Research and Development Programs for HTGRs in JAERI

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ABSTRACT

Since 1969, JAERI has conducted research and development (R&D) programs for High-Temperature Gas-Cooled Reactors (HTGR). And the High Temperature engineering Test Reactor (HTTR), which will be the first High Temperature Gas Cooled Reactor (HTGR) in Japan, is under licensing process now [1].

In this paper, some of results of R&D are outlined in the following fields which are closely connected with the HTTR design, that is, i) fuel, ii) nuclear design, iii) thermal-hydraulic design, iv) graphite structure and v) high temperature metal structure.

In the field of fuel, extensive investigations have been performed to develop the fabrication technology of coated particle fuel (cpf). In parallel, data of coated fuel particle failure and fission product release in in- and ex-reactor experiments as well as mechanical properties data were obtained and irradiation tests have been done using the Oarai Gas Loop No.1 (OGL-1) to verify the integrity of mass-produced fuel.

Concerning the nuclear design, critical experiments were conducted using the Very High-Temperature Reactor Critical Assembly (VHTRC). Also carried out were hydrodynamical and thermal experiments using the Helium Engineering Demonstration Loop (HENDEL).

On the graphite structures which compose the reactor internals, design criteria have been developed based on ASME B&PV Code Section III Div.2, subsection CE and design data have been accumulated on a domestic graphite material.

High temperature metal structure is also one of major subjects of R&D for HTGRs. Hastelloy XR, which is a modified version of Hastelloy X, was developed and various tests have been conducted

which include creep tests, creep-fatigue tests, etc. to establish design criteria and allowables. Component tests of the Intermediate Heat Exchanger (IHX) have been also performed.

1. Introduction

The High Temperature engineering Test Reactor (HTTR), which will be the first High Temperature Gas Cooled Reactor (HTGR) in Japan, is under licensing process now [1].

The HTTR consists of a core of 30MWt, a main cooling circuit, an auxiliary cooling circuit and related systems as shown schematically in Fig.1. Table 1 summarizes major design parameters of the HTTR. The reactor pressure vessel is 13.2m high and 5.5m in diameter and contains the core, graphite reflectors, core support structures and the core restraint mechanism as shown in Fig.2.

This paper presents a brief overview of R&D programs in connenction with the HTTR design.

2. Fuel design

As is shown in Fig.3, a fuel element assembly of the HTTR is made up of fuel rods and a hexagonal graphite block. The fuel consists of TRISO coated particles of low enriched uranium oxide whose average enrichment is about 6% and the kernel diameter is 600μm. The particles are dispersed in the graphite matrix and consolidated to form a fuel compact. These compacts are contained in a sleeve to form a fuel rod and these fuel rods are contained within vertical holes of a graphite block.

In the fuel design of HTGRs, it is very important to retain fission products within particles so that their release to the primary coolant may not exceed an acceptable level. From this point of view, in the HTTR safety criteria, the failure fraction of as-fabricated fuel coating layers is limited to 0.2 % and the fuel temperature is limited below 1495°C under normal operating conditions and below 1600°C under abnormal transient conditions in order to avoid additional fuel failures during operation.

Extensive investigations have been performed to develop the fabrication technology of coated particle fuels (cpf). In parallel, coated fuel particle failure and fission product release behavior has been investigated by in-pile, e.g., Oarai Gas Loop No.1 (OGL-1) and out-of-pile experiments. Data on fuel properties have been obtained for the thermal-hydraulic design and safety analysis.

In the OGL-1, irradiation tests of the HTTR fuel have been performed mainly to investigate the integrity under normal operating conditions [2].

The OGL-1 is a high-pressure in-pile gas loop installed in the reflector zone of the Japan Material Testing Reactor (JMTR). The OGL-1 was put into service operation in March 1977 and has been operated for more than 22,000 hours. A flow diagram of the OGL-1 is shown in Fig.4. The coolant gas pressure is 3MPa, the maximum gas temperature of 1000°C, thermal and fast neutron fluencies are $6x10^{13}$ $/cm^2s$ and $1x10^{13}/cm^2s$, respectively.

The conditions and results of irradiation tests of the HTTR fuel conducted in the OGL-1 are listed in Table 2 [3]. The change of the release rate to birth rate (R/B) of ^{88}Kr during irradiation is shown in Fig.5 [3]. As is observed, the R/B is almost constant throughout showing a good performance and that there is no significant increase of the fuel failure fraction.

The fuel behavior under accidental conditions has also been investigated by constant- and ramp-temperature heating tests on the irradiated coated particles [4].

3. Nuclear design

The flow of the HTTR nuclear calculation is shown in Fig.6

schematically; a computer code, DELIGHT is used to obtain the neutron spectrum of a fuel cell and to produce group constants based on the nuclear data from ENDF/B-3, -4. The calculation of a control rod cell is performed by the TWOTRAN-2 which is based on the two-dimensional transport theory. CITATION-1000VP, which is a vectorized version of the CITATION [5] is used to calculate the three dimensional core performance.

Accuracy of the nuclear codes has been examined by various experimental data obtained using the Semi-Homogeneous Experimental Assembly (SHE) [6] and the Very High Temperature Reactor Critical Assembly (VHTRC) which was reconstructed from the SHE. Main specifications of VHTRC are shown in Table 3.

A schematic view of the VHTRC is shown in Fig.7. The VHTRC is composed of a movable half and a fixed half both are made up of hexiagonal graphite blocks in the horizontal position and inserted in holes of blocks are fuel rods, control rods, safety rods and heaters. A fuel rod contains fuel compacts in which TRISO coated particles are dispersed as shown Section 2.

Major items of code verification are

a) the effective multiplication factor,

b) the control rod reactivity worth,

c) the burnable poison rod reactivity worth,

d) the power distribution and

e) the temperature coefficient.

As an example, a comparison of the effective multiplication factor obtained by the above codes and experiments in the VHTRC is shown in Fig.8 in which 2, 4 and 6 % enriched fuels are combined with effective core diameter ranging from 104cm to 140cm. The calculated results are in very good agreement with the experimental results.

The errors estimated by these comparisons between experimental and calculated results are reflected in the nuclear design of the HTTR.

4. Thermal-hydraulic design

As pointed out in Section 2, in the HTTR safety criteria, the maximum fuel temperature is limited under the normal operation and the abnormal transients condition. Thus, the core should be designed so as to maintain the sufficient core flow rate and to keep the maximum fuel temperature as low as possible, with the structural features of the core and fuel are taken into account. The flow of the thermal-hydraulic design is shown in Fig.9. Flow and temperature distributions of coolant in the core are calculated by the FLOWNET in which the flow network model is employed. A flow network consists of branches and nodes which simulate various flow paths in the core such as main flows, leakage flows through the gaps between the permanent reflectors and between fuel blocks. Based on the result of the FLOWNET and the power distribution obtained by the nuclear calculation, the temperature of the fuel is estimated by the TEMDIM code, in which compacts and a graphite sleeve of a fuel rod are modeled by coaxial cylinders and the maximum fuel temperature is calculated with systematic and random errors considered.

Various R&D programs have been performed to verify the thermal-hydraulic design. Appearing below is a research program in which the leakage flow rate within the core support structure is estimated using the Helium Engineering Demonstration Loop (HENDEL) and the experimental results are compared with the analytical ones of the FLOWNET.

The HENDEL was constructed to perform large scale demonstration tests of high-temperature components for the HTTR in March 1982. A schematic flow diagram of the HENDEL is shown in Fig.10. The HENDEL consists of the Mother (M), Adapter (A) and Test (T) sections. The Mother and Adapter (M+A) section circulates helium gas at a flow rate of 4 kg/s, a pressure of 4 MPa and at a maximum temperature of 1000°C. The M+A section has been operated for more than 10,000 hours since 1982. The test section is made up of the Fuel Stack Test Section (T_1 test section)[7,8] and the In-core Structure Test Section (T_2 test section)[9]. A general view of core bottom structure of the T_2 test section is shown in Fig.11. The T_1 test section has been in operation since March 1983 and the T_2 test section since June 1986 and various data have been obtained in the fields of thermal-hydraulic and high temperature structural design.

Shown in Fig.12 are experimental results of leakage flow rate from outside of the permanent reflector blocks into the hot plenum of the T_2 test section as a function of the pressure difference between the hot plenum and the outer side of the permanent blocks at inner and outer temperatures 950°C and 400°C, respectively. The analytical results obtained by the FLOWNET are shown also and an excellent agreement between experimental and calculated results is obtained.

We note references [7-10] for the results of R&D in the field of the thermal-hydraulic design.

5. Graphite structure design

The reactor core is composed of fuel blocks, control rod guide blocks and replaceable reflector blocks and is supported by the core support structures of graphite and metal. They are bound by the core restraint mechanism as shown in Fig.2.

A fuel block is graphite hexagonal right prism as shown in Fig.3 and is made of the IG-110, isotropic fine-grade graphite. Control rod guide blocks and replaceable reflector blocks have the same external shape as the fuel blocks and are also made of the IG-110.

The graphite core support structure is made up of permanent reflector blocks, hot plenum blocks, seals, keys, support posts and the thermal insulation blocks. They are made of the PGX, structural grade medium-to-fine grained molded graphite except seals, keys and support posts which are made of IG-110 and the middle layer of the insulation which is made of carbon.

In order to assess the integrity of the above graphite structures, a design code has been developed and design data have been accumulated on the IG-110 .

In the code the graphite components are categorized into core components and core support components and different allowances are adopted in view of the differences of their functions as summarized in Table 4.

The design code is based mainly on the ASME CE Code, however, the ASME CE Code is modified regarding the bi-axes failure theory, buckling limit and oxidation effects[11]. A comparison between the JAERI code and the ASME code is given in Table 5.

The strength of oxidized graphite, for example, is specified in the code as follows:

The region where amount of oxidation exceeds the 80% burnoff should be deemed as having no load carrying capacity and for the region where the burnoff is below 80%, the allowance is calculated based on the reduced strength which is obtained by experiments. The tensile strength decrease of grade IG-110 is shown as a function of burnoff in Fig.13 [12].

In order to develop the design criteria and their allowables, the research work has been carried out on including the high-temperature Young's modulus, impact strength, fracture mechanics properties, low cycle fatigue life, irradiation creep properties and so on. We note ref.[11] for details.

6. High temperature metal structure

High temperature metal structure is also one of major subjects of R&D for HTGRs. In the HTTR, a He/He intermediate heat exchanger (IHX) of 10MWt is used as shown in Fig.1 and the heat tubes and the central gas duct of the IHX constitute a part of the pressure boundary of the primary coolant at a temperature of about 900°C. A bird's eye of the IHX is shown in Fig.14. Hastelloy XR[13], which is a modified version of Hastelloy X, is used for the very-high temperature structures in the IHX.

A structural design code for the HTTR was developed by the JAERI based on both the Elevated-Temperature Structural Design Guide for Monju (ETSDG)[14] and the ASME Code Case N-47 and tests for creep, fatigue, fracture toughness, corrosion and other critical items have been undertaken to accumulate design data of Hastelloy XR [15].

Component tests of the Intermediate Heat Exchanger (IHX) have been also performed.

In the IHX, the pressure of both primary and secondary gas is about 4 MPa and their pressure difference is very small in the normal condition. If the secondary gas were lost by an accident, however, the tubes and header would be subjected to the external primary gas of 4 MPa and the possibility of the creep collapse would arise. Therefore, the experiments have been and being performed to evaluated the integrity of the tubes subjected to external pressure [16] and a simplified method for the prediction has also developed [17].

Another important potential failure mode of the IHX is the creep-fatigue failure induced by cyclic relative displacement between the hot header and tubes due to the difference of thermal expansion.

In order to verify the design criteria for the creep-fatigue in the structural scale, a IHX structural model has been made and the test apparatus will be completed in July 1989. The model consists of a hot header, 8 helical tubes and 8 connecting tubes, which are installed in an electrically heated retort. The test apparatus is shown in Fig.15. Each connecting tube is attached to the hot header in one end horizontally and the other end is extended to the upper loading grid vertically. These tubes are subjected to both repeated loading by the actuater and an internal pressure at a temperature of 950 °C. Deformation and the life obtained by the experiment will be compared with predictions by inelastic analysis and the JAERI code.

7. Concluding Remarks

Some of R&D results were provided in the field of fuel, nuclear and thermal-hydraulic design, graphite and high temperature structures in connection with the HTTR design. R&D of the HTGR, however, extends over far wider range than outlined in the above. An overview of the HTGR R&D program may be found in the annual report of JAERI [18].

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Table 1 Major design parameters of the HTTR

Thermal power	30 MW
Outlet coolant temperature	850°C/950°C
Inlet coolant temperature	395°C
Fuel	Low enriched UO_2
Fuel element type	Prismatic block
Direction of coolant flow	Downward-flow
Pressure vessel	Steel
Number of main cooling loop	1
Heat removal	IHX and PWC (parallel loaded)
Primary coolant pressure	4 MPa
Containment type	Steel containment
Plant lifetime	20 years

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Table 2 Irradiation test conditions and results [3]

Fuel Assem. Number	Number of compacts	Fast neutron fluence (Max.) [n/m², E>29fJ]	Burnup [MWd/t]	Fuel temp. (°C)		Coated particle failure fraction	
				Max.	Ave.	after irr.	before irr.
1	54	1.2×10^{24}	5,800	1490	1190	2.7×10^{-5}	3.7×10^{-6}
2	60	2.0×10^{24}	8,500	1520	1220	1.4×10^{-5}	1.0×10^{-4}
3	20	8.9×10^{23}	4,800	1370	1240	7.1×10^{-4}	8.5×10^{-4}
4	60	2.3×10^{24}	18,000	1400	1120	7.3×10^{-5}	1.1×10^{-4}
5	60	3.8×10^{24}	30,000	1410	1230	3.1×10^{-3}	1.9×10^{-3}
6	20	4.1×10^{23}	3,900	1530	1410	9.6×10^{-5}	5.0×10^{-5}
7	60	1.6×10^{24}	12,000	1430	1230	7.1×10^{-5}	2.7×10^{-5}
8	20	1.2×10^{24}	9,100	1440	1300	1.1×10^{-4}	5.9×10^{-6}
9*	60	2.8×10^{24}	24,000	1390	1260	1.4×10^{-3}	8.7×10^{-4}
10*	20	———	26,000	1550	1280	4.9×10^{-4}	2.5×10^{-4}

* fabricated by scale-up facilities

Table 3 Main specification of the VHTRC

Items	Specifications
Core Shape	Horizontal Hexagonal Prism Sprit in Two Half Assemblies on Tables
Core Dimensions	Core Across, 2.4 m Axial Length, 2.4 m
Maximum Thermal Power Output	10 W
Maximum Core Temperature by Electric Heating	210°C (800°C for Single Fuel Rod)
Fuel Element	
Type	Fuel Compact of Coated Particles Dispersed in Graphite Matrix
Enrichment	2,4 and 6 wt%
Maximum U-235 Loading	10.4 kg
Moderator and Reflector	Hexagonal Prism Graphite Blocks
Control Rod	Cd Cylinder Sheathed with SUS Motor and Pneumatic Drive, 2 sets
Safety Rod	Cd Cylinder Sheathed with SUS Pneumatic Drive, 6 sets
Instrumentation	Six Neutron and One Gamma Ray Monitors

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Table 4 Comparison between the core and the core support components of the HTTR

	Core component	Core support component
Main component	• Fuel block • Graphite sleeve • Control rod guide block • Replaceable reflector block	• Hot plenum block • Permanent reflector block • Core support floor thermal insulation layer • Support post
Replaceability Irradiation effects Design life	Routine Major 3y	Difficult Negligible 20y

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Table 5 Comparison between JAERI Design Code and ASME CE Code

Items		JAERI's criteria	ASME CE Code
Core support component	Failure theory	Maximum principal stress + modified Coulomb-Mohr theory	Maximum principal stress theory
	Buckling limits	Rankin-Gorden type	Karman type
	Pure shear stress limits	Considered	Not considered
	Oxidation effects	Considered	Not specified
	Quality control	Specified	Not completed
	Stress evaluation method Stress category Safety factor Mimimum ultimate strength	Same for both	
Core component		Fundamental concept is same as the core support component with some exceptions (safety factor, irradiation effects ... etc.)	Not specified

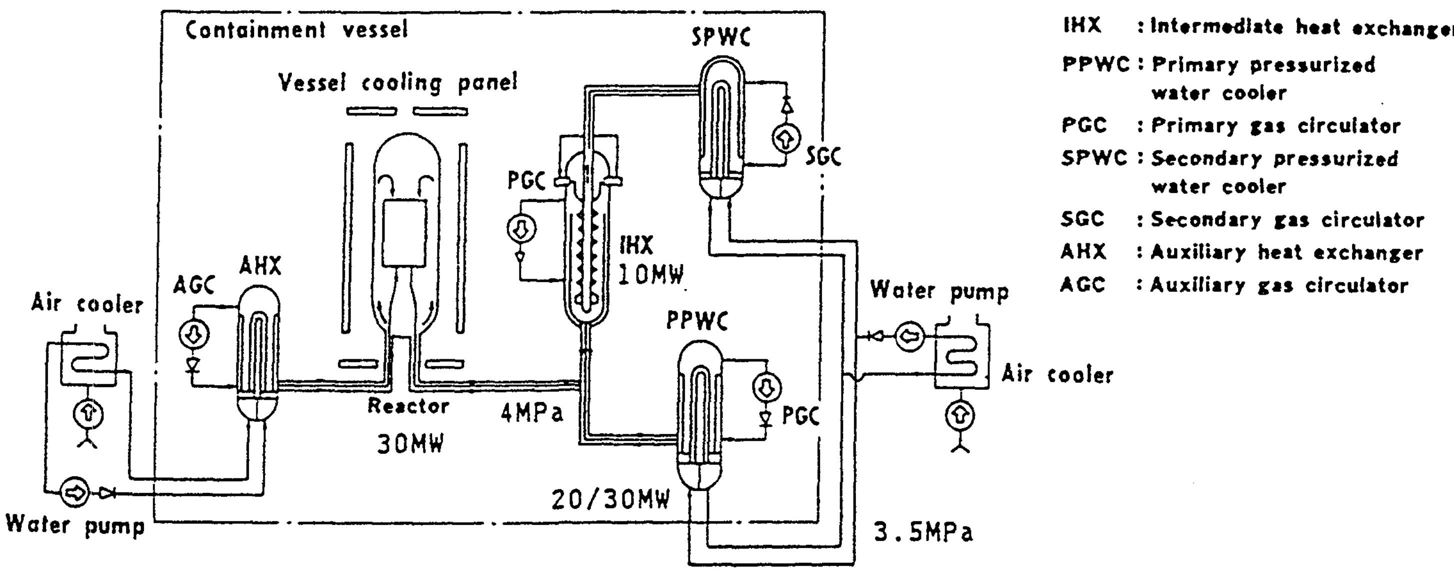

Fig.1 Cooling system of the HTTR

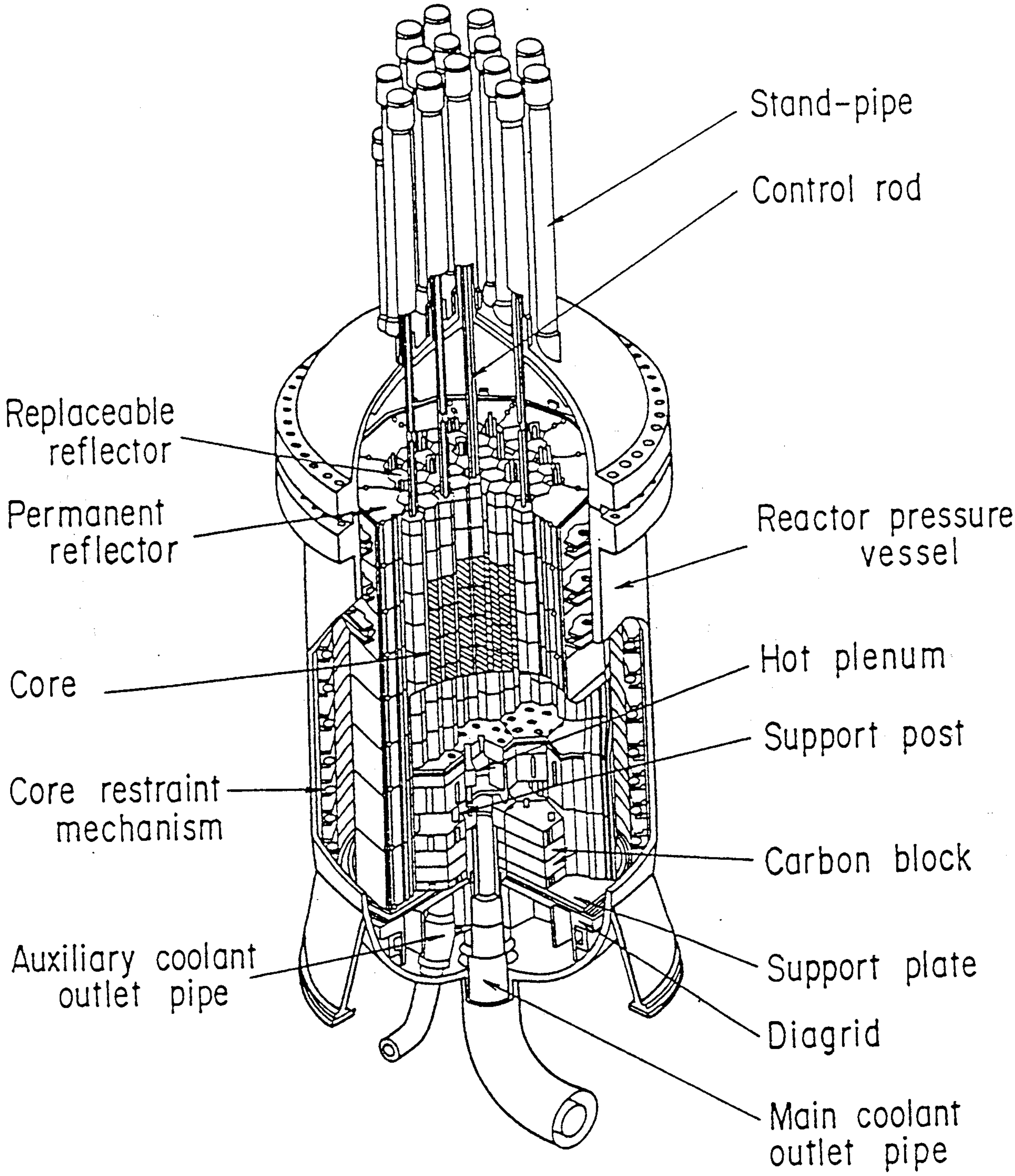

Fig.2 Bird's-eye view of the reactor vessel and core

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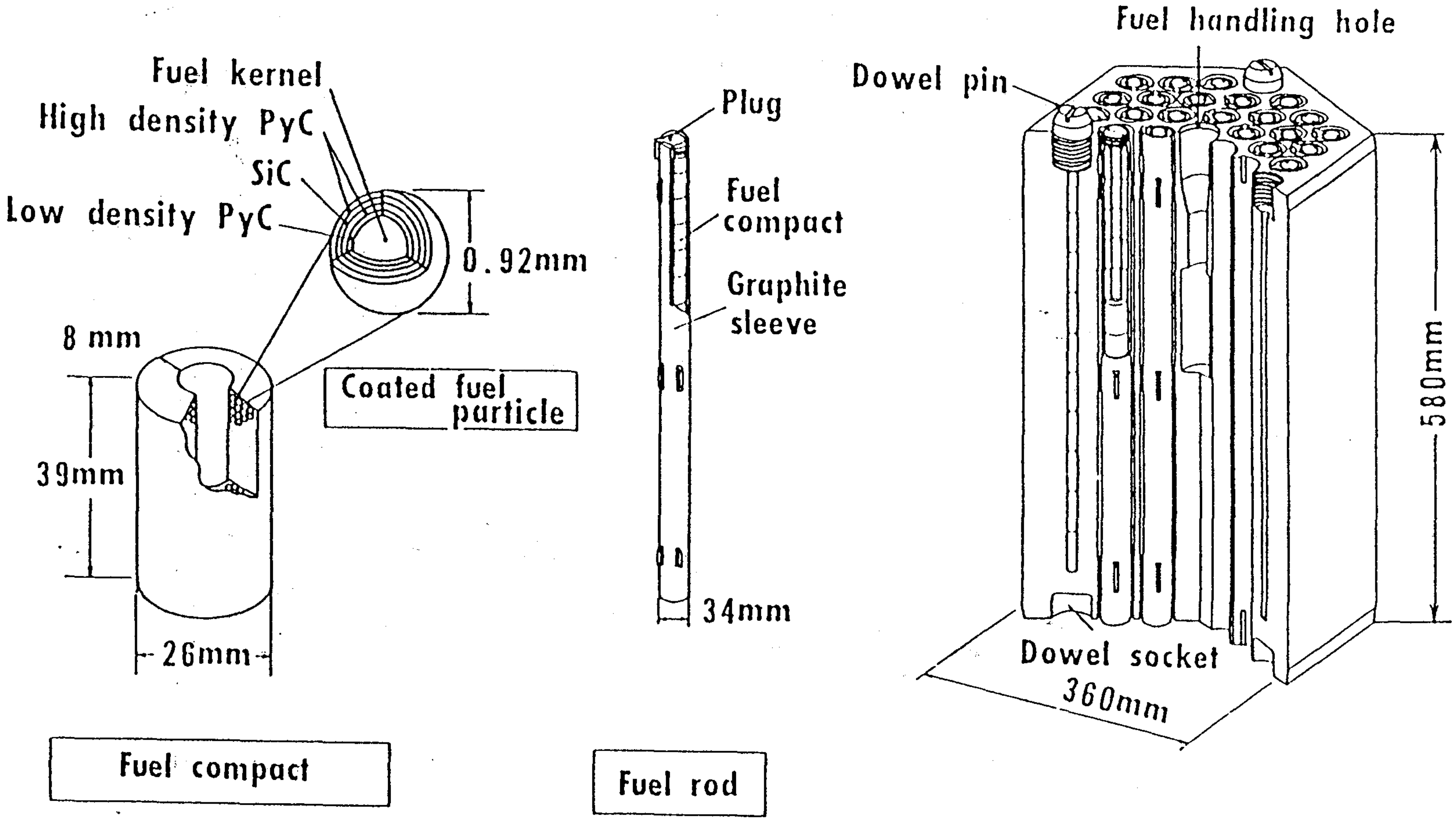

Fig.3 Block type fuel of the HTTR

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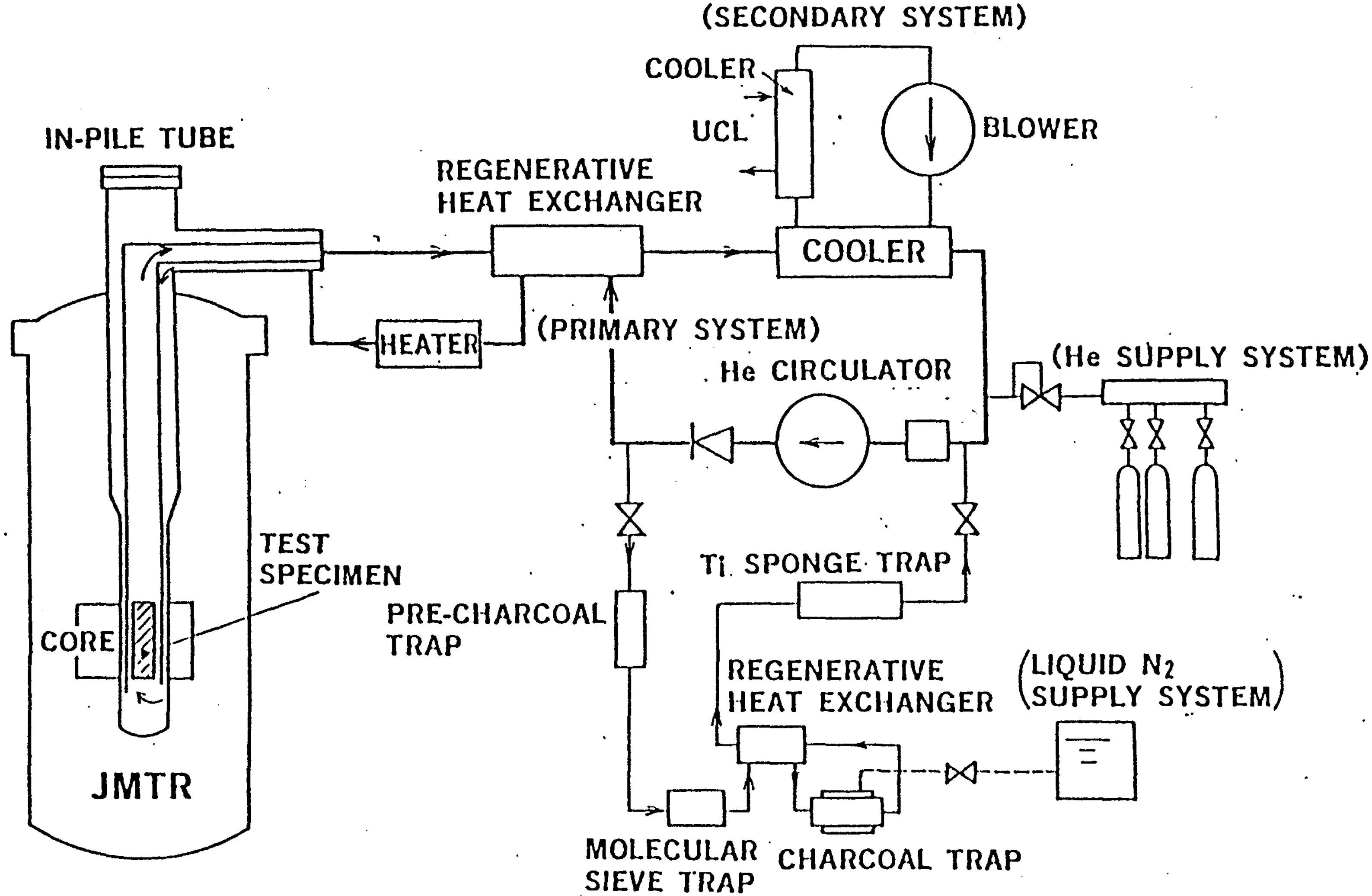

Fig.4 Flow diagram of the OGL-1

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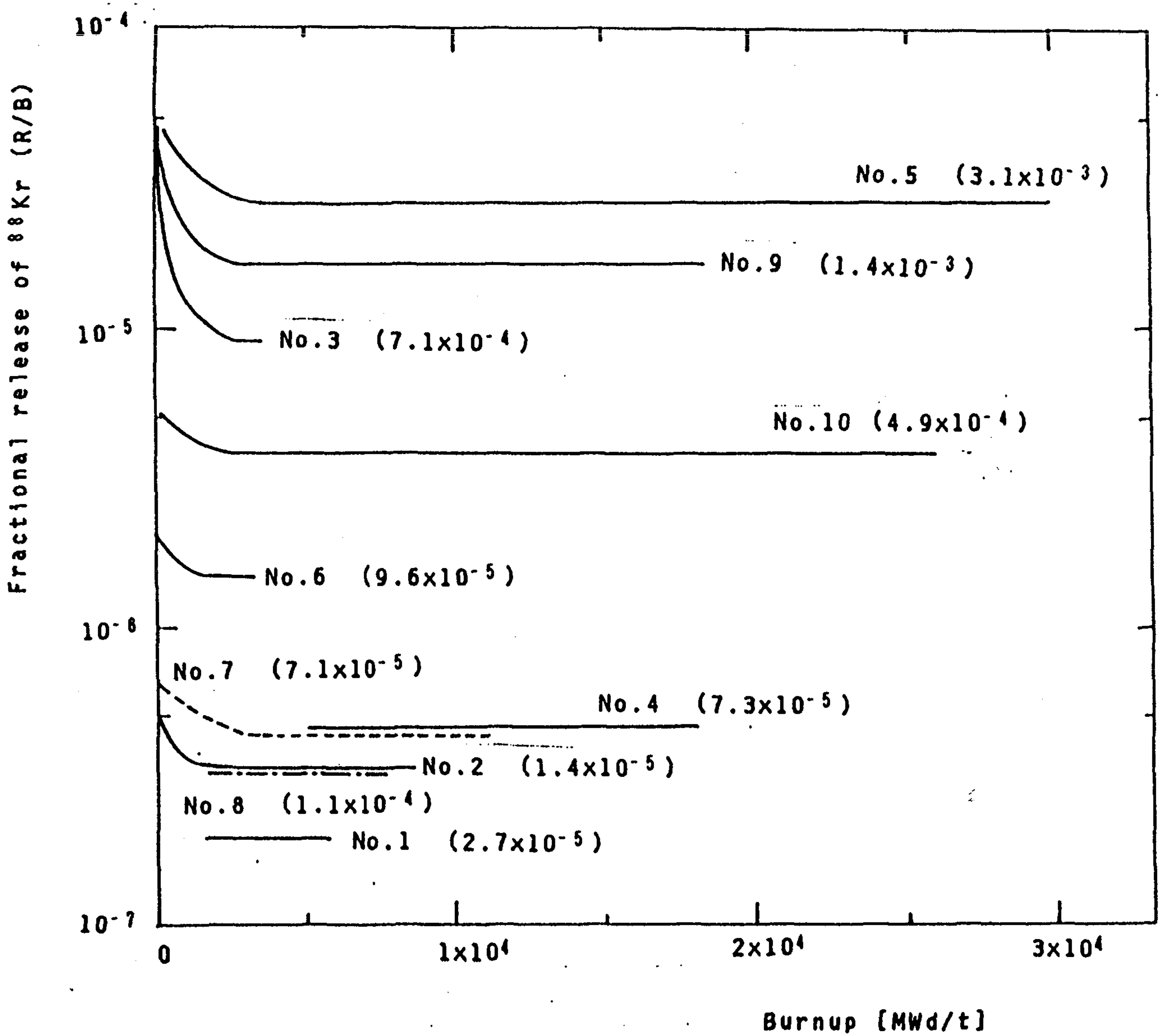
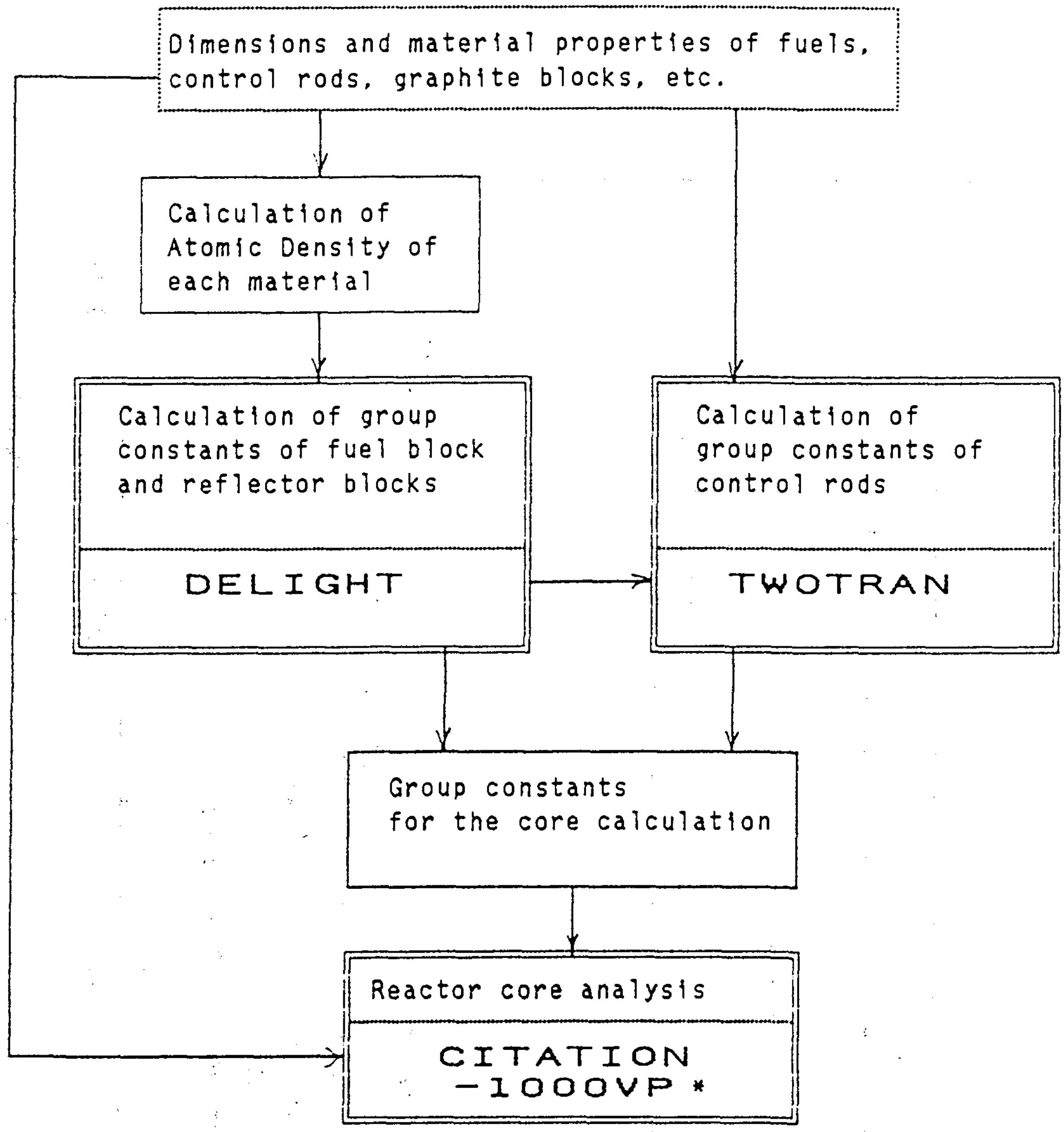


Fig.5 R/B of ^{88}Kr during irradiation. Figures in parentheses show failure fraction after irradiation [3].



* Vectorized version of the CITATION

Fig.6 Flow of the HTTR nuclear design

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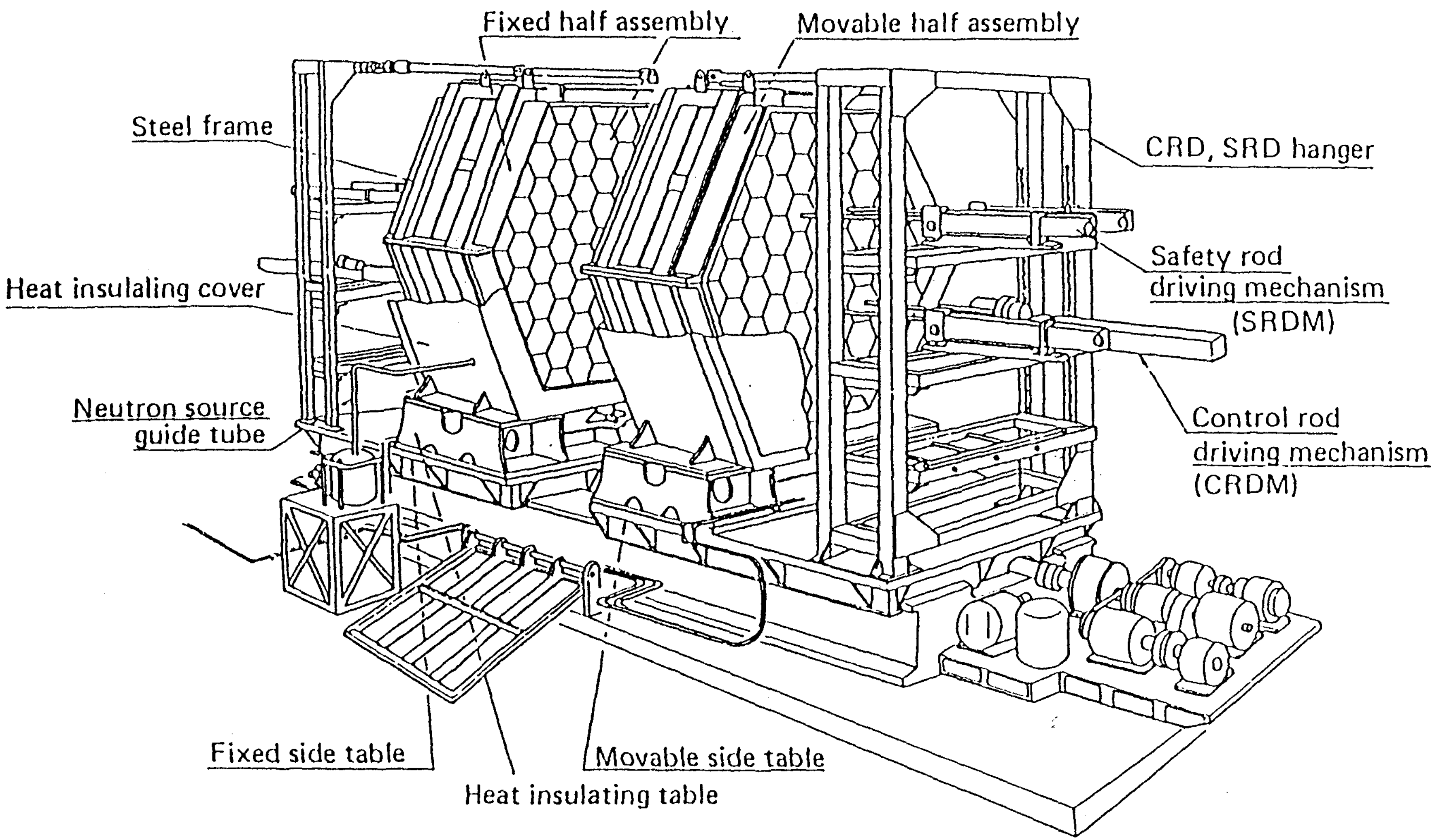

Fig.7 Bird's-eye view of the VHTRC

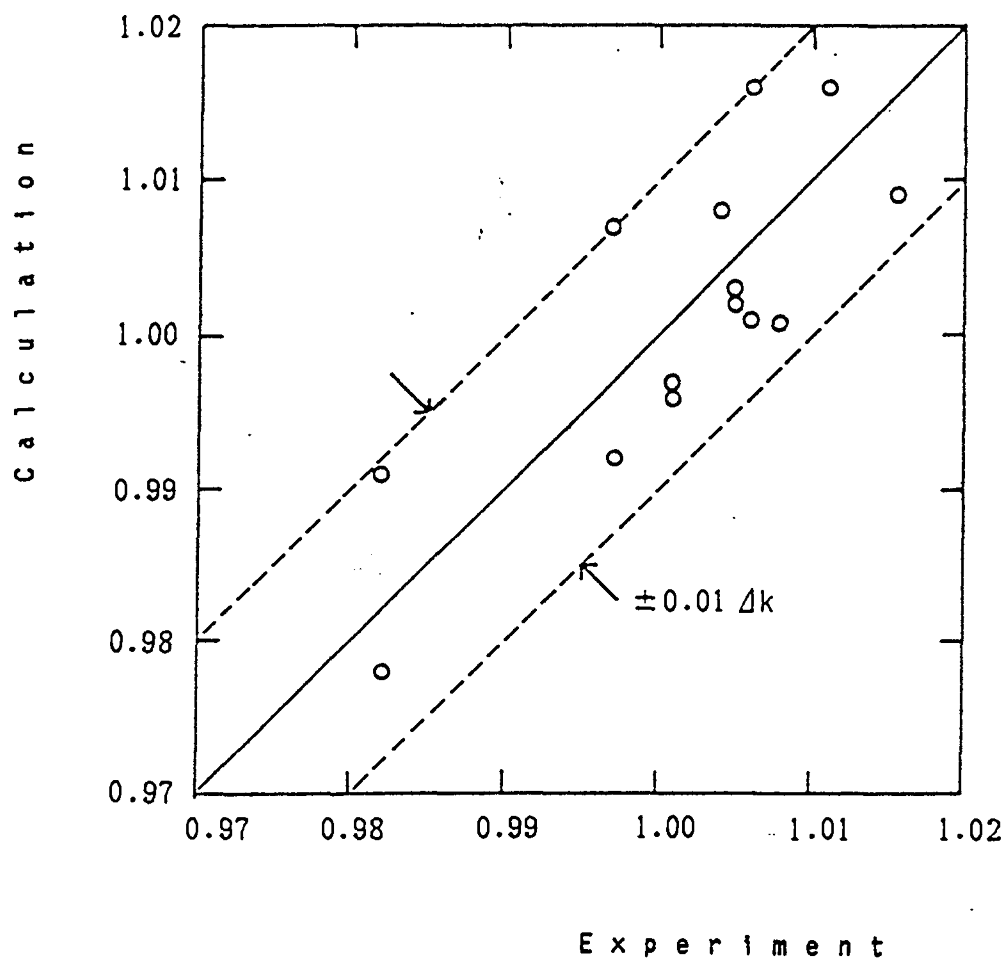

Fig.8 Effective multiplication factor. Camparison between experiment and calculation.

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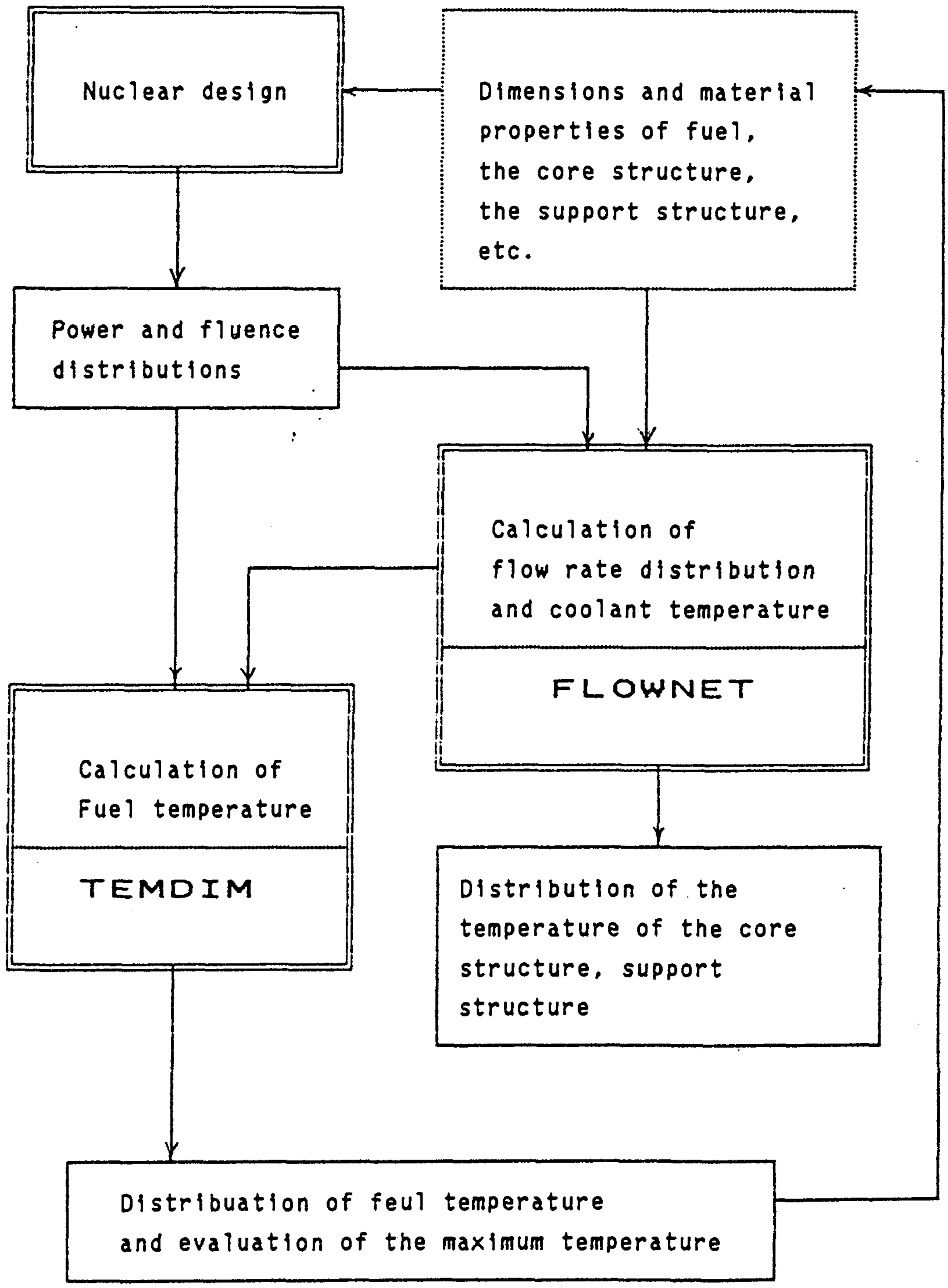

Fig.9 Flow of the HTTR thermal-hydraulic design

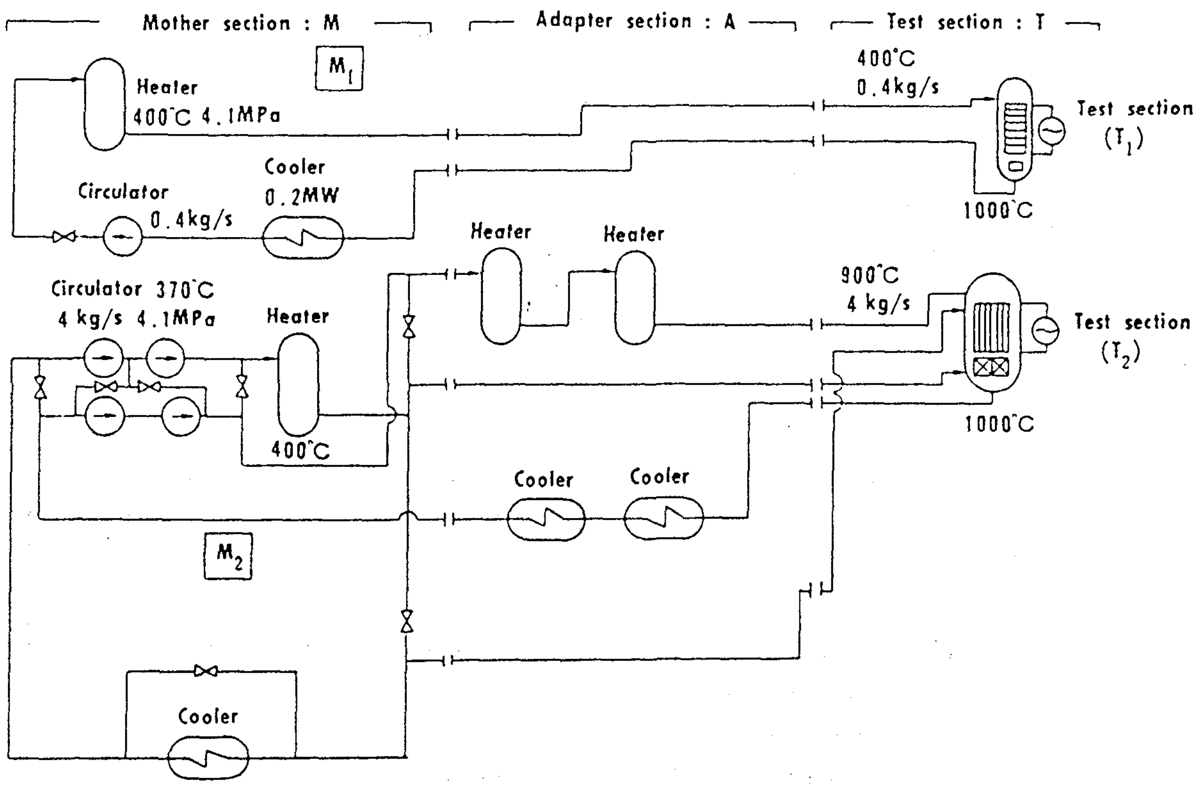

Fig.10 Flow diagram of the HENDEL

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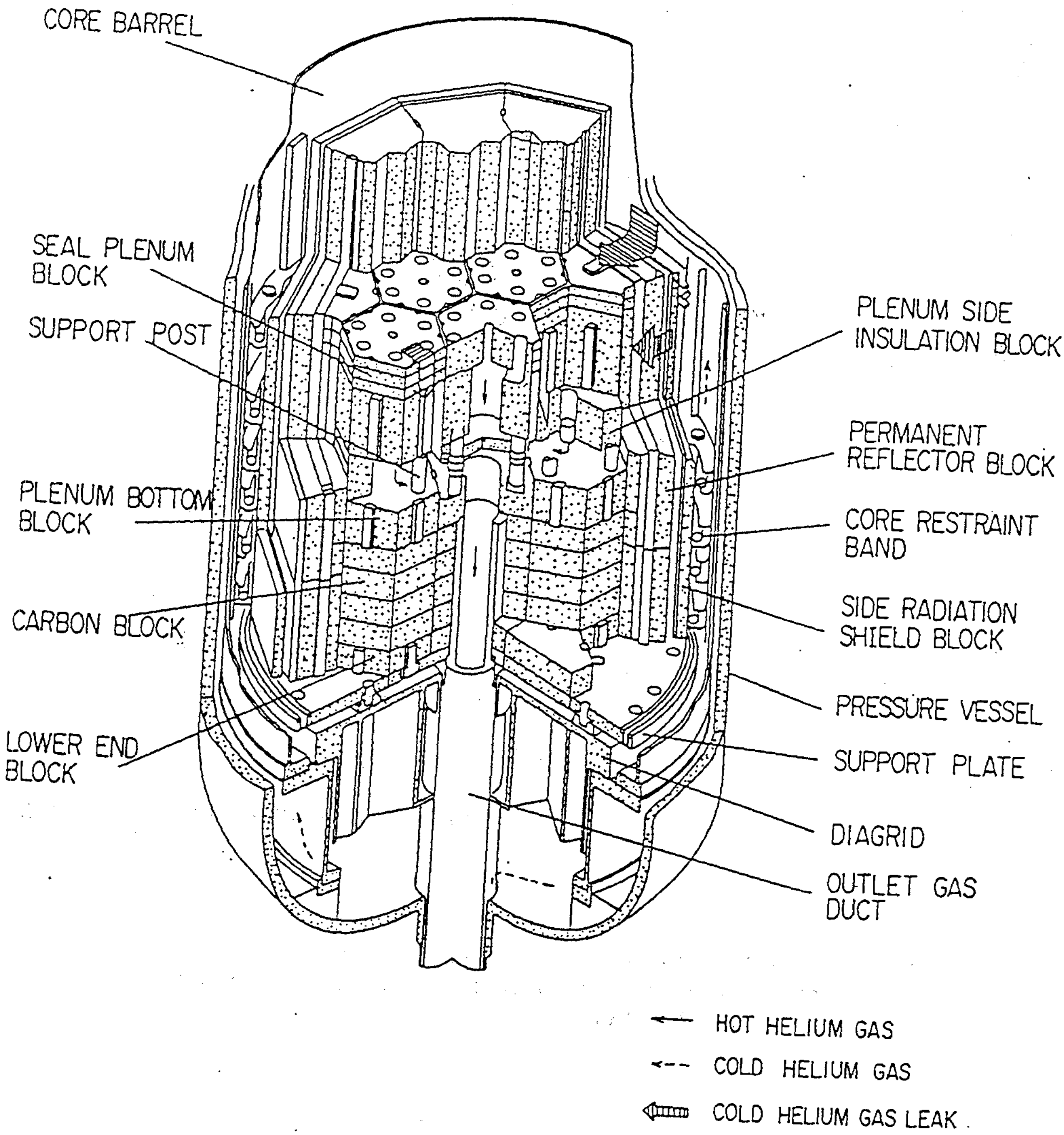

Fig.11 Bird's-eye view of the core bottom structure of T_2 test section

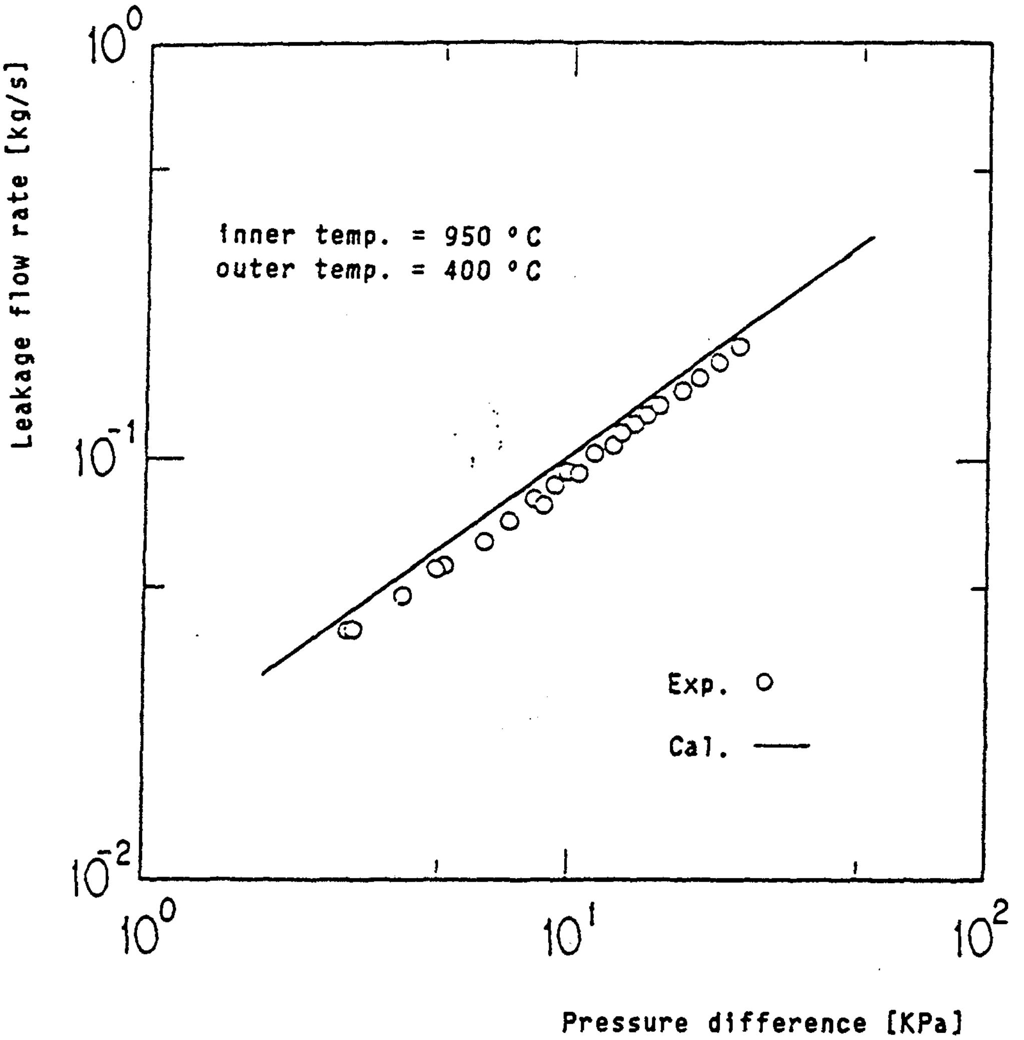

Fig.12 Relationship between leakage flow rate and pressure difference between inner and outer of the permanent reflector blocks.

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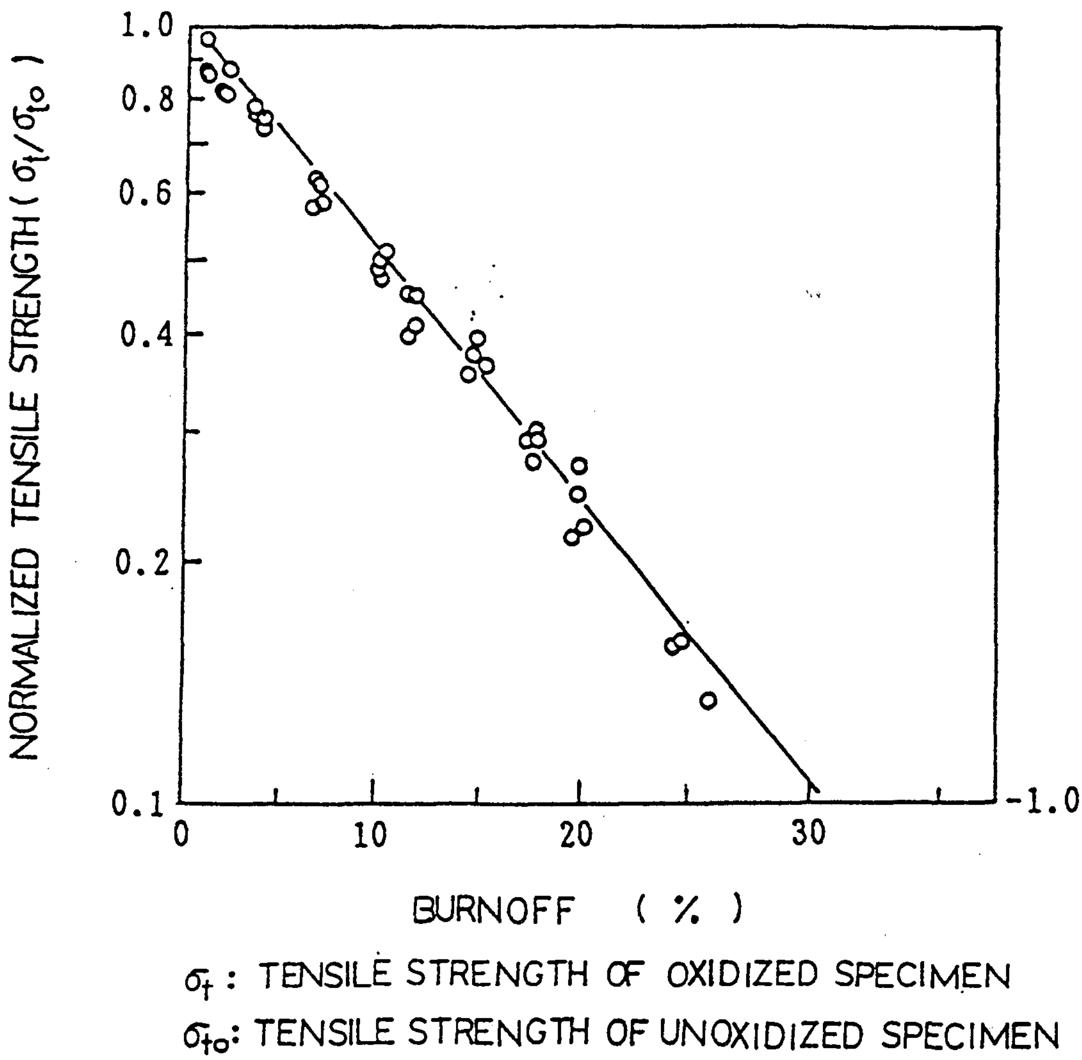

Fig.13 Dependence of strength on burnoff in uniformly oxidized IG-110 graphite

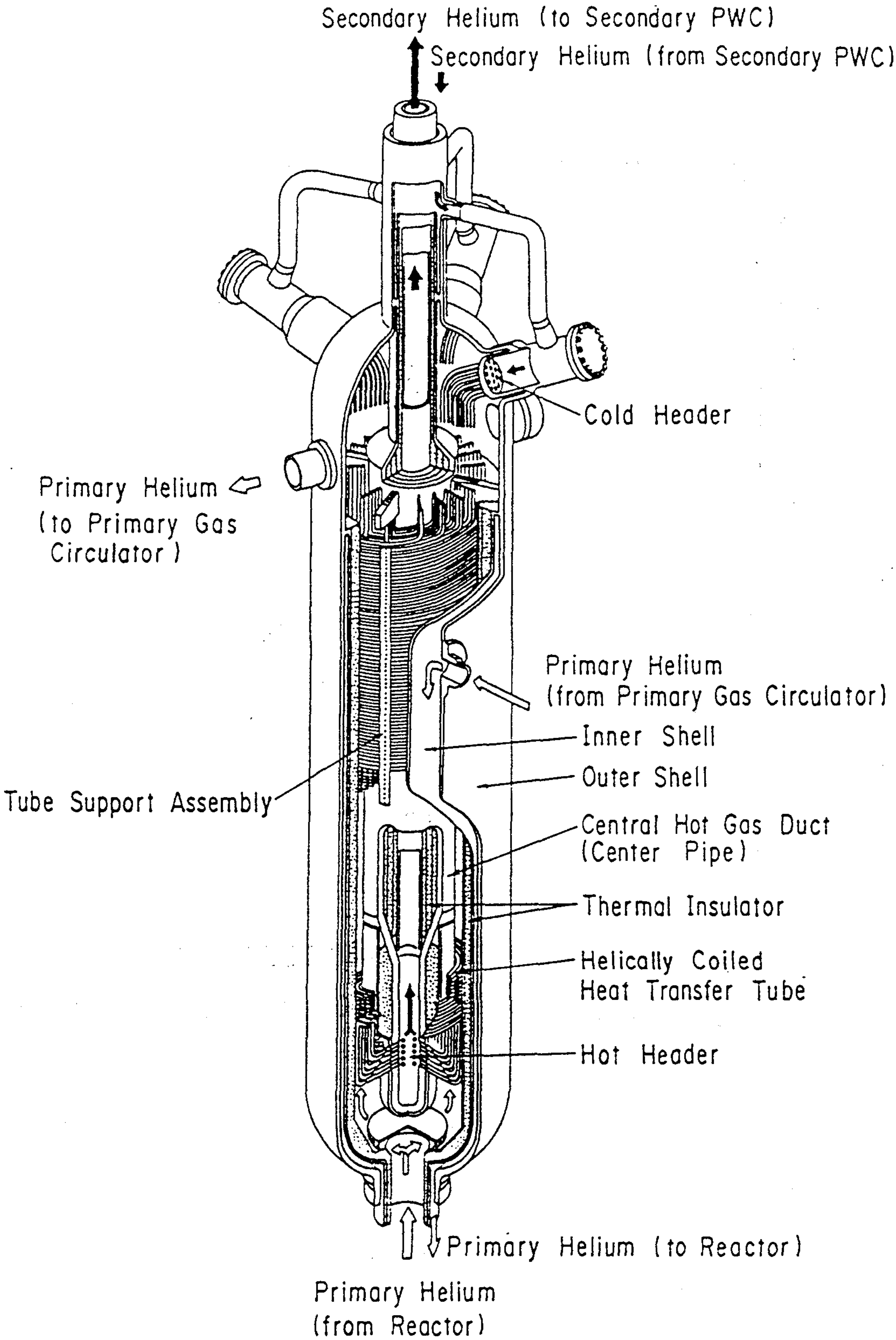

Fig.14 Bird's-eye view of the IHX

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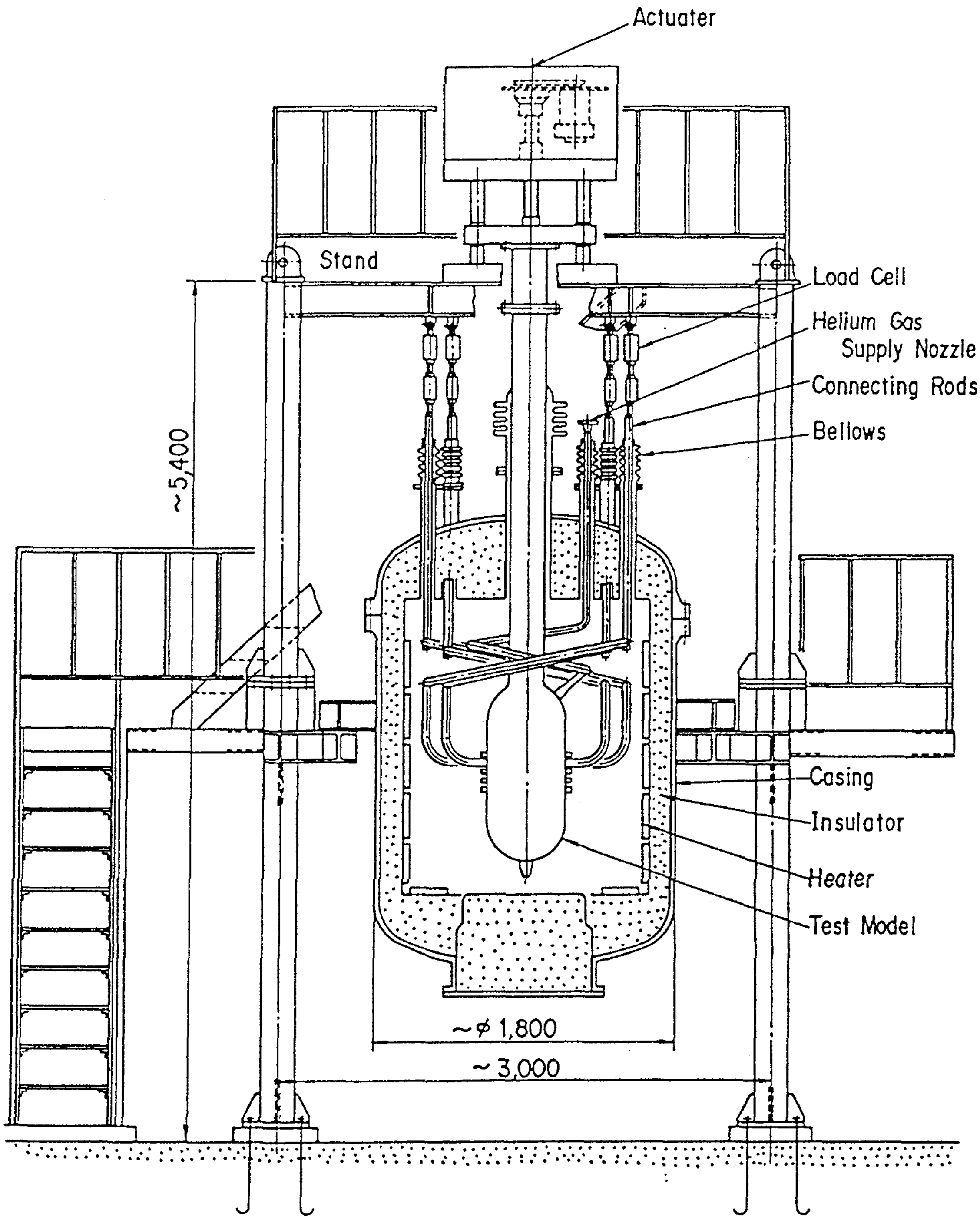

Fig.15 Test apparatus for the IHX structural test