KERNFORSCHUNGSANLAGE JÜLICH GmbH
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The Pebble-Bed High-Temperature Reactor as a Source of Nuclear Process Heat

Volume 3

System Considerations on the Nuclear Reactor

by

K. Kugeler, R. Schulten, M. Kugeler, H. F. Nießen, M. Röth-Kamat
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A Common Study by
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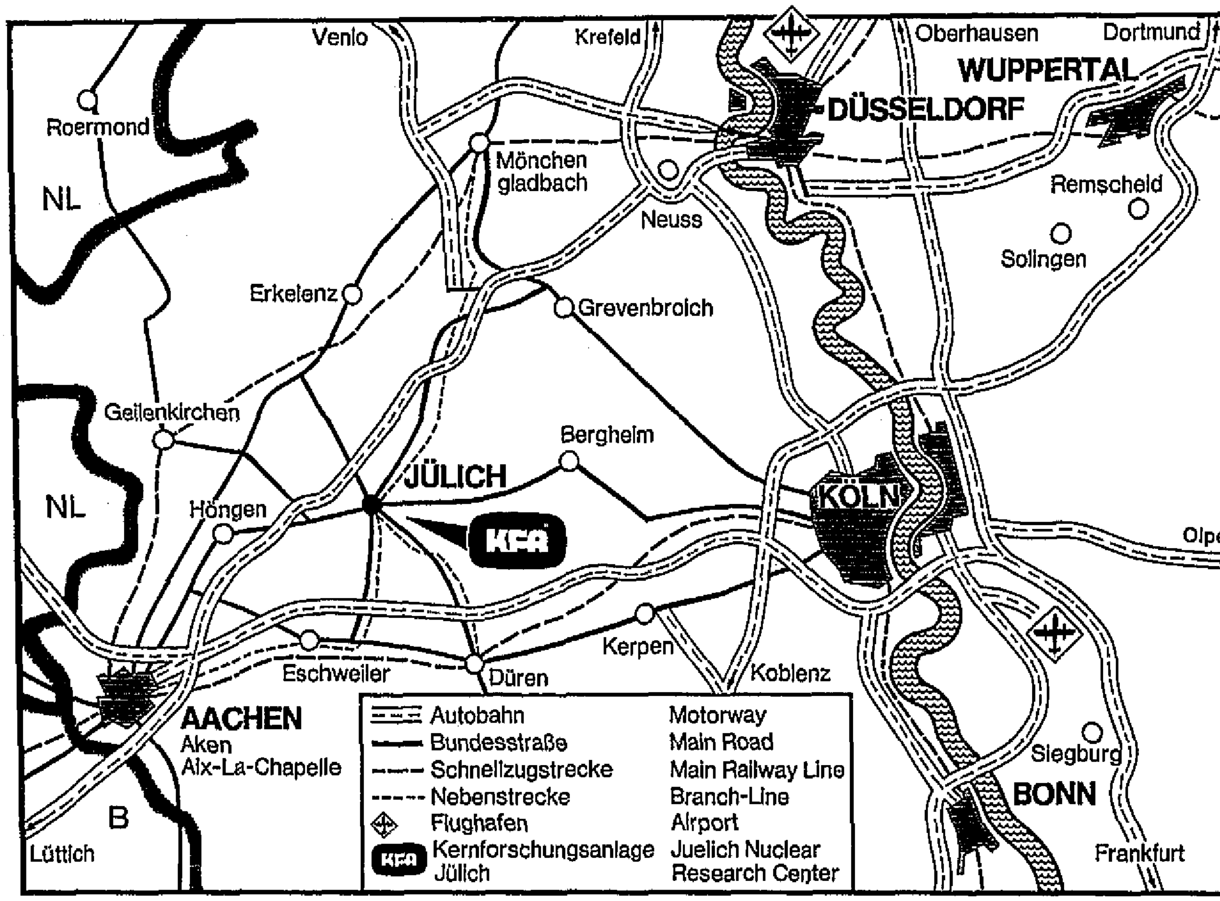

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Volume 3

SYSTEM CONSIDERATIONS ON THE NUCLEAR REACTOR

ABSTRACT

The characteristic questions concerning a process heat reactor with high helium outlet temperatures are dealt with in this volume like e.g. fuel element design, corrosion, and fission product release. Furthermore, some possibilities of the technical realization of the hot-gas ducting and intermediate heat exchangers are described. Important parameters for the design of the reactor such as core power density, helium inlet and outlet temperatures, helium pressure and fuel cycle burn-up and conversion and the effect of these on the primary circuit are investigated.

The important question regarding which reactor vessel is to be chosen for nuclear process heat plants is discussed with the aid of the integrated and non-integrated concepts using prestressed concrete, cast iron and cast steel. Thereafter, considerations on the safety of the nuclear plant are given. Finally, mention is made of the availability of the nuclear plant and of the status of development of the HTR technology.

Volume 3

SYSTEM CONSIDERATIONS ON THE NUCLEAR REACTOR

Introduction

The technical questions occurring in the case of the high-temperature reactor for the production of process heat can be divided into two categories. Firstly in a fundamental one which pertains to the possibility of realization of reactors with high outlet temperatures:

- Are helium temperatures realizable in the region of 950 °C without too high temperatures, temperature gradients and stress on the fuel elements?
- Are the contamination values of the systems for solid fission products sufficiently low so that the availability is not impaired?
- Are corrosion phenomena at high temperatures tolerable?

Secondly, some special features typical for nuclear process heat plants result for the reactor design:

- How is the hot-gas ducting to be designed?
- Is an intermediate circuit necessary?
- Which form of reactor vessel is to be chosen?
- Are the novel accidents controllable?
- How are the reactor data (power density, temperatures, pressures, fuel cycle) to be chosen appropriately?

3.1 Fuel Element Design for High Coolant Temperatures

The requirements regarding helium temperatures of the reactor are given in table 3.1-1.
A maximum temperature as given should be applied, depending on whether an intermediate circuit is planned or not

	Process	Reaction temperature	Reactor outlet tem. without IHX	Reactor outlet temp. with IHX
1)	steam reforming of methane	800...850 °C	950 °C	950 ... 1000 °C
2)	steam gafification of hard coal	800...900 °C	950 °C	950 ... 1000 °C
3)	steam gasification of lignite	750...850 °C	900 ... 950 °C	950 ... 1000 °C
4)	olefine production from gasoline	750...850 °C	900 ... 950 °C	950 ... 1000 °C
5)	water splitting	800...900 °C	950 ...1000 °C	1000 ... 1050 °C

Table 3.1-1 Helium Outlet Temperatures from Reactor depending on the Processes

High-temperature reactors, which are to be used for the production of process heat, have to show as low a contamination of the primary circuit as possible because such plants could possibly require inspection and repairs of the heat exchangers oftener than would be necessary for steam generators. As the rates of release are strongly temperature-dependent, fuel element temperatures should be kept as low as possible during continuous operation. According to all considerations on fuel element cycles and fissile material distributions in HTRs which have been employed up to now, the following method should achieve the lowest fuel element temperatures at defined gas outlet temperatures [1, 2]:
Fresh ball-shaped fuel elements are introduced at the top of the core and flow once through the core due to gravity. After reaching their specified burn-up, they are removed at the bottom of the reactor. The helium coolant gas flows in the same direction as the flow of the fuel elements, and is heated to the desired tem-

peratures (see fig. 3.1-2).

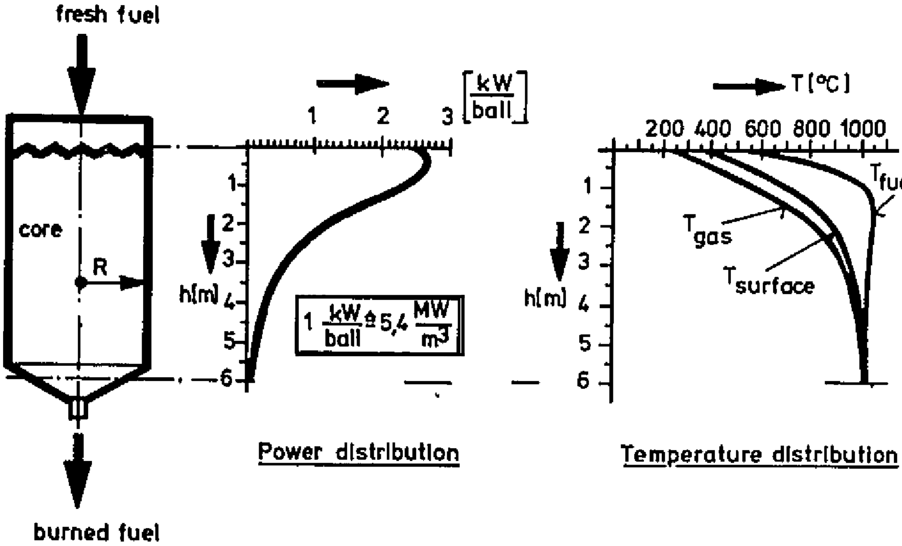


Fig. 3.1-2 Typical Axial Power and Temperature Distribution in a Reactor Core with the OTTO Loading Principle

By using this method, the following results are achieved: in the axial direction, the fissile material content of the fuel elements in the core decreases strongly from top to bottom. The thermal flux and power density show the same behaviour in the axial direction. Most of the power is therefore produced in the upper part of the core i.e. where the coolant enters in a relatively cold state. The power density of the fuel elements is very low at the bottom of the core, only an after-burning occurs, which results in an equalization of burn-up. With such a power distribution, the coolant is very rapidly heated in the upper part of the core, where large temperature differences occur between the gas and the fuel elements, while in the lower part of the core the differences become very small.

The following review shows the conditions in a reactor with the OTTO (once-through-then-out) fuel loading explained above. Detailed consideration of a single element shows that for a power density of 1 MW/m^3, a temperature difference of about 35 ^{o}C exists between the gas and the coated particle. Fig. 3.1-3 shows a typical temperature profile in a fuel element in a position in the lower part of the reactor core (on the axis) and in the middle of the core.

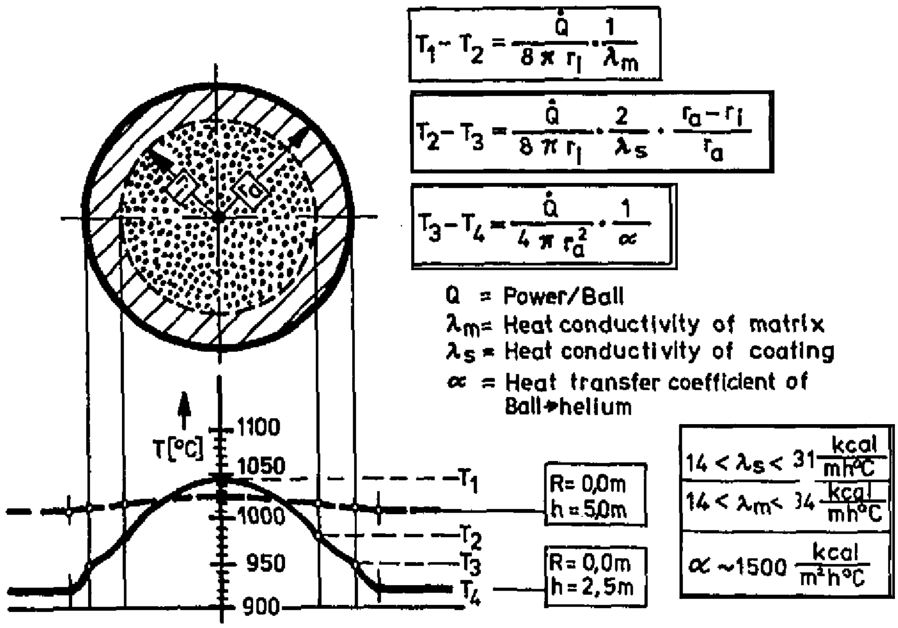


Fig. 3.1-3 Temperature Profile in Fuel Element for Different Core Heights

To obtain high coolant outlet temperatures the above OTTO cycle has the best prerequisites, in which a low power density automatically occurs at the coolant outlet and not through control of various levels of enrichment in the axial direction. In general, a two-zone loading of the core will be suitable, so that a maximum temperature difference of less than 100 °C is to be expected for the radial temperature profile. Thus the following status results for the fuel element design: For a specified coolant outlet temperature of 980 °C, and a mean power density in the core of 5 MW/m^3, the surface temperature of the fuel element reaches a maximum value of about 1100 °C, and the coated particle reaches a maximum temperature of about 1115 °C. As against this, the design data for ball-shaped fuel elements usually used today, and which are guaranteed for the THTR, have the following values: surface temperature, 1050 °C, coated particle temperature, 1250 °C. While the fuel temperature still includes a high safety factor, the surface temperature is slightly exceeded. However, this only leads to an increase in the reaction velocity, which can be considered in the design of the gas purification plant (see § 3.3).

It has to be tested, however, whether predictions regarding the typical, new type of behaviour of the fuel elements (low temperatures at high dose and high temperature gradients during the first half of reactor operation, high temperatures at low dose and low temperature gradients in the second part) can be made from results of the fuel element irradiation program undertaken up to now.

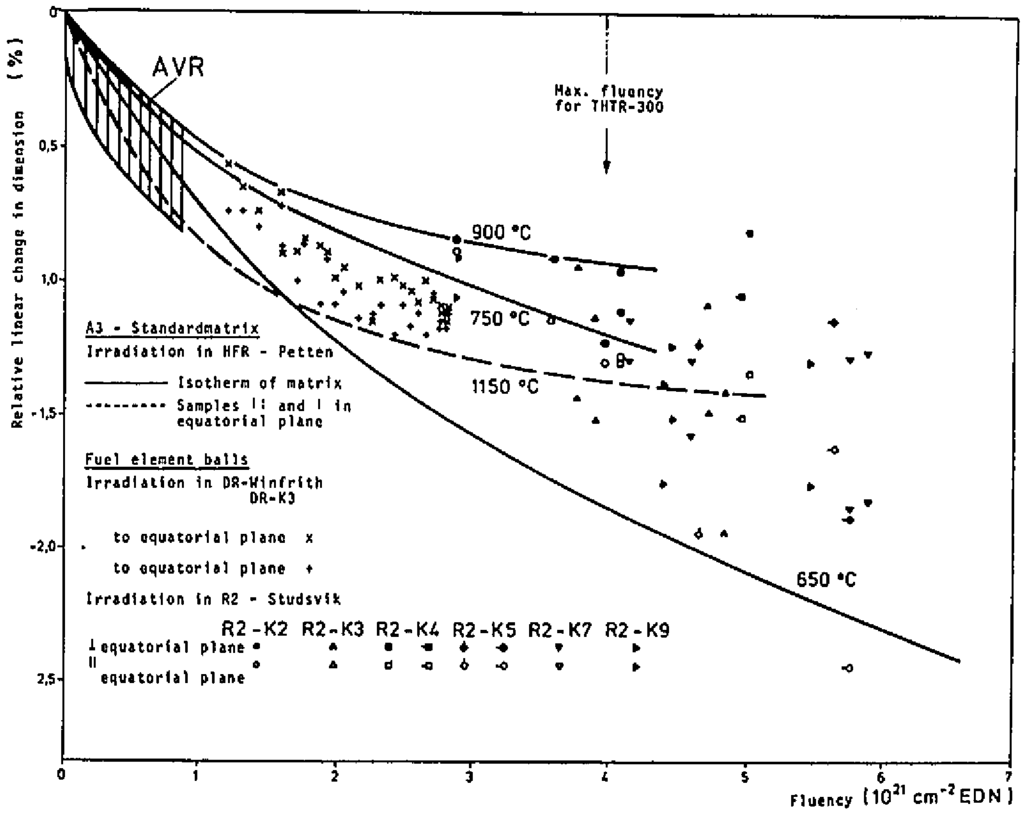


Fig. 3.1-4 Dependence of the Relative Linear Change in Dimensions of Ball-shaped Fuel Elements with THTR Temperature Cycling (R2-Tests) or with Max. Surface Temperatures of 900-1150 ^{o}C (DR-Tests) in Comparison to that of A3-Matrix (isothermally irradiated) on the Fluency [3].

Fig. 3.1-4 shows the status of irradiation of ball-type fuel elements. The burn-up conditions of process heat designs are already experimentally verified with good results. A possibility of considerably lowering the release of Cs-137 as well as other solid fission products occurs in doping the nucleii of the coated particles with oxidizing additives like Al_2O_3 and SiO_2 and so of obtaining a retaining capacity which is better by several decimal powers (see fig. 3.1-5). This new type of coated particles have still to be tested.

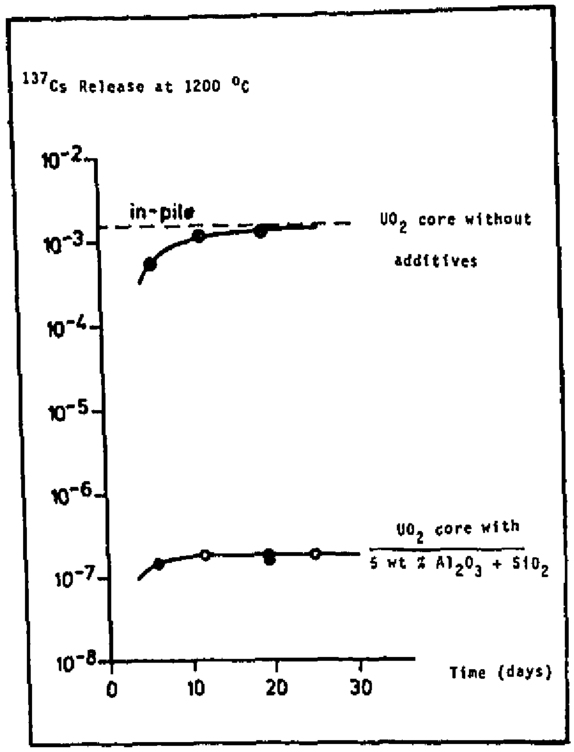


Fig. 3.1-5 Cs-Release from Fuel Material Coated with PyC during Annealing after Irradiation (Burn-Up: 8 % FIMA [4])

3.2 Contamination of the Helium Circuit

Fission products in the coolant circuits of nuclear reactors, and here especially in the heat exchanger aggregates, are of large importance for the following reasons:

- the accessibility to the components during operation and during inspection or repairs is rendered difficult or even impossible when the coolant circuit contamination is very high
- the solid fission products contained in the primary circuit and deposited on its walls (plate-out) may be released to the environment during an accident. Furthermore, it is very difficult to handle such components with plate-out for repairs.
- as some of the plants discussed here are open two-circuits plants, the consumer may come into contact with tritium, which can diffuse into the product gas stream (e.g. in synthetic methane).

From the technological safety point of view, the interesting fission products are such which are released in large amounts during fission, show high γ- or β-activity, have long half-lives and have high biological activity. Considering the above features, the following gaseous and solid fission products are of particular importance in high-temperature reactors today (table 3.2-1).

Isotope	Radiation	Mean Energy	Half-Life	Fission Yield
T	β^-	0.018 MeV	12.3 a	$8.7 \cdot 10^{-5}$
Kr^{85}	β^-	0.67 "	10.8 a	0.293 %
Xe^{133}	β^-	0.35 "	5.3 d	6.6 "
$Te^{129}m$	β^-	1.6 "	33 d	0.35 "
Te^{132}	β^-	0.22	78 h	4.7 "
I^{131}	β^-	0.61 "	8.05 d	3.1 "
I^{133}	β^-	1.4 "	21 h	6.9 "
Sr^{89}	β^-	1.46 "	50 d	4.8 "
Sr^{90}	β^-	0.54 "	28 a	5.77 "
Be^{140}	β^-	1.02 "	12.8 d	6.4 "
Cs^{134}	β^-	0.6 "	2.1 a	
Cs^{137}	β^-	0.51 "	30 a	6.15 "
$Ag^{110}m$	β^-	0.09 "	260 d	
Ag^{111}	β^-	1.05 "	7.5 d	0.025 "

Table 3.2-1 Properties of Some Important Fission Products (data acc. to Reactor Handbook Vol III A, Physics Sec., ed. 1962)

3.2.1 Steady-State Coolant Activity

The steady-state coolant activity in the HTR test reactors Dragon, Peach Bottom and AVR has been studied for several years of operation. The measured values are extremely low due to the coated particle techniques.
In the AVR, for example, the β-activity of the coolant is about 2×10^{-2} Ci/Nm^3. The present activity in the primary system of the AVR for approx. 2000 Nm^3 gas is only about 40 Ci.
Measurements and investigations up to now of noble gas fission products in HTRs show that in undamaged fuel elements the magnitude of the coolant activity is determined by the outer uranium contamination of the fuel elements, which depends on the fabrication process [5].

In a process heat reactor with a coolant outlet temperature of 980 °C, a mean power density of 5 MW/m^3 and by application of the OTTO principle, somewhat higher specific activities result. Assuming a value of 1 Ci/MW(th) for the AVR, the steady-state value for a 3000 MW(th) process heat plant can be expected to be approx. 10 Ci/MW(th). This takes into account the number of fuel elements in a definite temperature range in the AVR and in the process heat reactor which is being discussed. The overall steady-state coolant circuit activity would then be approx. 30 000 Ci (noble gas) or about 1 Ci/Nm^3 He without assuming broken fuel elements. Such a value in normal operation represents no great problem for the accessibility of the circuits (e.g. loading, gas purification).

Assuming a leakage of 1 ‰ per day under normal conditions, a release of radioactivity to the environment via the chimney of approx. 3.10^{-4} Ci/sec would occur. This value is not hazardous from the technical safety point of view. This value can be considerably reduced by an intermediate acculumulation of the leakage gas in the containment.

3.2.2 Deposition of Solid Fission Products

Many irradiation experiments have shown that solid fission products are released from fuel elements. The experience today ist that these fission products are deposited on the internal installations of the nuclear reactor as also on the walls of the heat exchangers and in tube ducting. For process heat plants, it is especially important to know this concentration in the heat exchangers, which have to be accessible for replacement of catalyst or of entire reformer tubes. The knowledge about the behaviour of solid fission products in the coolant is as follows: It is known that the deposition behaviour of individual isotopes depends strongly on the temperature, the material, the mass flow velocities, the geometrical dimensions, the surface structure and similar parameters.

There are a series of experiments, especially recently also in the AVR, which show the following results:

- the surface of dummy balls in the AVR were examined for deposits [6].
- near the helium compressor of the AVR, a filtering candle was installed. Relatively cold coolant at a temperature of 125 °C was led through this, and the filter was analysed [7].
- In the space between the reactor core and the steam generator of the AVR, hot-gas samples were removed and their content of solid fission products were measured (VAMPYR-experiment). These experiments are continuing [8].
- In the Pegase reactor (Cadarache), irradiation experiments are carried out in a loop on fuel element balls; the thereby occurring deposition of solid fission products is measured (SAPHIR-experiment)[9].

The following table 3.2.2-1 is obtained for some important solid fission products and shows some values for deposited activities on surfaces:

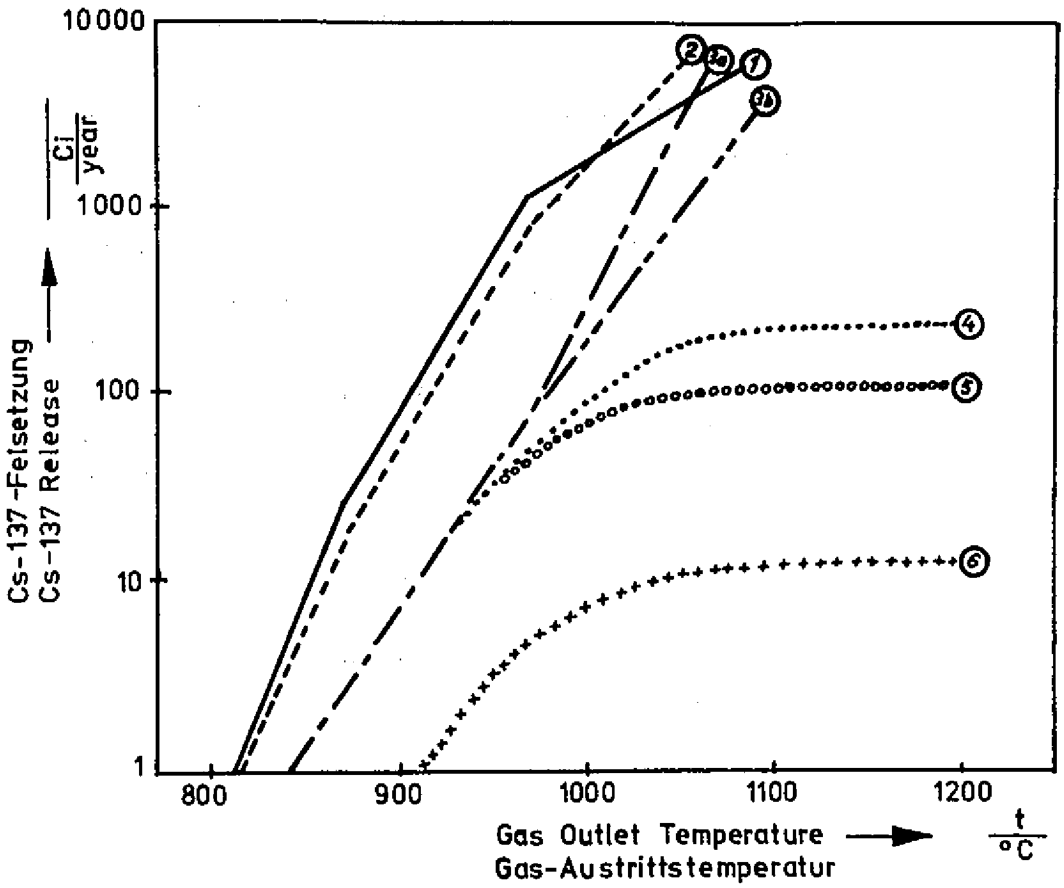
	Dim.	Deposition on AVR-balls	AVR-filter	Saphir 100 °C	Saphir 500 °C	Vampyr 850 °C
Iodine 131	$\frac{Ci}{cm^2}$	$1\text{-}2x10^{-9}$	$2.2x10^{-9}$	$0.5\text{-}1x10^{-6}$	10^{-9}	$7.6x10^{-9}$
Cesium 137	"	$1\text{-}1.5x10^{-9}$	$1.5x10^{-9}$	$5x10^{-10}$	$5x10^{-9}$	$2.1x10^{-10}$
Barium 140	"	$1\text{-}1.5x10^{-9}$	$5x10^{-9}$	$5x10^{-10}$	$5x10^{-10}$	$1.6x10^{-10}$
Silver 111	"			$5x10^{-6}$	$10x10^{-6}$	$5.2x10^{-8}$

Table 3.2.2-1 Experimental Data for Solid Fission Products in Reactor Circuits [6-9]

From other experimental measurements it is furthermore known that isotopes such as Iodine 131 are deposited on cold spots of the coolant circuit i.e. at 250 °C and lower temperatures. Therefore it is not to be expected that this isotope is deposited on the surface of the reformer tubes. However, attention has to be paid to the presence of this isotope on the cold spots of the gas-ducting system. In the case of Cs-137, it can be expected that a large part is deposited on the ceramic bottom of the nuclear reactor; however, a part may also deposit on the reformer tubes.

Also silver, which was unexpectedly detected in large amounts, could possibly be deposited preferably on the reformer tubes. The absolute amount of solid fission products in the coolant gas of a 3000 MW(th)-plant can naturally at present only be roughly estimated. It has to be expected that similar to the gaseous fission products, an increase in the outlet temperature to 980 ^{o}C would lead to an increase in the release of solid fission products as compared to the measurements in the AVR.

Based on the present experimental data, the problem can be stated numerically as follows: For the application of mixed oxide particles with layers of pyrolytic carbon and an outer uranium contamination of 2.5×10^{-4}, which is technically realizable today, a release of about 1000 Ci Cs-137/a is expected at 980 ^{o}C (see Fig. 3.2.2-2). Cs is considered today to be an especially critical isotope as regards maintenance. Lowering of the outer uranium contamination to 2.5×10^{-5}, a measure which is strived at, reduces the value to about 200 Ci/a. By use of silicon carbide layers for the mixed oxide particles and by lowering the uranium contamination further by a factor of 10, this value could be reduced to 5 Ci/a. After 10 years of operation of the reformer tubes,about 0.21 Ci/reformer tube or 5 μ Ci/cm^2 reformer tube surface would be deposited assuming 100 Ci/a (assumption: 4500 reformer tubes, uniform deposition of all the cesium on the reformer tubes). Catalyst manipulations can be carried out for such values without any big difficulties (s. § 4.9). However, the other fission products have to be considered, too, which possibly contribute with a similarly high contamination as Caesium.



Cycle:	OTTO
Power:	3000 MW_{th}
Burn up:	100 000 MWd/t
Gas Inlet Temperature:	250 °C
Fuel Residence Time:	890 days
Power Density:	9 MW/m^3

Curve No.	Fuel	Matrix Contamination	Fraction of Failed Particles	Fuel Element
1	Feed-TRISO Breed-BISO	2.5×10^{-4}	10^{-3}	Conventional
2	Mixed-BISO	2.5×10^{-4}	None	Conventional
3a	Mixed-BISO	2.5×10^{-5}	None	Conventional
3b	Mixed-BISO	2.5×10^{-5}	None	Zoned
4	Feed-TRISO Breed-TRISO	2.5×10^{-5}	10^{-4}	Conventional
5	Mixed-TRISO	2.5×10^{-5}	None	Conventional
6	Mixed-TRISO	2.5×10^{-6}	None	Conventional

Fig. 3.2.2-2 Estimation of Cs-137 Release in a 3000 MW-Reactor [10]

3.3 Corrosion

Water leakages from the steam generator and the steam reformer as well as hydrogen, which permeate from the steam reformer into the helium circuit, cause corrosion of the fuel elements and of the graphite installations of the core.

The reactivity of the reactor core is increased by transition to higher helium outlet temperatures. Thus it has to be investigated whether the specifications, which have been made up to now for fuel elements and for graphite installations e.g. for the THTR, can be applied for a reactor system with a gas outlet temperature, which is 200 ^{0}C higher. However, it has to be observed that through the introduction of the OTTO loading, the maximum surface temperatures of the fuel elements are not exceeded very much.

The following reactions could occur caused by the impurities in the helium circuit:

$C + H_2O \rightarrow CO + H_2$

$CO + H_2O \rightarrow CO_2 + H_2$

$C + 2H_2 \rightarrow CH_4$

$C + CO_2 \rightarrow 2CO$

Fig. 3.3-1 shows the equilibrium position of the individual reactions in dependence on the temperature.

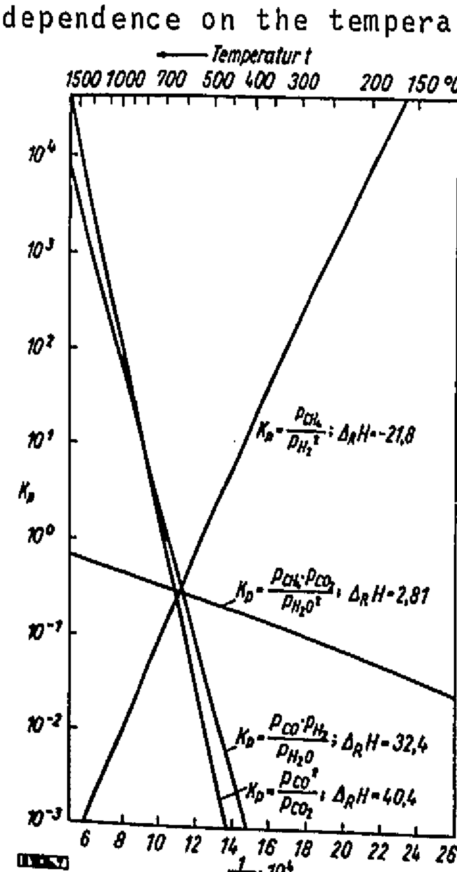
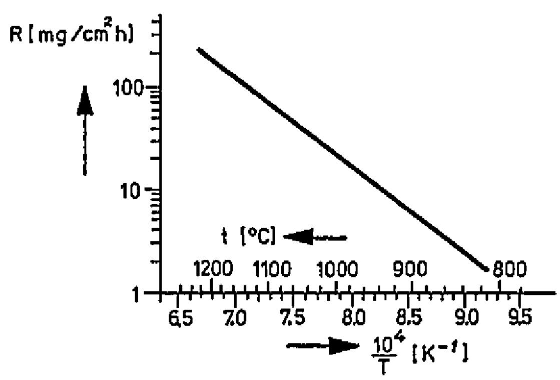


Fig. 3.3-1 Position of Equilibrium of the Reactions [11]

It is known that fuel element graphite and reflector graphite have strongly differing reaction rates for the various corrosive media.

A3-graphite, which is used for fuel elements, shows e.g. the following dependence on temperature and pressure for reaction with water [12].



$$R(P_{H_2O}, T) \sim e^{-\frac{C}{T}} \cdot \sqrt{PH_2O}$$

Fig. 3.3-2 Reaction Rate for A3-Graphite Fuel Elements [12]

The following specification is derived from this for the THTR reactor: for 1 Vol % H_2O in helium at 1b, the corrosion rate at 900 °C is limited to 0.5 mg/cm^2 h, and correspondingly at 1000 °C it is limited to 1.5 mg/cm^2 h (test duration:10 h) [13].

For the fuel elements, a graphite corrosion of 0.1 mm from the surface is considered to be tolerable. This means ∿ 1 % loss in weight or a total loss of ∿15 mg/cm^2 fuel element surface.

The amount of water, which can enter the core due to this specification, is estimated as follows:

reactor power	:	3000 MW
core volume	:	600 m^3
number of fuel elements in the core	:	3.25 x 10^6
number of fuel elements removed annually	:	10^6
weight of a fuel element	:	160 g

This amounts to a total loss in weight of 1.6 t C/a or an equivalent amount of converted water of 2.4 t H_2O/a. For a duration of operation of 8000 h/a, leckage amounts of 300 g H_2O/h can enter the helium circuit.

A leckage amount of 300 g H_2O for a 3000 MW-plant seems to be technically realizable under the aspect of the THTR steam generator specification of 60 g/h. No difficulties should therefore arise here caused by an increase of helium temperatures. The specification for the surface corroding can probably be somewhat disarmed due to the handling of the fuel elements only once in the OTTO loading.

The hot core bottom should have a service life of 30 years. Therefore,it has also to be protected from very high material corrosions: As the reaction rate of reflector graphite as seen in fig. 3.3-3 is a factor of 30 to 50 lower than that of A3-graphite, the design of the core bottom causes no difficulties.

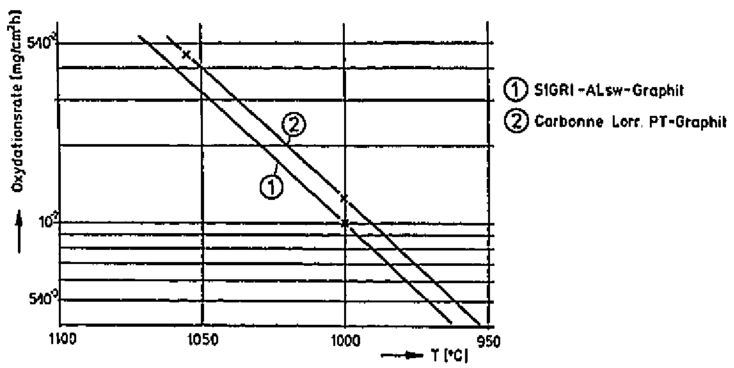


Fig. 3.3-3 Oxidation Rates for Typical Reflector Graphites [14]

The fuel element graphite is so reactive that it reacts with all the available steam in the pebble-bed. If, however, it is supposed that 2 - 3 % of the water can cause corrosion of the core bottom, the following observation can be made: About 30 to 50 kg graphite/a are converted; in the course of 30 years of operation this leads to a maximum corrosion of 1.5 t C. Corresponding to a hot graphite surface of 1000 m^2, which is affected, this means a corrosion of 1 mm, which can be tolerated.

The corrosion of the fuel elements and the structural graphite by H_2 can be rated as follows:
Assuming a tolerable corrosion of 0.1 mm from the surface of the fuel elements and therefore 1600 kg C/a as shown above, 5000 Nm^3 H_2 must be available as gasification medium. For 8000 hours of operation, therefore, 0.6 Nm^3 H_2/h would be tolerable as leckage rate into the primary circuit. The actual rate of permeation through the reformer tube is estimated to be < 0.5 Nm^3 H_2/h for a 3000 MW(th) plant. However, due to the largely differing reaction rates for steam and hydrogen, it is not to be expected at all that

reactions can occur to such an extent. By the way, the effectiveness of the gas purification just in the case of H_2 is counted as an additional safety factor.

The following table shows a comparison of the reactivities of H_2O and H_2, in which the well-known experimental results are converted to the same conditions (for fuel element graphite).

Medium / Parameter	H_2O	H_2
Temperature	1000 °C	1000 °C
Total pressure	40 b	40 b
Partial pressure	1 Vol % H_2O	1 Vol % H_2
Rate of reaction	∿10 mg/cm²h	≅1 μmole CH_4/h·g
Conversion based on C+)	$6 \cdot 10^{-3}$ g/h·g	$2 \cdot 10^{-5}$ g/h·g

Table 3.3-4 Comparison of Reactivity of Fuel Elements for H_2O and H_2

+) related to volume

This consideration indicates that the corrosion by hydrogen is much smaller than that by water.

For the structural graphite also, similarly low reaction rates in hydrogen as for A3-graphite were obtained [15]. Taking into account the small amount of additonally available hydrogen from the steam reformer and the effectiveness of gas purification, it can be shown by a similar consideration as before that here, too, no difficulties occur in the design of the core bottom for a helium outlet temperature of 950 °C.

3.4 Hot-Gas Ducting between the Reactor Core and Heat Exchangers

The transport of hot helium from the reactor core to the heat exchangers presents a considerable problem for the technical realization of nuclear process heat.

Experience with this problem until now has only been gained from the AVR, in which the coolant has actually attained a temperature of 950 °C. The problem has been solved here by passing the hot gas through structures made of graphite and carbon stone (s. fig. 3.4-4). However, the design concept chosen here of the placement of the steam generators above the core cannot be transferred to large-scale plants. Besides, in the new pebble-bed reactor design concept, the direction of the gas flow has been reversed.

In the case of the HTR test reactors Dragon and Peach Bottom and in the Windscale AGR (see fig. 3.4-5), the coaxial ducts used are cooled in counterflow by cold helium. However, the maximum helium temperature here is only 750 °C, so that it is difficult to make an extrapolation to present design concepts.In addition, these ducts have small dimensions only as compared to large-scale plants.

In the THTR (300 MW(e)), the problem of the transport of hot gas is solved by a ceramic duct between the reactor core and the heat exchangers (s. fig. 3.4-3). In the ducts themselves, there is additional insulation made of metal foils. Since all the primary circuit components are integrated in the large prestressed concrete vessel, the problem of insulating the vessel itself against hot gas does not occur.

In the Fort St. Vrain Reactor (330 MW(e)) (s. fig. 3.4-2), the above problem is solved as follows: The gas is led from the reactor core through the reactor support plate to the steam generators which are placed under the reactor core. These penetrations through the support plate are lined and insulated with "Kaowool", which is pressed together by metal plates.

In all new design concepts for large-scale plants, like pod-boiler vessels or loop systems, a new problem arises of leading the hot gas itself through the vessel wall and/or protecting the vessel i.e. the corresponding penetration duct, against excess temperature. The following principles of insulation, which are described in detail and discussed, can be used here (s. fig. 3.4-1):

- Insulation of the penetration duct by metal foils, wool or ceramic lining (variant 1)
- Use of a coaxial duct, and cooling the hot-gas duct in counter-flow with cold helium (variant 2) to temperatures of about 550 ^{0}C
- Use of a coaxial duct, in which the hot-gas duct is internally insulated (variant 3)
- Design of a coaxial duct system using tubular pieces of carbon stone or other ceramics(variant 4).
- Use of a cast chromium steel structure with holes for the cold gas, and insulated inner duct for the hot gas (variant 5).

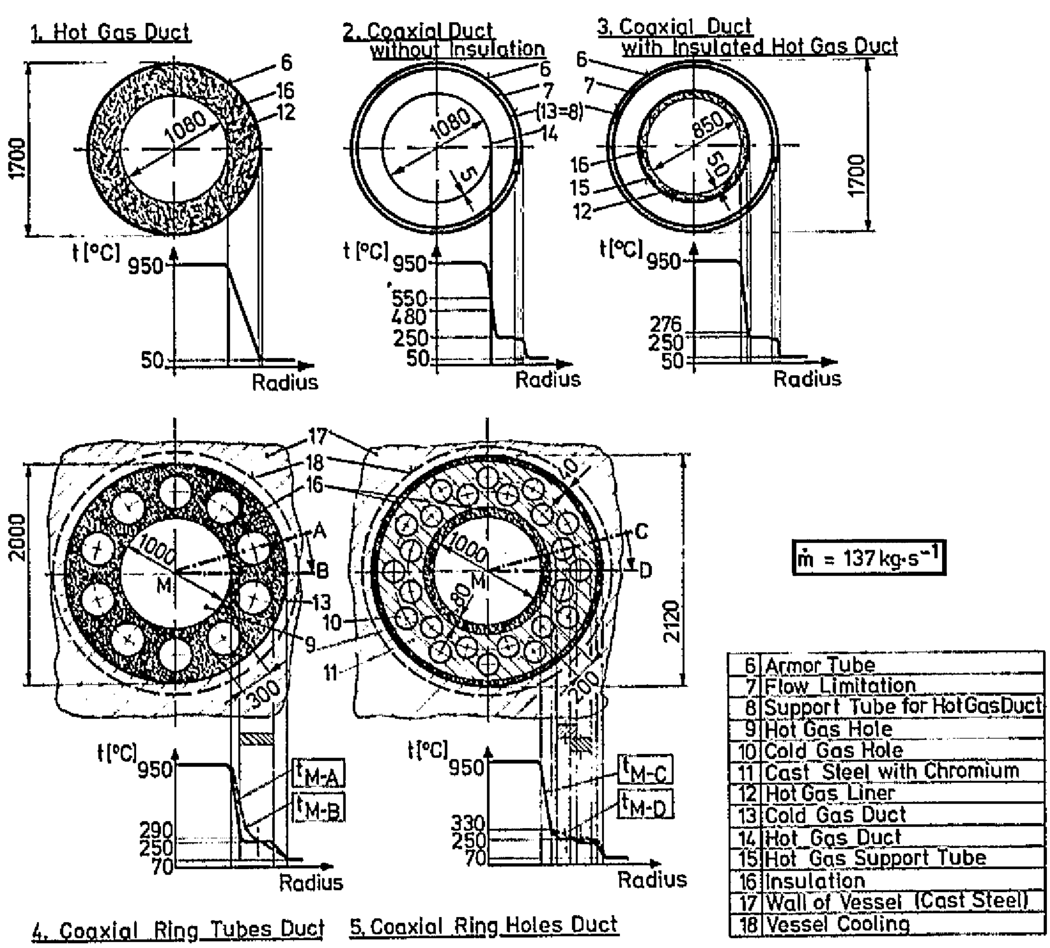


Fig. 3.4-1 Possible Types of Hot-Gas Ducting between the Reactor Core and Heat Exchangers

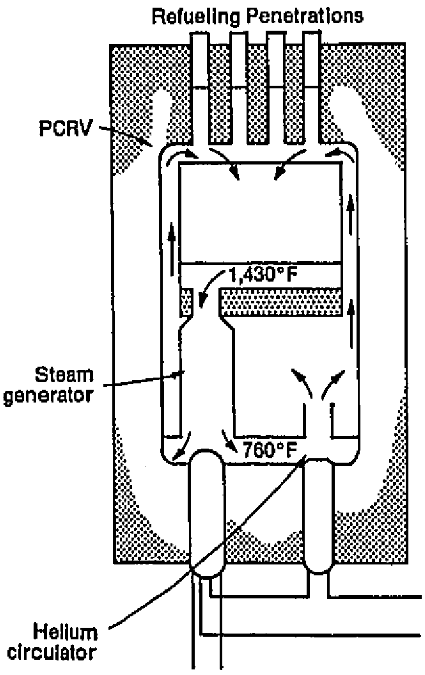


Fig. 3.4-2 Gas Ducting in the Fort St. Vrain Reactor [16]

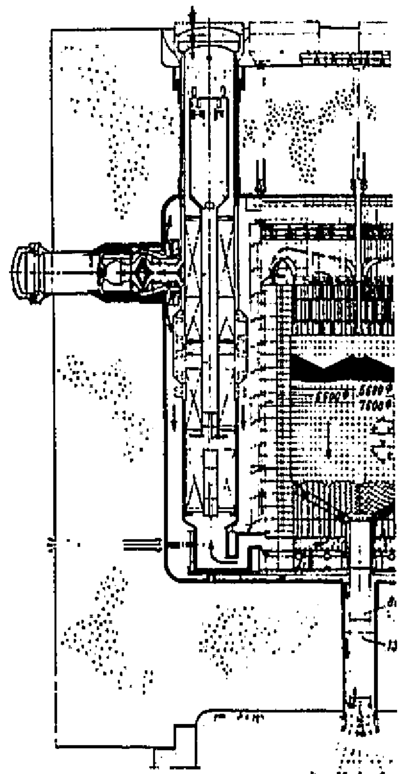


Fig. 3.4-3 Gas Ducting in the THTR [17]

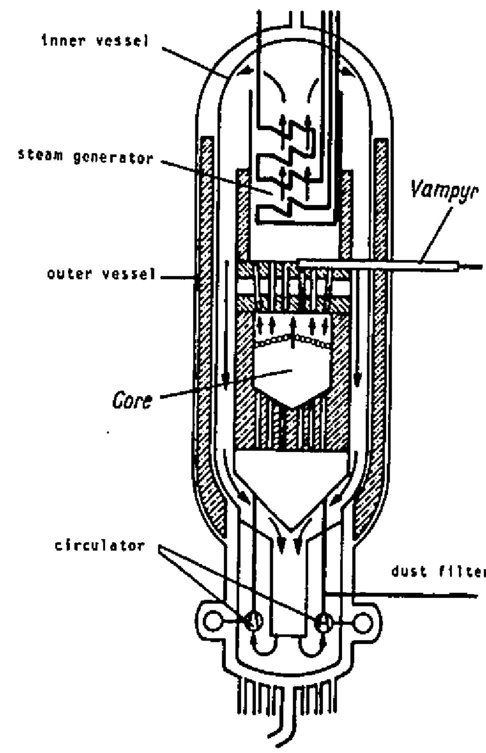
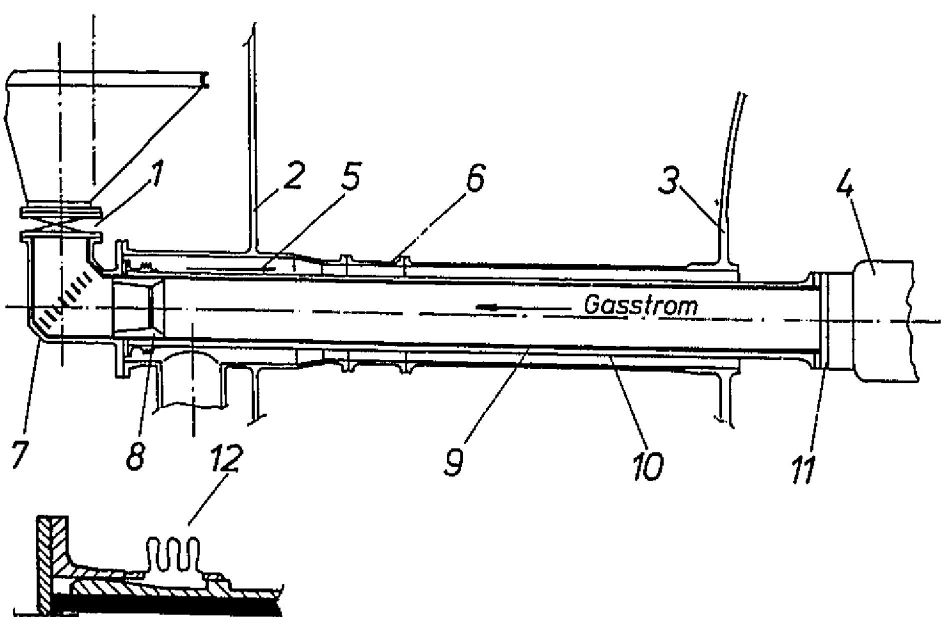


Fig. 3.4-4 Gas Ducting in the AVR [18]



1) Shut-off valve compensator
2) Heat exchanger
3) Reactor vessel
4) Hot-gas plenum
5) Gas stream
6) Removable part of duct
7) Elbow joint
8) Flow limitor
9) Insulation
10) Inner gas duct
11) Replaceable connection
12) Detail of friction spot at compensator

Abb. 3.4-5 Gas Ducting in the Windscale Reactor [19]

If the dimensions of the tube ducting for the 3000 MW(th)-plants discussed is not to exceed a diameter of 1 m, the number of loops to be chosen should be between 8 and 12, so that the flow rate and therefore the pressure losses in the ducts are not to become too high. Fig. 3.4-6 demonstrates this:

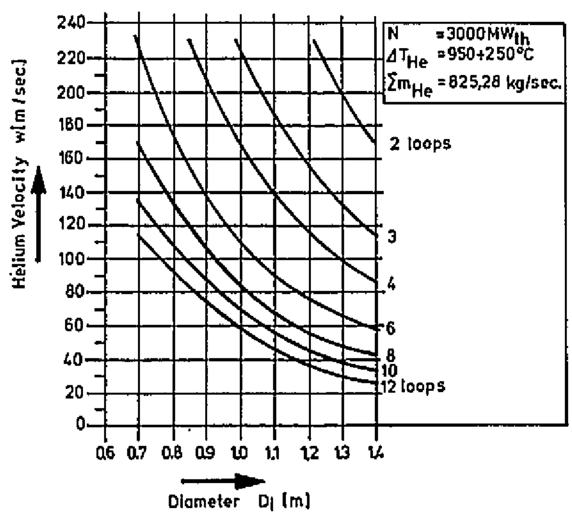


Fig. 3.4-6
Connection between Number of Loops, Helium Velocity and Tube Diameter

Variant 1

In variant 1, the penetration ducts of the reactor vessel are insulated on the inside with metal foils or with wool insulation of sufficient thickness (s. fig. 3.4-7 to 3.4-11). For the flow of the hot gas, sheet-metal liner is necessary on the inside. This liner e.g. Hastelloy, operates at the helium outlet temperature of the reactor of 980 °C. The temperature of the insulation decreases linearly up to the water-cooled penetrations ducts. The temperature of the penetration ducts is held at a maximum of 50 °C, because of the adjoining concrete. The usual thermal conductivity of metal foils is 0.3...0.5 W/m °C and that of wool insulation is (e.g. Kaowool) 0.1...0.25W/m°C. Therefore wall thickness of about ∿ 35cm or∿15 cm are required. The heat flux through the penetration ducts will in this case average ∿ 1 kW/m^2. This is a value which can be accommodated by water cooling surrounding the penetration ducts. The advantage of this design of gas ducting is that the diameter of the penetration ducts is smallest here as compared to the variants described in the following. The following are the disad-

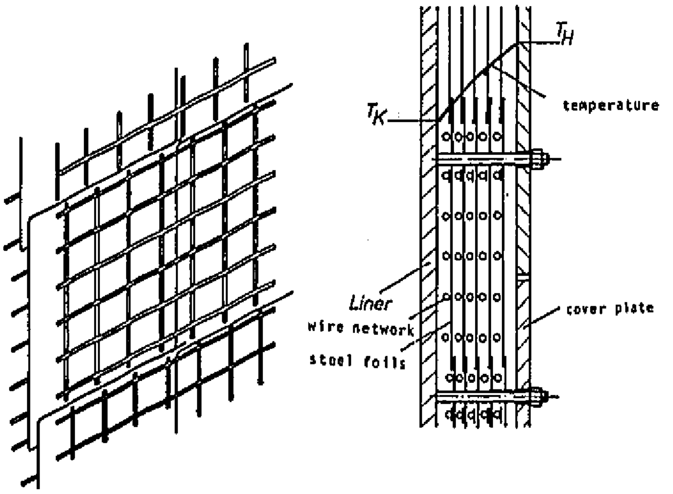


Fig. 3.4-7 Structure of a Foil Insulation [20]

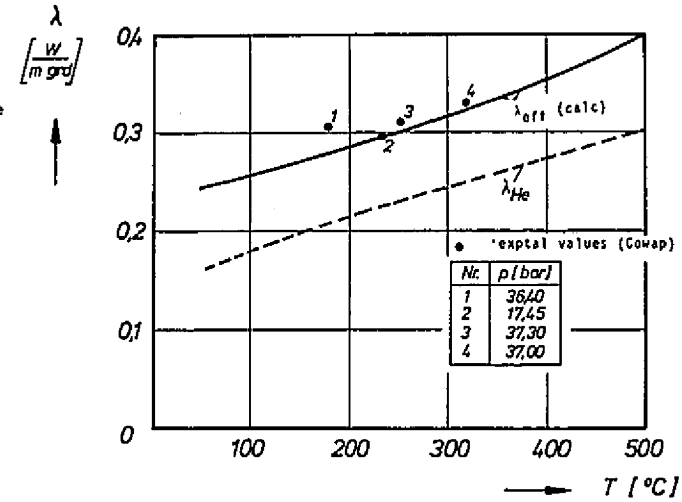


Fig. 3.4-8 Thermal Conductivity of a Foil Insulation [20]

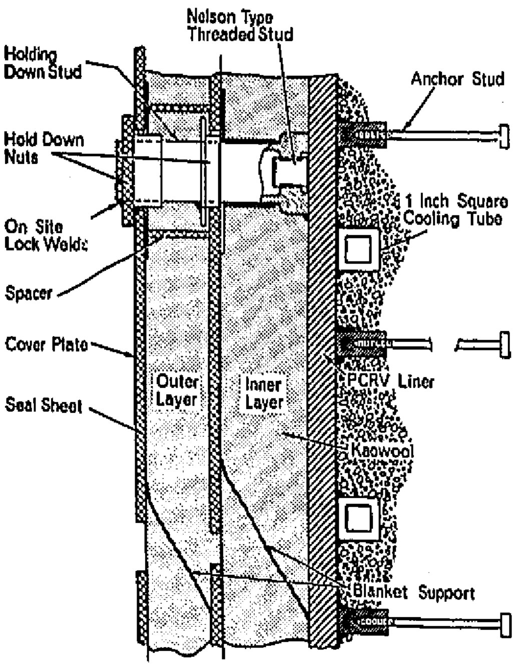
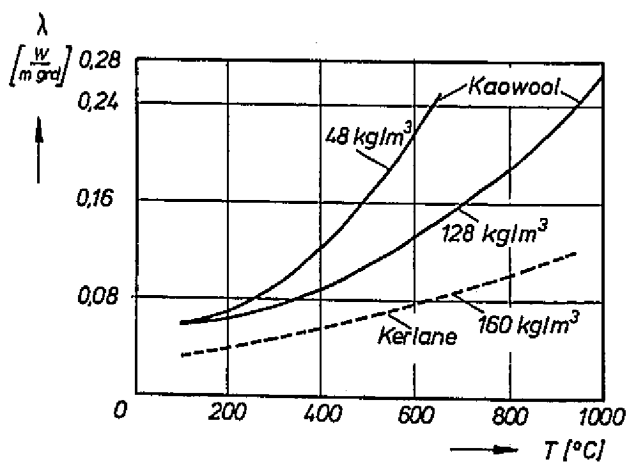


Fig. 3.4-9 Structure of a Wool Insulation [21]



composition [%]

	Al_2O_3	SiO_2	Fe_2O_3	TiO_2	MgO	CaO	Na_2O	B_2O_3
Kaowool	45,1	51,9	1,3	1,7	Spuren	0,1	0,2	0,08
Kerlane	44,0	51,8	1,6	1,3	0,6	0,3		

Fig. 3.4-10 Thermal conductivity of Kaowool and Kerlane in Air [20]

vantages especially at high temperatures:

- Operation of a material at high temperatures (T = 980 °C) for more than 30 years, whereby a replacement seems very difficult.

  It is possible that the impurities in helium i.e. traces of H_2O, H_2, CO, CO_2 and CH_4, can cause changes in the metallic structure.

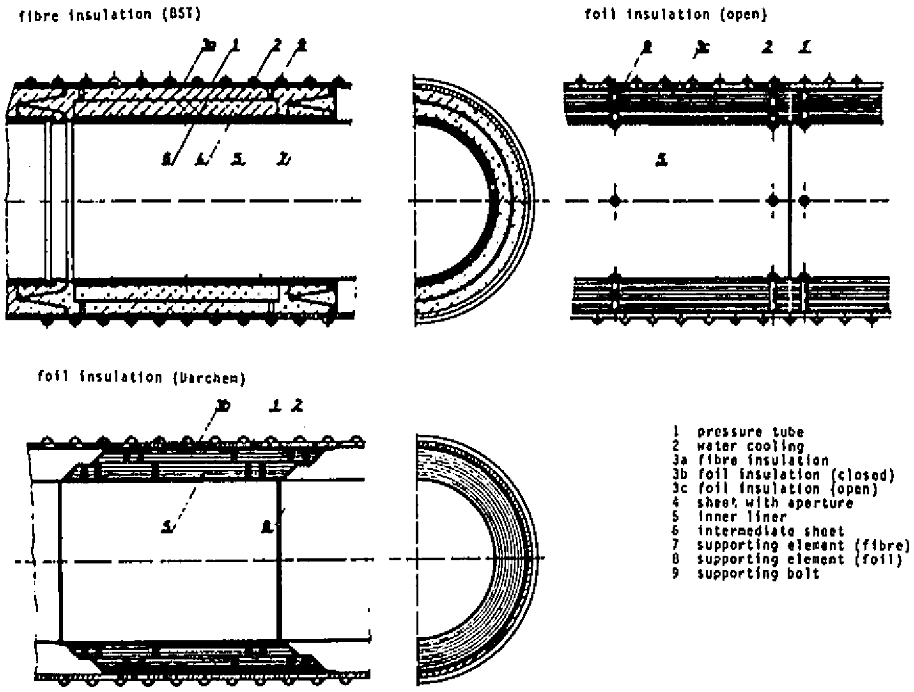


Fig. 3.4-11 Design of Internally Insulated Ducts [22]

Especially for the thin-walled sheet-metal liner required here, this can lead to difficulties

- Leakage streams are difficult to avoid and can result in azimuthal and axial dissymmetries in the duct wall temperatures.
- Possible damages to the thin-walled metal due to sudden pressure drops limit the permisible pressure gradient per unit time
- Deposition of solid fission products makes repairs and inspection work on the insulation and penetration ducts very difficult.
- The proposed materials and designs cannot be tested for the corresponding time of operation.
- Risk of friction welding at adjoining spots of the sheet metal.

<u>Variant 2</u>

In this variant 2, the principle of coaxial ducts is applied. The hot-gas duct is cooled in counterflow with cold helium at 250 ^{o}C. The transport of hot gases at 980 ^{o}C and more is thus possible, without wearing out metals at high temperatures. Through a suitable choice of velocities of the hot gas and the cold gas, as well as the wall thickness of the hot-gas duct, the duct wall temperatures can be held under 550 °C. As the pressure difference between the hot gas and the cold gas is only about 0.7 b (pressure decrease in core and core internals), the mechanical design of the hot-gas duct can be achieved even considering all thermal stresses. In fig. 3.4-12 the corresponding data and results are shown.

Data

Reactor power	:	3000 MW
Number of Loops	:	6
Helium flow/loop	:	136.6 kg/sec
Length of duct	:	10 m
Dimension of pipes	:	see figure
Pressure hot gas	:	39.3 bar
Pressure cold gas	:	40 bar
Temperature hot gas	:	950 °C
Temperature cold gas	:	250 °C
Pressure drop hot gas	:	0.007 bar
Pressure drop cold gas	:	0.03 bar
Velocity hot gas	:	95.8 m/sec
Velocity cold gas	:	61.2 m/sec
Material of hot duct	:	NIMONIC ALLOY 80 A
$\sigma_{allowed}$ (580 °C)	:	20 kp/mm^2
σ_{max}	:	16.5 km/mm^2

Temperature Profile

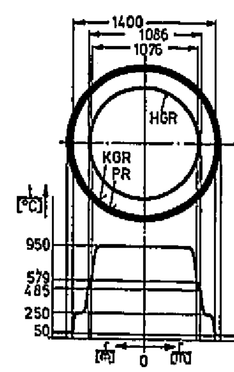


Fig. 3.4-12 Data of a Coaxial Duct cooled in Counterflow without Inner Insulation of the Hot-Gas Duct (Variant 2)

This variant has a number of design difficulties: The temperature difference between stand-still and the operating state is over 500 °C. For a hot-gas duct length of about 8 m, this means a change of length of 5 cm has to be compensated. Use of compensators or sliding seals can solve this technical problem. However, for duct diameters of 800 mm and more, both these solutions do not correspond to the status of technology. The hot-gas duct must be placed on roller-bearings, to allow movements during start-up and shut-down.The hot-gas duct is firmly welded on to the cold-gas plenum at the reactor bottom and so a clear separation of hot gas and cold gas is possible. At the same time, the hot-gas duct can thus be cooled indisputably by cold gas in the region of the hot-gas plenum also. Inside the penetration duct there is an additional duct, which fulfills the following three functions:

1) heat protection duct for the penetration duct;
this is achieved by stagnant helium between the penetration duct and the above-mentioned duct.
2) As flow limiter for helium in case of damage to the outer tube.
3) It serves as carrier duct, as the rollers for the hot-gas duct are welded on to this.

For a loop design the outer pressure-withstanding cold-gas duct must also have a compensator, which, however, only has to operate at 250 ^{o}C and 40 b. Alternatively the duct must be water-cooled to avoid a compensator.
In the described design concept, problems which occur by friction of metals in dry helium or possibly by welding friction can be solved by placing a layer of suitable material (e.g. titanium carbide or nitride) on the bearing contact. A further design problem, which has to be solved, is the question of vibrations, which could occur due to the circulator noises. Ribs for stiffening the duct on the outside could solve this question.

<u>Variant 3</u>

Variant 3 is a compromise between the two previously described variants. Through inner insulation of the hot-gas duct e.g. by replacable slide-in pieces, the temperature of the hot-gas duct can be lowered to under 300 ^{o}C. In this way it is possible to design the hot-gas duct with great stability without consideration of thermal stresses, so that vibratory problems do not occur. However, if wool insulation or metallic foil is used, here also the problem occurs that the inner sheet-metal liner takes on the temperature of the hot gas. Therefore, the design should allow for a replacement, which is possible in a loop construction.

In order to prevent thermal stresses in the carbon stone duct, this can also be divided into several segments in the circumferential direction. In fig. 3.4-13 a possible design of such a duct is shown. The especially difficult point here is the connecting part to the core structure, and the fastening of the ceramic pieces inside the supporting tube.

Variant 4

In variant 4 the difficulties of the connection to the ceramic core structure can be avoided, when the overall gas ducting is carried out in carbon stone. Favourable temperature distributions can be attained here, too, by counterflow cooling through borings for cold gas arranged symmetrically around the hot-gas channel. It still has to be found out, however, whether the occurring temperature gradients of 80 ^{0}C/cm allow tolerable stress distributions. The sealing off between hot gas and cold gas can be improved by a possible placement of metallic ducts in the cold-gas borings. The required diameter of the penetration duct or the cold-gas duct will, however, be considerably enlarged in this design. Other materials such as Al_2O_3 or similar ceramics can be considered, too.

Variant 5

To avoid some of the difficulties with the carbon-stone structure, the cold-gas ducting can be constructed in the form of a telephone disc made of cast chromium alloy. The inner hot-gas duct can be insulated by different known materials.

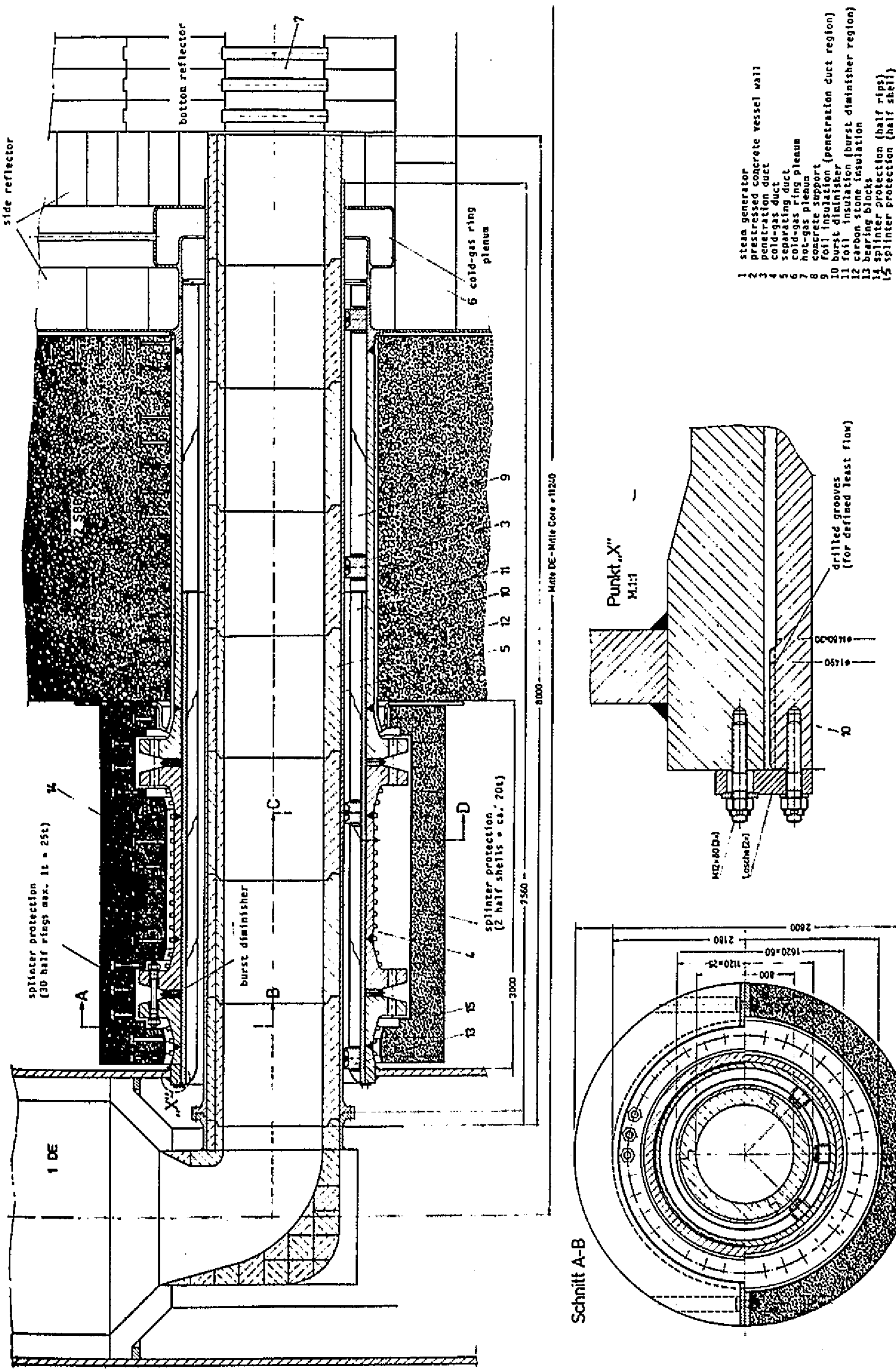


Fig. 3.4-13 Coaxial Duct with Inner Insulation of the Hot-Gas Duct (Variant 3)

3.5 Intermediate Heat Exchangers between the Primary Circuit and the Chemical Process

An intermediate heat exchanger (IHX) circuit separating the nuclear and the chemical plant is often proposed for the following reasons:

- Hydrogen permeation from the process to the helium circuit can be reduced.
- The rate of tritium permeation from the helium circuit to the process can be lessened.
- Contamination of the process heat exchangers is prevented.
- A complete separation of the chemical and nuclear plant is attained.
- The in-rush of water and gases into the core in case of an accident (tube rupture in steam generator or in steam reformer) is prevented.
- For some special processes (e.g. steam gasification of coal, water splitting processes) it is better for the process heat exchangers to be placed outside the containment, when solid materials are handled.

According to the explanations in Vol 4 § 8 and 9, the first two arguments do not indispensably justify an intermediate heat exchanger circuit, as a removal of tritium, when necessary and technically feasible, could also take place in the primary circuit. The contamination of the process heat exchanger causes no problem so long as accessibility to the catalyst is guaranteed and so long as no repairs have to be carried out on the tube bundle.

The first requirement can be fulfilled by placing the filling and removal position of the catalyst also in a helium-filled plenum at the same pressure as the primary circuit above the tube support-plate. By means of a connection, the uniformity of helium pressure is guaranteed; at the same time, however, the entry of solid fission products into the plenum is prevented. The operation life of the reformer tube is assumed to be $\geq 10^5$ h, after that an exchange of the overall tube bundle takes place. An individual reformer tube, which is damaged during its life-time, is removed from operation by plugging off. Reserve tubes, however, have to

Aspect	Plant with IHX	Steam Reformer in the Primary Circuit	Evaluation *
1) Separation of nuclear and chemical plant	separation is very attractive because of explosive gases	partial inert gas filling of the containment (gas pipes in N_2-filled channels) explosive gas mixtures have to be avoided	(+)
2) Hydrogen permeation from process to helium circuit	IHX lowers permeation rate to core	measurements show that with a H_2O/H_2-ratio of 0.2...1 in the process, the permeation rate for a 3000 MWth-plant is limited to ~ $1 Nm^3/h$, which is no problem	(/)
3) Tritium permeation from helium circuit to process gas	tritium can be removed from IHX-circuit by gas purification	tritium can be removed from primary helium circuit, too. Calculations show that if all tritium which is released to the helium circuit permeates to the process gas, there are no major problems	(/)
4) Plate-out of solid fission products on components	steam reformer totally free of contamination	if fission products are plated out on reformer tubes, there are no problems for change of catalysts for a pebble bed reactor at 980 °C	(/)
5) Corrosion of metals in primary helium by impurities	the thin-walled IHX-tubes are easily corroded. Additionally the wall temperature is higher by 50...70 °C in comparison to reformer tubes	thick-walled reformer tubes (15 mm) are influenced not very much because the wall temperature is < 900 °C	(-)
6) Reactor outlet temperature	either the reactor outlet temperature must be 1000...1020 °C in order to have the same CH_4-conversion, or with 900 °C in IHX circuit the reformer heat transfer area must be larger	950 °C is sufficient for the process and allows heat fluxes of the order of 60...70 000 $Kcal/m^2$ h	(-)
7) Corrosion in the reactor core	in case that all the reactor heat is transferred by the IHX, the specification for graphite corrosion in the core and structural graphite can be omitted	corrosion rate is no problem for 950 °C, if amounts of H_2- and H_2O-leakages are limited	(+)
8) Accident "water ingress into core by rupture of steam generator tube"	is avoided	no major problem as in steam generating HTR	(+)
9) Choice of pressures in the circuits	possibly more flexibility for choice of reforming pressure	the best choice is $p_{reforming} = p_{helium}$ · 40 b with a slightly higher pressure on the process side	(/)
10) Additional barriers for separation of primary circuit from the outside	1 more barrier	possibly same effect if duplex tubes are used for the steam reformer	(+)
11) maximum wall temperature	T_{wall} · 950 °C with T_{He} 1000...1020 °C	T_{wall} < 900 °C for T_{He} < 950 °C	(-)
12) Efficiency of the plant	if the total reactor heat is transferred, ~ 50 MWel additional power for IHX is needed	----	(-)
13) Heat fluxes in the heat exchangers	in IHX of the order of 30000 Kcal/m^2h; because thin-walled tubes with small diameters are used, power density in IHX can be ~1.5 MW/m^3	steam reformer bundles have ~60...70000 Kcal/m^2h; because thick-walled tubes with large diameters are used, power density in reformer is 1.5 MW/m^3	(/)
14) Arrangement of heat exchangers in reactor vessel	pod boiler, same conditions because nearly same power density in heat exchangers. Question not important for loop system		(/)
15) Additional components	- gas ducts: inner diam. D_i ~ 1 m - valves: D_i ~1 m, T_{wall} 900 °C and hot, fast acting, if IHX outside containment - hot penetrations for vessel and containment	---.	(-)
16) Technical problems for the additional components	- insulations - heat exchangers, if T_{He} 950 °C - valves not available - IHX-vessel closure very difficult because of penetrations for hot gas	if inner pigtails are used for the reformer tubes, the temperature for the penetrations is limited to 600 °C	(-)
17) After-heat-removal from the reactor core	separate auxiliary loops (4 x 50) are needed	can be achieved by 8...12 parallel production loops	(.)
18) Reliability of the plant	by addition of a IHX-circuit poss. higher probability for failures	---	(-)
19) Status of development	valves, pipes, penetrations, heat exchangers have to be developed and tested on a large scale	steam reformer bundles will be tested in large sections in near future	(-)
20) Costs	by addition of IHX which transfers the total power of the reactor, the costs of the nuclear heat are higher by 30	---	(-)
21) Safety	- no penetrations through vessel and containment are weak points - no fast acting valves are possibly not sufficiently available	accident "rupture of steam reformer tube" is easy to handle, because production of 1 tube is limited to ·400 Nm^3 H_2 + CO/h	(-)

* (+) = for or (-) = against IHX. '/ means that this is not a strong point for decision

Table 3.5-1 Arguments for and against a He/He Intermediate Circuit be installed from the beginning (5...10 % more capacity).

As far as the contamination values are considered, even a deposition on the reformer tubes of 50 μCi/cm^2, which seems to be pessimistic

The complete separation of the chemical and nuclear plant would certainly present an optimum solution from the operational and safety point of view, if apart from the heat exchangers it did not require additional components, like long hot-gas pipes, compensators, quick-acting hot-gas valves and hot-operating penetrations through the reactor vessel and the containment.

The in-rush of water into the core, and the formation of water gas connected with it, is a known and controllable accident for two-circuit HTR plants. The additional penetration of hydrogen into the core after the rupture of reformer tubes is limited and causes no special problem.

An exact technical analysis of the problem is justified because an IHX seems to be necessary for steam gasification of coal and because there could be the wish to possess an alternative solution.

In table 3.5-1 all known arguments for and against the use of intermediate circuits are compiled again.

Flow Schemes for IHX Circuits

The following figure 3.5-2 shows 5 basic flow schemes:
- Steam reformer (SR) and steam generator (SG) are directly integrated into the helium circuit - solution without an IHX (1)
- The SR is heated by an IHX, the SG is operated directly in the helium circuit of the reactor. The following different variants are thinkable:

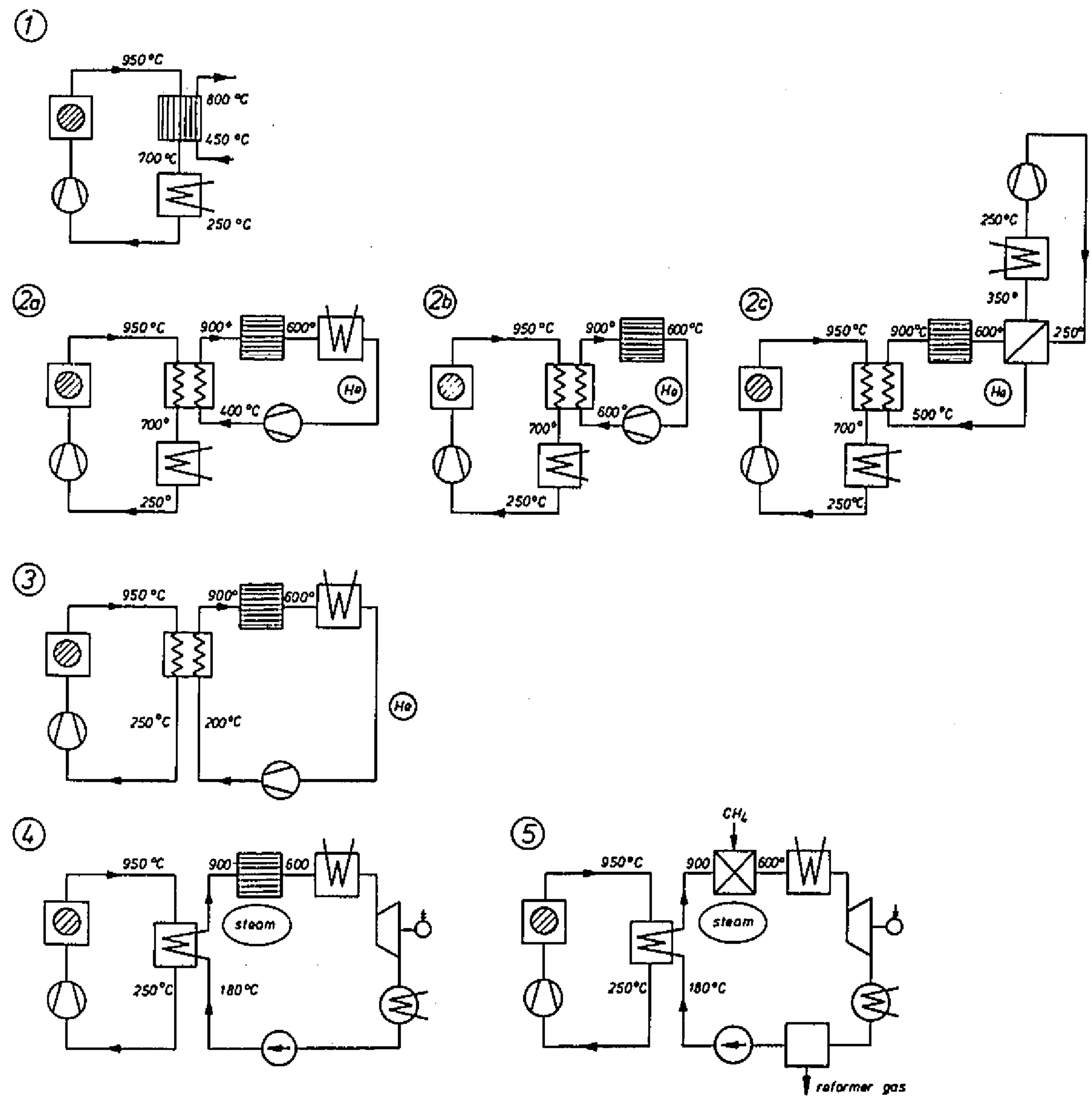


Fig. 3.5-2 Flow Schemes for Intermediate Heat Exchanger Circuits

- additional SG in the IHX (2a)
- hot circulator in IHX (2b)
- recuperator in IHX (2c)

- Both SR and SG are heated by an IHX, the total reactor heat is transferred via the IHX(3)

When steam is used as an IHX medium, special circuits (4 and 5) can be considered:

- SR is heated by steam, steam is expanded in the turbine, condensed and is pressurized as feed water. The SG would be operated at about 41 bar(4) at the steam exit.
- Operation of a catalyst-filled bed with a steam/methane ratio of ∿ 8/1, so that the sensible heat of the hot steam is utilized as heat for reforming. After expansion in the turbine and separation of the reformer gases, the feed water is again pressurized (5).

The circuits 2a, 2b and 2c vary especially regarding the quota of high-temperature process heat, heat fluxes as well as circulator power in the IHX circuit as is shown in table 3.5-3.

Parameter	1	2a	2b	2c	3
Q = SR	60,000 $\frac{kcal}{m^2h}$	36,000	36,000	36,000	36,000
Q = IHX	-	60,000	30,000	45,000	22,000
Q = Recup.	-	-	-	-	-
N_{el} IHX	-	40 MWel	85 MWel	60 MWel	80 MWel
$N_{process}/N_{reactor}$	0.36	0.22	0.36	0.27	0.36
Problems		small amount of process heat	hot blower in IHX circuit	large heat transfer areas	space in reactor vessel in integrated design

Table 3.5-3 Comparison of Values for Heat Transfer and Circulator Power for Different IHX-Variants

Media for IHX

Next to helium, basically CO_2, N_2, steam, Ne or Ar could be used as intermediate circuit media. An analysis of the required heat transfer areas as well as of the necessary pumping power (see fig. 3.5-4) in the IHX circuit shows helium to be the most economical medium. CO_2, N_2 and steam have very high pressure losses and therefore require large circulator powers. Steam as IHX medium only has advantages when it is condensed in the circuit .

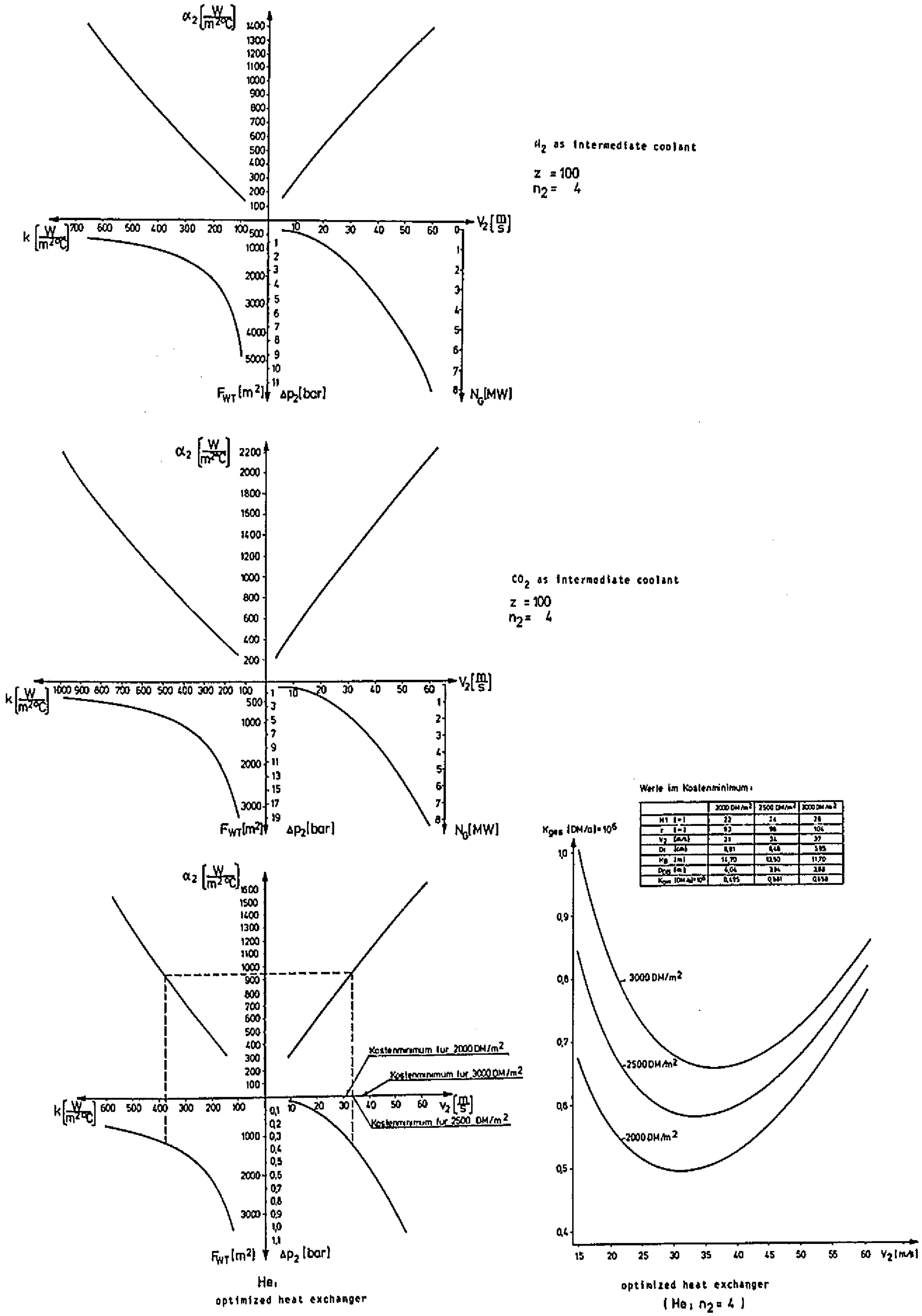


Fig. 3.5-4 Heat Transfer and Pressure Loss for Different Intermediate Circuit Media

Arrangement of IHX, SR and SG

When an IHX is used, the following arrangements are basically possible (see fig. 3.5-5 and table 3.5-6).

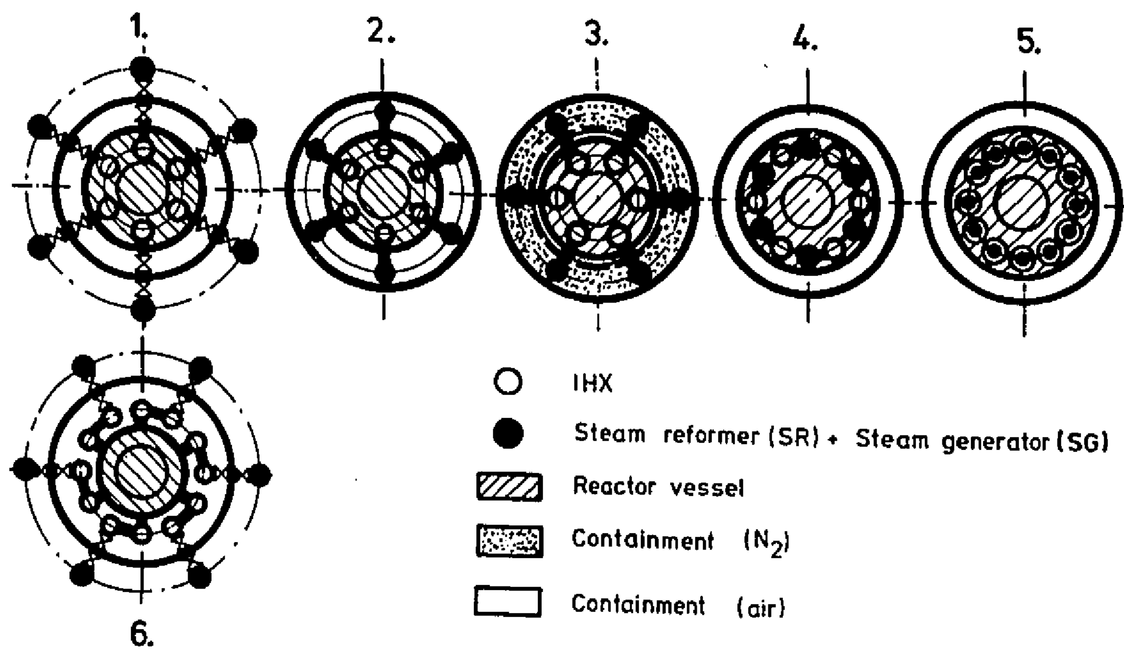


Fig. 3.5-5 Possible Arrangements of Heat Exchangers in the Primary Circuit

Variant	IHX arranged in	SR and SG arrangement	Gas filling in containment	Remarks
1	reactor vessel	outside the containment	air	- fast-acting valves necessary (T = 900 °C, Di = 1 m) - long helium transport ducts
2	" "	inside the containment	air	- danger of formation of explosive gas mixtures in containment
3	" "	inside the containment	inert gas	- rupture of gas ducts in containment is controllable
4	" "	in additional cavities in reactor pressure vessel	inert gas	- large-volumed reactor vessel in integrated design
5	" "	in the same cavities as IHX	inert gas	- use of double-walled tubes - hardly any need for more place in the vessel

Fig. 3.5-6: Possible Arrangements of Heat Exchangers in the Primary Circuit

Variant 6 is the correspodning arrangement of variant 1 for a loop design.

From the point of view of costs, feasibility and knowledge about intermediate circuit components, the most attractive design seems to be solution nr. 5.

Data and Design of an IHX-System

In order to convey an idea of a possible design of such a heat exchanger, a design example is given in the following in which all the reactor heat is transferred. The IHX circuit is divided into 2 parts: a heat exchanger for the high-temperature region (970 ... 550 ^{o}C) and a corresponding one for the low-temperature region (550 ... 270 ^{o}C). Both heat exchangers contain the same modules, in which the individual modules consist of coaxial ducts. Fig. 3.5-7 shows the principle of such a heat exchanger.

Hot Helium comes from the nuclear reactor through a coaxial duct to the inner part of the heat exchanger, which contains the high-temperature modules. From here the primary helium is led outside to the low-temperature part. All the coaxial heat exchanger tubes which are insulated from the inside to prevent too high heat losses, are welded on to a tube-plate (T $\sim$ 500 ^{o}C). Above this tube-plate, there are 2 gas collecting spaces for the secondary helium, one for the gas entering at 200 ^{o}C and the other for gas at $\sim$ 480 ^{o}C. The hot secondary helium (900 ^{o}C) is collected outside.

The diameter ($\sim$5m) and the height ($\sim$ 15 m) of such heat exchangers is such that they could also be integrated in a pod-boiler vessel.

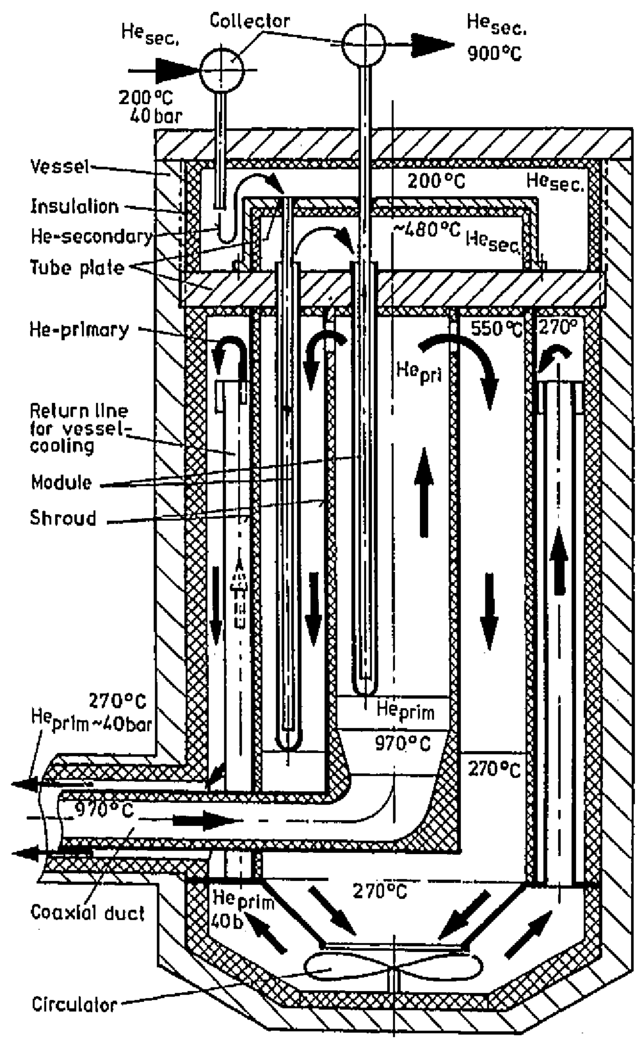


Fig. 3.5-7 Principle of Gas Ducting in an Intermediate Heat Exchanger

Reactor power	: 3000 MW
Number of loops	: 12
Mass flow per loop	: 68 kg/sec
Temperature on primary side	: 970 $\longrightarrow$ 270 ^{o}C
Temperature on secondary side	: 200 $\longrightarrow$ 900 ^{o}C
Pressure on both sides	: $\sim$40 b
Heat flux	: $\sim$35 000 kcal/m^{2}h
Area per loop	: 6200 m^2
Length of tubes	: $\sim$10 m
Diameter of bundle	: $\sim$5 m

Table 3.5-8 Design Data of an Intermediate Heat Exchanger

3.6 Choice of Reactor Design Parameters

The important thermal design data of HTR plants, which have been built or planned, are given in the following table 3.6-1. In the last row, the data of the PR 3000 are given. Reasons for this choice of reactor design parameters will be given in the following by employing a qualitative analysis of the influences of changing these values.

Plant	Thermal Power (MW)	Core Power Density (MW/m^3)	Helium Pressure (b)	Core Inlet Temperat. (°C)	Core Outlet Temperature (°C)
AVR	46	2.6	11	260	950
DRAGON	20	14	20	350	750
Peach Bottom	115	8.3	24	343	728
THTR	750	6	40	262	787
Fort St. Vrain	851	6.3	48.4	405	785
DRAGON Reference	1250	7.75	40	330	750
GA 1160	3000	8.4	49.9	319	741
interesting region for NPH-plants	2000..3000	4..6	30..50	250..450	900..1000
PR 3000	3000	5	40	280	980

Table 3.6-1 Design Data of already Constructed and Planned High-Temperature Reactors

3.6.1 Power Density in the Reactor Core

If the core power density is varied between 5 and 10 MW/m^3, which is considered to be interesting for HTR, results of the following form are obtained:

1. Regarding the fuel element design, mean power densities for the OTTO circuit of approximately 5 MW/m^3 can be achieved without resulting in unpermissible fuel element temperatures (calculation: T_{max} = 1100 °C in the fuel, ΔT_{max} ∿ 300 °C in the matrix, T_{max} on the surface = 1050 °C, limits: 1250 °C fuel temperature for THTR, 1050 °C surface temperature for THTR)

2. The capital costs of the nuclear part decrease with increasing power density. At the same time, the energy costs for the increased pumping power rise with increasing power density. On the whole, the following tendency results for the total costs influenced by the power density (see fig. 3.6.1-1).
   The cost curve has a parabolic form with a minimum in the region of 5-6 MW/m^3 (at electricity costs of 3 DPf/kWh (el)).
   When the plant is once put up, however, (i.e. the annual capital-dependent costs remain constant) and when simultaneously, the electricity generation costs rise in general due to a rise in prices, the minimum of this cost curve is gradually shifted towards lower power densities during the period of operation. After 20 years it would probably be at 4 MW/m^3. On the whole, there is no reason on economy grounds to aim towards power densities of above 5 MW/m^3.

3. The fuel cycle costs are practically independent of the power density. Savings in manufacture cost and low interest rates for the first reactor core (at high power densities) are compensated for by higher credit notes for bred fuel at low poer density (0.53 DPf/kWh(el) at 5 MW/m^3, 0.51 DPf/kWh(el) at 9 MW/m^3 at 1973 ore and separation costs).

   In 30 years of operation of a plant, the influence of the credit note towards better conversion will certainly lead to a tendency

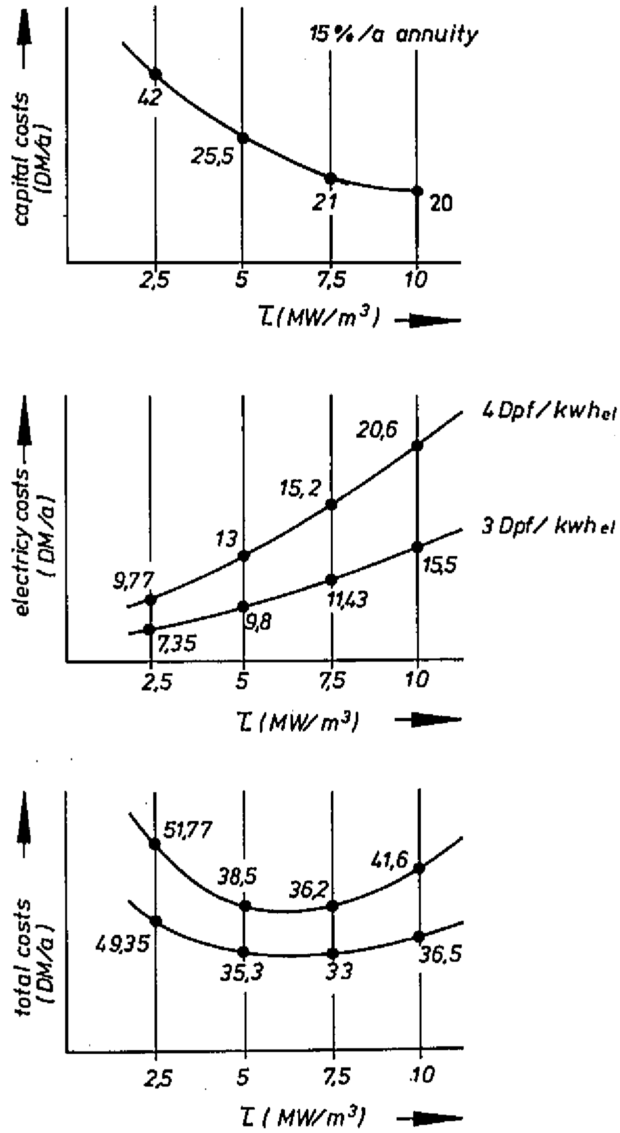


Fig. 3.6.1-1
Costs of Primary Parts dependent on the Core Power Density (including Reactor Vessel, Control Rods, Core Installations, Loading Installations and Circulator;) Consideration made for a Loop Design

towards low power densities due to the increasing uranium costs. In addition, due to the increased throughput of fuel elements, the influence of manufacture costs will decrease in future, too.

4. Temperatures in the core during normal operation and accidents, (e.g., in-rush of water, fall-out of the coolant) are lower at low power density.

5. The release of fission products and the plate-out on the components connected with it is less at low power denity because the temperatures are lower.

6. The irradiation dose on the side and top reflector decreases with lower power density. At 5MW/m^3 mean power density, a design life for the upper side reflector of 30 years seems to be reasonable.

Despite the calculations, these parts of the reflector should be removeable to enable repair and exchange.

By raising the power density to e.g. 10 MW/m^3, the reflectors would have to be changed within the amortization period (17 years) in any case.

7. Flow behaviour of the pebble bed is more favourable at high power density (smaller core diameter). However, it can be made sufficiently uniform by several parallel disloading exits (3...7 exits).

   Present investigations show that the fuel element ball flow is uniform in the upper half of the reactor core, when the height/diameter ratio is 0.646 (see fig. 3.65-1). In the upper part of the core the major portion of the heat is generated. Therefore, dissymmetries in the ball flow in the lower half of the core do not strongly affect the deviation in burn-up.

   It may be necessary to increase the number of fuel ball exits to as many as 6 to achieve adequate uniformity in a core with a smaller height/diameter ratio. There are other possibilities for the design of the core bottom. These possibilities have, however, to be thoroughly considered and designed yet.

8. The stress on the fuel elements is less at low power density because the temperature gradients. are smaller

9. The conversion factor increases with decreasing power density (protactinium absorbs neutrons parasitically and this absorption is power density dependent.)This is important due to the rising uranium prices for a high-converter system.

10. At higher power density very large circulators are necessary because the pressure drop in the core rises with power density. This can cause space problems, especially for integrated designs.

11. At low power densities a larger prestressed concrete vessel is mecessary. Today vessels with internal diameters of 25 m (at 40 b) can be constructed.

12. The core has a higher heat capacity at low power density, i.e. after accidents the starting of the emergency cooling can be delayed depending on decreasing power density.

This has certain consequences for the design of the emergency cooling system (it can be much smaller). In the case of a failure of all active after-heat removal systems, which is a hypothetical accident not considered in the licensing procedure today, the core is heated up due to the heat production by β-decay of the fission products. Depending on the core power density the following results are obtained using a very rough estimation:

After-heat production: $P_N(t) = P_o \times 0.13 \times t^{-0.283}$

Energy balance : $\int_0^t P_N(t) \times dt = M_c \times c_p \times \Delta T$

Heating up of core : $\dfrac{\int_0^t 0.13 \times t^{-0.238}\, dt \times \bar{L}}{c_p \cdot Z \cdot \frac{4}{3}\pi R^3 \cdot \rho_c} = \Delta T$

P_o = reactor power (3000 MW)
c_p = specific heat of graphite (0.4 kcal/kg °C)
Z = number of fuel elements per unit volume (5400)
R = radius of ball (3 cm)
ρ_c = density of graphite (1.6 g/cm^3)
L = core power density
ΔT = heating up of core

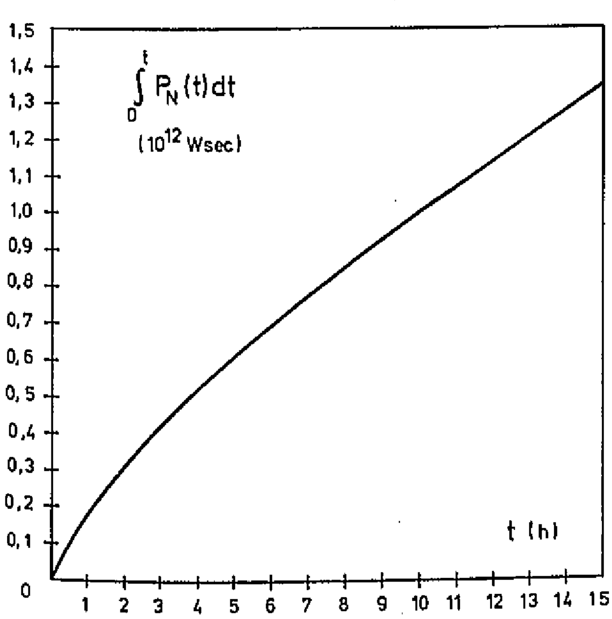


Fig. 3.6.1-2 Accumulated After-Heat Production in a 3000 MW Reactor

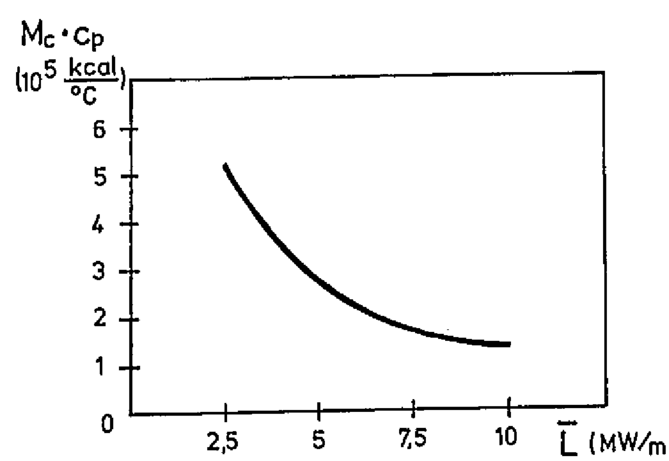


Fig. 3.6.1-3 Storage Capacity of the Core for After-Heat at Varying Power Density (1kcal≘ 4.186 x 10³ wsec)

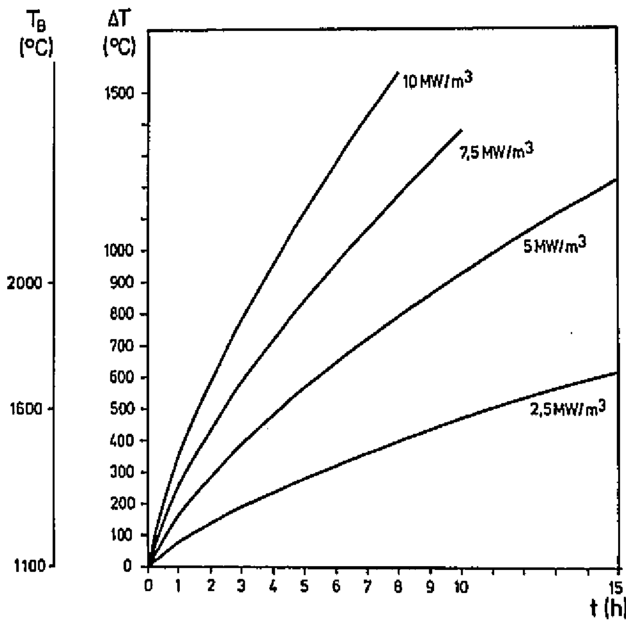


Fig. 3.6.1-4: Dependence of Adiabatic Heating Up of the Core on Time and Power Density (T_B = Maximum Temperature of the Fuel Element)

Fig. 3.6.1-4 can be interpreted as follows: When a power density of 5 MW/m^3 is chosen there are $\sim$4 hours time in which important measures for the starting of the after-heat removal or extreme manipulation such as emergency disloading of the core can be accomplished. During this time, the fuel element temperatures remain under 1600 °C. At this temperature during this short time, release of fission products is not intolerably high. At 10 MW/m^3 the fuel element temperature during the same time would have risen above 2000 °C and the temperature of metallic installations would already be so high that some of these structures would have melted (e.g. suspension of the top reflector).

In reality a part of the heat is dissipated by the cooling arrangement of the vessel liner surrounding the core structure before such high temperatures are reached. After 10 hours, the after-heat production is about 6 %o of the nominal reactor power. This heat can on principal be removed by the liner cooling (normally 2 kW/m^2, 1000 m^2, ΔT(water) $\sim$ 5 °C) by increasing the heating up interval to 100 °C. Additionally, the natural convection in the core will dissipate heat, too.

For such a consideration it is assumed, however, that heat can be transferred from the core limits without too large a temperature elevation in the core. Small dimensions or ring cores with a diameter of 4 - 5 m would be advantageous from this aspect.

By decreasing the power density to 2.5 MW/m^3, the conditions for after-heat removal would again be considerably improved.

13. The arrangement eat exchangers in the pod-boiler vessel automatically leads to low power densities as certain spacing has to be left for the heat exchangers between the cavities (pod diameter/spacing $\sim$ 2/1 at 40 bar design pressure).

In the case of the pod-boiler, a power density of 10 MW/m^3 for nuclear process heat cannot be realised at all because of the space needed for the heat exchangers, as the following consideration shows:

For a distribution according to 950 ... 700 ... 250 ^{o}C (steam reforming of methane,steam generation), a power density of 1.2 MW/m^3 is possible in the steam reformer. This leads to e.g. 8 pods of 4.4 m each for the steam reformers and 4 pods of 4.4 m each for the steam generators. In order to have a sufficiently large spacing between the heat exchangers in the container, they have to be placed in a circle with a diameter of approx. 25 m.

This leads automatically to a core cavity of 15 m diameter. In this cavity, a core with 5 MW/m^3 fits in.

A power density of 10 MW/m^3 in the core can only be actually realized by steam generating systems (with 6 steam generators, height approx. 8 m, diameter approx. 4 m, i.e. $L \sim 6$ MW/m^3 in the steam generator) if pod-boiler designs are used (see fig. 3.6.1-5)
In case of a loop arrangement of the heat exchangers, it would be possible to raise the power density of the reactor core. However, cost estimations made before show that here no economical

interests exist for increasing the power densities.

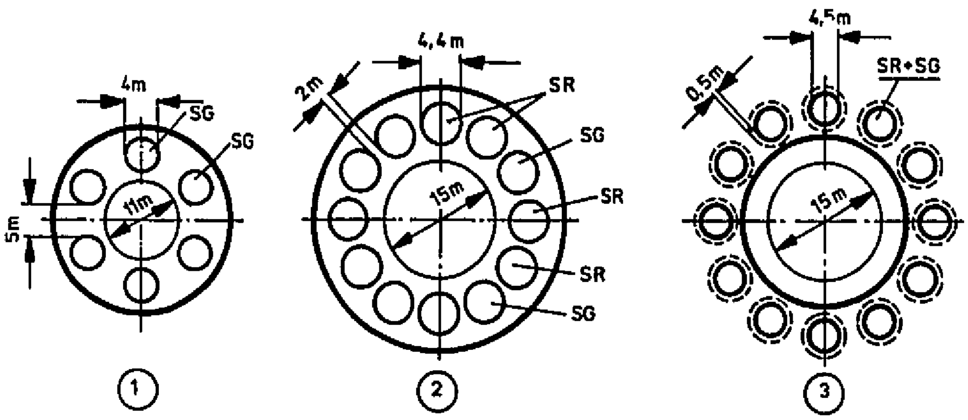


Fig. 3.6.1-5 Dimension of the Vessel by Changing Over from Electricity Production to Process Heat Applications.

1) Gulf 1160 MW(el) pod-boiler

2) PR 3000 pod-boiler for steam reformer (950...700 °C)and steam generator (700 ... 250 °C)

3) PR 3000 loop system for steam reformer (950 ... 700 °C) and steam generator (700 ... 250 °C)

From all the considerations made here, it is concluded that increasing the core power density above 5 MW/m^3 brings no advantages if economy is considered, but it means a lot of disadvantages, if the technology and the safety of the system are discussed.

Table 3.6.1-6 summarizes the main aspects once more.

Aspect	Remarks	Argument pro (+) or against (-) a higher value of power density than 5 MW/m³
1) Costs	- fuel cycle costs are nearly independent of power density (P.D.) - investments are lower for higher P.D. - cost for power of He-circulators is raised with higher P.D. - total costs show a flat minimum for a P.D. of 5... 6 MW/m³	(/)
2) Fuel element behaviour	- with higher P.D. all temperatures (fuel temp., surface temp., temp. gradients) are higher	(-)
3) Fast dose on reflector	- with higher P.D. all fast doses at top- and side reflector are raised. Reflector must be removed earlier	(-)
4) Conversion rate	- with higher P.D. the conversion rate drops because of higher parasitic absorption in protactinium	(-)
5) After-heat removal	- heat capacity in the core is higher for lower P.D. The time after which the after-heat system must be operated is longer for lower P.D.	(-)
6) Flow of balls in core	- for small cores, i.e. cores with high P.D. the flow of balls in core is much more parallel than in big cores. For cores with low P.D. more discharge tubes (4...7) or special core designs (ring core or core type bottoms) must be used	(+)
7) Diameter of cavity and dimensions of reactor vessel	- for a podboiler design automatically a P.D. of 5 MW/m³ results because of the lower power density of the heat exchangers (12 pods of 4.5 m ⌀ are needed instead of 6 as in case of steam generating plants) - for a loop design from the point of view of vessel diameter a higher value of P.D. could be used, but no cost advantage can be seen	(/) (/)
8) Power of the blowers, dimension of blowers	- with higher P.D. bigger blowers, possibly problems with space in integrated designs	(-)
9) Tritium generation	generators of tritium in ^3He and in fuel elements and structural graphite increases with P.D.	

Table 3.6.1-6 Aspects regarding Power Density

3.6.2 Reactor Inlet Temperature T_i

The determination of the reactor inlet temperature has considerable influence on the following:

- The design and life expectancy of the upper and side reflectors
- The temperature rise of the coolant through the core, and with it the pressure drop and the circulator power
- The temperature distribution in the core, and with it the release of fission products
- Insulation of the entire system
- The splitting of reactor heat into process heat and electrical power
- Design of the heat exchangers, in particular the size of the economizer in the steam generator
- Fuel element design

Regarding the details, the following comments can be made:

1. By increasing T_i and keeping all the remaining data in the primary circuit constant, the temperature rise through the core becomes smaller, and at the same time the coolant mass flow increases. The nuclear reactor power which can be utilized for process heat is increased.
2. By increasing the coolant mass flow, the heat transfer to the reformer tubes improves: The heat transfer coefficient on the helium side can be considerably increased.
3. The heat flux of the economizer is considerably increased by raising the reactor inlet temperature.
4. By increasing the coolant mass flow, the circulator power is raised very much. Thus the efficiency of the plant is decreased.
5. In addition, the circulator power is increased with an increase in core inlet temperature T_i.

6. Circulators with higher inlet temperatures are more difficult to construct.
7. With higher coolant mass flow, the velocity or the size of all primary gas-containing components have to be increased. In the first case the pressure drop will be higher with corresponding power, in the second case the plant costs are increased.
8. The permissible fast neutron dose in the graphite installations (the upper reflector and upper part of the side reflector) becomes less as the reactor inlet temperature is increased due to the typical irradiation behaviour of reactor graphite. Thus the service-life of these installations decreases with increasing core inlet temperature (above 350 ^{o}C).
9. The cooling of the entire primary circuit system (heat exchanger walls, ducting, thermal shield, cold-gas plenum) is more difficult with rising core inlet temperature. This also calls for the use of materials of higher quality. Increased insulation of all primary gas-containing components is necessary.
10. The cooling of the control rods is more difficult with increasing inlet temperature.
11. Bearings in dry helium cannot be lubricated with MoS_2 at a temperature of more than about 250 ^{o}C - 300 ^{o}C. Therefore, increased problems of friction and fretting have to be considered.
12. The principle of counterflow cooling of hot-gas ducts with cool helium becomes less effective at a higher inlet temperature.
13. At an inlet temperature of more than $\sim$ 350 ^{o}C, cast steel has to be used for the thermal shield instead of cast iron.

After weighing all of these considerations, the most appropriate reactor inlet temperature seems to be 250 ^{o}C. For hydro-gasification of lignite, this temperature is even necessary to make the required energy available for the drying of the raw coal. Also, for all the other process heat methods discussed, considerable amounts of energy from the lower temperature region of helium is required for steam and gas purification, so that the superfluous production of electrical energy is not too high. However, for applications of the chemical heat pipe (long-distance energy) an inlet temperature of 450 °C should be aimed at in order to make the chemical heat pipe(long-distance energy)portion as high as possible (from 60 % at T_i = 250 °C to 75 % at T_i = 450 °C).

3.6.3 Reactor Outlet Temperature T_o

A raising of the reactor outlet temperature above 950 °C has the following consequences:

1. According to the present status of technology, fuel element design is possible at gas temperatures of 1100 °C without exceeding the technical limits for coated particles of 1250 °C. The present fuel element surface specification of 1050 °C can be exceeded if the entire reactor heat is transferred through an intermediate heat exchanger IHX (helium). This is suitable for the coupling of the reactor with steam gasification of coal. All processes which require a steam reformer, can be operated with sufficient conversion at a reactor outlet temperature of 950 °C (without an IHX). Even steam gasification of hard coal can be accomplished with 950 °C helium (including an IHX).
2. At otherwise constant temperatures in the primary coolant circuit, the process heat portion of the reactor power increases with increasing outlet temperature T_o. This is especially important for the steam gasification of hard coal.
3. The heat fluxes in the steam reformer increases with increasing T_o without simultaneous higher pressure drops on the helium
4. The release of solid fission products (for example, Cs 137) increases above 950 °C by a factor of 15 by raising the temperature to 1050 °C. Similar variations are expected for the remaining solid fission products (I, Ba, Sr, Ag, Te) as the temperature is raised. Above 1000 °C, silicon carbide particle coatings (TRISO) must probably be used.

5. If the raising of reactor outlet temperature results in a higher process temperature in the steam reformer, then the wall temperatures in the reformer tubes would rise. This would result in decreased strength values as well as increased permeation rates of hydrogen and tritium.

6. The hot-gas ducting between the reactor core and the heat exchangers becomes an increasing problem with increased outlet temperature. Probably only ceramic ducts can be applied above 1000 ^oC.

7. Coolant mass flow in the primary circuit decreases with decreasing T_o, therefore the gas circulator power increases

8. Without an intermediate heat exchanger, and with increasing temperature, one must contend with increased leakages of H_2O and H_2 and CO_2 in core installations and fuel elements. The gas purification of the primary circuit becomes increasingly less effective as the corrosion rate increases by a factor of 10 per 100 ^oC.

9. The design of the hot core bottom presents no problem even at higher values of T_o from the point of view of radiation damage by fast neutrons, since with the OTTO-cycle the dose at the reactor bottom is sufficiently low.

A consideration of all these points shows: a reactor outlet temperature of 950 ^oC is sufficient for application of steam reformers.
No additional advantages result from the increase of temperature. For the steam gasification of hard coal,temperatures of $T_o \geqslant$ 950 ^oC are desirable when an intermediate heat exchanger is used, with a tendency towards higher temperatures.

3.6.4 Coolant Pressure

A variation in the nuclear reactor pressure mainly affects the reactor pressure vessel, the heat exchangers, the helium circulators as well as the helium circuits and the containment.

In detail, the following comments can be made:

- The ratio of circulator power to reactor power decreases with increased pressure.
- The heat transfer coefficient on the helium side varies with $\sim p^{0.7}$. Therefore the necessary heating surface becomes less with higher pressure.
- At a constant mass flow, an increase in pressure either decreases the velocities or decreases the required cross-sectional areas for flow. It therefore decreases either the pressure drop or the plant costs because the components become smaller.
- The wall thickness of all the primary gas-containing vessels increase with increasing pressure.
- After an accidental loss of coolant, the size of the containment must be larger for increasing system pressure, or the containment must be designed for a higher resulting pressure.
- When a steam reformer is used in the primary circuit, the pressure difference between the helium circuit and the process gas should be as small as possible, in order to obtain a reliable mechanical design of the reformer tubes. In this system, the nuclear reactor pressure should be between 30 and 40 b.
- The present state of development for prestressed concrete vessels is in the pressure range of 40 to 50 b.
- Components such as the gas purification system, fuel loading system, gas circuits, control rods, and circulators, must be constructed to withstand additional system pressure, if higher pressure is chosen.
- The after-heat removal with a separate auxiliary loop becomes more difficult at higher system pressure in the event of total loss of coolant.

- By varying the nuclear reactor pressure from 40 to 20 b or to 60 b would have the following effect upon economy, according to presently available data:

By decreasing the coolant pressure from 40 b to 20 b, the costs of nuclear heat increase by about 10 %; by increasing the pressure from 40 b to 60b,the costs remain nearly constant. There is a marked tendency, therefore, that for two-circuit plants a considerable decrease of the heat costs cannot be attained by increasing the reactor pressure. However, a decrease in pressure is connected with considerable economical disadavantages. According to plant technology conceivable today, an optimum from the economical point of view of the HTR pressure lies in the region between 40 and 50 b.

For all designs, the operational pressure was therefore taken to be 40 b. The main argument should be to have the same pressure on the process- and on the helium side of all process heat exchangers, to make the mechanical design of these components feasible.

3.6.5 Ratio of Core Height/Core Diameter

The determination of the ratio of core height/core diameter in the pebble-bed reactor is important for the fuel ball flow, the core pressure drop, the control rod penetration, as well as the dose stress of the reactor internals.

- By increasing the h/d-ratio the core pressure drop is increased very much.
- The penetration of the control rods into the core for cold shut-down is easier for low-height cores ($h \leq 5$ m), as the penetration depth then does not exceed 2.5 ... 3 m.
- Low-height cores have a higher fast neutron dose in the side reflectors in the critical temperature range of 500 - 600 ^{o}C. Presumably this would require an earlier replacement of the reflector.
- According to the measured curves of the reactor core for 3, 4 and 6 fuel ball disloading exits, the velocity of the fuel balls in the radial direction is the same for a h/d ratio between 0.5 and 0.6 at least in the upper half of the core, which is of great importance for the power production. The dissymmetries in the lower half of the core lead to undesirable deviations in the burn-up (s. fig. 3.6.5-1).

Considering these features, a suitable compromise for the choice of the core height seems to be 5.5 ... 6 m. For a mean core power density of 5 MW/m^3, the following data result for a 3000 MW(th) plant: h = 6 m, d = 11.3 m, h/d = 0.53.

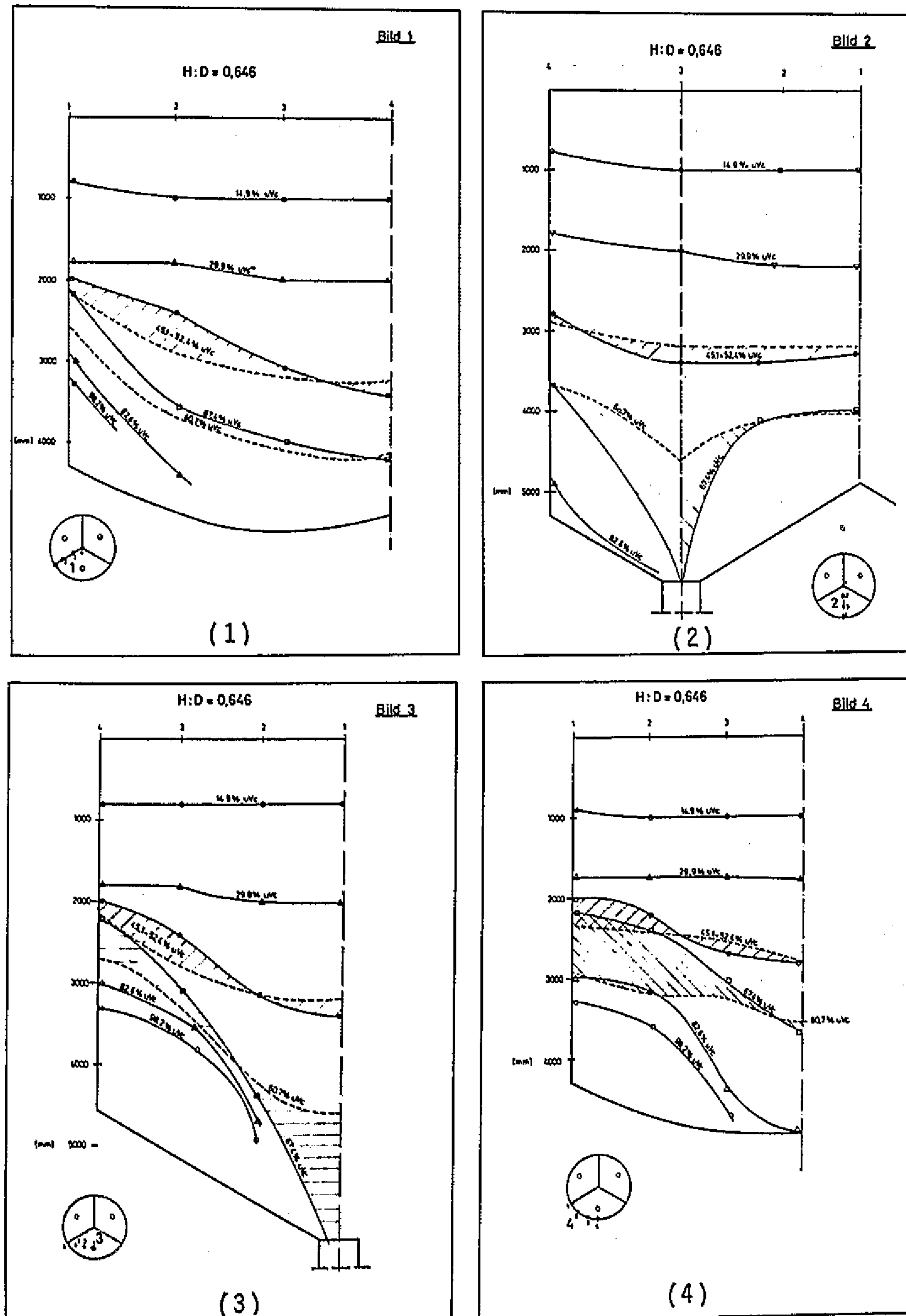


Fig. 3.6.5-1: Model Tests on the Flow Behaviour of Fuel Elements in a 1500 MW Reactor with 3 Disloading Exits [23] (Model 1:20)

(1) Section 1
(2) Section 2
(3) Section 3
(4) Section 4

3.6.6 Burn-Up and Conversion Factor

The price trend on the uranium market in the past years and the expectations of the continually increasing application of nuclear reactors force the design of fuel cycles of all reactor systems to higher conversion factors and so to a smaller supply of fresh uranium. As extensive calculations have shown [24], the pebble-bed reactor can be operated with a conversion factor of C ∿ 0.95 without any big further development, if the following limiting conditions are adhered to:

- There has to be a small parasitic absorption. For continual loading i.e. when there is no surplus reactivity to be kept low, the conditions are most favourable.
- The neutron leakage from the core has to be small. This is guaranteed for the large core dimensions discussed here and the complete embedding of the core in graphite.
- The parasitic absorption of neutrons in the nuclides Np and Pa of the conversion chains ($U^{238} \rightarrow U^{239} \rightarrow Np^{239} \rightarrow Pu^{239}$, $Th^{232} \rightarrow Th^{233} \rightarrow Pa^{233} \rightarrow U^{233}$) have to be kept as low as possible. This requirement is fulfilled by the low power density of 5 MW/m^3 already chosen due to other reasons.
- The absorption rate of the neutrons in the fertile materials should be high. This is made possible by the presence of much heavy metal in the core, i.e. in the fuel elements the heavy metal content has to be increased (from 12 g to more than 30 g).
- As a further measure, which would possibly even make the operation of the reactor as breeder possible, however with a very high doubling time (C ∿ 1.01), BeO comes into question for application.
- Calculations[25, 26] and experiments [27] show that by an addition of a certain amount of BeO balls in the core, about 2 % to 3 % more neutrons are available through the (n, 2n)-reaction of beryllium.
- When the separation of U^{236} under economical conditions is successful, two additional advantages are attained. The parasitic absorption of U^{236} is eliminated and building up of higher

isotopes, which being extremely long-lived alpha emitters cause difficulties during the end-storage, is lessened.

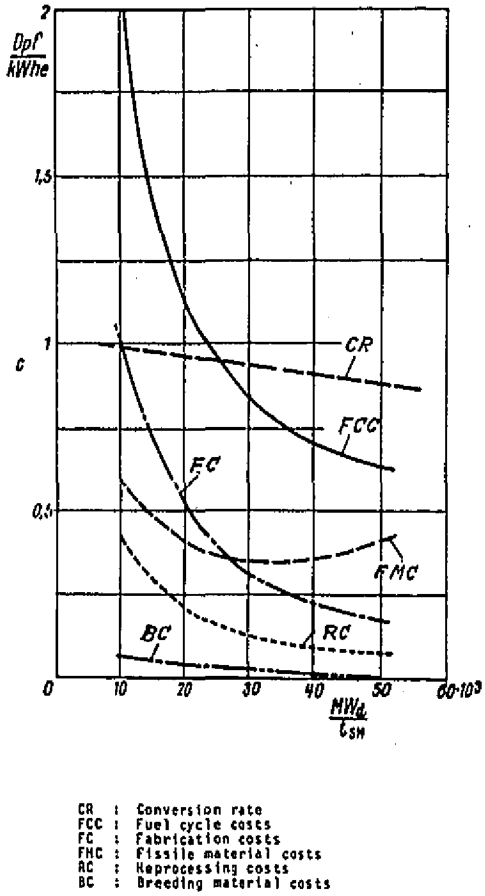
The following table shows a comparison of the design up to now of fuel elements, which due to the still high fabrication costs of the fuel elements (small numbers) aim at a high burn-up, as well as data for the high converter and near-breeder reactors.

Data	Dim.	Normal Converter	High Converter	Near Breeder	Breeder
Heavy metal content of the balls	g	12	30	∿30	∿40
Burn-up	$\frac{MWd}{t}$	100000	30000	10000	10000
Ball material		graphite	graphite	graphite	graphite/ BeO ∿ 1/1
Conversion factor		C∿0.6	0.94	0.98	1.01
Power density	$\frac{MW}{m^3}$	5	5	5	5...2.5

Table 3.6.6-1 Data of Different Fuel Cycles

However, the fuel cycle costs increases largely as fig. 3.6.6-2 shows, when the burn-up is reduced from approx. 30000 MWd/t for which the conversion factor is already 0.94 to 10000 MWd/t. This is due to the increased expenditure for reprocessing and re-fabrication. Besides, a large amount of fissile material is always found in the outer part of the fuel cycle.

However, the increase of the conversion factor is in all cases of such interest that the corresponding fuel element development has to be carried out. Here a possibility of making a thermal, inherently safe breeder system can be seen.
In any case, in the next twenty years, reactor systems which use the uranium and thorium reserves much more effectively than the systems known today must be introduced.



Annual load factor	0.8
Efficiency	40 %
Deadline for payments:	
Fissile material	360 d
Fabrication	360 d
Reprocessing after unloading	360 d
Credit note for fissile material after unloading	360 d
Interest rate	8 %
Tax on fissile material used	2 %
Dissipation factor during reprocessing	0.99
Diminishing value factor for fissile material during unloading	0.9
Diminishing value factor for thorium during unloading	0
Reprocessing costs	400 DM/kg HM
Fabrication costs for HM	400 DM/kg HM
Fabrication costs per ball	20 DM
U-233	45000 DM/kg
U-235	40000 DM/kg
Th-232	50 DM/kg

Fig. 3.6.6-2 Dependence of Fuel Cycle Costs and the Conversion Rate on the Burn-Up [24]

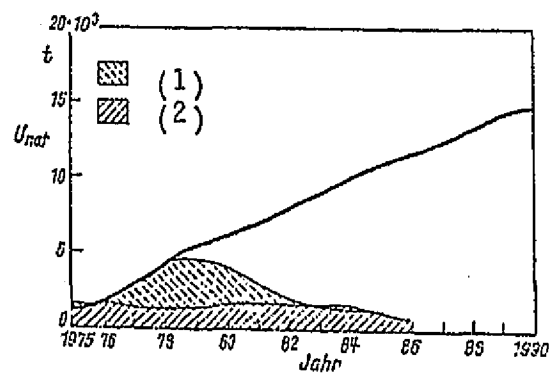


Fig. 3.6.6-3 Demand and Supply of Natural Uranium for the FRG [38]

Amounts ensured by contracts:
(1) Contracts with EVU
(2) Acceptance obligation of uranium companies

This is illustrated by fig. 3.6.6-3 which shows the expectations of uranium supply of the FRG for the near future.

3.7 Choice of Vessel Design (Loop/Pod-Boiler System, Concrete or Cast Steel or Cast Iron)

3.7.1 Technical Status of the Prestressed Concrete Vessel

The present development of gas-cooled reactors with prestressed concrete vessels shows a variety of principles of arrangement, as shown in Figure 3.7.1-1. While all the light and heavy water reactors constructed up to now favour a so-called loop design (many parallel heat exchanger circuits connected with the reactor pressure vessel by ducts, entire primary cycle placed in a reactor containment building), in AGR- and HTR-reactors the aim nearly always was to place the entire nuclear part in a reactor vessel, however, up to now, always without a containment.

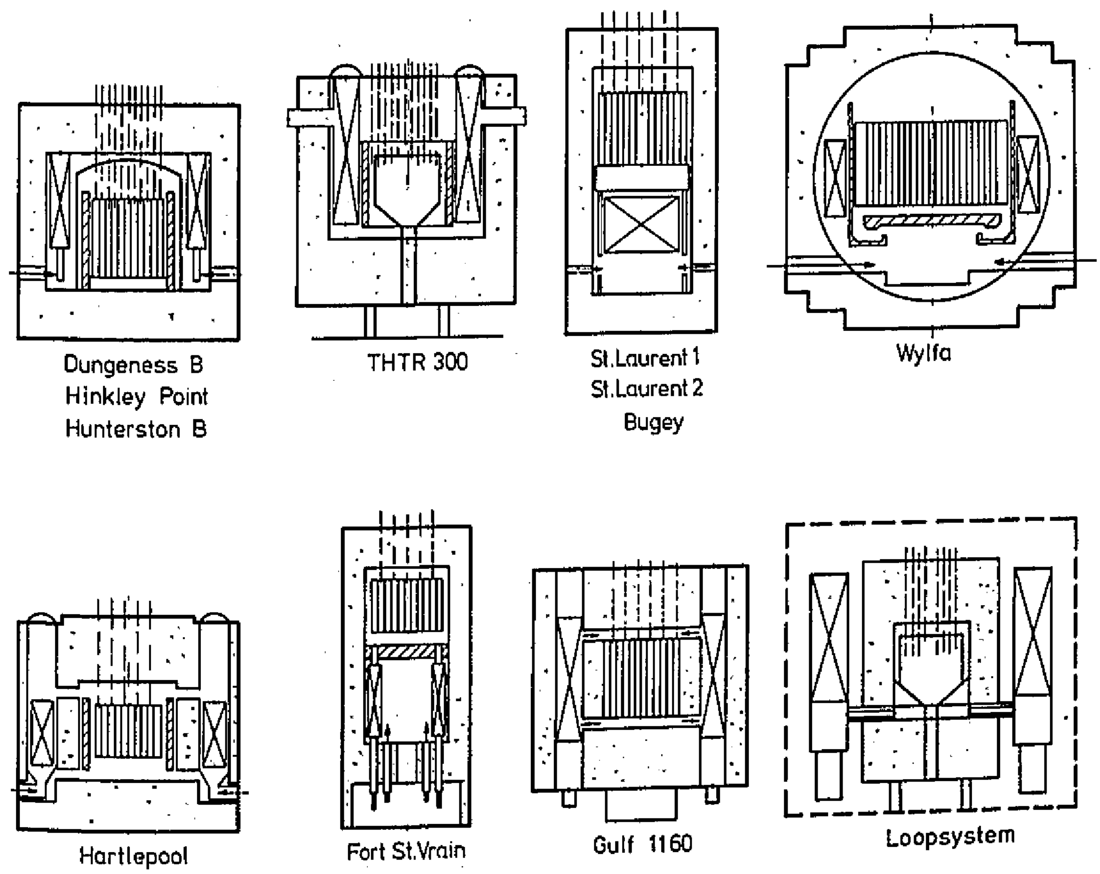


Fig. 3.7.1-1: Designs of Prestresssed Concrete Vessels (already constructed as well as planned plants)

Table 3.7.1-2: Prestressed Concrete Vessel

Plant	Marcoule	EDF 3	EDF 4	Oldbury	Wylfa	Dungeness B	Buggy	Hinkley Pt B	Fort St.Vrain	THTR
Country	France	France	France	U.K.	U.K.	U.K.	France	U.K.	U.S.	Germany
Power MWe	30	480	480 - 515	280	590	600	560	625	330	300
Gas pressure (at)	16.2	31	31 - 33.5	27	27	33.8	43.7	42.2	46.5	40
Vessel form	horizont. Cylinder		vertical Cylinder		sphere			vertical cylinder		
Inner vessel size:										
diameter	14	18.9	18.9	23.4	29.2	20	17	18.9	9.4	15.9
length	15.5	20	36.2	18.2	29.2	17.6	38	19.3	22.8	15.3
wall thickness	3.04	4/5.5	4.5/6.4	4.5/6.7	3.35	3.8/6.2	5.5/7.5	-	2.7/4.5	4.45/5.1
Temperature on the hot side of the insulation	49	410	230-240	245	250	290-675	230		400	260
Coolant gas	CO_2	CO_2	CO_2	CO_2	CO_2	CO_2	CO_2	CO_2	helium	helium

Up to now arrangement in the reactor containment was not necessary in this type of primary circuit layout, as the prestressed concrete vessels are considered to be burst protected and all vessel closures or terminations are doubly protected. Table 3.7.1-2 shows a review of the essential technical data of some of the prestressed concrete vessels, either already built or under construction. In contrast to steel vessels, which are used for water reactors and permit a diameter of only about 5 ... 6 m, prestressed concrete vessels can be constructed with very large free volumes.

3.7.2 Pod-Boiler and Loop Arrangements in Prestressed Concrete

For the application of nuclear process heat, 3 different principles are possible for the arrangement of the nuclear reactor and the heat exchangers (see fig. 3.7.2-1).

- Arrangement of the reactor core and heat exchangers in a common large cavity, similar to that in the THTR
- Arrangement of the core inside a prestressed vessel, location of the heat exchangers in so-called loop vessels outside the reactor vessel made either of boiler steel or made in a so-called burst protected way.

- Arrangement of the core in a central cavity in the vessel, location of the heat exchangers in separate small cavities situated around the core in the wall of the vessel (pod-boiler)

As fig. 3.7.2-1 shows, the first solution can be discarded due to the large dimensions, when prestressed concrete is used as vessel material. The conditions for cast iron or cast steel will be discussed in the following. The remaining solutions for concrete: loop arrangement and pod-boiler, have to be evaluated according to different properties regarding their suitability for process heat plants as is shown in table 3.7.2-2.

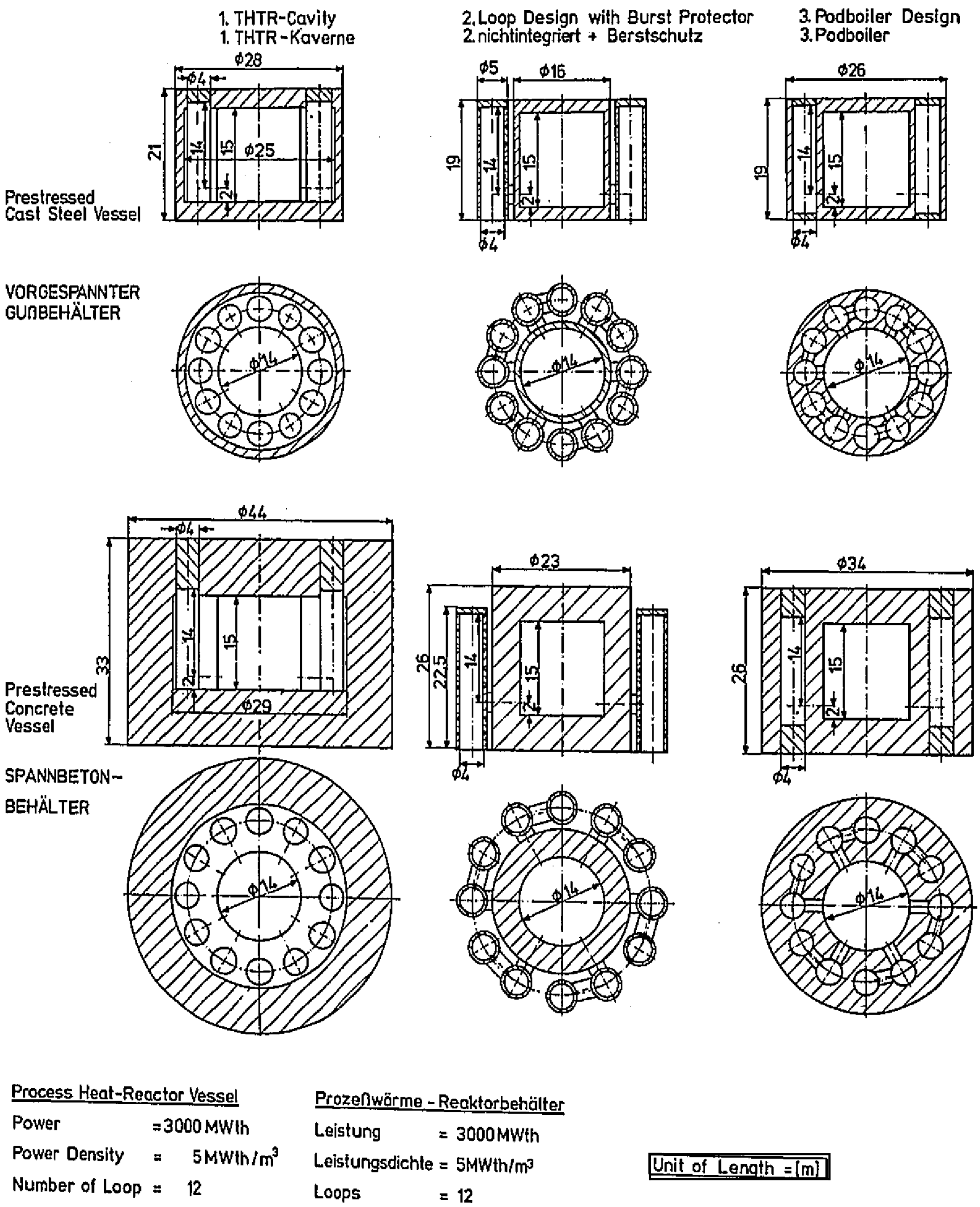


Fig. 3.7.2-1: Comparison of the Size of the Vessels for Different Principles of Arrangement and for Prestressed Concrete and Cast Steel

As table 3.7.2-2 shows, all these features are in favour of a choice of the loop system, if the problem of a fast depressurization of the primary system due to damages in the coaxial ducts or in the heat exchangers can be tolerated.

Aspect	Pod Boiler	Loop System
1) States of development	- some AGR reactors have been built following this principle - GAC intends to use this principle for their large HTGR-plants - for steam reforming applications 12 big cavities for the heat exchangers are used instead of 6 for steam generation plants; therefore larger vessels have to be developed	- vessel for a 3000 MWth-plant using the loop principle corresponds in dimensions exactly to the THTR-vessel - loop concept has been used until now only for small HTR-plants (Dragon, Peach-Bottom) and for small AGR (Windscale) - all light water reactors are built using the loop concept
2) Flexibility	- for the various applications of process heat, different vessels are always needed (different processes or different splitting of nuclear heat for steam reforming and for steam generation)	- one standard vessel can be connected with various different process heat exchangers
3) Dimension of vessel	- for a 3000 MWth-plant, a vessel with a volume of 30 000 m^3 is needed	- for a 3000 MWth-plant a vessel with 10 000 m^3 is needed
4) Extrapolation to larger units	- estimation for a 6000 MWth-unit: vessel with a diameter of 50 m	- vessel for 6000 MW has a diameter of 37 m, number of loops could be doubled
5) Components with big risk involved	- closures for the heat exchangers with 4...5 m diam. are new components	- coaxial gas ducts to the heat exchangers (1 m ∅) require compensators or complicated cooling systems
6) Depressurization of the primary system	- if double closures are used, a rapid loss of coolant to the containment is avoided (< 1 b/sec)	- if the number of loops is large enough (12) and with a special design of the inner gas ducting system in the reactor, the maximum pressure transient can be limited to 10 b/sec without any burst protection. This does not cause failures of structures which are needed for after-heat removal
7) Safety against earth quakes	- pod boiler system is inherently safe for this accident	- safety against earth quakes has to be attained by engineered design
8) Inspection and repairs	- all ducts and pods for heat exchangers and core cavity are insulated. Very difficult to inspect something	- all walls of the loops are made of normal vessel steel and are not insulated. Therefore they can be inspected and repaired
9) Height of the containment	- heat exchangers must be removed to the top of the vessel, therefore there has to be more space above the vessel	- heat exchangers can be removed to the side and be taken out by a central opening in the containment. Therefore height of the containment is smaller and underground arrangement becomes more attractive
10) Prefabrication of heat exchangers	- pods must be constructed at the site of the plant	- heat exchangers together with vessels can be prefabricated in a factory and be transported to the site
11) Construction time of vessel	- erection time is of the order of 4 years	- erection time of vessel possibly shorter by 1 year
12) Costs of the vessel	- in the order of $220 \cdot 10^6$ DM	- in the order of 80 Mio DM for the reactor vessel - additional 50 Mio DM for the heat exchanger vessels

Table 3.7.2-2 Aspects regarding Integrated or Non-Integrated HTR-Systems

This problem is important because after such an accident it has to be possible for the reactor system to be after-cooled i.e. damages must be prevented to the core internals or the remaining loops, which serve for after-heat removal.

Although the probability of a coaxial duct rupture is very small according to the present status of knowledge, the consequences of a sudden rupture or the bursting or a vessel have to be discussed.

According to recent calculations [28], the forces connected with fast depressurization of the primary circuit working on all core internals in the gas flow are sufficiently low so as to prevent damages. By the additional insertion of an inner flow limiting duct in the cold-gas duct, the emanation of the coolant during damages to the ducting is largely delayed.

The following rough estimation gives a picture of the pressure gradient after a rupture of a co-axial duct.

$$\dot{m}_1 = F \cdot \Psi \cdot p \cdot \sqrt{\frac{2}{RT_o}}$$

$$\dot{m}_1 + \dot{m}_2 = 0$$

$$pv = m_2 \; RT$$

$$\Psi = \left(\frac{p}{p_o}\right)^{\frac{1}{k}} \cdot \sqrt{\frac{k}{k-1} \left\{ 1 - \left(\frac{p}{p_o}\right)^{\frac{k-1}{k}} \right\}}$$

$$\Psi_{max} = \left(\frac{2}{k+1}\right)^{\frac{1}{k-1}} \sqrt{\frac{k}{k+1}}$$

$$- \frac{dp}{dt} \cdot \frac{v}{RT} = F \cdot \Psi p \cdot \sqrt{\frac{2}{RT_o}}$$

$$p = p_o \cdot e^{-F} \cdot C \cdot t$$

$$C = \Psi_{max} \cdot \sqrt{2RT_o} \cdot \frac{1}{v}, \quad \Psi_{max} = 0.5$$

T_o ~ mean temperature of helium

v = reactor volume

p_o = initial system pressure

k = 1.33 f. helium

An evaluation of these formulae for the different reactor vessel types shows the following results (see fig. 3.7.2-3).

In contrast to the integrated designs THTR and GAC 1160, high pressure gradients occur at the beginning of the loss of coolant accident for loop systems without burst protection. It has to be shown, therefore, that a pressure change of ∿ 10 b/sec can cause no damages to the core internals, insulations or other remaining loops, which have to be used for the after-cooling capacity of the core. The depressurization tests on metal foils and fibre insulations carried out up to now [22] show that damages can be avoided by appropriately constructing these components. Use of carbon stone for the hot-gas ducting would considerably lessen the problem. These core internals cause no problems when they are suitably designed according to the calculations carried out [28]. In any case, it is aimed at decreasing the free outlet area when the duct ruptures so that the pressure gradients are further decreased. A further step possibly in this direction could be the introduction of flow limiters in the hot-and cold-gas ducts. Such a measure, however, causes high pressure loss and the economy of the system is impaired.

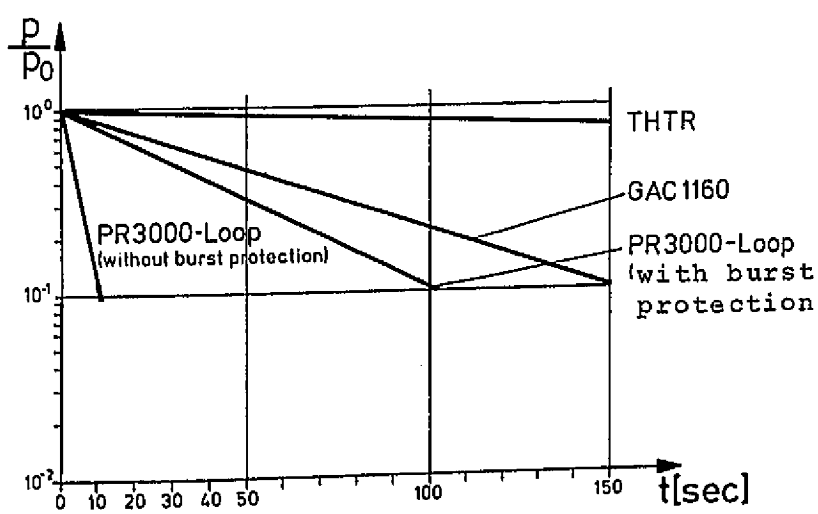


Fig. 3.7.2-3: Change of Pressure for the Emanation of Coolant from the Different Reactor Vessels

An additional possibility, which has been proposed already for light water reactors, is to surround the loop with a so-called burst protection, which would limit the flow rate during accidents so that pressure change rate could be retained to a few bar/sec. These burst protection measures could consist of rings either of cast iron or concrete or cast steel, which are surrounded by axial and radial bracing cables. The problem thereby is to keep the free volume between the vessel wall and burst protection as small as possible to minimize the free volume. At the same time there must be the possibility of removing the structure between the vessel wall and the burst protection to be able to carry out inspections.

The following table 3.7.2-3 shows a comparison of data of the known integrated reactor concepts and the corresponding data for loop systems in case of a depressurization of the system.

Reactor / Parameter	THTR	GAC 1160	PR 3000-Loop with Burst Protection	PR 3000-Loop without Burst Protection
Operation pressure	40 b	48 b	40 b	40 b
Containment pressure	1 b	4 b	3 b	3 b
Maximum free area	33 cm^2	645 cm^2	1000 cm^2	15000 cm^2
Reason for accident	Rupture of a duct NW 65	Damage to free area NW 290	Rupture of a loop vessel or a coaxial duct	Rupture of a loop vessel or a coaxial duct
Time required for pressure equalization with containment	∿ 2 h	∿230 sec	∿150 sec	∿10 sec
Maximum pressure transient	0.045b/sec	0.45b/sec	0.6b/sec	∿15b/sec

Table 3.7.2-3: Conditions for Total Loss of Coolant for the Different Reactor Concepts

As is explained already, an important stipulation for whether loop systems are constructable from technical safety aspects is that the pressure gradients do not affect the after-cooling capacity of the core. Other technical questions, such as the behaviour of the system in case of earthquakes, have to be considered carefully.

As all other arguments are in favour of the loop system this principle should find application for nuclear process heat plants, if all safety requirements can be fulfilled.

3.7.3 Use of Cast Iron or Cast Steel as Vessel Material

Prestressed concrete as vessel material can be replaced by cast iron or cast steel. When cast iron is used, it is arranged in the form of blocks around the liner and the penetration tubes. The vessel is stressed on the outside by radial and axial cables such that there are only pressure stresses in all parts of the vessel.

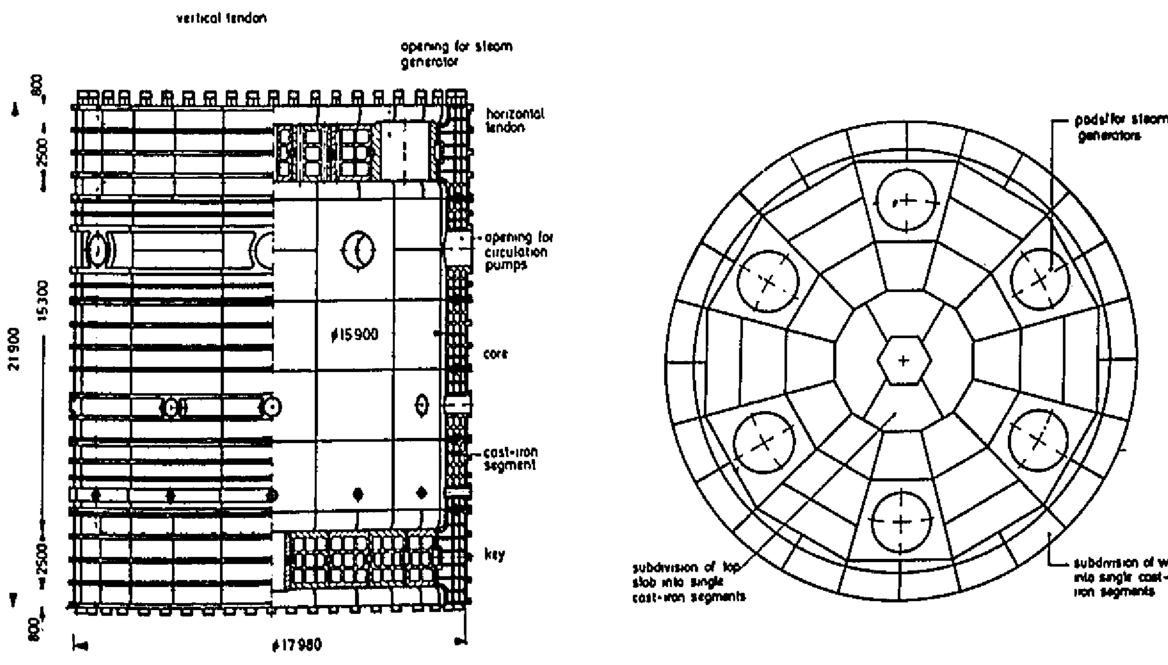


Fig. 3.7.3-1: Design of a Prestressed Cast Iron Vessel [29]

The separate blocks are connected by shearing wedges. Special designs are required for the liner attachment to the blocks. The necessary burst protection of the vessel is guaranteed by many parallel cables and successive shearing wedges. In operation, the prestressing is partly compensated by the reactor coolant pressure of 40 b.

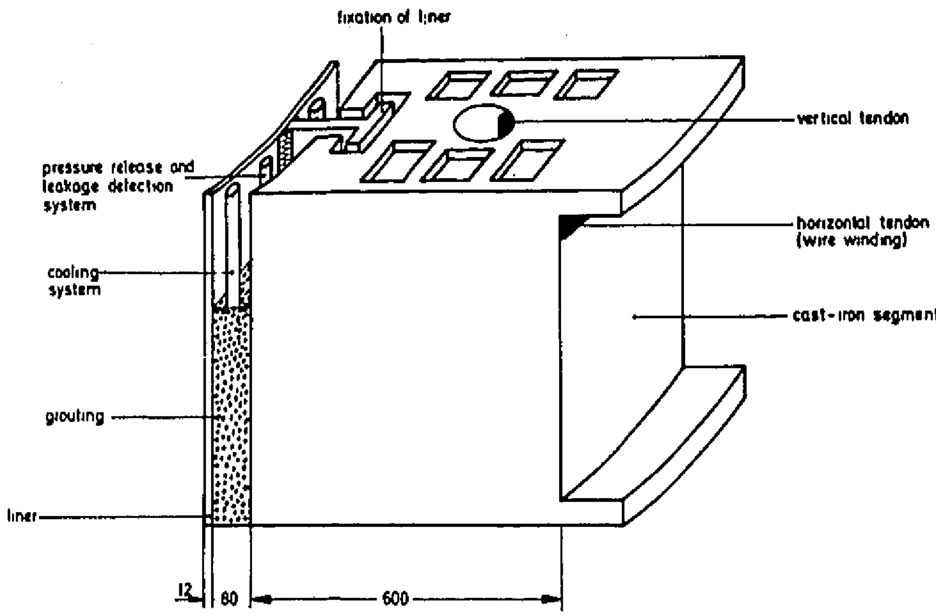


Fig. 3.7.3-2: Attachment of Liner in the Design in Fig. 3.7.2-1 [29]

By use of cast steel as vessel material, there is a possibility of avoiding the liner and the penetration tubes. The steel blocks are connected and sealed by stress-free welding seams between them and the arrangement is thus made helium-tight. These welding seams are on the outside of the vessel and so can be tested and repaired. In this design, cooling of the vessel occurs through water-filled cooling chambers in the blocks or by holes which are drilled inside the blocks.

The replacement of concrete by cast iron or cast steel has a number of big advantages:

- Use of well-known, well specified materials, not sensitive to temperatures up to about 400 °C (concrete in comparison cannot be heated above ∿ 80 °C)
- High strength coefficients
- Good and well-known long-time behaviour, no creep and shrinkage as in concrete
- Definite behaviour for limiting stresses (important in the case of accidents)
- Small thickness of wall of the vessel, saving of weight as com-

pared to concrete

- A smaller size of containment foundation plate and much smaller dimensions of the vessel as compared to concrete (see fig. 3.7.2-1)
- Vessels can be constructed in large dimensions, especially for a ring-shaped reactor core (i.e. vessel with a supporting middle column).
- More flexibility in the arrangement of reactor vessel and loops
- Prefabrication of the individual parts of the vessel in the factory, short final installation seems possible
- Shortening of the construction time of the reactor plant
- Possibility of exchange or repair of individual blocks
- Inspection of the vessel walls possible
- Demounting seems possible after shutting-down of the reactor plant
- Possible saving of costs for the reactor vessel due to prefabrication and shortening of construction time

In the case of the cast steel blocks, the following are the additional advantages over cast iron:

- No need for liner and penetrations, because steel is gas-tight for helium and easy to weld.
- Gas-tight welding seams on the outside of the blocks are easy to inspect and repair.
- Design of the top easier, which allows many penetrations for the control rods and the feed equipment ducts.
- Problem of connection between the liner and the cast iron blocks are avoided.

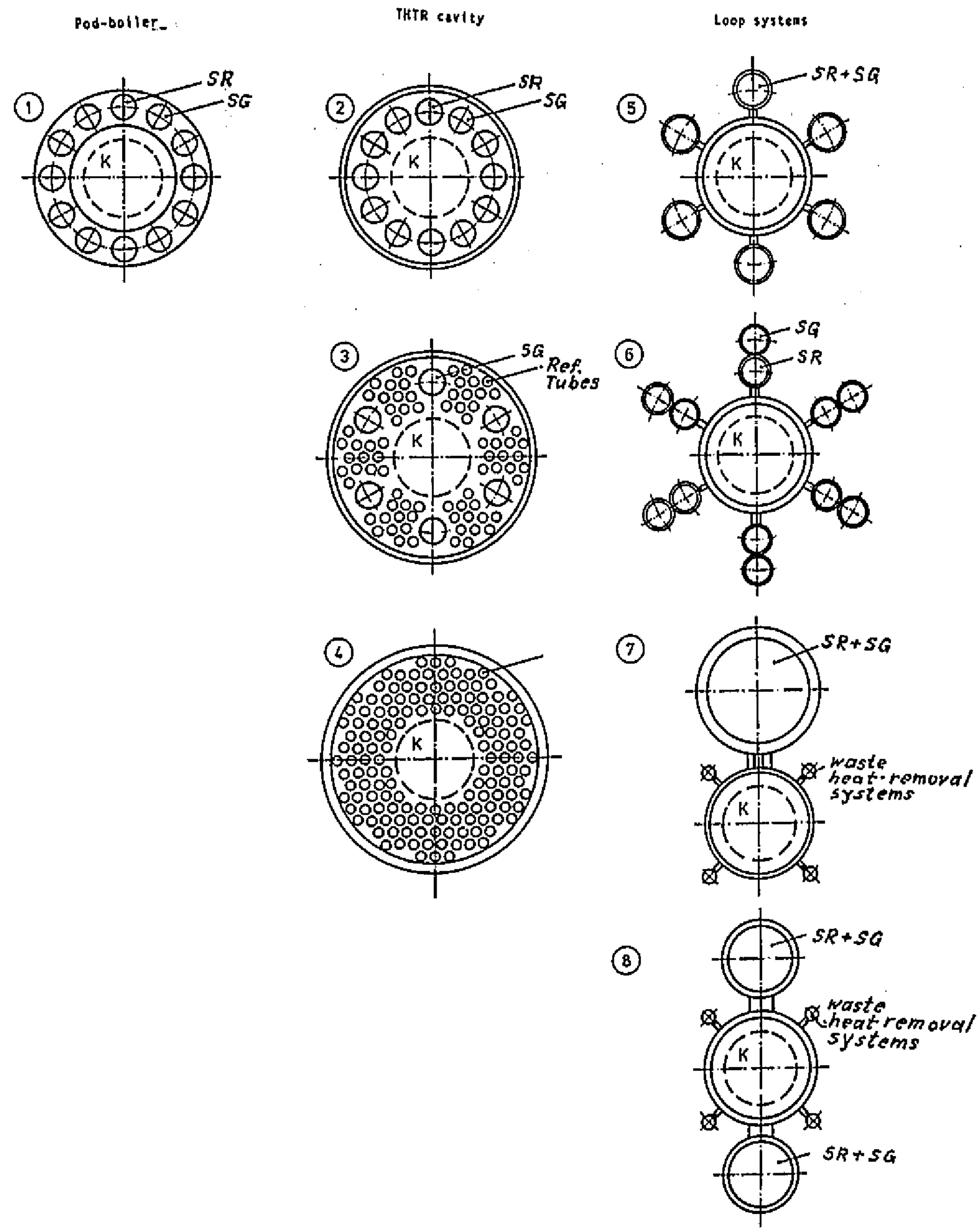
However, the costs of cast steel are considerably more than those for cast iron. Therefore it is necessary to confirm that the higher costs of cast steel are compensated by the saving of costs for the liner and penetration required with cast iron. According to existing information, this seems to be the case.

In table 3.7.3-3 there is a comparison of the most important data of the materials which can be used for cast iron or cast steel vessels. Especially the strength values for thick-walled cast iron blocks are relatively unknown today and have to be confirmed by development programmes.

Comparison of the Properties between Steel, Cast Iron and Cast Steel (the table is an enlargment of the DIN value)

Material Data	Dim.	Temp. °C	St 42.3	GS 45.3	GS 22 Mo 4	GGG 42.3	GG 20	SKMU 84
		DIN	17100	1681	17245	1693	1691	
		Firm.						Siempelkamp
Tensile Strength σ_{zB}	$\frac{kp}{mm^2}$	20 300	42 — 50	45	45 — 60	42 — 50	20	19 15
Yield-Point 1) σ_s 2) $\sigma_{0.2}$ 3) $\sigma_{0.x}$	$\frac{kp}{mm^2}$	20 100 300	26 1)	23 1)	25 1)	28 — 35 2) 27 22	---	18.8 3) 18.5
Compressive Strength σ_{dB}	$\frac{kp}{mm^2}$	20 100 300				76 — 91	60 — 83	100
Shear Strength τ_B	$\frac{kp}{mm^2}$	20 300	30 — 35	30	30 — 45			28
Bending Strength σ_{bB}	$\frac{kp}{mm^2}$	20 100 300					29 43	40
Allowable Surface Pressure p_{zul}	$\frac{kp}{mm^2}$	20 100 300	8 — 15	8 — 10	8 — 15		7 — 8	7 — 8
Impact value 1) DVM DIN 50115 a_K 2) ISO-V	$\frac{kpm}{cm^2}$	20	3.5 2)	min.4 1)	min.5 1)	1.8 — 2.5 1)		
Impact Strength 1) cycl. exam. ∅ 20 2) DIN 50115	$\frac{kpm}{cm^2}$	20 100 300					0.3 1)	0.35 2) 0.5 0.6
Hardness HB 30 DIN 50351	$\frac{kp}{mm^2}$	20	150	150		140 — 180	200	195
Young's Modulus E	$\frac{kp}{mm^2}$	20 300	21000	21000	21000	17500 17000	10500	10500
Elongation δ_5	%	20 100 300	22	22	22	11 — 25 10 — 21 8 — 19		0.5 1.2
Density ρ	kg/dm³	20	7.8	7.8	7.8		7.2	7.2
Coefficient of therm. Expansion α	$\frac{1}{°C}$	20 — 300	11 — 13	11 — 13	12	12.5	10 — 11	10 — 11
Heat Conductivity λ	$\frac{kcal}{m\,h\,°C}$	20 300	36	36	42 37.5		40	40

Table 3.7.3-3: Comparison of the Material Data for Cast Steel and Cast Iron



1) Pod-boiler principle
2) THTR cavity principle (steam generator + reformer tube bundle)
3) " " " (SG + individual reformer tubes)
4) " " " (for use of reformer tubes)
5) Loop system(steam reformer + steam generator both in same loop vessel)
6) " " (SR + SG in separate loop vessel)
7) " " (only an additional vessel for all heat exchangers)
8) " " (only two additional vessels for all heat exchangers)

Fig. 3.7.3-3: Degrees of freedom in the Design of the Reactor Primary Circuit Using Cast Steel

The use of prestressed cast iron or cast steel allows a series of design possibilities as shown in fig. 3.7.3-3. The principle of the THTR cavity does not lead to prohibitively large dimensions here as is the case for concrete.

The variants 3) and 4) seem to be possible alternatives only for cast steel vessels because of the plurality of penetrations in the vessel top.

3.8 Reflector Design

The reflector graphite is damaged in the course of operation by high-energy neutrons. Due to this radiation damage, neutron-induced changes in dimensions occur in the graphite(Wigner shrinkage) and neutron-induced creep occurs. Also the physical properties of the material (e.g. Young's modulus,heat conductivity) change under neutron radiation. A typical course of change in dimensions is shown in fig. 3.8-1 for an isotropic graphite.

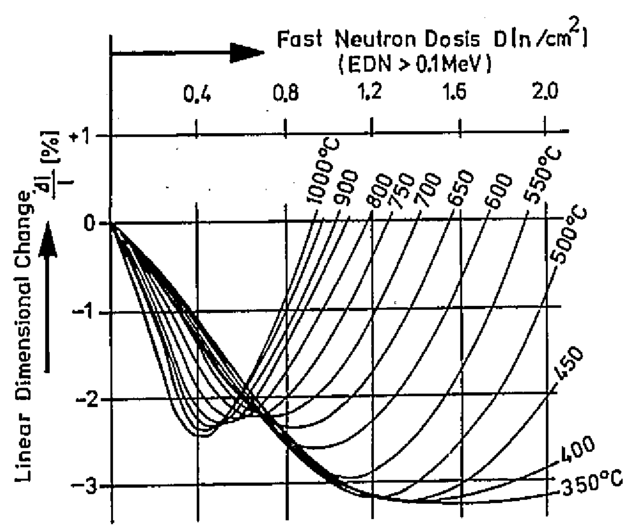


Fig. 3.8-1: Curves for the Dose and Temperature dependent Shrinkage of Isotropic Reflector Graphite [30]

A first approximation for the allowed design range of reflector graphite can be obtained from a curve, in which points of maximum shinkag are connected and a curve which joins the zero passages in fig. 3.8-1. More exact calculations using the method of finite elements are used today to calculate the stresses during normal and transient operation of the reflector blocks.
The fast neutron dose actually occurring during the life-time of the graphite reflector should in a first approximation always stay below the curves in fig. 3.8-2 in order to avoid stresses in the core internals.

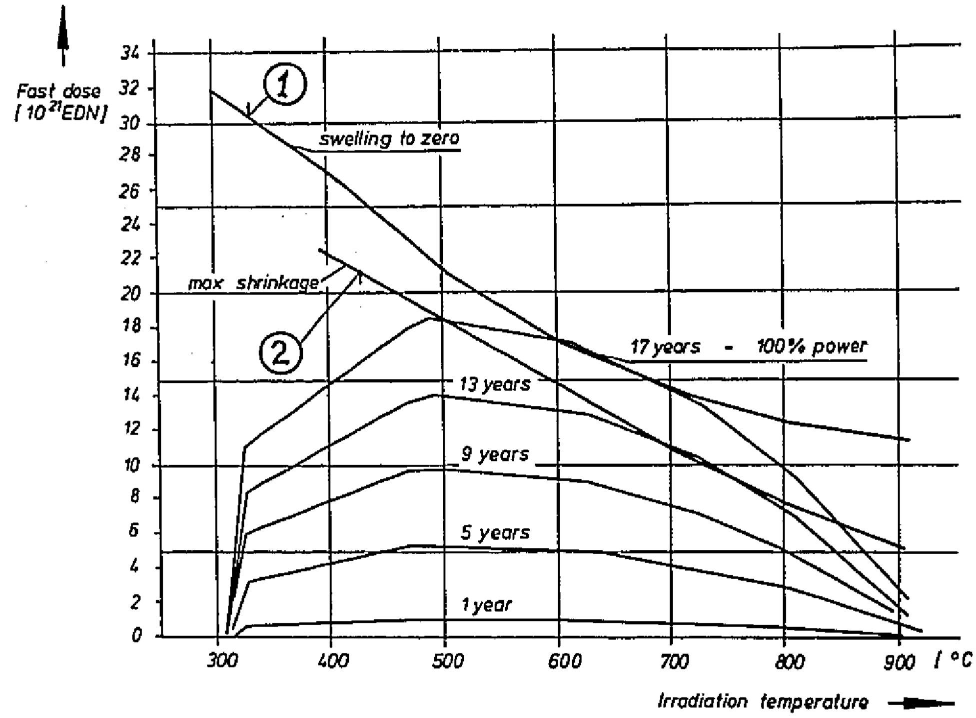


Fig. 3.8-2: Connection between Tolerated Dose and Radiation Temperature

(1) $\Delta L/L = 0$

(2) $\frac{d}{dD}\left(\frac{\Delta L}{L}\right) = 0$

The design of the reflector, which according to fig. 3.8-2 causes difficulties when the reflector is simultaneously subject to high temperature and high fast neutron fluxes (upper part of the side reflector), causes no problem in the hot region of the HTR after introduction of OTTO-loading. This is because the fast dose at the core bottom decreases by a factor of 10 in comparison to the top reflector or the upper part of the side reflector. Fig. 3.8-2 shows the dose distribution on the inner side of the side reflector, whereby places are marked by their temperatures in the axial direction. According to this, parts of the side reflector in the temperature region around 600 ^{o}C are especially endangered (i.e. places about 1 m below the core surface). Without any counter measures this part of the side reflector would have to be changed after about 17 years (load factor 0.9). In order to realize a service-life of 30 years in spite of everything, the following meausres which limit the dose in the side reflector are conceivable and appear to bring success:

- Poisoning of the side reflector and therefore decreasing the flux at the core edge
- Arrangement of a wandering layer of graphite balls in front of the side reflector to reduce the fast flux (ball layer of ∿0.2 ... 0.3 m width)
- Cooling of the upper part of the side reflector through bypass of small amounts of cold gas (∿ 5 % sufficient) and so shifting of the temperature profile to regions of higher allowed fast dose.
- In any case, the extension of the top-and of the side reflector, must be possible, because nobody can really guarantee for the life-time of the reflector.

The top reflector, which has a temperature of about 320 ^{o}C due to the core inlet temperature of 280 ^{o}C raised by γ-heating of the graphite structure, causes no problems for an extrapolation of the existing radiation values (up to 370 ^{o}C). If these values cannot be confirmed for low temperatures, an increasing of the core inlet temperature to 300 ... 320 ^{o}C is recommended.

For application to the reactor system discussed here, the following holds good: Through a small bypass of coolant gas through the reflector surface, which in any case could not be completely avoided, the temperatures are so lowered that not undesirable dose values are obtained in the critical temperature region. In spite of this, concepts were worked out for exchanging the upper parts of the side reflector as well as the top reflector. The same consideration, however, must be made for the total graphite structure of the core.

The following limiting conditions and aspects were thereby considered:

- The reflector should be designed at first for the reactor lifetime (∿ 27 full-load years) based on the radiation data and the expected stress relief caused by the neutron-induced creep.
- The possibility of changing of direct core structures can be looked upon as a back-up solution 1) in case of unexpected material behaviour 2) for controlling any hypothetical accident effects 3) to enable an adaptation to the varying different ope-

ration cycles of the pebble-bed core 4) for carrying through of reactor operation possible in principle for HTR, maybe even for longer than the planned power-producing operation life, by substitution of the corresponding construction parts which have a limited life-time 5) to facilitate the necessary removal after the final shut-down.

- A periodical, regular access (like for GAC Design: Change of reflector after 8 years, annual change of a quarter of the fuel elements), therefore, does not have to be planned in the operation scheme. Therefore, such possible measures do not have the same economic aspects as a before-specified operation does, but they should be undertaken realistically with the smallest possible danger to the personnel.

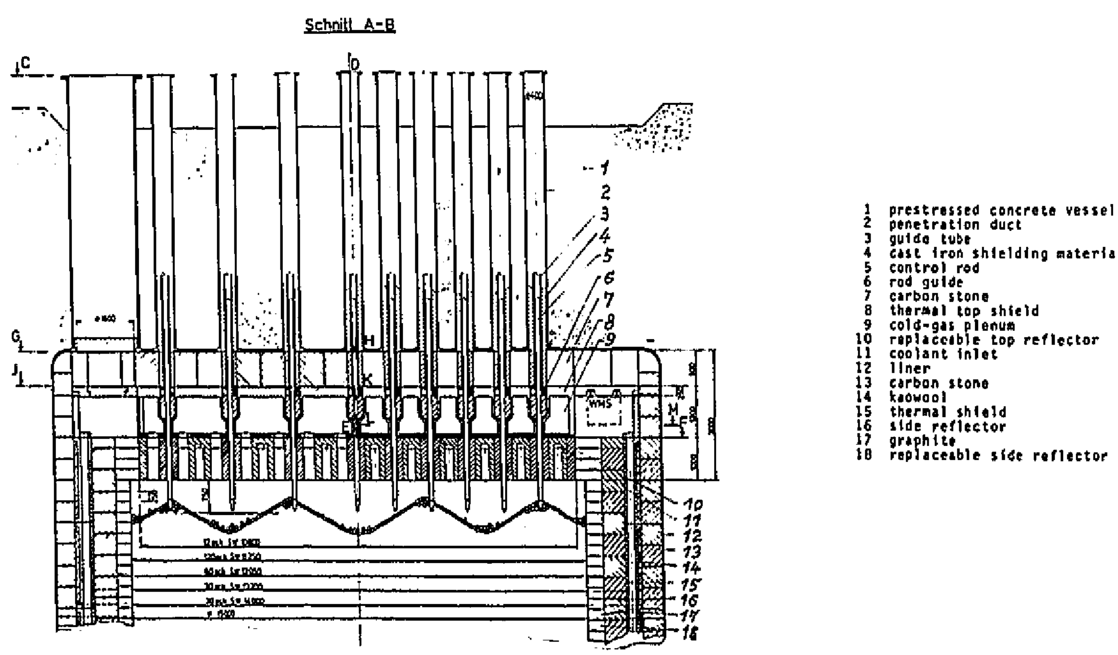


Fig. 3.8-3 Design of the Reactor Top in which Exchange Procedures can take place [31]

The following succession of structures surrounding the core is characteristic for the core design: graphite reflector, thermal shield and carbon stone insulation in blocks up to the liner of the pressure vessel. As shown in fig. 3.8-3 and fig. 3.8-4, these layers can be removed through the top of the vessel.

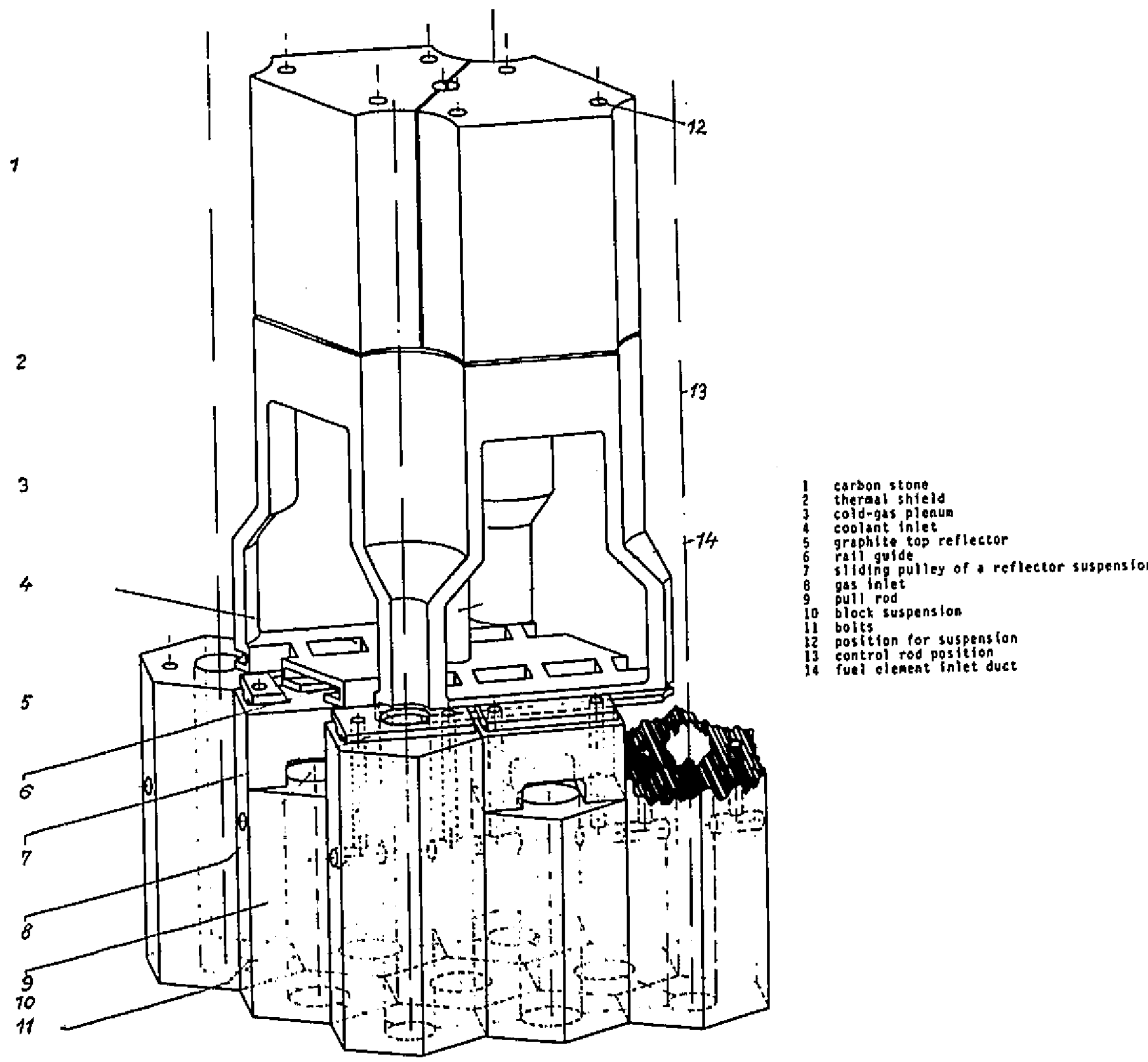


Fig. 3.8-4 Typical Design of Top of a HTR with Ball-Shaped Fuel Elements

3.9 Considerations on Safety and Accidents in the Nuclear Plant

3.9.1 Shut-Down of the Reactor

For the fast shut-down of the reactor, the control rods are used as the first shut-down system. These are inserted in about to a depth of 0.5 m into the pebble-bed and thus guarantee a sure shut-down.

The only compulsion for scram is the in-rush of water during a steam generator tube rupture. The fuel element temperatures do not rise considerably hereby.

The second shut-down system consists either in the use of boron trifluoride, which is put into the core after the failure of all the control rods or of small absorbing balls which are thrown into the core [32]. These small balls wander from the more important upper core half from the neutron physical point of view into the lower burnt-up core zone after the ending of the accident. They are removed by taking out small amounts of fuel elements from the core.
A further very effective shut-down system is the strongly negative temperature coefficient of the reactor.

3.9.2 After-Heat Removal

The normal condition for the removal of after-heat is a helium pressure of 40 b in the primary circuit. In this case, one of the production loops is sufficient in order to cool the core. Using this principle of after-heat removal,designs with as many loops as possible are most suitable as it is naturally of interest to have sufficiently large redundancy. Otherwise 3 or 4 auxiliary loops could be used parallel or additionally. The energy supply of the circulators has to be ascertained by auxiliary diesel engines, the feed water supply of the heat sinks (steam generator of the production loops or separate coolers) have to be guaranteed by a redundant design.

If a total loss of coolant from the primary circuit occurs caused by rupture of larger or smaller helium ducts, e.g. in auxiliary systems also,so there is a resulting mixture pressure of about 4 b in the containment. In this case one loop is still able to sufficiently cool the core. However, it has to be guaranteed by the design form of the primary circuit that during an accident the other remaining loops have not been damaged by pressure waves (s. § 3.7).

If the containment is simultaneously damaged, a pressure of 1 b results for the after-heat removal. There is no chance of this accident occurring according to the present safety measures, but should be however discussed as it is of interest to estimate the residual risk of nuclear plants. In such a case 7 loops would be sufficient to cool the core under the assumption of a plant with 12 production loops.

For the still more improbable case that there is no active after-heat removal system, which can be operated,which can have catastrophic results for all known reactor systems, there is the possibility of a suitable design preventing core-evaporization. The experiment carried out in the AVR of a breakdown of coolant circulation [33] and calculations for small plants [34]show that even in this extreme case the temperatures of the fuel remain uncritical. In the AVR a temperature constancy should be attained by natural convection. Calculations show that for reactors with 5 MW/m^3 power density and dimensions of about 4 m diameter, the maximum fuel temperatures remain below 2000 ^oC for a very long time even without natural convection. This does not yet have catastrophic results for fission product release. The reason for this favourable behaviour is that at high temperatures heat is transported by radiation from the core of the pebble-bed reactor. In this case, the liner cooling, which can remove more than 1 % of the after-heat, can be activated for this purpose.

For an extrapolation of these ideas to larger units (3000 MW), the removal of after-heat according to the above-mentioned principle is then alleviated when the geometrical distance is small for the heat from the place of generation to the heat sink i.e. in the order of magnitude of $\sim$ 2 m. This indicates the suitability of a ring-core with a ring width of approx. 4 m. This core geometry could, in addition to advantages of after-heat removal, also favourably influence the constructability of large prestressed cast steel vessels (due to the introduction of a central column), shut-down and regulation(effective use of reflector rods) as well the ball flow behaviour (charateristic dimensions similar to AVR).In any case HTR systems are thus conceivable in which even improbable accidents could be without grave effects for the surroundings.

System / Data	LWR	HTR
Core power density	100 MW/m^3	5 MW/m^3
Structural material in core	Zircoloy 4	graphite
Meling point of structural material (subl. temp.)	1700 °C	3500 °C
Melting point of fuel (UO_2)	~2800 °C	~ 2800 °C
Heat capacity in core	$\sim \frac{1tZry}{m^3} + \frac{2,8tUO_2}{m^3}$	$\sim \frac{0,9tC}{m^3}$
Maximum operation temp. in fuel	~2300 °C	~1100 °C
Time limit after which the after-heat removal has to start	20 sec	>2 h

Table 3.9.1-1: Comparison of the Conditions for the After-Heat Removal of Different Reactor Systems

For fast breeders this time limit is even shorter caused by the very high core power density of ~500 MW/m^3.

In comparison to the other reactor systems, the requirements for the after-heat removal are much less for a HTR-system caused by the small power density, the large heat capacity in the core, the low fuel temperatures and the high sublimation temperature of graphite as table 3.9.1-1 shows. In case of the hypothetical accident that no after-heat removal works, there is a large delay time for the fission product release.

3.9.3 Accidents in Heat Exchanger Systems

According to the present considerations, the following accidents could occur which are directly connected with process heat exchangers:

- rupture of one or more reformer tubes
- breaking off of a reformer tube
- rupture of an internal collector
- rupture of a lid penetration for process-or product gas through the primary circuit closure
- rupture of a collector or a gas duct in the containment (for process gas as well as reformer gas)
- rupture of one or more steam generator pipes
- rupture of an internal steam collector
- rupture of a steam penetration through the primary circuit closure
- rupture of a steam collector or a steam duct in the containment
- sudden pressure loss on the helium side
- sudden pressure loss on the process side
- rupture of the tube-plate for the reformer tube
- earthquake connected with intolerable vibration of the steam reformer bundle
- catalyst damage caused by carbon deposition or mechanical influences.

The following figures 3.9.1-2 and 3.9.1-3 show the schematic principle of the primary circuit as well as the distribution of the reformer tubes and collectors for the design shown before in Vol. 1 (concrete variant with pod-boiler vessel)

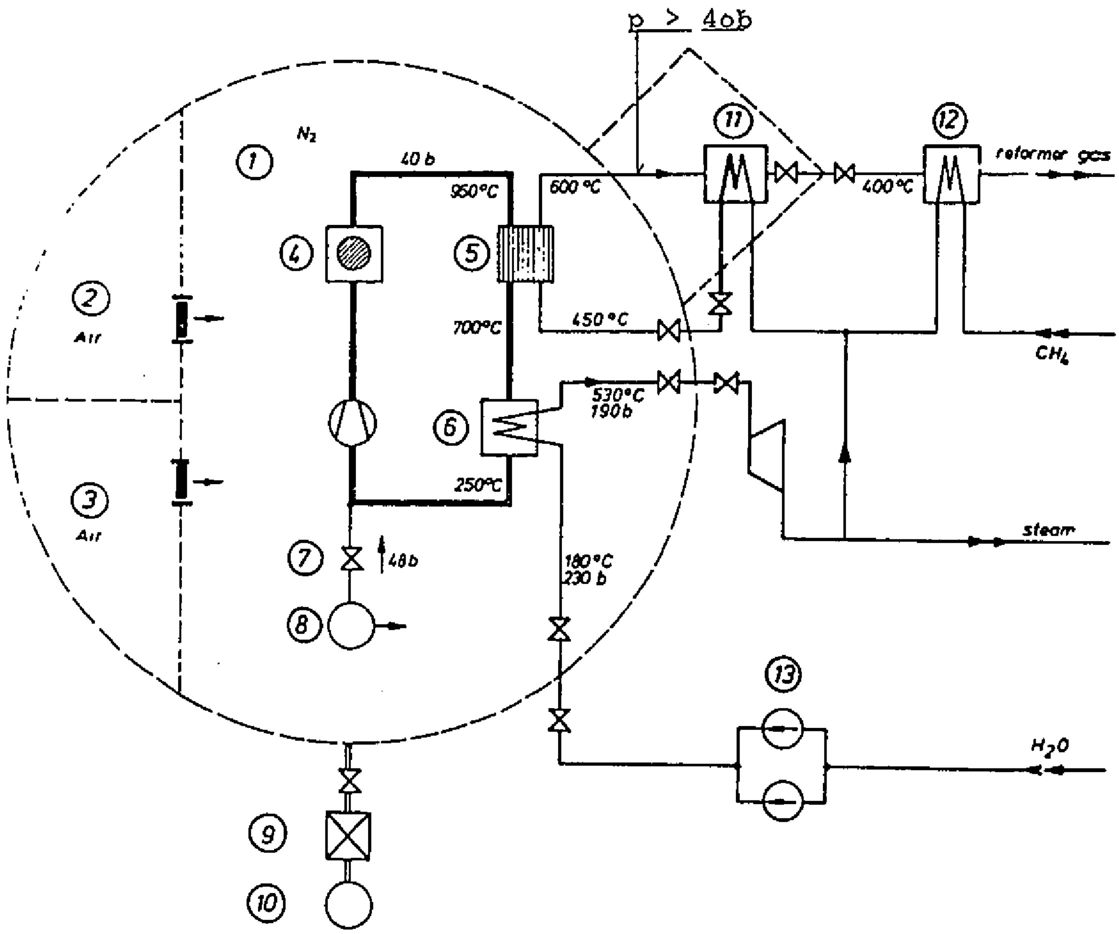


Fig. 3.9.1-2: Schematic Principle for Safety Considerations

1 *Containment (N_2)*
2 *Room for controlrods and loading systems*
3 *Fuel element discharge room*
4 *HTR*
5 *Steam reformer*
6 *Steam generator*
7 *Safety relief valve*
8 *Mixed cooler*
9 *Filter*
10 *Stack*
11 *Cooler*
12 *Cooler*
13 *Feedwater pumps*

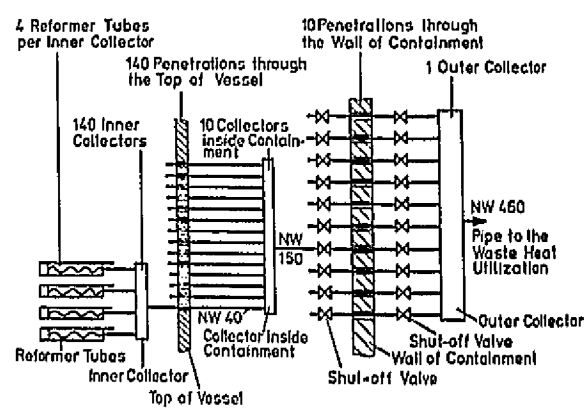


Fig. 3.9.1-3: Possible Connecting Up of Reformer Tubes under the Aspects of Safety

The most important inner accidents of the heat exchanger are the following:

Rupture of Steam Generator Tubes

After the in-rush of water,there is an increase of pressure in the primary circuit, the reactivity of the core increases, corrosion of the fuel elements and the graphite installations occur.

The following measures have to be taken after this accident occurs:

- the reactor is shut down by inserting the control rods to a depth of about 0.5 m
- the water supply to the corrosponding loop is stopped
- the safety valve of the primary circuit operates at 48 b, the mixture of helium and steam is blown into a mixing cooler (water vessel). Steam is condensed there, helium is introduced into the containment and a mixing pressure of $\sim$ 4 b results
- the after-heat removal is put into operation, the core is cooled very quickly below the temperature which is critical for corrosion.

The amount of water during the rupture of a tube should be about 2 t. This is attained by dividing up the pipes in the steam generator and by appropriate choice of the response times of the shut-off valves in the feed water ducts.

Damages to the Reformer Tubes

a) When the pressure in the reforming process is so chosen that it is always higher than the helium pressure, the process gas mixture (H_2, CO, CO_2, CH_4 and H_2O) flow into the primary circuit in case of tube damages. This accident can be handled exactly like that of rupture of a steam generator pipe. The only difference it that the throughput of a reformer tube is considerably smaller than the throughput of a steam generator strand. A tube produces about 400 Nm^3 H_2+CO/h on an average. The amount of reformer gas which enters the reactor core depends very much on how many tubes are connected together internally in the primary circuit. When each reformer tube is separate and can be plugged off separately,about 3 Nm^3H_2 + CO per accident ėnter the core. According to the proposal of connecting the tubes to collector systems, which is shown in fig. 3.9.1-3, about 12 Nm^3H_2 + CO enter the core. This amount can easily be removed by the gas purification. The safety valve for blowing off primary circuit gas does not have to be operated thereby. No activity is released to the surroundings by this accident.

b) If the pressure on the process side is under that of the primary circuit, helium enters the reformer gas during rupture of a reformer tube.

The flow area for helium is limited to the cross-section of the pigtail so that a fast depressurization of the primary circuit is prevented. However, the input of process gas must be stopped and valves have to be placed behind the first step of waste heat utilization in order to prevent the contaminated helium entering the chemical plant. The temperature of the reformer gas is sufficiently low here so as to allow technical realization of the valve. Helium is led to the containment via safety valves. This step of waste heat utilization is thus placed in the containment from the technical safety point of view. In any case, contaminated helium comes out of the containment by this choice of pressures.
According to present knowledge regarding safety,solution a) is more advantageous.

Damages to Steam Ducts in the Containment

In this case the pressure in the containment increases. It should not exceed 4 b. Through suitable limitation of the cross-section and sufficiently short response time of the shut-off valves, the pressure increase in the containment can be limited.

Damages to Collector Ducts for Process Gas and Reformer Gas in the Containment

The mass flow must be suitably limited here, too, to avoid intolerable pressure increase in the containment. The ducts must be surrounded by inert gas to prevent the formation of explosive mixtures. Conditions for explosions in the containment can be seen from the curves in fig. 3.9.1-4, which were obtained for the AVR in connection with a similar problem-posing.

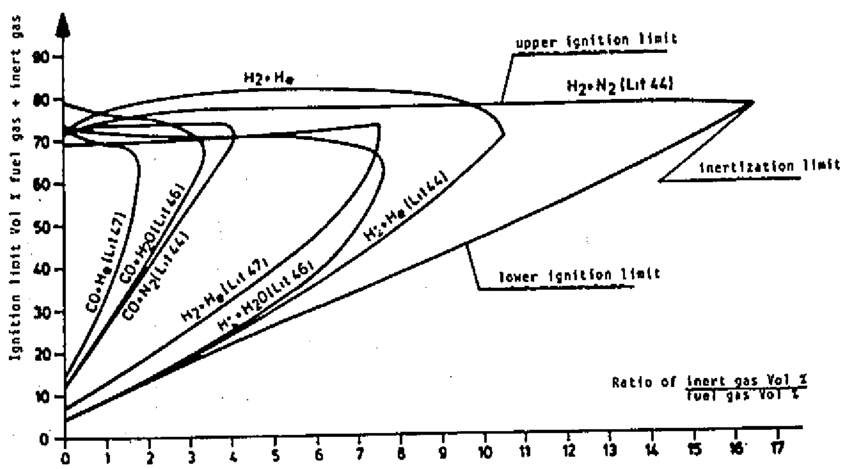


Fig. 3.9.1-4: Conditions for Explosions in the Reactor Containment [12]

Additionally, the conditions of gases in the gas production plants must be considered carefully to prevent explosions on the outside of the containment. This can be realized e.g. for the hydrogasification of coal or all other steam reformer applications by a large enough distance between the reactor and the chemical plant and by special measures for the connecting gas pipes.

3.10 Availability of the Nuclear Plant

For the economy of nuclear process heat plants, a high work- and time-availability of the nuclear part is of decisive importance.

Due to the continual loading in the pebble-bed reactor, it seems to be able to achieve the required 8000 h/a.

Fig. 3.10-1 shows that in the AVR test reactor, a work-availability of more than 7000 h/a was attained.

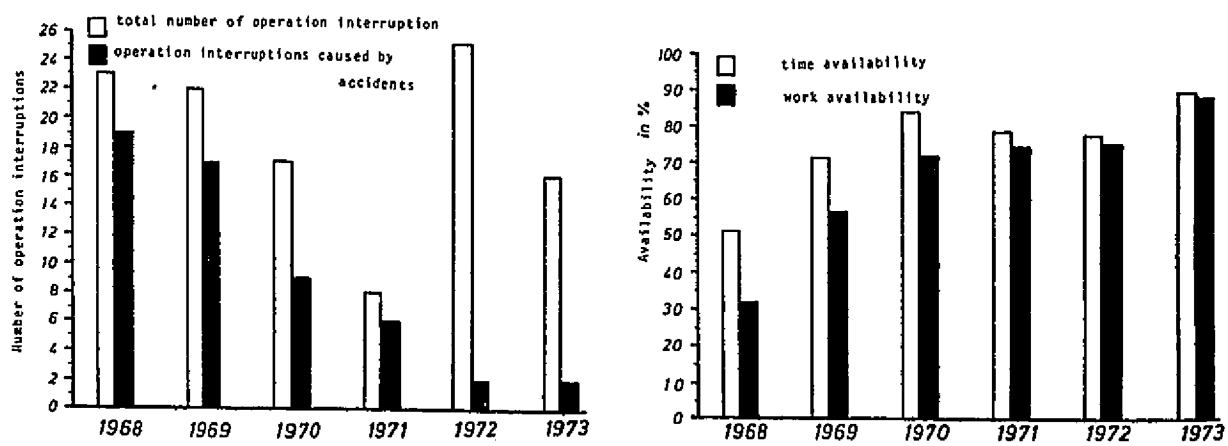


Fig. 3.10-1: Availability of the AVR [33]

The components requiring inspection such as circulators, control rods, gas circuits, gas purification and measuring instrumentation are so developed and well-known today that with a 4-year maintenance rhythm the desired high availability can be expected. The steam reformer as new and susceptible component should be so designed that the stresses (e.g. according to the pressure difference in the hot part of duct) are at a minimum. Furthermore, it is expedient to install 5 % more reformer tubes so that after rupture and plugging off of a tube, the plant can continue to be operated without removing the heat exchanger. According to today's knowledge, it is expected that after the projected service-life of 10^5 h, a removal of the whole tube bundle will take place.

One component, which could limit the availability of the HTR, is the upper part of the side reflector. According to the status of knowledge today a life-time of 30 years will be reached at a power density of 5 MW/m^3 and slight cooling of the side wall of the core structure by slowly flowing cold helium. In any case, the entire side and top reflector parts should be replaceable. The time required for such a manipulation lies in the region of about 3 months [35]. Considerations on being able to inspect and repair the entire core installations have to be continued. The fuel elements and their handling will probably cause no limits to the reactor operation. The fuel elements can continually be added and removed or discontinually during a slight disturbance. Repairs on the loading installations can be carried out also during normal operation if the necessary repair valves are provided.

The hot-gas ducts have to be so designed that it is possible for inspection and replacement to take place. According to considerations up to now, a loop system should have advantages here.

An inertization of parts of the containment which contain the heat exchanging aggregates is advantageous as protection against the formation of explosive gas mixtures or to prevent the burn-up of graphite in the core but naturally impedes the operation of the plant. Therefore, a placement of the gas ducts in inertized shafts is possibly advantageous.

3.11 Status of Development

After the successful operation of the AVR at 850 °C and 950 °C and after completion of the THTR program as well as corresponding work on the DRAGON project and by the firm GAC, the status of development of the HTR technology is as follows:

Fuel Elements [37, 5]

- The THTR fuel elements, which up to now have been successfully tested for the conditions there, can be qualified for application in process heat reactors at a helium outlet temperature of 950 °C.
- The release of fission products is the limiting factor in the fuel element design.
- BISO particles with LTI-coatings show sufficiently low fission product release.
- TRISO particles will be necessary above a helium temperature of $\sim$ 1000 °C.
- The ball form with its small dimensions enables testing of a lot of fuel elements under extreme conditions.
- The contamination of the matrix and the coating determines the release of the BISO particles and should be decreased by a factor of 10 by improvement of the production technique.
- The low packing density of the coated particles in the fuel element matrix prevents the breaking of the particles through radiation and enables a high heavy metal content
- The production process of the fuel elements can be automated due to the ball-shaped fuel elements.

Although the THTR fuel element seems to be suitable basically for application in nuclear reactors with outlet temperatures of 980 °C, the fuel elements will have to be modified for high outlet temperatures as well as higher conversion rates and their behaviour has to be elucidated in extensive radiation tests.

Three considerably different elements can be defined according to the requirements (s. table 3.10-2).

Element / Parameter	Dimension	Normal Burn-Up Element	Highest Temp. Element	High. Converter Element
Burn-up	$\frac{MWd}{t}$	∿100 000	∿ 100 000	< 30 000
Helium temper.	^{o}C	980	1050	980
Heavy metal content	g	12...15	15	∼30
Power density	$\frac{MW}{m^3}$	5	5	<5
Max. fuel temperature	^{o}C	1100	∼ 1200	1200
Max. surface temperature	^{o}C	1080	1180	1180
Max. fast dose	$\frac{n}{cm^2}$	$5 \cdot 10^{21}$	$5 \cdot 10^{21}$	$<4 \cdot 10^{21}$
Max. ball power	$\frac{kW}{ball}$	<4	<4	<3

Table 3.10-2: Data of Process Heat Fuel Elements

The main designs are demonstrated by table 3.10-2. The difference consists mainly in the heavy metal content as well as the fast neutron dose due to the varying burn-up and the maximum power of a ball. The specific fuel temperature for the high converter fuel element is about 100 ^{o}C above the usual values as this cycle has a flatter axial power profile than a normal OTTO-cycle. According to the existing fission product release calculations TRISO layers have to be applied to the element of type 2 and 3. For the type 1 both BISO mixed particles as well as separate feed (TRISO) and breed (BISO or TRISO) particles should be qualified. As an additional measure for mixed oxide particles the doping of the nucleii with oxide to retain the solid fission products should be further developed.

Core Structure

As shown in § 3.8, the graphite qualities developed together satisfy the requirements of a process heat reactor, if the power density and reactor inlet temperature are kept sufficiently low (i.e. $\sim 5\ MW/m^3$, $T_i \sim 250\ ^oC$). Due to the already mentioned reasons, however, the replacing of the top and upper side reflectors should be guaranteed. In order to obtain fully reparable systems, this technique will have to be applied to the reactor bottom, too. However, deloading of the reactor core is necessary for this. An increase in the number of fuel disloading ducts would be advantageous for the reactor bottom in order to have a more uniform pebble flow also in the lower part of the core. Tests on larger model scales are needed for this. Also ring-core structures, which would be of interest for prestressed cast steel reactor vessels with a supporting column in the middle, are not well-known enough yet and still require extensive model testing

Loading Equipment, Control Rods, Circulators, Steam Generator, Measuring Instrumentation

These components have been partly developed in the scope of the existing pebble-bed projects and have to be adapted to the special conditions of process heat plants. Especially, the spiral control rods should be further developed in order to decrease the forces on the fuel elements during insertion of the rods into the pebble-bed.

Hot-Gas Ducting

This component , whose design possibilities and problem-posing have been given in detail in § 3.4, certainly requires extensive development work and qualification tests before an objective choice for nuclear plants can be made. New directive results can be expected from the testing stands SUPEREVA, HHV and accompanying special tests.

Reactor Vessel

The prestressed concrete pod-boiler vessel can be considered to be technically accepted today in the HTR technology. However, it has yet to be shown that the increase of the pod number and pod diameter necessary for process heat application brings no basically new difficulties for this vessel concept. The idea of placing the heat exchangers outside the reactor vessel and so to reduce the dimensions of the reactor vessel considerably can only be realized when it has been ascertained that in case of rupture in the heat exchanger vessels or of rupture of the connecting coaxial ducts for the cold-gas ducting no intolerable pressure gradients result. Recent calculations for such accidents lead to the expectation that for an appropriate design, the pressure gradients can be held below 10 b/sec even without burst protective measures, so that it would be possible for the core to be after-cooled after such accidents. The reactor vessel for such a loop design would correspond in its dimensions and design data for a 3000 MW-plant to the THTR vessel. The development of prestressed cast iron or cast steel vessels would probably possess a whole series of advantages compared to prestressed concrete vessels for reasons mentioned in § 3.7. However, there are important questions to be answered, especially the type of seal welding, the design of the cooling of the individual blocks, the tolerance in the top and bottom parts and the form and design of prestressing. A test vessel will have to be constructed based on models which have solved the individual detailed problems.

Containment

This is a new component for HTRs. Neither the AVR, THTR nor Fort St. Vrain are placed in protective containers. For application of process heat reactors,the new accidents and the connected effects on the containment especially require yet exacter analysis. The proposed inertization and the new problems introduced therewith require extensive investigations. The proposed underground arrangement due to safety reasons has also to be investigated regarding its realization and the technical safety advantages. Especially the propagation of radioactivity in the ground has to be closely studied in this connection.

Safety and Licensing

A series of safety experiments have yet to be undertaken for HTRs. The possibility of the in-rush of air into the reactor core and the connected process of a possible burn-up of the core should be investigated. Furthermore,the possibility of activating the liner cooling in case of a more than hypothetical accident of failure of all active emergency cooling measures for after-heat removal should be investigated and the reactor design be adapted to the resulting requirements. In this connection, the technical safety advantages of a ring-core should be exactly studied. Finally, the licensing procedure for the process heat HTR, which has no prototype up to now, has to be introduced and completed. Especially, all questions regarding explosions have to be taken into consideration. According to the existing plans to possibly start with the construction of a demonstration plant in 1980, this problem must and will be tackled in the next 5 years in the scope of the project "Prototype Nuclear Process Heat".

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