STATUS OF THE FAST BREEDER REACTOR TECHNOLOGY IN CHINA

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Abstract

According to the China long-term energy strategy the FBR development is strongly suppoted. In the near term nuclear programme it is intended to build the experimental First Fast Reactor (FFR) in the year 2000. Design work is in progress.

1. Introduction

There is plenty of coal, oil and hydraulic resources in China, but it is not plentiful with respect to the per capita energy resources. Furthermore, the coal was concentrated in the North, especially in Inner Mongolia Autonomous Region and Shan Xi Province. And it is only plenty of hydraulic resources in the southwest of China. On the contrary, the large amount of population and industry are concentrated in the southeast littoral of China. In order to avoid the heavy transportation of coal from the north to the south and to avoid the expensive transfer of electricity from the west to the east, the Government has decided to adopt nuclear energy as the supplementary or replacement energy for the regions lack of general energy resources.

Qinshan 300 MWe PWR and Daya Bay 2x900 MWe PWR are under construction. They mark the beginning of the nuclear power programme of the Country.

In order to make full use of the uranium resources in the future, China is interested in developing of the FBR technology.

2. The history of FBR technology development in China

The FBR technology development in China is still in the basic and preliminary stage. This situation is due to the later development of thermal reactors for nuclear power and limited funds offered by the Government.

Since the late 1960s, a small group of people has begun the basic research work on FBR technology in China with only a small amount of funds and material resources. The emphasis of research work was put on the core neutronic physics, thermo-hydraulics, sodium purification and impurities analysis, compatibility of materials with sodium and some sodium facilities in small scale. So far more than ten small sodium loops and facilities have been set up (some have been decommissioned). It is shown in Table 1

2.1 Reactor Neutronic Physics

On the zero power facility some reactor neutronic physics experiments have been done, which are:

The research of the criticality and of the characteristics of safety;

The measurements of reactivity coefficients of small specimens;

The measurements of the relative distribution of fission rates and of the ratios of fission rates;

The research of the methods for reactivity measurement including oscillator technique, numerical inverse kinetics technique and source multiplication technique, the time-range analysis technique of neutron noise of the zero power fast neutron facility:

TABLE 1. SODIUM LOOPS AND FACILITIES

Name and Function	Parameters	Commissioning Time
Fast neutron zero power facility	Load(max), 50kg ^{235}U	July 1970
Isothermal sodium loop	Temp.(max), 600℃ Sodium velocity(max), 12m/s Oxygen < 50ppm	July 1970
Sodium purification loop	Sodium volume, 150kg Oxygen after purification, < 20ppm	Sep. 1970
Sodium heat transfer facility, for single pin	Flow rate (max), $20m^3/h$ Sodium temp. (max), 550℃ Heat transfer power, 50kw Pump head, $5.5kg/cm^2$	Oct. 1970
Thermal convection sodium loop	Sodium temp. (max), 700℃ Flow velocity, 6cm/s Sodium volume, 4 l Oxygen, < 15ppm	1972
Control rod drive mechanism component test facility	Medium, water Flow rate (max), 1t/h Driving range, 800mm Eccentric distance, ±30mm	Oct. 1979
Sodium loop for Plugging meter	Temp. (max), 450℃ Flow rate, $1m^3/h$ Sodium volume, 28 l	Oct. 1981
Sodium stress corrossion test facility	Temp.(max), 700℃ Load (max), 600kg Sampte deformation range, 0—10mm	Dec. 1981
Alternative current electromagnetic pump test sodium loop	Flow rate (max), 5t/h	1968
Alternative current electromagnetic pump test sodium loop	Flow rate (max), 10t/h	1969
Direct current electromagnetic pump test sodium loop		1968
Sodium valve test loop	Temp.(max), 450℃ Flow rate, $18m^3/h$	1984

The measurements of fast neutron spectrum; and

The research of on-line real-time measurement.

Because the core of this zero power facility is too small it could not simulate the experimental fast reactor that will be set up in around 2000 in China. This facility is only used with the aim to master the diverse experiment techniques and to research experiment methods.

2.2 Thermohydraulic Studies

On the sodium heat transfer facility with flow rate 15-20m^3/h, the following experiments have been done:

The heat transfer of liquid sodium turbulence flow in the circular tube;

The heat transfer of liquid sodium flow with low Peclet number in the circular tube;

The heat transfer of liquid sodium turbulence flow in the concentric annular with inner side heating;

The heat transfer of liquid sodium flow in the eccentric annular with inner side heating;

The influence of two sides heating on the heat transfer coefficient when the sodium flows through the concentric annular.

2.3 Sodium technology

So far as sodium technology is concerned, the research of sodium purification and impurities analysis have mainly been carried on. Sodium purification facilities have been installed, the maximum output of purified sodium is up to 240 kg per day. Table 2 presents the sodium quality after purifying by two purification facilities. The methods of impurities analysis adopted in our labs are presented in Table 3. Besides those, a small chemistry sodium loop has been set up on which the research of measurement

TABLE 2. SODIUM QUALITY AFTER PURIFYING

(ppm)

Purification loop	O	C	Fe	Co	Ni	Cr	Mn	Si
1	10	20						2
2	2	3	0.15	0.01	0.01	0.01	0.005	1.1

on line, mainly by manual plugging meter has been carried on. And expansion graphite methods have been established to extinguish sodium fire in small scale.

2.4 Materials

With the isothermal flow sodium loop, the corrosion selection research has been done for more than 30 types of chromium-nickel austenitic stainless steel and some alloys based on nickel. We have carried out the study about the corrosion aspect, mass transfer and subsequent microstructure changement of alloys based on vanadium and Ti-modified type 316ss in the thermal convection sodium loop. The study of stress corrosion characteristics on type 316 ss has also been done in the stress corrosion testing facility with high temperature sodium.

2.5 Codes

About ten years ago, we have written some computer codes for FBR core design with ALGOL Language used in the computers made in China. But now they are not used due to those computers have fallen into disuse and nuclear data were too old.

These years we import and get through international cooperation some computer codes, mainly from US, for nuclear

TABLE 3. ANALYSIS METHODS FOR IMPURITIES IN Na

Element	Method
Oxygen	vacuum distillation
Carbon	1) combustion in high temperature 2) combustion of the residue after distillation
Iron Calcium Cobalt Nickel Chromium Manganese	vacuum distillation—atomic absoption spectrophotometry (with flame or graphite furnace)
Silicon	Molybdenum blue spectrophotometry
Potassium Cadmium	flame spectrophotometry

data, reactor neutronics and sheilding, thermohydraulics, fuel pin design, mechanics and safety.

2.6 Components

During the past years, the experiences about design, construction and operation have been gotten for small sodium loops. We have trial-produced successfully small magnetic pumps, valves, cold traps and some sodium instruments. For control rod mechanism, some key components have been fabricated and tested for example, bellows, buffers, grapples....

3. Strategy study for FBR development

According to the needs of the electric generation, technical bases of FBR technology in China and the trends of the FBR development in the world, as the preliminary study results, the long term strategy of FBR techndogy development in China would be divided into three steps:

- Experimental Fast Reactor
- Modulized Fast Breeder Reactor
- Large Fast Breeder Reactor

3.1 Experimental Fast Reactor

As the first step of the FBR technology development in China, it is planned to set up an experimental fast reactor (named FFR as it will be the China first fast reactor) with 65 MWt (25MWe) and to complete it in the year 2000, the main purposes of which are to get the experiences of the design, construction and operation of the FBR for electric generation purpose and to have a fast neutron irradiation facility for the development of fuels and materials.

3.2 Modulized fast Breeder Reaclor (MFBR)

For the second step the preliminary study indicated that it is reasonable to select the modulized fast breeder reactor combination with the modular size 100～150MWe as the first demonstration FBR which would be set up in about 2015 in China. The main reasons for making this selection are:

- Less technical-economic risks;
- Easier to match local electricity grids with units providing either 600, 900 or 1200MWe.

In order to get the experience a modular fast breeder prototype will be set up in about 2010. After 2015, it is envisaged that the modulized FBR combination will be constructed in series.

TABLE 4. TECHNICAL CONTINUITY

	FFR	MFBR	LFBR
Primary System Arrangement	Pool	Pool	Pool
Fuel	driver, $(U,P_u)O_2$ test, (U,P_u,Z_r) and other advanced fuel	$(U,P_u)O_2$ or(U,P_u,Z_r) or others	(U,P_u,Z_r) or others
Cladding material	316 (Ti)ss test, advanced materials	316 (Ti)ss or advanced material	advanced material
Outlet Temp	510～530℃	510～530℃	510℃

3.3 Large Fast Breeder Reactor (LFBR)

In China case the purpose to develop FBR technoloty aims at not only the effective use of uranium resources, but also needs of shorter doubling time for FBR, in other words to get the good beneficial result of the energy supplied by uranium resources in rather short period. It is envisaged that the large fast breeder reactor (LFBR), as the third step, which has high breeding capability will be set up in about 2025 and then to have its deployment.

The technical continuity of the development of FBR is shown in Table 4.

We have selected $(P_u,U)O_2$ as the driver fuel for the FFR loading, because mixed oxide fuel has been adopted for fast reactors in the world for about 100 reactor・years. The operation and irradiation results have shown the mixed oxide fuel has the high temperature stability and the good irradiation performance. However the fast reactor with mixed oxide fuel has weaker breeding capability due to lower heavy atom density and softer neutron spectrum. We are going to pay great attention to the alloy fuel, encouraged by the excellent success of ARGONNE National Laboratory of US in this domain. We intend to irradiate the alloy fuel (U, Pu, Zr) as test fuel. Other advanced fuels with which the fast reactor could have high breeding retio also will be interested, we would progressively develop them, depending on the economic ability.

After FFR, for MFBR and LFBR it is intended to pass from oxide fuel to alloy or other advanced fuel for the breeding purpose.

As for the cladding and hexagonal tube material Ti-modified type 316 ss has been chosen for FFR, considering that it could meet the needs of the maximum irradiation damage for experimental or prototype fast reactor.

As irradiation tool, the FFR will irradiate and develop other advanced structure materials, for example ferritic steel, nickel alloy, ODS and so on. It is intended to use the advanced material in LFBR for the improvements of breeding ratio and fuel subassembly life.

For the primary system arrangement of the liquid metal fast breeder reactors two principal design concepts have been used: the loop type and pool type. Generally the advantages (and disadvantages) of both concepts roughly balance each othar out[1]. But it is especially emphasized that in the pool concept, leakage in the primary system components and piping does not result in leakage from the primary system, the mass of sodium in the pool is rather big, thus providing a large thermal inertia of the reactor. And we also find the fact of the pool type favoured for designed large scale fast reactors: SPX-2, SNR-2, CDFR, BN-800, BN-1600, PFBR (India) and EFR. And we have seen the success of EBR-II, phenix, PFR, BN-600 and SPX-1. Considering all above the pool concept has been chosen for the FFR, MFBR LFBR.

According to this long term strategy of the FBR development, if the deployment of the PWRs and fuel cycle systems run smoothly and 30 GWe assumed of the total capacity are reached by PWRs in the year 2015, more than 120 GWe of the total capacity could be realized by FBRs, in the year 2050. At that time the nuclear power could play in China an important role in the national electricity supply.

4. FFR Project

In the near term nuclear program established by the China National Nuclear Industry Corporation (CNNC) and in the hi-tech program led by the National Science and Technology Committee (NSTC) , and it is decided to build an experimental fast reactor (FFR) and it is intended to be commissioned in the year 2000.

The relation of the leadership and the responsibility for the FFR Project is shown in Fig 1. The schedule of FFR is presented in Fig 2. Right now the project is still in the conecptual design stage. The main optins and boundary conditions for FFR, as presented in table 5, have been decided in the end of 1988.

The main parameters of FFR have been obtained in the first phase of the conceptual design, the results are shown in Table 6.

The cross section of the core is presented in Fig 3, the drawing of the fuel subassembly is shown in Fig 4. The conceptual design of the FFR Reactor bloc are shown in Fig 5 and 6.

5. R&D for the FFR Project

Since 1988, the new R&D program for the FFR has been begun, which includes the following subjects:

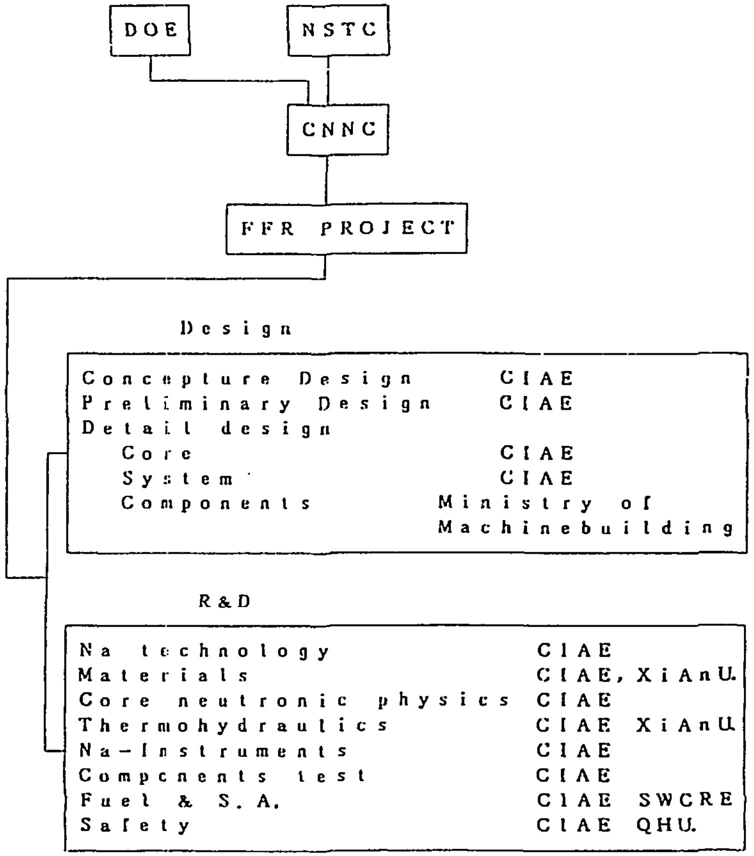


FIG. 1. FFR project.

1) Reactor design study

The emphasis is put on establishing the design ability, that is to collect and evaluate the nuclear date and materials data, to compile them, to develop and import the computer codes etc.

2) Sodium technology

In this subject we are studying the sodium purification method which could be enlarged to the scale corresponding to the needs of the FFR. The supplement of impurities

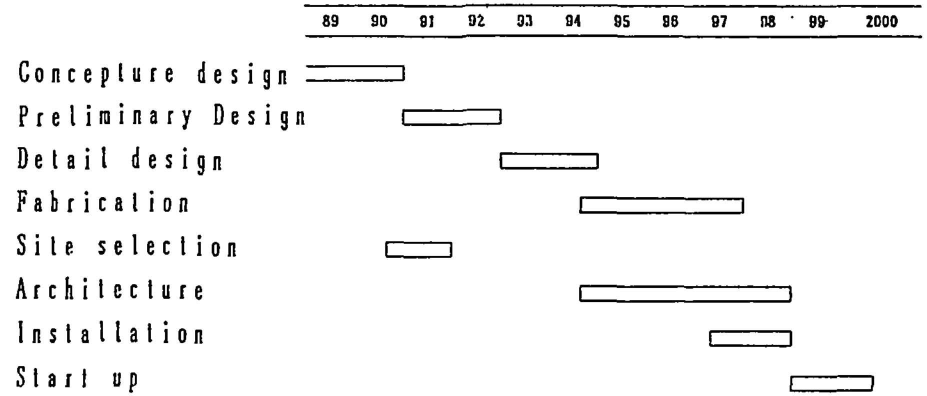


FIG. 2. FFR schedule.

analysis methods is being done. The economic washing process for the facilities contaminated by the sodium is going to be studied.

3) Fast neutron zero power experiments

The new building is under construction in which the fast neutron zero power facility will be set up. The experiments will be done to check some codes used in the FFR design.

4) Thermo-hydraulic study

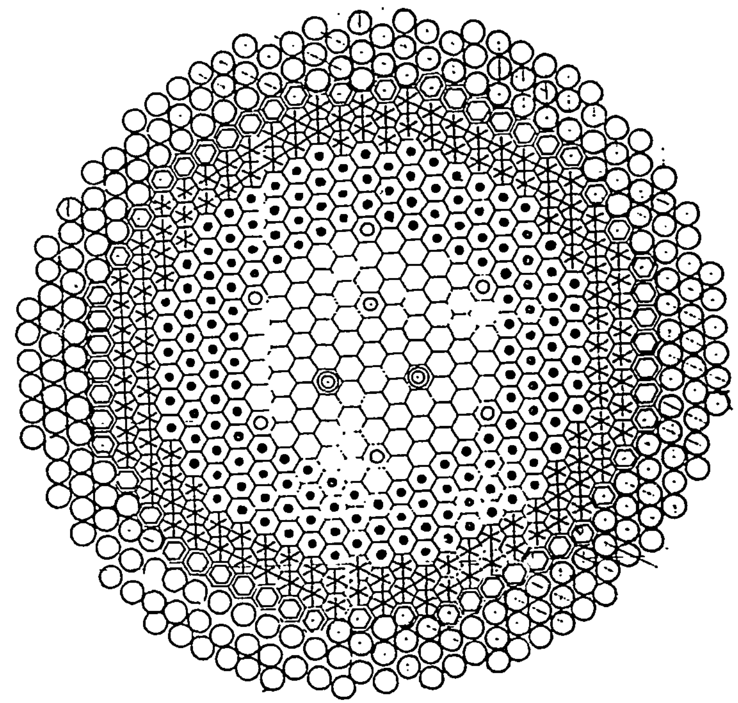
The sodium blockage experiments in the bundle will be done, which results will be used to check some transient codes. For these experiments, the electric heating elements have been prepared. In the same subject the simulation experiments of sodium natural convection in the reactor tank are under preparation, also, one pin boiling experiments in the sodium loop are under preparation.

5) Materials

Concerning the materials research for the FFR Project we pay attention on the application features, for example, corrosion and mass transfer properties in sodium, material creep and fatigue properties under sodium and the changement of the mechanical features after the irradiation damage.

TABLE 5. MAIN OPTIONS AND BOUNDARY CONDITIONS FOR FFR

Size (power)	MWe	25
	MWt	~65
Diver fuel	$(Pu, U)O_2$	
Test fuel	(U, Pu, Zr)	
Cladding material	316 (Ti) SS	
Arrangement	Pool Type	
Irradiation Cond.	No independent Irr. loops	
Spent fuel storage	Inner storage	
Refuelling scheme	Two rotating plugs with straightdraw machine	
Safety consideration	· two independent shut down systems · passive DHRS	
Outlet Temp. of the core	℃	530
Steam Temp.	℃	480
pressure	bar	90
Cladding Temp. Permited (max)	℃	650
Burnup of the fuel (max.)	MWD/T	5×10^4



⬡	Fuel Subassembly	82
⬢	Blancket Subassembly	162
✳	Reflector Subassembly	126
⬡	Spent fuel storage position	54
◎	Control rods	7
◉	Safety rods	2
○	Sheilding rods	216

FIG. 3. FFR core.

TABLE 6. FFR MAIN PARAMETERS IN THE FIRST PHASE OF THE CONCEPTUAL DESIGN

Specification	Unit	Parameter
Thermal output power	MWt	65.5
Electrical power	MWe	25
Core Fuel Equivalent		
Diameter	mm	595
Height	mm	500
Enrichment, Initial		
Pu	%	27
U-235	%	~30
Fuel Inventory		
Core (Pu + U-235)	kg	~220
Cladding outside		
Diameter	mm	6.0
Cladding Thickness	mm	0.4
Fuel S.A dimension		
Width across flats	mm	58.5
Height	mm	~2600
Number of fuel pins per S.A		61
Neutron Flux (max)	$n/cm^2 \cdot sec$	2.97×10^{15}
Core average	$n/cm^2 \cdot sec$	1.76×10^{15}

6) fuels

In this subject, we have done some trail-fabrication of $(Pu,U)O_2$ and UO_2, the former contains 27% plutonium in metal weight which is the same percentage as the selection of the FFR driver fuel. The latter is particularly for the preliminary restructuring experiments in Swimming Pool Test Reactor, so its enrichment of U-235 is only 10%.

TABLE 6. (cont.)

Specification	Unit	Parameter
Number of first shutdown rods		7
Number of second shutdown rods		2
Linear power density (max)	W/cm	430
Main tank		
Height	m	~10
Diameter	m	~8
Reactor Inlet/outlet Temp.	℃	400/530
Number of loops		2

Some facilities, that is, induction melting and injection casting are under preparation for metal fuel study.

7) Sodium-water reaction study

Water micro-leakage $Na-H_2O$ reaction experiments are being prepared. H-meter development is carried on. The injection rate of less than 1 g/s has been chosen for the experiments. The purposes of the experiments is to master the early diagnostic method of the $Na-H_2O$ reaction and to select the materials which are anti-damage spread for the steam generators.

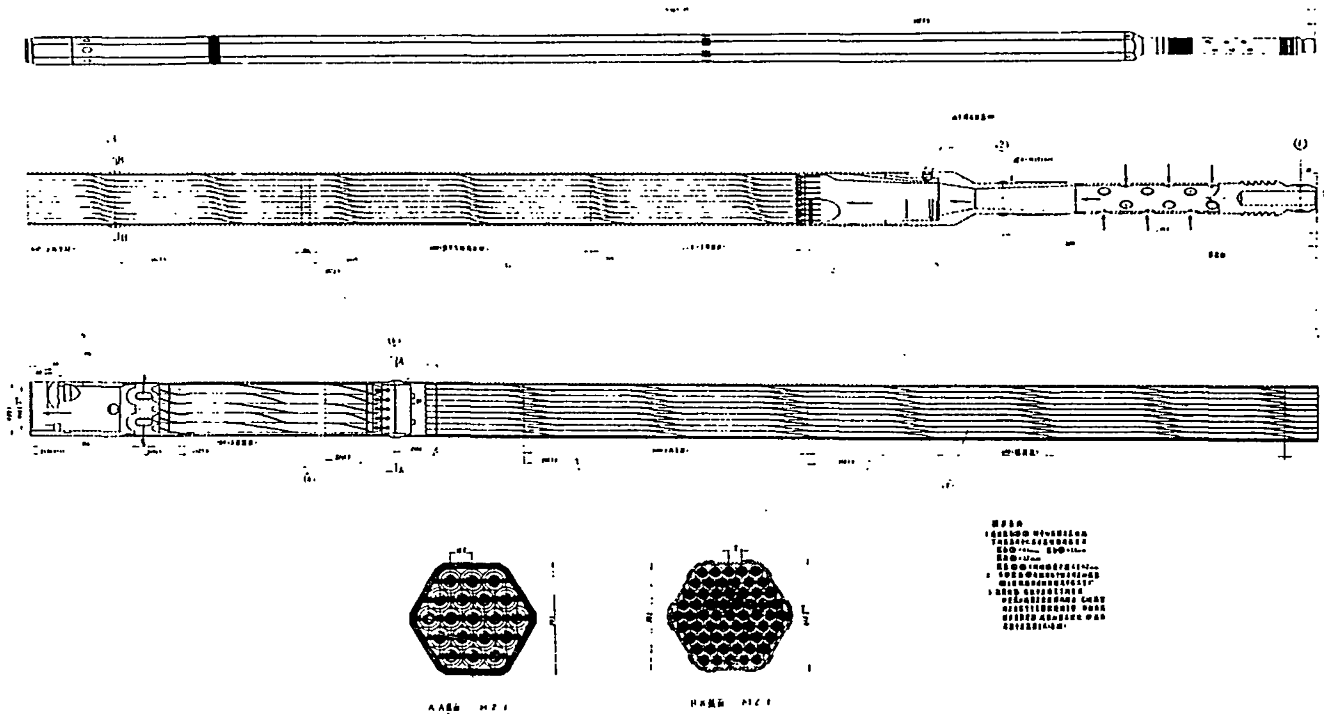


FIG. 4. FFR fuel assembly.

8) components

The trail-fabrication of some components is arranged to get the design and fabrication experience. Valves, control rod drive mechanism, fuel subassembly, sodium instruments etc, are being or will be trail-fabricated.

Above researches are all contributing to the FFR Project. Table 7 indicates some sodium loops and facilities under construction and Table 8 gives some others planned.

When the conceptual design is finished, some R&D subjects which need sodium loops and facilities of rather big scale will be proposed.

6. International cooperations

Since ten years ago, in the domain of the nuclear science and technology, the Chinese Government has established some cooperation relations with Italy, France, Germany and Japan. There are more than 30 seientists and engineers on the FBR technology who have been trained or have worked in above countries .

In 1988 we have had some memoranda and proposals of the FBR technology cooperation with foreign countries, we are waiting the approval by the Governments.

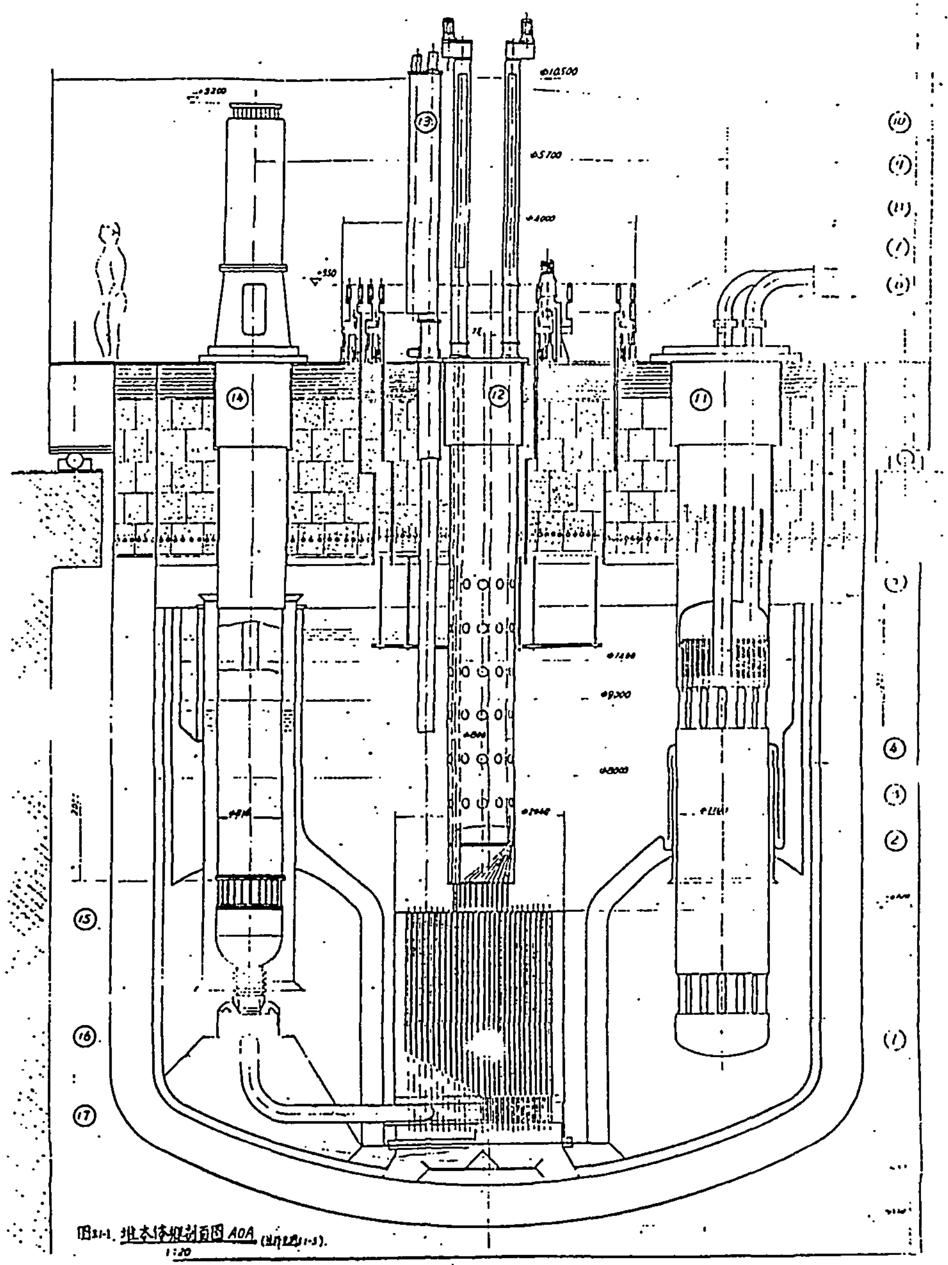


FIG. 5. FFR reactor block (1).

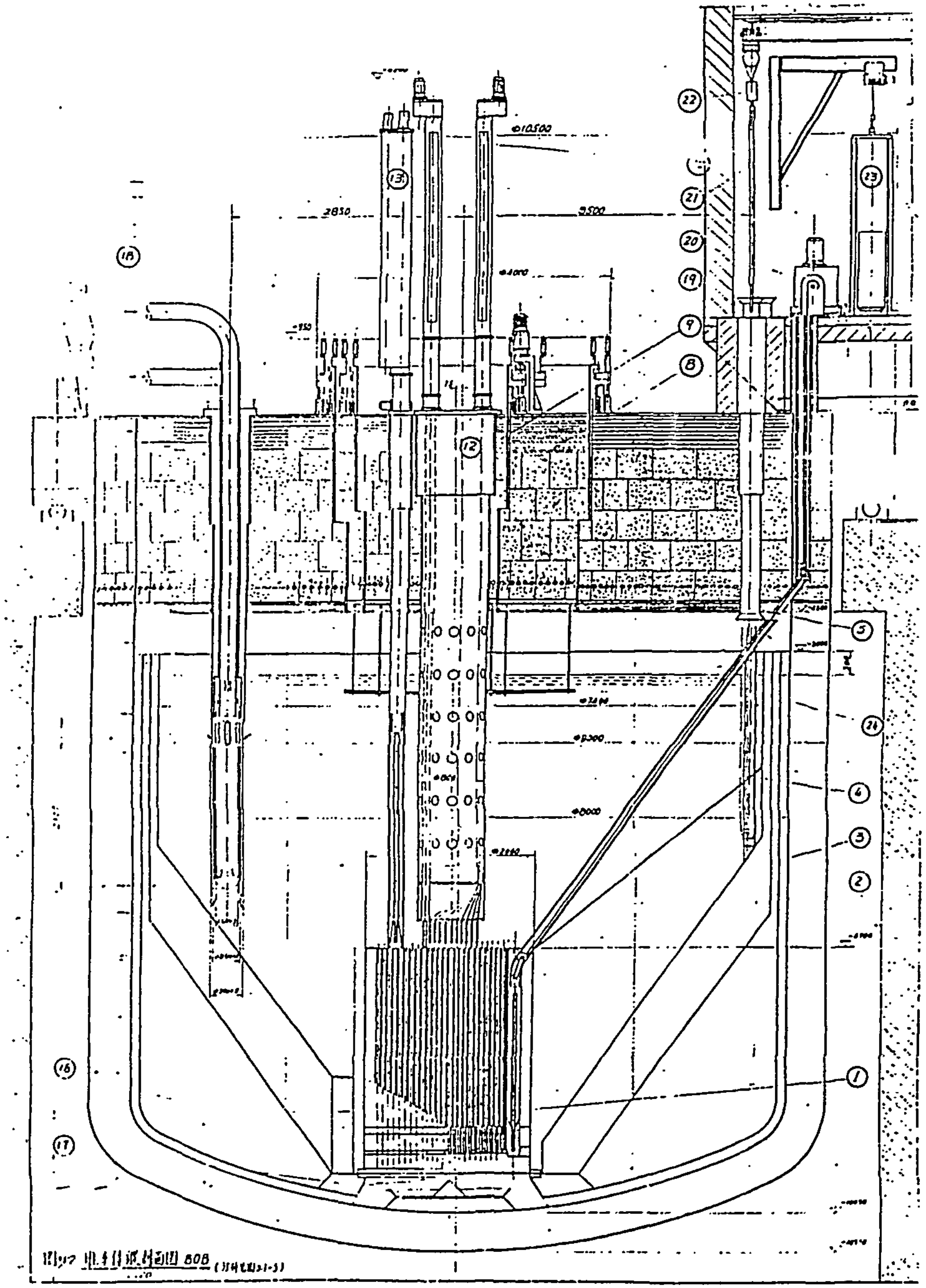


FIG. 6. FFR reactor block (2).

TABLE 7. SODIUM LOOPS AND FACILITIES UNDER CONSTRUCTION

Name and Function	Parameters		Commissioning time planned
1.Sodium heat transfer loop ,for bundle blockage experiments	Flow rate(max): Sodium Temp.(max): Test power:	20 m^3 /h 550 ℃ 300KW	Dec. 1990
2.Sodium Purification loop,Na transported by a special vessel	Volume: Temp.(max): (nominal) Flow rate Volume of C.T.	68.5Kg Na 520 ℃ 400 ℃ 1m^3 /h 9×10^{-3} m^3	Dec. 1990
3.Sodium heat transfer loop with a single pin boiling test section	Flow Rate: Temp.(max)	20 m^3 /h 1000℃	March,1991
4.Mass transfer test sodium loop, for materials test	Temp.(max) Temp.(low) Flow rate O＜20ppm C＝0.1ppm～5ppm	550 ℃ 450 ℃ 2m^3 /h	Oct. 1990
5.Material corrosion test sodium loop, for materials test	Temp.(max) Flow Rate O＜10ppm C＜1ppm	600 ℃ 12m^3 /h	Oct. 1990

TABLE 7. (cont.)

Name and Function	Parameters	Commissioning time planned
6.FCCI simulating test facility,to simulate F.P. attack to cladding material	Temp. 550 ～700 ℃ O/M: 1.96～2.00 B.U. 5 ～10%	Oct. 1990
7.Material mechanical feature sodium loop, for fatigue and creep test in Na	Temp.(max) 600℃ Na velocity 1～3m/s	Dec. 1990
8.Bi-axis creep test facility,for fuel cladding tube	Temp.(max) 900℃ pressure(max) 100bar	Dec. 1990
9.U-Zr induction melting and injection casting facility,for metal fuel study	～300g U-Zr	March,1991
10.Na- H_2O reaction facility	H_2O Injection Rate＜1g/s Temp. 300 ～500 ℃ presure 10 bar	March,1991

TABLE 8. SODIUM LOOPS AND FACILITIES PLANNED

Name and Function	Parameters	Constrution time
1.Small Scale Na facilities test loop,for valves and some instruments test in Na	Temp.(max) 550℃ Flow Rate 20 m^3/h	1991～1992
2.Sodium heat transfer loop. for S.A. blockage test	Temp.(max) 550℃ Flow Rate 60m^3/h	1991～1992
3.Core Subassemblies test sodium loop, for endurance and thermo-shock test of S.A.	Temp.(max) 600℃ Flow Rate 250 m^3/h	1991～1992
4.material corrosion sodium loop,for material corrosion test in simulation condition of FFR	Temp.(high sec.) 500℃～ 600℃ Temp.(Low sec.) 380℃～ 450℃	1991～1992
5.Rotating plug seal test facility,for O-rubber ring test	Temp.(max) 200 ℃ Pressure(max) 2bar	1991～1992

For the FFR design and construction we are naturally interested in seeing the enlargement, as possible, of the exchange and cooperation with foreign countries.

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REFERENCE

1) Status of Liquid Metal Cooled Fast Breeder Reactors, IAEA, VIENNA, P.348-350 1985