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GCFR CORE THERMAL-HYDRAULIC DESIGN*

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ABSTRACT

The approach for developing the thermal-hydraulic core assembly designs for the gas-cooled fast reactor (GCFR) is reviewed, and key considerations for improving the core performance at all power and flow conditions are discussed. It is shown how the thermal-hydraulic core assembly designs evolve from evaluations of plant size, material limitations, safety criteria, and structural performance considerations.

INTRODUCTION

Several design developments and improvements have been considered for the GCFR demonstration plant. The most notable of these is the change of the helium coolant flow direction through the core from downflow to upflow. The principal consideration motivating the change to upflow cooling was the enhancement of natural circulation cooling capability for residual heat removal (RHR). The change in the flow direction had an effect on the nuclear steam supply system (NSSS), as well as on the core design of the GCFR demonstration plant.

The reactor and its associated coolant circuits of the upflow GCFR are shown in Fig. 1, and system parameters are given in Table 1. The plant design is discussed in detail in a separate paper (Ref. 1).

The general configuration of the reactor core for this design is illustrated in Fig. 2. The active core region consists of 169 hexagonal

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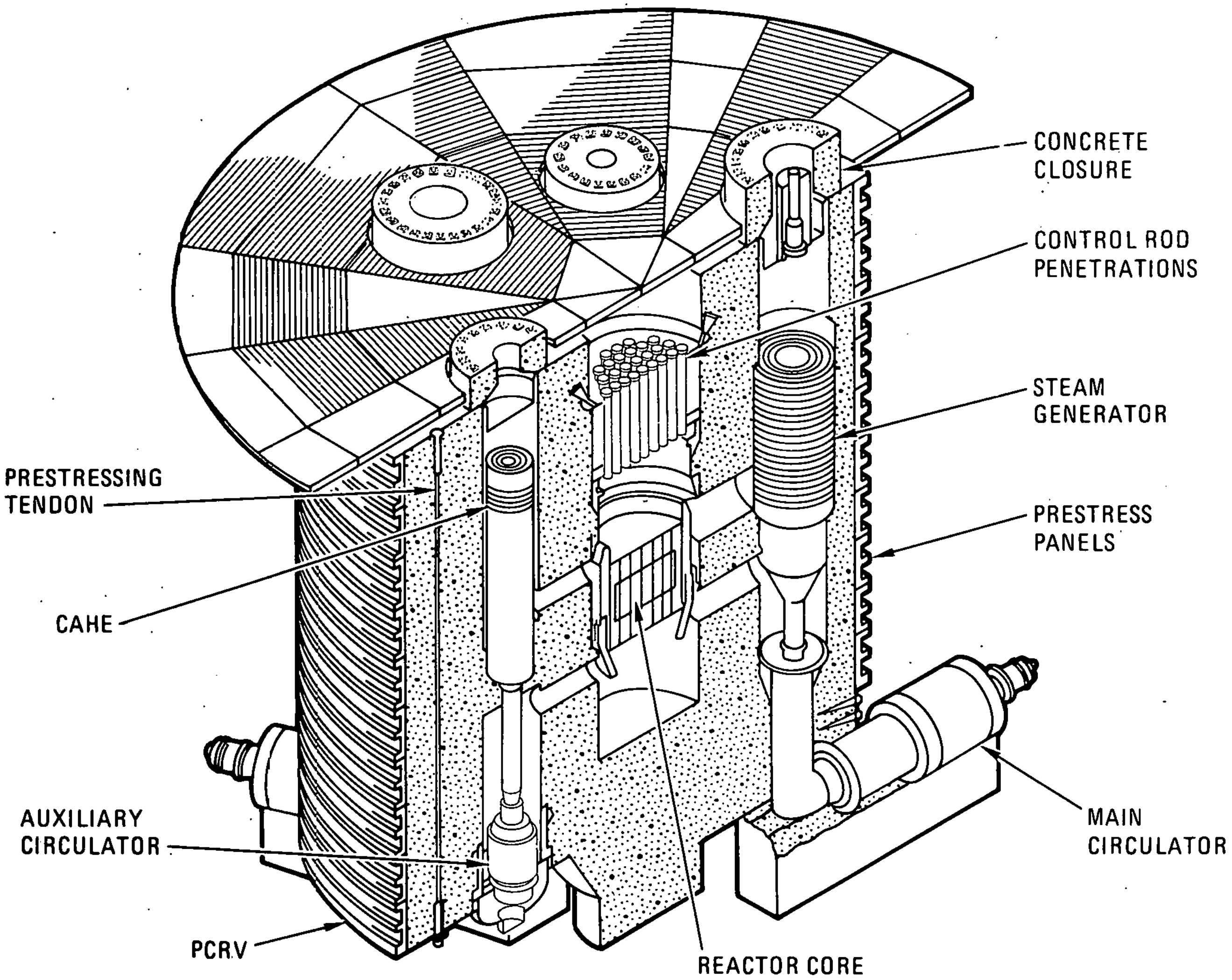
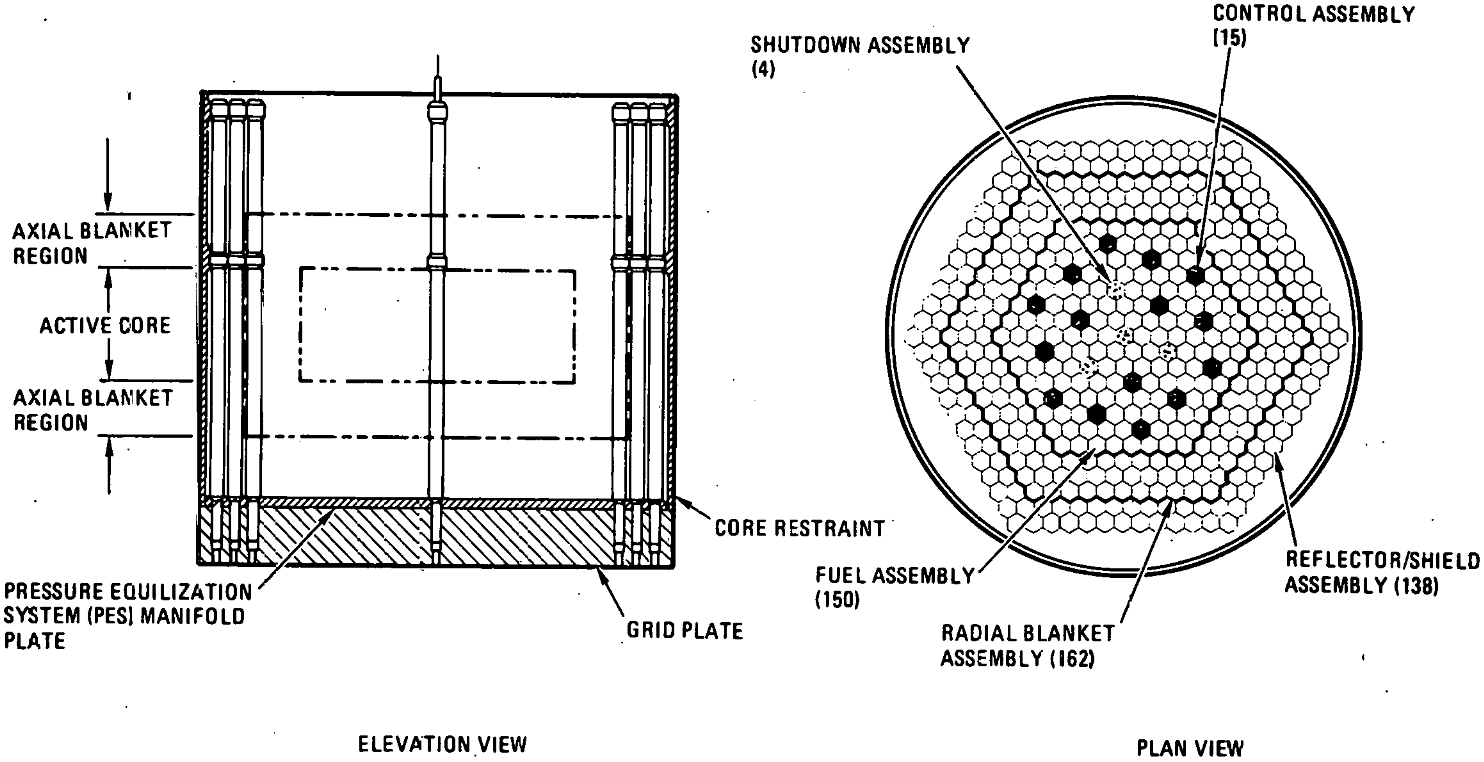


Fig. 1. GCFR demonstration plant NSSS



ELEVATION VIEW

PLAN VIEW

Fig. 2. GCFR upflow core, demonstration plant, general configuration

assemblies; 150 of these are fuel assemblies, 15 are control assemblies, and 4 are secondary shutdown assemblies. The central core region is surrounded by three rows of radial blanket assemblies. The radial blanket region is surrounded by two rows of reflector/shield assemblies. The process of thermal-hydraulic core assembly design and key considerations affecting the core performance are discussed.

TABLE 1
CORE GENERAL OPERATING CONDITIONS

Reactor power, MW(t)	1088
Reactor inlet temperature, °C (°F)	293 (560)
Reactor outlet temperature, °C (°F)	526 (978)
Reactor pressure, MPa (psi)	10.5 (1523)
Reactor ΔP, kPa (psi)	183 (26.5)
Clad maximum temperature, °C (°F)	750 (1382)
Maximum rod rating, W/cm (kW/ft)	385 (11.74)
Total flow rate, kg/s (lb/s)	891 (1964)

CORRELATIONS

Proper design and analysis of GCFR core assemblies depend on the accuracy and reliability of thermal-hydraulic correlations used in the analysis. The following types of correlations are required for two types of surfaces (rough and smooth), two types of rod spacers (grid and wire wrap), three types of coolant flow channels (interior, wall, and corner), and all flow regimes:

1. Friction factor.
2. Stanton number.
3. Spacer effect on pressure drop and heat transfer (loss coefficient.
4. Mixing factors.

As a result of a literature search, interaction with international experts, a comprehensive experimental program, and benchmark calculations, a set of reference thermal-hydraulic correlations has been developed and documented (Refs. 2-4).

THERMAL-HYDRAULIC CODES

Two principal codes are used for thermal-hydraulic design of the GCFR core assemblies. The first code, CALIOP (Ref. 5), is used for core parametric and scoping studies. The second code, COBRA*GCFR (Refs. 6, 7), is for detailed thermal-hydraulic analysis and design of the assemblies. These codes are discussed briefly below.

CALIOP Code

The CALIOP computer program is a fast breeder core system design tool that considers core system geometries, thermal-hydraulics, and nuclear physics. The code is used for initial core parametric and scoping studies, core assembly coolant orifice evaluations, and initial fuel management calculations.

Thermal-hydraulics and nuclear physics included in CALIOP are simplified representations of rather complex phenomena and core characteristics. The intent of the simplified and analytical approach is to provide a fast and economic means to study the impact of a large number of different core design and performance objectives. This means that calculational accuracy has been somewhat compromised in favor of calculational speed and cost. Still, high confidence in the calculated results is required to reach valid conclusions that should remain valid when more detailed thermal-hydraulic and physics calculations become available on specific core designs.

A number of options are available in the CALIOP code. For example, for a given power level, core ΔP, number of fuel elements, number of rods per element, core length, blanket thickness, and maximum permissible hot spot temperature, CALIOP will determine the fuel rod pitch-to-diameter ratio and the core outlet temperature. The preliminary configuration of the core assemblies obtained by the CALIOP code is analyzed in detail by the thermal-hydraulic subchannel code COBRA*GCFR (Ref. 6).

COBRA*GCFR Code

The COBRA*GCFR code is a thermal-hydraulic subchannel analysis code (Ref. 6) which has been adapted for GCFR applications from the COBRA-IV code (Ref. 7) developed for light water and liquid metal reactors. The code calculates flow, pressure, and temperature distributions by subdividing the assembly into subchannels. Figure 3 shows a subdivision used for GCFR fuel assembly analyses. The principal features of the COBRA*GCFR code include proper thermal-hydraulic treatment of smooth, rough, and wire-wrapped bundles; rod circumferential heat conduction; radiation heat transfer; and buoyancy effects.

The preliminary verification of the COBRA*GCFR code is currently in progress through a series of benchmark calculations where code calculations are compared to those of other existing codes and experimental results. The COBRA*GCFR has shown good agreement at all flow ranges in these benchmark calculations. Figure 4 shows one such comparison with the 12-rod BR-2 calibration experiment* conducted at the Institute of Neutron Physics and Reactor Research (INR) at Karlsruhe, West Germany (Ref. 8).

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*12-rod BR-2 calibration experiments are out-pile thermal-hydraulic tests conducted in pressurized helium for calibrating a 12-rod bundle designed for radiation tests in the BR-2 test reactor in Belgium.

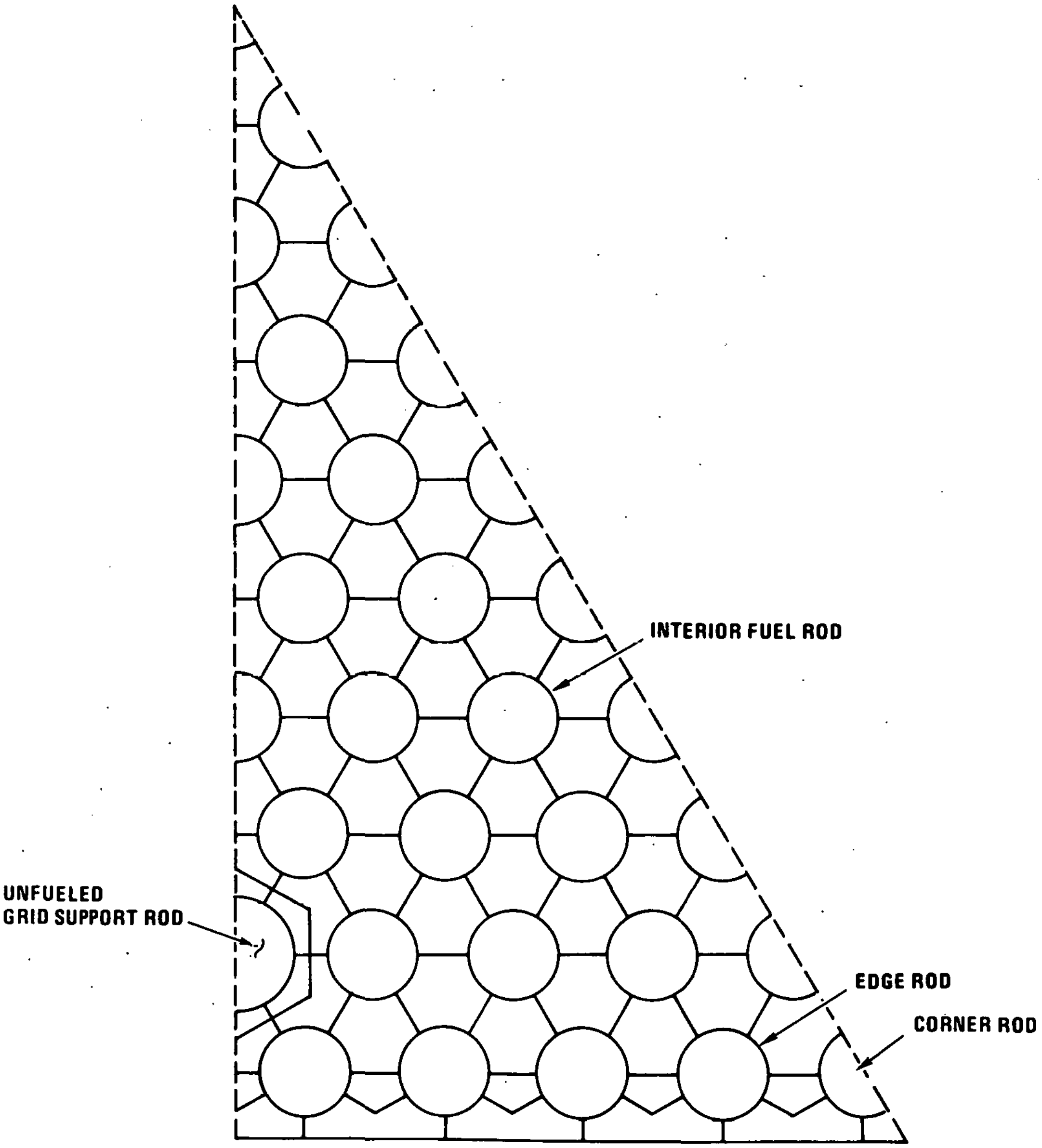

Fig. 3. One-twelfth section model of the GCFR fuel assembly

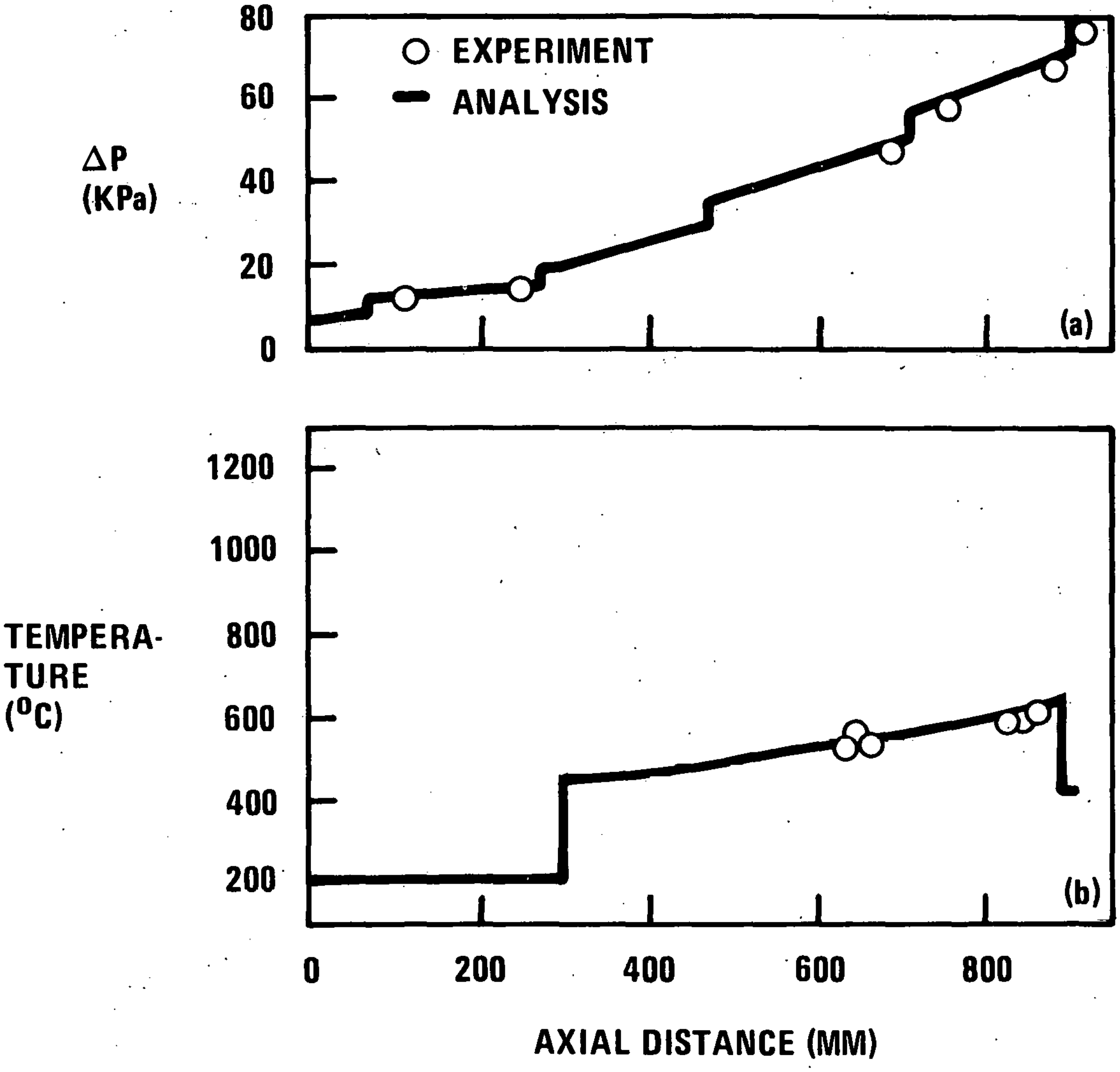

Fig. 4. Comparison of BR-2 calibration experiment and COBRA*GCFR analysis: (a) bundle pressure drop, and (b) rod surface temperatures

HOT SPOT ANALYSIS

Thermal-hydraulic analysis of GCFR assemblies using subchannel codes predicts nominal temperatures; i.e., it does not consider uncertainties such as a change in geometry from nominal dimensions or uncertainties in thermal-hydraulic correlations, properties, physics calculations, etc. All these uncertainties are accounted for by defining hot spot factors that are applied to the nominal temperatures to calculate the hot spot temperatures. For the GCFR demonstration plant, the hot spot midwall cladding temperature is limited to 750°C.

To calculate the midwall hot spot cladding temperature, three hot spot factors are defined:

F_{ch} = channel hot spot factor,
F_f = film hot spot factor,
F_{cl} = cladding hot spot factor,

and the midwall hot spot cladding temperature is obtained as follows:

$$T_c = T_{in} + (\Delta T)_{ch} * F_{ch} + (\Delta T)_f * F_f + (\Delta T)_{cl} * F_{cl} \quad ,$$

where T_c = hot spot midwall cladding temperature,

T_{in} = coolant inlet temperature,

$(\Delta T)_{ch}$ = channel temperature rise to the location of hot spot,

$(\Delta T)_f$ = film temperature drop at the location of hot spot,

$(\Delta T)_{cl}$ = nominal temperature drop from surface to midwall cladding at the location of hot spot.

The hot spot factors are calculated by using a semistatistical method. In this method, the uncertainties are divided into two groups: (1) uncertainties that occur randomly (e.g., manufacturing and assembling tolerances, material properties, and correlations) are treated as statistical subfactors, and (2) those that occur nonrandomly (uncertainties in calculations and measurements) are directly added as cumulative subfactors. The subfactors are combined by a method similar to the one used for the Clinch River Liquid Metal Fast Breeder Reactor (LMFBR). Table 2 summarizes the hot spot factors calculated for the GCFR fuel assembly. The 2σ deviations are used for all but faulted operating conditions.

TABLE 2
HOT SPOT FACTORS FOR GCFR FUEL ASSEMBLY

	Channel F_c	Film F_f	Cladding F_{cl}
Turbulent flow			
3σ[a]	1.131	1.361	1.195
2σ	1.111	1.311	1.153
Laminar flow			
3σ	1.236	1.212	1.138
2σ	1.187	1.173	1.116

[a] σ = standard deviation; thus, 2σ indicates 97.72% confidence level and 3σ indicates 99.87% confidence level.

SCOPING STUDY

The GCFR core design is determined by various parameters, which depend on a number of overall considerations.

Plant Size

The plant size is selected depending on plant purpose (i.e., reactor experiment, demonstration, commercial application) and component feasibility. For GCFR demonstration plants, sizes of 800 to 1100 MW(t) have been considered; for commercial plants, usually six loops with about 600 MW(t) each have been evaluated.

Component Technology

Component technology has a limiting impact on key components that results in limits on major operating conditions. For example, the pressure containment capability of the reactor vessel introduces an upper limit for the primary system pressure; the circulator power limit influences the core pressure drop; and fuel handling and assembly duct dilation considerations impose limits on fuel assembly size.

Fuel Life Goals

Fast breeder fuel economics require long fuel in-pile times and have resulted in fuel life goals equivalent to burnups of about 100 MWd/kg heavy metal. This goal has a direct impact on fuel assembly components, such as assembly and duct size and duct wall thickness. For the GCFR demonstration plant, 70 MWd/kg has been recommended.

Material Limitations

Overall plant economics dictate high temperatures of the working fluid, which results in high fuel and fuel cladding temperatures. Both temperatures are limited by material properties. For proper fuel performance, the centerline temperature must stay below the melting point; for the cladding temperature, long time effects of radiation at high temperature and interaction with the fuel are important. Cladding irradiation tests for the LMFBR and GCFR have focused on feasibility demonstrations of a maximum cladding temperature of 700° to 750°C; maximum linear ratings range from 329 to 395 W/cm (10 to 12 kW/ft) for the GCFR, which results in centerline fuel temperatures well below the melting point of mixed oxide fuel.

For the GCFR demonstration plant core, a hot spot cladding temperature of 750°C has been selected. This appears to be significantly higher than the 730°C indicated in the Clinch River breeder reactor analysis. However, it should be remembered that the hot spot temperature is the maximum temperature considering all possible uncertainties. The nominal maximum temperatures should be compared for the different breeder concepts; this temperature appears to be lower for the GCFR as shown in Table 3.

TABLE 3
COMPARISON OF CLADDING TEMPERATURES

	LMFBR	GCFR	
Maximum hot spot cladding temperature, °C	730	700	750
Maximum nominal cladding temperature, °C	660	600	640

CORE CONFIGURATION AND CORE ASSEMBLY DESIGNS

The discussion so far has indicated that a large number of core design and operating parameters are dictated by plant economics and component technology. Key parameters that remain to be determined are: fuel and radial blanket rod diameter, rod pitch, number of rods per assembly, and core length. These parameters strongly affect the thermal as well as the nuclear performance, which is measured by breeding ratio and doubling time. The parameters are selected based on sensitivity studies.

Fuel Rod Roughness Configuration

Through an extensive research program at EIR,* the optimum roughness configuration has been determined (Ref. 9). The thermal-hydraulic correlations for the optimum roughness shape are applied when fuel rod diameter and pitch are selected. Subsequently, a calculation is performed to determine the optimum roughness height. The results of these rib height calculations for the current demonstration plant core are shown in Fig. 5.

Thermal-Hydraulic Core Assembly Design and Performance

Using the nominal operating data obtained with CALIOP in the scoping study, detailed assembly analyses are conducted using the subchannel analysis code COBRA. A typical fuel assembly model for COBRA analyses is shown in Fig. 3. This detailed design analysis results in determination of parameters such as edge channel size; location and diameter of grid spacer support rods; flow, pressure, and temperature distributions; and wire-wrap pitch for the radial blanket.

The overall plant design and nominal core operating conditions are presented in Ref. 1. Details of the mechanical designs are described in Ref. 10. The description of the thermal-hydraulic design of fuel and radial blanket assemblies is depicted in Table 4.

During the evolution of core assembly design, careful attention must be given to an efficient use of the pressure drop in various portions of the assembly. Although a very compact assembly design is desirable, the pressure drop associated with each component has to be minimized. The pressure drop distribution obtained for the current fuel assembly design is shown in Table 5. The largest portion of the pressure drop (~45%) is caused by the bare rod bundle; a rather large fraction is caused by the grid spacers (~22%). The pressure drops in the inlet and outlet regions are caused mainly by shielding requirements.

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*Federal Institue of Reactor Research, Switzerland.

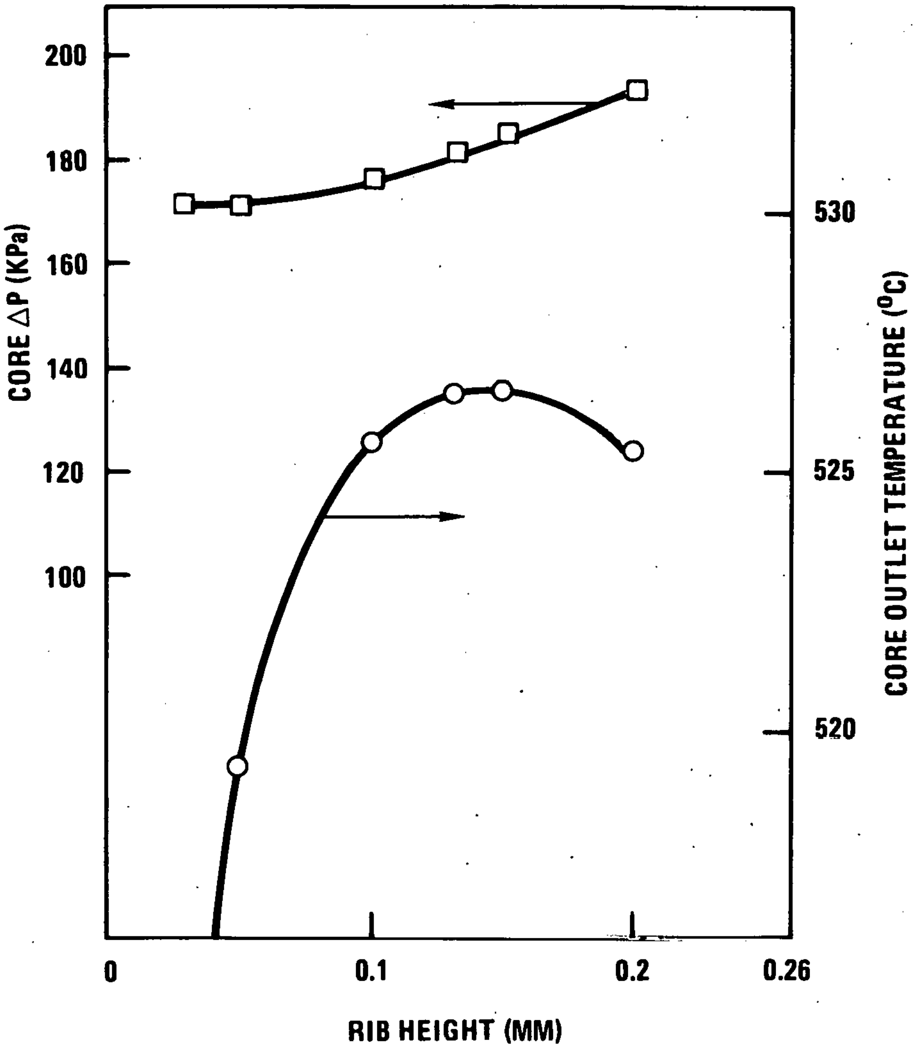

Fig. 5. Rib height study for three-loop GCFR demonstration plant

TABLE 4
CORE ASSEMBLY DESIGN AND PERFORMANCE

Fuel assembly	
Active core height, mm	1,200
Number of fuel rods per assembly	265
Fuel rod diameter, mm	8
Fuel rod pitch, mm	11.5
Duct-to-rod clearance, mm	2.87
Roughness height, mm	0.13
Roughness pitch, mm	1.56
Maximum assembly flow, kg/s	6.6
Reynolds number	90,300
Inlet temperature, °C	293
Outlet temperature, maximum assembly, °C	561
Radial blanket assembly	
Rod diameter, mm	22.2
Rod pitch, mm	24.2
Wire-wrap pitch, mm	300
Maximum assembly flow, kg/s	1.5
Reynolds number	40,000

TABLE 5
PRESSURE DROP IN VARIOUS PARTS OF THE GCFR FUEL ASSEMBLY (THREE-LOOP DEMONSTRATION PLANT)

Region	ΔP (%)
Inlet	8.3
Lower axial blanket	4.9
Core region (rough)	43.8
Upper axial blanket	7.1
Spacers	22.6
Acceleration	2.1
Outlet	11.2

The selection of nominal design parameters has to account for acceptable performance of all assembly components and has to include special effects such as fabrication tolerances that go beyond the hot spot analysis.

Edge Channel Design

In the current reference design, the fuel assembly edge channel is composed of rough fuel rods and a smooth flat duct wall. For this design, a large edge channel is required because of four different effects:

1. Under low power and flow conditions, the heat input to the edge channel sees a relative increase while the flow rate experiences a relative reduction. These disadvantages are slightly counterbalanced by overcooling under normal operating conditions.
2. In-pile creep causes the duct to dilate more midway along the flats than at the corners. The relative impact of this dilation can be reduced by making the channel large and by including flow resistances to avoid excessive overcooling of the edge channels.
3. Mechanical design requirements of the grid spacers require a relatively thick edge band, which results in a flow blockage in the edge channel that tends to be larger than that in interior subchannels. This, in turn, leads to flow redistributions and heat transfer modification at the spacer level and creates sawtooth-type fuel rod temperature distributions, as shown in Fig. 6. From a flow and temperature distribution point of view, uniform relative flow blockages are desirable. This can be achieved by large edge channels.
4. Due to fabrication tolerances of duct inner diameter and outer spacer dimensions, the possible edge channel geometry covers a wide range. The fractional variation of the edge channel due to

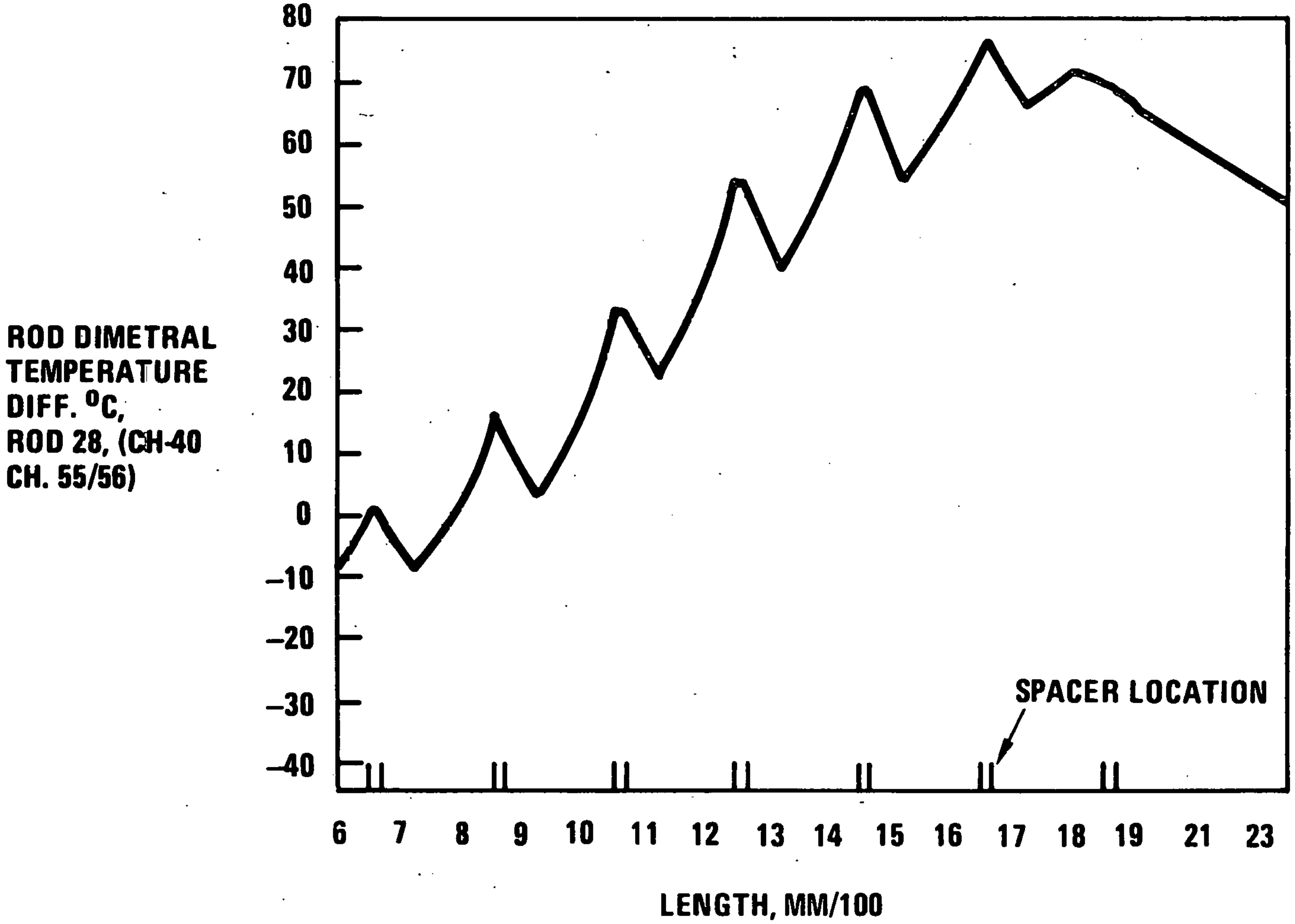

Fig. 6. Rod diametral temperature difference versus length

the fabrication tolerances can be minimized by using large edge channels.

The above findings have led to the recommendation of a large nominal edge channel. The duct-to-rod clearance selected is 80% of the rod-to-rod clearance. However, this large edge channel leads to a temperature difference across the edge rods of about 75°C and a temperature difference swing associated with the sawtooth temperature distribution of 32°C, which together with radiation-induced differential metal swelling leads to unacceptable fuel rod bowing and to a fuel rod life equivalent to about 80% of the life goal of 70 MWd/kg. Analytical studies have shown that roughening of the inner duct surface results in additional flow resistance sufficient to reduce the temperature difference and the difference swing to 53° and 24°C, respectively, and reduce bowing such that a rod life equivalent to a burnup of 70 MWd/kg can be reached. The results are summarized in Table 6. The need for a roughened duct strongly depends on the fast fluence and on the metal swelling correlation; both are currently associated with a relatively large uncertainty.

Corner Rod Design

The corner subchannel has a relatively small flow area. Under normal operating conditions, the power-to-flow in that channel is similar to that in the other channels. No large temperature gradients exist. However, the hydraulic diameter of the corner subchannel is much smaller than those of the other channels so that, under off-normal operating conditions, i.e., at low power and flow conditions, this channel tends to show unfavorable power-to-flow ratios. The selection of a large rod-to-duct spacing largely compensates for this effect.

TABLE 6
EDGE ROD TEMPERATURE DISTRIBUTIONS AND FUEL ROD LIFE

Assumptions

Edge channel spacing, %	80
Spacer band thickness, mm	
Interior channel	0.4
Edge channel	0.7

Analytical Results

Inner Duct Surface	Fuel Rod Temperature Difference (°C)	Temperature Difference Swing (°C)	Fuel Rod Life (h)
Smooth	76.3	32	14,400
Roughened	53	24	17,100

Analytical studies have shown that an 80% edge spacing leads to an overcooling condition for the corner rods with a less severe temperature distribution than that of the edge rod; under steady-state low power and flow and considering negative fabrication tolerances, the corner rod still remains cooler than the interior rods. This was not the case for previous designs that had smaller rod-to-rod and rod-to-duct clearances. Also, the analysis technique was greatly improved to include circumferential conduction in the fuel rods. Because of the small hydraulic diameter of the corner channel, the temperatures in this channel are sensitive to γ-heat produced in the duct; therefore, transient analyses are required to fully describe the thermal characteristics of the corner channel.

Another design option that is being evaluated is relocation of the unfueled grid spacer support rod into the corner position. Analyses have shown that the unfueled corner rod leads to even lower temperatures because of reduced heat input. Transient analyses need to be conducted to find the most suitable corner rod design.

Effect of Fabrication Tolerances on Fuel Rod Life

The temperature gradients across edge and corner fuel rods are strongly affected by the fabrication tolerances. Deviations from nominal design due to fabrication tolerances lead to large positive and negative flow area deviations in the edge channel. As shown in Table 7, both extreme tolerance effects lead to unacceptable rod bowing and fuel life reduction effects. In the case of a stack-up of plus tolerances, a large temperature gradient develops across the edge rods and in the case of a stack-up of minus tolerances, the temperature gradient swing due to spacer solidity effects reduces the fuel life. In addition, the negative tolerance case develops excessive temperatures during low flow operation.

From a comparison of data shown in Tables 6 and 7, it can be seen that selections of 80% edge spacing and roughened duct surfaces alone do not sufficiently correct the detrimental effects of fabrication tolerances in the edge rod performance. Depending on the ultimate swelling correlations and the fast fluence at end of fuel life, a drastic reduction of duct and spacer fabrication tolerances may be required.

Advanced Design Considerations

Because of different hydraulic characteristics of the different flow channels within the rod bundle, uniform temperature distributions at all flow conditions cannot be achieved. Advanced design alterations are being reviewed to compensate for, or to correct, nonuniform rod temperature distributions. The introduction of roughness on the inner surface of the flow duct represents a substantial improvement in the thermal-hydraulic assembly design and performance. A further design improvement is seen in flow mixing vanes, which are successfully employed in light water reactor (LWR) cores. The use of mixing vanes in the GCFR fuel assembly may enhance coolant mixing and result in more uniform coolant temperatures at all flow conditions.

TABLE 7
EFFECT OF FABRICATION TOLERANCES ON
EDGE FUEL ROD TEMPERATURE DISTRIBUTION
(SMOOTH DUCT WALL, 80% NOMINAL EDGE SPACING)

Stack-up of Fabrication Tolerances	Fuel Rod Temperature Difference (°C)	Temperature Difference Swing (°C)
Plus	107	--
Nominal	76.3	32
Minus	44	44

Both design features are being reviewed because of their potential for reducing temperature differences within fuel assemblies at all flow conditions. In addition to increasing the fuel rod life expectancy, there are several other attractive advantages. Because of smaller differences between maximum and average temperatures, the mixed mean core outlet temperature will be increased and coolant flow and pumping power requirements will be decreased. In addition, mixing vanes may have a direct impact on hot spot factors. These investigations are currently in progress.

Fuel and Blanket Assembly Orificing

Depending on the life of the core assemblies and their location in the core, the fuel and blanket assembly power varies by a large factor from the average values. To obtain a maximum mixed mean core outlet temperature, the assemblies are orificed to control the flow through the individual assembly. Ideally, each core assembly can be orificed so that the mass flow through the assembly is just enough to keep the midwall hot spot temperature below 750°C during a particular cycle. However, this requires a large number of orifice sizes; hence, a trade-off must be made between the number of orifice sizes and the thermal performance.

Figure 7 shows such a study for the GCFR fuel assemblies. With a continuously variable ideal orificing, a mixed mean core outlet temperature of 544°C is possible. With fixed but ideal orificing, the mixed mean outlet temperature is reduced by 10°C to 534°C. Neither of these options is feasible due to the large number of orifices required. Reducing the number of orifices for the fuel assemblies reduces the outlet temperature. Since the reduction in the mixed mean outlet temperature, by using five types of orifices compared to 50, is quite small (i.e., 3°C), the GCFR demonstration core will have only five orifice groups for the fuel assemblies. A similar study indicates five groups for the radial blanket assemblies. This scheme yields a mixed mean core outlet temperature of 526°C.

SUMMARY

The approach for developing the thermal-hydraulic core assembly designs for the GCFR are reviewed using fuel and radial blanket assemblies as examples. The effects of edge channel size, grid spacers, and fabrication tolerances on fuel rod temperature distributions and fuel rod life are introduced. Thermal-hydraulic design considerations and their potential effects on fuel assembly performance are reviewed. The design fuel life can be reached in the GCFR demonstration plant core. However, depending on fast fluence and correlations for radiation-induced metal swelling, special thermal-hydraulic design measures, such as roughened duct walls and/or flow-mixing vanes, may be required to generate more uniform temperature distributions.

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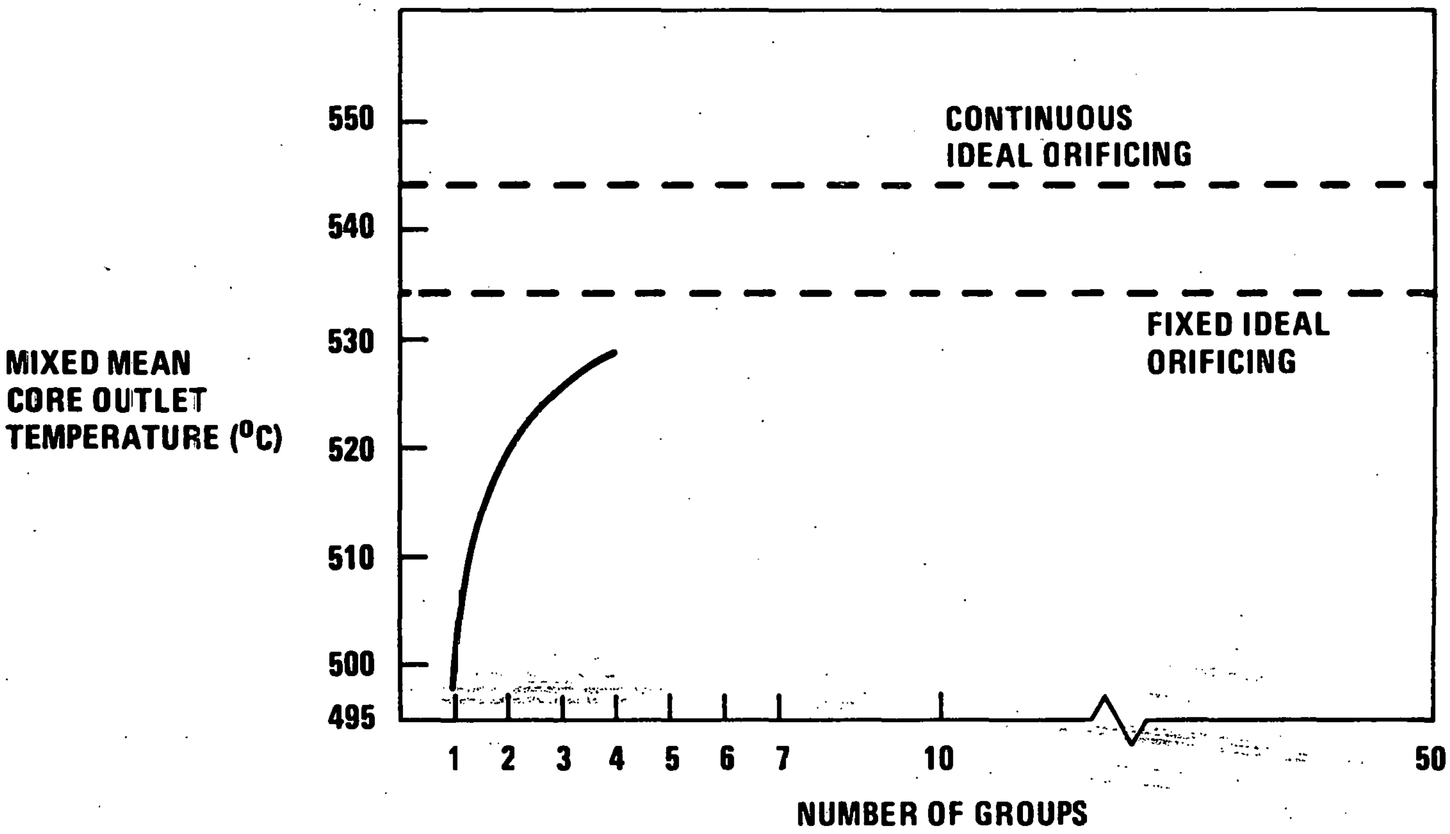

Fig. 7. Effect of fuel orificing groups on outlet temperature (assumes ideal fixed blanket orifices)

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