost, due to PNP development plans which share common technologies with the HTR-K. Both the HKV and WKV versions of the PNP concept include steam plants for the generation of process steam and electricity. The steam will be produced by steam generators not unlike those planned for the HTR-K. In addition, much of the PNP technology (e.g., PCRV, circulators, etc.) is very similar to that needed for an HTR-K plant. Therefore, future resurrection of an HTR-K program would have large portions of needed development work already conducted within the PNP development program.

2.3.3 HTR-K EVALUATION

Section 2.3.1 showed that there has been reasonably extensive experience in gas-cooled reactors which is either partially or completely applicable to the HTR-K concept. Sections 3, 4, and 5 describe the components and systems of the HTR-K to the extent that information was provided by the German participants. Those sections also provide evaluations for specific components and systems.

It generally appears that the HTR-K concept is a reasonable extrapolation of existing technology. The basic designs seem to be based on sound engineering principles. Particular strong points are fuel cycle flexibility and good basic fuel irradiation experience in the AVR reactor. Another area of strength is the improved attention to inspectability and maintainability, i.e., better improved access to the PCVR liner and major components for repairs. Straight tube superheaters in the steam generators, while not without design problems, such as wear on the tubes and thermal stresses, do offer improved inspectability over helical superheater tubes. The steam plant is based upon proven Rankine Cycle technology and includes a steam reheater in lieu of the helium reheater, which should result in improved plant reliability, although at a loss in plant efficiency (2%).

Experience at AVR and at the full-scale THTR fuel handling mechanism test facility has shown that the on-line fuel handling methods appear operationally acceptable. Operation of the THTR will provide additional proof of the concept. The on-line refuelling capability has been shown, in the report "Assessment of Gas-Cooled Reactor Economics," to offer major economic advantages through potentially higher capacity factors than attainable by batch reloading plants.[20]

The pebble bed fuel has been extensively tested at AVR. Fuel dynamics and performance have been examined in the 1:6 scale model of the THTR core and the KAHTER critical facility. These and other tests give reasonable confidence in the fuel flow behavior, physics behavior, thermal-hydraulic behavior, rod insertion behavior, etc. (refer also to Section 3). It appears that the fundamental information has been established and with satisfactory completion of ongoing test programs that development of a 3000 MWth core is achievable.

The report "Safety and Licensing Evaluation of German Pebble Bed Reactor Concepts" concludes in part that there do not appear to be any aspects of HTR-K that would preclude U.S. licensing.[22] There are, however, a number of areas that would require further qualification to be acceptable to the U.S. Nuclear Regulatory Commission, such as the imposition of simultaneous seismic event and depressurization accident loads on the plant and full-scale testing of a 1/6 region of the core. The differences between US and FRG licensing requirements pose many other questions similar to these that will require resolution.

The development plan for PNP (Section 2.3.5) should resolve many areas of uncertainty, including upper side reflector behavior, spatial Xenon detection and control, transient behavior, etc. It should also be pointed out that the low power density of the pebble bed core and the OTTO cycle fuel temperature profile appear to offer improved post-accident behavior, but further development work is needed.

It should be noted that the apparent acceptability of the HTR-K concept is based upon engineering considerations only. Economic incentives for development of a steam cycle gas-cooled reactor have not been considered in this Topical Report.

2.3.4 PNP TECHNOLOGY BASE

The twenty years of steam cycle gas-cooled reactor experience, including that of the recent AVR and THTR pebble bed plants, forms a significant portion of the technology base for PNP as well as HTR-K. The general areas of the experience outlined in Section 2.3.1 applicable to the PNP concept include core design, PCRV design, helium heated heat exchanger design, electrical circulator design, etc. It can be seen that gas-cooled reactor experience applies to PNP primarily in the nuclear core and related support systems, such as gas purification and after-heat removal. A second equally important part of the PNP technology base to be examined, therefore, must be the balance of plant (i.e., experience in process heat equipment such as steam reformers, He/He heat exchangers, high-temperature gas ducts, etc.). The experience existing for the reactor and the balance of plant is summarized in the following subsections. It should be made clear that the German chemical process plant data was excluded from the General Electric work scope. Therefore, only a general review of the gasification technology was made. The PNP components in the nuclear plant have been examined in Section 4.

2.3.4.1 Reactor Experience at PNP Conditions

To date the only experience at PNP temperatures (950°C at the core outlet) has been at the AVR pebble bed test reactor. The outlet temperature in February 1974 was elevated to 950°C. The three years' experience has allowed some preliminary conclusions about pebble bed fuel performance at PNP temperatures. Another report discusses AVR experience in more detail and concludes in part that the fuel did not exhibit higher corrosion or damage rates at the higher temperature and that the overall plant availability (70-90%) was not affected.[23] It appears that fission product releases can be maintained at an acceptable low level at the higher temperatures. In short, there is evidence that the pebble bed fuel is capable of sustained operation at 950°C.

2.3.4.2 Hydrogasification Experience

There has been a fairly extensive base of experience for conventional fossil fuel heated steam reformers. The processes used to date have started with materials such as natural gas, refinery gas, and light benzenes, which are desulfurized and mixed with steam to the desired steam/methane ratio (2/1 to 5/1, depending on the input material. The current reforming temperatures have been in the range of 750-850°C, which is directly applicable to the nuclear heating helium steam reformers.

The EVA reformer facility at the Nuclear Research Center in Jülich, West Germany, has used single tubes (i.e., partial-length tubes that examine incremental changes in process gas at various positions in a full-length tube) to test helium-heated reformers and methanators. These tests have generated much useful data as discussed in Section 4.

The hydrogasification process has been examined under laboratory conditions and tested for about two years in a German pilot plant (HKV-I) operated by Rheinkraun AG in Wesseling, FRG. The pilot plant has a coal throughput of 200 kg carbon per hour with the following preliminary results:

Hydrogasification Pilot Plant

Lignite Throughput	800 kg/hr
Carbon Burnup	64%
SNG Methane Content	30%
Fluidized Bed Pressure	80 bar
Reaction Temperatures	800-900°C
Methane Production per Liter of SNG	$2NM^3CH_4/\ell$

An analysis of the pilot plant preliminary performance data by the Germans indicated that the basic performance assumptions for the PNP hydrogasification plant tend to be pessimistic.

2.3.4.3 Steam Gasification Experience

The steam gasification of coal is a more extensively (several decades) developed process both in the U.S.A. and Europe using such processes as LURGI, Koppers-Totzek, and Winkler. As in the case of hydrogasification, laboratory tests for nuclear steam gasification have been conducted. Bergbau Forschung in Germany has operated a pilot plant (WKV-I) for about one and one-half years in Essen. This plant has a 200 kg carbon per hour throughput with an input of hard coal. Preliminary results appear to support the general assumptions made for nuclear steam gasification.

2.3.5 PNP DEVELOPMENT PLAN

The German PNP program is advancing through the concept definition phase, during which designs for large commercial plants have been evaluated and alternatives narrowed to the HKV and WKV plants described in Section 2.2. Simultaneously with these commercial plant concept studies, efforts are under way to define the character of a PNP Prototype (demonstration) plant that would actually be a small (500-750 MWth) pebble bed process heat plant. Recently the design was settled on a two-loop concept with a pebble bed heat source of 500 MWth.

The steps planned to extrapolate the current PNP technology base to a 3000 MWth commercial plant are discussed in the following subsections and consist of moving from the completed laboratory and pilot plant experience to a semitechnical plant and finally the prototype plant. The general character of these steps and the time scale are shown in Table 2-6 and Figure 2-20.

2.3.5.1 Semitechnical Plants

The next stage in steam reformer/methanation development will be the operation of the EVA-II/ADAM-II facility (Figure 2-21) at KFA in Jülich, West Germany. The 30-tube reformer will be heated by electric heaters (10 MWe) and will be connected to a 6 MW methanation plant. The main objectives of this facility are to test a full-scale reformer tube bundle, short-term materials performance evaluation testing, test-out of multistage methanation techniques, and to prove the chemical heat pipe concept (i.e., closed loop transport of nuclear heat energy by chemical means). The facility is described in more detail in Section 4.

TABLE 2-6

EVOLUTION PROCESS FOR PNP PROCESSES

Process	Laboratory Tests	Pilot Plant	Semitechnical Plant	Prototype	Commercial Plant
Steam Reforming	Differential Reactor (10 KW)	1 Tube EVA-I	30 Tubes EVA-II	300 Tubes	400 Tubes
Methanation	Differential Reactor	250 kW ADAM-I	6 MW ADAM-II	48 MW	100 MW
He/He Heat Exchanger	-	-	IHX Plant 70 MW Bundle	125 MW Bundle	125-250 MW Bundle
Steam Gasification	1 kg carbon/hr	200 kgC/hr	2 tonsC/hr WKV-II	30tC/hr	Parallel 30tC/hr
Hydrogasification	1 kg carbon/hr	200 kgC/hr	2 tonsC/hr HKV-II	40tC/hr	Parallel 40tC/hr

The intermediate heat exchangers (IHX) needed for the steam gasification PNP will be first tested in a planned IHX facility. The IHX facility (Figure 2-22) will be designed in such a way that the IHX test section can accommodate either of the two alternative designs currently under consideration (helical and U-tube). The facility will test hot helium ducting as well as the IHX. The IHX test section has primary loop conditions (950°C inlet) on one side and intermediate loop conditions on the other side (900°C outlet).

The HKV-II and WKV-II plants are planned as the next stage in development of hydrogasification and steam gasification, respectively. These facilities will be located with the pilot plants and operated by the same companies who are testing the pilot plants. It is anticipated that these large-scale tests will test out components such as isolation locks, confirm material behavior, and produce more data on operational performance.

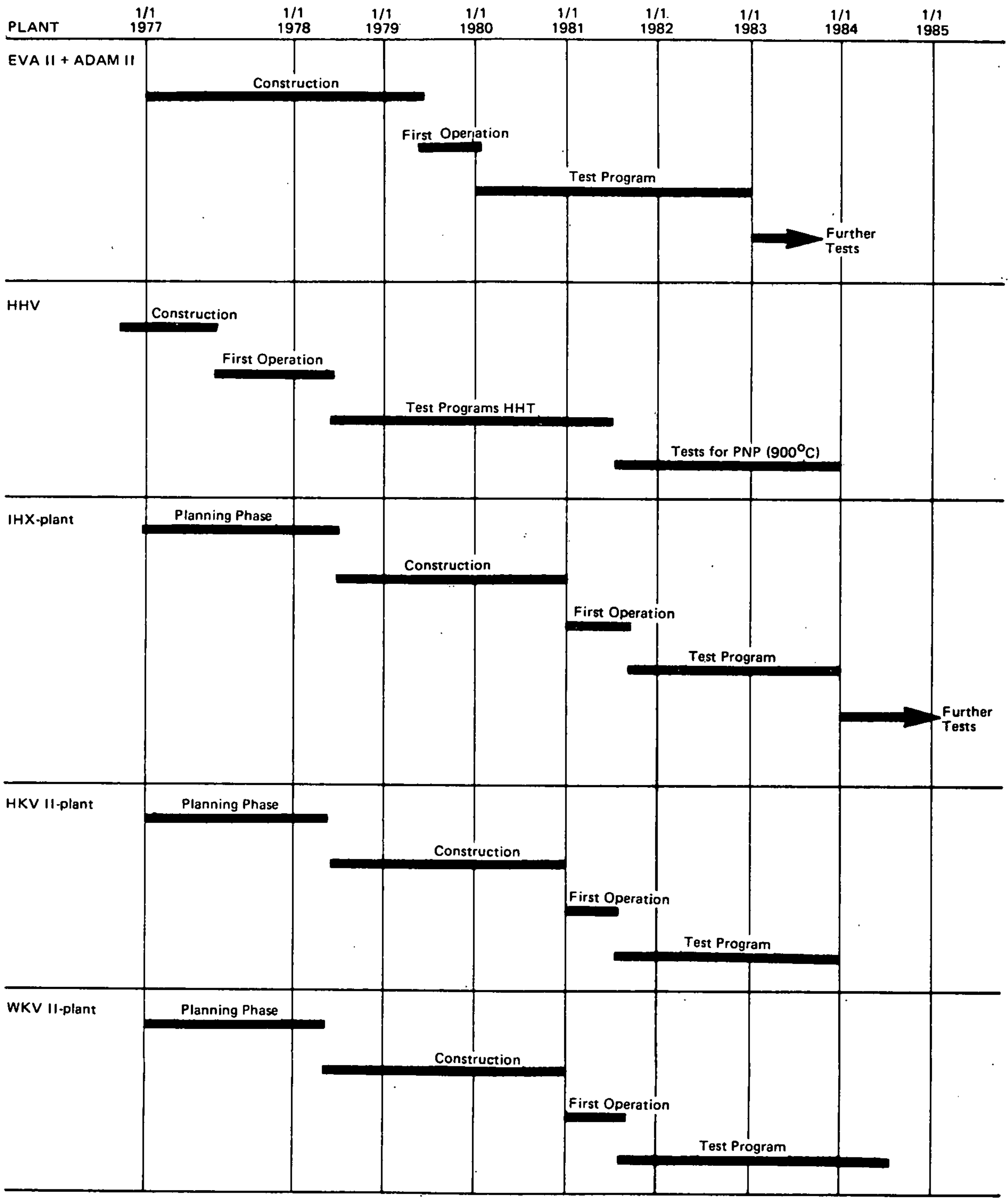


Figure 2-20. Semitechnical Plant Development Schedule

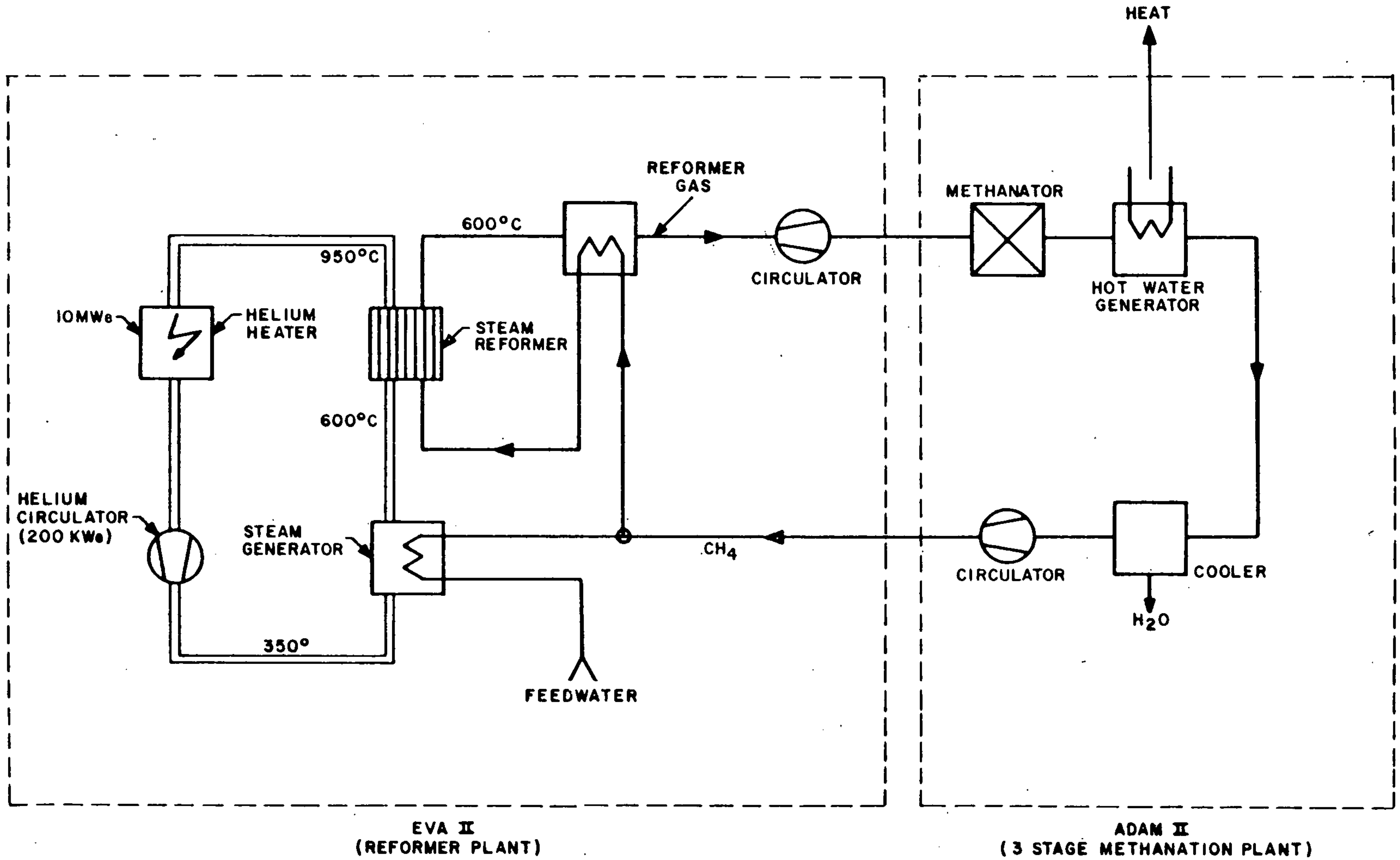


Figure 2-21. ADAM II/EVA II Test Plant

HKV-II is shown in Figure 2-23. Coal is added to a bunker and moved through the process by a CO_2 pneumatic system. After crushing and drying, the coal moves to the chemical reactor through isolation locks. In the reactor the coal reacts with preheated hydrogen and steam (oxygen is provided for process startup). The raw gas generated is processed and the residual coke is removed.

The WKV-II plant (Figure 2-24) is somewhat different since the chemical reactor is heated by helium whose temperature is controlled by a separate loop consisting of an electric heater and a recuperator. The coke is inserted into the reactor (gas generator), and ashes are removed from the bottom. Input water is heated in the first steam generator. The steam passes to a steam drum and through a superheater to the reactor. Some of the condensate in the steam drum is reheated in the raw gas cooler and in a second steam generator. The generated gas is processed as shown in the figure.

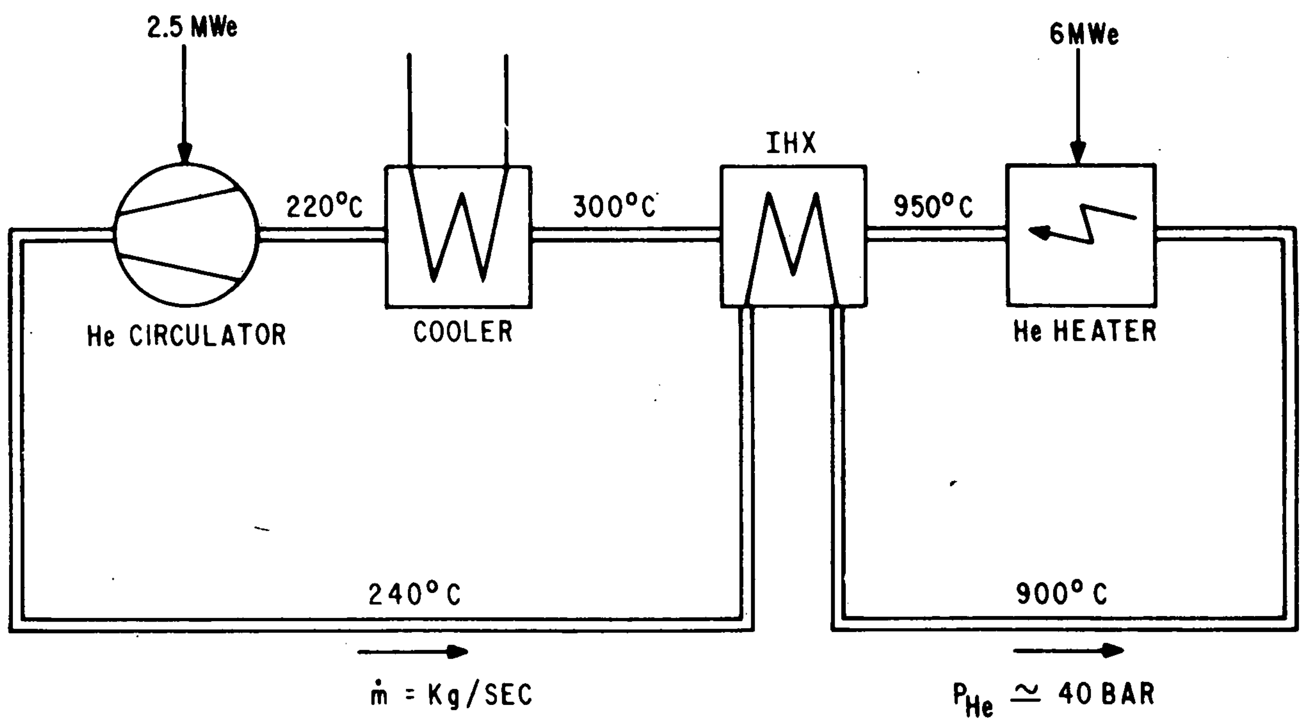
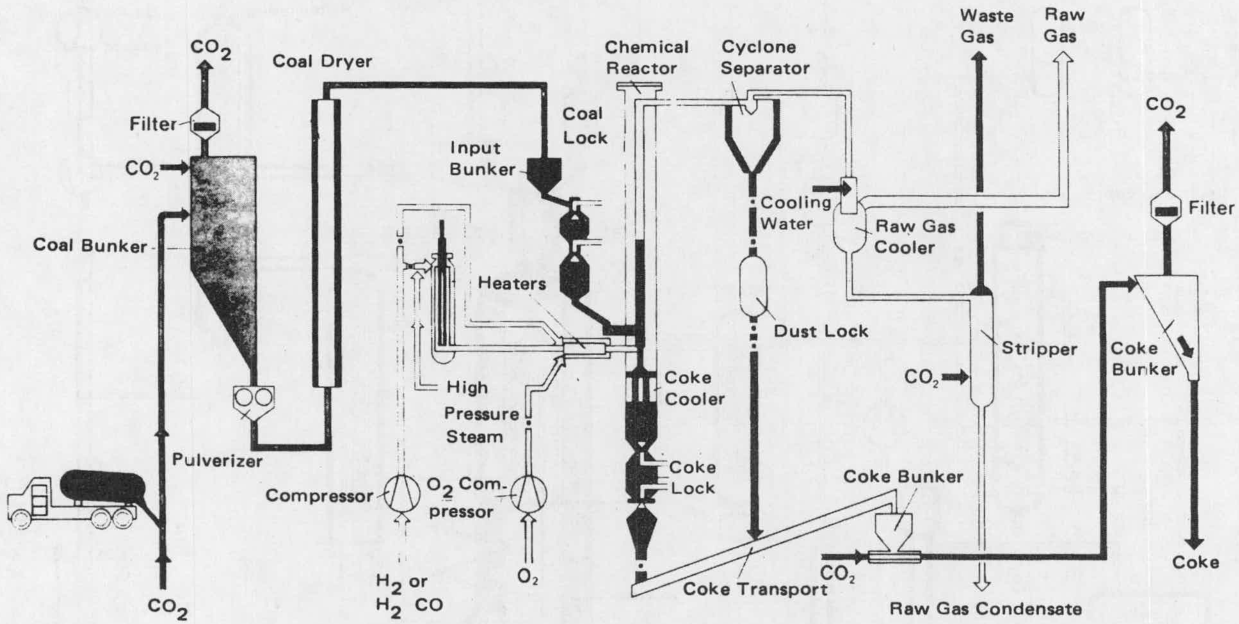


Figure 2-22. IHX Test Facility



PNP

Figure 2-23. HKV-II Test Facility

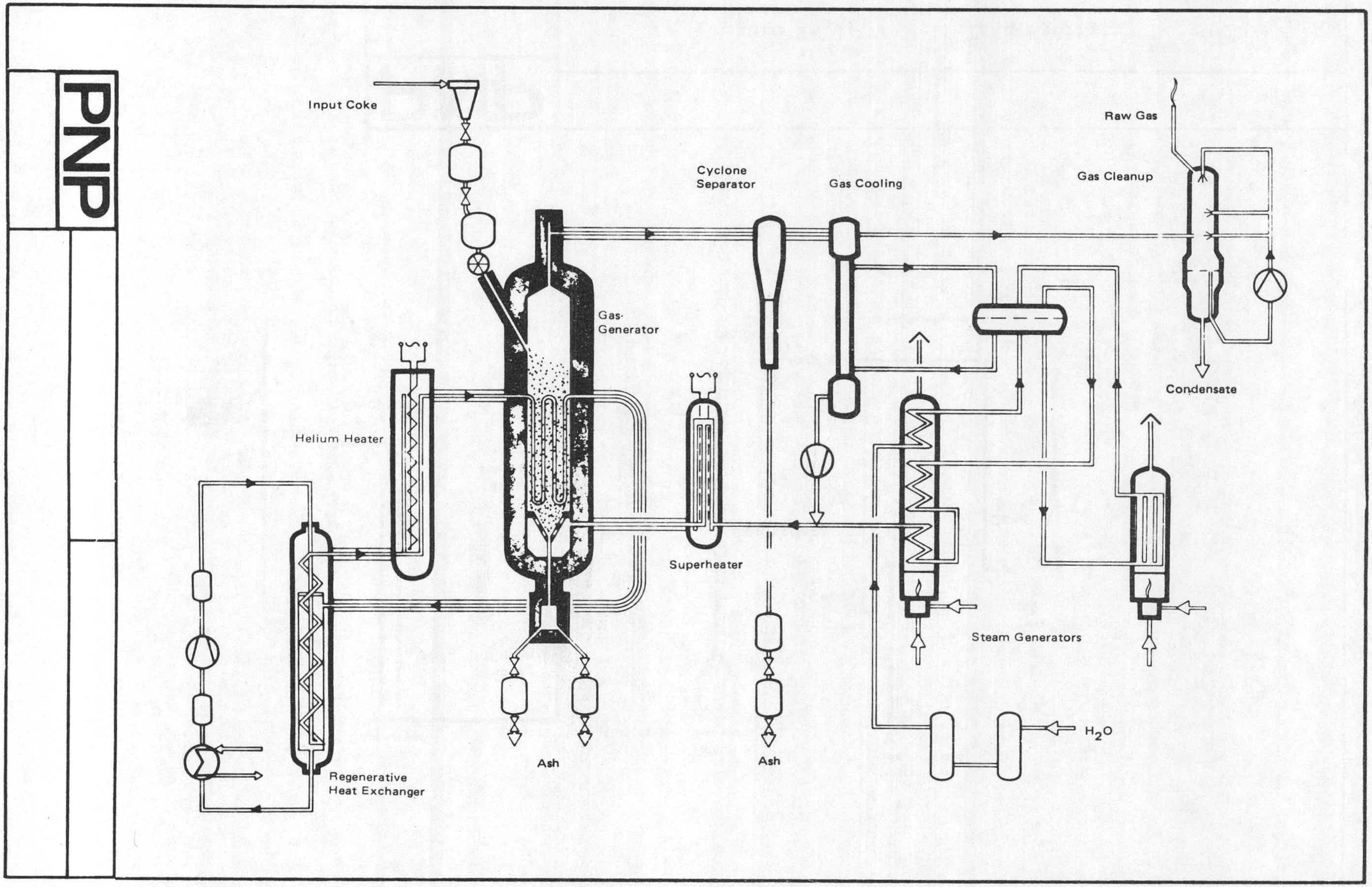


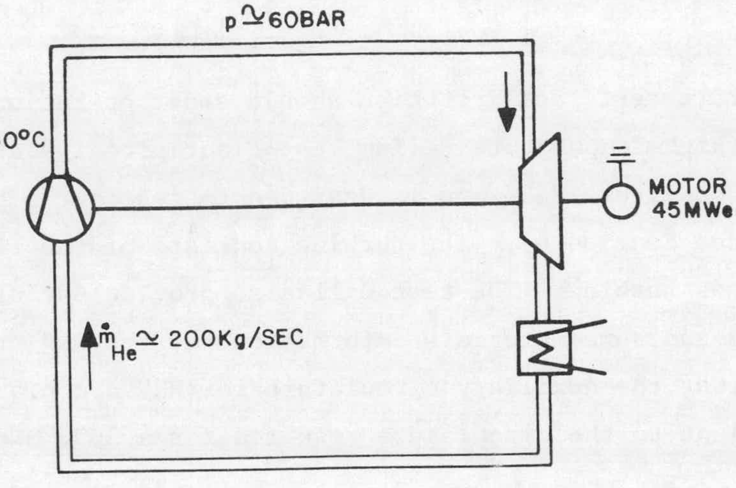
Figure 2-24. WKV-II Test Facility

A separate test facility which should start operation in 1978 is the HHV facility (High-Temperature Helium Test Plant) located at KFA. The test loop is shown on Figure 2-25 and is designed to test out a closed-loop helium gas turbine (see Section 6). The turbine consists of the first two stages of a 300 MWe gas turbine. The test will also provide useful high-temperature component data and some materials information for the PNP project. It should be mentioned that the auxiliary circulators for HHV are 6.5 MW machines and nearly equivalent to the circulators required for a 3000 MWth PNP plant.

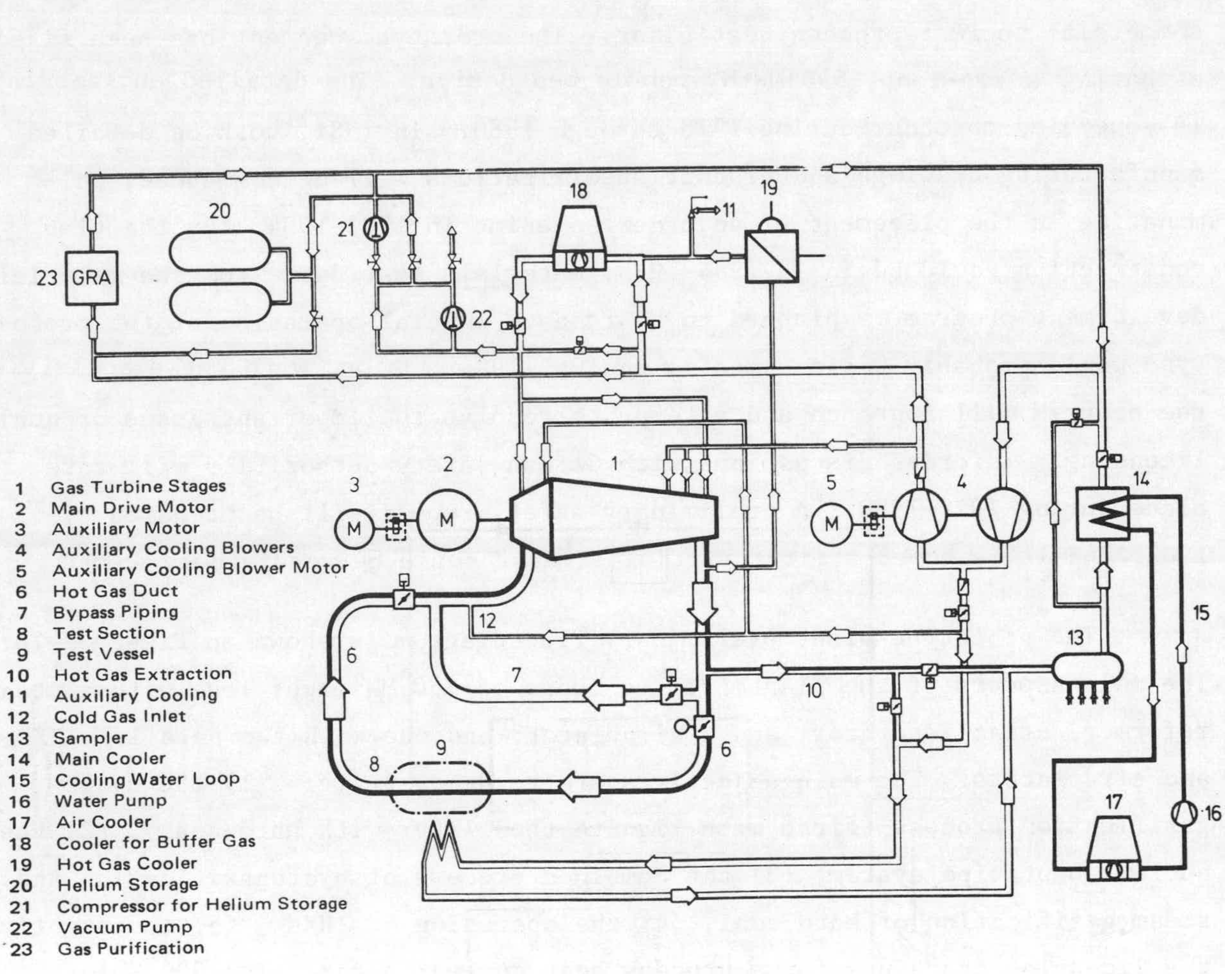
2.3.5.2 PNP Prototype Plant

The planned PNP prototype will be a key stage in the development of a commercial nuclear process heat plant. The prototype concept has been fixed as having a two-loop, 500 MWth, pebble bed design. The detailed engineering is scheduled to occur during 1978 through 1980. In 1981, work on detailed manufacturing drawings and product specifications will be commenced, culminating in the placement of an order sometime in 1983-1984. By the time construction could begin in 1984-1985, materials test data from the materials development program is planned to be ready. Initial operation of the prototype would probably begin sometime in 1993-1994. Along with these activities, the program will approach and attempt to resolve the important issue of nuclear licensing. Informal discussions with German safety authorities will take place during 1978-1980. A preliminary safety report will be prepared by 1982, and it is hoped a first partial license could be received by 1984.

The prototype plant preliminary flow diagram is shown on Figure 2-26. The main aspects of the plant are two loops of 250 MW each: one with a steam reformer, steam generator, and a circulator; and one with two parallel IHXs and circulators. The main objectives are to demonstrate: (1) the hydrogasification process (first with lignite, then later with hard coal), (2) the nuclear heat pipe system, (3) the combined process of hydrogasification and steam gasification of hard coal, (4) the operation of IHXs, (5) to establish the licensing procedure for a process heat nuclear plant. The 500 MW of nuclear heat is divided into 250 MW for the combined process loop and 250 MW for the reformer loop (50% to nuclear heat pipe and 50% to hydrogasification).

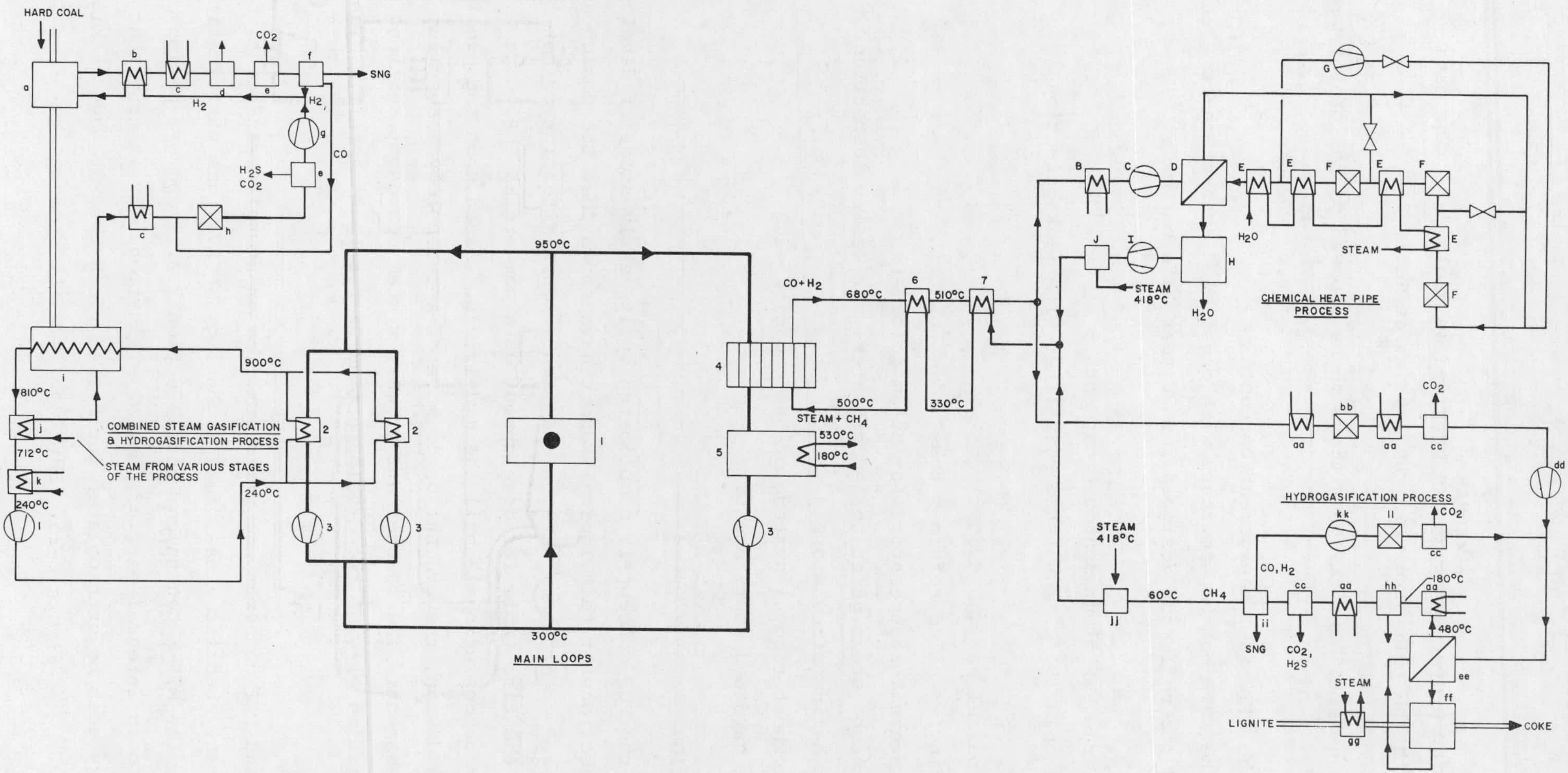


a. Simplified HHV Flow Schematic



b. HHV Flow Diagram

Figure 2-25. HHV Test Facility



MAIN LOOP COMPONENTS

1. Pebble Bed Reactor
2. Intermediate Heat Exchanger
3. Main Loop Circulators
4. Steam Reformer
5. Main Loop Steam Generator
6. Steam Reformer Internal Recuperator
7. Steam Reformer External Recuperator

COMBINED GASIFICATION COMPONENTS

a. Hydrogasification
b. H_2 Recuperator
c. Cooler
d. Water Wash
e. Gas Purification
f. Low Temperature Separation
g. H_2 Compressor
h. Shift Conversion
i. Steam Gasification
j. Steam Superheater
k. Steam Generator
l. Intermediate Loop Circulator

CHEMICAL HEAT PIPE COMPONENTS

A. Recuperator
B. Steam Generator
C. Reformer Gas Compressor
D. Recuperator
E. Steam Generator
F. Methanation Stage
G. Recycle Compressor
H. H_2O Separator
I. CH_4 Compressor
J. Steam-Methane Mixer

HYDROGASIFICATION PROCESS COMPONENTS

aa. Cooler
bb. Shift Converter
cc. Gas Purification
dd. H_2 Compressor
ee. Recuperator
ff. Hydrogasification
gg. Coal Drying Plant
hh. Water Wash
ii. Low Temperature Separator
jj. Steam-Methane Mixer
kk. CO Compressor
ll. Shift Converter

Figure 2-26. PNP Prototype Plant Schematic

The main data for the prototype plant are provided on Table 2-7. The data can be compared with the full size plant by looking at Table 2-2.

Some of the general boundary conditions to be imposed on the prototype plant include:

- Use of the OTTO fuel management concept
- Use of the THTR bottom structure with one fuel discharge chute
- Side and top reflector designed for 30 year life but made removable
- Use of large PNP plant shutdown concept
- Use of a cold (50°C) insulated PCRV liner with leak detection system
- Use of coaxial hot gas ducts
- Use of single-walled reformer tubes
- Use of internal recuperators for steam reformers
- Straight tube steam generator superheaters for easier inspection
- Electric helium circulators
- Four afterheat removal systems of 50% capacity each
- Aircraft hardened containment.

2.3.6 PNP EVALUATION

Evaluating the PNP concept is complicated by its early stage of development. The other sections of this Topical Report have shown that the design has not yet been narrowed to a true "Reference Design". This situation is not unexpected, given the time table for development of a commercial PNP. The prototype will not be operational until the mid-1990's. Based upon the preliminary nature of the design, the evaluation is more of a technology assessment that a system assessment. The reactor and primary system component sections provide a more detailed view of the development needs for those specific plant aspects.

The HTR-K and PNP concepts are different from an overall standpoint, but quite similar in many specific areas (fuel technology, PCRVs, gas purification, etc.). Therefore, the HTR-K technology base (Section 2.3.1) and evaluation (Section 2.3.3) are an integral part of the PNP evaluation. It is true that the base technology must be extrapolated further to reach the PNP, due to higher

TABLE 2-7

PROTOTYPE PLANT CHARACTERISTIC DATA

NUCLEAR REACTOR

Power of reactor	500 MW
Production loops	2x250 MW
Auxiliary cooling loops	4
Helium temperature rise	$300^{o}C \rightarrow 950^{o}C$
Helium pressure	40 bar
Helium mass flow	148 kg/sec
Core power density	$5.5\ MW/m^3$
Active core height	5.5 m
Core diameter	4.75 m
Pressure drop in core	0.5 bar
Number of fuel elements	~530,000
PCRV core cavity diameter	9.75 m
PCRV cavity height	14.3 m
PCRV outer diameter	26.5 m
PCRV outer height	29.3 m
Diameter of pods for steam-reformer, steam generator, IHX	~4 m
Diameter of pods for auxiliary cooling loops	~2 m
Containment building inner diameter	38 m
Containment building inner height	66.3 m
Containment wall thickness	1.5 m
Fuel element	THTR type
Particle concept	BISO
Heavy metal content	11.24 g/fuel element
Burnup	100,000 MWd/t HM
Mean power/ball	~1 kW/ball
Maximum coated particle temperature	$\sim 1030^{o}C$
Maximum surface temperature	$\sim 990^{o}C$
Maximum dose of balls	$\sim 4.8 \times 10^{21}\ n/cm^2$ (E>0.1 MeV)

TABLE 2-7 (Cont'd.)

HELIUM-PRIMARY CIRCUIT

Helium outlet temperature (reactor)	950°C
Helium inlet temperature (reactor)	300°C
Helium pressure (core outlet)	40 bar
Pressure drop in primary circuit	1.5 bar
Helium mass flow	148 kg/sec
Power of the reformer loop	250 MW
Helium inlet temperature (steam reformer)	950°C
Helium outlet temperature (steam reformer)	700°C
Helium inlet temperature (steam generator)	700°C
Helium outlet temperature (steam generator)	300°C
Power of the IHX system	250 MW
Helium inlet temperature IHX (primary side)	950°C
Helium outlet temperature IHX (primary side)	300°C

DATA ON HYDROGASIFICATION OF LIGNITE

Power input	125 MW
Input lignite	161 t/h
Production of SNG	25550 Nm^3/h
Production of residual coke	19.8 t/h
Electricity demand	27 MW
Steam for reformer (50%)	19.35 kg/sec
Steam for drying coal	10.5 kg/sec

DATA ON NUCLEAR HEAT PIPE SYSTEM

Power input	125 MW
Electricity production	-
Reformer gas input to methanation (wet)	23.8 kg/sec
Steam production in methanation (540°C/115 bar)	12.8 kg/sec
Methane recovery in methanation	4.45 kg/sec
Methanation power output	48 MW

DATA OF COMBINED GASIFICATION

Power of the loop	250 MW
Coal input	64.8 t/h
SNG-output	46100 Nm^3/h
Electricity demand	55 MW
Process steam demand	101 t/h
Conversion in hydrogasification (relative to input	54%
Conversion in steam gasification (relative to input)	41%

temperatures and the linking of a nuclear plant to a chemical plant. Extrapolation of the THTR fuel is summarized in Table 2-8. The AVR data at 950°C is very important and, based upon AVR and THTR data, the fuel extrapolation appears sound.

The PNP concepts were not evaluated specifically with respect to chemical processes, due to work scope limitations. However, a general review indicates that the processes selected by the German program (Nuclear heat pipe, steam gasification, hydrogasification, and combined steam and hydrogasification) are technically feasible and that the development plan includes appropriate work to prove the processes.

The higher temperatures of PNP, although generally acceptable from a fuel standpoint, raise an entire spectrum of development requirements. The most wide spread area of difficulty is that of materials. The materials difficulties manifest themselves in the design of almost all the hot (600°C) components, particularly hot gas ducts, steam reformers, He/He heat exchangers. The materials development program addresses these questions although some concern exists that the number of candidate alloys is insufficient to allow for backups which may be required if serious problems are found in the present candidate alloys. Graphite development is included in the FRG fuel program and the more significant ceramic problem areas are development and qualification of new, nearly isotropic graphites for core structure and resolution of the upper side reflector deterioration under irradiation.

PNP primary circuit components appear to be attainable; however, many are characterized by major development needs, e.g., the steam reformers, the steam generators, the He/He heat exchangers, and the hot gas ducts. The semitechnical plants and the prototype plant address these needs. At this preliminary stage, it appears that these components are complex and potentially present significant manufacturing and maintenance problems, however, the testing programs allow for continued engineering assessment and changes as the concepts become more refined.

The new and unique character of a PNP presents an entire spectrum of unresolved licensing and safety issues, with respect to both the FRG and U.S. safety authorities. Some examples include: control and shutdown schemes (screw rods and KLAK), use of only one barrier between the primary circuit and the environment (HKV and nuclear heat pipe), coupling of a nuclear plant

TABLE 2-8

THTR/PNP FUEL COMPARISON

Data	Dimension	THTR 300	PNP 3000
Fuel element	-	ball	ball
Fuel cycle	-	Thorium (93% U235)	Thorium (93% U235)
Ball flow	-	6 passes thru core	1 pass thru core (OTTO Cycle)
Diameter ball		6	6
Heavy metal	-	U/ThO_2	U/ThO_2
Heavy metal content ball	g	11.2	11.2
Diameter coated particle	mm	400	400
Coating	-	BISO	BISO
Mean core power density	MW/m^3	6	5.5
Core helium temperature rise	^{o}C	250-750	300-950
Burnup	MWd/tHM	100000	100000
Max. fuel element surface temp.	^{o}C	950	1020
Max. coated particle temperature	^{o}C	1020	1050
Surface limit temperature	^{o}C	1050	1050
Coated Particle temperature limit	^{o}C	1250	1250
Fast dose (E 0.1MeV)	n/cm^2	6.3×10^{21}	4.5×10^{21}

to a chemical plant generating explosive gases, transient and accident behavior, environmental effects, and inservice inspection of the primary circuit components. The development plan appears to allow scheduled resolution of these issues first through informal evaluations by FRG safety authorities, and then through the detailed licensing procedure for the prototype plant. With respect to U.S. safety criteria, similar evaluation work needs to be performed; however, the licensing issues appear resolvable.[22]

2.3.7 CONCLUSIONS

Both the HTR-K and the PNP appear to be technically attainable, with the former much closer to commercial introduction. The experience base for the pebble bed core is good, and the ongoing PNP program presents an excellent foundation for extrapolation to large size nuclear plants. HTR-K balance of plant does not present any major development problems. The novel PNP components offer developmental challenges, but appear within engineering resolution.

In conclusion, the German programs for development of the PNP address the major problem areas, and they seem to offer an excellent chance for the achievement of a commercial nuclear process heat plant.[22]

2.4 CONFORMANCE WITH GENERAL DESIGN CRITERIA

An evaluation of the German technology against required design criteria must be performed from two distinct perspectives. First, the various systems and components of the HTR-K and PNP must be designed to, and evaluated against, the codes and standards of the Federal Republic of Germany (FRG). Second, the U.S. Government is interested in the ability of the FRG technology to be utilized in this country. Therefore, the various systems and components must also be examined against U.S. Codes and Standards. The report "U.S./FRG Nuclear Licensing Comparison" describes German nuclear licensing procedures and criteria in some detail.[24] The report then compares the U.S. and FRG criteria and assesses the impact of identified differences on an international cooperative program. The report "Safety and Licensing Evaluation of the Pebble Bed Gas Cooled Reactor" provides a detailed examination of the ability of the HTR-K and PNP to meet U.S. licensing requirements.[22] These reports should be referred to for details; however, brief summaries are provided below.

2.4.1 CONFORMANCE WITH GERMAN LICENSING CRITERIA

A detailed examination of compliance with FRG rules has not been performed; however, the technical evaluation has been useful in identifying plant design characteristics caused by unique FRG requirements. Several examples are described below. FRG safety criteria specify redundancy and diversity requirements for safety systems that, in practice, have been interpreted to require diversity (i.e. based upon different design concepts) and (N-2) redundancy. The N-2 criterion means that the number of parallel trains in a safety system is reduced by two for accident conditions. The scenario is that one train is out of service for repair and that a second train fails to operate when needed. The consequences of this scenario are safety systems consisting of 4x50% capacity trains (e.g., the PNP afterheat removal system) or 3x100% capacity trains (e.g., the HHT afterheat removal system). The U.S. regulations only specify N-1 redundancy making the German design excessive for U.S. applications.

A second area of design dictated by FRG licensing criteria involves loadings for structures, systems, and components. The German approach is to look at the various possible accidents and then impose the worst case upon the plant design. In practice this means that the design depressurization accident loads would not be imposed simultaneously with seismic loadings. In the U.S., on the other hand, the plant would be designed to withstand the combined loading. The main reason for this difference is the low level of seismic activity in Germany. With respect to aircraft crashes, the opposite is true. The small size and high population density of Germany makes the country vulnerable in this respect. Therefore, all nuclear plants in Germany are designed to withstand aircraft impact. In the U.S., only the few nuclear plants near airports are aircraft-hardened (e.g., Three Mile Island in Pennsylvania)

As a final example of how the German safety criteria affect plant design, consider the reactivity requirements of control rods. The control rod designs under consideration are based upon two rod failures. Licensing criteria specify that the worst case control rod be assumed to fail upon a demand signal. Additionally, the German utilities require the plant to be operable with one failed rod; therefore, the plant must be capable of shutdown with the two worst case control rods stuck out.

The above examples indicate that compliance with FRG criteria has been an integral portion of the design process. Reference No. 24 discusses the lack of criteria specific to gas cooled reactors. The HTR-K design has the benefit of THTR licensing experience and some of that experience can be extended to the PNP and HHT concepts. There are no precedents for either a nuclear process heat plant or a nuclear gas turbine plant which will force regulatory judgement to be utilized in lieu of criteria, until appropriate regulations are developed.

It is, therefore, concluded that the HTR-K is in general conformance with existing FRG criteria as extended to HTRs by the THTR experience. Some problems can be anticipated if a specific licensing proceding is begun, but not of sufficient severity to prevent commercialization. The PNP and HHT designs appear to be based on an assumed safety criterion in those areas where a vacuum exists in present criteria. It is important to note that early participation by regulatory authorities is planned as part of the German Safety Development Program.

2.4.2 CONFORMANCE WITH U.S. LICENSING CRITERIA

The report, "Safety and Licensing Evaluation of German Pebble Bed Reactor Concepts" is a detailed assessment of the safety and licensing aspects of the HTR-K, PNP, and HHT concepts with respect to U.S. licensing requirements.(22) As discussed in another report, the development of gas cooled reactor criteria in the U.S. is quite extensive compared to the situation in Germany.(24) LWR criteria is, of course, far more detailed than for gas cooled reactors. However, it generally appeared that significant development work (See Section 2.3) would be needed to qualify the designs to comply with U.S. requirements. The situation with regard to HHT and PNP is similar to the U.S. since specific criteria for such designs do not exist. The creation of criteria for advanced gas cooled reactors should, therefore, be an important ingredient in any national or international development program.

SECTION 3

NUCLEAR REACTOR (DESCRIPTION AND EVALUATION)

In this section, the reactor proper is described. This includes the core, fuel elements, and control system, but excludes the PCRV, liner, and ducts. Although there are differences at present between the HTR-K and PNP reactors in the areas of the core support, control philosophy, and the like, it is clearly the German goal to have a single reference core design which can be used for HTR-K, HHT, and PNP. To date this goal has not been completely achieved, partly because work on the HTR-K has been deferred in favor of the HHT and partly because the complete German study is not finished.

3.1 REACTOR GENERAL DESCRIPTION

Figures 2-1 and 2-2 provide vertical and horizontal views of the HTR-K primary circuit and are representative of the general pebble bed core layout.

Each reactor is enclosed in a prestressed concrete pressure vessel (PCRV) which serves as the primary containment structure. A central cavity holds the reactor core structure. Heat transfer equipment, consisting of helium-to-helium heat exchangers, steam reformers, steam generators, and after-heat removal heat exchangers, are located in multiple cavities surrounding the central core cavity. The primary helium coolant is circulated through the core, heat transfer equipment, and connecting ducts by means of electrically driven circulators.

The reactor core structure consists of graphite blocks and metallic thermal shields located in the central cavity. Within this graphite holder rest the spherical fuel elements. The graphite structure serves both as a neutron reflector and as part of the insulation. The lower portion is porous to let the hot helium pass out from the core. Control rod drives are mounted above the core so that the rods can be moved through the suspended upper reflector, through void regions above the fuel, and into the fuel bed itself.

The fuel elements themselves are 60 mm-diameter graphite spheres. They fill the core cavity to an average height of 5.5 m. After they are fed into the core cavity from the top by means of a fuel handling system, they drop to the surface of the fuel bed, pass slowly downwards through the core, and are eventually removed by means of unloading chutes in the core and PCRV bottom.

The coolant flow is downwards, in the same direction as the ball flow. Because of this flow direction and because the fuel elements are only passed through the core once, slowly, this arrangement is called the OTTO cycle, (Once-Through-Then-Out). A distinctive axial thermal profile, discussed in Section 3.2.2, results.

The fuel elements, described in detail in Section 3.5, are graphite spheres, 60 mm in diameter, containing coated fuel particles of mixed thoria-urania. They are identical with those specified for the THTR and proven in the AVR.

TABLE 3-1

PEBBLE BED REACTOR
GENERAL DESIGN PARAMETERS

Parameter	Units	Reactor Type		
		HTR-K	HHT	PNP
Power Level, Thermal	MW	3000		
Power Density	MW/m^3	5.5		
Inlet Gas Temperature Exit Gas Temperature	°C °C	260 700	457 850	300 950
Pressure Level, Maximum	bar	60	72	40
Mass Flow	kg/s	1320	1508	890
Number of Fuel Elements	-	3×10^6		
Thruput of Fuel Elements	balls/d	2654		
Number of Fuel Inlet Tubes	-	43		
Number of Discharge Tubes	-	6		
Core Height	m	5.5		
Core Diameter	m	11.2		

Table 3-1 shows a comparison of some key design parameters of the HTR-K, HHT, and PNP reactors.

3.2 REACTOR REQUIREMENTS AND PERFORMANCE

The nuclear requirements and the expected performance of pebble bed reactors are discussed in the following subsections.

3.2.1 REACTIVITY REQUIREMENTS

Based on the German regulatory requirements, the safety and control systems are divided into two separate systems. For this reason, the reactivity requirements are also divided into two groups. The first must be able to halt all transients which may occur and bring the reactor to a subcritical, hot condition and hold it there for a period long enough for further decisions. The second must be able to handle power control (maneuvering) and also bring to and hold the reactor in a longtime cold subcritical condition. The details and reactivity worths of the system which perform these functions are described in Section 3.4, while the reactivity requirements are discussed in this section.

Table 3-2 shows the primary reactivity effects and requirements for the first and second shutdown systems in a large pebble bed reactor (the HTR-K is used as an example). The maximum assumed accident is water ingress, for instance from a failed steam generator. This condition is assumed to allow 1000 kg of water to enter the core. Analysis showed a 0.5% Δk increase, which is conservatively doubled to give the 1% Δk shown in Table 3-2.

Two temperature effects occur in a short time after shutdown. One is caused by the change in the temperature distribution as the core thermal power drops from 3000 MW to the hot standby level. The other is caused by overall cooling of the core, about 200^{o}C, in the first half hour. The net effect is that the first shutdown system must be able to control approximately 2.8% Δk.

The second shutdown system has two functions: first, to allow for normal changes in power level and distribution (primarily xenon override and control of spatial power oscillations) during maneuvering; and second, to hold the core subcritical for long periods in the cold shutdown condition.

The xenon override requirement is different for the HTR-K and the PNP. For use on an electric grid, the HTR-K must be able to follow a load change from 100% power to 25% power and back to 100%. This requires an Xenon override of 3.5% Δk. The maneuvering requirements in total give a total power control requirement of about 4.4% Δk.

TABLE 3-2

HTR-K REACTIVITY REQUIREMENTS

Parameter Description	Reactivity (% Δk)
First Shutdown System	
Maximum Accident - Water Ingress	1.0
Temperature Equalization - Full Power to Hot Standby	0.5
Cooling Effects - ∿ 0.5 Hours	1.0
Subtotal	2.5
10% Uncertainty	0.3
Total Reactivity to be Controlled by First Shutdown System	2.8
Second Shutdown System	
Excess Reactivity for Load Following (Xenon Override; 100/25/100% power)	3.5
5% Uncertainty	0.2
Base Reactivity Reserve (For Transient Initiation)	0.2
Power Distribution Control (For Control of Xenon Oscillations)	0.5
Loading Uncertainties Equilibrium (2% for First Core)	0
Subtotal for Power Control	4.4
Temperature Defect	3.5
Decay of Xe-135	3.8
Decay of Pa-233 → U-233	4.9
Subtotal	12.2
10% Uncertainty	1.2
Long-Term Effects	13.4
Total Reactivity to be Controlled by Second Shutdown System	17.8

Long-term reactivity effects listed in Tables 3-2 include the overall temperature defect to a cold condition, the decay of Xe-135, and the decay of Pa-233 to U-233. This gives a total long-term effect of about 13.4% Δk. The net effect for the second shutdown system is, therefore, about 17.8% Δk for these large pebble bed reactors.

For the PNP reactor the requirements for the first shutdown systems are nearly the same.

The net requirement for the second shutdown system in PNP is nearly the same as that for HTR-K. There is a change in the distribution of the reactivity values because the PNP core requires more reactivity for the long-term shutdown due to the higher gas outlet temperature (950°C instead of 750°C) and because the excess reactivity in the PNP core is reduced due to the 100/40/100% part load requirement in PNP instead of 100/25/100% part load requirement of HTR-K. These two differences nearly compensate for each other.

3.2.2 THERMAL PERFORMANCE

In this Section, the unique features of the thermal performance during reactor operation are described. Performance during after-heat removal is described in Section 5.3.

3.2.2.1 Basic OTTO Cycle Performance*

The preferred mode of operation for the pebble bed reactor, esepcially for process heat applications, is the OTTO cycle. In this cycle fresh fuel elements are introduced at the top of the core, flow through the core, and are removed at the bottom for long-term storage or reprocessing. Typically, the desired burnup is reached after about three years in the core. This cycle is in contrast to that used in the AVR and planned for the THTR in which the fuel elements are circulated fairly rapidly, checked for damage and burn-up at each discharge, and then reloaded. The helium coolant flows in the same direction as the fuel elements and reaches its maximum temperature at the core exit.

Figure 3-1 shows the results of this scheme for the PNP conditions. In the axial direction, the fissile content of the fuel elements decreases

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*Paraphrased from Reference (3) page 33 ff.

from top to bottom. The heat flux and power density have a corresponding distribution, as shown by curve 4 in Figure 3-1. Thus, the highest power density occurs at the top of the core, where the coolant is at its lowest temperature. The power density of the fuel elements is very low at the bottom of the core. The temperature differences between the fuel element and the gas are large at the top of the core and very low at the bottom

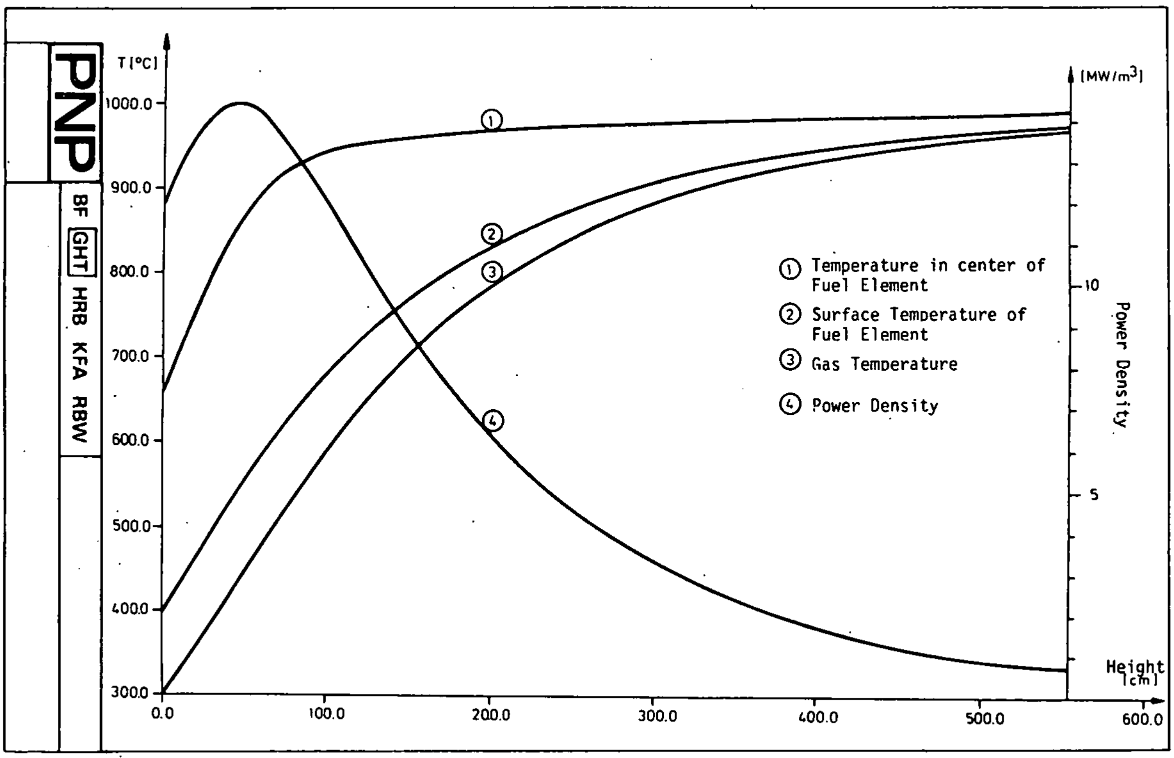

Figure 3-1. Core Axial Temperature and Power Density Profiles (Center Axis)

of the core, typically of the order of 30°C to 50°C between the center of the element and the gas temperature. As an example, the PNP design shows a maximum ball center temperature of 1012°C with a maximum gas exit temperature of 976°C (average exit gas temperature is 950°C). It appears that, for a given maximum fuel particle temperature, the OTTO cycle permits the highest exit gas temperatures; this is in comparison with recirculating schemes such as the AVR and THTR use, and with a fixed fuel system, such as the HTGR.

Figure 3-2 shows typical radial temperature profiles in the pebble bed reactor. In this particular reactor, the fuel elements in the outer 1.0 m of the core are designed with a slightly greater fuel loading than those in the central region, thus tending to flatten the radial power (and temperature) distribution. As can be seen, the range of gas temperatures at the core exit is about 50°C.

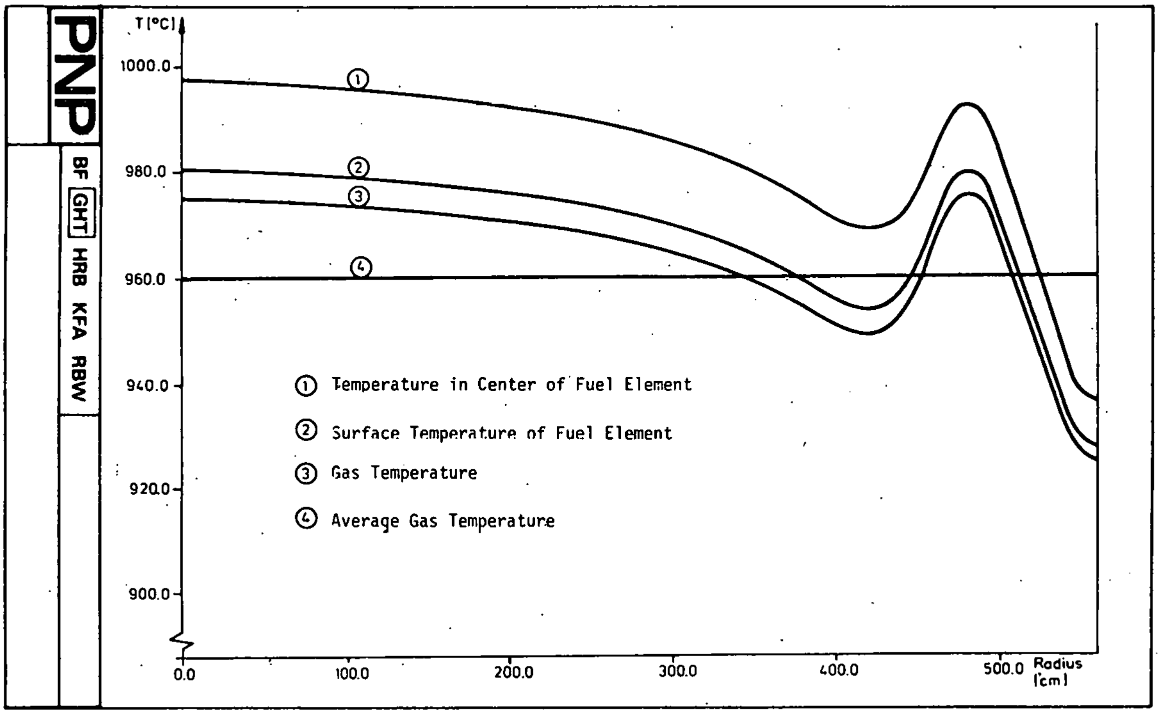

Figure 3-2. Core Outlet Radial Temperature Profile

3.2.2.2 Thermal Performance Uncertainties

It is especially important for the PNP plant to have a very uniform exit gas temperature profile. The metallic materials downstream of the cores are operating near practical strength limits and have not been designed to withstand large overtemperature transients. GHT has performed calculations leading to estimates of the local temperature variations at the core exit, one of the parameters which determine thermal variations seen by the downstream metallic components.

Table 3-3 shows the estimates made for various effects. Some of the more subtle effects are explained below.

TABLE 3-3

UNCERTAINTIES IN LOCAL GAS OUTLET TEMPERATURE

Cause	Estimated Effect, °C
Upper Void Space	0
Variation in Top of Bed (Cones of Pebbles Below Feed Chutes)	+30 ; -15
Nonuniform Flow of Balls (Due to Variable Length of Core Support)	+7 ; -18
Reduced Packing Fraction at Core-Reflector Interface	0 ; -8
Mixing of Two Enrichment Zones	0
Loading Cycle and Effect of Long-Term Shutdown	± 10
Effect of Control Rod Motion in the Bed	+16 ; -30
Estimate of Net Effect, Assuming Control Measures are Used *	± 30*

*This value is not a single summation. It assumes that, as temperature and power variations are detected, control rod motions will be used to limit the variation.

As the balls drop from the fuel inlet tubes, they form cones projecting into the void space above the core. In effect, the top of the core is uneven. Experimental work has shown that, although some radial flow mixing occurs between fuel elements, the axial gas flow tends to follow vertical paths. Thus, the vertical flow paths are of different lengths depending on the locations of the fuel element cones.

In a similar way, the conical bottom support changes the vertical length for gas flow, as well as gives a different residence time for balls at different radial and azimuthal locations. Figure 3-4 of Section 3.2.3.2

shows this effect measured in a 1/6th scale model. Another effect is due to the lower ball packing fractions near the reflector.

When the control rods penetrate the bed during long-term shutdown, their insertion and withdrawal disturbs the bed. Depending on the type of rod, balls are either displaced upwards or forced downwards from their normal location. The effect of this motion depends on the number of operations performed and how often they are repeated. Frequent rod motions would obviously "churn" the top of the bed more than occasional motions.

There are other effects which are not explicitly accounted for in Table 3-3. These include:

- Xenon oscillation (until brought under control)
- Overall variations in ball packing fractions
- Uncertainties in temperature measurements
- Uncertainties in calculations.

It is expected that nuclear instrumentation in the reflector will detect xenon oscillations so that they can be controlled using the control rods. Likewise, measurements of exit gas temperature throughout the core bottom will indicate the extent of control rod action required to control the system.

3.2.3 FUEL ELEMENT FLOW

3.2.3.1 Introduction

Uniform flow of fuel elements through the core is important to achieve an even fuel burn-up and a flat gas exit temperature profile. This is especially true for the PNP plant, which operates at a 950°C exit gas temperature because the heat exchanger components operate near the limits of material performance, and large temperature gradients across the components cannot be withstood.

The reference core bottom design for the 3000 MW core has six exits arranged symmetrically on a circle with a diameter of two-thirds that of the core. Each exit is located at the bottom of an inverted cone which guides the balls to them. The design of these cones is based on the bottom design of the THTR. However, the large, flat core of the 3000 MW reactor cannot achieve sufficiently uniform flow with a single exit chute such as that used in the AVR and THTR; the flow would completely stagnate in some parts of the 3000 MW core if only one exit chute were used. Hence, multiple exit core bottoms are employed.

A backup design has been proposed, and it is described in Section 3.2.3.3, "Backup Core Bottom." The reference and backup designs are discussed from a structural standpoint in Section 3.3.2.3, "Bottom Reflector and Core Support Structure."

3.2.3.2 Reference Core Bottom

The 1:6 scale model which has been used to investigate the flow behavior of the fuel elements in the reference design is shown in Figure 3-3 . A major method used in this investigation was the "Verweilspektren Methode," the method of residence spectra. In this method, test spheres (TS) are distributed in a thin layer over the smoothed surface of the bed to form the "test sphere layer." The test spheres are distinguished from the other spheres by a slightly smaller diameter. After the test layer has been formed, spheres are removed in small intervals from the bottom of the core and other spheres are added to keep the height of the bed constant. The fraction of the test sphere layer (ΔTS) in each interval is measured. The residence spectrum, ε, is then defined by Equation 3-1 , where ΔV is the fraction of the core volume in each interval.

$$\varepsilon = \lim_{\Delta V \to 0} \frac{\Delta TS}{\Delta V} = \frac{dTS}{dV} \qquad (3\text{-}1)$$

The integral of ε with respect to the number of core volumes removed gives the cumulative fraction of the test spheres removed as a function of circulated core volumes (CCV), where CCV is the number of core volumes removed from/added to the pebble bed during the measurement.

The results of three residence spectra experiments for the 1:6 scale model are given in Figure 3-4 , where the integral of ε is plotted against the circulated core volumes. These curves indicate that 90% of the test spheres had residence times between 0.8 and 1.2 circulated core volumes. The curves also indicate that the results were reproducible.

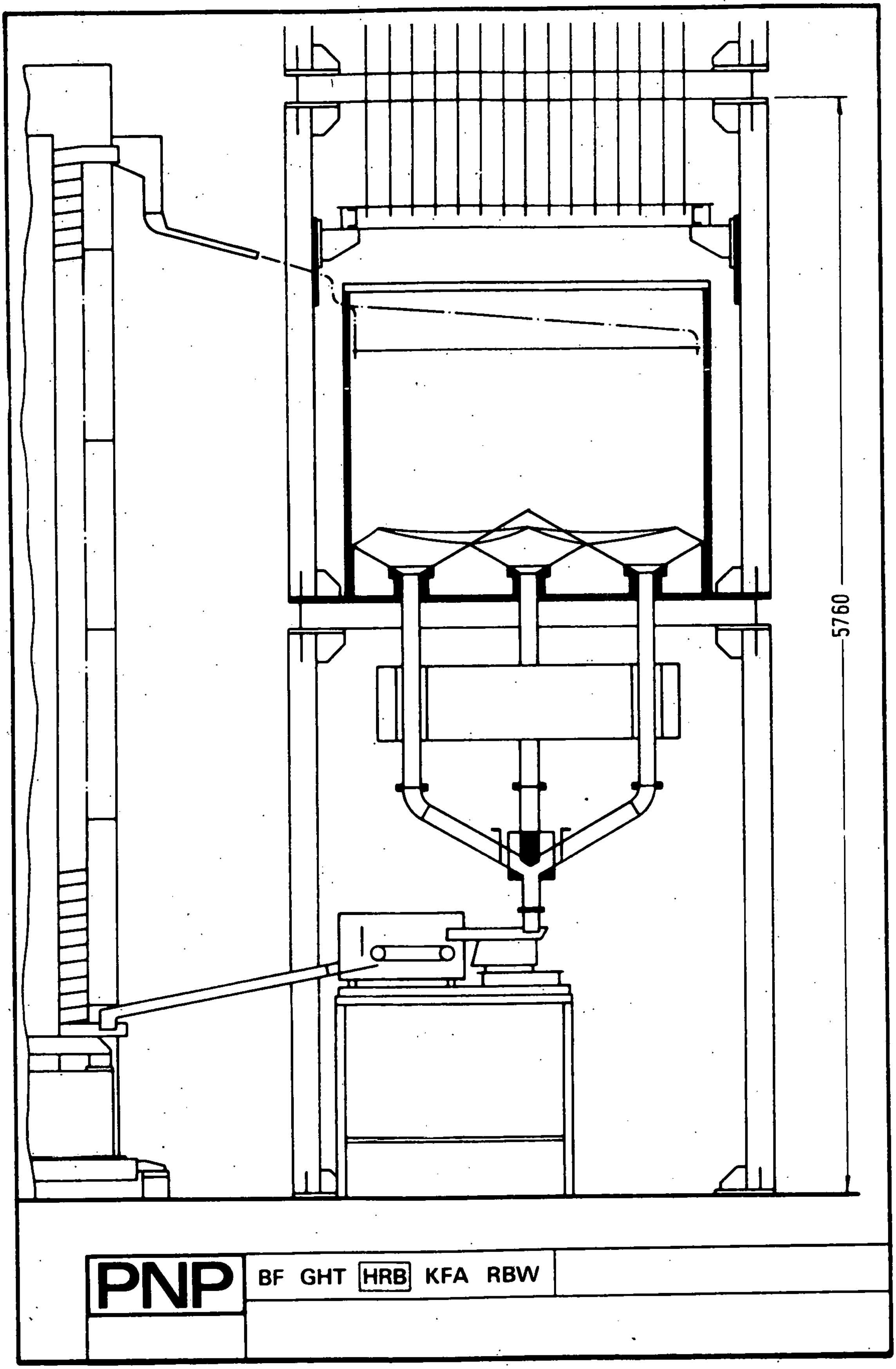

Figure 3-3. PNP-3000 Core Model (1:6 Scale)

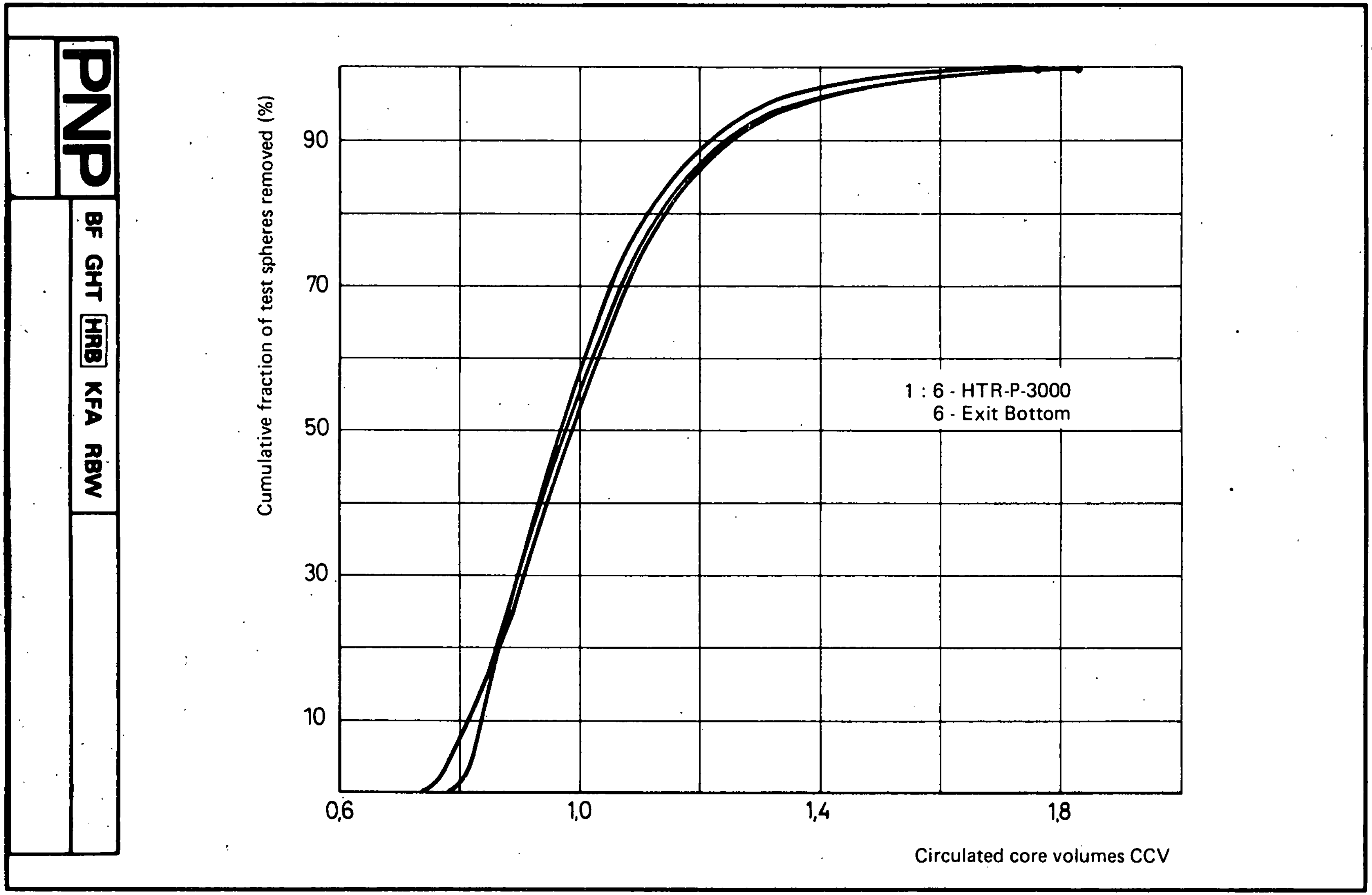

Figure 3-4. Fuel Flow Data (Residence Behavior)

A second test method was used to determine the flow paths of spheres at different radial positions. The detailed mechanics of this method were not available. Figure 3-5 shows the results of this test for nineteen radial positions between the core centerline and the side reflector. The points on the figure represent the movement of the test spheres resulting from circulation of equal portions of the core volume. The data indicates that flow is very uniform to a depth of 70% of the core height. After this, the flow begins to be influenced by the shape of the core bottom; however, this has little effect on gas outlet temperature since only a small fraction of the power is generated at this depth in the core.

3.2.3.3 Backup Core Bottom

The backup core bottom design described in Section 3.3.2.3 has an upright cone located in the center of the core bottom. The cone has a slope of 25°, and its base has a diameter one-half that of the core. The cone rests on a cylinder which projects one meter above the bottom reflector. There are twelve fuel element exits symmetrically spaced around the side of the cylinder on which the cone rests. The core floor extending from the side reflector to the flow cone has a downward slope of 35°.

Some preliminary investigations of this design have been performed in a 1:20 scale model. These tests, however, did not give conclusive results on the uniformity of the fuel element flow.

3.3 CORE INTERNALS

The reactor has a pebble bed core of nearly three million spherical fuel elements contained in a ceramic vessel with an external cast iron thermal shield. The cylindrical bed of fuel elements has a mean height of 5.5 meters and a diameter of 11.2 meters. The fuel elements pass through the core once, moving from top to bottom with an average speed of approximately 5 mm per day. The bed of spherical fuel elements has an average porosity of approximately 0.39 which allows for the downward flow of helium. The cylindrical configuration of the pebble bed is maintained by the core vessel, which is also cylindrical and is assembled from graphite blocks. Around the outside is located the cast-iron thermal shield. In addition to containing the core, the vessel acts as a neutron reflector and thermal shield, and it guides

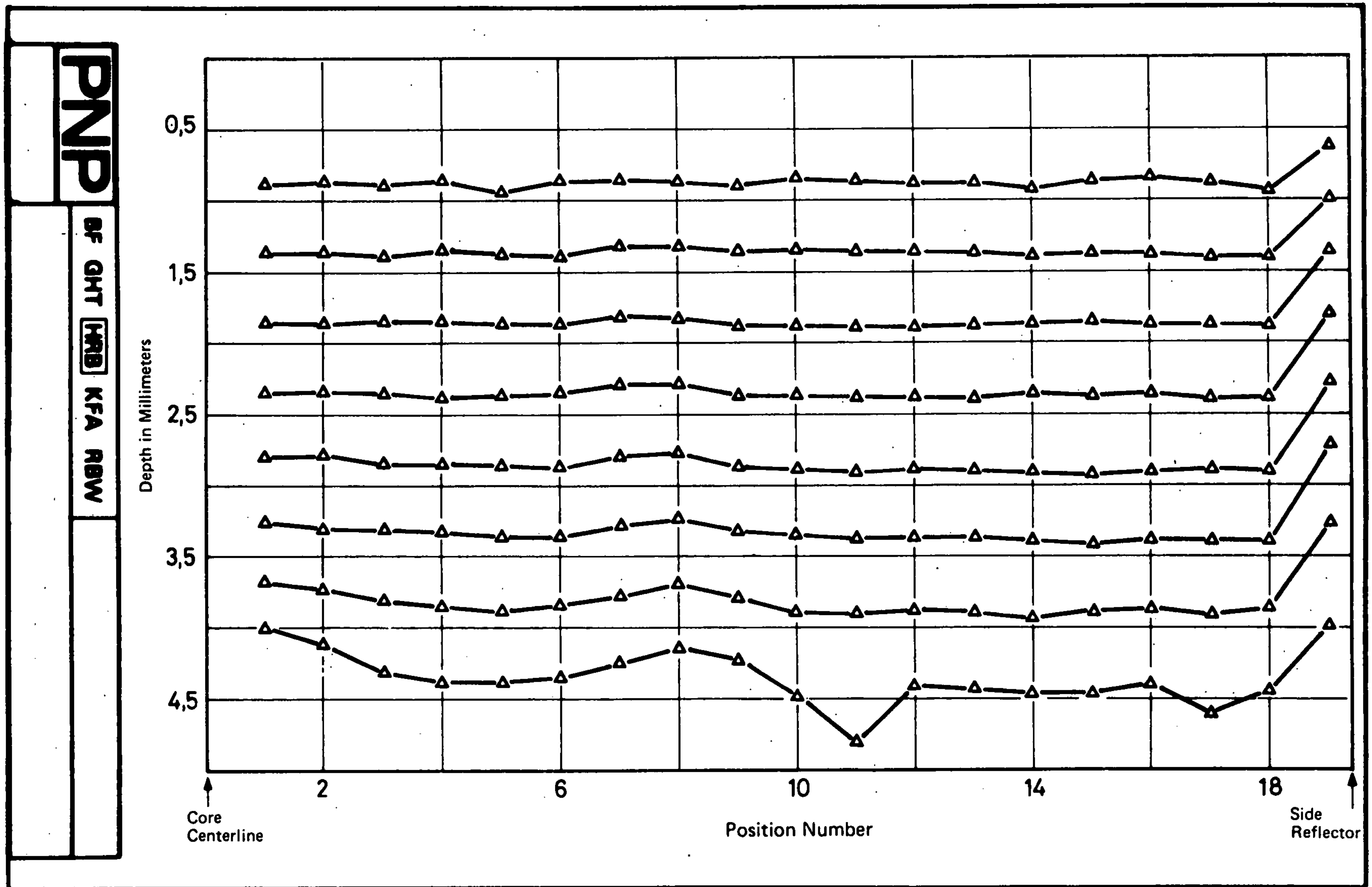

Figure 3-5. Fuel Flow Data (Radial Variance)

the helium flow through the reactor. In both the HTR-K steam cycle electric plant and the PNP process heat plant, the physical arrangement of the core and its container are quite similar. However, the use in the HTR-K plant of separate helium inlet and outlet ducts somewhat alters the gas flow path in the reactor. In addition, the designs of the bottom reflector, core base, and underlying support structure are continuing to receive further study to improve the reference design.

3.3.1 REACTOR GAS FLOW PATH

The primary purpose in passing helium gas through the reactor is to remove heat from the fuel. Virtually all the helium that enters the reactor eventually passes through the core bed, and its course through the reactor is considered to be the primary gas flow path. In some instances secondary flow paths representing a relatively small fraction of the total flow are intentionally diverted from the primary path to achieve specific objectives (such as providing localized cooling of important reactor components).

3.3.1.1 HTR-K Reactor Gas Flow

In the HTR-K reactor arrangement, separate inlet and outlet ducts are used (Figure 3-6). Six horizontal inlet ducts introduce 260°C helium near the top of the reactor. The primary flow stream passes radially inward toward the center of the core moving between the upper thermal shield and reflector. In this region the flow passes around the control rods and turns downward to pass through the upper reflector and the pebble bed, and then into the porous bottom reflector. Here, in the reference bottom reflector design, the helium flows directly downward in vertical passages until it enters a cylindrical exit plenum located in the base of the graphite vessel. In the plenum, the helium no longer flows vertically but rather turns to again flow radially outward, this time passing around vertical core support columns protruding through the plenum. Upon reaching the core circumference, the helium, now at an average temperature of 700°C, passes into the six horizontal exit ducts, thus completing the primary flow path of the reactor.

Secondary flow paths are used to cool the thermal shield and control rods and to limit the temperatures to which the PCRV liners are exposed. In the HTR-K plant, cool helium is allowed to back-flow in the annular space surrounding

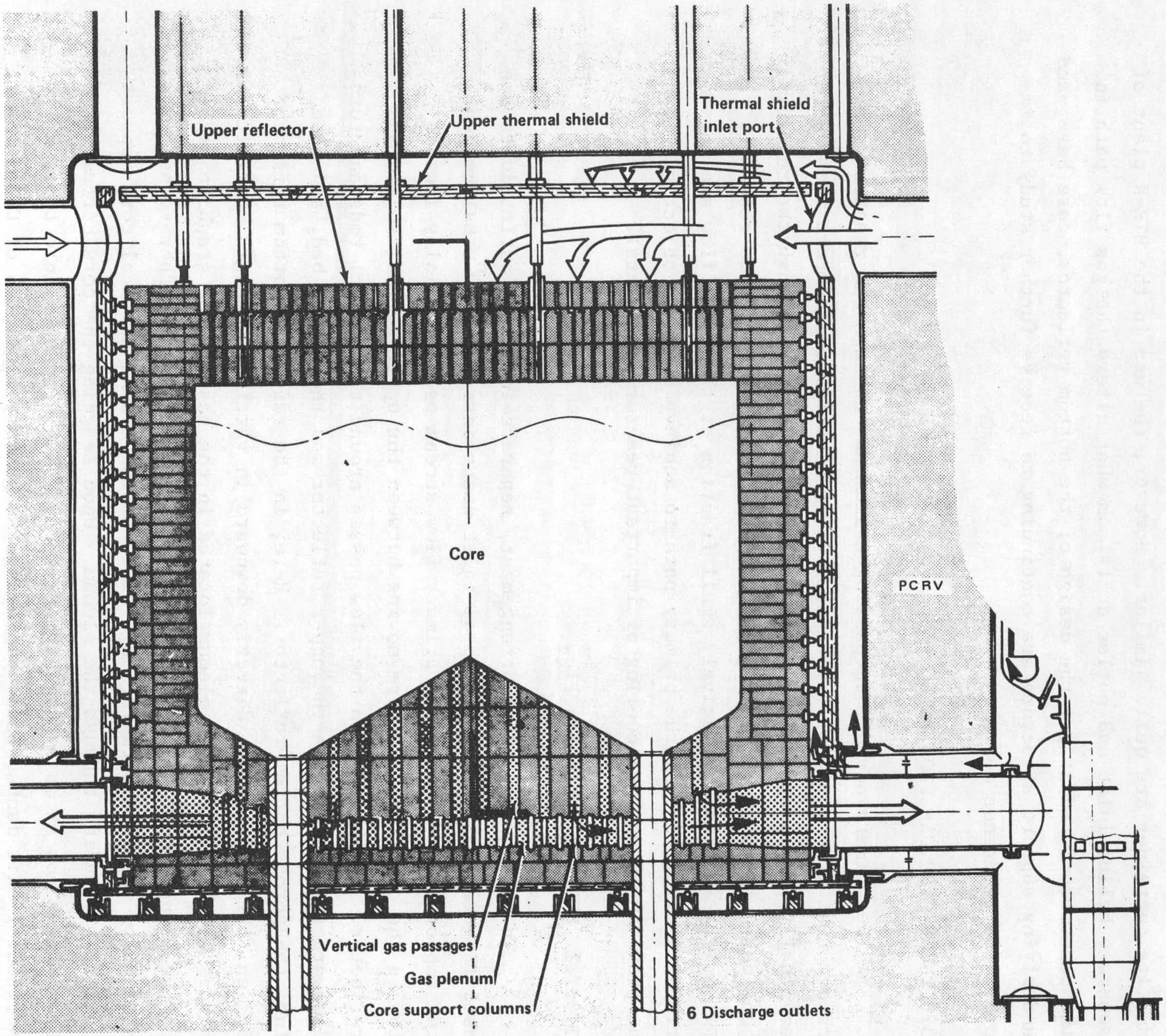

Figure 3-6. HTR-K Reactor Arrangement

the six reactor outlet ducts (Figure 3-6). A fraction of the circulator discharge passes down the inside of the steam generators through the annular duct space to the reactor. Once inside the reactor cavity of the PCRV, this secondary flow cools the thermal shield by passing between it and the PCRV liner and by flowing between the core reflector and thermal shield. In this manner the PCRV liner is exposed only to the temperature of the relatively cool secondary flow stream and not directly to the hot reactor discharge temperature. The magnitude of this secondary flow is controlled by flow restrictors in the annular duct space and by the pressure drop of the primary helium stream in the thermal shield inlet ports near the top of the reactor.

The upper thermal shield is cooled by gas from the inlet ducts which passes between the upper thermal shield and overhead PCRV liner. The control rods are cooled by an internal helium flow path passing along an axial channel running the length of the rod. The helium enters the internal passage through entrance ports located in the spaces between the upper reflector and thermal shield and between the upper thermal shield and overhead PCRV core cavity liner. Once inside each rod, this secondary gas stream flows downward the length of the rod to emerge through exit ports at the rod tip. At this point the secondary stream rejoins the primary flow path in passing through the pebble bed.

3.3.1.2 PNP Reactor Gas Flow

In the PNP reactor design (Figure 3- 7) the objectives in cooling the core and various reactor parts are the same as in the HTR-K plant; however, there are some significant variations in the gas flow paths. Six horizontal coaxial ducts are used for helium flow to and from the reactor in the PNP reactor design. High-temperature (950°C) reactor discharge helium flows inside the central ducts while cool (300°C) helium passes back to the reactor in annular space surrounding the central ducts. Upon entering the reactor, the primary flow stream passes downward a short distance to enter a space below the reactor base. There it passes around the core support columns while it flows radially inward toward the core center. At the center of this gas space the flow turns upward, turns once more, and again flows radially, this time outward, passing along the underside of the reactor core base. The inward and outward bound flow streams are for the most part separated by a thin horizontal diaphram. At the circumference of the graphite core vessel, the flow turns upward, moving in vertical cooling passages in the thermal shield until it emerges above the core in the space between the upper reflector and thermal shield. There, as it turns downward to pass through the upper reflector, it is rejoined by a secondary flow stream which had passed upward in the annular space between the side thermal shield and PCRV liner. This secondary stream entered the annular space directly from the inlet ducts, and its magnitude is restricted to approximately 20 percent of the total flow by restricted flow passages in the upper thermal shield and control rods. From above the core the primary helium flow stream passes downward through the pebble bed, enters the porous bottom reflector, and moves through vertical passages to the outlet plenum in the graphite base of the core, as shown in Figure 3-6. In the plenum the flow is radial and passes into the six exit ducts. This is the flow path of the reference core base design. The core bottom design shown in Figure 3- 7 is a backup design currently undergoing study as a possible alternate to the reference configuration of Figure 3-6.

In the PNP reactor as in the HTR-K design, helium is used to cool the control rods. As in the HTR-K design, helium enters the control rod cooling passages through inlet ports in the area of the upper reflector and thermal shield. It passes downward inside the control rod assemblies and emerges at the rod tips.

3.3.2 CORE CONTAINER

The pebble bed or core of the reactor core is contained in a cylindrical graphite vessel. Since the properties of graphite are altered when it undergoes long-term neutron exposure, the design of the graphite core container must compensate for such effects. For example, Figure 3-8 illustrates typical data for graphite shrinkage as a function of neutron exposure and temperature. Consideration of such changes is particularly important in preparing a stable long-lasting design for the upper region of the side reflector, where the fluence is greatest. At the base of the core the exposure levels are lower; however, the substantial loads associated with supporting the weight of the core, accommodating thermal stresses, and allowing for possible seismic disturbances all require careful consideration of the physical properties of the material. The approach used in designing the

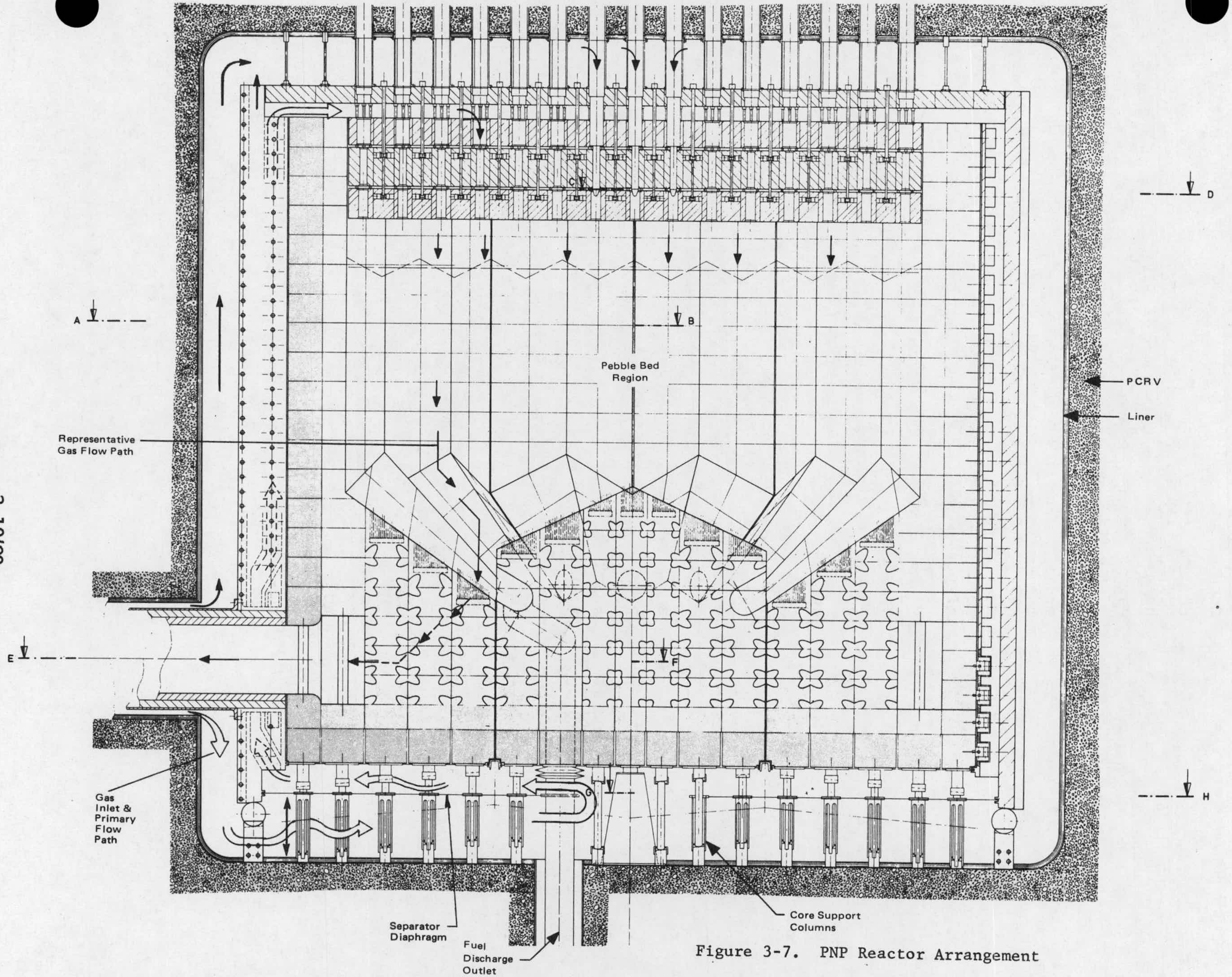

Figure 3-7. PNP Reactor Arrangement

core container is to minimize the use of large, single graphite pieces of complex shape. Rather simpler, smaller blocks are assembled and keyed or fastened together and then the assembly is positioned and held in contact using spring and gravitational forces. The specific features are discussed below.

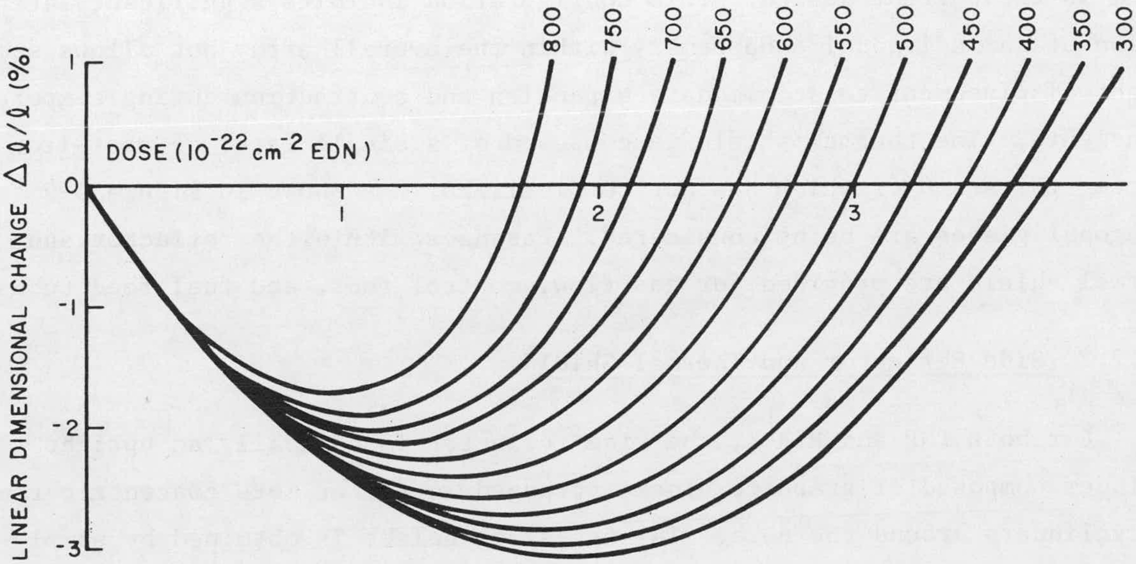

Figure 3-8. Graphite Shrinkage Behavior

3.3.2.1 Upper Reflector and Thermal Shield

For both PNP and HTR-K, the design of the upper reflector and thermal shield and their method of support are similar to those used on the THTR plant under construction in the Federal Republic of Germany. The reflector consists of an array of graphite block subassemblies suspended from overhead as shown in Figures 3-7 and 3-9. In the current design, the individual subassemblies are composed of three graphite blocks stacked on top of each other and connected together with anchoring rods. At the top of each subassembly, anchoring rods extend upward to permit the subassembly to be fastened to the cast-iron thermal shield located above the reflector. In turn, the thermal shield is suspended from the PCRV liner overhead. The length of the rods above the subassemblies is sufficient not only to attach the rod to the thermal shield but also to extend upward through the cooling gas gap between the thermal shield and upper reflector. This gap is needed for helium flow and is obtained by spacer pins that maintain the proper gap when the subassembly anchor bolts are tightened to support the reflector assembly below. The entire upper reflector is composed of an array of these subassemblies positioned side-by-side above the core. As depicted in Figure 3-9 the shape of the graphite blocks is hexagonal in the current design. This configuration inhibits significant lateral motion of an individual subassembly within the overall array but allows some slight displacement to accommodate expansion and contraction during temperature transients. The thermal shield is composed of a single layer of cast-iron pieces, the shape of which has not be finalized. As shown in Figure 3-9 hexagonal pieces are being considered. Passages within the reflector and thermal shield are provided for gas flow, control rods, and fuel feed tubes.

3.3.2.2 Side Reflector and Thermal Shield

For both PNP and HTR-K, the side reflector is basically an upright cylinder composed of graphite blocks arranged in two or more concentric rings or cyclinders around the core. The necessary height is obtained by simply stacking circular rows of blocks on top of one another. Key-ways are cut in the sides of the blocks and keys are inserted to align the blocks. Circumferentially around the outside of the blocks where they contact the cast-iron thermal shield, fasteners are used to connect the graphite blocks to the metallic thermal shield. Toward the bottom of the side reflector in the area adjacent to the core base, springs (located between the thermal shield and outer reflector blocks) are used to apply a radial force inward toward the center of the core. This force pushes the graphite blocks inward so that each block bears against the ones on either side, and the entire ring is constrained to the desired circular shape. The design of the springs is basically similar to that of a belleville washer. During assembly of the reflector, access to the springs is gained through holes in the thermal shield. Once the springs are adjusted or positioned to provide the proper compressive force on the blocks, the access holes are plugged to prevent undesired cross flow of helium between the gas passage in the thermal shield and the gas space between the PCRV liner and thermal shield.

The side thermal shield is constructed of cast iron sections bolted together to form a rigid cylinder. The thicknesses of the sections are 20 cm

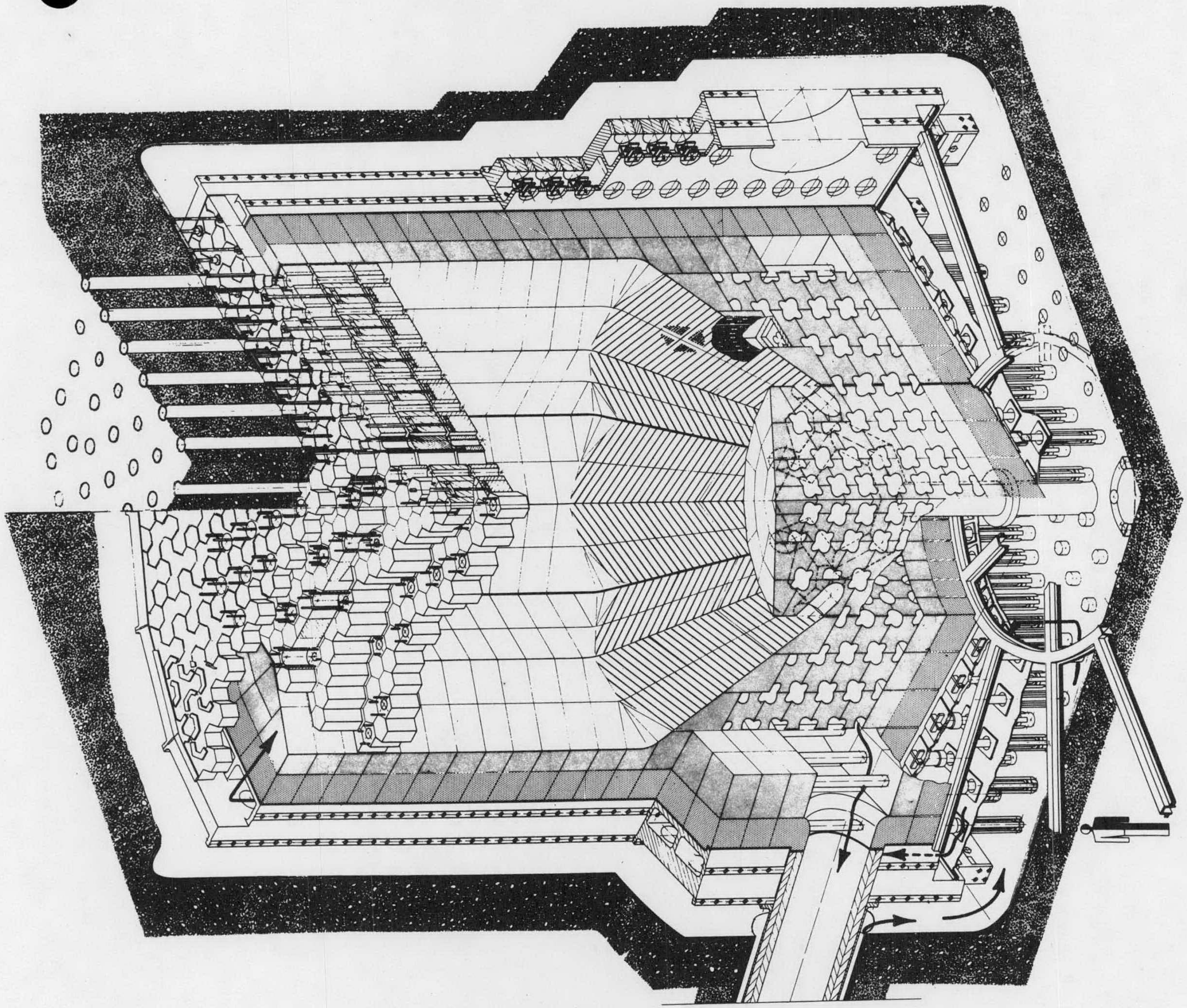

Figure 3-9. PNP Core Structure (Vertical Section)

for the HTR-K and 40 cm (including gas space) for the PNP plant. In general, the method of fabrication is similar to that of the THTR plant. In this arrangement, the shield is built up of annular rings (Figure 3-6), each composed of twelve panels bolted together. The necessary height is obtained by simply stacking and fastening sufficient rings to reach the desired height above the core. In the PNP design, the sections contain internal gas passages to cool the shield (Figure 3-9), while in the HTR-K design the shield has cooling gas flowing on both sides (Figure 3-6). The weight of the shield and, in part, that of the core is borne by circumferential support columns at the base of the core.

When viewed from above, the inner reflector surface is not truly circular. Rather, it appears as a polygon with many sides that, when taken together, approximate a circle. This design is used in the HTR-K plant (Figure 3-10) together with a bottom reflector utilizing six fuel discharge openings. To achieve satisfactory ball flow, the bottom reflector has six conical segments, each centered on a discharge opening and thus effectively funneling the spent fuel elements toward each discharge port. While this arrangement has been designed as the reference design for both the HTR-K and PNP plants, Figures 3-9 and 3-11 depicted a backup arrangement currently undergoing evaluation. In this design, the side reflector is decidedly not smoothly cylindrical but is, rather, shaped more like a star with twenty-four sides (Figure 3-11 sections A-B and C-D). The configuration of the side reflector is directly related to the shape of the core bottom, and both are designed to channel fuel elements toward the discharge openings.

One of the main structural objectives of the reflectors discussed above is to contain the pebble bed core in the desired configuration without impeding the flow of fuel elements downward. However, under certain conditions, the inner surface of the side reflector can actually inhibit fuel flow. This can occur when the fuel balls at a given core elevation tend to move downward in unison with little or no relative motion of one to another. In such instances the fuel balls tend to pack into a horizontal array or lattice pattern that, if large enough, can inhibit the downward movement of all the fuel in the array. Such lattice patterns most frequently start at the walls of the container, since fuel movement there is least apt to be disturbed by other fuel. To preclude the occurrence of such lattice patterns, it is necessary to promote some relative movement of the fuel at the reflector surface. This is accomplished by simply machining grooves on the inner surfaces of the reflector blocks to alter the smooth surface periodically.

During long-term operation of the reactor, radiation-induced damage such as surface cracking of the side reflector may occur. Such damage is most likely to occur in the upper part of the side reflector near the top of the pebble bed, where the use of the OTTO fuel cycle results in the greatest neutron fluence. Upon being irradiated, graphite initially shrinks, then expands (Figure 3-8). In the German analysis, spalling is possible once the initial shrinkage has been recovered and the graphite begins to expand beyond its original size. This corresponds to the zero-point absissca in Figure 3-8 . By plotting the locus of points where $\Delta\ell/\ell$ equals zero, a graph representing the onset of possible spalling can be plotted in terms of temperature and dose. This has been done in Figure 3-12 (curve A). Also plotted in the same figure is the calculated axial dose to the side reflector (curve B), expressed in terms of temperature. This latter curve is obtained simply by plotting the dose and temperature at various axial elevations in the core from axial temperature and dose profiles. Curve B is drawn based on a cumulative dose over the 40-year planned life of the reactor. Since a reactor utilization factor of 0.8 is used, curve B represents an equivalent 32 years of full power operation. For shorter periods of operation, the cumulative dose would be less and curve B would be lower. To find the onset of potential spalling, it is necessary to find the reactor "age" where curves A and B first touch, as the cumulative dose is increased during the life of the reactor. This first occurs at a temperature of about 480^{o}C (this condition is depicted by curve C). As a first approximation, if it is assumed that dose varies linearly with age, then the time for the potential onset of spalling can be determined by reducing 32 years by the ratio of the vertical distances below curves A and B at a temperature of 480^{o}C. From Figure 3-12, it can be seen that (A'-C')/(B'-C') times 32 is equal to .867 times 32 or 27.7 years. Thus, in the German analysis, potential spalling would begin after 27 to 28 effective full-power years of operation. Since the dose decreases rapidly inside the reflector, spalling would be limited to approximately the first five centimeters of the side reflector surface near the top of the pebble bed.

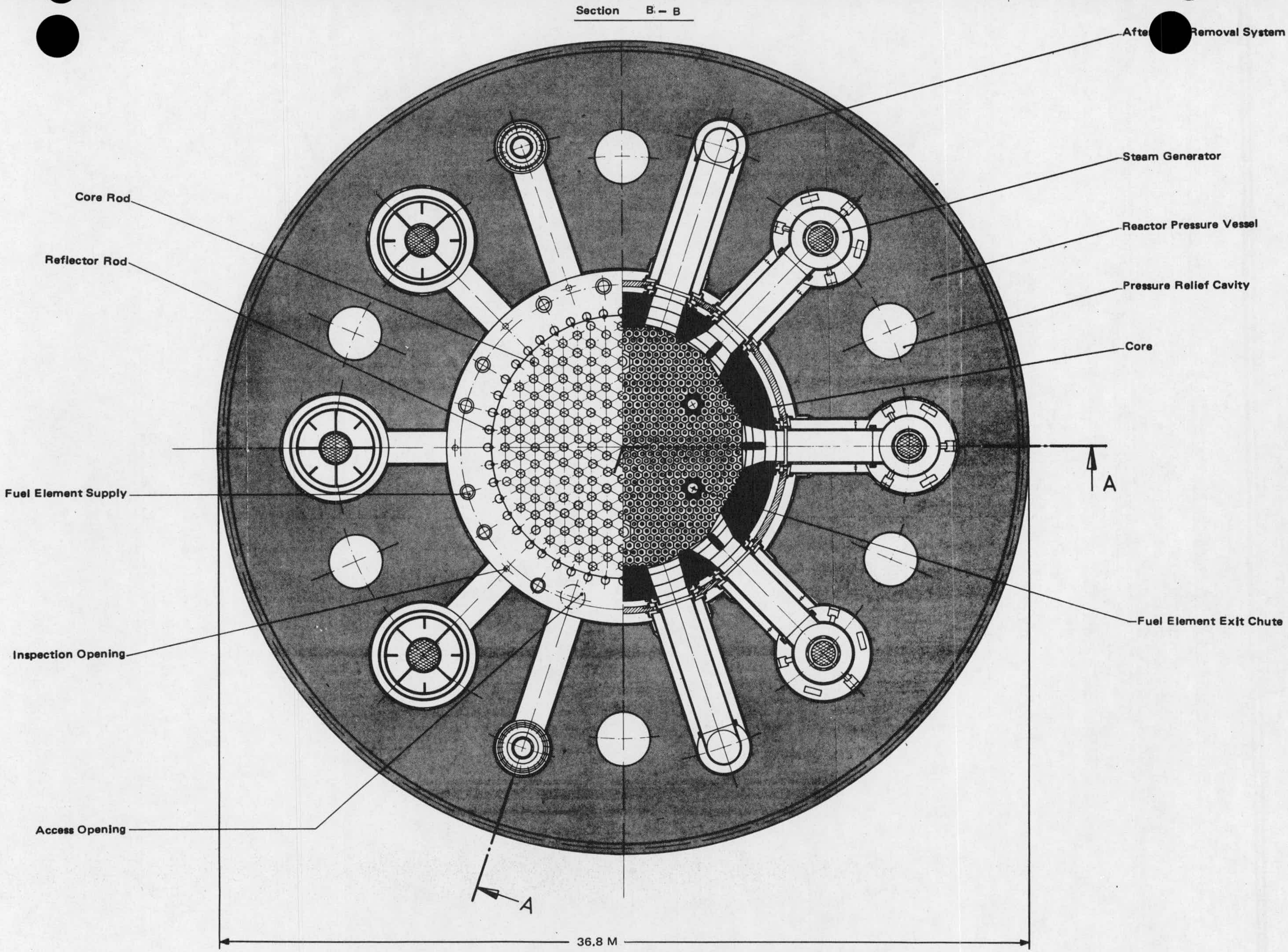

Figure 3-10. HTR-K Core Structure (Plan View)

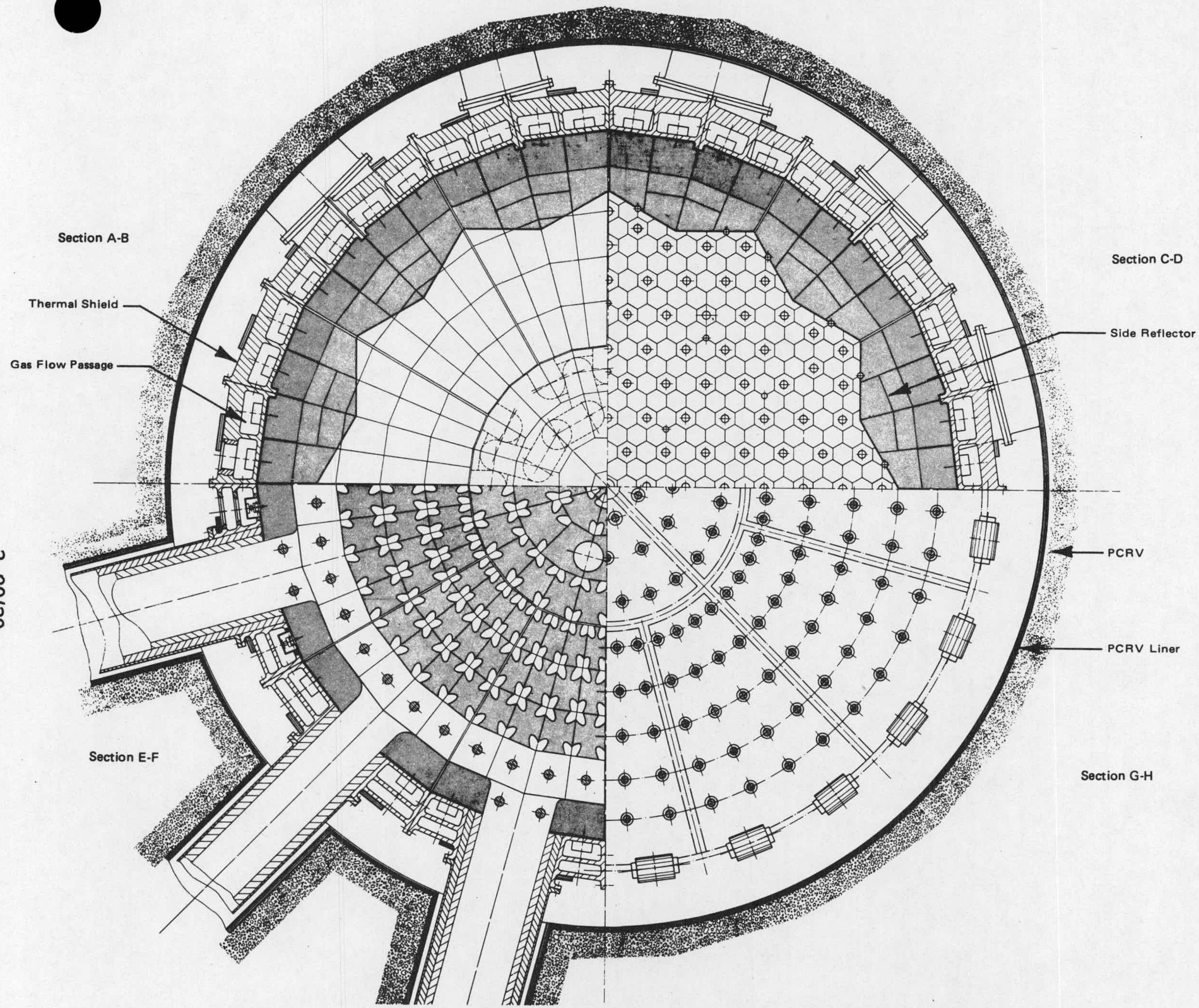

Figure 3-11. PNP Core Structure (Plan View)

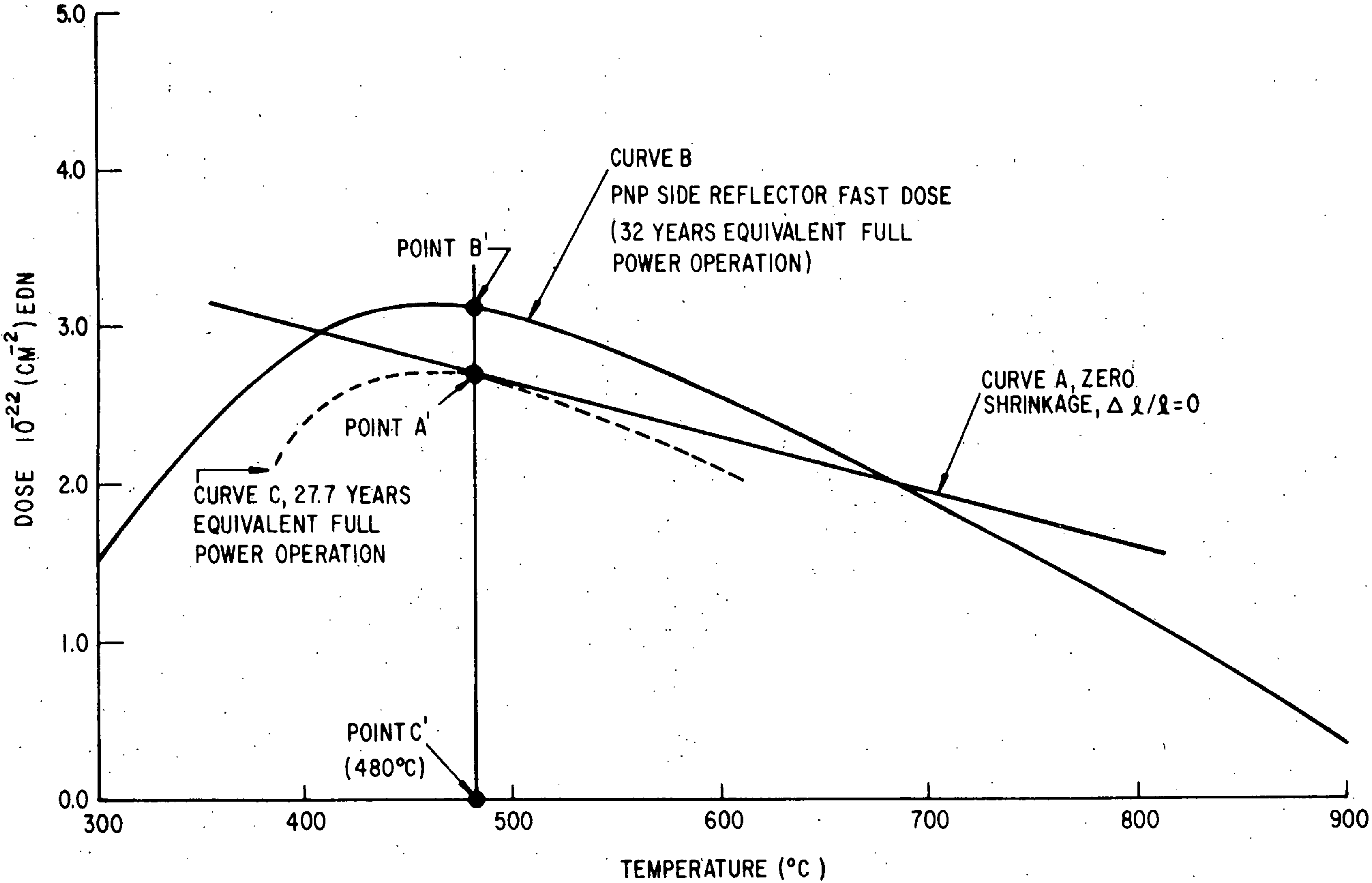

Figure 3-12. Reflector Graphite Dose and Temperature Behavior

Such spalling is considered tolerable in Germany, and the designs of the reactor and its related systems have been prepared to accommodate limited spalling without reducing plant performance or safety. In the final design, maximum cooling of the side thermal shield and reflector will be employed to limit reflector temperatures. If necessary, small gaps will be cut in the surface of the reflector blocks to limit the size of graphite pieces separated from the reflector, and the helium purification system will be sized to remove graphite dust potentially produced by spalling. As an additional contingency measure, the graphite blocks of the inner reflector will be designed so that they are replaceable. Such replacement would require the reactor to be shutdown, cooled-down, and emptied of fuel. In addition, investigations are continuing in Germany regarding the possible use of graphite that is highly isotropic to minimize long-term growth and distortion that in time could potentially lead to spalling. In any case, the current side reflector design is considered to have sufficient thickness--approximately 120 centimeters of which the outer 70 centimeters has 1.5% boron content--to tolerate some limited spalling of the inner surface.

3.3.2.3 Bottom Reflector and Core Support Structure

As indicated previously in Section 3.2.3 the large diameter of the 3000 MWth reactor core requires multiple spent fuel discharge ports to achieve satisfactory fuel element flow to obtain a satisfactory power distribution and, in turn, an acceptable gas temperature profile. Three factors are among those of key importance in achieving these objectives: first, the shape of the bottom reflector surface, and second, the interior construction of the bottom reflector gas flow passages and the gas mixing they achieve. The third factor is the design of the base support structure and how it carries the weight of the core and other transmitted forces.

The bottom reflector and underlying core base is composed of graphite blocks. The reflector itself is made of high-grade relatively isotropic graphite, while for the underlying base a more economical grade is used. The reference bottom reflector design is shown in Figure 3-6 and has six discharge ports located azimuthally 60° apart around the core at a radius equal to two-thirds of the core radius. The reflector configuration in the vicinity of each exit port is conical with the port at the center of the cone. Thus the surface of the bottom reflector approximates that of six intersecting cones.

A backup design is shown in Figure 3-7 and utilizes a central flow cone below which are twelve fuel discharge openings. The twelve spent fuel passages are merged within the base of the flow cone to three discharge pipes that pass downward through the PCRV to the spent fuel facility. In the core, fuel elements are guided toward the exit ports by virtue of the slope of the bottom reflector and are directed to each of the discharge openings by inclined V-shaped channels in the reflector (Figures 3-7 and 3-9).

The cooling gas also passes through the bottom reflector, but nearly all of it flows in passages different from those used by the fuel. Figure 3-13 shows the bottom reflector flow passages used in the reference design. The hexagonal cross section graphite blocks in each of the six conical regions contain numerous vertical flow passages connecting the pebble region of the core to small collection chambers at the base of surface blocks. The openings are of such dimension and shape so as not to be blocked by the fuel elements. Larger vertical gas passages in the next lower layer of blocks that are keyed to the upper blocks connect these individual collection chambers to the large central hot gas exit plenum located in the core base. As shown in Figure 3-13, vertical graphite core support columns pass through the plenum and bear the weight of the core. In passing through this series of flow passages considerable gas mixing takes place in the collection chambers and hot gas exit plenum as well as by virtue of the turbulent flow in the passages themselves.

In the backup bottom reflector design the flow passages in the base of the core are arranged in a different manner, as shown in Figure 3-14. Upon leaving the pebble bed region of the core, the cooling gas passes vertically downward through porous regions of the first layer of graphite blocks to collection chambers. From the individual collection chambers the gas passes through diagonal flow passages cut in each of the underlying graphite blocks. Using these passages, the gas migrates to the annular hot gas exit plenum chamber and then leaves via the six outlet ducts. Within the array of diagonal flow passages the mixing of the cooling gas is considered to be somewhat improved over that of the previously described design.

The weight of the reactor core is borne by the metallic thermal shield along with the underlying core base and is transferred to the PCRV by support columns located below the reactor (Figure 3-6). The columns employ a top-mounted roller assembly that bears against the base of the thermal shield to compensate for thermal expansion. Lateral motion of the entire core assembly is restricted, and the horizontal hot gas ducts are mounted to allow for limited movement and thermal expansion of the ducts relative to the core assembly.

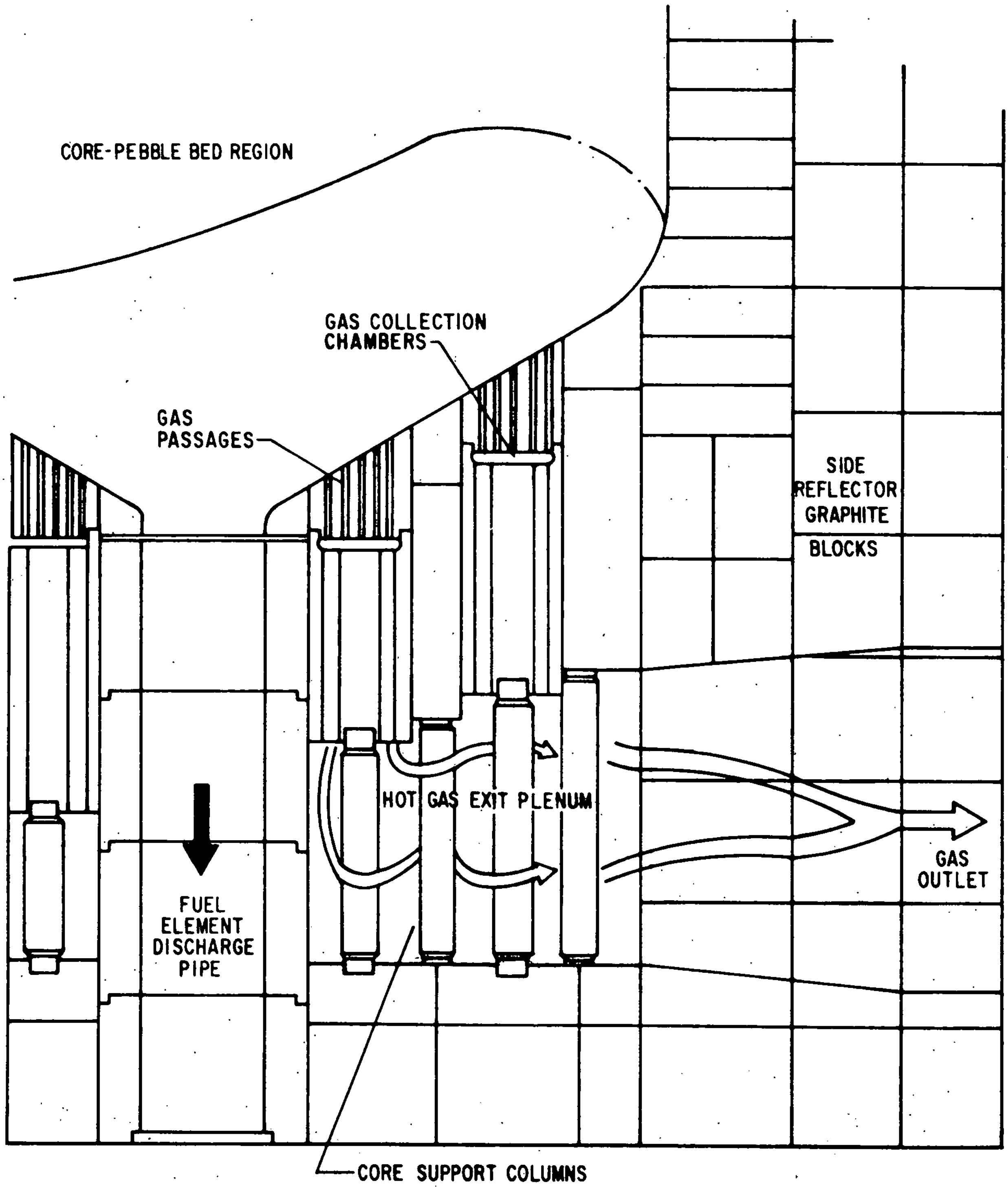

Figure 3-13. Reference Core Bottom Structure

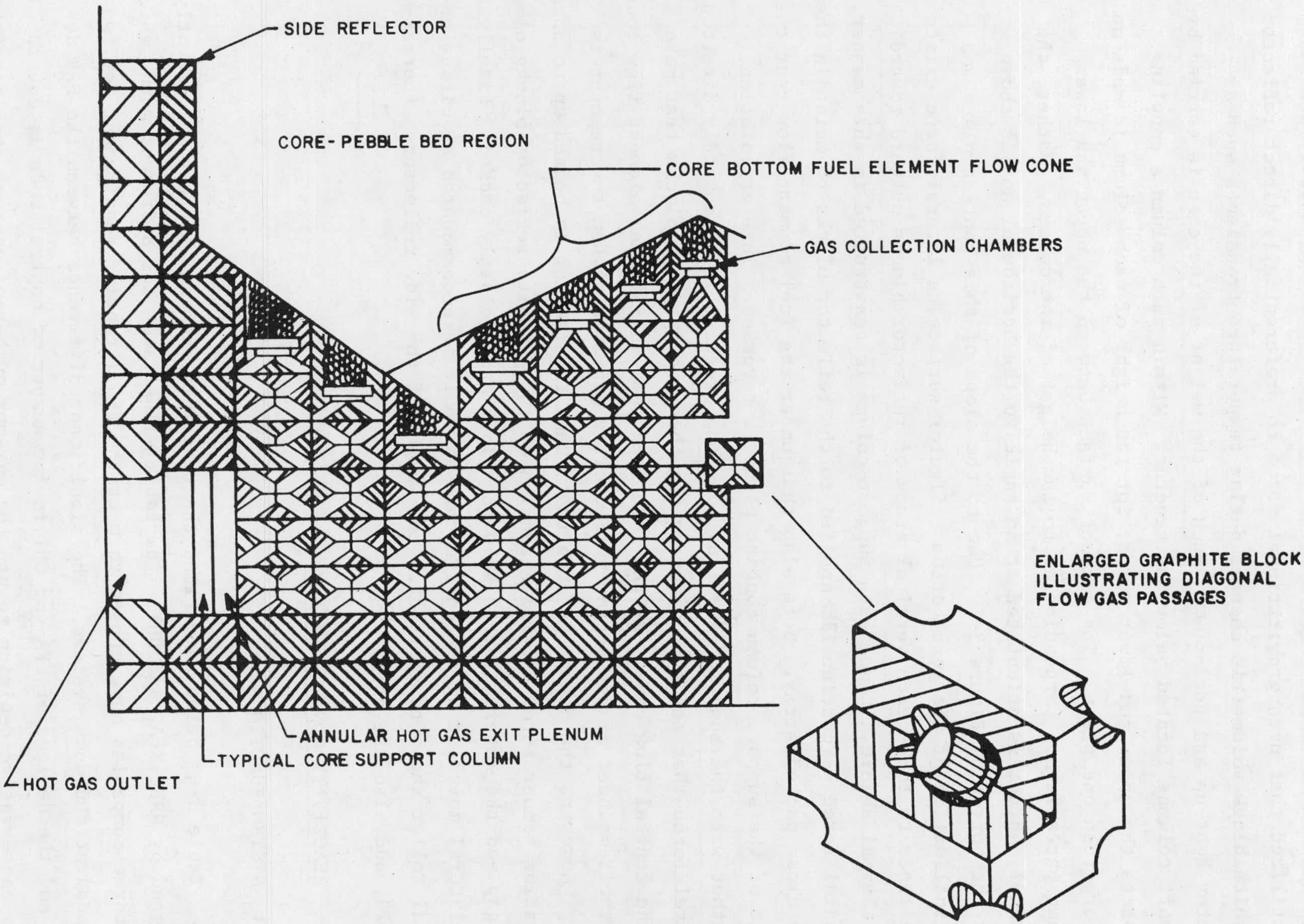

Figure 3-14. Backup Bottom Reflector Design

In the backup core design (Figure 3-7) a novel core support scheme is utilized that uses gravitational force to preferentially direct reflector graphite block volumetric changes during temperature transients such as reactor heat-up and cool-down. Much of the weight of the core is carried by support columns located below the reactor. Within each column a coupling connects the upper and lower parts, for the height of each column is made up of more than one piece. The contact surface between the upper and lower column parts is not perpendicular to the height of the column. Rather, the plane of contact is orientated at an angle to the horizontal and is shown near the bottom of Figure 3-7. Due to the slope of this contact plane and the inclination of the core bottom reflector surface, a lateral force exists that tends to force each vertical stack of reflector blocks outward toward the thermal shield. Since each support column is constructed in this manner, a radial force is continually applied to the reflector blocks to maintain them in a close-packed array. Note also that under the fuel element flow cone the angle of the support column contact plane is reversed. This orientation, together with the shape of the flow cone top, results in an inwardly directed lateral force that tends to compress together the graphite blocks that make up the central flow cone. At the interface between the fuel element flow cone and the remainder of the bottom reflector, a gap exists when the reactor is cold. However, the design is such that, when the reactor is heated up to normal operating temperature, thermal expansion of the graphite reflector blocks closes the gap and the lateral forces keep the block array closely packed. Finally, any lateral movement of the entire core assembly is accommodated by circumferential roller supports that carry the weight of the side reflectors, thermal shield, and also the core.

3.4 REACTIVITY CONTROL SYSTEM

3.4.1 SYSTEM FUNCTIONS

Table 3-4 below summarizes the reactivity control systems as currently designed by HRB, KFA, and GHT. The table illustrates the different means which the three companies have chosen to meet the FRG licensing requirement for two independent shutdown systems. The significant difference between the HRB design and the designs of KFA and GHT in the number of control rods is due to HRB's conservative decision to use the maximum number of control rods permitted by the design limitations on the PCRV head and to have some contingency for the detailed design phase. On the other hand, the KFA and GHT designs use the smallest number of rods which would meet the reactivity requirements for the reactor.

At present, the reference reactivity control system for HTR-K is the HRB design, and for PNP, it is the GHT design, but the systems are exchangeable with some modifications. The HRB and GHT control systems and their components will be discussed in further detail in the remainder of Section 3.4.

TABLE 3-4

CONTROL CONCEPT SUMMARY

	Total System	First Shutdown System	Second Shutdown System
HRB (HTR-K reference design)	198 core rods +48 reflector rods	42 lifting rods (pneumatic drives) +24 reflector rods (gravity/electric drive)	156 lifting rods (hydraulic drives) +24 reflector rods (gravity/electric drive)
KFA	168 core rods	36 lifting rods (pneumatic drives)	132 rotating rods (spindle drives)
GHT (PNP reference design)	156 core rods	156 rotating rods (spindle drives)	156 rods (spindle drives) + KLAK*

3.4.1.1 HTR-K System Functions

Reactivity control for the HTR-K is provided by 198 core control rods and 48 reflector control rods, which are split into two independent systems. The first system contains 42 core rods and 24 reflector rods, and the second system contains the remaining 156 core rods and 24 reflector rods. The core rods are of the lifting type and have either pneumatic drives (first system) or hydraulic drives (second system). All the reflector rods have gravity/electric drives. The drives for these rods are contained in the top head of the PCRV, and they are positioned about the core as illustrated in Figure 3-15.

*The 156 rods and the KLAK are capable of independently shutting down the reactor.

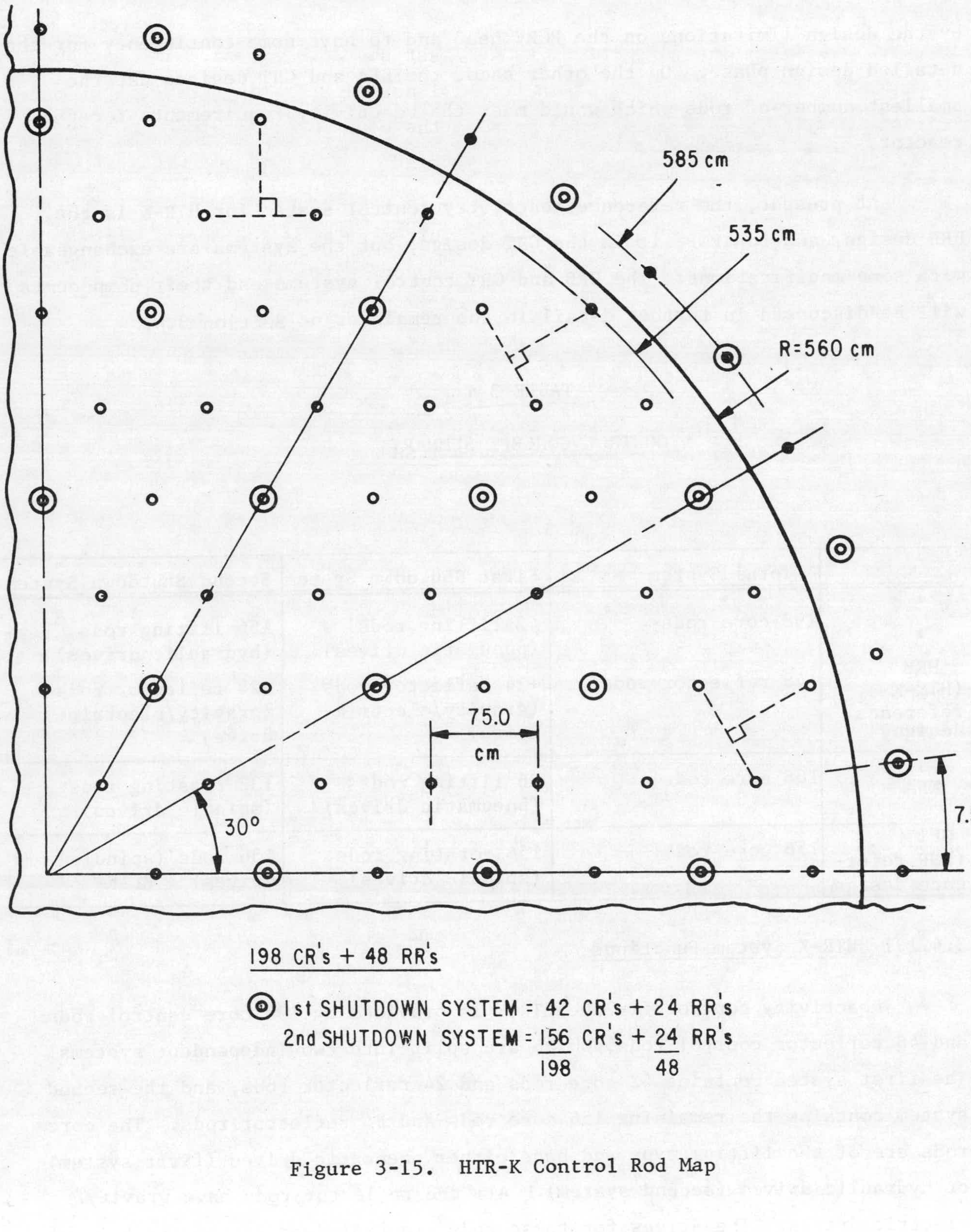

Figure 3-15. HTR-K Control Rod Map

The first system functions as the fast shutdown system. It has the capability to shutdown the reactor from all normal operating or accident conditions. This system is designed to hold the core subcritical for a short period of time, when it is possible to override the negative reactivity introduced by xenon build-up; hence the reactor is capable of a fast restart after being shut down. Table 3-5 shows the requirements and the worth of the system. As can be seen from the table, the contingency of the system is relatively high.

TABLE 3-5

REQUIREMENTS AND WORTH OF THE FIRST SHUTDOWN SYSTEM (HTR-K)

Requirement	Reactivity, %Δk
• Accident (water ingress)	1.0
• Temperature equalization ($T_{fuel} \rightarrow T_{mod}$) (hot standby)	0.5%
• Temperature reduction by cooling the core	1.0
• 10% uncertainty	0.3
• Total demand	2.8
Worth of System (42 Core Rods + 24 Reflector Rods)	
• Insertion to core 4.5 m	8.0
• Reduction of worth by influence of second shutdown system	-0.8
• 10% uncertainty	-0.8
• Loss of 2 most effective rods	-1.2
• Net worth of system	5.2
• Required demand	-2.8
• Minimum shutdown margin	-0.5
• Contingency of the system	1.9

The second system serves as both the power control system and the long-term shutdown system. For power control, this system must provide reactivity for load following over a range of 100-25-100% and for control of the power distribution. For long-term shutdown, it must hold the reactor subcritical at ambient temperature for an indefinite period of time. Hence, it must overcome the complete decay of fission product poisons as well as the core's negative temperature coefficient.

Table 3-6 shows the requirements and the worth of the second shutdown system. As can be seen from this table, the contingency of this system is relatively high. However, some reserve is needed to allow for changes during the detailed engineering of the core.

TABLE 3-6

REQUIREMENTS AND WORTH OF THE SECOND SHUTDOWN SYSTEM (HTR-K)

Requirement	Reactivity, %Δk
• Temperature reduction ($T_{oper} \rightarrow 20^{\circ}C$)	3.5
• Decay of Xe 135	3.8
• Decay of Pa 233	4.9
• 10% uncertainty for long term effects	1.2
• Excess reactivity load following 100/25/100 (including 5% uncertainty)	3.7
• Control of Xenon oscillations	0.5
• Reactivity reserve for initiating transient	0.2
	17.8
Worth of System (156 Core Rods + 24 Reflector Rods)	
• Insertion to core 4.5 m (including 5% uncertainty)	24.9
• Loss of 2 most effective rods	- 3.0
• Worth of system remaining	21.9
• Required demand	-17.8
• Minimum shutdown margin	- 0.5
• Contingency of the system	3.6

The backup design for the HTR-K reactivity control system is the reference design for the PNP, which is described below.

3.4.1.2 PNP System Function

The PNP reactivity control system consists of two completely diverse systems. The first is a set of 156 core control rods; the second uses small, high absorbing spheres. The reference rod for this system is a rotating rod with a spindle drive. As in the HTR-K, the drives are located in the PCRV head, and the rods are distributed as shown in Figure 3-16. This system provides the same three capabilities as the HTR-K system. First, it provides the reactivity to hold the reactor in a hot, subcritical condition for a short period of time. Second, it provides the long-term capability to hold the reactor cold subcritical, and, finally, it provides the reactivity necessary for power control.

The second system using small absorber balls has the acronym KLAK, which comes from the German Kleine Absorber Kugel (which means small absorber balls). The KLAK system introduces about 20% Δk, which is sufficient to maintain the reactor in a cold, subcritical condition. It is used in the PNP design to satisfy the licensing requirement for two independent shutdown systems. In practice, it would only be used in the event of significant failures in the rod system.

The backup design for the first PNP system is an HTR-K type system, which utilizes the KLAK as an emergency backup.

3.4.2 REACTIVITY CONTROL SYSTEM COMPONENTS

3.4.2.1 Absorber Elements

The absorber elements of both the lifting and rotating rods are constructed from two concentric, metal tubes, but the rotating element has a helical profile, while the lifting element is smooth. The design of the rotating rod absorber element is shown in Figure 3-17. Alloys presently being considered for the outer tube are Incoloy 800H and 802 and Inconel 519 and 625.™

B_4C in the form of rings is placed between the concentric tubes and provides the needed neutron absorption.

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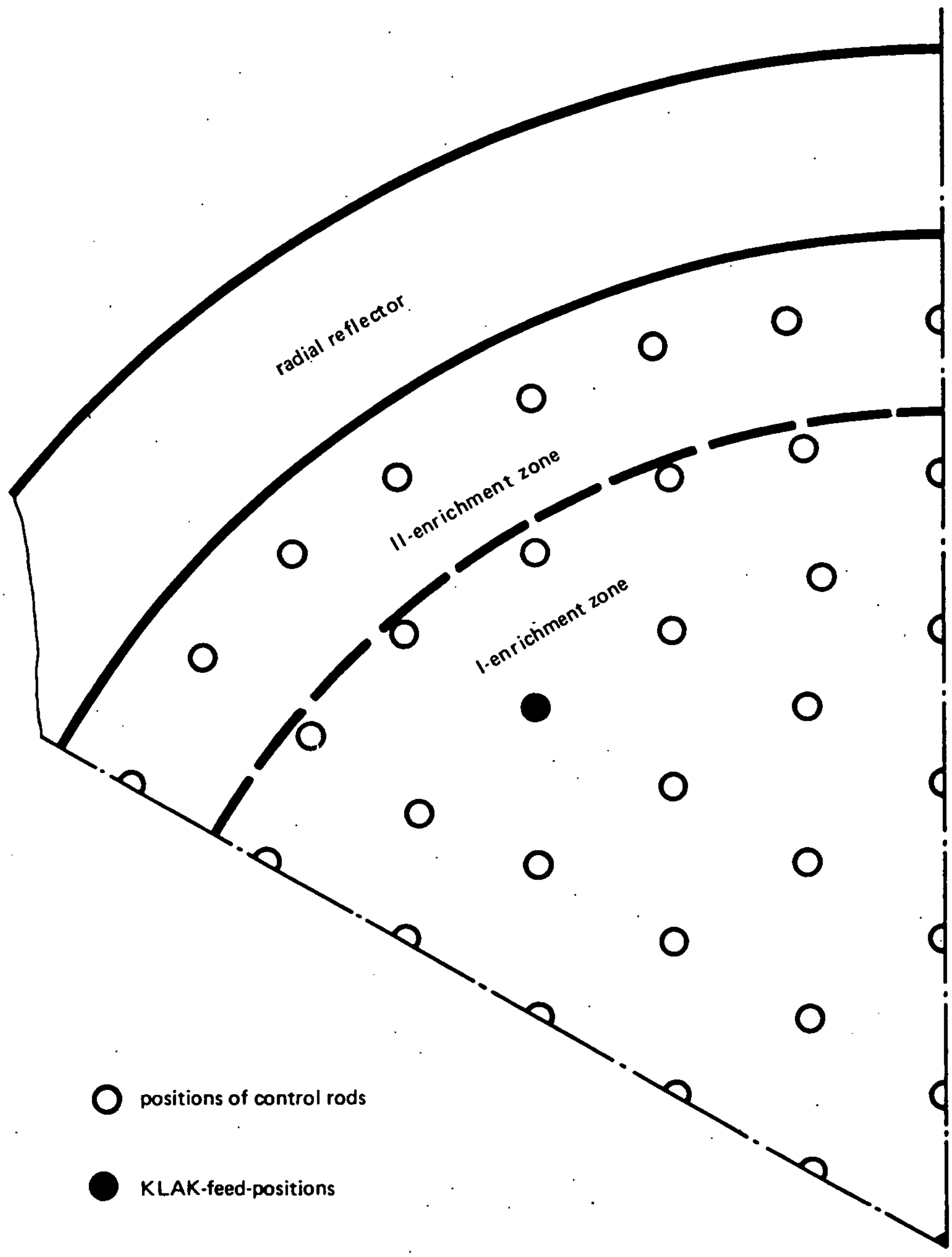


PNP | BF GHT HRB KFA RBW

Figure 3-16. PNP Control Rod Map

Internal cooling of the absorber element is provided by helium flow. Cold helium enters intake openings along the rod and exits through openings at the tip of the rod. In order to keep the cooling flow pressure losses to a minimum and to prevent upward flow of helium in the rod, two sets of intake openings are provided; the shape and position of these openings are illustrated in Figure 3-17.

3.4.2.2 Pneumatic Control Rod Drive

The design of the pneumatically driven control rod which was developed for the THTR is shown in Figure 3-18. Because this control rod must perform power control and shutdowns, it has a short-stroke piston (step insertion for power control) and a long-stroke piston (continuous insertion for shutdown). In the HTR-K, however, the pneumatically driven core rod is used only for shutdowns; hence, it need only have continuous insertion capabilities. Thus, the short-stroke piston is eliminated in the HTR-K pneumatic drive, and this represents an important simplification and cost reduction over the THTR rod.

3.4.2.3 Hydraulic Control Rod Drive

The hydraulically driven control rod of the HTR-K is shown diagramatically in Figure 3-19. The basic components of this drive are

- Valves for controlling the flow of the driving fluid (oil)
- The drive casing
- A purged seal that separates the hydraulic fluid from the primary helium
- The piston and the push rod which connects it to the absorber element
- Stops which define the maximum rod stroke.

Insertion of the hydraulic rod is accomplished by pressurizing the piston on the upper side, which causes a continuous downward movement of the absorber element. After it has attained the desired position, the pressures above and below the piston are adjusted to stop the motion and fix the location of the absorber element. To remove the rod, the procedure is reversed, with the lower side of the piston being pressurized; this forces the piston upward and extracts the absorber element.

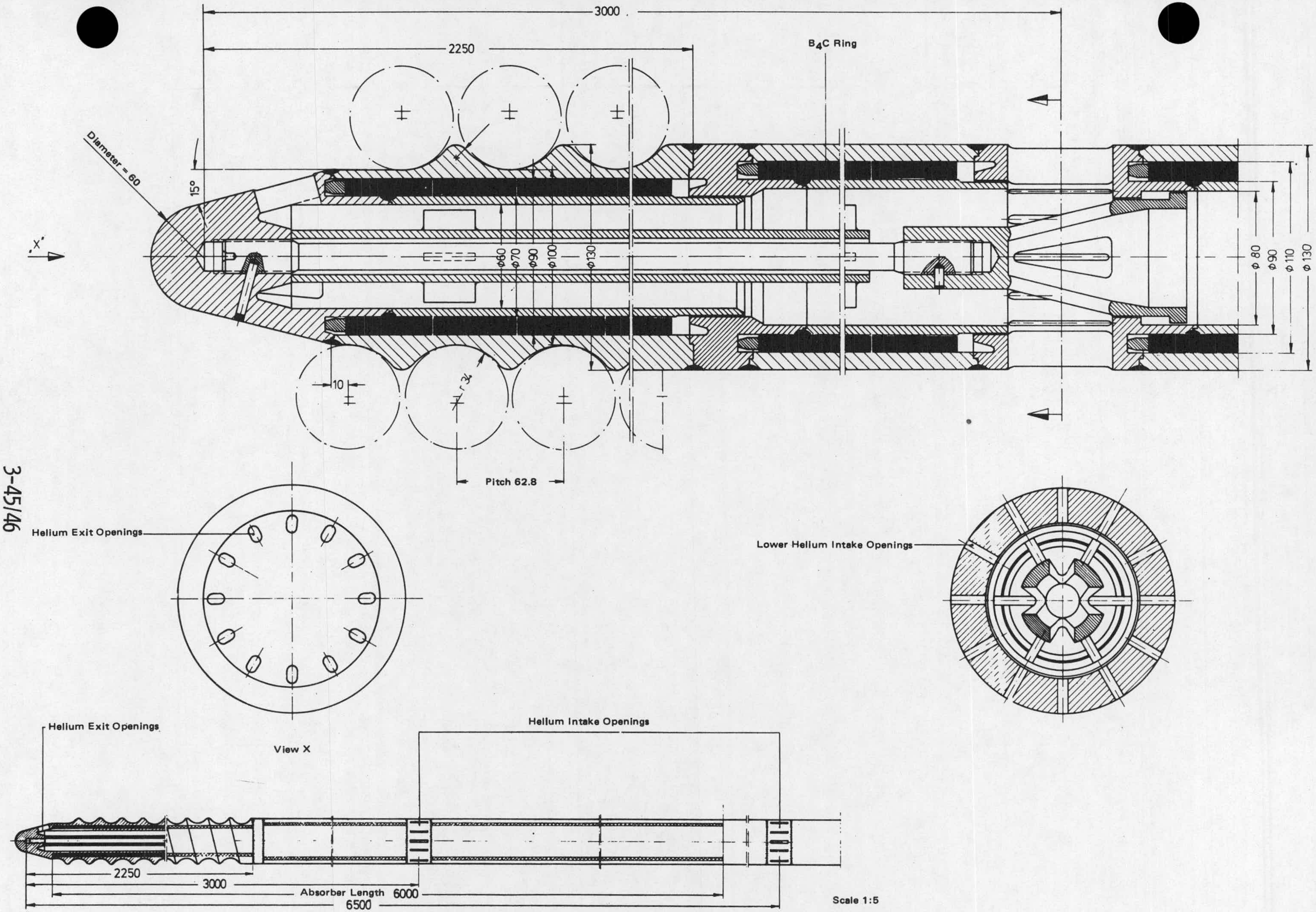
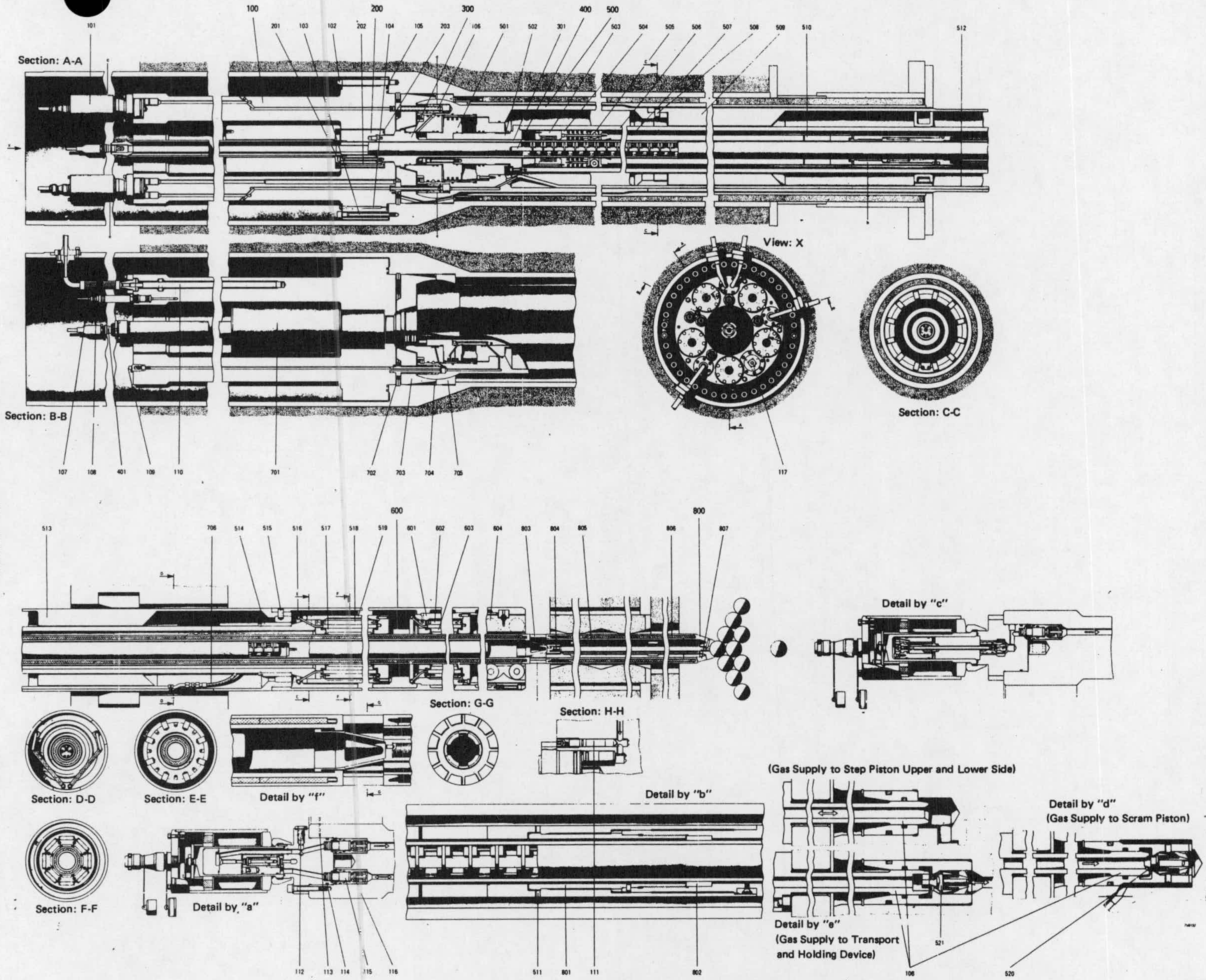


Figure 3-17. Control Rod Absorber Element

Note: All dimensions are in millimeters



Designation of Main Parts

Pos.	
100	Valve Block (Complete)
101	Solenoid Valve
102	Guide Pin
103	Necked-down Bolt
104	Screw Plug (Control Bore Hole)
105	Seal (at Valve Block)
106	Guide Piece ("Finger")
107	Hose Line
108	Pressure Gauge
109	Locking Cap (Greasing Point)
110	Filter
111	Setting Pin
112	Setting Pin (in Valve)
113	Control Bore Hole
114	Seal (at Valve)
115	"Finger" (at Valve)
116	Check Valve
117	Electric Feed-through
200	Closing Element
201	Necked-down Bolt
202	Screw Plug (Control Bore Hole)
203	Seal
300	Imersion Tube
301	Protection Tube
400	Position Indicator Bar
401	Plug Connection
500	Upper Section of Casing (Complete)
501	Step Piston
502	Guide Piece
503	Magnetic Head
504	Long-stroke Piston
505	Upper Guide Bearing
506	Push Rod Tube
507	Push Rod
508	Bolt
509	Shielding
510	Coolant Slits (Push Rod)
511	Bore Holes for Uncoupling
512	Tube line
513	Shielding
514	Guide Sleeve
515	Bolt
516	Annular Tube Line
517	Flat-oval Tube and Bar with Brake Lining
518	Cage
519	Casing (Transport Device)
520	Check Valve (Inlet for Scram)
521	Check Valve (Inlet for Holding Device and Transport Device)
600	Lower Section of Casing (Complete)
601	Casing
602	Cage
603	Flat-oval Tube and bar with Brake Lining
604	Lower Guide Bearing
701	Shield Plug
702	Safety Ring
703	Holding Ring
704	Safety Ring
705	Receiving Ring
706	Hose Line
800	Shutdown Rod
801	Claw
802	Guide Sleeve
803	Central Coolant Slits
804	Safety Check Rod and Holding Plug
805	Centering Tube
806	B_4C Rings
807	Lower Coolant Slits

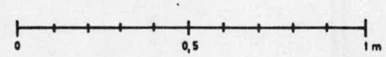


Figure 3-18. THTR Pneumatic Control Rod Drive

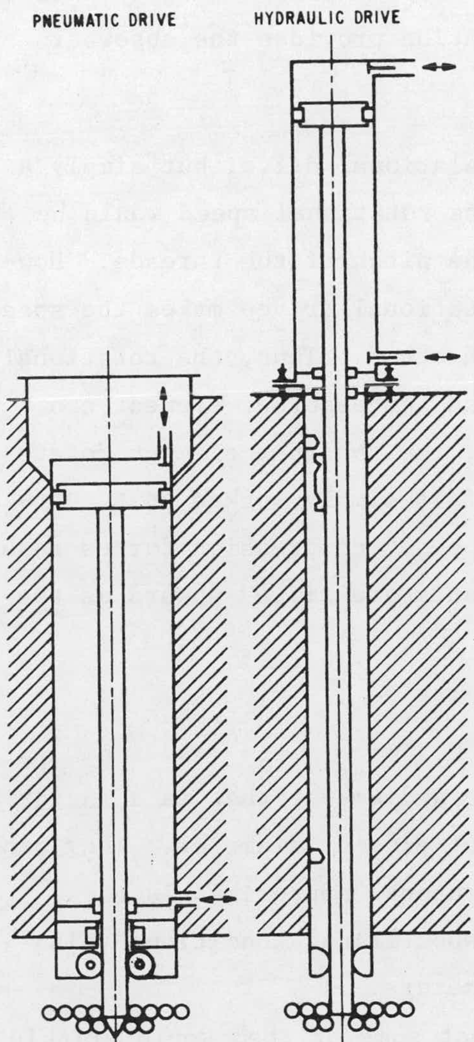


Figure 3-19. Control Rod Drive Options

3.4.2.4 Spindle Control Rod Drive (with rotation)

The reference control rod design for the PNP reactor is a rotating rod with a spindle drive; this control rod design is shown in Figure 3-20. Its major components are

- A primary gas seal, which is a rotating bushing located in the top of the penetration liner
- The spindle and spindle nut
- The axially fixed, torsion tube, which is attached to the absorber in a torsion-proof manner
- The absorber element, which is threaded over approximately half of its length.

During operation, torque is applied to the coaxial shafts by an electric drive motor. The inner shaft rotation provides the translational motion via the spindle drive, and the outer shaft rotation provides the absorber elements' rotation via the torsion tube.

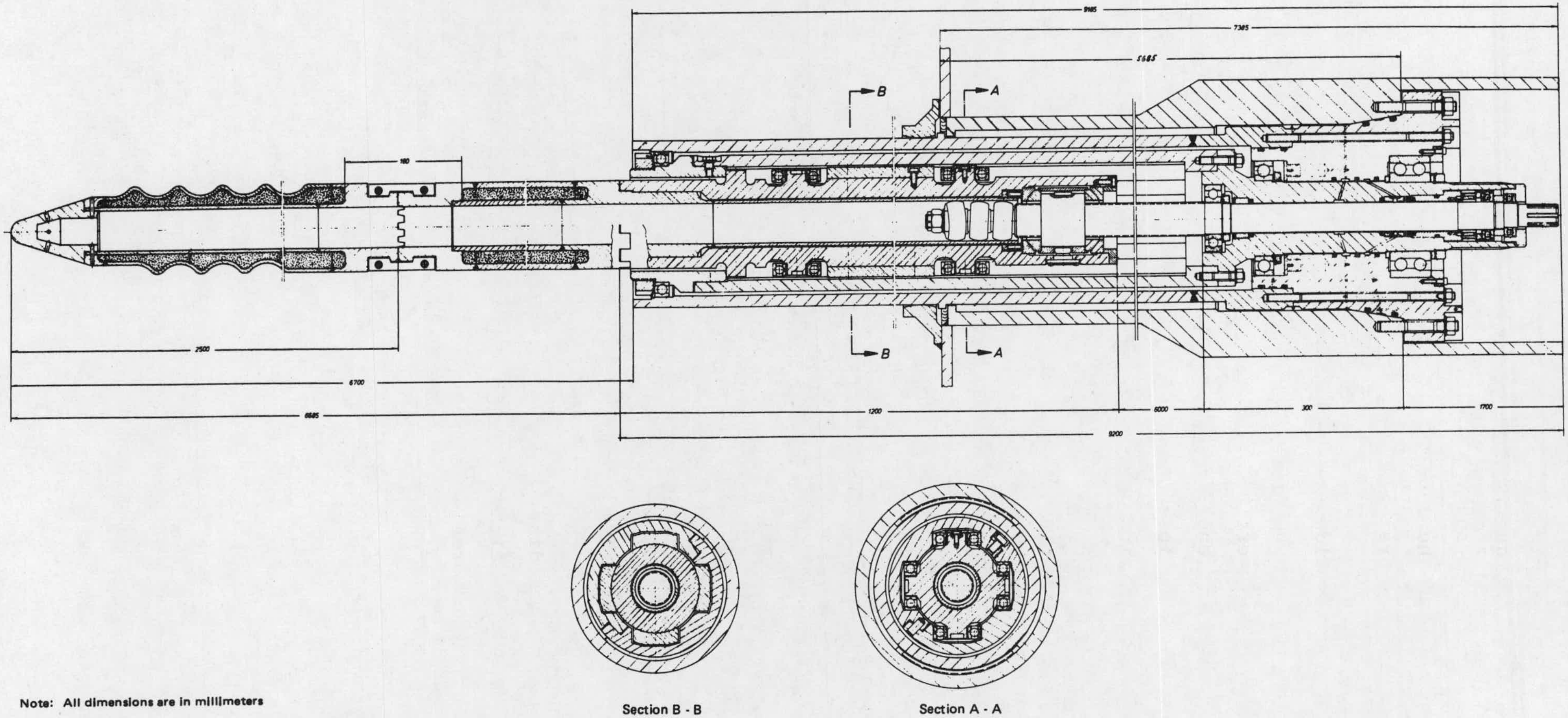
If this control rod did not have a translational drive, but simply a rotational one, it would act as a screw, and its rotational speed would be defined by the allowable insertion speed and the pitch of the threads. However, the use of separate translational and rotational drives makes the speeds of rotation and translation independent of each other. Thus, the rotational speed is not defined by the insertion speed, and the absorber element can rotate over a range of speeds. In this design, the absorber element rotates fifty percent faster than the speed required if it simply worked as a screw. The advantage of this excess rotation is that no net compression forces result from the use of this rod because the fuel elements are lifted upward as the rod inserts.

3.4.2.5 <u>KLAK System</u>

The KLAK are graphite spheres which have a diameter that is a factor of approximately 6.3 times smaller than the fuel element diameter. They contain one volume percent of B_4C. At this B_4C content, approximately three million KLAK are necessary to achieve a cold, subcritical condition; this quantity of KLAK would occupy about two cubic meters.

The size of the KLAK was selected so that some of them would trickle through the larger voids in the core, while others are trapped by the minimum size voids. The resulting distribution of KLAK is shown in Figure 3-21 in terms of the ratio of the weight of small balls (W_A) to the weight of large balls (W_B) in a given core region. This distribution closely follows the axial flux distribution for an OTTO-type core.

In the present concept, the KLAK would be held in seven containers above the PCRV and would be manually activated. Upon activation, the membrane at the bottom of each container is broken, and the KLAK showers onto the pebble bed. The method of removing the KLAK from the pebble bed is still under investigation. Present indications are that approximately 70% of the KLAK can be dislodged by inserting the control rods and that removal of all the KLAK requires that ten percent of the fuel elements be removed. The manner in which the KLAK actually exit from the core is not yet decided.



Note: All dimensions are in millimeters

Figure 3-20. PNP Spindle Control Rod Drive

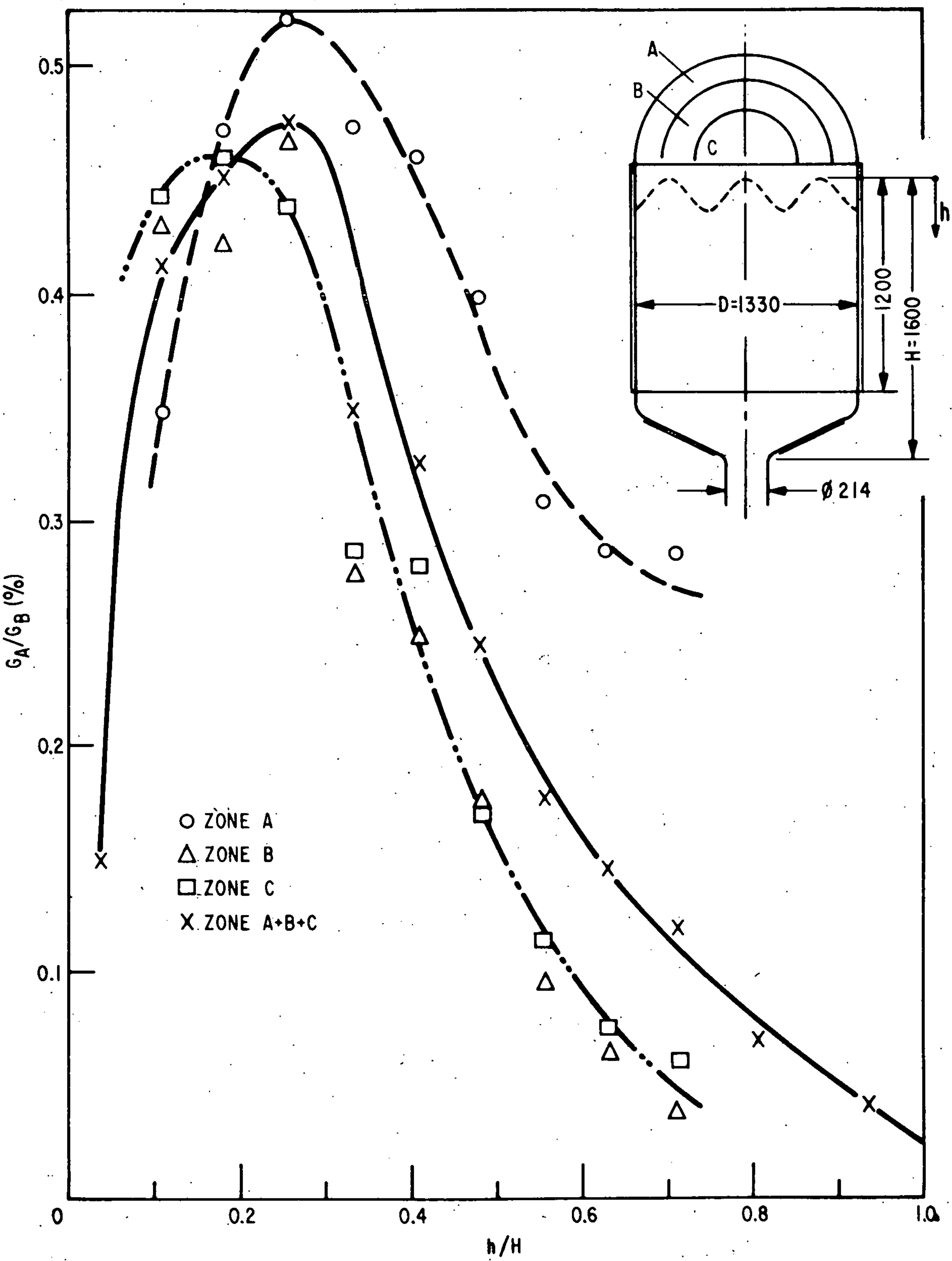


Figure 3-21. KLAK Characteristic Distribution

3.4.3 OPERATIONAL CHARACTERISTICS

The reactivity control systems for the HTR-K and PNP have three basic functions: power control, short-term shutdown, and long-term shutdown. In the following two sections, the operational characteristics of the HTR-K and PNP systems will be described for each of the three basic functions.

3.4.3.1 HTR-K Operational Characteristics

In the HTR-K, the second control system, consisting of 156 core rods and 24 reflector rods, is used for power control. During these maneuvers, the core rods move within the top reflector at a speed of 2 cm/s with a position accuracy of ±2 cm; the reflector rods would move in a similar manner in the side reflector. The reactivity necessary for power control is 4.4% Δk. While performing power control, the maximum absorber element surface temperature would be approximately the cold gas temperature of 260°C.

The first control system, 42 core rods and 24 reflector rods, is used for the short-term shutdown of the HTR-K. Upon reception of a scram signal, all these rods are completely inserted. In this case, the core rod speed is 30 cm/s, and the rods penetrate to a depth of 4.35 m. When the system is activated, ammonia is injected into the reactor core to act as a lubricant between the fuel elements and the absorber elements; this is necessary to reduce the insertion forces to acceptable levels. This system must provide 2.8% Δk reactivity with the two highest worth rods unavailable, one failed and one in repair (to satisfy German licensing and utility requirements).

Long-term shutdown capability is also provided by the second control rod system. During this function, all rods in the system insert to their deepest position and provide 13.4% Δk reactivity. The insertion speed is again 2 cm/s.

3.4.3.2 PNP Operating Characteristics

In the PNP, all three functions are provided by the control rod system with the KLAK system serving as the second independent shutdown system to satisfy the German licensing requirements. For power control, the absorber elements move within the upper reflector at a speed of 2 cm/s to an accuracy of position of ± 2 cm, in a manner similar to the HTR-K core rods. During these maneuvers, the maximum surface temperature of the absorber element is approximately 300°C, the cold gas temperature.

For most short-term shutdowns, all control rods are inserted to a depth of 0.5 m. The exception occurs during full-load operation immediately after a long-term shutdown. Under these conditions, the insertion is to one meter due to the lack of the negative reactivity from the equilibrium xenon and protactinium concentrations. Because of the smaller compression forces exerted by the rotating rods, no ammonia injection is necessary. The rods insert at 2 cm/s, and an accuracy in position of ± 2 cm is necessary to insure that no rod travels too far and exceeds the surface temperature limit of 700°C. The amount of reactivity required to achieve this hot, subcritical condition is 2.8% Δk. Again, this must be provided with the two most valuable rods unavailable.

To achieve a cold, subcritical condition for long-term shutdown, the absorber elements are inserted to a depth of 4.5 m at a speed of 2 cm/s. This insertion is performed stepwise to preclude the possibility of an absorber element surface temperature exceeding 700°C. At the present time, it appears that such a shutdown can be accomplished in three steps without exceeding an absorber surface temperature of 600°C.

As discussed above, the PNP reactor uses rotating control rods for power control and for both short- and long-term shutdowns. However, this system alone cannot satisfy the German licensing requirement for two independent shutdown systems; therefore, the PNP reactor has the small absorber balls, KLAK, which can maintain a long-term, cold, subcritical condition in the core. However, the KLAK system will be used only in the event of significant failures in the rod system.

3.4.4 NUCLEAR INSTRUMENTATION

The purpose of nuclear instrumentation is to provide information for the reactor protective system, in addition to the measurements necessary for reactor control during all normal operating or transient conditions. In the case of large pebble bed cores with low power density and an OTTO fuel cycle, new techniques are needed for measuring the flux distributions, because in-core measurements are difficult in a pebble bed, and large cores have a tendency toward xenon oscillations. In the pebble bed core, axial xenon instability is limited by the relatively low core height. However, analysis has shown that damped radial and azimuthal oscillations can occur because the core's diameter is greater than 10 m. Hence, an essential task for the instrumentation is the detection of radial and azimuthal xenon oscillations in the upper half of the core.

Apart from the measurements for the reactor protective system, neutron flux measurements must provide the axial flux and power distributions. Because the axial flux distribution is only slightly "blurred" by the side reflector, detectors placed behind the side reflector provide excellent information on the axial distribution inside the core. Hence, the detectors are placed between the side reflector and the thermal shield at four points displaced by 90° around the circumference of the core.

For the startup and transition range, two fission chambers at each location are used. These chambers are in position during startup, but once the power range is reached, they are removed. Therefore, they are not subjected to excessive temperature or radiation-induced stress.

For the power range, six vertically positioned detectors are uniformly spaced along the height of the core at each measurement position. In order to obtain a signal closely proportional to the total power output, regardless of the nature of the axial power distribution, the signals from the six detectors are summed to provide a single, integrated signal. Because of the relatively high temperature ($450^{\circ}C$), the use of large area (n,β) detectors was investigated, and the results showed that this type of detector would be satisfactory.

Measurement of the radial and azimuthal power distribution is performed by fast flux detectors in the top reflector. Because the radial fast flux distribution in the top reflector is a facsimile of the fast flux in the upper regions of the core, such measurements provide information on the power distribution in these regions. The local thermal flux in the top reflector cannot yield information of the power distribution directly below the measuring point because of the relatively large size of the void between the top of the pebble bed and the top reflector.

3.5 FUEL SYSTEM

The base fuel system selected for both the PNP and HTR-K reactor is the same one specified for the THTR. This decision limits the required development and makes maximum use of the on-going fuel testing and development associated with the THTR. Alternate fuel systems are being studied separately, and both proliferation-resistant and higher conversion systems could be used if appropriate. The report "Fuel Cycle Evaluation" discusses fuel cycle analyses in detail.(21)

3.5.1 FUEL ELEMENT DESIGN

Table 3-7 shows the data for the fuel element for the reference fuel. The basic design consists of coated fuel particles in a graphite matrix. This matrix, 50 mm in diameter, is enclosed in a 5 mm-thick graphite shell to form the 60 mm-diameter fuel element. The coated particles are of the BISO type. They consist of a kernel of mixed thorium-uranium oxide 400 μm in diameter covered with three graphite layers. Next to the kernel is a low-density ($\leq$1.0 g/cm^3) pyrolytic carbon layer to act as a buffer and a trap for fission products. Surrounding this is a double pyrolytic carbon layer, the first a sealing layer of high density ($\sim$1.6 g/cm^3) and the second outer layer at $\sim$1.85 g/cm^3.

3.5.2 Fuel Element Performance

Many references to the performance of coated fuel particles have been published. This basic type of fuel has been studied by the U.S. (GA and ORNL) and the British (Dragon Project), as well as by the Germans. Little doubt exists as to the performance of the particles themselves. Work continues on reducing uranium contamination of the coatings (a major source of fission product release) and improvement of the product by means of process refinement.

The performance of the fuel elements themselves has been demonstrated in the AVR reactor, which has been operated at an exit gas temperature of 950°C for more than three years. Many thousands of THTR-type balls have been tested in the AVR. Development continues to improve the manufacturability of the elements and to evaluate the limits of performance under simulated accident conditions. Work is also under way to develop fuel elements for the alternate fuel cycles, particularly elements with the higher fuel loadings required for the high-conversion and recycle schemes.

TABLE 3-7

FUEL DATA FOR PNP AND HTR-K

Parameter	Value
Fuel Element	
Type of Element	Spherical
Type of Fuel	Th-U Mixed Oxide
Diameter of Fueled Zone	50 mm
Diameter of Element	60 mm
Density of Matrix and Shell Material	1.7 g/cm^3
Heavy Metal Loading (Th,U)	11.24 g/ball
Particle Volume Fraction	∿ 9%
Coated Fuel Particles (BISO)	
Kernel Diameter	400 μm
Thickness of the Three Coatings	85/30/80 μm
Particle Diameter	790 μm
Density of Kernel	9.5 g/cm^3
Density of Three Graphite Coatings	1.0/1.6/1.85 g/cm^3
Fuel Cycle	
Type	OTTO
Uranium Loading (93% U-235)	0.85/1.04 g/ball*
Average Residence Time	1160 Full-Power Days
Conversion Ratio	0.59

*Lower Value for Central Zone of Core.

The similarities between the PNP fuel elements and their environment and the THTR fuel elements and their environment is shown in Table 3-8. Operation of the THTR will further demonstrate adequate fuel element performance characteristics for the PNP reactor.

3.5.3 ALTERNATE FUEL CYCLES

There are several alternate fuel cycles under consideration for both the HTR-K and the PNP. They are summarized below, and discussed in more detail in another report.(21)

3.5.3.1 Low Conversion Alternates

HRB is interested in using a fuel cycle like the GA HTGR cycle. In this option, the uranium is formed into one (TRISO)* fuel particle and the thorium into another BISO particle. These particles are of different sizes, so that, in principle, they can be processed separately after discharge from the reactor. This would allow separation of high-purity U-233 for recycle. Since other means exist to do this, as discussed below, the major advantage of this scheme would appear to be commonality with the GA HTGR work, especially similar reprocessing requirements.

In addition to the two-particle system, HRB has investigated the use of two kinds of fuel elements for HTR-K, one using as much fuel as technically feasible (they assumed 15-20 g of heavy metal per ball), and the rest completely unfueled (dummy balls). The rationale is basically that an optimization of fuel cycle costs results because the savings achieved by fabricating fewer balls containing fuel overrides the increased fabrication cost of balls with heavier (than THTR type) fuel loading. Overall performance is not changed significantly.

*TRISO-coated fuel particles have an extra layer of pyrolytic silicon carbide between the outer two graphite layers. This is intended to improve fission product retention at exit gas temperatures above 950°C. However, there are no current plans to exceed 950°.

TABLE 3-8

COMPARISON OF THTR AND PNP FUEL

Data	Dimension	THTR 300	PNP 3000
Fuel element	-	spherical	spherical
Fuel cycle	-	Thorium (93% U-235)	Thorium (93% U-235)
Ball flow	-	6 passes through core	1 pass through core (OTTO)
Diameter ball	cm	6	6
Heavy metal	-	U/ThO_2	U/ThO_2
Heavy metal content ball	g	11.2	11.2
Fuel particle kernel diameter	μm	400	400
Coating	-	BISO	BISO
Mean core power density	MW/m^3	6	5.5
Temperature rise of cooling gas	^oC	250 → 750	300 → 950
Burnup	MWd/tHM	110 000	100 000
Max. fuel element surface temp.	^oC	950	1020
Max. coated particle temperature	^oC	1020	1050
Surface temperature limit	^oC	1050	1050
Coated particle temperature limit	^oC	1250	1250
Fast dose (E>0.1 MeV)	n/cm^2	6.3×10^{21}	4.5×10^{21}

3.5.3.2 High Conversion Alternates

One of the most interesting features of pebble bed reactors, even net breeders, is the potential to reach very high conversion ratios, in essentially the same reactor. Reprocessing and recycling of the fissile fuel is required. Listed below are some of the requirements for high conversion.

- Increased fuel loading per ball. This decreases the carbon-to-heavy-metal ratio, thus increasing the conversion ratio.

- Control of the U-236 buildup (which is a nuclear poison) from the U-235 fissile material.

- Decrease in the average burnup, which minimizes losses to fission products.

- Use of bred U-233 in a separate reactor (No U-235 or U-236).

Many schemes have been investigated in order to increase the conversion ratio without incurring unduly high fuel cycle costs. All use variations of the thorium cycle to achieve recycle of U-233. Conversion ratios up to ~0.85 can be achieved without the use of separate U-233-fueled reactors. The use of separate U-233-fueled reactors, high fuel loadings, frequent reprocessing, and radial fertile blankets permit the achievement of ratios as high as 1.05. The use of a decoupled flow of fertile material could possibly further increase the breeding ratio up to 1.10.

One key requirement for these cycles is the development, testing, and qualification of fuel balls containing high heavy metal loadings. To achieve high conversion, loadings approaching 45 g/ball will be required. Present experience is shown in Table 3-9. There seems to be no barrier to achieving the required high loadings if they become economically attractive.

3.5.3.3 Proliferation Resistant Cycles

Although the base fuel system uses highly enriched U-235, the Germans recognize that nontechnical pressures may require a less proliferation-prone fuel form. Two main lines of study are being pursued:

- Full scale fuel elements well qualified
  - Macro configuration constant
- Coated particles are the same for all
  - Burnups to 160,000 MWd/t or greater

TABLE 3-9

PBR FUEL ELEMENT DEVELOPMENT STATUS

Fuel Description	Heavy Metal Loading (Grams/Ball)	Volume Fraction of Coated Particles in Fuel Matrix	Comments
AVR Fuel	6	∿ 0.1	Off-the-shelf, fully qualified. 3.5 years with 950°C exit gas.
THTR Fuel	11.2	∿ 0.10	Developed, tested in the AVR. Qualified for use.
Developed Fuel	16 → 20	0.12 - 0.17	Developed, under test in AVR. Not fully qualified.
Projected Fuel	30 → 45	0.25 - 0.35	Needs manufacturing development, which is under way.

The first is a cycle using low enriched U-235, about 8%, with no thorium. This type of cycle resembles that of the present-day light water reactors. It has all the problems and benefits associated with present LWRs, especially the production of fissile plutonium isotopes which can (in principle) be separated chemically to make material for nuclear explosive devices. By assuming the same type of no-reprocessing environment, this cycle would probably be as acceptable (or unacceptable) as that of present-day LWRs.

A proliferation-resistant cycle which is especially suited to the PBR is the 20% enriched thorium cycle. In this cycle, U-235 enrichment is limited to a value (say 20%) which is accepted as safe from a proliferation viewpoint. Thorium is added to allow the breeding of U-233, which is burned without reprocessing. The discharged fuel contains about 1/10th of the fissile plutonium of LWR, and it is largely diluted with non-fissile plutonium isotopes. It is probably as "safe" a cycle as has been proposed.

German activity in this area has included the testing, in the AVR, of fuel elements of both types. It is expected that either system can be developed to reach an acceptable status.

3.6 PNP AND HTR-K REACTOR CHARACTERISTICS LISTING

This section contains Table 3-10 and a summary tabulation of HTR-K and PNP reactor characteristics.

TABLE 3-10

REACTOR CHARACTERISTICS FOR HTR-K & PNP

	Units	PNP	HTR-K
Core			
Thermal power	MW	3000	
Average power density	MW/m^3	5.5	
Average core height	m	5.53	
Core radius	m	5.60	
Average height of void above the core	m	1.0	
Burnup (average)	MWd/ MT	100 000	
Fuel cycle		OTTO	
Number of core zones		2	
Thickness of the outer core zone	m	1.00	
Average cooling gas temperature-inlet/outlet	^{o}C	300/950	260/700
Operating pressure	bar	40	60
Fill fraction of pebbles dropped in the core		0.61	
Number of pebbles in the core		$3x10^6$	
Direction of helium flow		top to bottom	
Helium flow rate	kg/s	888	1320
Pressure drop (core and bottom reflector)	bar	0.50	0.63
Load range	%	40/100	25/100
Design lifetime	full-power years	32	
Capacity factor	%	80	

	Units	PNP	HTR-K
Fuel Element			
Geometry		spherical	
Fuel		$(ThU)O_2$	
Radius of fuel zone	cm	2.5	
Outer radius of the pebble	cm	3.0	
Density of matrix and shell	g/cm^3	1.7	
Coated particle volume fraction	%	9	
Heavy metal loading per pebble	g	11.24	
U-235 enrichment	%	93%	
U-235 loading per pebble	g	0.85/1.04	
Average residence time	FPD	1160	
Conversion ratio		0.59	
Max./average output in the core		2.7	
Max. power per sphere	kW	2.8	
Max. surface temperature	^oC	981	732
Max. center temperature	^oC	1012	781
Coated Particles			
Diameter of the kernels	mm	400	
Thickness of the 3 coatings	mm	84/30/80	
Density of the mixed-oxide kernels	g/cm^3	9.5	
Density of the 3 coating layers	g/cm^3	1.0/1.6/1.85	
Neutron Doses (E 0.1 MeV)			
Pebble at discharge	n/cm^2	$4.5 \cdot 10^{21}$	
Max. top reflector	$n/cm^2/yr$	$1.6 \cdot 10^{21}$	
Max. side reflector	$n/cm^2/yr$	$1.8 \cdot 10^{21}$	

	Units	PNP	HTR-K
Fuel in the Core			
Heavy metal	kg	31987	
Fissile material U-233	kg	541	
U-235	kg	530	
Pu-239	kg	1.9	
Pu-241	kg	0.9	
Reflector			
Material		graphite	
Construction			
top reflector		hexagonal blocks suspended from the top thermal shield	
side reflector		segmental blocks supported by the side thermal shield	
bottom reflector		hexagonal blocks with perpendicular holes for gas flow, supported in upright position by graphite columns	
Thermal Shielding			
Material		gray cast iron	
Construction		stacked annular rings	
top shield		circular plate attached to & supported by the control rod penetration liners	
side shield		free standing cylindrical shell; in PNP, with channels for internal cooling gas flow	

	Units	PNP	HTR-K
Reactivity Control System			
System design		core rods plus KLAK	core rods & reflector rods
Number of control rods, core		156	198
reflector		none	48
Type of control rods		rotating	lifting
Outer diameter	cm	13.0	10.5
Active length	cm	600	635
Drives, core rods			
first shutdown system		electric spindle	Pneumatic long stroke piston
second shutdown system		electric spindle	hydraulic long stroke piston
Drives, reflector		none	gravity/ electric
Accuracy of rod position control	cm	±2	±2
Maximum insertion velocity	cm/s	2	30
Maximum insertion depth	m	4.5	4.35
Reactivity requirements			
first shutdown system	%Δk	2.7	2.8
second shutdown system	%Δk	20.3	17.8
Specific Characteristics			
Fuel Handling System			
Throughput of fuel elements	d^{-1}	2654	
Number of inlet tubes		43	
Number of fuel exit tubes		6	
Loading time per day	h	1.5	

	Units	PNP	HTR-K
Fuel Element Storage			
Fresh fuel elements			
storage capacity	years	1	
number of fuel elements		775,000	
number of fuel element containers (1000 elements per container)		775	
Spent fuel elements			
storage capacity	years	8	6
number of fuel elements		6×10^6	4.5×10^6
number of fuel element containers (2100 elements per container)		2860	2140

3.7 ANALYSIS METHODS

3.7.1 CORE DESIGN METHODS

The core design methods at the HRB, GHT, and KFA Laboratories have been developed independently but are quite similar in most respects, including the various approximations which are utilized in carrying out the design calculations. The methods have been under development since the middle 1960's, and there is a high degree of experience in the application of the methods to the design effort.

The HRB, GHT, and KFA Laboratories all utilize a modular package of loosely coupled design codes as shown in Figure 3-22. In following the flow of the calculations in Figure 3-22, the neutron spectrum and data processing module is utilized to generate the nuclear data parameters for the design analysis. Of particular importance in this module are the heterogeneity calculations which are carried out for the specified fuel cell model. The neutron flux, eigenvalue, and reaction rate calculations are performed in the neutronics module, and the output is utilized in the burnup routine to compute the changes in the fuel isotopic distribution for a given burnup time step. The output from the neutronics module is also utilized in the thermal-hydraulics routines to generate the fuel, coolant, and moderator (structure) temperature distributions throughout the core regions. (A detailed description of each of the modules in the code package is given in the sections which follow.) Most of the systems also include a routine for computing the fuel cycle costs. However, this routine is not necessarily an integral component of the overall system. In setting up the code package as shown in Figure 3-22, there is generally a trade-off between the accuracy and the costs of the design computations. Thus, for example, two-dimensional synthesis diffusion techniques are utilized in place of direct two-dimensional methods for the neutronics calculations. In general, the methods are highly developed and there is evidence that the synthesis techniques, including both methods and reactor modelling approximations, have been checked out and, in some cases, normalized with respect to the more sophisticated computational design tools (e.g., transport and Monte Carlo methods).

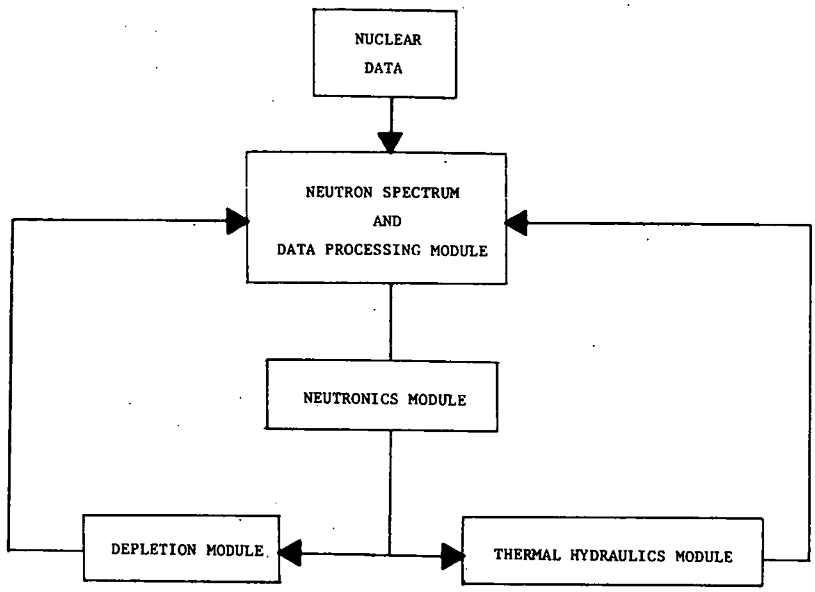


Figure 3-22. PBR Modular Analysis Code

The computer costs associated with the code packages vary significantly with the type of problem being considered. For a detailed design analysis in which the reactor history is traced from a freshly loaded core to an equilibrium operating cycle, the processing time on the IBM-370 series computer may take from 4 to 8 hours. This processing time may be significantly reduced for scoping-type calculations based only upon an equilibrium burnup cycle and limited iterations on the flux solutions. However, at best, the computations are moderately expensive relative to the design calculations which are performed for the light water reactor systems.

In describing the principal components of the linked system shown in Figure 3-22, it is noted that each of the German design groups has a variety of codes with which to analyze problems of varying degrees of complexity. For example, KFA utilizes the two-dimensional diffusion code EXTERMINATOR-II in carrying out the xenon calculations. In addition, the two-dimensional transport code DOT and the three-dimensional diffusion code CITATION are used to analyze the effects of control rod movements. The Monte Carlo code KENO-II has also been used to study the effects of the void region between the upper core and the reflector, and has been employed in the analysis of several experiments performed in the KAHTER critical facility located at the KFA Laboratory. Similar codes are utilized by the HRB and GHT design groups. It is noted that all of the design codes listed above have been originally developed in the United States with some further development in Germany, and with the exception of several recent code development programs in this country (e.g., the three-dimensional neutronics and burnup code VENTURE, KENO-IV, etc.), the design capabilities in Germany and the United States are generally comparable. In the material which follows, the principal components of the code package shown in Figure 3-22 are described relative to those modules which are most frequently used in the Pebble Bed core design analysis.

3.7.1.1 Neutron Spectrum and Data Processing Module

The thermal neutron spectrum calculations at KFA are carried out with a modified version of the thermal transport theory code THERMOS.[5] The modified THERMOS code is utilized to compute the space-and energy-dependent flux distribution, utilizing collision probability theory in the treatment of the lumped coated particle fuel zone. The fuel lumping heterogeneity treatment provides a somewhat more accurate calculation of the thermal cross sections relative to a simple homogeneous cell model. That is, the heterogeneity treatment gives thermal cross-section values which differ, in some instances, by as much as 2 to 3 percent from the homogeneous values. In contrast to the KFA methods it is noted that the HRB design group tends to utilize a homogeneous model for the spectrum calculations which are carried out with the data processing code MUPO. In validating the accuracy of the HRB Pebble Bed design studies, it will be necessary to ensure that the differences between the heterogeneous and homogeneous calculations have been factored into the design analysis.

The epithermal and fast neutron spectrum calculations at KFA are based upon the P1 multigroup approximations in the GAM-1 code.[6] The Nordheim numerical integration method[7] is used for the lumped resonance absorber region. In addition, a double heterogeneity treatment based upon collision probability theory[8] is utilized to account for the probability that a neutron born in one coated particle will be absorbed in any other fuel kernel. (The double heterogeneity treatment refers to the heterogeneity approximations which are first performed for the coated particles and then for the fuel elements.)

The methods for performing the epithermal and fast neutron spectrum calculations at GHT are very similar to those at KFA. However, a different technique is utilized at HRB. There the resonance calculations are carried out on the basis of the Russian f-factor approach.[9] That is, resonance cross sections are calculated for an average total background cross section, σ_o, due to the presence of all other materials. For a specific reactor composition (and thus a specific σ_o), the cross sections for the material of interest are determined by extrapolating between precalculated data points for which the values of the σ_o's are known. A double heterogeneity treatment, again based upon collision probability theory, is used to generate equivalent σ_o's for the lumped fuel region. This approximation technique results in a reduction in the computer running time relative to the direct calculations in GAM-1. However, experience with the f-factor approach has indicated that the accuracy is poor in the epithermal energy regions where the narrow resonance approximation is no longer good. It will again be necessary to make direct comparison of the HRB and KFA methods to ensure that the differences in the methods do not lead to inaccuracies in the HRB Pebble Bed design analyses.

3.7.1.2 Neutronics Module

The neutronics calculations at HRB, GHT, and KFA are carried out using the CONDOR, CIRCUS, and FEVER codes, respectively. as was noted above, extensive use is made of two-dimensional synthesis techniques to reduce the costs of the design calculations. Thus, for most design analyses the neutron flux in R-Z geometry is computed from a series of linked one-dimensional diffusion calculations. The synthesis approach is very similar to that used in the GE two-dimensional synthesis program BISYN[10], and is based upon a reactor model with several radial and axial zones which serve as channels for the one-dimensional computations. The perpendicular flux solutions are linked to each other through the perpendicular groupwise buckling terms. In general, the calculations are carried out first in the radial direction, then in the axial direction (generally utilizing 2-4 different flux solutions in the different axial channels) and concluded with a final radial run. The two-dimensional flux solution is given by a simple combination of the radial and axial flux solutions.

Experience with the two-dimensional flux synthesis approach in both Germany and the United States (relative to the direct two-dimensional calculations) has indicated that this approach is generally adequate for predicting the neutron flux in core regions. However, there is a significant uncertainty in the calculation of the fast neutron flux level in the graphite radial reflectors of the Pebble Bed Reactor using the synthesis approach, and this method is probably not adequate for predicting the fluence in the reflector regions. Therefore, the calculation of flux levels in the reflector regions is carried out using direct two-dimensional methods.

3.7.1.3 Depletion Module

The depletion calculations based upon the finite difference methods of the FEVER code are generally typical of the type of burnup calculation which is carried out in the Pebble Bed design analysis. The reactor model in FEVER may consist of up to 200 subregions which contain different nuclide concentrations. One or more subregions are assigned to a diffusion zone for which the average neutron flux has been determined in the neutronics calculations. The flux solutions are computed over a burnup time interval of approximately 20-30 full power days. In addition, this interval is divided into a number of smaller time steps (3-6 steps) in which the flux level is adjusted by renormalizing to the reactor power. Forty explicit fission products, plus one accumulated fission product, are included in the depletion routine.

The burnup routine generally includes a fuel management subroutine for the relocation and accountability of both the in-pile and out-of-pile fuel. In this operation the 200 distinct fuel subregions may be handled separately or in groups.

3.7.1.4 Thermal-Hydraulic Module

Two distinctively different approaches are used in the thermal-hydraulic analysis of the Pebble Bed system. First, at GHT the thermal-hydraulic calculations are carried out on the basis of a rod bundle fuel element model using the thermal-hydraulics code COBRA-IIIC.[11] In this approach it is necessary to correlate the treatment of the Pebble Bed system and the equivalent rod bundle. Specifically, it is necessary to

a) Mockup the fuel pin model to represent the Pebble Bed spherical fuel elements.
b) Specify the bundle channels to give the correct pressure drop over the Pebble Bed core, and
c) Provide the correct correlation of the Pebble Bed radial coolant streaming and energy deposition.

The rod bundle representation of the Pebble Bed system is aided in part by the fact that there is very little radial mixing in the Pebble Bed core. In setting up the rod bundle model approximately 25 annular regions are utilized to describe an equivalent Pebble Bed zone.

The thermal-hydraulic calculations at HRB and KFA are carried out with the codes MÜNSTER and TIK-LSD, respectively. In general, the two codes use a similar approach based upon the gas flow in a porous medium. A coupled series of two-dimensional mass and viscous momentum transfer conservation equations, together with an empirical pressure drop equation, is used to determine mass flow, velocity vectors, and Reynold's numbers. Similarly, in TIK-LSD the gas temperature distribution is determined by using the energy conservation equation and an equation of state. A description of the thermal-hydraulics module, including the approximations which are used in the code routines, has been given by C.E. Lee of LASL in an overall evaluation of the KFA VSOP system.[12]

The power density profile is input to the TIK code for the gas temperature determination. The gas temperature distribution from TIK is used in turn in LSD to solve for the fuel center temperature. In computing the fuel temperatures, the average surface heat transfer coefficient is determined by utilizing the Reynold's number in an equation for the free flowing sphere in a homogeneous mixture. C.E. Lee notes that the uncertainty in the central fuel transfer coefficient leads to an estimated uncertainty in the central fuel temperature of approximately 50°C in the upper core region, where the power density is high. In the lower part of the core the corresponding uncertainty is about 5°C. These coefficients are being investigated experimentally.

3.7.2 NUCLEAR DATA

The HRB, GHT, and KFA design groups make extensive use of the Evaluated Nuclear Data File/Version B.[13] At HRB and GHT these files are used almost exclusively in the Pebble Bed design studies, although different versions of the files are utilized at the two nuclear centers: That is, the HRB design group uses ENDF/B-II, which is based upon data evaluations up through 1970-1971. In contrast GHT uses ENDF/B-IV, which includes data evaluations through 1974-1975. From the standpoint of the Pebble Bed analysis the differences between ENDF/B-II and ENDF/B-IV files are small.

The multigroup data libraries at KFA are generated on the basis of a number of data evaluations. An old version of the ENDF/B file (thought to be ENDF/B-II) is generally utilized as a starting point in generating the multigroup cross sections. The principal modifications to the ENDF/B data at KFA are

a) Utilization of the detailed resonance information of J.J. Schmidt[14]

b) Utilization of the thermal data for Th-232, U-233, U-235 and U-238 from GA

c) Utilization of the fission product data from the KFA libraries, and

d) Utilization of the scattering matrix for graphite as generated using the Young phonon spectrum.[15]

Multigroup cross sections for approximately 200 materials are available for use in the Pebble Bed design analysis.

3.7.3 TESTING OF THE DESIGN METHODS

Testing of the Pebble Bed design codes and data sets has been carried out for several of the KAHTER critical assemblies and for the AVR Experimental Nuclear Power Station. However, only limited results of the testing program have been obtained to date. The Monte Carlo code KENO-II has been used to predict the reactivity of the KAHTER facility with good accuracy. Similarly, the dose rates in the upper reflector of the KAHTER assembly have been predicted with good accuracy using the Monte Carlo methods. However, differences between calculations and measurements, on the order of 4% in critical mass, have been obtained when design methods have been used to calculate some KAHTER critical experiments.[16] Part of the problem is attributed to the difficulty in modelling the KAHTER facility in two dimensions. Work has continued to determine what computer model changes should be made to reduce the differences.

The results of testing the Pebble Bed design codes against the AVR measurements have been reported as giving good agreement between the calculated and measured eigenvalue. However, in testing the burnup behavior in the AVR it was noted that the codes did not accurately predict the changes in reactivity. It was not clear which codes were used in the AVR calculations, and additional information will be required to evaluate this testing program.

3.8 REACTOR EVALUATION

This section contains an evaluation of the reactor components and systems. Areas which are evaluated include core internals, reactivity control system, fuel elements, nuclear and thermal performance, analysis methods, and fuel element flow through the pebble bed. Specific areas of concern, from either a technological or licensing viewpoint, are discussed. In general, reactor components and systems in large pebble bed reactor designs in Germany are characterized by a highly developed state of technology. For most of the areas of concern discussed below, programs are in place or planned in Germany to provide the required technological information or improvements. In no case was an area uncovered in this evaluation which would conclusively prohibit the eventual successful operation of a large pebble bed reactor.

3.8.1 CORE INTERNALS EVALUATION

The cylindrical configuration of the pebble bed core is maintained by the cylindrical core container assembled from graphite blocks enclosed by an external cast iron thermal shield. In addition to serving as a thermal shield, the container utilizes the graphite blocks to serve as a neutron reflector, and the entire assembly is also used to guide the flow of helium through the reactor. While the size of the container is markedly larger than other previous gas cooled reactors, its design and configuration are in substantial measure based on successful engineering practices developed for the operational AVR reactor and the THTR plant currently under construction. (Refer to Figure 3-23 below and to Table 3-11.) In addition, considerable laboratory experimental data has been gained which supports the core design methods. However, several aspects of the core container represent new situations due either to its large size and configuration or its operating conditions. Such topics are discussed below and are considered to require further engineering study and evaluation.

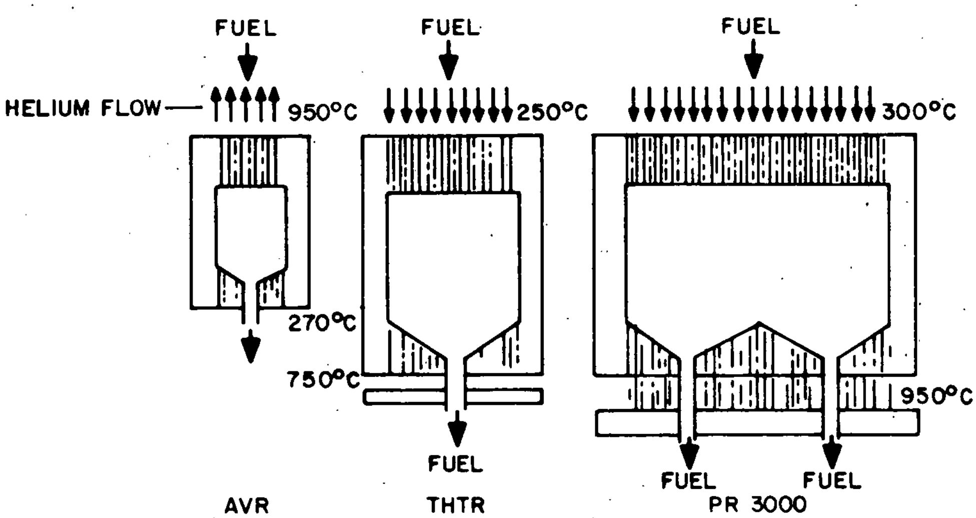


Figure 3-23. Core Container Scaling

TABLE 3-11

SIZE COMPARISON FOR PEBBLE BED REACTORS

Parameter	Units	AVR	THTR	PNP
Thermal power	MW	46	750	3000
Core power density	$\frac{MW}{m^3}$	2.5	6	5.5
Gas ΔT	^{o}C	$270 \rightarrow 950$	$250 \rightarrow 750$	$300 \rightarrow 950$
Helium pressure	bar	10	40	40
Number of fuel elements	-	100 000	675 000	$3 \cdot 10^6$
Core diameter	m	3	5.6	11.2
Core height	m	3	5	5.5
Number of discharge tubes	-	1	1	6
Fuel handling	-	6 passes through core	6 passes through core	OTTO
Temperature at core bottom	^{o}C	270	750	950
Max. fast dose (30 years) on side reflector (E>0.1 MeV)	$\overline{cm^3}$	$<1.5 \cdot 10^{22}$	$3 \cdot 10^{22}$	$5 \cdot 10^{22}$
Max. fast dose (30 years) on core bottom (E>0.1 MeV)	$\frac{u}{cm^3}$	$<1 \cdot 10^{22}$	$3 \cdot 10^{22}$	$5 \cdot 10^{21}$

3.8.1.1 HTR-K Thermal Shield Cooling

The HTR-K reactor uses separate inlet and outlet ducts for each of the six primary helium circuits. In this design, cool helium enters the reactor near the top of the core and passes directly downward through the pebble bed; only a secondary flow stream is diverted to cool the thermal shield. While the size of this cooling stream cannot be readily determined from available information, it is judged to be a small fraction of the total flow, since the forcing pressure drop is simply that which exists across the thermal shield inlet duct and port. It is doubtful that such a stream can adequately cool the cast-iron thermal shield. As a consequence, it may be necessary to use a more expensive thermal shield design or material, or even to change the primary helium flow path to provide more cooling to the thermal shield. This latter possibility might be accomplished by using the PNP duct design which employs coaxial ducts near the bottom of the core to introduce and remove helium. Available PNP data indicates that approximately 80% of the helium flow passes through the shield to provide cooling in the PNP design, and there is no apparent reason in the available data that precludes use of this arrangement for the HTR-K plant also. In any case, additional analysis appears required before adequate cooling for the HTR-K thermal shield can be demonstrated.

3.8.1.2 Core Gas Duct Attachment

In one of the designs for attaching the gas ducts to the core container, a bolted metal flange is used to join the duct to the graphite blocks. The metallic bolts affixing the flange are inserted in tapped holes in the graphite blocks. There is experience showing that graphite can be bored and tapped for use with such metallic bolts. However, it is not equally clear for long-term reactor operation under the conditions expected in the large 3000 MWth pebble bed core that reliable operation can be expected for the life of the plant. Further testing is needed to demonstrate that flow-induced vibrations, mismatch of thermal expansion coefficients, possible corrosive effects of impurities in the helium, or even manufacturing uncertainties do not lead to premature failure of the duct connections. Further evaluation, now under way, is needed to finalize a reliable design.

3.8.1.3 Lateral Core Support

It is clear that the vertical forces associated with the weight of the core are borne and transferred to the PCRV by columns located below the core. However, less information is available regarding how lateral or radial support is provided to the core container. Such radial support is needed to position and, if necessary, to restrain the entire core assembly during a seismic disturbance, for instance. Without radial support, very high and potentially damaging loads can be applied to joints in the helium gas ducting. In the backup PNP core bottom design, a central pedestal and underlying grid of structural members are used to restrain lateral core motion and provide support. However, no such arrangement is evident in the HTR-K core base. Instead, the six spent fuel discharge pipes are to be stiffened and used to provide lateral support through base plates located under the bottom graphite blocks. Furthermore, the overall core support design is to provide for an inspection/repair space of approximately 1 m between the core container and PCRV liner. Such arrangements appear feasible, but additional analysis and experimental data are needed to demonstrate satisfactory lateral support for the heavy, large-diameter, 3000 MWth core container.

3.8.1.4 Side Reflector Spallation

The region of the side reflector adjacent to the upper region of the pebble bed is subject to neutron-induced graphite irradiation with subsequent spallation. In the worst case, stresses produced by growth and distortion of the graphite blocks may lead to late-in-life surface cracking and breakage with pieces of graphite actually being separated from the reflector. Because of this potential problem, substantial engineering analysis has been applied in the Federal Republic of Germany with the conclusion that, by using a properly chosen reflector design, the spalling can be tolerated.

The problem, however, is complex, and further information is needed on irradiation effects for graphite. The reflector is designed for a nominal life of 40 years with a plant utilization factor of 0.8. For an OTTO cycle core with an average power density of 5.5 MWth/m^3 and a normal converter-thorium cycle, the reflector dose is shown in Figure 3-12, based on 32 years of full power operation. Based on the estimate described previously in Section 3.3, it is concluded that radiation-induced spallation would not become a problem during the first 27 to 28 years of equivalent full power operation. In the remainder of core life some graphite spallation could be anticipated; however, it is expected to be limited in depth to within 5 cm of the surface, since the dose drops off exponentially within the graphite. Currently, the German approach to this problem is to accommodate the spalling rather than to try to preclude it, since they consider that the safety, stability, and reliability of the graphite blocks is unaffected. Thus, consideration is being given to machining small grooves in the reflector surface to limit the size of graphite pieces caused by spalling. Further, it is planne to build the inner reflector blocks so that they can be replaced if necessary. Finally, the size of the helium purification system will be selected to provide added capacity to remove spalling-related dust during the last part of core life.

For potential applications in the United States, further evaluation is necessary before the German solution can be adopted. For example, experiments would be necessary to demonstrate that spalling occurs in the manner anticipated and that the sizes of the pieces can be controlled. A renewable reflector design must be prepared and its cost compared with that of a non-spalling reflector. Included in this analysis must be the cost of the added purification system capacity needed to remove the additional dust potentially introduced by graphite spalling. Currently in Germany a graphite irradiation program is under way that will provide additional information useful for the continuing evaluation of side reflector spallation.

3.8.1.5 Bottom Reflector Design

The reference core bottom design for the 3000 MWth reactor consists of six adjacent conical shaped regions, each containing a central fuel discharge tube. Each of the six regions is individually similar to THTR core bottom design and uses the same sort of hexagonal graphite blocks arranged to achieve the desired conical shape. Below each graphite block is a vertical graphite column which passes upward through the exit gas plenum. Beneath the columns is a base plate that underlies the whole core and is used to position and support the columns so that large gaps do not form between blocks. While this design appears to be feasible, it should be tested on a large scale, duplicating operational temperatures and core-related forces. In addition, the behavior of the core base assembly should be modelled and tested to determine its response to seismic disturbances, since it must remain functional for the removal of after-heat during reactor shutdowns.

Considerable effort has been devoted in Germany to preparing an improved backup core base and bottom reflector design. While definite advantages appear possible, the new backup design which uses a central fuel ball flow cone may produce substantial stress levels in the graphite base blocks. This situation may occur because some of the blocks used in the backup design are rather large and have high porosity because of gas flow passages bored in the blocks. Due to the large block size and rather small ligament distances, high stresses will be difficult to avoid. Further, while the novel core support columns (Section 3.3) offer a potentially improved means of maintaining the base blocks in a close-packed array, its workability and practicality should also be demonstrated with large-scale core bottom model experiments simulating actual in-core temperatures and loads.

3.8.2 REACTIVITY CONTROL SYSTEM EVALUATION

3.8.2.1 System Evaluation

The reference HTR-K reactivity control system is physically similar to the THTR system; both use lifting core and reflector rods. However, the operations of the systems differ due to the recent FRG licensing requirement for two diverse shutdown systems. In the HTR-K system, this requirement resulted in the use of two shutdown systems with different drives for the core rods. Whether this solution will satisfy the requirement has not yet been determined. A potential advantage for this system is that the drives used are available today, but this advantage is diminished in importance by the FRG move from the HTR-K to the longer range HHT plant. Thus, the HTR-K system is, to a certain extent, an extension of the THTR system, and it could be built today. However, its licensability is still an open question.

The reference PNP reactivity control system concept is unique, with little similarity to the THTR system or other nuclear control systems. It is also quite complex due to the requirement of step insertion of the rods to prevent absorber element overheating. The uniqueness and complexity may raise a number of licensing issues. The PNP system does appear more likely to satisfy the diversity requirement than the HTR-K system, but this is still an open licensing question. Unlike the HTR-K drives, the PNP rotating rod drives are not available today nor is the KLAK system. In conclusion, the PNP system is a unique and complex one which will require much further design and testing. Also, its licensability is still an open issue.

Recent personal communication indicates that significant progress is being made toward a unified reactivity control system. This system would utilize both core and reflector rods, divided into two systems as in the HTR-K. In this concept, one system would have lifting core rods, and the other would use rotating core rods. Also, the total number of core rods would be smaller than the 198 for the HTR-K system. The system would use KLAK as an emergency shutdown system.

3.8.2.2 Component Evaluation

The pneumatically driven control rod used in the HTR-K utilizes the long-stroke piston drive of the THTR control rod. Thus, this rod was developed using much of the THTR design and testing experience. The pneumatic drive is considered developed and available today and should be acceptable to the licensing authorities due to favorable THTR experience.

The technology for a hydraulic drive is well known from other applications; however, it should be tested to assure that it will operate satisfactorily under reactor conditions. Hence, the drive is considered available but in need of testing. One drawback of the hydraulic drive is that it extends 8 m above the PCRV, and is therefore exposed to external damage. At the present time, this drive has not been accepted by the FRG licensing authorities.

Two generic issues concerning the lifting rods are the forces of insertion into the pebble bed and the quality of position indication. The first issue has been resolved to the satisfaction of the three FRG companies through scaled experiments done by HRB. The experiments were performed in 1:6 scale model with 198 rods and a bed of graphite spheres. The analysis of these tests indicated that the insertion forces were similar to those of THTR and that the forces were within acceptable limits. The issue of rod position indication still remains open, especially for the pneumatic drive. Developmental work is being done in this area by GHT.

The rotating rod with a spindle drive has been completely designed, and several components of this drive have been tested--in particular, the spindle and spindle nut, the bearings, and the penetration through the PCRV for the drive shaft. One complete rotating control rod has been ordered for a feasibility test which is scheduled to begin in 1979 at KFA. Coincident with this feasibility test, a detailed design of a prototype rotating rod will begin. The prototype design is scheduled for testing during 1981. Thus, further design and a good deal of testing are scheduled for this rod design, which would be completed at the earliest by 1982. The rotating rod has a major advantage over the lifting rods because its use results in no net compression forces in the pebble bed, and it does not require the use of ammonia injection. Also, the rotating rod has no difficulty in achieving the required accuracy in position indication.

The KLAK system proposed for the PNP second shutdown system is similar to an emergency shutdown system used in the GA HTGR. In the GA system, small, high-absorbing balls held in containers are released to fill channels in the prismatic blocks. If it is available, data from the development of the GA system would be useful in developing the KLAK system. At present, the KLAK system is in an early design stage. The inlet system has been considered; however, a means of modifying the core bottom to facilitate removal of the KLAK has not been considered in detail. The behavior of the KLAK balls in the pebble bed has been studied. Also, the earthquake behavior of KLAK is being studied at the University of Aachen. However, the entire KLAK system will have to be tested once it is completely designed.

The development needs of the KLAK system would be significantly changed if the system were used as an emergency shutdown system with a probability of use of approximately 10^{-6} per reactor year. Using the KLAK in this manner has been proposed for the unified control concept. Due to the low probability of usage, no special modifications of the core bottom for removal of the KLAK would be required. Also, complete periodic testing of the system would not be needed as it would be if the KLAK system was used as a second shutdown system. Elimination of the requirement for complete testing of the system could be an advantage in overall plant availability.

3.8.2.3 Control Rod Materials

One major materials concern is whether the proposed metallic structural components will retain the necessary toughness, ductility, and strength throughout the expected service life. These properties can be significantly degraded by long exposures at elevated temperatures in gas reactor environments. Also of concern are potential galling, seizing, and self-welding that can occur at points of metal-to-metal contact. Probably most critical in this context are seals, especially sliding seals with long service life.

When the control rods are inserted into the core, they will be exposed to high temperatures. It is well known that most alloys lose toughness and ductility after extended exposure at elevated temperatures, particularly between 550-950°C. Strength may be either increased or decreased due to such exposures. Of the proposed top candidate alloys, Incoloy 800 should be affected the least. In this regard, Inconel 625, although considerably stronger than Incoloy 800, suffers considerable loss of toughness and ductility (particularly lower temperature toughness and ductility - below about 450°C after exposure between 650-800°C), and the properties of Inconel 519 would be expected to fall somewhere between those of Incoloy 800 and Inconel 625.

Most alloys of the above types appear to carburize extensively due to elevated temperature exposure in "reactor purity helium" (helium with very low levels of oxygen and low levels of H_2, H_2O, CO, CO_2, and CH_4). This effect is apparently particularly significant at temperatures above about 750°C. Extensive carburization can be expected to significantly reduce toughness and ductility, particularly at temperatures below about 450°C; and it will probably also reduce strength (particularly low cycle fatigue strength). Of the alloys previously mentioned, Inconel 625 appears least susceptible to carburization in HTR-type environments. All should be satisfactory, provided exposure to temperature above 700°C is kept to a minimum, as is the intent of the present design.

Neutron exposure, particularly to fast neutrons, also results in loss of toughness and ductility. Little information is presently available on the effect of the fluence and energy spectrum expected in pebble bed reactors, particularly at temperatures above 700°C, for alloys of interest.

Because of the low oxygen potential (or more simply, very low ratio of oxidizing to reducing species) of reactor-purity helium, the types of oxides formed on most alloys at high temperatures in air are not stable. For this reason most contact between alloys will be actual metal-to-metal contact which, under conditions of high temperature, high contact stresses, and long times, can result in self-welding. When relative motion is required under such conditions, severe galling or seizing (and self-welding) is possible.

All the proposed control rod structural materials are susceptible to these problems; where conditions conducive to self-welding or galling are foreseen, protection with wear-resistant materials (tested for compatibility with these service conditions) will be required.

Available information indicates that the German partners have recognized these problems. They have an ongoing materials development program in which these problems are being addressed. At present, the materials testing only simulates the helium and temperature conditions; however, a search is under way for a facility where irradiation tests can be performed.

3.8.2.4 Nuclear Instrumentation

Most of the information available on the subject of the nuclear instrumentation system indicated that the design was still in the conceptual stage with much work yet to be done. However, recent personal communication has revealed that more detailed work has been done in selecting the number, type, and locations of flux detectors used. Also, the basic system concept using ex-core detectors remains unchanged.

The German companies involved argue that in-core instrumentation is not needed because the fuel operates far below its technical specifications and well below its physical limits (such as the melting point of the fuel). Also, it is pointed out that only in hypothetical accidents will the fuel overheat. Even then, the only result is an increase in fission product release; the fuel elements remain intact. Given these circumstances, the companies believe that ex-core instrumentation will provide sufficiently detailed information for safe operation of the reactor. However, this system has not yet been approved by the German licensing authorities for the 3000 MWth core. A similar system was approved for the 750 MWth THTR core, but the THTR does not have the potential for xenon oscillations that the 3000 MWth core does. Therefore, licensability in Germany is not yet resolved. It should also be mentioned that U.S. licensability is not clear. The experience in the U.S. has been that new design commercial cores have been required to have in-core instrumentation. The acceptability of the German design cannot be resolved without an actual review by the U.S. Nuclear Regulatory Commission.

3.8.3 FUEL ELEMENT EVALUATION

Fuel elements for the HTR-K and PNP reactors are probably the most qualified major component of the reactor system. Thousands of elements of the THTR type, which is the reference design for the HTR-K and PNP plants, have been tested in the AVR. In addition, many fuel element tests have been carried out in different test reactors in Europe. The similarity between the fuel elements and their environment in the THTR and in the PNP was shown in Table 3-8. There is little doubt that the fuel system will be fully qualified by the time the first 3000 MWth plant is built.

Section 3.5 summarized the Pebble Bed Reactor fuel system. A detailed description of the system is the subject of another report.[21] The results of the decades of fuel development, the ten years' operation of the AVR, and the ongoing work in Germany form the basis for the confidence placed in the fuel elements.

Without contradicting the above very positive statements, there are several areas of uncertainty which will have to receive attention during future reactor development work.

- If nuclear proliferation arguments require a change from the present THTR fuel (highly enriched uranium mixed with thorium as an oxide), the effect of a new fuel form will have to be evaluated. Tests of elements containing 20% enriched uranium are now under way.

- Present testing to evaluate the limits of performance, primarily under postulated accident conditions, must present no significant unexpected surprises. Many tests with much higher specifications than 1250^{o}C for the coated particles, however, have already been performed with good results.

- Any changes in element composition due to manufacturing development must not degrade the present excellent performance.

As reported in Section 3.5, the future advanced development potential of the fuel element is high. Given the necessary economic and conservation incentives, advanced fuel cycles can be developed which have very high conversion ratios, up to net breeding. Development of TRISO particles can raise the exit gas temperature several hundred degrees, provided that heat transfer equipment can be built to use the higher gas temperature.

3.8.4 NUCLEAR AND THERMAL PERFORMANCE EVALUATION

All developmental work performed to date indicates that the system can meet the thermal and nuclear performance requirements. The remaining area of concern rests with the fact that no large OTTO cycle reactor has yet been built. However, the successful operation of the AVR for ten years and the considerable design, analysis, and testing which has been performed provide a substantial basis for designing to meet nuclear and thermal performance criteria. In addition, the THTR, which is nearing completion, will contribute further to the testing and understanding of these performance characteristics.

The introduction of the OTTO cycle, while improving the thermal performance of the reactor, represents a change from present-day fuel cycles. At present, the fuel elements are passed through the core several times, and the in-core residence time and the percentage burnup for each pass-through are less than those in the OTTO cycle. Thus, in the OTTO cycle, whereby the fuel element is depleted to the maximum extent in a single pass through the core, it is particularly important that analysis methods accurately calculate accumulated fuel depletion and fission product history in order to properly calculate fuel power density and temperature within the pebble bed. It appears, as shown in Section 3.7, that these methods are well developed for the most part, but some additional work is required to qualify the fuel burnup calculation. For the HTR-K, with its low exit gas temperature, and steam cycle power conversion equipment, there is reasonable confidence that satisfactory operations can be achieved. In the HTR-K case, there is much greater thermal margin than for the PNP plant. This is evident because the maximum fuel temperature specification limit is 1250°C for both plants, and the maximum calculated fuel temperature is 1020°C for the PNP plant and 820°C for the HTR-K case. Thus, the PNP margin of 230°C is only half that of the HTR-K plant. In addition, the temperature-sensitive PNP chemical process plant equipment requires a relatively uniform reactor gas exit temperature with variations to be held within only ±30°C (Table 3-3). For the PNP plant, additional work is needed to assure that the requirement on temperature uniformity can be met. Likewise, continuing work on verifying the actual fuel particle temperature in an OTTO cycle element should continue, especially for the PNP plant conditions.

It will be necessary to prove that out-of-core instrumentation can measure the flux and power shape. Present work indicates that this is the case, but it may be necessary to prove the method in an operating reactor. It is planned to do this in the PNP-prototype reactor. The licensing of a large plant with only ex-core instrumentation to detect power oscillation is an open issue.

In summary, there is high probability that the required thermal-nuclear performance can be achieved. The present German program, if continued at the appropriate level, is sufficient.

3.8.5 EVALUATION OF ANALYSIS METHODS

The design methods at HRB, GHT, and KFA have been under development since the middle 1960's, and they have incorporated many of the design codes which were originally developed in the United States (e.g., the three-dimensional diffusion code CITATION, the two-dimensional transport code DOT, the Monte Carlo code KENO-II, etc.). Particular attention in Germany has been given to the adaptation of the codes to the Pebble Bed system. Thus, the old thermal reactor codes GAM and THERMOS have been modified to include the heterogeneity calculations for the Pebble Bed fuel kernels and fuel ball matrix. In general, the design methods are highly developed, and each of the German design groups has considerable experience in the utilization of the methods for carrying out the Pebble Bed analysis.

A modular code package system has been set up for carrying out the core design and fuel cycle studies. This system for linking together the principal design codes provides a considerable degree of flexibility in carrying out a design study. However, the code systems are both cumbersome and expensive to run on a computer. The costs of running the linked code system are significantly reduced for scoping studies in which the depletion calculations are limited to the equilibrium operating cycle. The modular code systems have been specified to give a trade-off between accuracy and cost. This trade-off is accomplished primarily in the neutronics module of the linked system, where two-dimensional synthesis techniques are used to compute the flux solution. In comparing this approach to the direct two-dimensional computations, it is concluded that the synthesis technique is adequate over the Pebble Bed core regions. However, the synthesis techniques can be expected to result in large errors in the calculation of the fast neutron flux in the graphite radial reflectors. For this reason, the direct two-dimensional calculations are used to predict the flux behavior in the reflectors.

Two features of the Pebble Bed design are particularly difficult to model in the neutronics calculations. First, the effects of the void between the core and the upper axial reflector are difficult to model using diffusion theory, and because of the proximity of this region to the region of high neutron flux in the core, the effects of this void region are important. The control effects are also difficult to model because of the large number of rods and because they are frequently utilized in the core regions where the flux gradients are large. These modelling problems have been examined in detail using two-dimensional transport theory, three-dimensional diffusion theory, Monte Carlo methods, and response matrix techniques. In general, the approximations which are used in the void and control regions have been found to give good agreement with the results of calculations using the more sophisticated design methods.

The data processing methods at KFA are highly developed and can be utilized in the Pebble Bed analysis with a high degree of confidence. The HRB design group utilizes a Russian f-factor approach (see Section 3.7.1.1) for processing the data in the epithermal and fast energy ranges. Experience with this approach has indicated that it may give poor values for the data in the energy range where the narrow resonance approximation is not good. However, to evaluate the adequacy of this technique, direct comparisons with the KFA methods must be made.

The design methods which are used in the Pebble Bed analysis represent, for the most part, a high degree of development effort, and are generally adequate for most design studies. Great care must be exercised in application of the methods to Pebble Bed systems, in which the three-dimensional modelling effects are important. In addition, testing of the burnup routines by comparing calculations and measurements in the AVR has indicated that some caution should be taken in interpreting the burnup results, and therefore, more effort should be applied in qualifying the burnup calculation. Probably the limiting factors in the utilization of the methods are complexity of the input specifications and the computational costs. The work presently being carried out by LASL to improve the efficiency of the VSOP code has indicated that some cost savings may be realized. However, this work has not yet been completed.

In conducting the evaluation of the HRB, GHT, and KFA analysis methods, only a limited review was made of the comparisons of the calculations and measurements in the KAHTER criticals and the AVR experimental facility. A detailed evaluation of the results of these benchmark test calculations should be included in any further evaluations of the design methods.

3.8.6 EVALUATION OF FUEL ELEMENT FLOW

The 1:6 scale testing of the flow behavior with the reference core bottom indicates that the flow is sufficiently uniform to achieve the requirement that variation in the gas exit temperature profile shall not exceed $\pm 30^{\circ}C$. Although the 1:6 scale results indicate that flow was sufficiently uniform, a full-scale test of one-sixth of the core may be necessary for licensing purposes in the U.S. (See Reference No. 22).

Flow behavior with the backup core bottom design has been investigated in 1:20 scale; however, these tests did not yield sufficient information to determine the uniformity of the flow of fuel elements. Hence, further testing is required to obtain conclusive results and to compare the uniformity of fuel flow to that obtained with the reference design. One potential problem is with the backup design of the exit tubes below the core bottom where the 12 tubes are interconnected to form 6 exit tubes, which then are interconnected into 3 exit chutes; past HRB experience has shown that such interconnections have an adverse effect on the uniformity of ball flow.

3.9 CONCLUSIONS

No single problem or group of problems is seen at this time to preclude the orderly development of a large 3000 MWth reactor plant. Much work, however, remains necessary in order to resolve open areas of concern. Typical of these areas of concern are the following:

- The concern over the likelihood and seriousness of graphite spalling of the upper side reflector must be resolved.
- Workable and reliable control rod drive and rod position indication systems must be demonstrated for high in-core temperatures of the PNP plant.
- Confirmatory analysis and full agreement over the acceptability of not using in-core nuclear instruments must be obtained.

Resolution of these issues, together with what is considered to be a well established fuel technology, are judged to provide a satisfactory basis for the continued development of an operational 3000 MWth Pebble Bed reactor plant.

SECTION 4

PRIMARY SYSTEM COMPONENTS

The following material is provided to describe in detail the primary loop components of the systems described in Section 2 and to evaluate the component technology base and development requirements and identify possible potential problems. The components are treated as separate entities rather than as parts of a specific plant system because of the basic similarity of the component functions. Differences between various plant arrangements that can significantly influence the component design have been identified and discussed in the text.

Table 4-1 provides an overall listing of major characteristics for the primary loop components.

4.1 PRIMARY CIRCULATORS

4.1.1 PRIMARY CIRCULATOR DESCRIPTION

The primary circulators provide the energy to move primary-loop helium coolant at the required flow rate and overcome the flow resistance of the various primary loop components. Circulator power is directly proportional to the volumetric flow rate and pressure rise required. Since this power is a net loss to the plant output, there is incentive to minimize the pressure loss in the various components. Since volume flow varies inversely with system pressure, there is advantage in reducing circulator size by increasing system pressure. This must be balanced against other factors, however, such as reactor vessel strength requirements. In order to minimize volume flow and also to provide a reasonable temperature environment, the circulator is located at the region of lowest temperature in the primary loop. In the PNP plant arrangements for hydrogasification (Figure 2-7) and steam gasification, Alternate A (Figure 2-13), the circulators are located in the bottom region of

TABLE 4-1

COMPONENT GENERAL CHARACTERISTICS

COMPONENTS	Unit	PNP HKV	PNP WKV	HTR-K
Reactor Pressure Vessel				
Type		PCRV		
Construction		multiple cavity		
Operating pressure	bar	40 to 42		60
Design pressure	bar	46.2		66
Concrete temperature	^oC	66		66
Overall dimensions				
height	m	31	35	31.6
diameter	m	44	46	37.4
Core cavity				
height	m	17	17	15.4
diameter	m	16.4	18	16.3
Cavities for components:				
Steam reformer			NA	NA
number		6		
diameter	m	4.8		
He/He heat exchanger		NA		NA
number			24/12*	
diameter	m		3.25/5.5*	
Hot gas distributor		NA	3	NA
number			6	
diameter	m		4	
Steam generator			NA	
number		6		6
diameter	m	4		5
Process gas pipeline			NA	NA
number		6		
diameter	m	1.8		

*First entry is for helical design; second entry is for U-tube design
NA = not applicable

Table 4-1 (Cont'd.)

After-heat removal system	Unit	PNP HKV	PNP WKV	HTR-K
number		4	4	4
diameter	m	3	3.8	3.06
Circulator				
Number		6		
Type		radial		
Drive		electric motor		
Helium temperature (at exhaust)	°C	300		260
Helium pressure (at exhaust)	bar	40		60
Flow rate	kg/s	148		220
Pressure rise	bar	1.3		1.3
Power at motor terminals	MW	8		7
Steam Reformer			NA	NA
number		6		
type		counter flow with inside return		
Power derived from helium	MWth	115.3		
Heat regained from reformed gas within the steam reformer	MWth	21.5		
Total power for reforming	MWth	136.8		
Helium mass flow rate	kg/s	148		
Helium inlet temperature	°C	950		
Helium outlet temperature	°C	800		
Process gas flow rate	kg/s	58		
Process gas inlet temperature	°C	500		
Process gas maximum temperature	°C	810		
Process gas outlet temperature	°C	680		
Process gas inlet pressure	bar	44		
Process gas outlet pressure	bar	40		

Table 4-1 (Cont'd.)

	Unit	HKV	WKV		HTR-K
He/He Heat Exchanger		NA			NA
Construction			Helical Counter Flow	U-Tube Counter Flow	
Number			24	12	
Thermal power	MW		125	250	
Flow rate	kg/s		37/36.3+	74/73+	
Inlet temperature	^{o}C		950/240+	950/240+	
Exit temperature	^{o}C		300/900	300/900	
Operating pressure	bar		40/42	40/42	
Design pressure	bar		45	45	

	Unit	HKV	WKV	HTR-K
Steam Generator		6	not available	6
Number		helical & straight tube bundles	not available	helical & straight tube bundles
Flow		parallel in straight, counter in helical	counter	parallel in straight, counter in helical
Thermal power per unit	MW	385	not available	504
Total thermal power	MW	2308	2033	3020
Helium flow rate	kg/s	148.1	not available	220
inlet temperature	^{o}C	800	687	700
outlet temperature	^{o}C	300	240	262
Water/steam flow rate	kg/s	143.3	not available	201
inlet temperature	^{o}C	180	150	185
outlet temperature	^{o}C	540	540	515
outlet pressure	bar	115	185	175

the PCRV. This facilitates duct design, allows access to the other major components without removing the circulators, and permits ready removal of the circulators. In the HTR-K design (Figure 2-1) and in the steam gasification plant, Alternate B (Figure 2-15), the circulators are located above the heat exchangers, reportedly to simplify the total design.

Satisfying these two alternative mounting arrangements has resulted in the two circulator designs shown in Figures 4-1 and 4-2. The circulators have electrical drive motors, oil supply, cooling water supply, and internal sealing gas all fully integrated within their lined PCRV cavities. The drive motor is constant speed, and the circular inlet has a control device for flow control and shutdown reverse flow limitation. Main parameters are shown in Table 4-2.

TABLE 4-2

PNP AND HTR-K CIRCULATOR DATA

Characteristic	Unit	PNP	HTR-K
Number required	--	6	6
Coolant/molecular weight	--	He/4	He/4
Helium mass flow	kg/s	148	220
Volume flow	m^3/s	48	44
Operating pressure	b	40	60
Inlet temp.	^oC	300	260
Static Pressure Rise	b	1.3	1.3
Motor power	MW	$\sim$8.0	$\sim$7.1
Motor speed	RPM	2950	NA
Regulation	--	60-100%	70-100%

The basic circulator design can accommodate either a "suspended" or a "standing" arrangement inside the lined cavity. The "standing" arrangement locates the impeller wheel at the upper end of the vertical shaft, as shown in Figure 4-1. Circulator installation into the lined cavity is accomplished by sliding the unit into position from the bottom end of the cavity. On the other hand, the "suspended" arrangement (Figure 4-2) locates the impeller wheel at the bottom end of the vertical shaft. Circulator installation in this arrangement is accomplished by sliding the unit into position from the top end of the cavity.

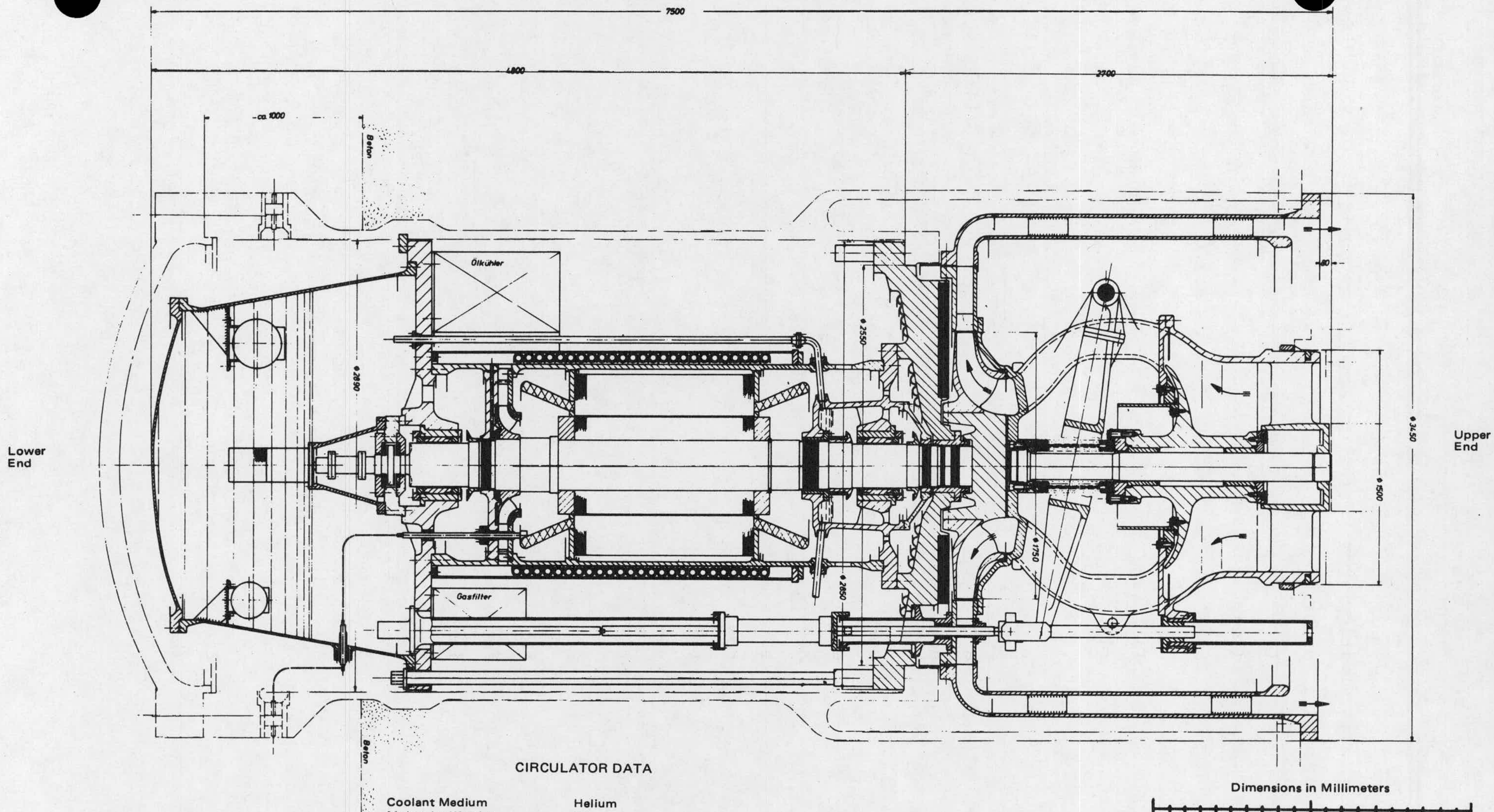
The following descriptions are specific to the "standing" type of circulator construction. Variations for the "suspended" type of construction can be seen in Figure 4-2.

The primary circuit impeller is designed as a single-stage radial compressor discharging into a bladed diffuser. From the diffuser, flow discharges to an annular duct to a dump plenum.

The impeller/diffuser and the drive motor with cooler and shutoff control device at the inlet-side (with its own actuator) are assembled in a steel cavity in the reactor vessel. The lined cavity is hermetically sealed by a gas-tight cover plate. The drive motor runs in primary-loop helium atmosphere under the reactor pressure existing at the given operating condition. All lines and fittings containing cooling water, lubrication, gas, electric wiring, and instrument leads are guided through the bottom flange region of the cavity liner. The liner cover plate for the primary containment has no penetrations.

In order to limit the release of primary helium into the reactor containment building should failure of the liner cover plate occur, a flow restrictor has been placed immediately behind the impeller wheel. This flow restrictor plate is attached to the inner flange of the cavity liner.

The flow restrictor, which includes the impeller housing, is additionally secured by a second restrictor plate that provides protection against rapid blowout. An alternate is being considered to the cavity cover shown in Figure 4-1. In this alternative, a reinforced concrete cover would be used over a welded or bolted steel membrane liner. The cover would be restrained by vertical tensioning rods through the PCRV. It is considered with this construction that the flow limiter plate may not be required.



CIRCULATOR DATA

Coolant Medium	Helium
Molecular Weight	4
Helium Mass Flow Kg/S	148
Volume Flow Rate M^3/S	46
Discharge Pressure P (Bar)	40
Inlet Temp, ^oC	300
Static Pressure Rise, $\triangle P$ Bar	1.3
Envelope Radius mm	1750
Motor Power KW	8700
Motor Speed, RPM	2950
Speed Regulation	with control valve – 60-100% of mass flow in normal operation

Dimensions in Millimeters

0 1 2 m

Figure 4-1. Standing Arrangement Circulator

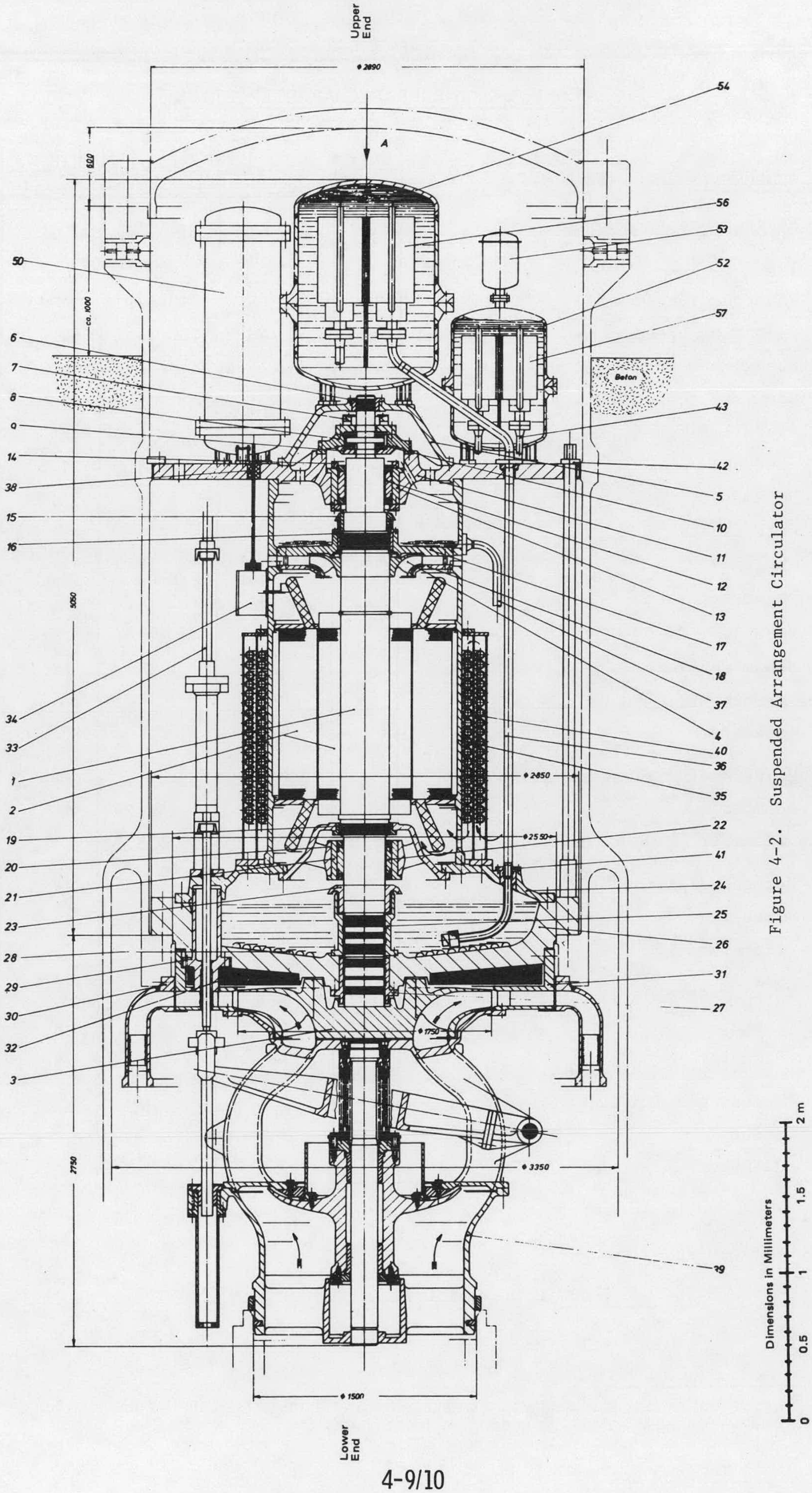


Figure 4-2. Suspended Arrangement Circulator

The perpendicularly arranged impeller shaft is guided by and positioned in oil-lubricated bearings. The vertical load is supported by a multicollar thrust bearing positioned below the circulator radial load bearing. The radial load bearings and multicollar thrust bearings are equipped with tilting segments containing Babbit metal inserts. They are suited for operating in both directions of rotation.

An oil supply system is arranged inside the slide-in circulator unit. This oil supply system provides the lubricating cooling oil for the bearings.

The drive motor of the circulator is a three-phase asynchronous motor with special deep-slot rotor. To obtain a high efficiency, low-loss lamination sheets are selected for the rotor core. The stator of the circulator motor is inserted and supported in a snug fit in the motor frame and is axially positioned and fixed by means of radial dowel pins. At the same time, these dowel pins transmit the torque. Motor cooling is similar in principle to that provided for the THTR circulators. An auxiliary radial fan wheel is arranged above the motor and maintains a cooling gas circulation through the motor and motor cooler. The motor cooler is arranged in a sheet metal casing concentric to the motor frame. The motor cooler consists of ribbed pipes which carry water to an external heat exchanger.

All the sliding bearing surfaces are supplied with oil from an oil sump located below the motor. In case of failure of the bearing oil pump, which is driven by the circulator shaft, safe shutdown of the blower has to be ensured while the impeller is coming down in speed. This is accomplished by providing a second standby oil pump designed for a continuous capability of 100% of the required pump output. This standby oil pump is located externally and is also switched on during startup and shutdown of the circulator.

To provide an effective seal against separation of the primary gas into the motor chamber, the space where the impeller shaft passes through the flow restricter plate is filled and charged with external pure helium buffer gas. The flow quantity required amounts to approximately 300 Nm^3/h.

The suction (intake) housing with flow control valve is flange-mounted to the impeller/diffuser. A piston ring is arranged at the flow inlet opening of the housing, which provides a seal between the suction side and the outlet. Axial and radial differential expansion are also accommodated by the piston ring seal.

The control actuation device is operated by means of a hydraulic cylinder and lever arrangement. The hydraulic cylinder, which is inside the cavity, is charged with high-pressure oil.

The control characteristic requirements of the throttle device are essentially the same as those imposed on the well-tested THTR circulators. For this reason, the design concept of the THTR-control device has been adopted. The circulator control device provides the following functions:

- Control of the mass flow rate between 60 and 100% of the rated mass flow at a fixed speed of approximately 2950 rpm.

- Limiting the back-flow quantity with the blower at stand-still, down to approximately 7% of the rated flow rate, while the other blowers are operating at rated flow rates.

- Throttling-down to a minimum quantity in order to facilitate the startup of one blower against five blowers which are already in operation.

The dimensions and design data for the process heat-primary circuit blower can be obtained from Figures 4-1 and 4-2. The assembly weight of one blower unit amounts to approximately 50 tonnes.

4.1.2 PRIMARY CIRCULATOR EVALUATION

The design of the circulator appears to be in the advanced stages of a preliminary design with further work awaiting decisions on the specifications on the final plant and the establishment of final product requirements. Both of these items are necessary to establish the size and operating conditions. Of particular concern will be development of the off-design and transient operating conditions for both normal and faulted operation which are used to establish inlet temperature and pressure variational conditions. FRG analysis indicates that 70-80°C overtemperature is the maximum attained during any mode of operation.

The general layout and design features of the proposed circulator are based on scaling up the design for the THTR main circulators but without the THTR variable-speed motor. The THTR circulators, while not yet proven in reactor operation, have been built and tested in a simulated environment. Similar circulators using both fixed- and variable-speed motors have been designed and operated as part of the United Kingdom CO_2-cooled Magnox and AGR Reactor Programs, at much smaller power levels, however. A summary of the THTR, Dragon, and other helium circulator characteristics is shown in Table 4-3. In addition, it is reported by the Germans that an electrically driven circulator of 11 MW has been used by Great Britain in their CO_2-cooled reactor work. Based on this experience and background, no "barrier" problems are anticipated in the development of the 3000 MW system circulators, providing that transient and off-design conditions do not significantly exceed the steady state conditions.

Similar to the circulator design for the THTR, the 3000 MW system unit is designed as a module for ease of removal and replacement. Where cavity liner diameter is not of critical concern, as in the hydrogasification design, the circulator and closure assembly may be removed as a unit. Where space is more limiting, the unit may be removed by removing the closure head, unbolting the mounting plate from the liner flange using the extended bolts provided, and lifting out the unit. Note that it is possible to separate the main blower assembly from the outlet diffuser for ease of assembly.

4.2 PRESTRESSED CONCRETE REACTOR VESSEL WITH LINERS (PCRV)

4.2.1 PCRV DESCRIPTION

The core, the heat transfer equipment, the after-heat removal systems. and the primary system circulators are all contained in a burst-proof prestressed concrete reactor vessel (PCRV). This multiple cavity PCRV is prestressed by means of axial cables and uninterrupted circumferential windings. A seismic-shock-resistant support is provided with flexible elastomer bearing pads between the primary structure and foundation. The primary structure is restrained from rotational motion.

TABLE 4-3

SUMMARY OF GAS CIRCULATOR CHARACTERISTICS

Data	Dragon	Dragon Refer.	Peach Bottom	Fort St Vrain	770 MW_e (Summit)	1160 MW_e (Fulton)	AVR	THTR
Medium	He	He	He	He	He	He	He	He
Number of circulation	6	13	2	4	4	6	2	6
Driving power, MW	0.07	1.26	-	3.88	10.8	10.8	0.128	2.5
Kind of drive	electr.	electr.	electr.	steam	steam	steam	electr.	electr.
Mass flow, kg/s	1.8	32.4	27.7	110	235	235	8.5	50.7
Rotational spaced, RPM	1100-12000	6000	-	9550	6750	6750	4400	5600
Inlet pressure, bar	8.25	40.2	24	46.4	45.9	45.9	10.2	38
Pressure rise, bar	0.45	-	-	0.96	1.43	1.43	0.083	1.22
Inlet temp., °C	350-420	325	343	395	-	330	175-330	250
Circulator design/stage	radial/1	radial/1	radial/1	axial/1	axial/1	axial/1	radial/1	radial/1
Circulator arrangement	horiz.	-	horiz.	upright	upright	upright	horiz.	horiz.
Regulation	speed	initial whirl throttle	-	speed	speed	speed	speed	bypass-frequency variation

The PCRV primary structure is generally similar for the various plants being considered but differs in detail, primarily in the number and size of penetrations required for the heat transfer equipment and the number, size, and location of interconnecting duct ways.

Examples of the various PCRV geometries considered were shown in the four figures (2-7, 2-13, 2-15, and 2-1) which depicted the primary system arrangements for hydrogasification of coal (HKV), two alternate arrangements for steam gasification of coal (WKV), and the primary system for dual-cycle steam electric power (HTR-K), respectively.

4.2.1.1 Structure and Geometry

The prestressed concrete construction has the form of an upright, right-circular cylinder, with outside dimensions, depending on plant type, of 31 to 35 m height and 36.8 to 44 m diameter, which includes the assembly for the winding process that establishes the desired prestress. Without these assembly units the diameter is decreased 1 m. The prestressed concrete construction consists of a central cylindrical cavity (core cavity) approximately 13 m in diameter and 17.0 m high. Around the core cavity are arranged the cavities for the heat transfer equipment. A representative PCRV configuration for one type of PNP plant (hydrogasification of coal), shown in Figures 4-3 through 4-6, contains the following.

- 6 cavities 4.80 m diameter for the tubular shaped steam reformers
- 6 cavities 4.00 m diameter for the steam generators
- 6 cavities 1.80 m diameter for the process gas pipelines
- 4 cavities 3.00 m diameter for the after-heat removal systems

These cavities are closed by pressure-tight top covers constructed of reinforced concrete. Various sealing systems are being evaluated. The configuration and size of the gas ducts can be seen from Figures 4-3, 4-4, and 4-5. The top head of the prestressed concrete reactor vessel, in the core region, contains up to 246 vertical passages for control rods and 48 vertical passages for charging fuel into the core. The bottom of the reference PCRV in the core region has 6 ball outlet tubes, each with a diameter of 1.20 m. Typical wall thicknesses of the concrete vessel are as follows:

bottom:	6.50 m
side portion:	12.60 m
top head:	7.50 m

All inside concrete surfaces are covered by a sealing steel liner in order to provide a leak-tight gas barrier and to provide thermal, erosion, and radiation protection for the concrete. The liner is insulated on the gas side, which is bathed with cold primary gas at approximately 300°C. On the concrete side a water cooling system (cooling channels) is welded to the liner. The water cooling channels are oriented vertically and terminate in headers at the top and bottom in groups of seven. Each of the headers is connected to the water supply in order to provide redundancy. The liner is constructed of fine-grain mild steel and is held into the concrete with 300 mm-long stud anchors resistance-welded to the liner. The liner is faced with layers of KAOWOOL™ insulation and a sheet-metal foil, all held in place with steel cover plates. The cover plates are attached with threaded fasteners attached to the liner. Redundancy is incorporated into the fastener design via welded studs to prevent loss of cover plate anchorage.

The PCRV design for the PNP plant for steam gasification of coal which uses an intermediate loop He/He heat exchanger of helical construction (Alternative A, Figure 2-13) differs from the construction discussed above mainly in the way the cavities for the equipment are arranged. Twenty-four intermediate heat exchanger cavities with diameters of 3.25 m are required. Groups of four are served from one hot gas duct and one hot gas distributor chamber via connecting ducts. The six cavities for the hot gas distributors are located on a pitch circle of 32 m diameter, each with a diameter of 4 m. Coaxial to the hot gas ducts, six annular, lined ducts return cold gas into the core cavity. The four after-heat removal systems in this case are located between the primary heat extraction systems. The size and location of the components result in an outside diameter of the PCRV of 46 m, including the prestress winding, and a height of 35 m.

The PNP concept using an intermediate loop He/He heat exchanger of U-shape tube construction (Alternative B, Figure 2-15) is of a different geometry. Twelve He/He heat exchangers are located on a pitch circle of 36 m diameter, and each requires a cavity diameter of 5.5 m. Groups of two heat exchangers

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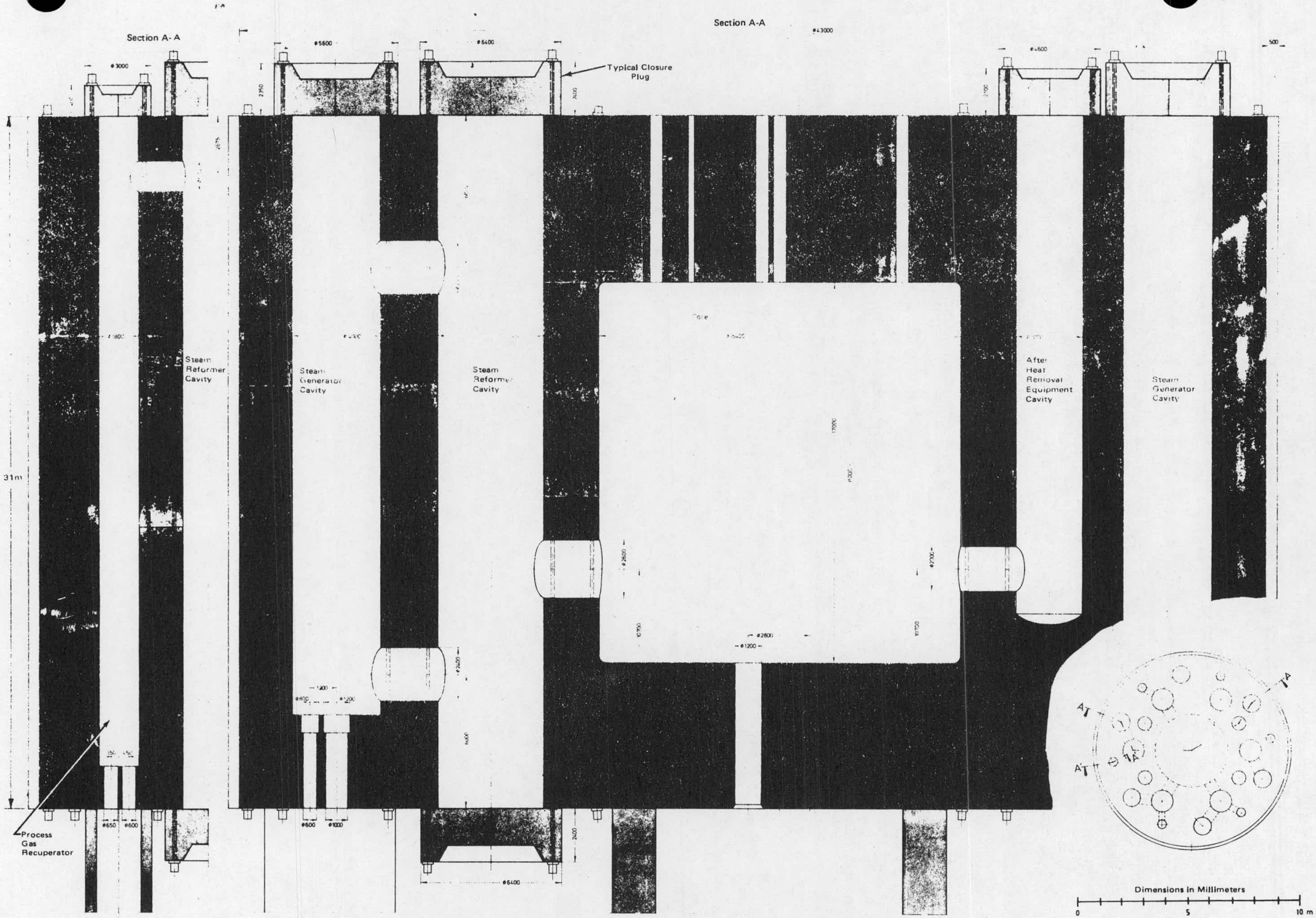


Figure 4-3. PCRV Vertical Section (HKV Plant)

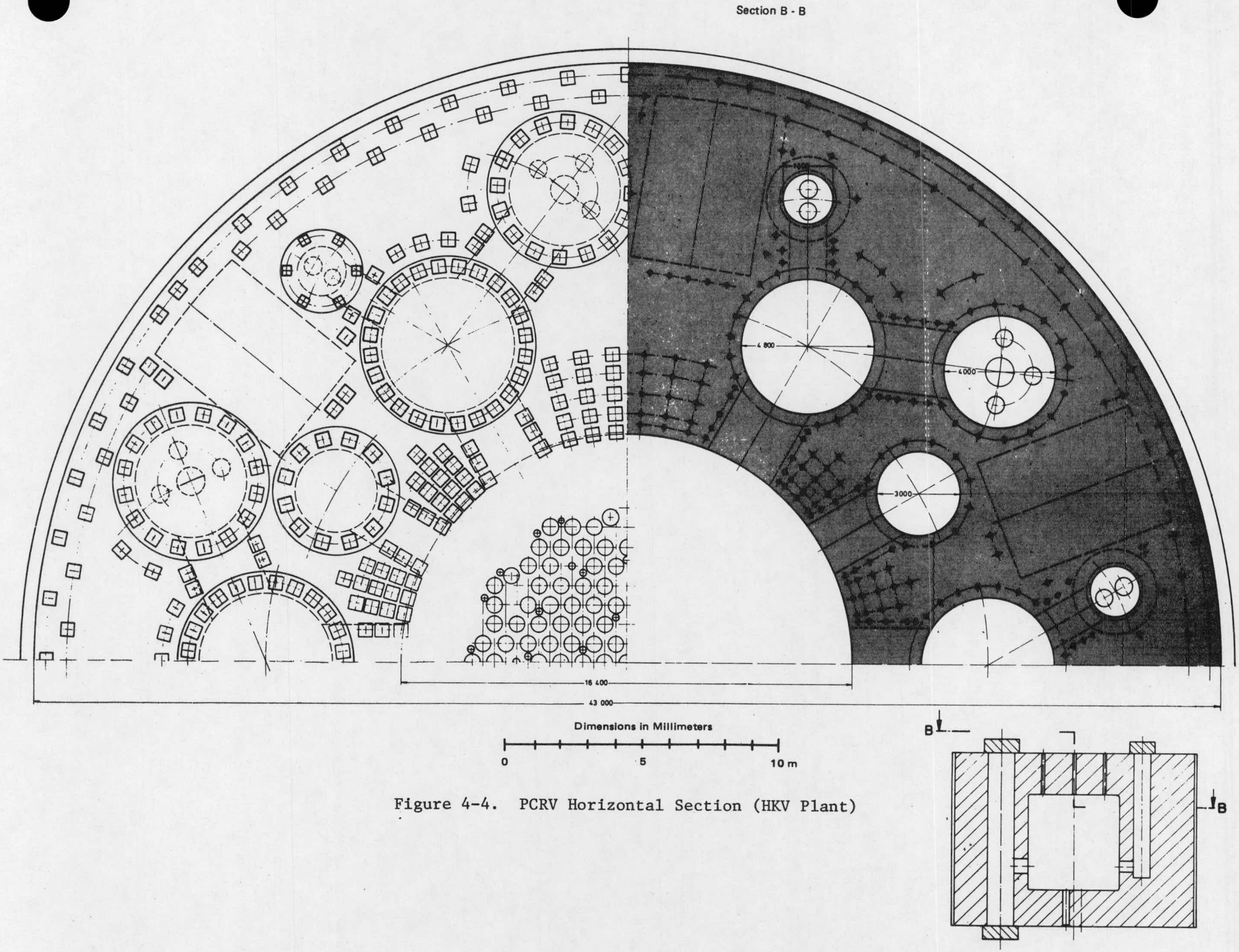


Figure 4-4. PCRV Horizontal Section (HKV Plant)

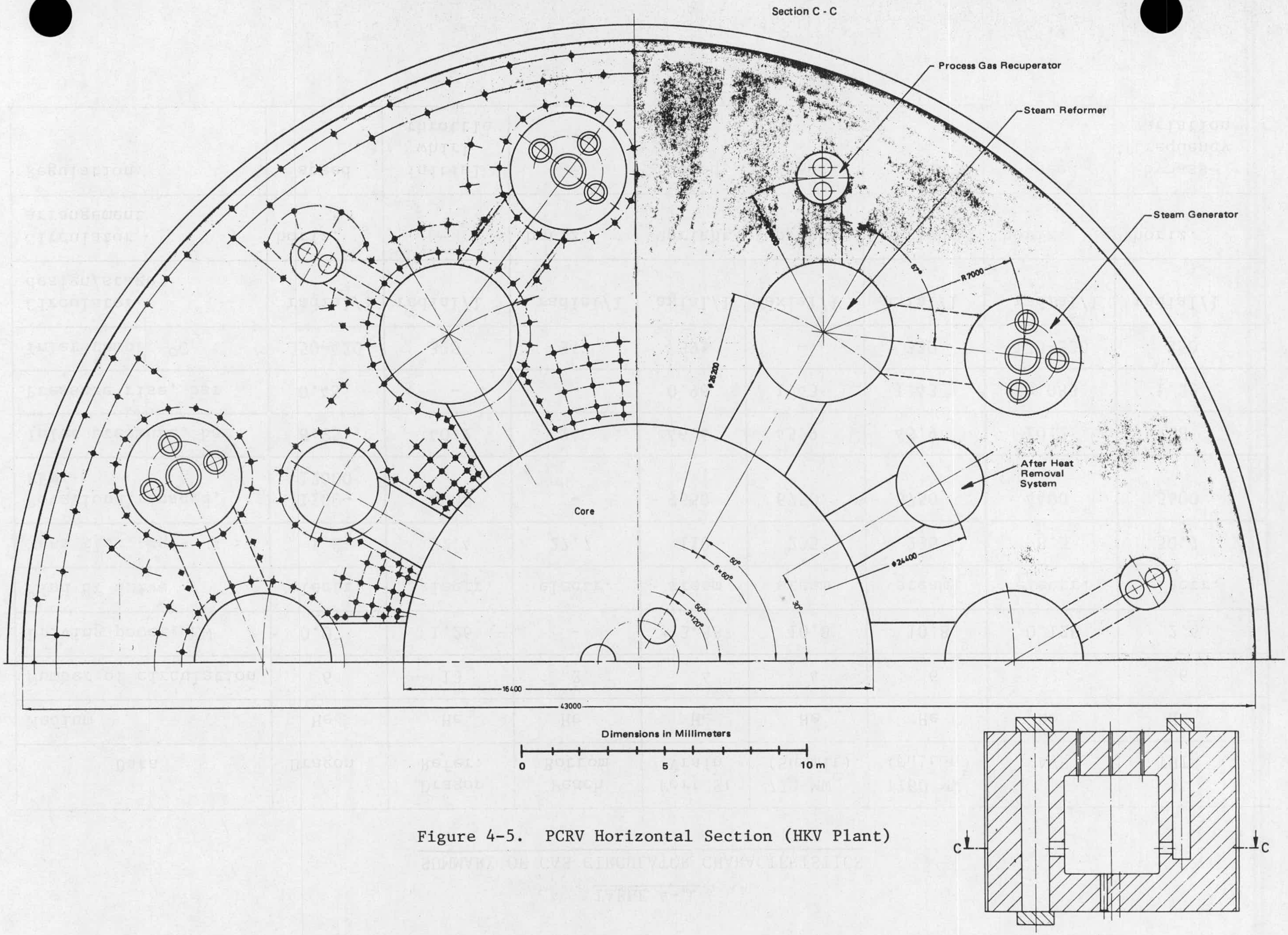
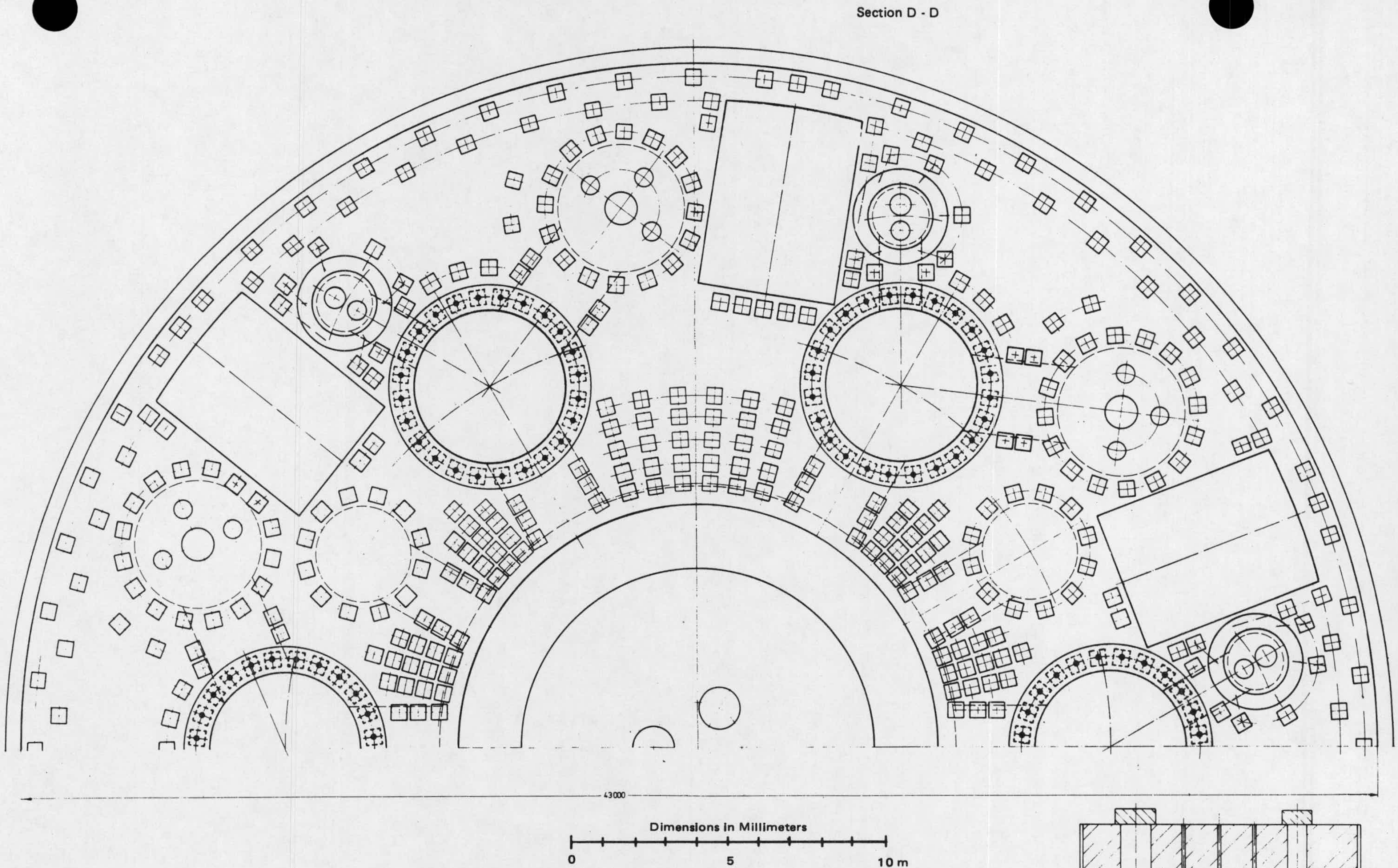


Figure 4-5. PCRV Horizontal Section (HKV Plant)



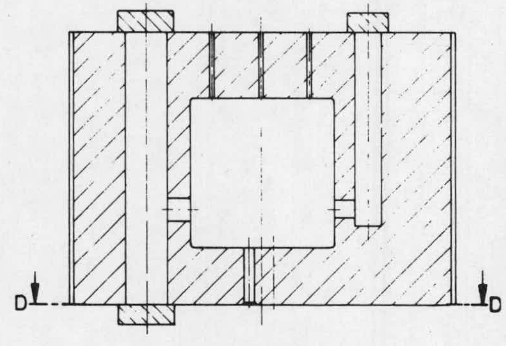


Figure 4-6. PCRV Horizontal Section (HKV Plant)

are individually connected to shared distributing cavities and then, via six coaxial ducts, to the core. Four cavities for the after-heat removal system are located between the primary heat extraction systems. This arrangement has the same PCRV outside diameter as the construction utilizing helical tube He/He heat exchangers, but the PCRV height is reduced from 35 m to 32.5 m.

The PCRV arrangement for the HTR-K consists of six steam generator cavities arranged symmetrically on a pitch circle of 26 m diameter. The cavities have a maximum diameter of 5.05 m and are connected to the core via separate inlet and exit ducts at the upper and lower levels of the core. Four after-heat removal cavities are provided. The arrangement results in a reactor vessel of 36.8 m diameter and 31 m height. Figure 4-7 shows the general configuration.

4.2.1.2 Design Data

The internal pressure during normal operation depends on the type of plant and ranges from 40 bar for the PNP plants to 60 bar for the HTR-K plant. The maximum pressure during operation is 1.05 times the normal operating pressure The design pressure amounts to 1.10 times the maximum pressure during operation.

The designs of the insulation and the liner cooling system are based on limiting the hot zone concrete temperature to 66^{o}C. The influence of hot spot zones on the prestressed concrete construction have not been investigated to date.

The materials employed (such as concrete, reinforcing bar steel, tension rod steel) are based on previously demonstrated construction.

For vertical prestressing, a procedure is employed which utilizes large-size cable bundles with an available tensile force of 645 tonnes.

For prestress established during the (tensioning) winding process, various methods can be employed (GA, Taylor-Woodrow, and others), which are all within the present state of technology. Load cells are installed on 10% of the tensioning rods in order to monitor performance during life.

During PCRV preliminary sizing, special attention was paid to the design feasibility of the reactor top head. The large number of shutdown and control rods, as well as fuel charging tubes, results in small clearances between the passages through the reactor head. In addition, the arrangement of the passages is so irregular, from a structural point of view, that a continuous run of reinforcement is not possible. For this concept then, with only the liner structure between the passages, no tensile stresses were permitted in the top head region. The effect of cooling-pipe-runs at the liner ceiling and at the passages of the shutdown rods have not been investigated.

The ligaments between the cavities are very heavily stressed, particularly in the top head region. This is a result of the large horizontal prestress required by the large diameter of the container and to assure that there is no tensile force in the region of the top head. Three-dimensional (3D) calculations were performed for the PCRV, to determine actual compressive force components introduced by prestressing at the outer edge of the PCRV, and these have been correlated against model tests.

4.2.2 PCRV EVALUATION

HRB industrial representatives consider that design of the PCRV has progressed to the point that design feasibility has been assured. They state that the design is based on US ASME Section III Division II Nuclear Code Requirements and has been analyzed by 3-dimensional, finite element methods plus dynamic relaxation for early and late life. The codes used have been developed in part by HRB with other FRG support and United Kingdom input. Previous PCRV designs have used scale models to verify stress distribution and crack evaluations. HRB considers that the technology is sufficiently developed so that modelling is no longer required. There is not total agreement in this regard, with KFA indicating that additional PCRV modelling is required. Prestressing technology is well developed in France, Germany, and Great Britain. HRB indicated that four constructions similar to the one proposed have been completed. The latest is Hartlepool (United Kingdom), which has recently been commissioned. It is noted that these constructions are much simpler in number of cavities and increased ligament dimensions and may not adequately support the HRB contention of demonstrated similarity.

A final design for the cavity closures for the PCRV has not been selected; however, this appears to be a straightforward design problem to be

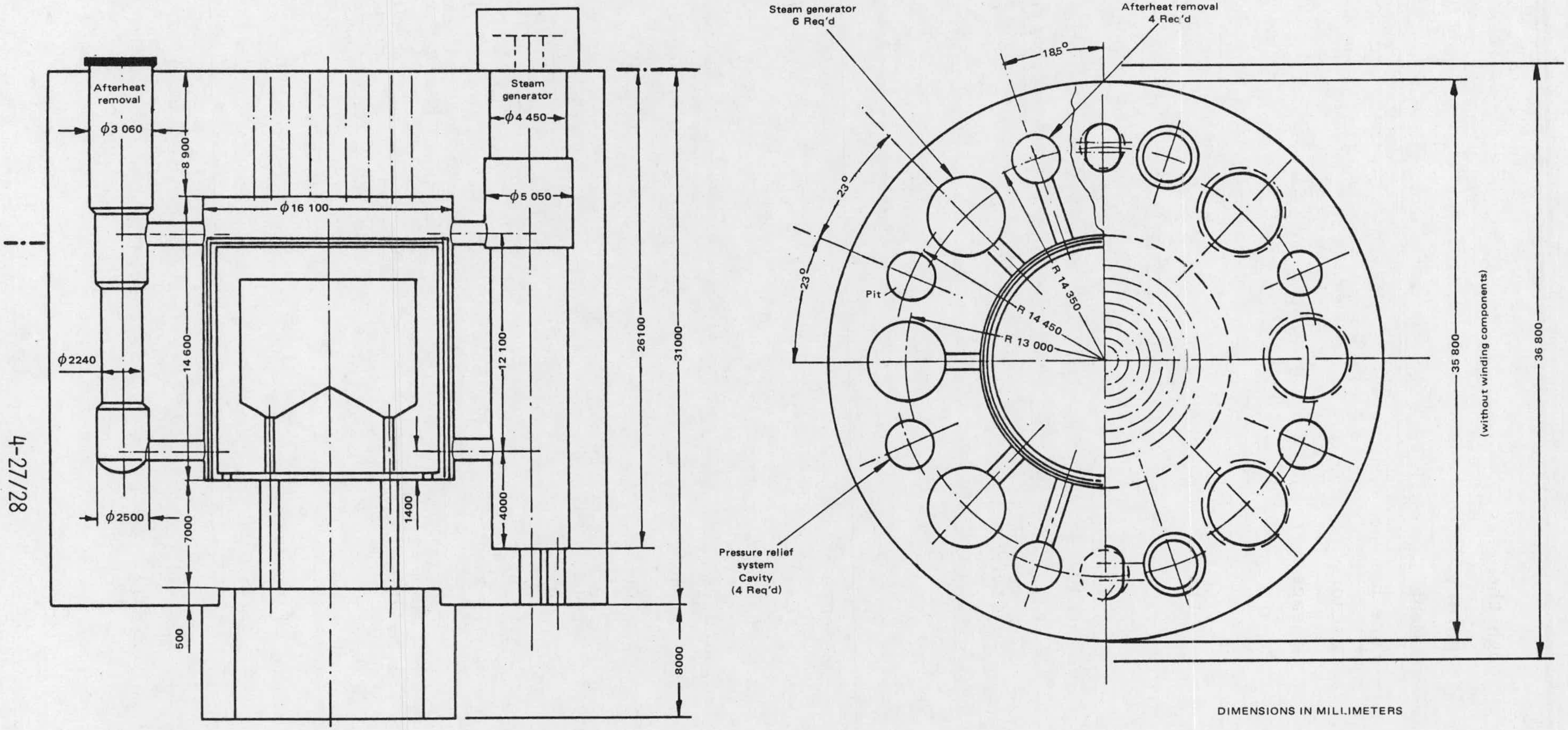


Figure 4-7. PCRV Arrangement for HTR-K

determined primarily by cost considerations. The HRB prestressed concrete closure is the most probable choice.

The liner and insulation design differs from FRG experience with the THTR but is generally similar to the GA-HTGR design. The THTR uses liner insulation consisting of multilayers of foil and a steel cover plate; but the PNP/HTR-K plants propose using fibrous insulation and steel-covered plates. No serious problems appear to exist; however, substantial testing is required in order to provide a qualified design. Plans being made to test 1 m^2 sections will include vibration tests to determine the effect on insulation material.

For liner maintenance, the FRG philosophy is to provide the capability for inspection and repair of all components. Several methods to determine liner failure are being considered, such as cooling water temperature monitoring, sampling for He leakage outside the liner, and ultrasonic inspection. FRG analysis indicates that liner weld repairing over concrete is conceptually possible, but this has yet to be proven by experiment. It is the FRG industrial contention that periodic in-service inspection is not necessary. However, this view has not been accepted by German utilities and remains an unresolved item.

4.3 HOT GAS DUCTS

4.3.1 HOT GAS DUCT DESCRIPTION

In all of the FRG plants being considered herein (i.e., HTR-K steam cycle, PNP - hydrogasification, and PNP-steam gasification), the primary-loop gas ducting is subject to steady state temperatures of 700°C and greater, or 300°C and lower. In all cases, the PCRV liners are subject to the lower temperature gas flows. The arrangements of the hot gas ducting depend on the type of plant and the arrangement of components. The most straightforward design is the concentric duct system for PNP hydrogasification as shown in Figure 2-7. Hot gas at 950°C from the core outlet is directed through the inner side of a concentric hot/cold duct to the inlet of the steam reformer. The gas exits from the steam reformer at 800°C and is directed through a similar hot/cold duct to the inlet of the steam generator. The gas leaves the steam generator at 300°C and flows to the primary circulator. Circulator discharge gas is returned to the core inlet and in transit is used to bathe the PCRV liner and hot duct outer annuli. This circulation is the same for each of the six primary loops. In the HTR-K steam cycle plant of Figure 2-1, hot gas at 700°C is carried to the inlet of the steam generator through the center of a concentric duct. After being cooled to 260°C it is returned to the core inlet, in part through a single cold duct and in part through the outer annulus of the hot duct. All six major loops of this system are identical. In the steam gasification systems of Figures 2-13 and 2-15 the core outlet hot gas at 950°C is ducted to one of six identical distribution plenum chambers by way of a concentric hot/cold duct. Depending on the type of heat exchanger used, the gas is then directed by way of concentric hot/cold ducts to either two or four heat exchangers. Cold gas at 300°C is returned to the reactor, with either all or partial flow cooling the hot gas ducts.

There are four designs being considered for the hot gas ducts. One design, which is no longer being actively pursued, was considered for use only with the lower temperature (700°C) HTR-K steam electric power plant. The other three designs, two by HRB and the other by GHT, are being evaluated for PNP application at 950°C, and a single design concept will evolve.

The GHT design has been applied to the steam gasification PNP plant arrangement of Figure 2-13, but could be applied equally to the hydrogasification PNP plant. The same is conversely true for the HRB design.

4.3.1.1 GHT Duct Design

The gas ducting is designed in such a fashion that, within the cavities of the PCRV, all the hot gas ducts are arranged inside the cold gas recirculating ducts, i.e., inside a cold gas bypass flow. Gas velocity in both cold and hot passages is limited to approximately 60 m/s.

Figure 4-8 shows the GHT design for a hot gas duct between the reactor and a steam reformer, while Figure 4-9 provides a representative view of a hot gas duct between the reactor and the helical He/He heat exchangers for the steam gasification plant. The ducting consists of a low-alloy-steel support tube lined on the inside with SiO ceramic insulation consisting of individual ceramic segments. A gas passageway consisting of graphite is arranged concentrically inside this support tube. Both the insulation and graphite are supported from attaching rings welded to the support tube. The ceramic segments

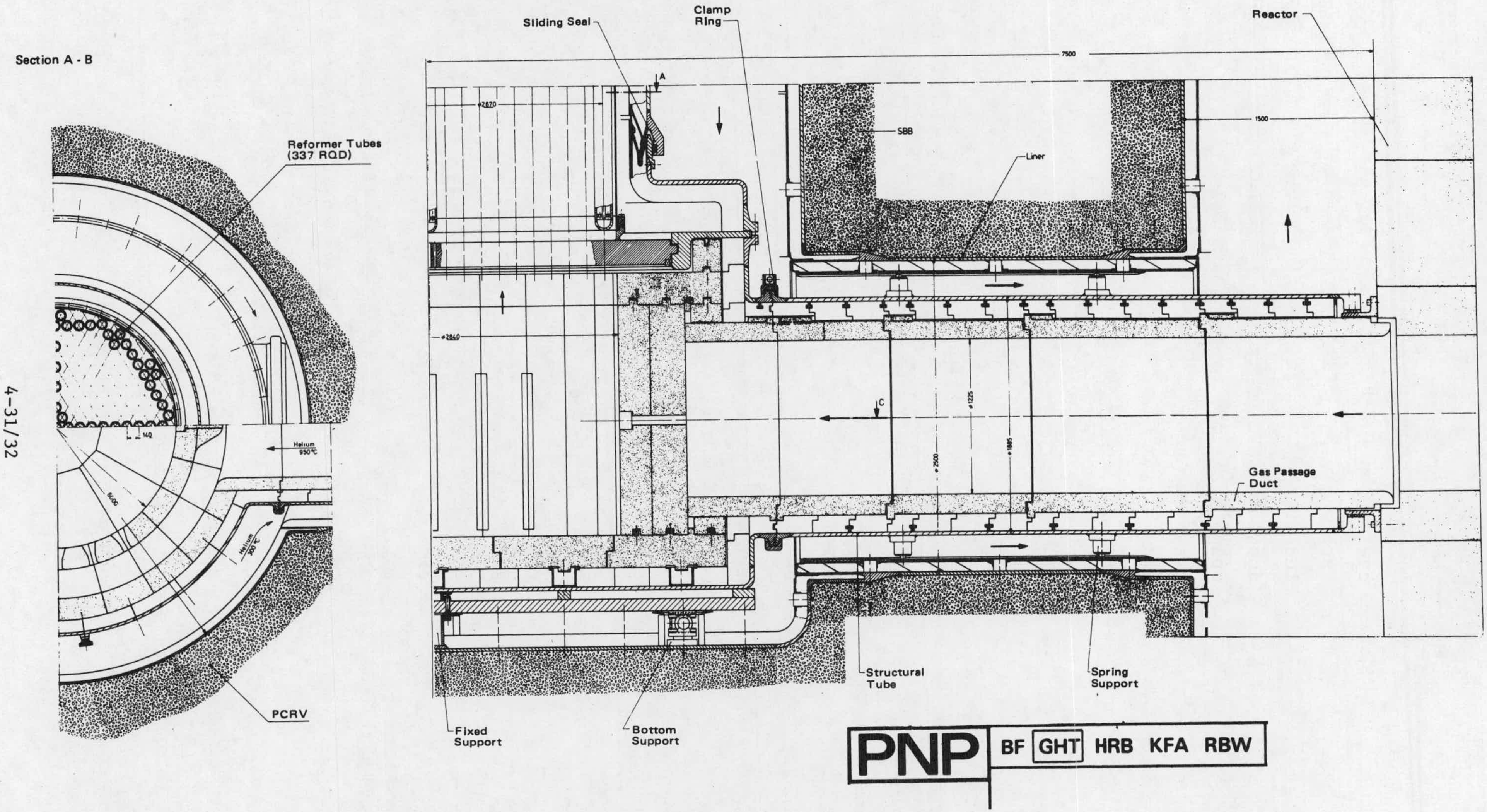


Figure 4-8. Hot Gas Duct: Core to Steam Reformer (GHT Design)

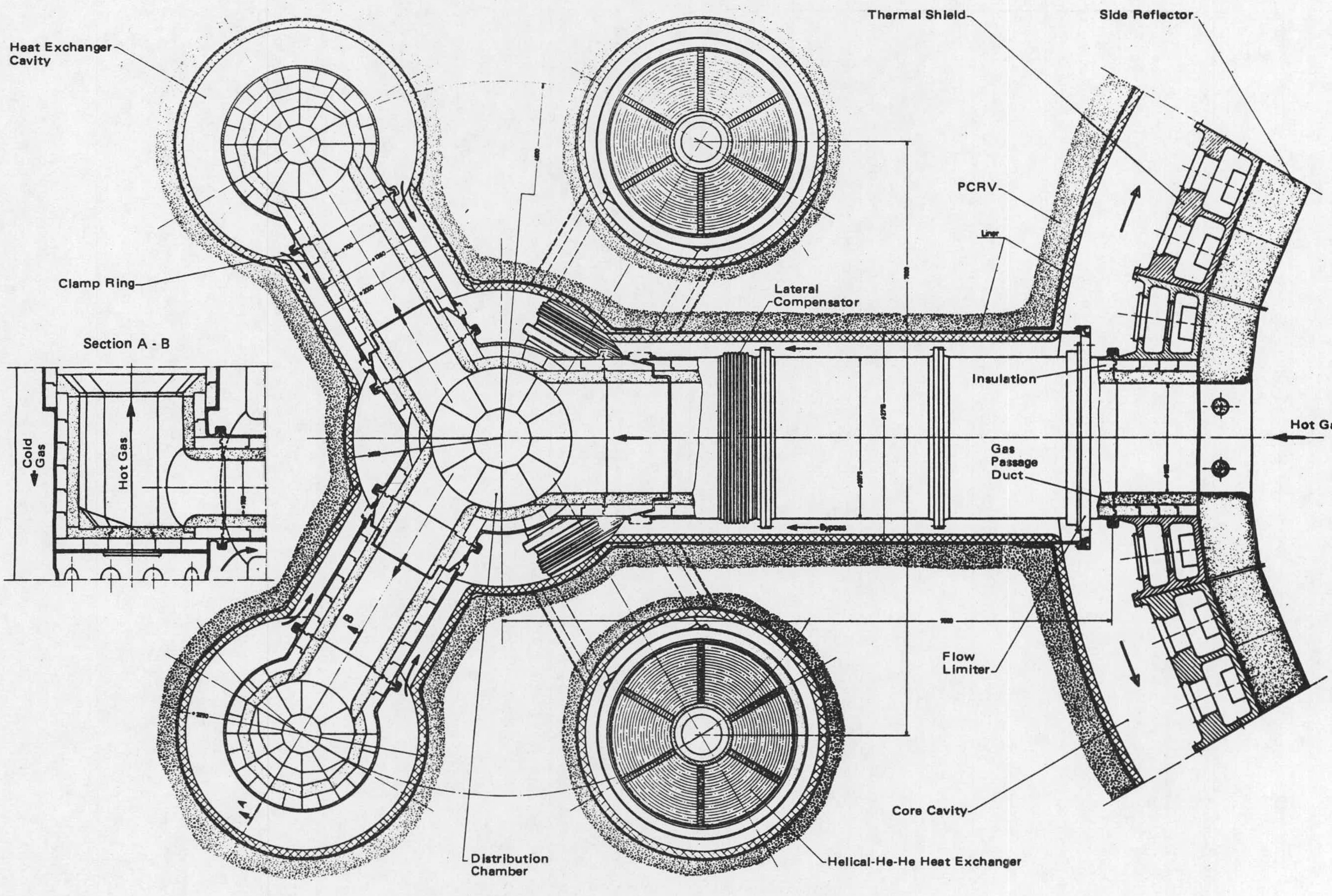


Figure 4-9. Hot Gas Duct: Steam Gasification Plant (GHT Design)

are supported by the tube in such a fashion that differential thermal expansions of the segments with respect to the support pipe are permitted. Furthermore, the segments are packed tightly so as to minimize the clearance gaps at their interfaces. Gas flow due to convection is, therefore, reduced to a minimum, thus ensuring that the support tube will maintain a temperature level close to the cold gas temperature. If required, fibrous insulting material can be placed in the gaps to further reduce gas flow. The graphite inner gas duct prevents direct exposure of the ceramic segments to the gas flow and protects the ceramic against damage caused by vibration and erosion. The support tube is supported within the PCRV by spring-loaded rolling elements attached to the support tube and guided in rails attached to the PCRV duct liner wall.

As a result of the two heat exchanger configurations selected and taking into consideration limiting size and ligament restraint conditions for the prestressed concrete reactor vessel, it was necessary to subdivide the hot gas flow to four (4) helical heat exchangers (on Version A), and two (2) U-tube heat exchangers (on Version B). To accomplish this, a distribution chamber was designed, which is insulated with the same basic principle as that applied to the hot gas duct. Gas flow distribution and velocity profiles into the components have to be determined by test.

To compensate for thermal expansion of the support tube, the following design techniques are being considered:

- Bellows-Type Compensators have the advantage of providing a seal which is completely leak-tight. In addition, they do not have any sliding components. Evidence of the gas-tightness of multilayer structures can be determined by means of pressure drop measurements or by surveying the temperature profile. Disadvantages of the bellows units are in their longer assembly lengths and larger diameters (when compared to the dimensions of a sliding contract-type), and susceptibility to fatigue damage.

- Sliding-Contact-Type Seals have the advantage of accommodating large deflections in a more compact size. Their disadvantages, when compared to bellows-type compensators, are the danger of fretting wear during operation and the resultant increased need for maintenance inspection. The sliding seal would consist of multiple "piston ring" seal elements cooled by controlled leakage from the cool gas outside. A titanium carbide coating would be used to minimize wear.

When the duct concept described above is used with heat exchanger Alternative A (helical), it is limited to the use of sealed bellows expansion compensators, because in some operating conditions it is possible to have higher pressure on the hot side than on the cold side. Leakage across a sliding-type seal could not be permitted under these conditions. The Alternate B heat exchanger duct arrangement does not have this characteristic, and either type of compensator may be used. Modifications to the duct design are being considered to eliminate this restraint. Present plans call for testing both the sliding-contact-type seals and the bellows-type compensators.

In the design of Alternative B the total flow discharge of the cold helium gas provides cooling of the hot gas channel. When considering other possible arrangements, Alternative A offers advantages with respect to cooling, by cooling portions of the hot gas channel on the outside through bypass gas flow. This bypass gas flow can be adjusted accurately and monitored with thermocouples.

The design of the hot gas duct was based on the requirement that the components susceptible to failure (such as compensators and sliding contact-type seals) be easily accessed and inspected. Details of the inspection methods have not been worked out as yet; however, all of the components of the hot gas channel are designed to be disassembled and repaired or replaced.

The hot gas ducting of the after-heat removal system can be arranged and structured similar to the hot gas ducting of the main loops. During after-cooling operation, heat losses have no great influence on system performance, and therefore it may be possible to obtain a significant simplification of the after-heat gas ducting.

4.3.1.2 <u>HRB Duct Design</u>

The two alternate HRB gas duct designs between the core and steam reformer are similar in concept to the GHT design but differ in design detail. The ducts consist of structural tubes cooled on the outside by cold helium (prior to its entry into the core region) and insulated on the inside. Figure 4-10 shows one alternate that has three layers of ceramic fiber insulation, 60 mm in total

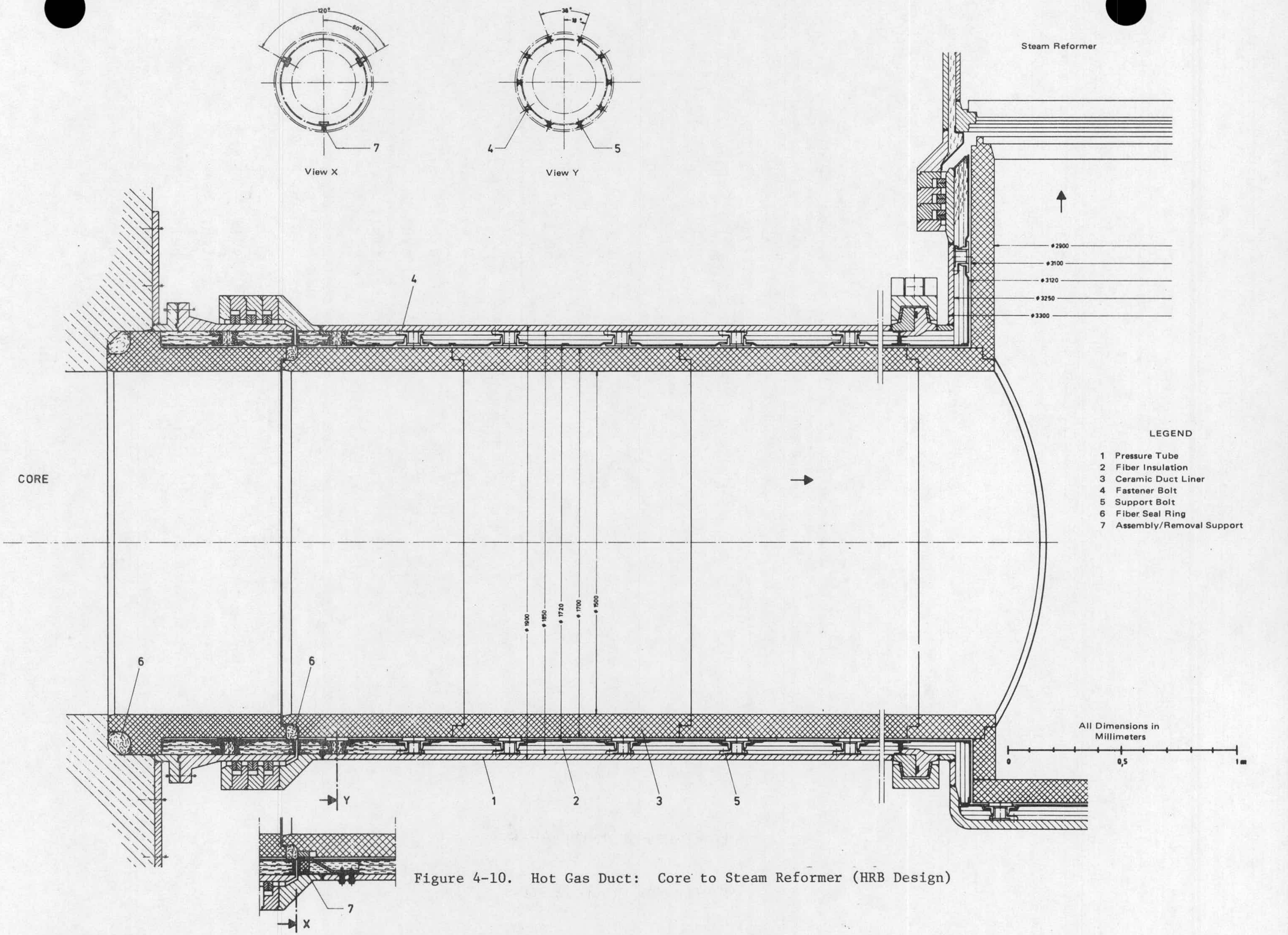


Figure 4-10. Hot Gas Duct: Core to Steam Reformer (HRB Design)

thickness, which are held to the structural tube by a thin foil and expansion-accommodating spacers. Inside the foil is a 100 mm-thick wall of silicon oxide in ring form to provide the hot gas duct. The silicon oxide rings are machined to overlap at their faces. In the design shown, a short length of similar duct construction is bolted to the core radial reflector (or thermal shield) and projects into a piston ring sliding seal on the duct proper (similar to the GHT design). The joint at the steam reformer is a rigid flange joint with a ring-type clamp.

In the alternate HRB design (not illustrated), four layers of ceramic fiber insulation, each of 25 mm thickness, are held in place by a 20 mm-thick cover plate attached by studs to the cold wall. KAOWOOL and SAFFEL fibrous silicon oxides are used for insulation, and fiber-reinforced graphite (CFC) is being evaluated for cover plate material. Studs will be of either CFC or TZM (titanium-zirconium-molybdenum). The structure is attached to the core reflector graphite with a flange and spring-loaded studs to allow for some relative motion between graphite blocks. The attachment structure is a separate short length and is attached to the duct structural tube through a split-ring-type flange joint. At the steam reformer end, the structural tube is guided and sealed to a stub tube attached to the steam reformer inlet plenum. Sealing and relative motion allowance is provided by a piston ring joint similar to the GHT design.

The current design requires removal of the steam reformer in order to remove the hot gas duct. Design changes are being evaluated to permit duct removal without removing the steam reformer.

Design conditions for the PNP plant reactor to steam reformer duct are the following:

	PNP
No. of ducts	6
Hot gas temp. ^{o}C	950
Cold gas temp. ^{o}C	300
Pressure, bar	40
Flow rate, kg/s	148
Maximum depressurization rate, b/s	10
Velocity, hot duct, m/s	61
Velocity, cold duct, m/s	21

For the HTR-K design the hot gas temperature is reduced to 700°C, and metallic cover plates can be used in place of the reinforced graphite design.

4.3.2 HOT GAS DUCT EVALUATION

Design of the hot gas duct has been finalized to the extent of general configuration; that is, the use of concentric hot/cold ducting with cold gas cooling of the structure has been decided on, but final materials selection and details of attachments and expansion devices are not yet firm.

The design appears to be a logical extension of previous gas reactor design experience in both FRG and elsewhere. FRG experience at similar temperatures was achieved with the AVR, where the coolant actually attains a temperature of 950°C. The AVR configuration (vertical arrangement of core, hot gas duct, and heat removal system) and a duct design consisting of passing the hot gas through structures of graphite and carbon stone are not applicable to large-scale plants.

In the HTR test reactors (Dragon and Peach Bottom) and in the Windscale AGR, coaxial ducts are used and cooled in counterflow by cold gas. However, the maximum helium temperatures are only 750°C, and extrapolating to the present design concept is difficult. In addition, these ducts have small dimensions compared to the large-scale plants.

In the THTR (300 MWe), hot gas is transported by a hot duct between the reactor core and the heat exchangers, with insulation within the ducts made of metal foils. Since all the primary circuit components are within one large cavity in the prestressed concrete vessel, the problem of insulating the vessel itself against hot gas does not occur.

The major hot duct problems appear to be in the selection of materials for the environment and long life requirements. Particular emphasis must be placed on obtaining design experience with ceramic ducting components. It is considered that partial ceramic construction will be needed because of the severe thermal and environmental conditions and the poor high-temperature characteristics of most weldable and formable metallic alloys. In addition, testing of the selected materials and configuration must be done to evaluate the effects of flow and acoustically induced vibration, depressurization transients, and long-term effects on conductivity. Component testing is required to select the optimum design of expansion compensators, attachment devices (particularly to the core reflector), and flange connectors for ready removal and service. Some preliminary testing has been done in regard to thermal insulating properties and depressurization but apparently at reduced size and velocity, and substantially more test and development work must be done before a design can be selected.

Of particularly high potential is the proposal by HRB to use fiber-reinforced graphite for insulation cover plate material. HRB indicates that test work in this area is being conducted.

Of particular concern is that primary emphasis on hot gas duct design has been placed on the core outlet concentric duct design, although, in actuality, the same insulation design problems appear in a number of other regions (such as at the inlet to and around the steam reformer, steam generator, and He/He heat exchanger).

4.4 STEAM REFORMERS

4.4.1 STEAM REFORMER DESCRIPTION

In the steam reformer, incoming process gas at 330^{o}C, consisting of a mixture of methane (CH_4 from process coal) and steam is heated to approximately 810^{o}C while being treated in a catalytic bed and converted to H_2 and CO plus some residual water and methane. The resultant products are then cooled to approximately 520^{o}C and returned to the coal processing plant. The heat necessary to provide the desired reaction temperature is provided by helium circulated through the reactor primary loop. The temperature conditions have been chosen to provide optimum system performance. Cooling of the output process gas from 680^{o}C to 520^{o}C is accomplished by heat transfer to the incoming process gas through a recuperator. A schematic representation of the primary loop system is shown in Figure 4-11. In the German effort, two recuperator designs have been investigated: one in which the recuperator is located in a separate cavity within the PCRV and one where the recuperator

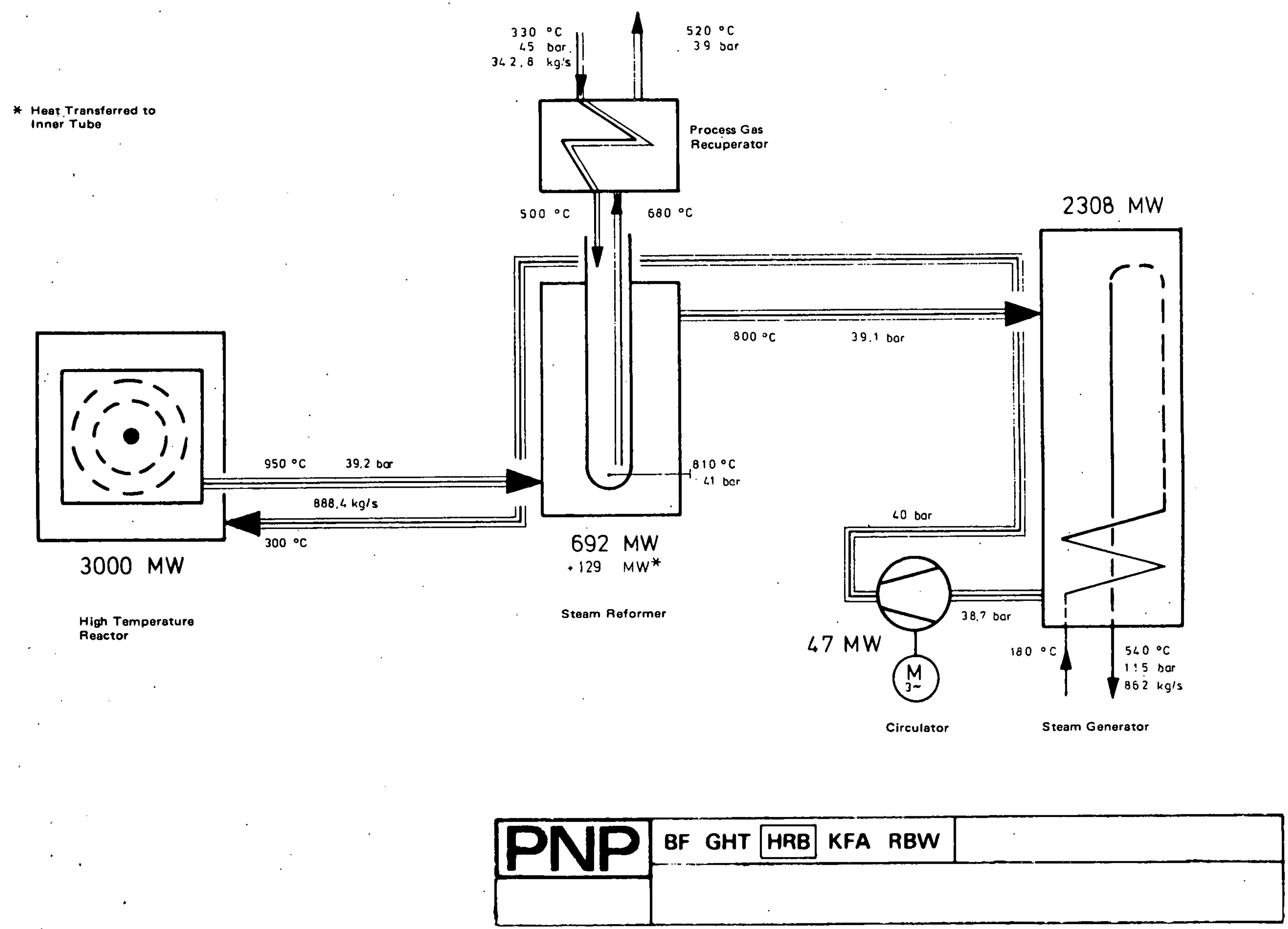


Figure 4-11. Hydrogasification Plant Primary Loop Schematic

is integrated into the steam reformer. The second method has been selected for the PNP reference design and provides the basis for current FRG effort. Both designs are described below.

The steam reformer illustrated in Figure 4-12 is without the integral recuperator, and Figure 4-13 shows one possible modification to the reformer upper end that incorporates the recuperator within the reformer cavity. Alternate designs to optimize the integral recuperator are being investigated but are still in the preliminary stages.

The steam reformer contains 313 reforming tubes of 150 mm outside diameter which are closed at one end with a hemispherical head and open at the other. The open end is welded to a carrier plate and the tubes supported vertically with the closed end down. A smaller tube is placed down the center of the reformer tube and the space between the concentric tubes filled with the nickel-based catalyst. The catalyst geometry being considered uses Raschig-rings, which are small-diameter, right-circular cylinders of approximately equal length and diameter with a central concentric hole (Figure 4-14). They are placed randomly in the reformer tubes. The mixture of steam and methane is brought into the outer tube at the open end. It then flows downward through the catalyst, where it is reformed. It is then turned 180°, flowing upward through the central tube. A temperature reduction from 810°C to 680°C is experienced on this return passage. Heating of the incoming process gas and catalyst is by the hot helium of the reactor primary circuit, which flows upward between the reforming tubes. The distribution of the incoming gas mixture into the individual reforming pipes and the collection of the reformed gas are carried out within a cylindrical superstructure above the carrier plate. Table 4-4 summarizes the steam reformer characteristic data.

The reforming tubes as shown in the representative design pictured in Figure 4-12 have a smaller diameter at their upper end than at that section which contains the catalyst. Therefore, the ligament section between tubes in the carrier plate is increased for strength. At the same time, the space between tubes provides a collecting plenum for primary gas between the carrier plate and a seal plate. Primary gas from the plenum flows radially outward through the reformer tube shell and is collected in an annular chamber, where it flows downward to the inner pipe of a concentric duct leading to the steam generator. The bottom of the carrier plate is cooled by cold helium from the primary circulator as it is being returned to the core. The carrier plate is insulated on the process gas outlet side, which is at 520°C and, therefore, is subjected only to the cold gas temperature of 300°C. The seal plate, which provides the upper boundary of the primary gas outlet plenum is insulated on the primary side, where gas temperature can reach 800°C during normal operation. The bundle of reformer tubes is located inside the primary gas shell, which will be manufactured in three sections to facilitate assembly. The top section with the carrier plate is attached to the cavity liner wall with a cylindrical mounting member to permit differential expansion between the plate and the liner. The midsection, which contains the hot duct to the steam generator, is supported by four flange feet from the cavity liner, and the lower section with primary gas inlet ducting is supported from the cavity liner floor. Expansion and movement between the sections is provided by multiple piston-ring-type expansion joints. A spacer plate is mounted at the bottom of the middle section, which provides lateral support and spacing to each of the reformer tubes through a pin mounted on the bottom of each tube and sliding in a graphitized bushing in the spacer plate.

In the construction of Figure 4-12, the structure at the top of the reformer is provided only to distribute and collect the incoming and re-formed process gas. Preheating and partial cooling of the process gas is done in a separate recuperator as shown in Figure 4-15.

A steam reformer with integral recuperator, as shown in Figure 4-13, provides for heat transfer between outlet and incoming process gas in the superstructure region above the carrier plate. The steam reformer is essentially unchanged from that shown in Figure 4-12. Incoming cold process gas is collected in individual tubes within the process gas cavity and ducted to a plenum between the carrier plate and an upper seal plate and then into the reformer tube. The hot process re-formed product gas is piped through this plenum between the carrier plate and an upper seal plate and then into the reformer tube. The hot process re-formed product gas is piped through this plenum within the reformer return tubes to an exit plenum above the inlet plenum. Heat transfer between the hot and cold process gas streams is enhanced by adding helical fins to the return tubes and flow ducting around the reformer tube inlets. The outlet gas is then ducted through individual ducts to the process gas cavity in a manner similar to that for the inlet gas.

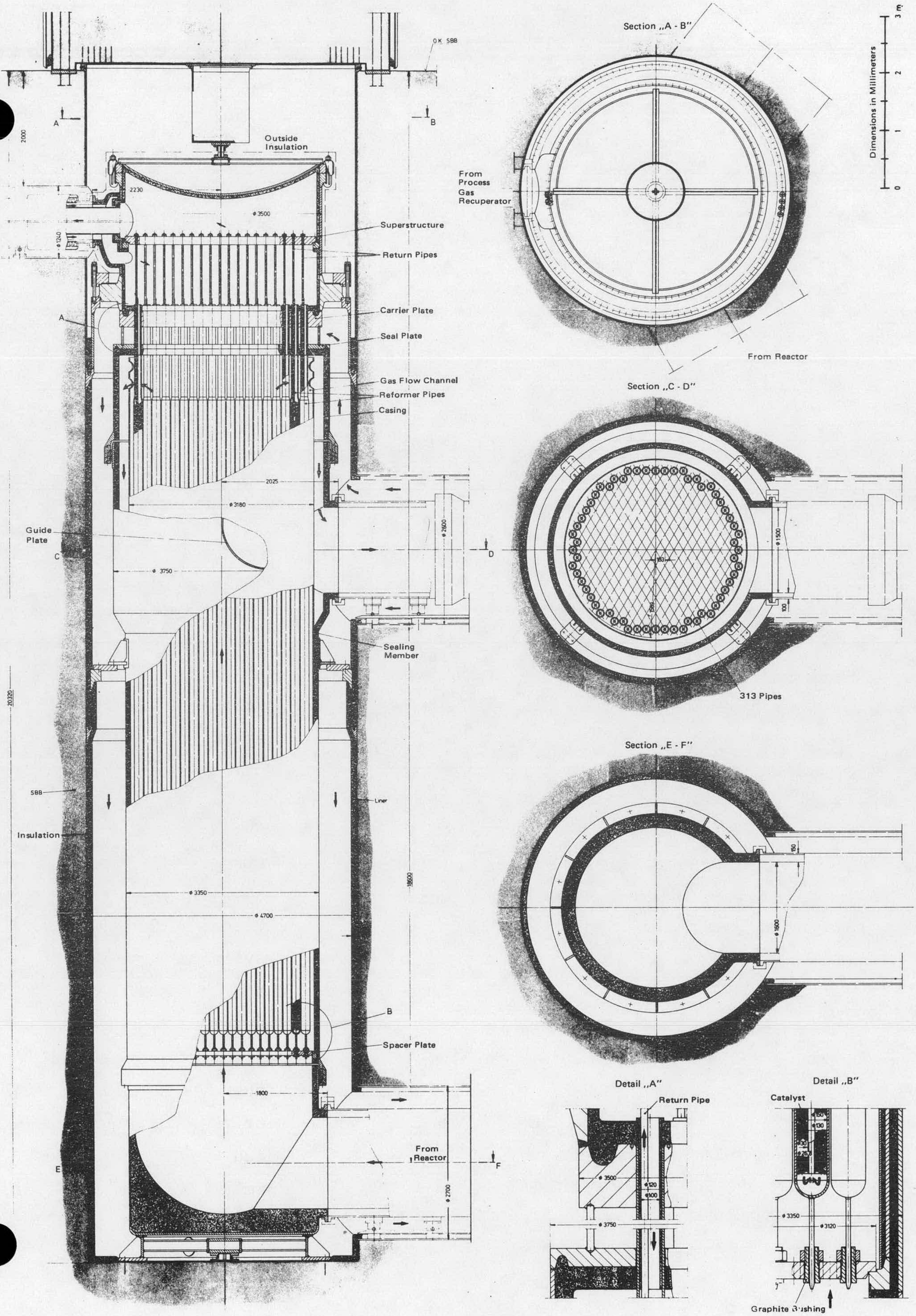


Figure 4-12. Steam Reformer Without Integral Recuperator

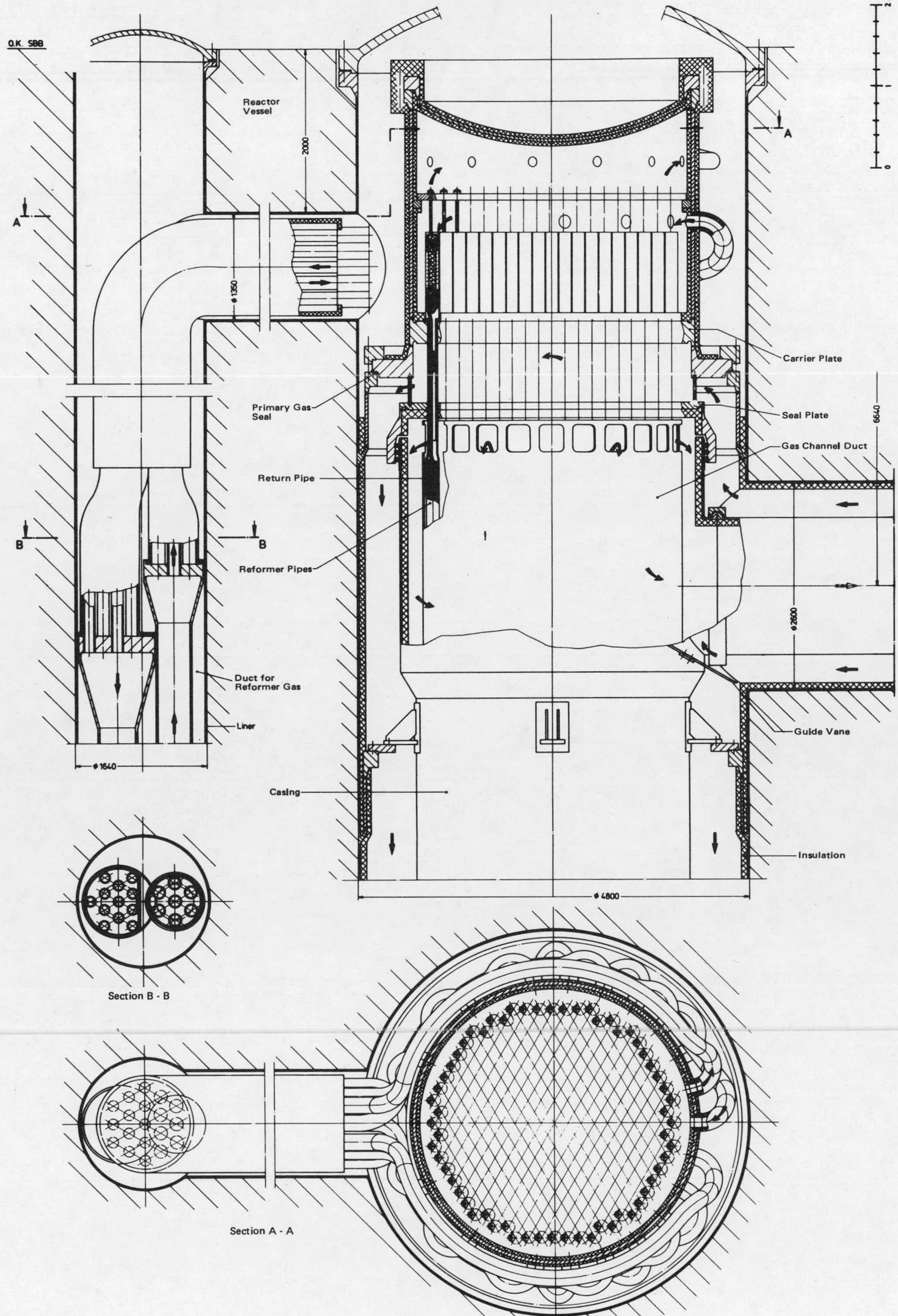


Figure 4-13. Steam Reformer With Integral Recuperator

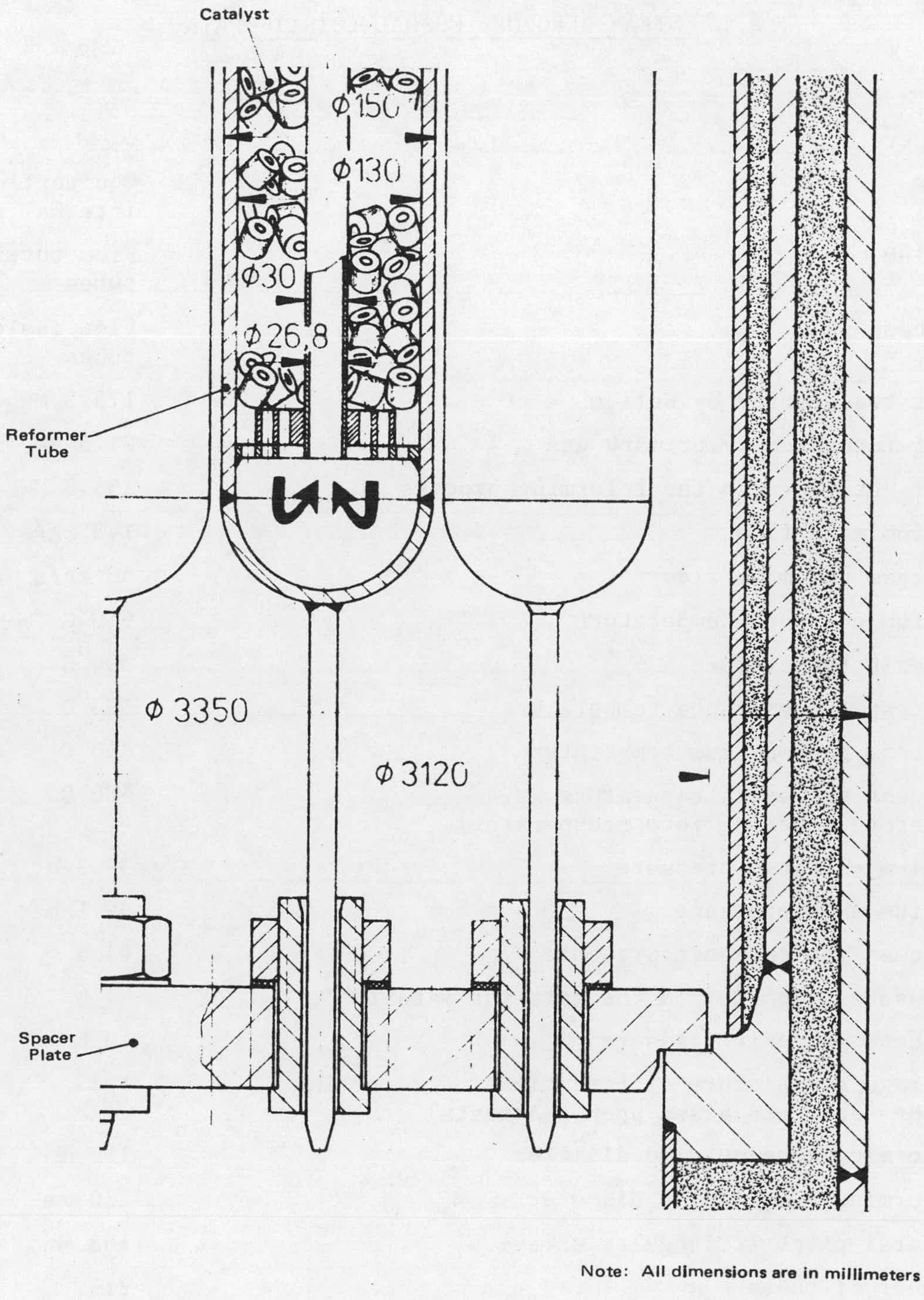


Figure 4-14. Raschig Ring Catalyst Arrangement

TABLE 4-4

STEAM REFORMER CHARACTERISTIC DATA

Number	6
Type	Counterflow with internal return
Helium	Flow outside of tubes
Process gas	Flow inside of tubes
Heat transferred by Helium	115.3 MW
Heat recovered by primary gas	21.5 MW
Heat entered into the reforming process	136.8 MW
Helium mass flow	148 kg/s
Process gas mass flow	58 kg/s
Helium entrance temperature	$950^{o}C$
He exit temperature	$800^{o}C$
Process gas entrance temperature	$500^{o}C$
Process gas maximum temperature	$810^{o}C$
Process gas exit temperature (before entering into recuperator)	$680^{o}C$
Helium entrance pressure	39.2 b
Helium exit pressure	39.1 b
Process gas entrance pressure	44 b
Pressure at exit from the catalyst material	41 b
Process gas exit pressure	40 b
Average temperature differential for reforming tube (estimate based upon EVA-tests)	$192^{o}C$
Reforming tube outside diameter	150 mm
Reforming tube inside diameter	130 mm
Lateral pitch (triangular array)	163 mm
Number of tubes	313
Casing inside diameter	3.15 m
Average helium velocity between pipes	40 m/s
Average heat transfer coefficient at the Helium interface	990 w/m^2K

Table 4-4 (Cont'd.)

Average heat transfer coefficient at the process gas interface	1020 w/m^2K
Average heat conduction coefficient	385 w/m^2K
Heating area	1560 m^2
Average heat flow density for reforming tube	73.9 kw/m^2
Reforming tube active length	10.6 m
Reforming tube total length	12.2 m
Steam reformer maximum diameter	4.43 m
Steam reformer maximum length	15.4 m
Weight	Approx. 250 tonnes
<u>Return Pipe</u>	
Outside diameter	30 mm
Inside diameter	26.8 mm
Type	Straight pipe
Total length	13.5 m
Average heat transfer coefficient outside surface	1020 w/m^2K
Average heat transfer coefficient inside surface	2140 w/m^2K
Average heat conduction coefficient	617 w/m^2K
Heating surface in active reforming tube segment	1 m^2
Average heat flow density	68.8 kw/m^2
<u>Catalyst</u>	
Type	Raschig rings
Expected change out frequency	6-8 years
<u>Process Gas Recuperator (PGR)</u>	
Location	Integrated into the reformer
Type	Helix-type return pipes above the re-former carrier plate
Input gas entrance temperature	330^oC
Input gas exit temperature	500^oC
Reformed gas entrance temperature	680^oC
Reformed gas exit temperature	520^oC
Helix diameter	110 mm
Number of helix turns	20
Pipe length	7.0 m
Height of helix	0.9 m

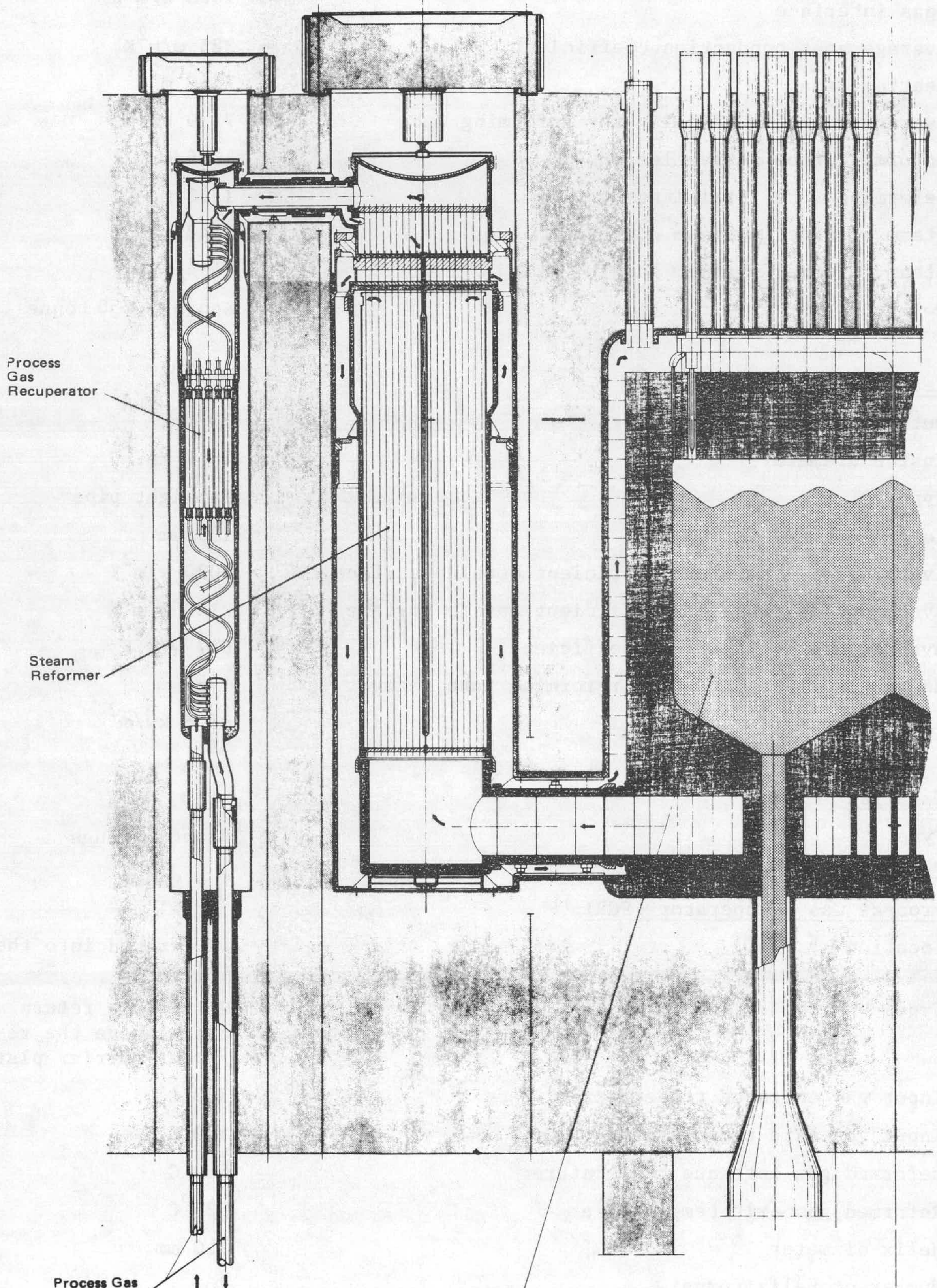


Figure 4-15. Steam Reformer Separate Recuperator

Variations of this general design concept (integral recuperator) are being pursued by HRB but are considered too preliminary for evaluation review. The catalyst must be renewed or replaced every six or eight years. The following replacement sequence is used:

a) Remove the concrete lid of the steam reformer cavity.

b) Loosen the clamping bolts at the dished cover of the superstructure and remove the cover.

c) Loosen and remove the bayonet connectors at the 313 return pipes.

d) Loosen the bolted connection between the perforated plate and flange and remove the plate.

e) Lower a grid onto the flange as a working platform.

f) Remove the catalyst with a suitable vacuum apparatus.

g) Insert the new catalyst.

h) Reassemble in the reverse order as described above.

The reformer is so designed that it may be replaced during plant life. In order to do so, the concrete lid is removed, the entrance and exit process gas pipes are cut, and the bolts which hold the carrier plate in place are removed. Subsequently the carrier plate, the superstructure, the seal plate, the upper section of the casing (including the upper interface seal), as well as the bundle of reforming pipes (with the lower spacer plate), can be removed as one unit. The middle and the lower sections of the casing will not normally be replaced. Their removal, however, is possible and would be necessary if the gas ducting between the steam reformer and heat exchanger or between steam reformer and reactor has to be removed.

All parts relevant to safety have to be tested and inspected at prescribed intervals and levels of detail. It is, therefore, anticipated that pressure tests, leak checks, visual inspections of the exterior and interior as well as other nondestructive testing will be done at different times (about every two, four, and eight years) and at different levels of detail. It is planned to perform these tests, as much as possible, in connection with the catalyst changeout. Inspection of the inside of the reforming tubes will require that the return pipes be removed after the catalyst has been removed. Inspection of the inside walls is then possible visually or by means of ultrasonic or eddy current measurement methods.

One possible mode of steam reformer operation is the startup procedure for one loop after it has been shutdown and the other loops are at operating power. Such operation requires that the steam reformer inlet be subjected to high core outlet temperature immediately after loop circulator startup. Some design effort has been conducted in this area. The objective is dilute core discharge gas with variable quantities of cold inlet gas in order to provide reasonable transient temperatures, which normally are limited to approximately $1^{o}C/s$. It is not known at this time if dilution is required or not, and this design effort is considered preliminary.

Final material selection has not yet been made for the steam reformer components, but candidate materials have been identified and testing is in progress. The critical component from a materials standpoint is the reformer tube, which must withstand the $950^{o}C$ primary loop helium and the process gas, steam, and products for a projected life of 100,000 hrs. Materials under consideration are Incoloy 802, Incoloy 800H, Manuarite 36X, and Inco-519.

4.4.2 STEAM REFORMER EVALUATION

The steam reformer concept as described above must be considered as not just a recent development but a culmination of previous Federal Republic of Germany effort and experience in chemical and nuclear steam reformer design and experiment. Any evaluation must reflect this consideration. An excellent summary with design detail and background is included in Reference (17), and it is reasonable to assume that this information has been incorporated into the reformer conceptual design. Some pertinent information from the reference includes the following:

- Conventional (nonnuclear) steam reforming uses sulfur-free fuel combustion for heating primarily by radiation at a maximum flame temperature of approximately 1500^{o}C. The reforming temperature is in the range of 750^{o}C to 850^{o}C with a pressure of 1-30 bar and with a steam/methane ratio of 2/1 to 5/1. A summary data comparison of nuclear and "conventional" steam reforming plants is shown in Table 4-5. The process is technically well developed today and is applied worldwide for the production of gases for ammonia and methanol synthesis as well as H_2 production for hydrocracking processes.

- The selection of reforming operating parameters is a complex evaluation of equilibria methane conversion factors and reaction kinetics influenced by the subsequent use of the product H_2 or H_2 + CO mixture. In general, for all applications which require a high operating pressure, a high reforming pressure is advantageous (compression energy can be saved by compressing the gas before the steam reforming process). A disadvantage of increasing pressure is that the unreformed methane content of the product gas will increase with increased pressure. As an example, for typical parameters of temperature and H_2O/CH_4 ratio, an increase in reforming pressure from 30 b to 40 b will decrease the CH_4 conversion by about 10%. Obviously optimization is required.

- At the operating regions of interest (i.e., temperature of 600 to 800^{o}C, pressure of 20 to 30 b, H_2O/CH_4 ratio of 2.5 to 3), the reforming reaction rates are found to be limited by heat flux to the reformer tube. Therefore, the thermal-heat transfer characteristics of the reformer are extremely important. Trade-off studies have been made of heat transfer coefficients and pressure drops on the process and helium sides versus gas velocities to provide size and helium pumping power requirements. It was also found that utilizing an inner gas

TABLE 4-5

COMPARISON OF CONVENTIONALLY HEATED (FLUE GASES) AND NUCLEAR-HEATED (HELIUM) STEAM REFORMERS

Parameter	Conventional Plants	Nuclear Plants
Tube length	8 ... 12 m	10 m
Internal diameter	100 ... 150 mm	100 mm
Wall thickness	15 ... 20 mm	15 m
Product gas removal	Outside reformer tube	Within reformer tube
Reforming pressure	1 ... 25 b	40 b
Reforming temperature	800 ... 850°C	800 ... 850°C
Heating side pressure	1 b	40 b
Heat transfer	Radiation	Convection
Space utilization	< 1 tube/m^2	~ 45 tubes/m^2
Max. heating temperature	1400 ... 1500°C	950°C
Max. tube wall temp.	900°C	900°C
Max. pressure difference across tube wall	0 ... 25 b	1 bar (hot part)
H_2O/CH_4 ratio	2/1 ... 5/1	2/1 ... 5/1
Mean heat flux	60 000 kcal/m^{2}h	60000...70000 kcal/m^{2}h
Heat flux max./min	10/1	1.5/1
Rate of gas flow	~50000 Nm^3 H_2 + CO/m^{2}h	~50000 Nm^3 H_2 + CO/m^{2}h
Service life aim	100,000 h (60,000 h attained today)*	>100,000 h
Reformer tube materials	G-X40 CrNiNb 2524 (W.-No. 1.4855; IN 519) G-X45 NiCrCoWNb 4625 (IN 643) G-X45 NiCrCoWNb 3626 (IN 638)	To be determined
Product gas tube materials	Incoloy 800, Incoloy 807	

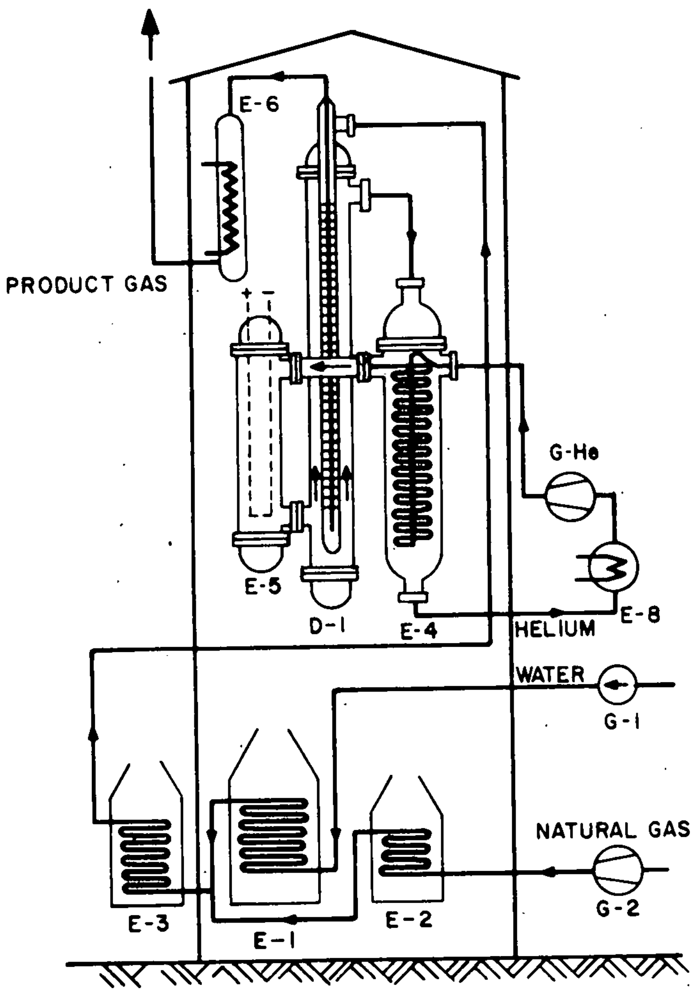
*Individual tube life can be considerably shorter, but in conventional plants repair is relatively easy.

return duct for hot product gas down to approximately 650°C can be used to transfer heat to the catalyst filling and leads to an approximate 20% increase of the heat transferred to the reformer tube.

- Stress analyses of reformer tubes in the reference were conducted for internal pressure, external buckling, thermal stress during startup and shutdown with a temperature transient of 1°C/min, and thermal stress during operation. Results indicated that the reformer tubes are capable of satisfying the conditions evaluated. Neither a fatigue analysis of the tubes nor an analysis of the carrier plate support for the tubes was provided. Based on this, it can be generally concluded that the design can be shown adequate if materials characteristics are known. Low and high cycle fatigue and creep/fatigue characteristics, however, must be demonstrated.

- Materials properties under actual operation conditions have been and are continuing to be evaluated, and this appears to be the greatest area of uncertainty. Creep properties in the 900°C range, hydrogen permeation through the reformer tube wall into the primary gas stream, and the corrosion effects of the reactor coolant on candidate tube materials and their possible control are all areas of concern. The problems and concerns of tritium permeation from a systems standpoint must be demonstrated and licensability must be determined. The design criteria and design code for use of properties must also be developed and accepted for the proposed materials at the temperatures of interest.

Data for steam reformer operation in a helium-heated loop simulating nuclear plant operation have been and are being established through operation of the EVA-I plant at KFA in Jülich, Germany. The loop schematic and operating condition parameters are shown in Figure 4-16. Tube sizes of 100 to 150 mm inside diameter and length of 10 to 15 m may be accommodated in the facility. Charateristics and results of a typical test specimen are shown in Table 4-6.



Main data:

Max. helium temperature	1000°C
Max. helium pressure	50 b
Max. helium throughput	0.4 kg/sec
Inlet-temperature process gas	450...550°C
Outlet-temperature process gas	750...850°C
Reforming pressure	30...40 b
Throughput CH_4 max.	200 Nm^3/h
Throughput steam max.	500 kg/sec

D-1	steam reformer
E-1	evaporator, superheater
E-2	CH_4-heater
E-3	H_2O/CH_4-superheater
E-4	helium heat exchanger
E-5	helium heater
E-6	cooler
E-8	heat exchanger
G-1	feed water pump
G-2	CH_4-compressor
G-He	helium-circulator

Figure 4-16. EVA I Plant Schematic

TABLE 4-6

TYPICAL TEST IN EVA I FACILITY

Dimensions	
Length of tube	14.38 m
Inner diameter	160 mm
Wall thickness	20 mm
Annulus for helium	12.5 m
Length of inner pigtail	30 m
Inner diameter of inner pigtail	25.4 mm
Wall thickness of inner pigtail	4 mm
Data on Helium Side	
Mass flow	0.405 kg/s
Pressure	39.6 b
Temperature inlet	950°C
Temperature outlet	700°C
Data on Process Side	
Mass flow	0.045 kg/sCH_4 + 0.116 kg/sH_2O
Pressure (inlet tube)	34.3 b
Pressure drop in tube	3.7 b
Temperature inlet	450°C
Temperature outlet	820°C

Work to date shows that the helium-heated steam reforming of methane is basically possible and has established fundamental design parameters. However, the design of a large tube bundle has to be proven. This testing is planned in the EVA II Facility at KFA. That facility is scheduled for completion and check out in late 1979. A 30-tube steam reformer test section is planned with 3 tubes each of Incoloy 800H, Incoloy 802, Manuarite 36X and the remainder of Inco-519. Major parameters for this test are shown in Table 4-7.

TABLE 4-7

DATA FOR EVA II FACILITY

Reformer Facility	
Power of electrical heater	10 MW
Temperature helium	950°C
Mass flow helium	3.2 kg/s
Pressure helium	40 b
Pressure drop helium	1.2 b
Outlet temperature helium reformer	600°C
Inlet temperature helium electrical heater	350°C-(450°C)
Reforming conditions	T=825°C p=40 b H_2O/CH_4=3/1
Methanation Facility	
Power	~6 MW
Mass flow	~10,000 Nm^3 Gas/h
Pressure	45 b
Maximum temperature	600°C
Heat production (steam)	T~225°C/18 b
Flow sheet	3-stage

In summary, it is apparent that considerable basic work has been done by the Federal Republic of Germany to determine the design criteria for steam reforming processes. Important parameters and relationships between temperature, pressure, H_2O/CH_4 ratio, and reformer heat transfer characteristics have been established. What remains to be done is the considerable effort to provide a steam reformer design that will satisfy the manufacturing, operating, maintenance life, and safety requirements of a nuclear plant installation. Of immediate concern are the properties of the materials to be used as affected by the service conditions. In the longer term design problems of weld joint fatigue, tube vibration, insulation attachment, and flow-induced vibration, large-diameter expansion joint design, transient safety analysis, and the general area of maintainability must be resolved.

4.5 INTERMEDIATE HEAT EXCHANGERS (IHX)

4.5.1 IHX DESCRIPTIONS

As described in Section 2.2.3, the processing plant used to convert coal to methane by steam gasification, unlike the hydrogasification plant, utilizes an intermediate circuit to separate the reactor plant from the rest of the gasification equipment. This intermediate loop uses helium as an energy transport medium and, therefore, requires a He-to-He heat exchanger between the secondary loop and the reactor primary helium system. The H_e-to-H_e heat exchanger is located within the PCRV as described in Section 4.2.1.

Two alternate design concepts for the intermediate heat exchanger are being evaluated in parallel in the Federal Republic of Germany. One plant design (Alternate A) was shown in Figures 2-13 and 2-14, and consists of 24 exchanger units, where 4 units are provided for each of 6 primary circulating loops. These units are of helical tube construction. The other plant design (Alternate B) was shown in Figures 2-15 and 2-16, and consists of 12 exchanger units each of 8 U-tube modules with 2 units provided for each of 6 primary circulating loops. The primary advantage of the helical tube units is that they are of such size that all construction can be done in the fabrication shop and no field welding is required. The advantage of the U-tube-type IHX is that the tubes are more accessible for visual inspection, and therefore it is easier to find and isolate a failed tube.

Design operating conditions for the two alternate IHX designs include:

- Design life = 20 years
- Full 40 bar differential pressure between primary and secondary gas and in both directions for 1000 hours at operating temperatures
- 5 bar/s primary side depressurization transient

- $10^{o}C/s$ startup transient for normal operation
- $30^{o}C/s$ emergency condition, 5 per unit life

4.5.1.1 Helical Tube IHX

Some design and mechanical data for the helical tube IHX is listed in Table 4-8, and the assembly is shown in Figure 4-17. The heat exchanger is basically a vertically mounted counter-cross flow design. Hot primary gas from the reactor at $950^{o}C$ enters at the bottom of the IHX and flows upward on the shell side and across the helically wound tubes. After cooling to $300^{o}C$, the primary gas, at the top of the tube bundle and below the tube bundle support plate, is turned radially outward through the shell wall and downward in an annulus formed between the IHX shell and the cavity liner. The primary gas is returned to the reactor through a concentric duct system at the bottom of the IHX. On the secondary side, intermediate-loop helium gas at $240^{o}C$ enters through multiple pipes at the top of the IHX into a plenum above the tube bundle support plate and flows downward through the helical tubes. The tubes are individually connected at the bottom of the IHX into a central return pipe, and the gas flows upward and out through the tube support plate, emerging at the top of the IHX at a temperature of $900^{o}C$.

The primary support structure for the IHX is the tube support plate at the top of the assembly which carries the weight of the helical tubes, the shell hot liner, and the center return pipe. This plate, which is supported on a low-stress flange extension attached to the cavity liner, is also used as the seal between the primary and secondary helium. The helical tubes penetrate the support plate into an annular ring on the top surface of the plate which, sealed by an additional plate, serves as a collector for incoming intermediate-loop He. Incoming He is directed via a collecting manifold through three pipes in the seal plate. This arrangement provides cool gas on both sides of the tube support plate and, therefore, minimizes insulation use and thermal stresses in the plate. The chamber above the support is unpressurized and is isolated by a second cover which is attached to the PCRV. A pressure relief valve prevents ejection of the cover in the event that mounting bolt failure occurs. Alternate concepts, not shown, provide a pressurized chamber and reinforced concrete cap.

TABLE 4-8

DESIGN AND MECHANICAL DATA, HELICAL INTERMEDIATE HEAT EXCHANGER

Design Data	Units	Primary Side	Secondary Side
Type		Helical Helix Counterflow	
Output	MJ/s	125	
Mass Flow	kg/s	37	36.3
Temperature Entrance	^{o}C	950	240
Temperature Exit	^{o}C	300	900
Operating Pressure	bar	40	42
Mechanical Data per IHX			
Heat Transfer Area	m^2	3880	
Design Pressure	bar	45	
Design Temperature	^{o}C	400-1050	400-1000
Number of Tubes		1990	
Tube Dimension (without corrosion coating)	mm	22.4 Dia x 2.25 wall	
Tube Length	mm	36340	
Tube Pitch	mm	33.6	
Tube Bundle Diameter (inner/outer)	mm	1000/2400	
Center Return Tube Diameter	mm	650/1000	
Weight	tonne	160	

The active tube bundle consists of 1900 tubes of 22.4 mm outside diameter with a wall thickness of 2.25 mm and a tube length of 36.3 m. The tubes are spaced and supported by hangars. At the tube support plate the tubes are bent in the axial direction so as to be perpendicularly attached to the plate. On the bottom they are connected radially to a central hot secondary collector. The overall length of the tube bundle is 16.5 m. The hot secondary collector is supported by the top support plate by an internally insulated hot gas return (exit) tube. The tube bundle is surrounded by a 2.45 m-diameter un-insulated inner hot shell that serves as a close fitting gas distributor and also as a transportation and shipping container for the tube assembly. An insulated outer shell is installed into the PCRV liner and serves as the inner wall for the cold primary gas return passage to the circulator. This outer shell, which interfaces with the coaxial hot duct inlet at the bottom of the IHX, is semipermanently installed and is not normally removed during IHX servicing or replacement. A rigid ceramic insulation is planned for the inside surface of this outer shell.

A bellows expansion compensator for the tube bundle and hot shell liner is incorporated into the design due to the differential expansion experienced during various operating conditions between the tube bundle hot shell liner and insulated outer shell. The IHX is supported laterally at the upper cover on the top, by the inlet duct at the bottom, and by a radial support at about the lower third height to withstand seismic loading effects.

Manufacturing of the heat exchanger is completed within the factory, and no additional fabrication is required at the site. The dimensions and weights of all components are such to permit rail shipment. The hot shell liner is used as a shipping container for the tube bundle and tube support plate assembly and is specially reinforced for this purpose. For installation the hot liner tube bundle assembly is upended with a suitable tilting mechanism and inserted into the prepared cavity after installation of the primary gas inlet ducting and the insulated outer shell.

Periodic tube leak testing and inspection can be conducted after reducing the pressure in the primary circuit and cooling down the component. The primary circuit need not be opened. First, the cover plate of the unpressurized

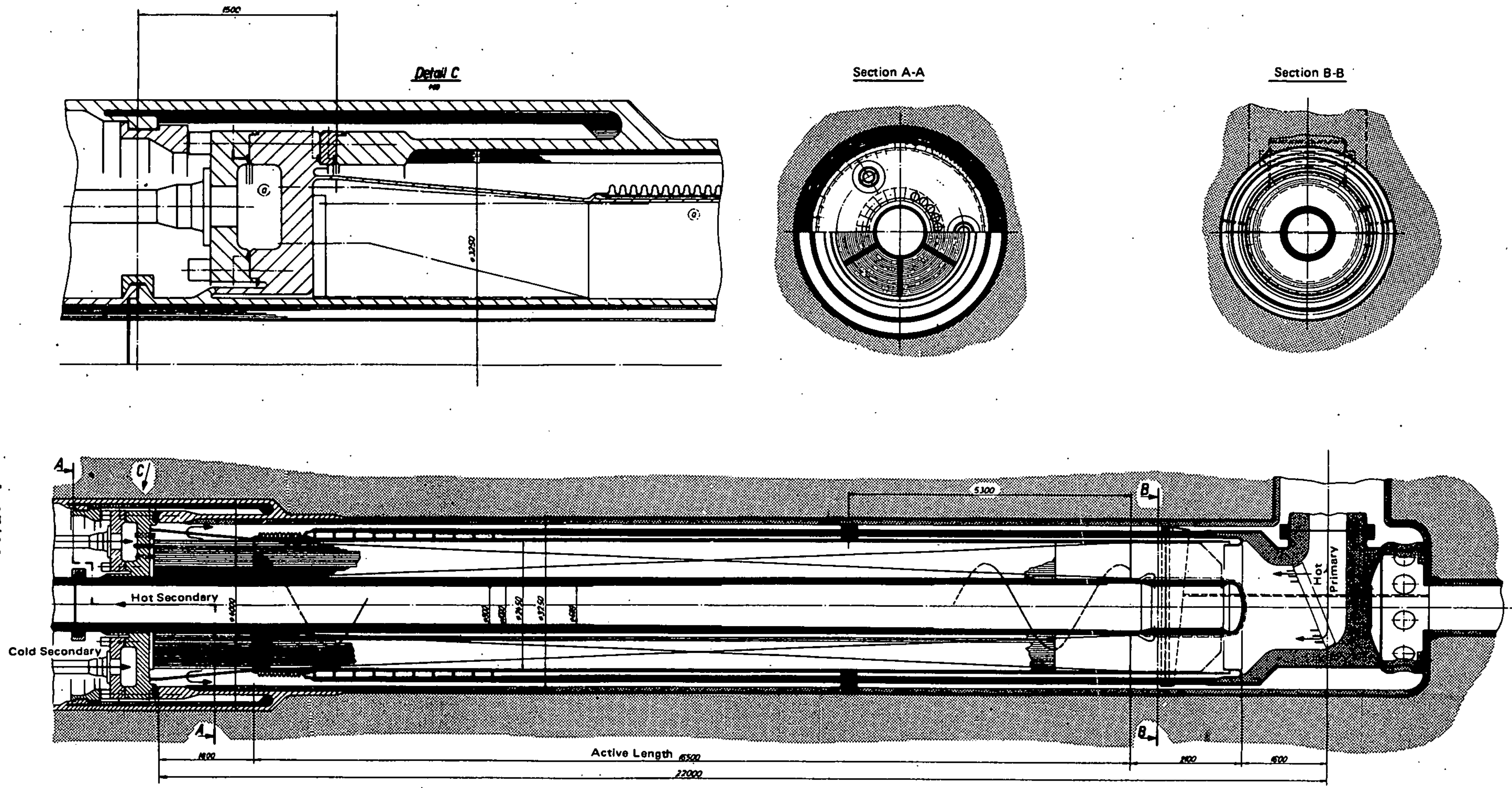


Figure 4-17. Helical Intermediate Heat Exchanger

chamber above the mounting plate is removed, then the secondary side gas entrance and exit pipes and the secondary gas annular cover plate are removed. Access is gained through the tube support plate to the mating IHX tubes and the hot gas center tube. Testing the covers is done by dye penetration methods and ultrasonic testing.

The mounting for the hot gas return tube and the tube itself can be tested ultrasonically after removal of the inside insulation. It is anticipated that the helical tubes can be tested with eddy current procedures presently being developed that may make inspection of individual tubes for defects possible. Critical components in this version of the heat exchanger are the hot gas collector and tube attachments in the hot zone above 800°C, the expansion compensator, the thin-walled tubing, and possibly the mounting covers. Based on the construction details and load factors, Incoloy-800H type material was chosen for the reference design.

All the essential components of the He-heat exchanger--tube sheets and hot tube collectors, as well as the insides of tubes--can be tested without opening the primary circuit. Since all the supply lines are located above the mounting plate, replacement of the He/He IHX can be accomplished without cutting and welding work in the primary circuit.

4.5.1.2 U-Tube IHX

Design and mechanical data for the U-tube IHX is listed in Table 4- 9 and the assembly shown in Figure 4-18. The heat exchanger consists of eight bundles of tubes in a "U" configuration of unequal length legs, spaced vertically around a central pipe. The central pipe serves as a common collector for secondary gas and as structural support for the bundle weight.

Primary He at 950°C is directed via a concentric duct arrangement to an annular inlet plenum (18)* at the bottom of the IHX and then routed to the inlet (20) of each module through 40 vertical riser tubes. The flow in the modules is with primary helium on the shell side in counter flow with secondary helium on the tube side. Primary helium, cooled to 300°C, exits from the upper leg of the "U" modules into the inlet plenum chamber of the primary circulator. Discharge from the circulator is dumped into the heat exchanger

*Numbers in parenthesis refer to Figure 4-18.

Table 4-9

DESIGN AND MECHANICAL DATA, U-TUBE

INTERMEDIATE HEAT EXCHANGER

Design Data	Units	Primary Side	Secondary Side
Type		U-Tube Counterflow	
Output	MJ/s	31,250	
Mass Flow	kg/s	9.25	9.13
Temperature Entrance	^{o}C	950	240
Temperature Exit	^{o}C	300	900
Operating Pressure	bar	40	42
Mechanical Data per Module - 8 modules per IHX			
Heat Transfer Area	m^2	1,090	
Design Pressure	bar	45	
Design Temperature	^{o}C	400-1050	400-1000
Number of Tubes		730	
Tube Dimensions (without corrosion coating)	mm	18 dia x 1.8 wall	
Tube Length	mm	26,300	
Tube Pitch	mm	23	
Tube Bundle Diameter	mm	hot 800 cold 250	
Center Support Tube Diameter	mm	900/1750	
Weight	tonne	30	

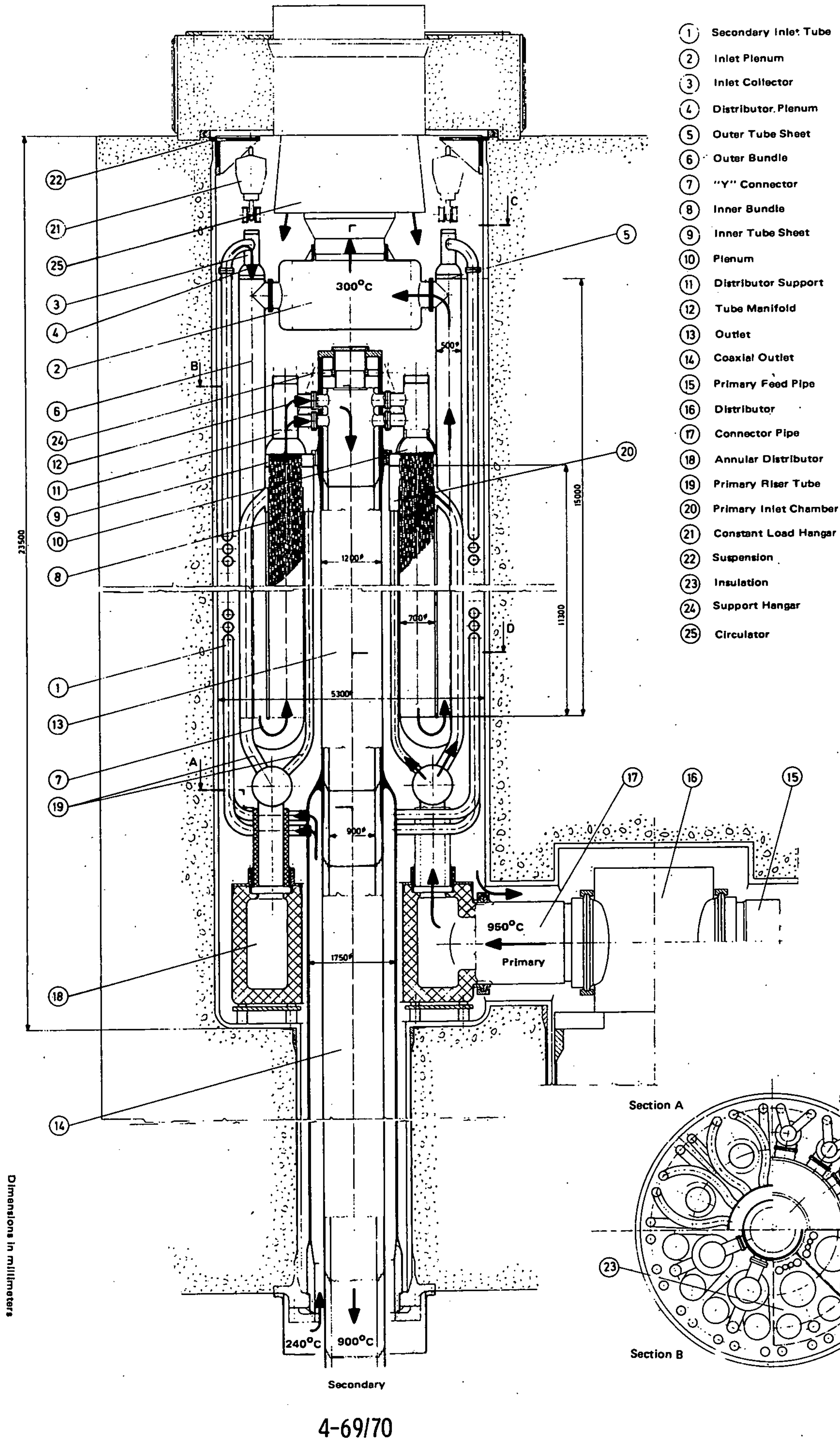


Figure 4-18. U-Tube Intermediate Heat Exchanger

cavity and flows downward to the concentric duct at the bottom of the IHX for return to the reactor. This He is also used to provide some component cooling.

Secondary helium enters and leaves the IHX by way of a concentric duct from the bottom of the PCRV. Inlet gas at 240^{o}C enters the outer annulus of the duct which feeds 32 small diameter tubes (1) that extend to two plenum chambers (4) at the top of each U module. These are the module "cold" legs, and two cold legs are joined at a "Y" connector (7) to one hot leg for each of the eight modules. The secondary gas flows down the module's two cold legs and up the single hot leg in a number of continuous tubes (8) which terminate in tube sheets (5) (9). The secondary helium, heated to 900^{o}C, is collected in outlet plenum chambers (10) and ducted through pipes (12) to the single central outlet duct (13) to the bottom of the PCRV.

The primary structure for the IHX consists of the central coaxial inlet/outlet duct column which is mounted to the bottom of the PCRV. The eight modules are supported from the top of this duct by support hangers (24). The module casings are structural members and are anchored at the cold (upper) end by constant force hangars (21) attached to the top of the PCRV cavity. The annular inlet plenum (18) is separately supported in the cavity liner. To accommodate differential thermal expansion between the components, a number of expansion devices are required. Sliding joints are provided at the connection between the inlet plenum (18) and the primary hot gas inlet collector. The cold gas secondary inlet tubes (1) are wound in helical fashion around the modules to permit expansion, and a sliding joint is provided between the circulator body and the circulator inlet plenum (2). Insulation (23) is provided around the U-tube modules and the hot primary gas riser tubes and also on the inside of the coaxial outlet duct to minimize internal heat losses and maintain reasonable gas-side PCRV liner temperature.

Periodic inspection of the U tube IHX, unlike the helical IHX, requires that the primary system be opened. The primary circulator can be designed to be removed either with or prior to removal of the reinforced concrete cavity cover, after which the circulator inlet plenum chamber may be removed. The U tube manifold end covers are then removed providing access to both ends of the heat exchanger tubes. This feature allows inspection from both ends of the tubes and possible plugging of faulty tubes (neither of which can be provided for the helical tube exchanger design). For testing of the lower end components or the module outer casing, the IHX must be removed, which can be done by removing the center coaxial connection to the bottom of the PCRV and sliding the IHX upward out of the cavity.

In addition to visual inspection, currently available ultrasonic testing or eddy current testing can be used for nondestructive examination, depending upon the type of construction used and the material. Both methods are indirect ones, since they can only detect physical effects, such as cracks due to material changes, and are not direct methods, such as pressure testing. Only a reflective procedure can be used when testing tube inside surfaces with ultrasonic methods. A disadvantage of this method is that stray echoes reflected from the grain boundaries of austenitic materials may cause a high noise level. The actual material condition must be evaluated for each individual case. An ultrasonic system is presently being tested in FRG for a ferritic U-tube heat exchanger with tube dimensions of 25 mm O.D. x 2.3 mm wall and 15.900 x 2 mm wall. Sensing heads mounted in a helical pattern on a carrier are moved along the axis of the tubes and allow the measurement of wall thickness and surface condition. Results to date show an accuracy to 0.0125 mm for the thickness measurements.

Testing methods using eddy current are also limited by the type of materials to be tested. Good results can be achieved from volumetric tests on austenitic materials. On interior surfaces proportionality between fault depth and signal is detectable in the range of 0.1 to 5 mm. Wall thicknesses up to 8 mm are measurable: an inside test coil can detect faults of 5% of the wall thickness and outside coil faults of 10 to 15% of the wall thickness. Ferritic steels permit mostly surface investigations. For straight ducts appropriate probes are available starting at 10 mm diameter. U-shaped pipes with an inside diameter of 12 mm and a bending radius of up to 64 mm are being tested. This testing procedure is very sensitive to surface changes such as those caused by corrosion, because such changes alter the electrical conductivity.

The movement of the probes (rotation and transverse motion) and the centering within the helix for helical heat exchangers make the testing rather difficult. Tests, however, are being conducted in order to solve these identified problems.

4.5.2 INTERMEDIATE HEAT EXCHANGER EVALUATION

Comparative evaluation results of the helical and U-tube IHX are reported in a General Electric Co. design study in Reference 18. This computer-optimized configuration study resulted in a U-tube configuration rated superior to a helical IHX in all categories which consisted of safety-related mechanical design, thermal hydraulic design, and cost aspects. The major advantages of the U-tube exchanger included ease of in-service inspection, ability to replace a module, leaky tube isolation, and cost differences resulting from less tube weight and significantly smaller tube sheets. One of the major inspection features was the ability, in the reference study, to borescope-inspect the U-tubes without opening the primary loop. This advantage is not available with the current FRG design but could be incorporated with redesign of the secondary loop flow. An advantage retained in the FRG design is the ability to pressure or vacuum test for and then isolate leaks by plugging both ends of the faulty tube. Inspection methods are being developed for the helical type exchanger but are much more complex, and the sensitivity is reduced.

A problem in most U-tube heat exchanger designs is the thermal stress and deflection due to restraining the ends of the U-tube, considering the difference in thermal growth rates between the cold and hot legs. This has been greatly reduced in the FRG design through the use of a constant load hanger to support one end of the U-tube and by making the cold leg of the exchanger longer than the hot leg to reduce the differential expansion.

As indicated in Section 4.5.1.1, the helical tube IHX can be completely assembled and tested in the fabrication plant and requires a minimum of field assembly and testing. In contrast, the U-tube IHX has a large number of joints to be made in the field. The current design shows seven flange joints for each module plus eight sliding seal joints for the total exchanger. These joints could become potential problems.

The problem of tube vibrations will appear in each of the tube designs and must be evaluated and the results substantiated by test. Since details of the designs (such as tube length between supports and the method of tube spacing/support) are not available, no evaluation can be made of this concern other than that it must be considered.

Similar to the situation for the steam reformer, the major effort remaining is to provide an intermediate heat exchanger design to be tested in sufficient size and power rating to demonstrate the operating, maintenance, life, and safety requirements of a nuclear plant installation. Efforts in this large-scale test area by the Federal Republic of Germany is further discussed in Section 2.3. Of immediate concern are the screening and selection of candidate materials for IHX components at the high temperatures for the long life and special environment required and providing the materials properties required for design.

4.6 STEAM GENERATOR

4.6.1 STEAM GENERATOR DESCRIPTION

The function of the steam generator (i.e., the production of high-temperature steam) is required for any steam gasification or hydrogasification PNP, as well as for the HTR-K plant. The steam generator is used in the primary loop for the hydrogasification PNP and HTR-K. The intermediate loop in the steam gasification PNP has a steam generator, but it will not be discussed here. The operating characteristics for the two uses are summarized in Table 4-10 and show a general similarity between the two units. The major difference in operating data is the power per unit and primary-side helium pressure. A mechanical arrangement difference does exist as a result of primary-loop layout. This is shown by comparing Figure 2-1 for the HTR-K plant and Figure 3-7 for the PNP plant. When used with the PNP steam reformer, the hot primary gas inlet is at the top end of the steam generator and the primary circulator at the bottom. In HTR-K, however, the hot primary gas inlet is at the bottom of the steam generator and the circulator at the top. In neither steam generator is internal reheat supplied using helium. Reheat is not needed for the coal gasification process for the smaller electric output required of the PNP turbines. Helium/steam reheat was not introduced into the HTR-K plant in order to simplify the heat exchanger design and to improve reliability. External reheat using a steam/steam heat exchanger is used in the HTR-K cycle with a resultant loss in plant efficiency of two percentage points.

TABLE 4-10

STEAM GENERATOR OPERATING CHARACTERISTICS

Parameter	Units	PNP	HTR-K
Number of SGs	-	6	6
Power of each SG	MW	385.4	500
Helium inlet temperature	°C	800	700
Helium outlet temperature	°C	300	260
Helium inlet pressure	bar	39.1	60
Helium outlet pressure	bar	38.8	59.55
Helium mass flow/loop	kg/s	148.1	221.7
Water inlet temperature	°C	180	185
Steam outlet temperature	°C	540	515
Water inlet pressure	bar	120	210
Steam outlet pressure	bar	115	175
Output steam/loop	kg/s	143.3	200

General design criteria for the steam generators include:

- 40-year design life (load factor $\sim$0.8).
- Helium-side pressure drop $\leq$ 0.45 bar to permit acceptable circulator size.
- Outer diameter limited to 4 m for transportation purposes.
- Minimum field fabrication.
- Gas ducts in the PCRV to be exposed to cold helium (300^{o}C for PNP, 260^{o}C for HTR-K).
- In-service inspection at least for the high-temperature superheater section and supporting structure.

4.6.1.1 HTR-K Steam Generator

Selected mechanical data for the HTR-K steam generator is listed in Table 4-11 and calculated data in Table 4-12. The general arrangement is shown in Figure 4-19. After being heated in the reactor to 700^{o}C, helium is ducted to the steam generator through a coaxial hot duct to an annular chamber at the bottom of the steam generator. The gas enters a central duct through holes from the surrounding chamber and flows upward on the shell side of the counterflow straight superheater tubes. The central gas duct serves as the superheater shell. At the top of the steam generator the primary helium turns 180^{o} and flows downward through the helical sections of the economizer and evaporator and the first stage of the superheater. The flow is separated from the main superheater straight tube flow by the superheater shell. An additional shell is placed on the outside of the helical tube section. After discharging from the economizer at 260^{o}C, the cooled primary helium is again turned 180^{o} and flows upward through an annular passage (provided by the helical section shell and the outside wall of the steam generator) to the inlet of the primary circulator. Circulator discharge gas is directed downward between the PCRV cavity liner and the outside wall of the steam generator for cooling purposes. Part of this flow is returned to the core through a duct at the top of the core, and part is directed to the bottom of the steam generator to cool the annular inlet chamber and hot gas duct.

TABLE 4-11

PNP AND HTR-K STEAM GENERATOR MECHANICAL DATA

Parameter	Units	PNP Straight Section	PNP Helical Section	HTR-K Straight Section	HTR-K Helical Section
Diameter of Central Section	m	1.7	-	1.8	-
Outer Diameter of Bundle	m	-	3.4	-	3.56
Length of Straight Bundle	m	13.7	47.2	11.2	-
Height of Helix Bundle	m	13.8	3.9	-	8.25
Outside (Inside) Diameter of Tubes	mm	25 (19)	25 (21)	25 (16)	22 (16)
Number of Straight Tubes	-	380	-	430	-
Number of Helix Tubes	-	-	380	-	430
Spacing of Tubes, Radial, (Longitudinal)	mm	72 (0)	40.5 (36)	67 (0)	42 (31)

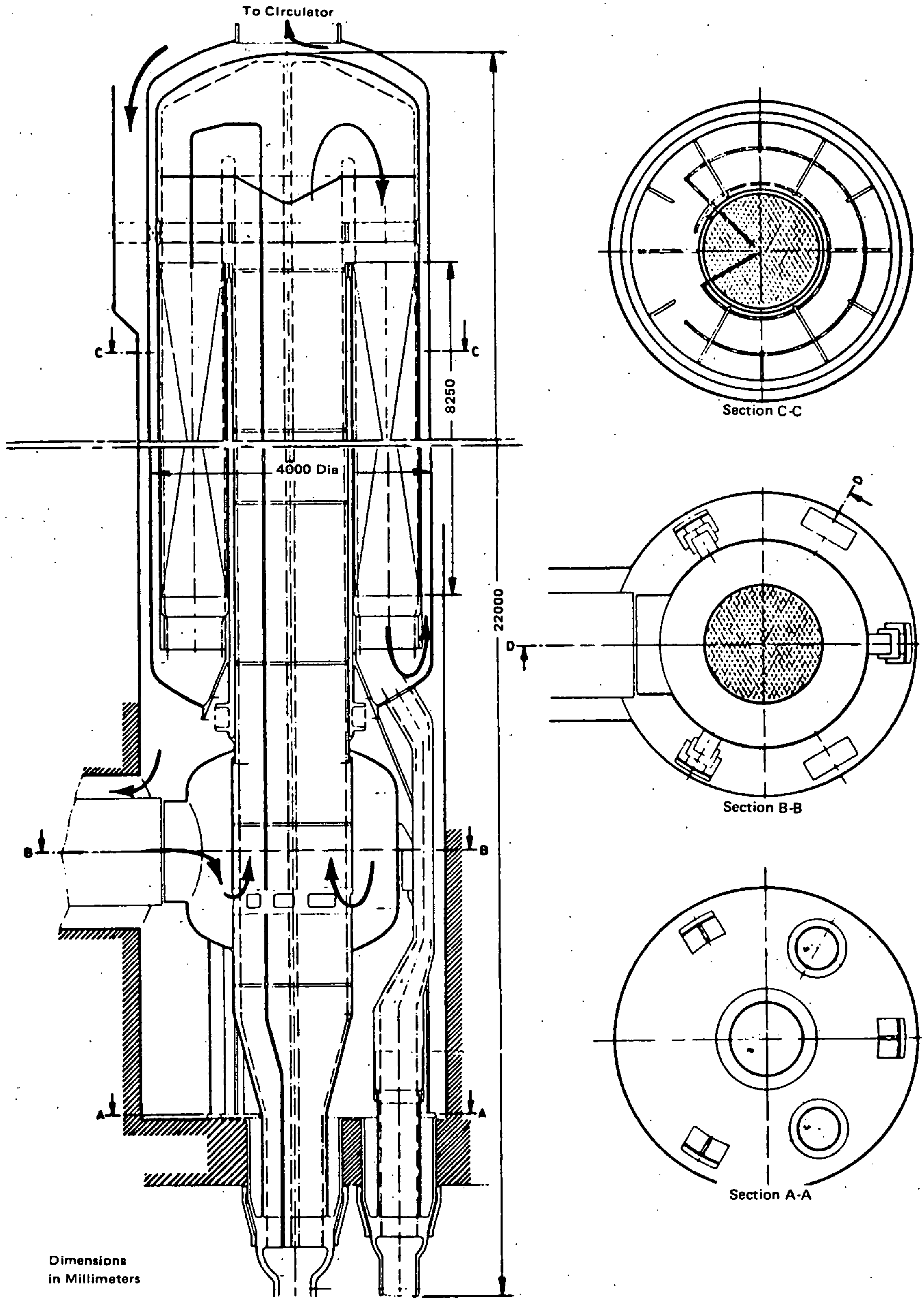


Figure 4-19. HTR-K Steam Generator

TABLE 4-12

PNP AND HTR-K STEAM GENERATOR CALCULATED DATA

Parameter	Units	PNP	HTRK
Total heat exchanger area/loop	m^2	1827	3959
Mean heat flux	$\frac{KW}{m^2}$	200	129
Max. wall temperature	^oC	~ 600	590

Feedwater enters through two pipes with tube sheets, supplying tube bundles that join with the helical tubes of the economizer/evaporator section. The steam leaving the helical bundle is led to the straight tubes of the superheater tubes inside the central portion of the heat exchanger. All flow is counterflow. The tubes of the helical bundle are supported in support structures similar to the THTR or Fort St. Vrain designs. The straight tubes are fixed at the bottom of the assembly and spacers are used to control vibration. The hot steam is taken out at the bottom of the steam generator in a single central pipe which is connected to all the straight tubes. The main structural support is provided by columns from the cavity floor.

4.6.1.2 PNP Steam Generator

Selected mechanical data for the PNP steam generator are listed in Table 4-11, and the general arrangement is provided in Figure 4-20. Some calculated data are listed in Table 4-12.

The design of the PNP steam generator is similar to that of the HTR-K. The main difference is that hot helium at 800°C rather than the 700°C of the HTR-K enters at the top, coming from the steam reformer through a concentric hot/cold gas duct. It leaves from the bottom to enter the primary circulator. In the central region is located the superheater straight tube section followed immediately below by the helical section evaporator and economizer. There is counter flow in the outer helical section parallel flow in the straight tube section. Cold helium at 300°C from the circulator discharge is directed upward to cool the heat exchanger outer shell and the incoming helium hot gas duct. Feedwater inlet and steam outlet are similar to those of the HTR-K design. The primary structural support for the steam generator tubes is from the bottom of the steam generator cavity. Primary shell support is from the cavity liner in the upper region.

4.6.2 STEAM GENERATOR EVALUATION

Information on helical-type steam generators is available from a number of gas-cooled reactors in France and Britain and from Fort St. Vrain in the U.S. It is considered that some testing, however, is necessary to confirm the feasibility of the proposed straight tube concept. Vibration testing of this section of the assembly should be done, and flow distribution testing is also needed. The requirement for in-service inspection (at least of the superheater tubes and the supporting structure) will necessitate changes from previous design experience. In the available literature, no stress analyses for the stationary and transient operations are given. Hence, the thicker walled supporting structures should especially be analyzed. It is stated that ferritic steels are used in the helix bundle and that ferritic steel and Incoloy 800 are used in the hot part of the superheater. The Incoloy material needs further qualification for long-term applications in helium circuits, especially for a temperature of 800°C. This work has started as a part of the FRG national program for the PNP Project. Methods for in-service inspection are available today for straight tubes using knowledge from the field of light

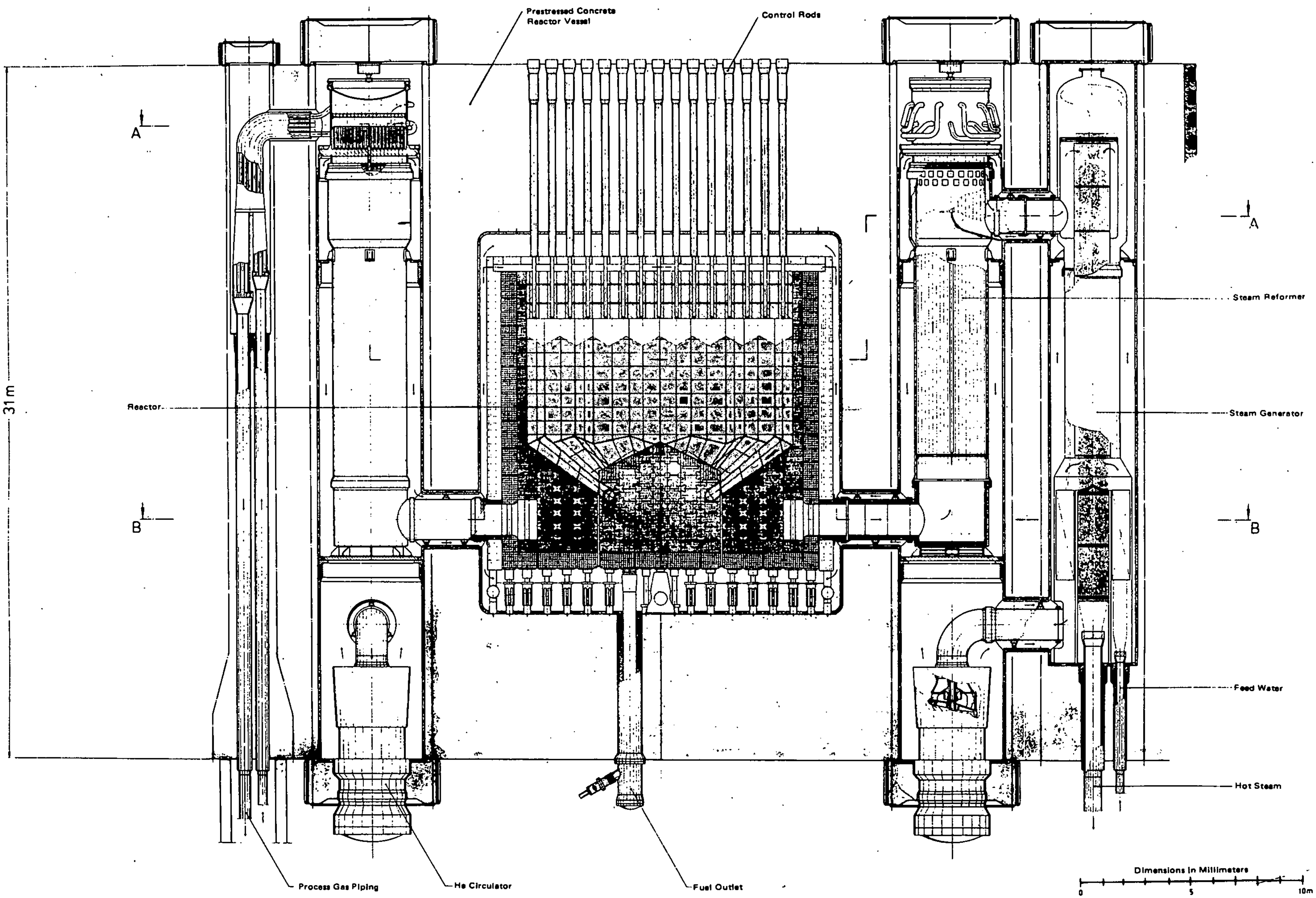


Figure 4-20. PNP Steam Generator (HKV Plant)

water reactors. Helical bundles today can only be tested if their length is less than 20 m and if the number of turns is no more than two. The tubes in the designs are much longer and cannot be tested with current techniques. The future requirements of safety authorities in this field are unknown, and perhaps further change to straight tubes in the economizer and the evaporation may be required. This would cause more space to be needed for the steam generator within the multicavity vessel. It can be generally stated that the basic elements of the proposed steam generators are known; however, some confirmatory tests remain to be done.

SECTION 5

AUXILIARY SYSTEMS (DESCRIPTION AND EVALUATION)

5.1 GENERAL CONSIDERATIONS

The nuclear portions of the PNP and HTR-K concepts require numerous support systems for proper long-term operation. Three of these support, or "auxiliary," systems have been selected for discussion in this section because of their central importance and the availability of descriptive information. Other auxiliary systems, such as the various cooling water systems, are also important, but sufficient data was not available.

The following subsections describe the Fuel Handling System, After-Heat Removal Systems (NWA), and Gas Purification Systems.

5.2 FUEL HANDLING SYSTEM

5.2.1 INTRODUCTION

The fuel handling system for the 3000 MW core must introduce fresh fuel at the top of the core and remove the spent fuel from the bottom. The fuel loading system consists of a network of tubing, plus numerous valves and devices for tracking the fuel elements' progress and for guiding them through the network. The fuel removal system receives spent fuel through six exits in the bottom of the core and transports them to containers that then empty into carts, which transport the spent fuel to storage.

The fuel handling system described below is the design done by GHT, and it is shown in Figure 5-1. (The descriptions in the following two subsections are keyed to the callouts shown on this figure.) Some differences exist between this design and the designs of KFA and HRB. These differences will be discussed in Section 5.2.4, "Design Alternatives."

5.2.2 FUEL LOADING SYSTEM

The reactor operates on a once-through-then-out fuel cycle, which means the fuel is not recirculated. Hence, no forced circulation system is needed to carry the fuel back to the top of the core, and the elements can be loaded strictly by means of gravity. This is a significant simplication over the AVR and THTR systems, which recirculate their fuel.

The fuel loading system consists of three separate similar systems, one for the outer core zone and two for the inner core zone. Fresh fuel elements are held in three storage containers (1) above the PCRV. When the loading system is operated, the fuel elements are removed one at a time from the storage containers and fall into the tube directly below the container. In this tube, they pass through a counting device and then through two shutoff valves (2). This counting device and all others in the loading system are used to monitor the path of the fuel elements. The shutoff valves control the flow of fuel. After these shutoff valves, the fuel elements reach a "branching point." The outlet tube of the storage container for the outer zone "branches" into four tubes, while the outlet tube of each of the inner zone storage containers "branches" into two tubes. At each "branching" there is a switching device which either lets the fuel element pass or directs it into the tube which has "branched" from the outlet tube of the storage container. Counting devices are located on each of the "branched" tubes.

Each "branched" tube carries the elements to a device called a distributor (3). The distributor's function is to direct the fuel element to the proper loading inlet tube through which the element enters the core. It has an angled piece of tubing which pivots and aligns itself with one of six (in one case, seven) loading inlet tubes. A total of 43 loading inlet tubes penetrate the PCRV ceiling through perpendicular, circular channels. Toward the top of each tube are a shutoff valve (4) and a safety valve (5). The safety valve would be activated and would seal off the primary system against leakage if a leak occurred in the loading system. Also, there is a counting device on each inlet tube just before it enters the PCRV top slab.

During operation, the entire loading system is pressurized to slightly above the primary system pressure. The loading system has a maximum capacity of 2900 elements/hr.; this is more than sufficient to accommodate the daily load of 2600 elements during a one and one-half hour period.

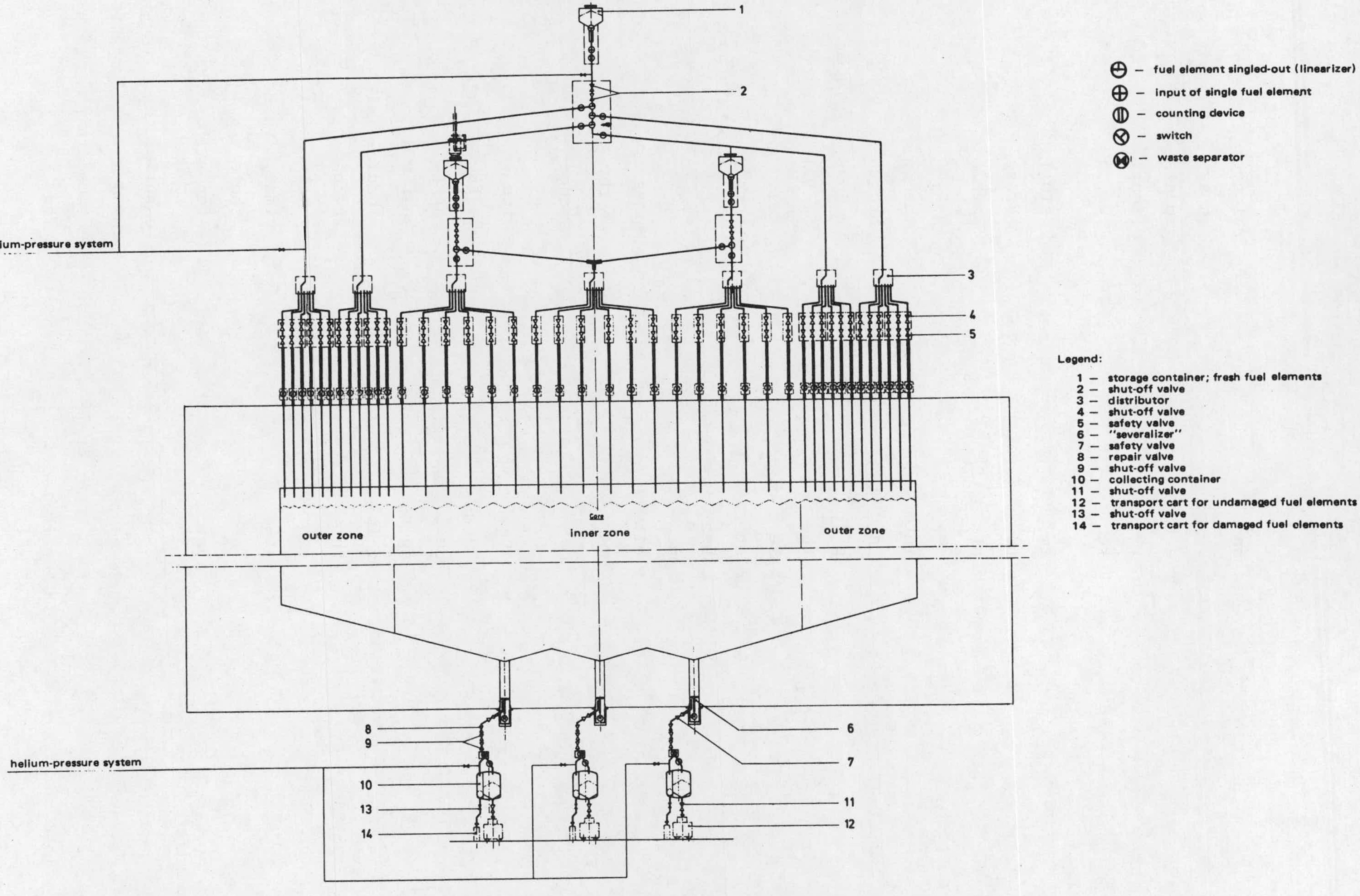


Figure 5-1. Fuel Handling System Schematic

5.2.3 FUEL REMOVAL SYSTEM

The fuel removal system consists of six identical systems, one for each fuel element exit. The fuel elements pass through the exits in the core bottom into fuel exit chutes, which are pipes with a diameter eight times larger than the fuel element diameter. At the bottom of the exit chutes is a device known as a "severalizer" (6), which allows small groups of fuel elements to fall from the exit chute into a tube. The "severalizer" is a disk with a hole large enough to allow one element to pass through. The disc is rotated by an electric motor, and a fuel element falls through only when it is positioned over the hole. Along the tube which receives the fuel elements from the "severalizer" are four valves. The first valve is a safety valve (7) like the one used on the loading inlet tubes. The second is a repair valve (8) used to isolate the system during maintenance. The other two are shutoff valves (9). After passing through all these valves along the tube, the fuel elements arrive at the damaged element separator. In the separator, damaged and undamaged elements are separated mechanically and pass by means of gravity to a subdivided collecting container (10). Just before the collecting container, there is a counting device to measure the number of undamaged elements. The fuel removal system has a capacity of 2100 elements per hour.

Each collecting container has a capacity of 8000 to 9000 fuel elements and operates under primary system pressure while fuel is being removed from the core. To empty a collecting container, the corresponding shutoff valves are closed, and the container is depressurized. Next, a spent fuel cart for damaged or undamaged elements (12 or 14) is connected to the exit tube; the shutoff valve (11 or 13) is opened, and the spent fuel falls into the cart for transport to the storage area in the reactor auxiliary building. The storage capacity for spent fuel will be 1.5 to 2 core loads, which is sufficient for approximately 6 to 8 years' operation. The storage area for fresh fuel will also be located in the reactor auxiliary building and will have a capacity sufficient for one year of refueling.

5.2.4 DESIGN ALTERNATIVES

The fuel handling system designed at KFA has two basic differences from the GHT design described above. First, the KFA design uses a "lock system" on the outlet tube from the loading containers. This lock system is composed of three shutoff valves which open sequentially to allow the elements to pass through, and then close. Thus, two of three valves are always closed as the fuel elements pass through the lock system. This lock system would replace the shutoff valves which are located on the loading container outlet tubes and the loading inlet tubes in the GHT design. The second difference between the KFA and GHT design is that the KFA design does not utilize damaged element separators in the fuel removal system. Since the fuel is not recirculated, it is not necessary to separate the damaged elements; therefore, the KFA designers considered the damaged element separator unnecessary.

The HRB fuel handling system differs radically from the KFA and GHT designs because most of the system components are located outside of the reactor containment building. This design requires the use of containment isolation valves on all fuel handling lines which penetrate the containment. It also requires much greater lengths of tubing than the KFA or GHT designs. The result of the HRB choice to locate most components outside the containment is a significant increase in system cost compared to the other designs.

5.2.5 FUEL HANDLING SYSTEM EVALUATION

All the components required for the fuel handling system are well known and tested from AVR operational experience and from the THTR development program. Also, the fuel handling system for the large PBR operating on an OTTO fuel cycle is simpler than the systems required for the AVR and THTR, since it does not recirculate fuel elements. Thus, the fuel handling system could be built and is available at this time.

5.3 AFTER-HEAT REMOVAL SYSTEM (NWA)

5.3.1 INTRODUCTION

The various pebble bed reactor concepts (PNP, HTR-K, or HHT) must all meet the safety criteria specified in German licensing regulations. A fundamental requirement of those safety criteria is that reactor residual heat be removed under all conditions. The normal heat transfer loops are not sufficiently reliable for all accident situations. Therefore, alternative heat removal systems must be provided. Gas-cooled reactors do not require emergency core cooling systems (ECCS) of the type used in light water reactors, due to the large core graphite heat capacity and temperature-resistant ceramic fuel, which allows some delay in removing the core residual heat. The pebble bed reactors under consideration are all designed to meet the need for post-accident heat removal by use of a Nachwärmeabfuhrsystem-NWA System (After-Heat Removal System).

5.3.2 GENERAL CHARACTERISTICS

The NWA Systems, whether for PNP or HTR-K, have the same general components and design bases. The sytem is made up of redundant primary helium loops located within the PCRV, and secondary water loops which dissipate the heat to water/air heat exchangers. The general flow scheme for NWA Systems for PNP and HTR-K is shown on Figure 5-2. Each of the four redundant NWA trains has enough capacity to remove 50% of the design basis after-heat under worst-case conditions. They therefore meet the German licensing requirement that the NWA System meet its design purpose, considering one train out for repairs plus failure of a second train on demand. It is interesting that the HHT design includes three NWA trains, each sized for 100% heat load, which therefore, also meet the licensing criteria.

The typical NWA helium loop consists of an electric motor-driven helium circulator, a reverse flow limiting check valve, a helium/water heat exchanger, and appropriate gas ducting. The secondary loops include water piping between the helium/water heat exchanger and a water/air heat exchanger outside the reactor building, water makeup provisions, circulating pumps, and overpressure control systems. The NWA system also includes appropriate actuation and control subsystems.

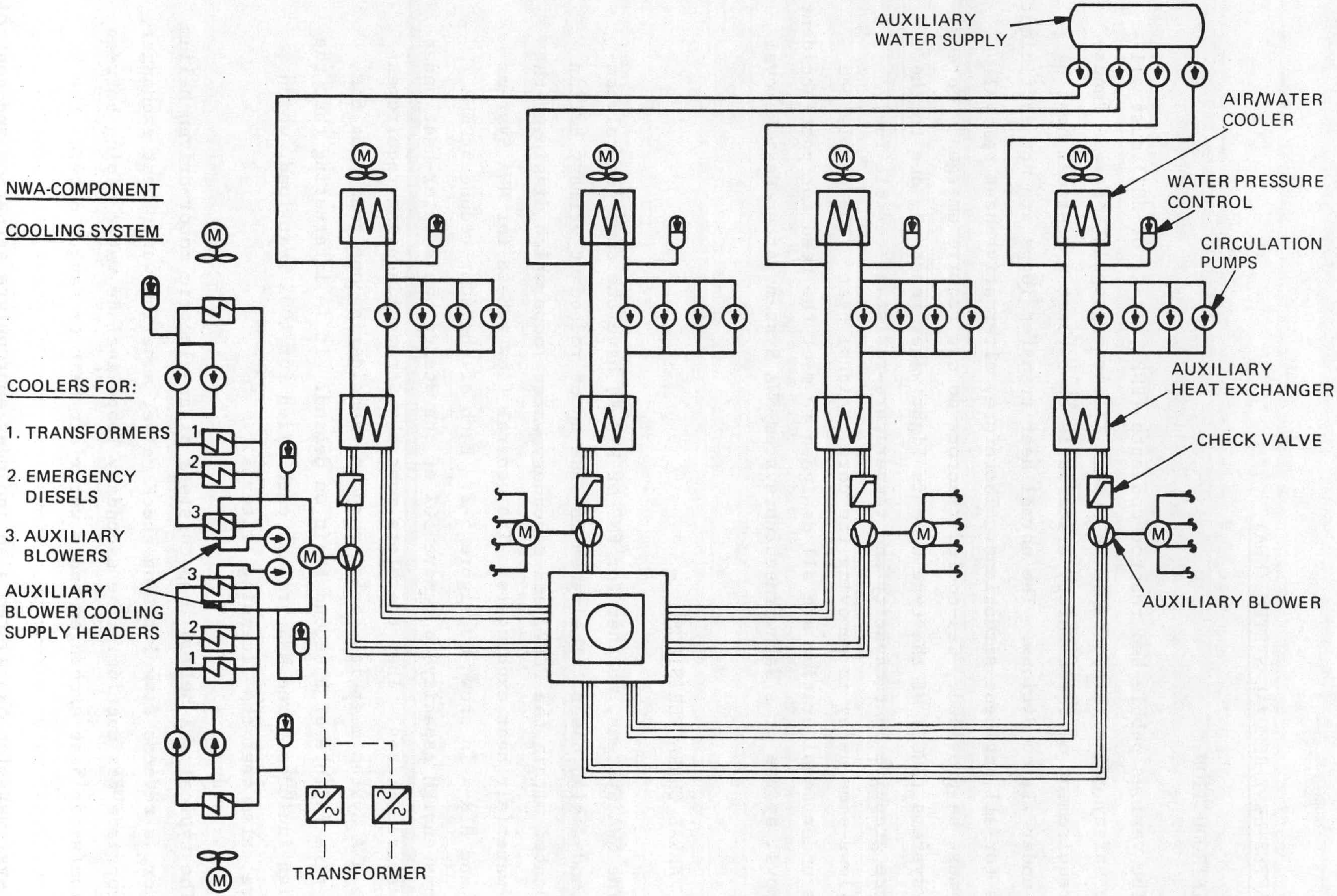


Figure 5-2. After-Heat Removal (NWA) System Schematic

5.3.3 NWA SYSTEM ARRANGEMENT FOR HTR-K

Figure 5-3 shows the arrangement of the NWA System primary loop within the PCRV. It can be seen that the auxiliary heat exchanger and circulator for each of the four redundant trains are located in separate cavities with the circulator on top of the heat exchanger. The downward flowing hot helium in the core is discharged from the bottom and passes to the NWA heat exchanger through a coaxial duct. The helium gives up heat while passing upward through the heat exchanger to the suction of the NWA circulator. Cool helium is then discharged back to the core inlet through a separate return gas duct, although part of the helium is bypassed downward to cool the liner in the NWA cavity, and the outer annulus of the inlet duct. The bypass cooling gas then flows upward around the thermal shields to the core inlet.

During normal operation of the HTR-K, core inlet pressure is higher than the outlet pressure, due to operation of the main gas blowers. This differential pressure causes some reverse flow back through the NWA heat exchanger and circulator, although the check valve at the NWA blower inlet limits the quantity (See Section 6.2.7).

Maintenance and inspection of the NWA System is performed by opening the top cover on the NWA cavities. The blower has to be removed in order to gain access to the heat exchangers below, thus making tube inspection somewhat difficult and time-consuming.

5.3.4 NWA SYSTEM ARRANGEMENT FOR PNP

The NWA system for PNP plants is arranged in a manner different from that described above for HTR-K. Figure 5-4 shows the current reference design. The four redundant trains are again located in separate PCRV cavities; however, the PNP design places the NWA circulator below the heat exchanger and utilizes a cool gas return duct which is below the coaxial hot gas inlet duct.

The hot helium passes from the core outlet at the bottom of the pebble bed through a hot gas coaxial duct into the inlet of the NWA heat exchanger where it flows upward. After giving up its heat, the cooler helium makes a 180° turn and passes downward to the blower inlet via the annulus around the NWA heat exchanger shroud. The electric motor-driven blower discharges the cold high-pressure helium upward into an outlet plenum, from which it flows through a cold gas return duct to the core bottom and upward around the thermal shield to the core inlet. As in the HTR-K design, some cold helium is bypassed from the cold gas return to cool the coaxial duct.

During normal plant operation the primary loop blowers force some reverse flow through the NWA system (limited by NWA blower inlet check valves), which keeps it cool.

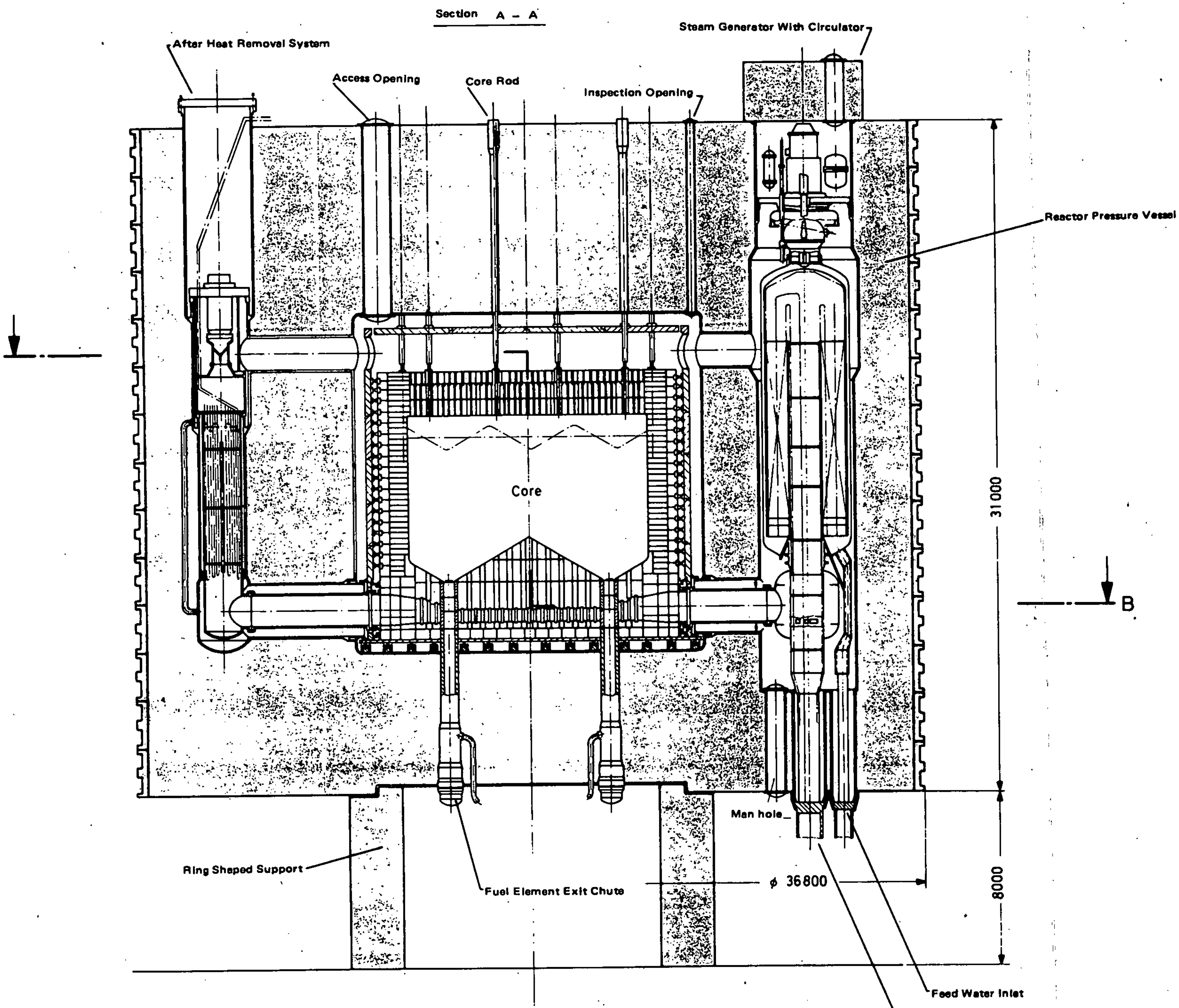
Inspection and maintenance of the PNP system is made easier than for HTR-K by the location of the heat exchanger above the blower. The exchanger tubes can be eddy-current-tested without removal of the blower. Blower performance can be checked in place.

5.3.5 NWA SYSTEM DESIGN BASES

As discussed above, the NWA system must be capable of removing reactor residual heat during the various design basis accidents. The worst case accident from an overall standpoint is the design basis depressurization accident (DBDA). For HTR-K, the DBDA implies a helium loss sufficient to result in a depressurization from 60 bars to 2 bars in about three minutes (4.67 psi per second). The NWA systems are sized such that each train can remove 50% of the design heat load under the worst hypothetical conditions, wherein the pressure is 1 bar and some air ingress has occurred. Should the more likely case accidents with a pressurized containment occur, the NWA trains can remove significantly more than 50% of the design heat load. In the worst case, two operable trains will meet all requirements.

The German licensing process requires that the probability of a light water reactor meltdown be on the order of 10^{-7} per reactor per year. The probability of a major loss-of-coolant accident is about 10^{-3} per reactor per year, which places the required ECCS reliability of 10^{-4} per demand. The NWA system is designed to meet the ECCS reliability specification of 10^{-4} per demand.

When the NWA System actuates, the NWA circulators start and accelerate to rated speed. At the same time the primary circulator inlet valves move shut. Without these valves shut, most of the NWA flow would bypass the core via reverse flow through the main loops; therefore, the capacity of the NWA circulators is such that a single failure of a primary circuit reverse flow



Note: All dimensions in millimeters

Figure 5-3. NWA Arrangement for HTR-K

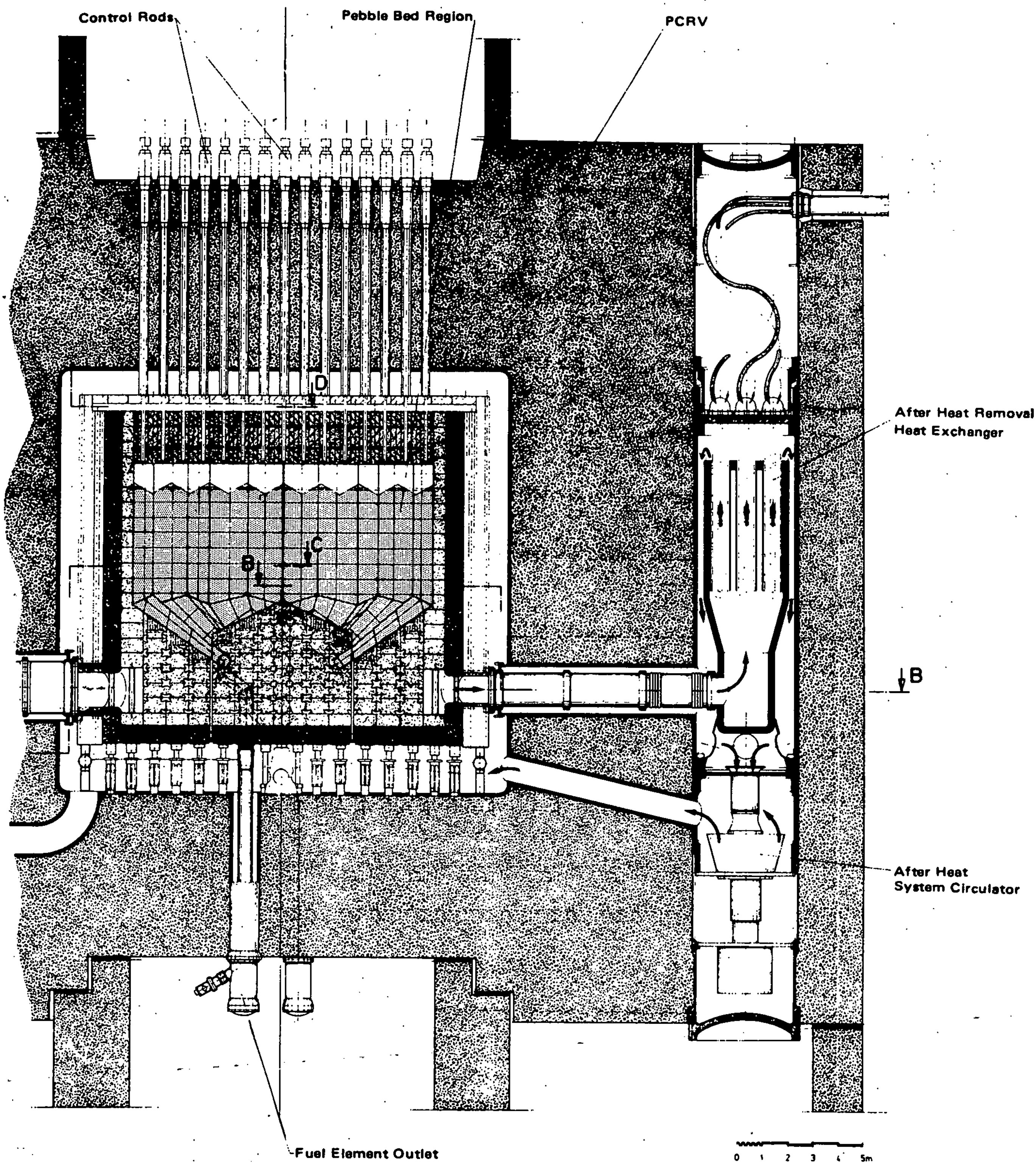


Figure 5-4. NWA Arrangement for PNP

shutoff valve can be accommodated. Some reverse flow occurs in those main loops with shut inlet valves, in order to keep the liner and components cooled. That reverse flow allowance is also included in NWA size requirements. The design NWA heat load is about 2% of rated power (i.e., 60 MWth). That number is somewhat greater than the amount of heat to be dissipated; this is discussed below. The reference curves are for the PNP core but are similar to curves for HTR-K.

A study was performed at KFA to determine the thermal performance during after-heat removal with various mass flows. The objectives were to define the minimum mass flow which would guarantee stable core flow during the entire period of after-heat removal, and to quantify the formation of transient hot spots.

Calculations were made with an analysis system consisting of coupled programs:

- A 2-D transient heat conduction program
- A 2-D quasi-stationary convection program
- A 2-D quasi-stationary gas temperature program.

A significant amount of data was generated, demonstrating that 2% flow is required in order to maintain a stable core flow and avoid local hot spots. Figures 5-5 and 5-6 show gas streamlines at time 4.5 hours for 1% and 2% NWA flow, respectively. The latter figure can be seen to have significantly better flow distribution. Figures 5-7 and 5-8 provide an overall view of the temperatures at various locations in the PNP core during the transient at 1% and 2% NWA flow, respectively. Again the 2% NWA flow is seen to result in superior performance. As the result of these considerations, the design NWA heat load is currently considered to be 2%. It should be mentioned that, although the design loads are fairly certain for the HTR-K, some future work remains for the PNP. Water ingress accidents (caused by steam generator failures) result in additional heat removal requirements due to the steam/graphite corrosion and the need for rapid core cool-down. It is possible that the PNP design basis could change.

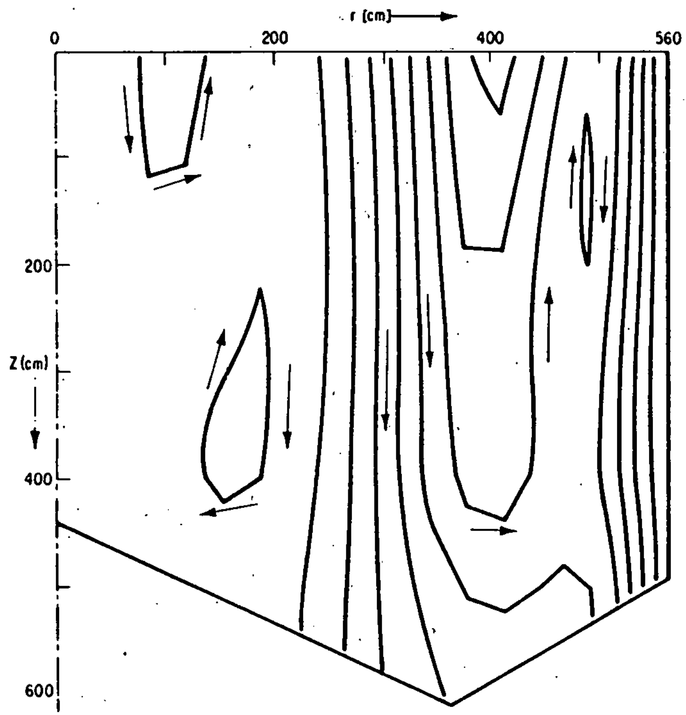


Figure 5-5. Core Streamlines with 1% NWA (4.5 Hours)

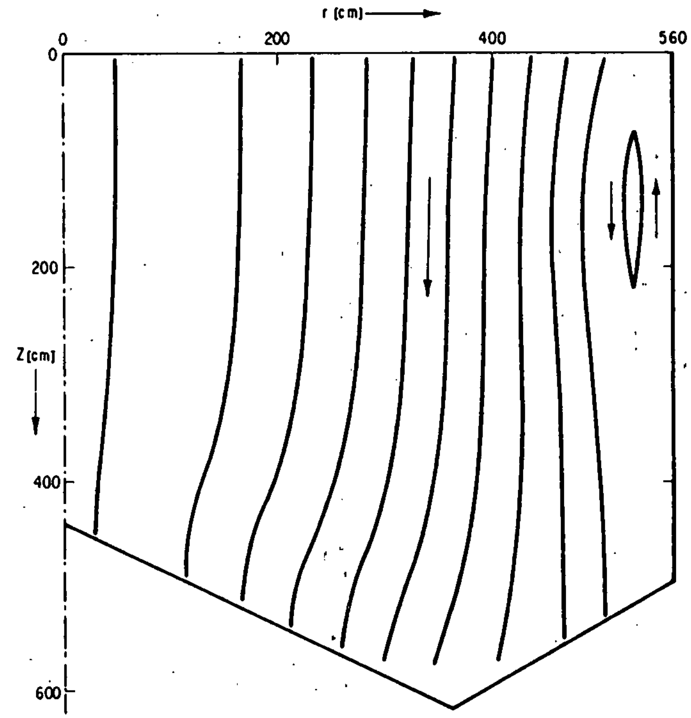


Figure 5-6. Core Streamlines with 2% NWA (4.5 Hours)

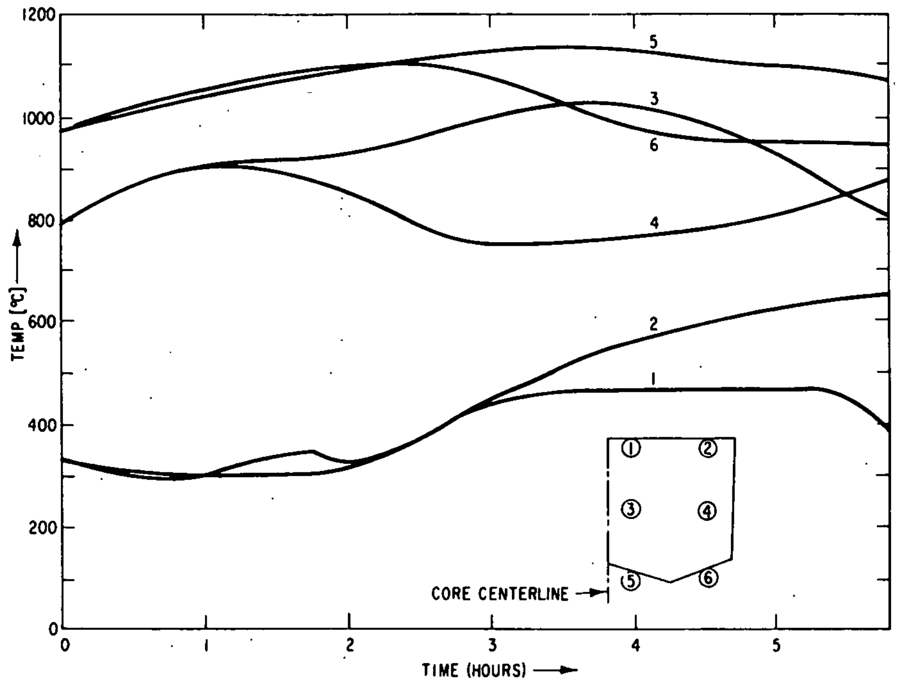


Figure 5-7. Core Thermal Transient with 1% NWA

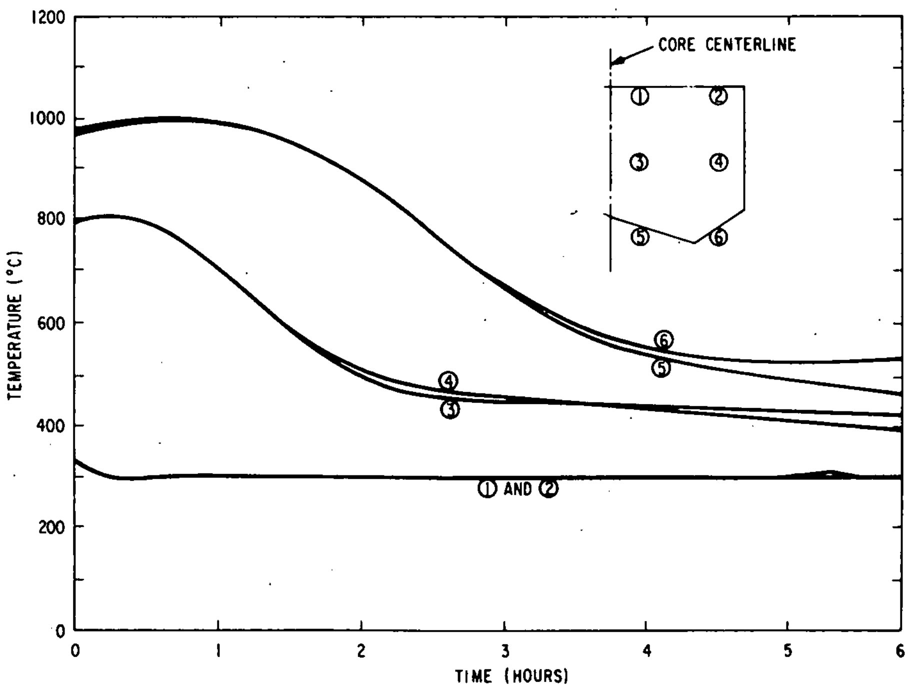


Figure 5-8. Core Thermal Transient with 2% NWA

5.3.6 COMPONENT DATA

5.3.6.1 NWA Circulators

The NWA circulators are presently considered to be of single-stage axial construction with speed regulation for flow control. The sizing is based on the worst case conditions of 1 bar pressure and a helium air mixture. These conditions result in the highest volume flow and the highest power requirements. As mentioned earlier, the design also considers bypass through one main circuit. Each blower will be driven by an electric motor that is powered by a separate and independent 2 MW emergency diesel generator through a frequency regulator. Major design parameters are shown below:

NWA Blowers

	PNP	HTR-K
Number	4	4
Drive	Electric	Electric
Power (at motor terminals) (KW)	616	875
Minimum Suction Pressure (bar)	1	1.4
Minimum Suction Temperature (°C)	200	207
Control Method	Speed	Speed
Flow Rate (kg/s)	8.6	11.6

5.3.6.2 NWA Heat Exchangers

The NWA Heat Exchangers are module-type U-tube heat transfer devices. Each of the four exchangers contains seven U-tube modules which are arranged next to each other in a circular array within the NWA cavity. As discussed earlier, the HTR-K design places the heat exchangers below the NWA blowers. This arrangement forces the water to be piped in around the blower from the top of the PCRV. Since the hot helium passes up from the bottom, the cooling water flows countercurrent to the helium until the U-bend, then flows upward in a concurrent manner.

The PNP arrangement uses the same flow scheme (i.e., countercurrent downward and concurrent after the U-bend). The main difference is that the heat exchangers are above the blowers and are much easier to inspect.

The current design parameters are listed below. The heat exchangers are sized for the depressurized case. The HTR-K analysis indicated that the 2% heat load (60 MW) is sufficient for rapid cool-down following a water ingress accident. The PNP heat exchangers have not yet been analyzed for adequacy during the water ingress accident.

NWA Heat Exchangers

	PNP	HTR-K
Number	4	4
Capacity (MW)	27	31.6
Type	U-tube	U-tube
Heat transfer area (m^2)	545	920
Gas inlet temperature ^oC	1000	880
Gas outlet temperature ^oC	250	230
Secondary water flow kg/s	80	200
Secondary water inlet temp. ^oC	60	110
Secondary water outlet temp. ^oC	140	148

5.3.7 PERFORMANCE

During normal plant operation and during routine shutdown/cooldown transients, the NWA blowers are inactive. There is a small reverse helium flow for cooling, which causes the NWA heat exchangers to be operated continuously at light load. It was felt that such a condition would improve system reliability and reduce stagnant water corrosion of the U-tubes. Calculations for the HKV-type PNP indicate that the NWA system removes about 47 MW during normal operation.

Upon receipt of an actuation signal, the NWA system is placed automatically into service. The cooling water flow and the NWA blowers are slowly increased to full load to avoid thermal shocks. The safety analysis assumes that a five-minute delay in NWA actuation occurs, which accounts for the startup cycle. The inlet gas temperature encountered by the NWA heat exchangers depends upon the particular transient but varies between 600 and 1050^{o}C for PNP and up to about 900^{o}C for HTR-K.

Overall plant transients have been analyzed for the HTR-K plant to a larger extent than for the PNP; however, after-heat removal analyses have been performed for both PNP and HTR-K with similar results. As an indication of NWA performance and core behavior, an HTR-K depressurization accident transient is shown on Figures 5-9 and 5-10. Note the decrease in core after-heat generation of Figure 5-9 accompanied by an increase in heat dissipation as the NWA system functions. Figure 5-10 shows the resulting core thermal transient.

5.3.8 DIVERSE BACKUP AFTER-HEAT REMOVAL

The reliability of the NWA System is such that it meets most of the German Safety regulations. The exception is in the area of diversity during a water ingress accident with the NWA System as the initiator. The scenario postulated assumes that one NWA heat exchanger fails, causing a water ingress at 100% reactor power. Due to the (N-2) failure criteria in Germany, one train is assumed out for repair and one other train fails on demand. The net result would be only one remaining train with 50% heat removal capacity.

This unlikely transient is answered in the HTR-K design by the incorporation of an Emergency Feedwater System for the normal steam generators. This backup system has 2x100% heat removal capacity using the normal heat transfer loops that are unaffected by the water ingress. The exact configuration for the PNP has not been selected, but it appears that a backup using the normal heat transfer loops is likely.

5.3.9 NWA SYSTEM EVALUATION

The NWA Systems as presently envisioned for PNP and HTR-K are founded on sound engineering principles. The design used for the HTR-K system has been widely used for other gas-cooled reactors both in the FRG and the U.S. Therefore, it does not seem to offer significant developmental uncertainties. The PNP design has not as yet been qualified against the spectrum of plant accidents possible for the PNP concept. The new dimension of transients, although potentially severe, does not appear to pose unsolvable developmental problems.

As mentioned in Section 2.4, the reference NWA concepts incorporate one more level of redundancy than required in the U.S.

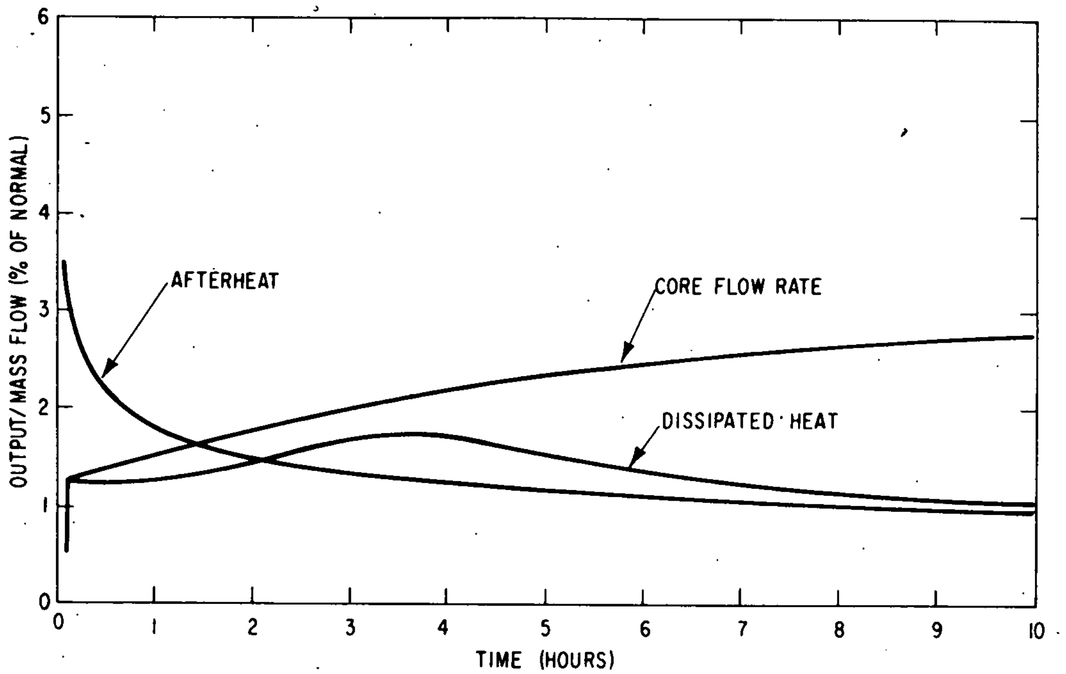


Figure 5-9. HTR-K Depressurization Accident

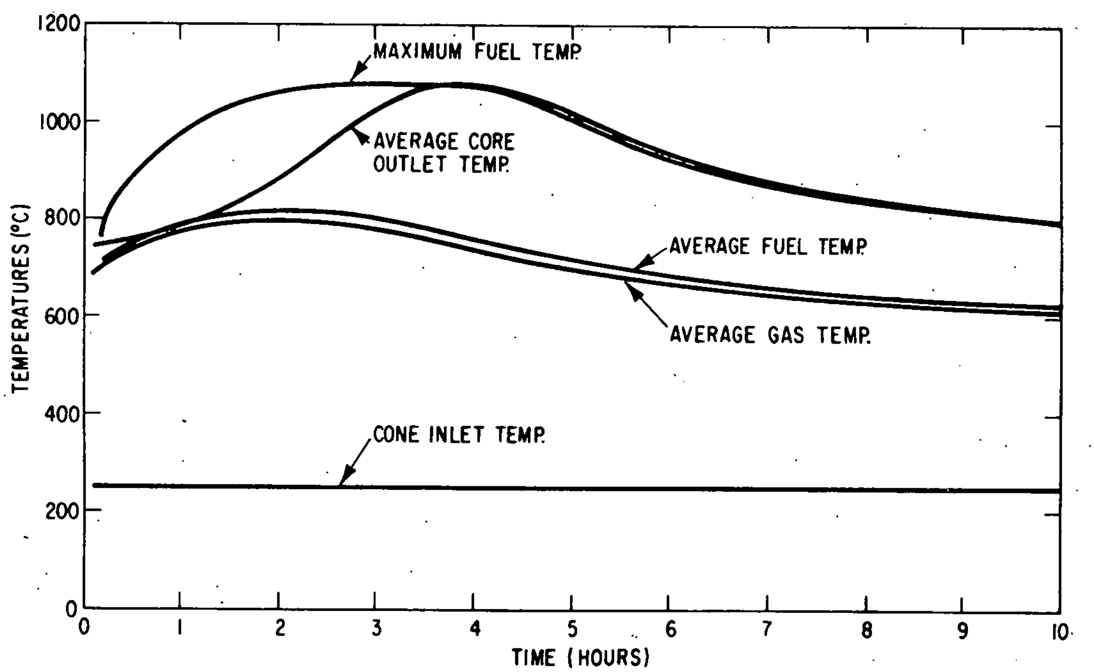


Figure 5-10. HTR-K Core Response for Depressurization Accident

5.4 GAS PURIFICATION SYSTEMS

5.4.1 INTRODUCTION

The various plant concepts under consideration (HTR-K, PNP, HHT) are all characterized by the use of helium as a reactor coolant. Impurities may enter that helium by erosion and corrosion of the graphite and other materials, by diffusion or leakage of fission products from the fuel through the core graphite, by diffusion or leakage of hydrogen and steam from secondary circuits, by the injection of impure makeup helium, and by air ingress after inspection and repair activities. Impurities are of concern for the following reasons: the potential deposition of fission products on primary circuit surfaces (maintenance problems); the potential for corrosion of the graphite; and the potential for diffusion into the secondary cycle (i.e., tritium and other fission products). In addition, clean helium for component seals must be provided at a pressure sufficiently above primary pressure to ensure that the flow is inward.

As the result of the above considerations, helium purification systems are an integral part of the design of all gas-cooled reactors. The need for purification is heightened in the PNP by the generation of process gas that passes to the general public for use. The direct-cycle HHT concept requires even closer control of activated impurities due to severe maintenance problems anticipated for the gas turbine and related machinery. The report "Safety and Licensing Evaluation of the German Pebble Bed Reactor Concepts" should be consulted for the safety and licensing implications of the purification system[22].

5.4.2 HTR-K PURIFICATION SYSTEM

General Electric was not provided with information on HTR-K purification systems. It is assumed that HTR-K purity limitations would be similar to those of the General Atomic HTGR. The system would have to meet the general functional requirements of the PNP listed below (Section 5.3.3). The capacity of the HTR-K system would probably be significantly lower than that of the PNP (10-20% helium turnover per hour instead of 100%), due to the more conventional secondary plant design (steam electric plant). The HTR-K purification plant design is sized for chemical impurity control and pump downtime, not radioactive impurity control, although the system does of course remove radioactive impurities.

5.4.3 PNP HELIUM PURIFICATION PLANT

The Helium Purification Plant for the PNP shown on Figure 5-11, consists of three subsystems: the Gas Purification Plant, the Exhaust Gas Retaining System, and the Regeneration Plant. The German design is based on consideration of primary circuit impurities (activated and nonactivated), helium supply for reactor plant components, and control of tritium release to the process gas. Note that the energy output to the consumer takes the form of process gas. Therefore, PNP has one less barrier between the fuel and the public than the HTR-K. The PNP purification plant is sized for the lowest possible tritium levels, in order to control doses to the consumer. The tasks and requirements for the system were specified to be:

- Cleanup of primary-circuit helium to control impurities below permitted levels.
- Supply of clean helium to reactor components.
- Supply of clean helium to the seals of the main and auxiliary (NWA) circulators.
- Pump-down of the primary (and/or) intermediate (by pump-down and pressurized helium storage).
- Regeneration of purification equipment.
- Monitoring of impurity levels.
- Storage and treatment waste gases.
- High availability.

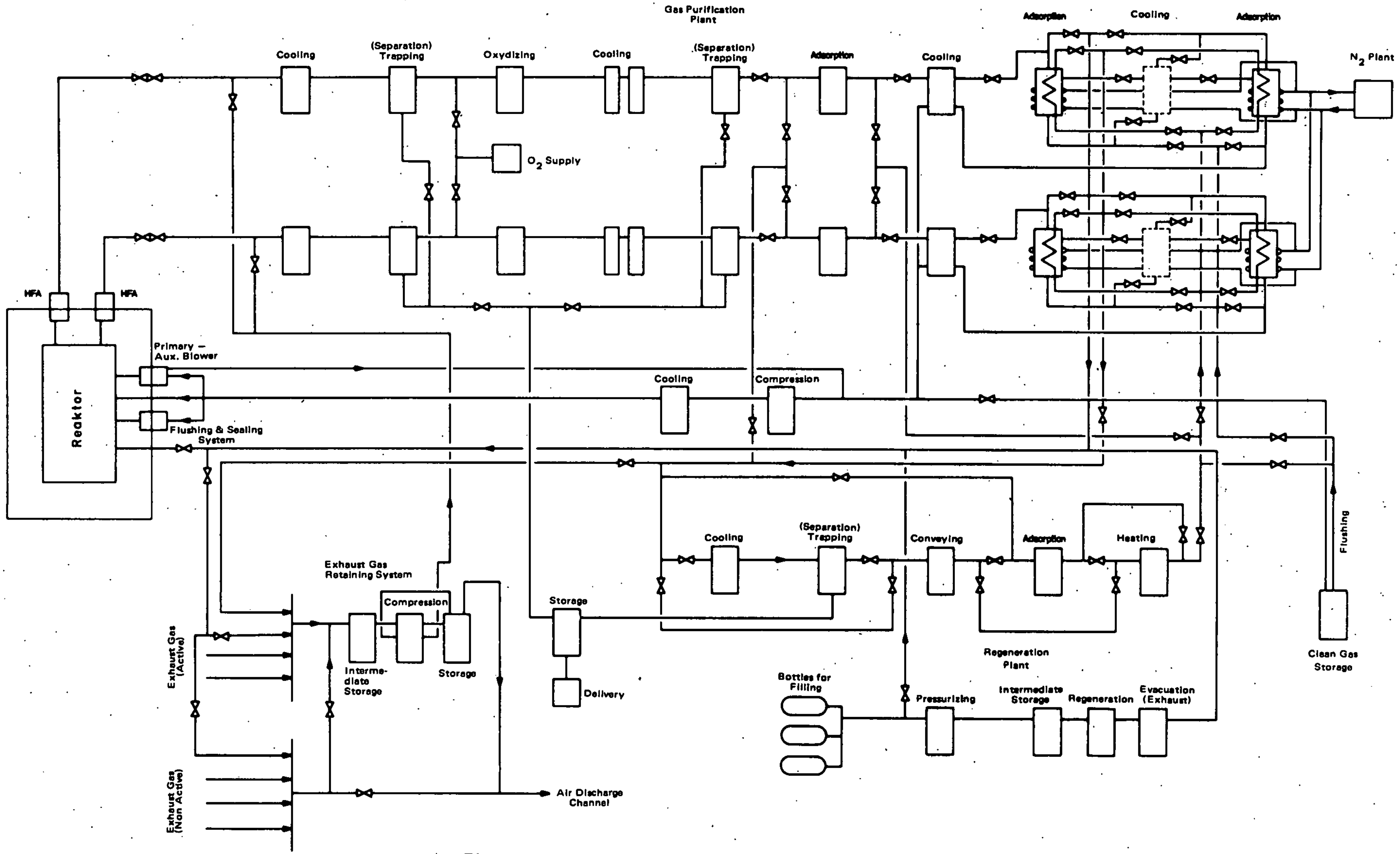


Figure 5-11. PNP Helium Purification System Schematic

The Helium Purification Plant is connected in parallel to the main helium loops and therefore acts as a bypass flow. The flow bypassed is sufficient to turn over the entire volume of the primary circuit through the purification loop each hour. This flow rate (40,000 m^3/hour) is substantially higher than that used in previous gas cooled reactors.

5.4.3.1 PNP Purification Plant Subsystems

The Gas Purification Plant consists of two redundant process lines, each capable of performing the required cleanup function. The helium removed from the primary circuit passes through high-temperature filter absorber units, where fission products other than noble gases and tritium (primarily iodine) are removed by adsorption on charcoal. Dust is also prevented from entering the purification train. The helium then passes through a series of coolers, trappers, and an oxydizer (copper oxide), which converts CO to CO_2 and hydrogen to water. Separators (molecular sieves) are provided to remove CO_2 and water. A low-temperature absorber system is then used to remove impurities such as krypton, xenon, carbon monoxide, methane, and some hydrogen and tritium. The low-temperature system operates at cryogenic temperatures (-300^{o}F) through the use of liquid nitrogen provided by the nitrogen plant. The output from the low-temperature system passes through a regenerative heat exchanger, where input gas is cooled and the returning gas heated. The warmer return helium flows to a compressor and cooler, and back to the reactor primary circuit either directly or via the sealing flow path (bearing buffer gas).

Normally, one train of the Gas Purification Plant is on-line, with the other train in standby. The operating train usually stays on-line for about six months, at which time the standby train is brought into use. The depleted train remains in a passive mode for about two months, during which the cryogenic portion is kept cold. The two-month period allows decay of the shorter lived radioactive impurities.

After the two-month decay period, the depleted train is regenerated by the Regeneration Plant. During regeneration, the equipment in the depleted purification train is brought back to operational conditions. The absorbers are heated and they off-gas the retained impurities. The regenerated train of the Gas Purification Plant is then placed on standby. The offgas is passed to the Exhaust Gas Retaining System, where further decay is allowed. When activity levels are low enough, the stored gas is discharged to the atmosphere in a controlled manner.

The PNP Helium Purification Plant is also provided with clean gas storage facilities, clean helium makeup connection, and appropriate instrumentation and control systems.

5.4.4 GAS PURIFICATION SYSTEM

As mentioned earlier, no information was provided on the HTR-K purification system. The discussions in Section 5.4.3 cover the extent of information available on the PNP purification concept. Although it is impossible to make conclusive statements about this important system, some general comments can be made subject to more detailed information.

The HTR-K should not require use of purification equipment substantially different from that of THTR or the U.S. General Atomic HTGRs. Therefore, it is not expected that purification problems will block development of the HTR-K.

The limited data on the PNP purification plant has given rise to some concerns. The experience to date in helium purification indicates that the equipment necessary to provide 100% helium volume turnover per hour will be extensive and costly. The need for a low-level tritium in the synthetic natural gas, and the desire for low hydrogen content in the primary helium, both impose increased size requirements on the purification plant. It may be that the purification processes planned by the Germans are different from previous plants; however, such information was not provided. Therefore, the German PNP development program should evaluate the economic incentives for providing such a large-capacity purification system. These concerns, however, are not felt likely to block development of an acceptable PNP purification concept.

SECTION 6

DIRECT CYCLE HELIUM TURBINE (HHT) CONSIDERATIONS

6.1 BACKGROUND

The contract between the U.S. Department of Energy and the General Electric Company for Fiscal Year 1977 was directed at the PNP and HTR-K concepts of the German Pebble Bed technology. During 1977 the decision was made by the German industrial companies (BBC and HRB) to support the development of the direct-cycle helium turbine concept with a pebble bed core (HHT) for electricity generation instead of the more conventional steam cycle HTR-K. This proposal is now being considered by the German Government and electric utilities.

The reasons for that decision are beyond the scope of this report; however, the selection of HHT and PNP as the reference concepts has a significant impact upon the overall development program.

The following sections are based on limited HHT technical information and, therefore, many of the issues discussed are not unique to the HHT but are generic nuclear-helium-powered gas turbine issues. The report "Safety and Licensing Evaluation of German Pebble Bed Concepts" should also be consulted for comments on the HHT concept.[22]

6.2 OVERALL DESCRIPTION

6.2.1 THERMODYNAMIC DESCRIPTION

The gas-cooled reactor concept for electricity generation (HTR-K) described earlier (Section 2.1) is based upon the Rankine Steam Cycle, which is the basis for the light water reactor plants currently in use. The direct-cycle helium turbine concept is a departure from these more conventional nuclear cycles and is based upon the closed Brayton Thermodynamic Cycle.

The Rankine Cycle for HTR-K and the closed Brayton Cycle as generally applied to HHT are shown on Figure 6-1 below:

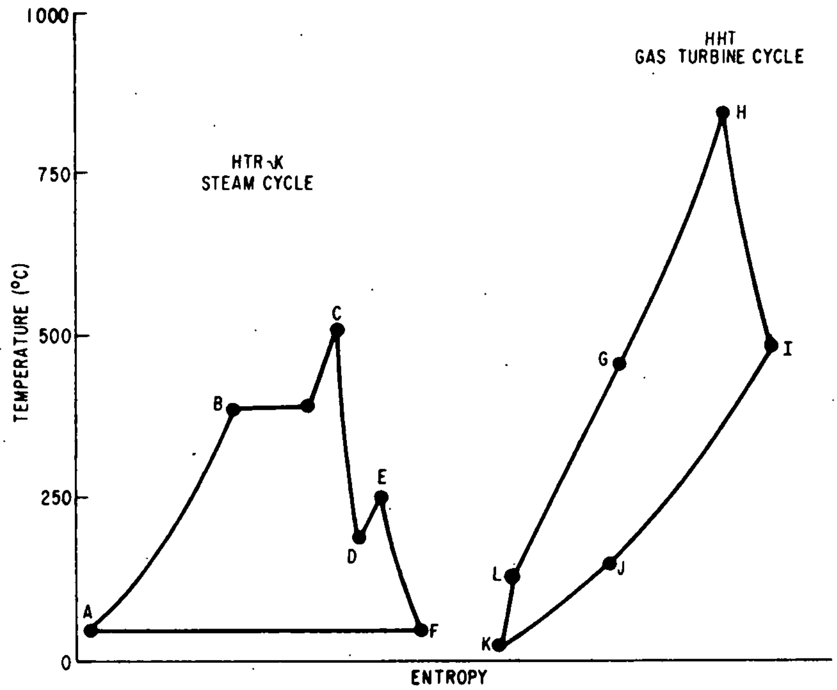


Figure 6-1. HTR-K/HHT Thermodynamic Cycles

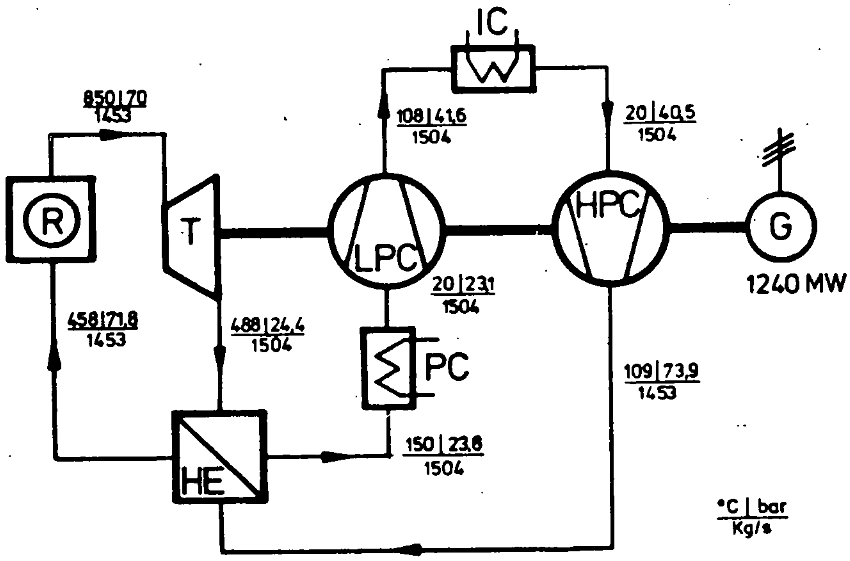
The HTR-K cycle includes the following general processes: heat input to feedwater in the steam generators to the boiling point (A-B), heat input through boiling transition to superheated conditions (B-C), expansion (work performed) in the high-pressure turbine (C-D), steam reheating (D-E), expansion and work in the low-pressure turbine (E-F), and finally heat rejection in the condenser (F-A).

It can be seen from the above right drawing that the HHT process is quite different. Reactor heat is added directly to the input helium (G-H), followed by expansion (work) in the gas turbine (H-I). The discharged helium gives up some energy to the input helium in a regenerative heat exchanger or "recuperator" (I-J) and is then cooled further in the compressor precooler (J-K). The helium pressure is increased in the compressor (K-L). Finally, the helium is preheated as it passes through the recuperator (L-G) before reentering the reactor. The HHT cycle (Figure 6-2) is slightly different in that two stages of compression with an intercooler are utilized.

It can be seen from the above figure that the heat is added at a higher temperature range in the gas turbine plant, which allows the potential for higher thermodynamic efficiencies than those possible for steam cycle plants. Another potential advantage is the higher temperature range at which waste heat is exhausted. The higher temperature waste heat can conceivably be used for distinct heating, as well as being much more amenable to dry cooling tower use. The dry cooling possibility has been considered to be a significant advantage for siting in regions of low water availability, although a recent report has indicated that dry cooling may not be the most economical choice. Peak-shaved dry-wet cooling was suggested as preferable (i.e., use wet cooling to supplement the dry towers during worst case environmental conditions).

6.2.2 HHT CYCLE DESCRIPTION

The gas turbine cycle of the HHT is consistent with the general thermodynamic description of Section 6.2.1. All of the components are housed within cavities of an integrated prestressed concrete reactor vessel (PCRV). The reference design includes only one gas turbine although other designs (those of General Atomic Company in the U.S.) are based on multiple loops, each with a separate, smaller sized gas turbine. The general data and a simplified flow diagram for the reference HHT are shown on Figure 6-2. The cycle includes heat addition from the pebble bed core (R), expansion in the gas turbine (T), heat exchange in the recuperator (HE), and compressor precooling (PC). After first-stage compression in a low-pressure compressor (LPC), the HHT cycle has interstage cooling (IC) before the helium passes to the high-pressure compressor (HE). Figure 6-2 should be consulted for the detailed helium conditions at the various points of the cycle.



REACTOR THERMAL POWER	MW	3000
TURBINE INLET TEMPERATURE	°C	850
TURBINE INLET PRESSURE	BAR	70
UPPER TEMPERATURE OF COOLING WATER	°C	70
AMBIENT AIR TEMPERATURE (NOMINAL)	°C	10
NET ELECTRICAL POWER	MW	1240
NET STATION EFFICIENCY	%	41.4

Figure 6-2. HHT Flow Diagram and Design Data

6.2.3 HHT GENERAL ARRANGEMENTS

The HHT primary circuit is housed within the PCRV with the reactor core located above the very large turbine cavity, which holds the turbine and the high- and low-pressure compressors. The other major components are located in separate cavities, including the recuperators, precoolers, and after-heat removal equipment. The gas turbine is directly connected to low- and high-pressure compressors via a single shaft, and there are two gas loops to and from the turbomachinery with two each of the other major components. The general arrangement of the HHT primary circuit is shown in Figures 6-3 and 6-4.

6.3 MAJOR COMPONENTS

6.3.1 PRESTRESSED CONCRETE REACTOR VESSEL (PCRV)

The HHT integrated design presents several aspects that are significant extrapolations from the HTR steam cycle PCRV: (1) a much larger size (48 meters) (2) higher operating pressure (72 bars instead of 60 bars); (3) significant increase in the magnitude of pressure transients (the Germans have estimated HHT transients as large as 100 bar/s, compared to about 5/bars/s for HTR-K); (4) the large horizontal cavity for the turbomachine; and (5) a warm liner concept.

All these items tend to increase the risk inherent in this advanced concept, since resolution cannot be assured. The warm liner concept offers some potential advantages over the more conventional insulated "cold" liner concept. In the reference HHT design, cold helium from the compressors flows between the hot components and the liner, thus avoiding the need for insulation. The potential benefit of this configuration would be improved access to the liner for inspection and maintenance, compared to the HTR-K and PNP concepts. There are, however, several problems associated with this liner concept, such as designing the warm liner to withstand extreme off-normal and accident pressure transients. Also, the large space required for inspection access will make post-accident liner cooling with the NWA system very difficult.

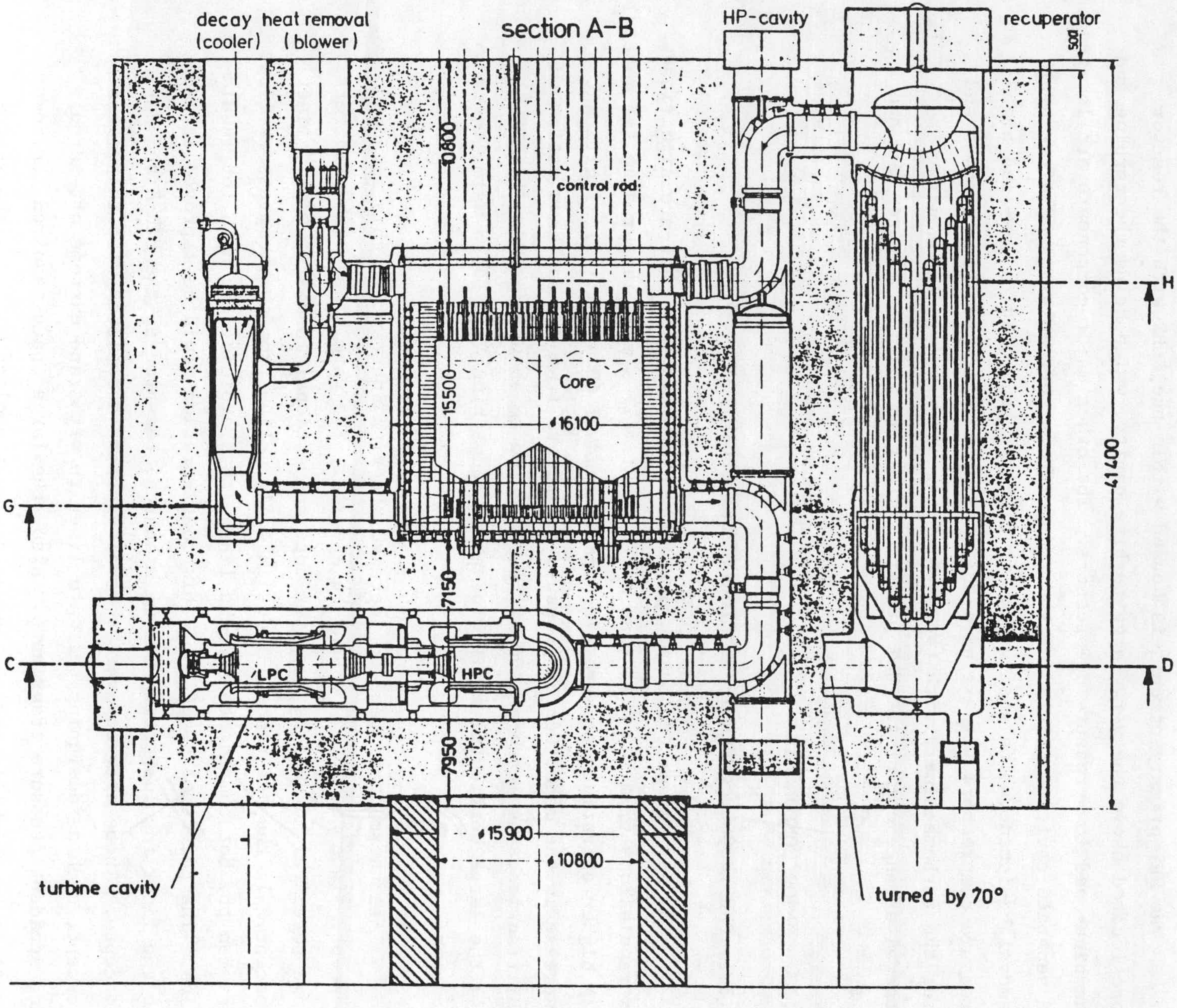


Figure 6-3. HHT Vertical Cross-Section

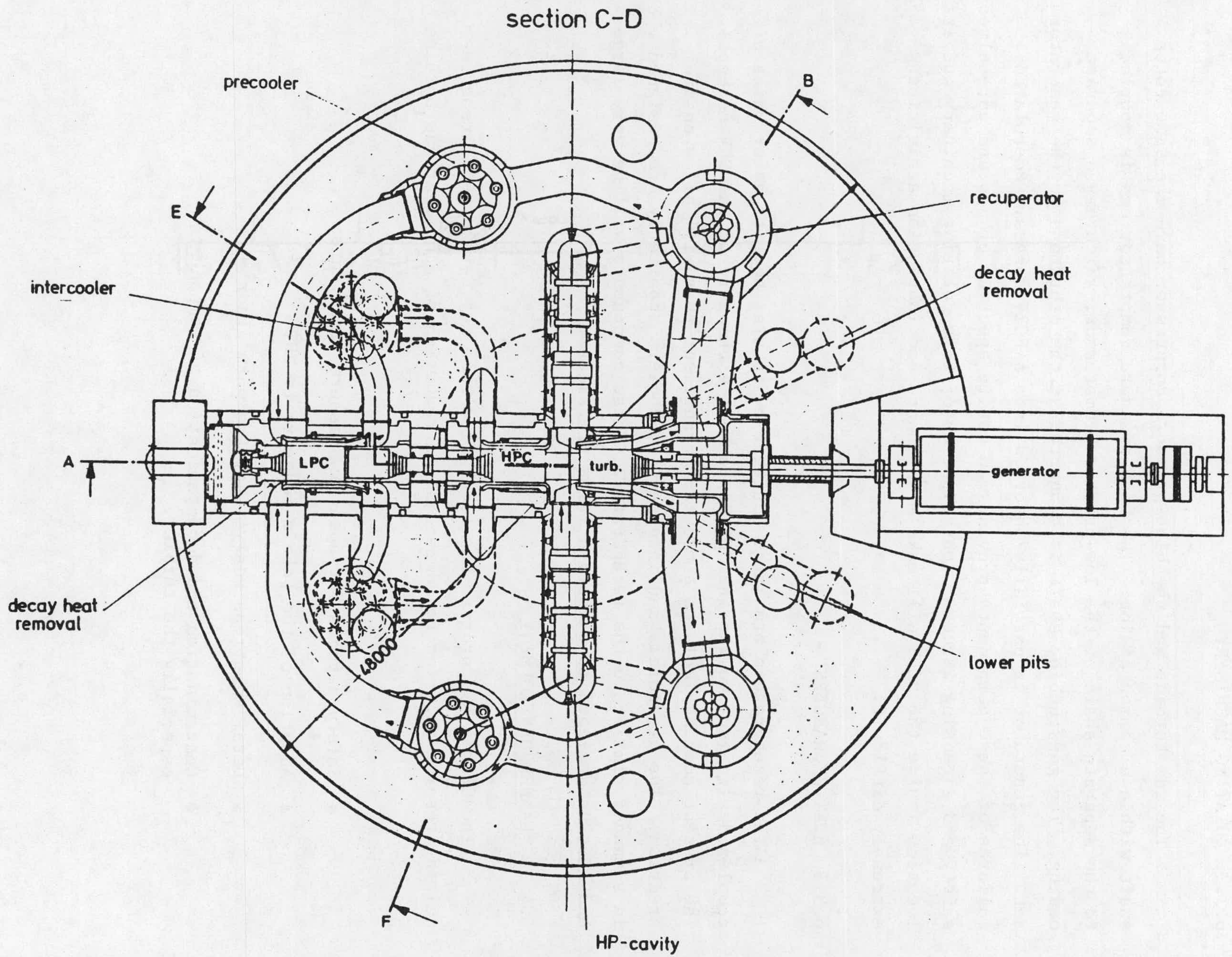


Figure 6-4. HHT Horizontal Cross-Section

6.3.2. ROTATING MACHINERY

The gas turbine and the high-pressure compressor have the same rotor shaft within a common casing. The high-pressure compressor end is coupled to the separate shaft of the low-pressure compressor, which has a separate casing. The gas turbine shaft is connected at the other end to the generator set. The generator is partially inserted into a PCRV recess. Postulated failures of these components provide an entire spectrum of new and unresolved safety and licensing issues. Contamination of this rotating machinery and its location inside the PCRV will make maintenance and inspection activities extremely difficult.

6.3.3 HEAT EXCHANGERS

The recuperators are large counterflow tubular heat transfer units of complex design (not unlike the PNP He/He heat exchangers in the WKV concept). The present concept includes a modular arrangement that will allow onsite erection. The precoolers and intermediate coolers apparently are designed in a manner similar to the recuperator. These components will be very large and must be designed to withstand large transients.

6.4 PLANNED DEVELOPMENT

The German HHT program has identified some key development areas requiring resolution prior to commercial introduction of a nuclear gas turbine plant:

- Safety aspects for special pressure transients
- Adaption of the pebble bed core to HHT
- Detailed stress analysis of the warm liner
- Contamination and decontamination of components, especially the turbine.

The program envisions development testing in two facilities:

- EVO (closed-cycle gas turbine plant, Oberhausen). This is a plant in Germany that is fossil-fuel-fired and is scheduled to reach full capacity (50 MW) late in 1977.

- HHV (High-Temperature Test Facility, Jülich). This is a high-temperature helium test loop located at the Research Center in Jülich. Initial operation during 1978 is envisioned.

- A prototype HHT plant is being considered to proof test the concept. The prototype plant would have a capacity of from 500-700 MWe and would possibly begin operation in the early 1990's.

6.5 GENERIC EVALUATION

The selection of the HHT as the companion concept for the PNP, instead of the more proven HTR-K, has made the high-temperature reactor program in Germany somewhat of a long-term research and development activity. The development needs for the PNP will now be paralleled by the major development needs of the HHT.

There are many areas where parallel HHT and PNP development will allow common programs, such as PCRV technology, graphite structures, fuel handling systems, etc. On the other hand, the conceptual differences will force a large number of other development programs to be separate. For example, the liner concepts are different with different development needs; after-heat removal requirements will be somewhat different; even the fuel requirements may have to be different, due to the HHT requirement of extremely low reactor coolant contamination levels. Materials development for many of the HHT components will have to be somewhat different, since the severe conditions that must be considered in HHT are not the same as those of PNP, particularly during transients.

The following is a brief list of some generic issues that have been identified for direct-cycle helium turbines:

- General Electric heavy-duty industrial gas turbine experience indicates that extrapolations, scaled from a proven design, normally result in an initial reliability factor of about 75% for the new design. Eventually the reliability factor increases into the 90% range after the industrial experience base expands. The closed-cycle gas turbine may be anticipated to have similar low reliability experience. Compounded by the presence of only one turbomachine, plant capacity factors may therefore be a potentially serious problem.

- Maintenance associated with radioactive turbomachinery inside the PCRV will raise man-rem exposure problems and cause more time-consuming and expensive repair outages.

- Formidable performance requirements for primary-circuit components due to extreme transients during off-normal and accident conditions.

- Severe materials requirements necessitate major development programs.

- Design of the very large PCRV with large horizontal cavities.

- Complex safety analyses need to be performed so that licensability can be determined.

Although none of the above items precludes HHT development, the task of engineering and commercializing a direct-cycle nuclear gas turbine plant is certainly formidable, particularly in view of the decision to develop two challenging concepts (PNP and HHT) simultaneously.

SECTION 7

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