Status of Fast Reactor Technology in China

For the Twenty-Fifth Annual Meeting of the International Working Group on Fast Reactor, IAEA

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April 27-30 1992, VIENNA

Abstract
The paper has introduced briefly the recent news about Chinese nuclear program on PWR and FBR. Concerning the FFR design, some issues under consideration have been presented including the matches between thermo-parameters of primary sodium and of steam, the arrangement of control and safety rods which correspond to first and second shut-down systems, the structure of inner vessel and the axial length of subassembly. With regard to the R&D of FBR technology, some results on sodium technology and on the cladding materials have been given in the paper. Finally some progress and troubles on site selection for this reactor have also been outlined.

1. Introduction

The following factors stimulate China to develop the nuclear power technology:

.the annual primary energy consumption per capita in China recently only about a half of the average in the world, thus energy industries have to be developed steady during a long time, requested obviously by the enormous population and economy modernization;

.uneven distribution of general energy resources compared with the needs by the industries and population;

.the pollution problem by burning coal has caused public concern. The acid rain already appeared in many areas. Some big cities have already had serious atmospheric pollution caused by burning coal.

China has started developing her nuclear power technology with following general purposes:

.up to the early of next century, nuclear power will play in the country a supplementary or replacement role in the regions lack of general energy resources;

.in the middle of next century, nuclear power will be the important part of total electricity production in China.

To implement these targets China has decided adopting thermal reactor PWRs matched with FBRs in the future as the primary strategy of the Chinese nuclear development.

In the end of the year 1991, Qin Shan-1 300 MWe PWR has connected to the electricity grid. Da-Ya Bay twin 900 MWe PWRs will be completed in the year 1993 and 1994 respectively. Qin Shan-2 twin 600 MWe PWRs project has been approved by the Government, which is in the design stage.

Since 1987 up to now the development of FBR technology is pursued under the framework of the State High Technology Program in China. In the last month the State Council has made the authorization for the FFR project which is 65 MWth and 25 MWe experimental fast reactor. It is planed that the fabrication of the components and facilities and its architecture engineering will be started in the year 1996. The criticality of the FFR is envisaged in the year 2000.

2. FFR design

The conceptual design of the FFR has been basically completed in the end of 1990, which was introduced at the last meeting[1], but of some auxiliary systems will be completed in the end of this year.

2.1 Review of Conceptual Design

Since last meeting we have done some fallen back into review on the conceptual design. The following main points are considered to be rational and to be kept to the preliminary design:

.design demands

--main technical selection should be consistent with the trends of the development of the FBR technology in the world;

--the basic thermo-parameters of commercial fast reactors should be adopted;

--the design is requested to have a self-stability reactor core and passive decay heat removal system;

--the technology used in the FFR must be proved either by our own laboratory or by others;

--systems and components should be striven to be simplified so that to get high reliability, due to which it will be easier to transmit from the FFR to next step.

.main technical selections and design boundary conditions for the FFR[1]

.main design parameters[1]

2.2 Some issues under consideration

2.2.1 Different arrangements of control rods

Three different arrangements of control rods have been considered, which are shown in Figs 1-3 and their main parameters are compared in Table 1. The preliminary evaluation pointed out that the values of safety rods and control rods in the three schemes, they all meet respectively the design criteria for the FFR. MK-II has better accessibility to each control rods mechanism, which will facilitate to its maintenance, but the linear power a little lower than MK-I and MK-I*. Considering the higher linear power of fuel pins and the larger shut-down margin, MK-I* has been selected.

Table 1 Three core arrangements

Scheme	MK-I	MK-I*	MK-II
no.of fuel subassemblies	82	82	85
fuel roading Pu / Pu-239 / U-235(kg)	121.6 / 93.2 / 97.6	121.6 / 93.2 / 97.6	126.0 / 96.6 / 101.2
no. of safety rods	2	2	2
no. of control rods	7	7	7
value of safety rods (%ΔK / K)	2×2.015	2×2.015	2×1.615
value of B.U. compensation rods(%ΔK / K)	3×0.487	3×0.970	3×1.567
value of Temp. compensation rods(%ΔK / K)	1×2.010	1×2.010	1×1.470
value of coarse regulation rods(%ΔK / K)	2×0.300	2×0.300	2×0.420
value of fine regulation rods(%ΔK / K)	1×0.220	1×0.220	1×0.420
total amount (%ΔK / K)	8.81	10.90	10.46
excess reactivity (%ΔK / K)	5.08	5.08	6.12
Margin of shut-down (%ΔK / K)	3.73	5.82	4.34

2.2.2 Temperature difference at pinch point of steam generators

Based on the conceptual design of FFR, the temperature parameters of the primary and secondary sodium, and of water-steam are listed as follows:

1,IHX primary sodium temperature inlet/outlet 525/385,°C
2,IHX secondary sodium temperature inlet/outlet 325/505,°C
3,Evaporator sodium temperature inlet/outlet 456/325,°C
4,Evaporator water temperature inlet/outlet 225/319,°C
5,Superheater sodium temperature inlet/outlet 500/456,°C
6,Superheater steam temperature inlet/outlet 319/480,°C

The coolant temperature distributions of primary, secondary and of water-steam are shown in Fig.4. The temperature difference at the pinch point in SG reaches 456-319=137 °C, much higher than usual temperature difference applied. The solution intended for this issue would be raise the tertiary circuit pressure and thus the saturation temperature of water.

2.2.3 Inner vessel

The 3D-thermo-hydraulic computer codes for flow distribution and temperature profile of primary sodium are under preparation, so the thermo-mechanic calculation hasn't started. It is tried to have an inner vessel with a near almost symmetry structure to facilitate the compensation of thermo-deformation of the structure.

2.2.4 Fuel Subassembly

The model of the FFR fuel subassembly has been successfully fabricated according to the conceptual design. It is found that the subassembly foot of about 300 mm is rather short compared with the total length of 2600 mm and it will be unstable when it stands onto the lattice plenum. It is also estimated the flow will be more uneven in the plenum chamber. So we will longer this foot up to about half meters, meanwhile we intend to shorter the total length by cancelling some shielding section to obtain good stiffness of the subassembly.

3. Research and development

During past years, the R&D of FBR technology have been emphasized on sodium technology, materials, fuels and safety studies, here only first two parts are touched.

3.1 Sodium Technology

Taking experiences abroad as reference and based on our own experiences of interactions between materials and sodium with some impurities. The quality standard for reactor sodium has been preliminarily decided as follows:

C	$<$	30 ppm
O	$<$	30 ppm
B	$<$	5 ppm
Cd	$<$	5 ppm
Halogen	$<$	30 ppm
Ca	$<$	10 ppm
Li	$<$	10 ppm
K	$<$	200-1000 ppm

This standard of sodium will be used for material testing sodium loops.

The industrial sodium which is produced from melt $NaCl-CaCl_2$ compounds by electrolysis contains the calcium contents of up to 400-500 ppm which is not suitable for fast reactor application. The first part of the multipurpose sodium purification loop (MSPL) [1] is used for de-calcium by adding NaO_2 to produce CaO, then which is filtrated at low temperature. By this method we could get sodium with calcium lower than 5 ppm [2]. Some sodium loops built during past years have been filled with sodium in our reactor grade, produced by the MSPL. The de-calcium system is shown in Fig 5 and 6.

It is intended to design sodium purification loop for the FFR by enlargement in scale based on MSPL.

3.2 Materials

In Chinese material market there is 316 SS on sale, but it is not in reactor grade and especially not suitable for fast reactor core materials. The cladding tubes of 316 Ti SS for FFR pins have been tri-fabricated in our laboratory, using twin vacuum smelting and fine rolling technology. The composition, thermo-characteristics, crystal grain size, mechanical properties (creep data have not been obtained yet) of the material and dimension, tolerance and surface quality of the cladding tube are all satisfactory to design needs.

The further demonstration for 316 Ti SS will be emphasized on the interaction properties with sodium and irradiation performances.

The following facilities for material demonstration have been almost ready to accept material specimen:

.material corrosion sodium loop
.mass transfer sodium loop
.stress corrosion sodium loop
.fatigue and creep sodium loop
.bi-axial creep test facility (no sodium) and
.cladding-fisssion products interaction test facility (out of reactor)

The pre-experiments out of reactor on cladding-fission produces interaction have been done, using the facility above-mentioned, to study the interaction between cladding and fission products simulated by Cs,I,Te,Se and by different oxygen potential with Cr/Cr_2O_3 (lower oxygen potential, corresponding to O/M 1.96) and Ni/NiO (higher oxygen potential, corresponding to O/M =2.00). The cladding materials tested are presented in Table 2.

Table 2 Cladding materials tested

specimen	CW%	contents										
		C	Cr	Ni	Mo	Mn	Si	Ti	P	S	B	Fe
1	15	0.07	14.97	15.23	1.20	0.73	0.68	0.45	–	–	0.006	balance
2	15	0.066	17.08	12.75	2.25	1.37	0.67	0.68	0.028	0.016	–	balance
3–1	30	0.040	16.8	12.5	3.17	1.67	0.73	0.63	0.016	0.0078	–	balance
3–2	29.7	0.042	16.8	12.8	3.18	1.71	0.72	0.63	0.016	0.0078	–	balance

It's found from the experiments that for different materials tested the maximum interaction depth is about 28 um and non-obvious attack under lower oxygen potential, but contrary, the deep intergranular corrosion and spalling section layer were observed and maximum attack depth is about 100 um under higher oxygen potential. In the experiments the isothermic test temperature is 700 ^{o}C and test duration 120 h with the contents Cs,I,Ts,Se corresponding to the burn-up of 100 MWd/Kg[3].

4. Feasibility study

In China the feasibility study of any big engineering project have to be made when the budget has been decided to offer by the government, as said the FFR project has only gotten the approval for the budget. After the feasibility study report is approval by the State Planning Commission the preliminary design could be started, including architecture engineer design.

For nuclear installation before the submittal of the feasibility report to the State Planning Commission, it is necessary to get the approval of the pre-safety analysis report by the Safety and Protection Department of the China National Nuclear Corporation if the project is belong to CNNC, also necessary to get the approvals of the environment impact report by the State and local government Environment Protection Administrations.

4.1 Pre-Safety analysis[4]

4.1.1 Transient of power with scram

The reactivity coefficients of MK-1* i.e. Doppler, sodium density, fuel expansion coefficients have been calculated, but control rod expansion and core deformation coefficients not included. Based on these conditions, TOP at low power and at full power have been calculated, which results are shown in Table 3 and 4.

Table 3 TOP at low power

power at TOP starting 1 MWt
sodium inlet temperature 300 °C
sodium flow in core 40% of nominal value
given limitation of power protection 35.6 MWt

reactivity adding rate(¢ / S)	3.02	2.27	1.82*	1.36	0.90**	0.63
time up to power protection(S)	41.85	49.57	59.51	77.32	***	***
maximum temperature in fuel(℃)	1714	1672	1677	1693	1200	1200
maximum temperature at mid-wall of cladding(℃)	519	513	513	516	434	434
maximum temperature in coolant(℃)	507	502	501	504	426	452

* related to Burn-up compensation rod withdrawal unforeseen.
** related to regulation rod withdrawal unforeseen.
*** The power is stable before the power protection.

Table 4 TOP at full power

power at TOP starting 102% full power
sodium inlet temperature 400 °C
flow rate in core 91% nominal value
Given limitation of power protection 120% full power

reactivity adding rate(¢ / S)	1.82*	1.36	0.90**	0.68
time up to power protection(S)	7.02	9.78	16.24	21.00
maximum temperature in fuel(℃)	2253	2253	2253	2251
maximum temperature at mid-wall of cladding(℃)	609	609	610	610
maximum temperature in coolant(℃)	585	584	585	585

* related to burn-up compensation rod withdrawal unforeseen.
** related to regulation rod withdrawal unforeseen.

It is seen from Table 3 that the reactor could appear the property of neutronic self-stability before scram triggered by power protection if the reactivity adding rate is less than 0.90 C/s. The temperature are not higher than maximum value limited for the fuel,mid-wall of cladding and sodium when one of the burn-up compensation rods or regulation rods, corresponding to 1.82 C/s and 0.90 C/s respectively is withdrawn from the reactor at its full power.

4.1.2 Transient of power without scram

It is also interested in two situation for TOP without scram, which are burn-up compensation rod is unforeseen withdrawn corresponding to 1.82 C/s reactivity adding rate, and the regulating rod, corresponding to 0.90 C/s. The results are listed in the Table 5.

Table 5 TOP without scram

incidents	time (S)	power (relative)	Temp. of fuel at hot spot(℃)		Temp. of mid–wall of cladding at hot spot(℃)		outlet temp. of sodium(℃)	
			hot channel	channel average	hot channel	channel average	hot channel	channel average
0.9¢ / S								
withdrawal rod starting	0.0	1	2148	1808	585	578	549	546
fuel starts melting at hot channel*	58.0	1.56	2750	2348	689	674	656	652
rod fully out of the reactor	62.5	1.60	2790	2393	699	686	663	660
reactor becomes stable	160	1.67	2901	2498	712	702	680	675
1.82¢ / S								
withdrawal rod starting	0.0	1	2148	1808	585	578	549	546
fuel starts melting at hot channel*	28.0	1.61	2750	2375	694	682	657	655
fuel starts melting at channel average*	44.0	2.0	3108	2750	769	752	725	72[illegible]
fuel starts failure at hot channel**	50.4	2.19	3299	2856	799	781	755	748

* means the temperature at one node of fuel has reached melting point, while the fuel is divided axially into 10 section and radially 12 nodes in the calculation
** means the temperature at 8th node of fuel has reached melting point

From the Table 5, it is found that only the temperature at one node of fuel has reached melting point in the case of 0.90 C/s reactivity adding rate. It means that the reactor will be neutronic self-stable and the all fuel pins will keep its integrity at the incident of an unforeseen withdrawal regulation rod in the FFR. However in the case of 1.82 C/s it will be inverse. It means we have to limit the raising rate of withdrawal compensation rod, despite of the completeness of reactivity feedback considered and the preciseness of the calculation.

4.1.3 Loss of flow without scram
The transient condition of loss of flow without scram has been calculation in which it is assumed that the reactor is in normal operation (full power) at the incident beginning and the duration of inertia rotating of pumps is 60 second (conservative assumption). The results are presented in the table 6.

Table 6 Loss of flow without scram

incidents	time (S)	power (relative)	full temp. at hot spot of hot channel(℃)	temp. of mid–wall of cladding at hot spot of hot channel(℃)	sodium outlet temp. at hot channel(℃)	temp. in upper part of hot sodium(℃)
beginning	0.0	1	2352	585	549	549.0
sodium starts boiling at outlet of hot channel*	17.0	0.660	2274	915	885	551.6
sodium starts boiling at outlet of channel average*	26.5	0.495	2185	1104	1080	558.3
sodium temp. reaches maximum at outlet	31.5	0.452	2142	1153	1107	562.0
calculation stops	250	0.362	1881	1077	1013	710.8

* Boiling occurred only at one axial section.

From the calculation we could find that there are no fuel melting, the temperature at mid-wall of cladding at hot spot will be over the limiting value and sodium will be boiling at four axial sections of hot channel and three sections of average channel.

Based some conservative assumptions, all calculation of transients presented here are preliminary and improved calculation will be carried on in the future.

4.2 Site evaluation

It is intended to build the FFR in the China Institute of Atomic Energy, only 40 Km far away from Beijing City. The site evaluation has been started.

4.2.1 Site evaluation rule

Which rule or standards have to obey for the site evaluation of an experimental fast reactor? As we understand the rule or standard for power reactors are much higher or more rigorous than for research reactors, for example the surface faulting must be 8 Km far away from the power reactor site (according to US ANS, in China 5 Km according to HAF) but 400 m for research reactors (in China, the same). If the FFR could be looked as a research reactor, it will be easier to get approval from the China National Nuclear Safety Administration and from the National Environment Protection Authority. But it was pointed out in IAEA-TECDOC-403 Siting of research reactors:"Fast reactors and cores with significant plutonium inventory are not considered".

In fact the safety of fast reactor are not worse than general research reactors. An experimental fast reactor even though including electricity generation should be looked as a research reactor, provided that we would make a detail environmental evaluation, especially to plutonium.

4.2.2 Siting

It is intended to build the FFR in the CIAE. The site investigation has been started, some of which are given below.

4.2.2.1 Region geology

It has been realized in the zone having a radius of 20 Km from the site that in the west of 2 Km away there is Ba Bao Shan surface faulting, in the north of about 2 Km away Lian Xian Bei surface faulting and in the south of about 4 Km away there is Lian Xian surface faulting. It is confirmed that there is no any surface faulting in this interior region, by chemical detection method, electrical detection method and shallow artificial earthquakes.

It is also realized that there was no any surface collapse for about million years in this region.

4.2.2.2 Earthquakes

We have collected the informations on historical earthquakes which have occurred in the zone having a radius of 150 Km from the site since 1983 B.C. up to 1990 A.D. It is envisaged that the maximum potential intensity of earthquakes will be 7.3 to 8.0. According to the safety regulation for nuclear installations, it is asked that the building will be designed in accordance with intensity 9.

5. Conclusion

The FFR project has been approved by the Government last month and the engineering target we planned is its first criticality in the year 2000. But up to now the technical preparation for this project is still not enough. We will extensively pursue the development using nation's own technology on one side and pay great attention on oversea's FBR experiences on other side.

Acknowledgment

Thanks are due to my colleagues Prof.s Zhang Yushan, Chen Yishao, Xiao Xunze, Senior Engineers Xie Guangshan, Xin Chaoqing, Xu Yongli, Hu Songde, Yang Fuchang etc. who have unselfishly given the contributions to this paper. Special thanks are due to Dr. Prof. Vuong Chau, Director of the committee on Fast Reactor Technology, CIAE for his guidance in the conceptual design of FFR.

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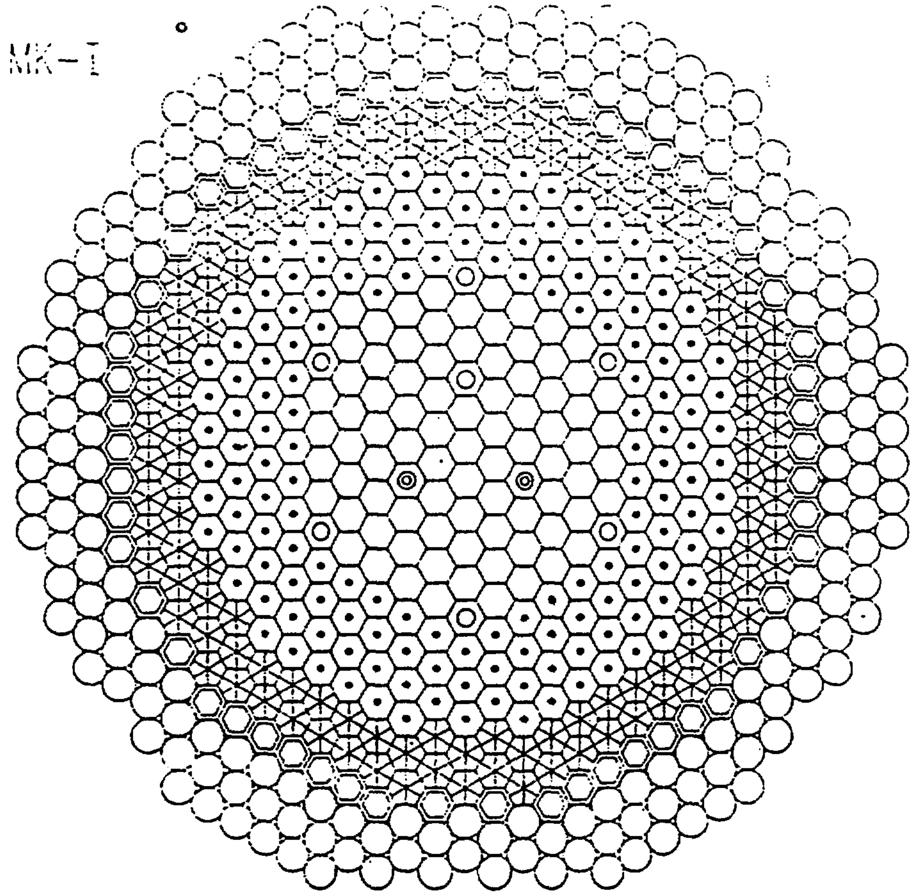


Fuel Subassembly	燃料组件	82
Blanckei Subassembly	再生区组件	162
Reflector Subassembly	反射层组件	126
Spent fuel storage position	乏燃料贮存位置	54
Control rods	控制棒	7
Safety rods	安全棒	2
Sheilding rods	屏蔽层组件	216

Fig. 1 FFR Core MK−1

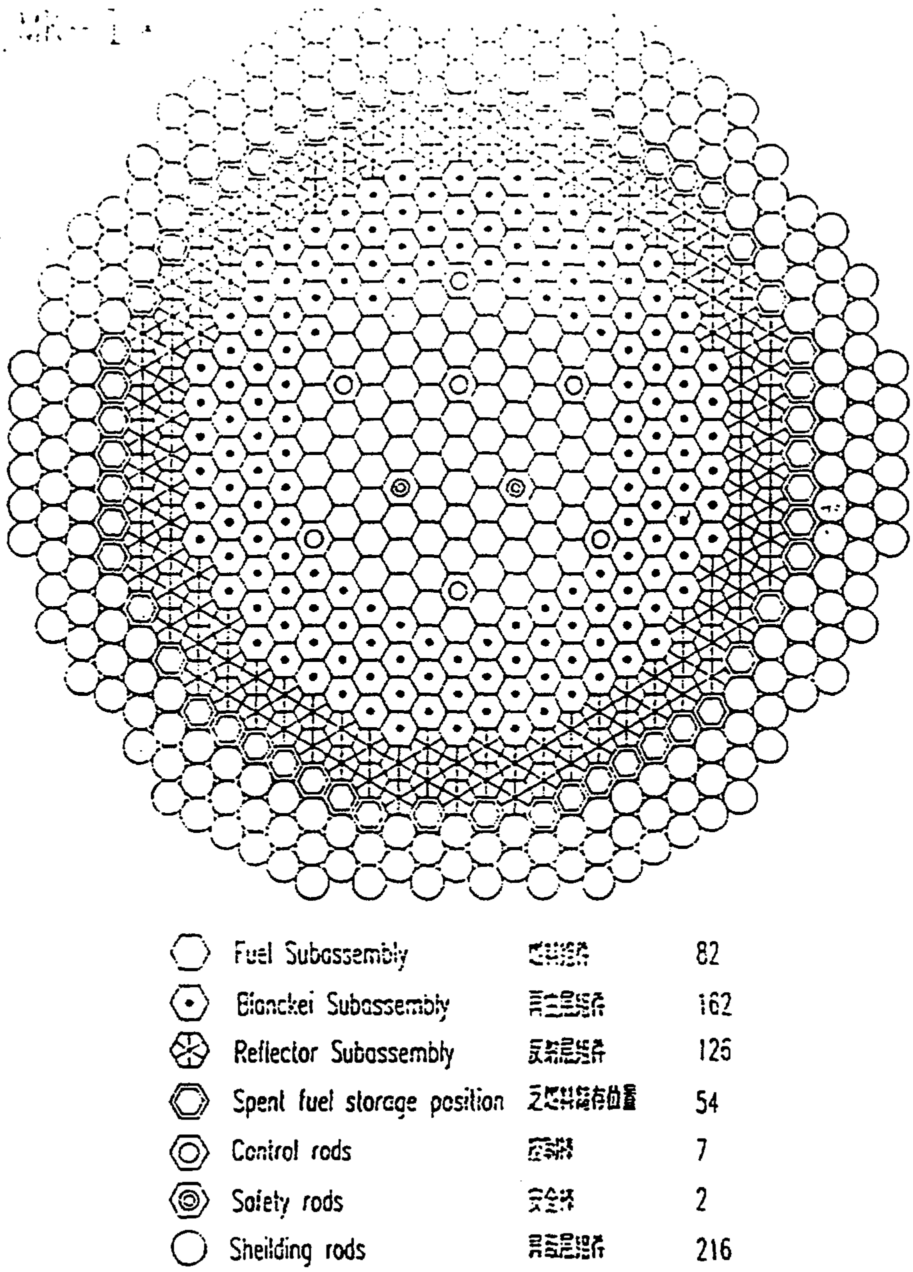
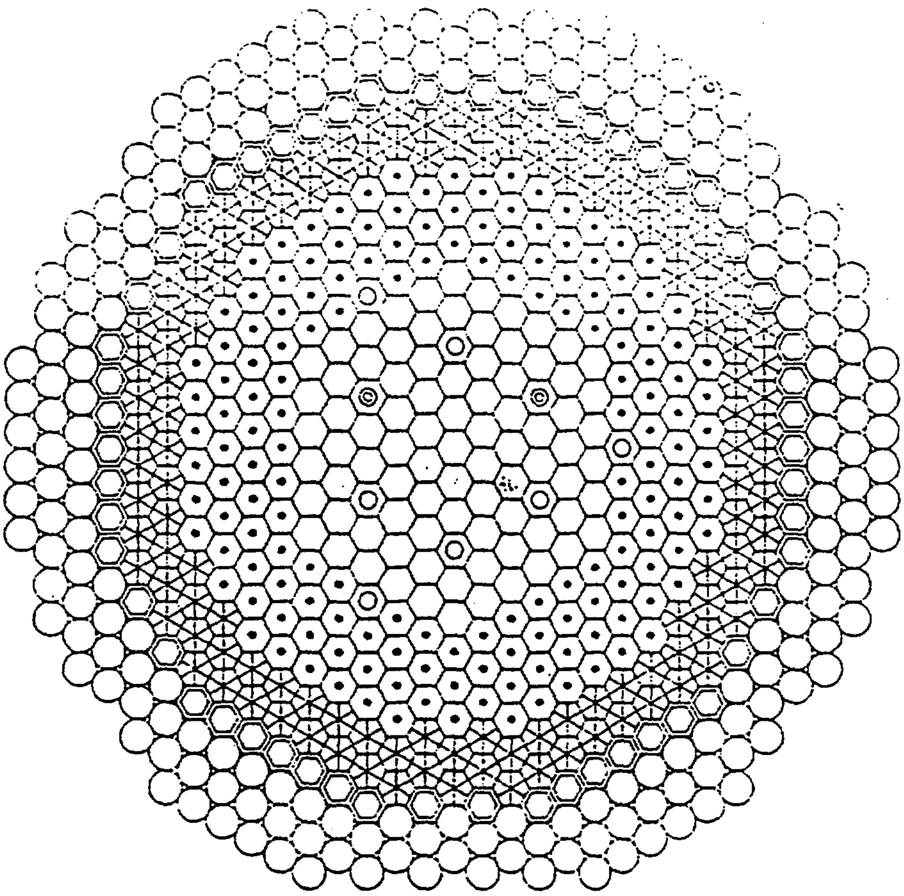


Fig. 2 FFR Core MK−1*



Fuel Subassembly	燃料组件	85
Blanckei Subassembly	再生层组件	159
Reflector Subassembly	反射层组件	126
Spent fuel storage position	乏燃料储存位置	54
Control rods	控制棒	7
Safety rods	安全棒	2
Sheilding rods	屏蔽层组件	216

Fig. 3 FFR Core MK–2

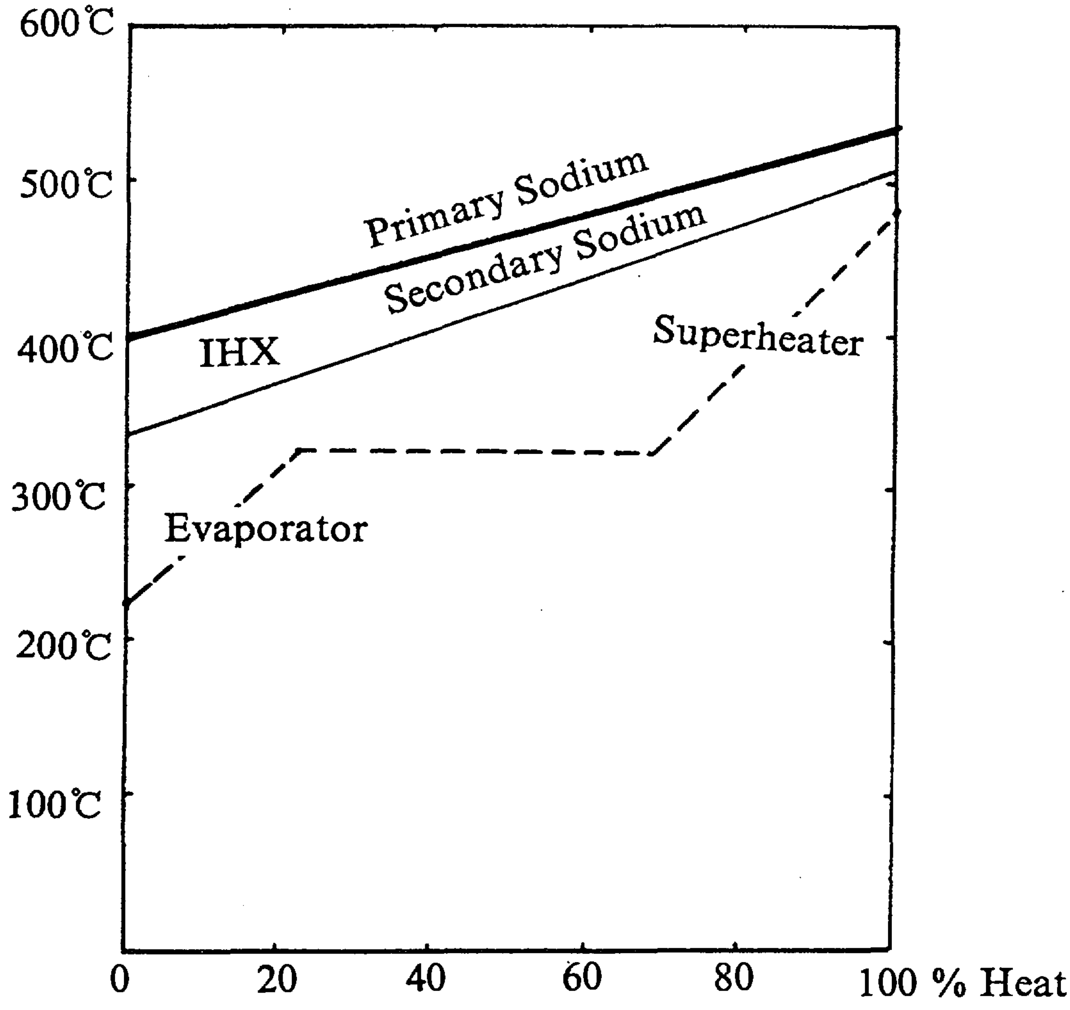
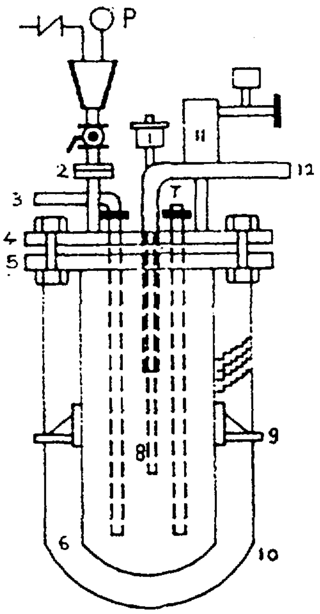
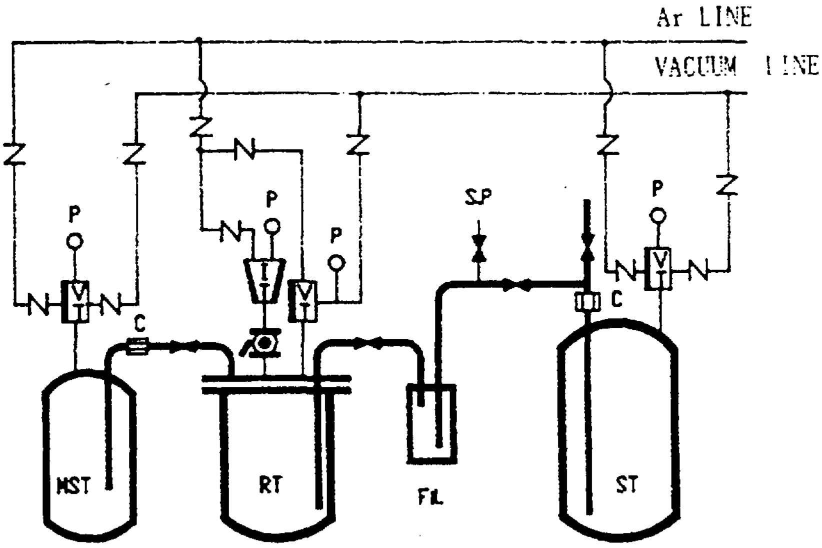


Fig. 4 Covlant temperature distribution in primary, secondary and tertiary coolant circuits of the FFR



1. lever detectors
2. L.T flange
3. outlet sodium tube
4. upper flange
5. down flange
6. thermal insulation
7. heaters
8. thermocouple
9. supports
10. casing
11. Ar-gas trap
12. inlet sodium tube

Fig.5 The Reaction Tank (RT)



—⋈—	valves for gas	P	pressure gauges	MST	Movable Sodium Tank
—▶◀—	valves for sodium	V.T	steam traps	RT	Reaction Tank
C	Connectors	L.T	Loading Tank	Fil	Filter
S.P	Sampling Points	ST	Storge Tank		

Fig.6 Schematic View of the Calcium Removal System