Acquisition and participation to the development of the CEA CASTEM code are proceeding; for its validation is now under consideration the setting-up of a specific Laboratory to test mechanical structures under different conditions.

Moreover Italy continues to actively participate in European initiatives in this field and specifically in the "Working Group Codes and Standards" of "Fast Reactor Coordinating Committee".

A REVIEW OF FAST REACTOR PROGRAMME IN JAPAN

Y. MATSUNO
Experimental Fast Reactor Division
O-arai Engineering Center, PNC, Japan

S. BANDO
Project Planning and Management Division, PNC
Minato-ku, Tokyo, Japan

1. Introduction

The fast breeder reactor development project in Japan has been in progress in the past twelve months and will be continued in the next fiscal year, from April 1981 through March 1982, at a similar scale of effort both in budget and personnel to those of the fiscal year of 1980. The 1981 year budget for R & D work and for construction of a prototype fast breeder reactor, MONJU, will be approximately 20 and 27 billion Yen respectively, excluding wages of the personnel of the Power Reactor and Nuclear Fuel Development Corporation, PNC. The number of the technical people currently engaging in the fast breeder reactor development in the PNC is approximately 530, excluding those working for plutonium fuel fabrication.

Concerning the experimental fast reactor, JOYO, power increase from 50 MWt to 75 MWt was made in July 1979 and three operational cycles at 75 MWt have been completed in August 1980 and the forth cycle has started in the middle of March 1981. With respect to the prototype reactor MONJU, progress toward construction has been made and an environmental impact statement of the reactor was approved by the concerned authorities.

Preliminary design studies of large LMFBR are being made by PNC and also by utilities. A design study being conducted by PNC is on a 1000 MWe plant of loop type by extraporating the technology to be developed by the time of commissioning of MONJU. A group of utilities is conducting a similar study, but covering somewhat wider range of parameters and options of design.

Close contact between the group and PNC has been kept. In the future, those design efforts will be combined as a single design effort, when a major effort for developing a large demonstration reactor will be initiated at around the commencement of construction of the prototype reactor MONJU.

Highlights and topics of the fast breeder reactor development activities in the past twelve months are summarized below.

2. Experimental Fast Reactor JOYO

2.1 General Status

Although JOYO was designed aiming at a target power of 100 MWt, it was planned to increase the reactor power stepwise through three phases, 50 MW, 75 MW and 100 MW, considering that the reactor was the first sodium cooled fast reactor built in the country. Construction of JOYO was started in the spring of 1970 at the site of the O-arai Engineering Center and the initial criticality was achieved on April 24, 1977. The low power physics tests and high power tests at 50 MWt were carried out until the middle of September 1978. The 50 MW normal operation began in October 1978 and successfully completed in the end of September 1978. The 50 MW normal operation began in October 1978 and successfully completed in the end of February 1979.

Test for power increase to 75 MWt was started in the beginning of July, 1979 and the rated power level was achieved in the middle of July. Normal operation at 75 MWt began in the beginning of 1980 and will be continued until the end of 1981 in order to accumulate technical data of the present core. Concerning the performance characterics of the reactor, the experimental results so far obtained by various tests have been in general satisfactory, being in good agreement with the predicted values.

2.2 Operational experiences

Operational experiences with the reactor are given below. The initial criticality test began on March 16, 1977 and the criticality

was achieved with 64 fuel subassemblies on April 24, 1977. After the criticality achievement, low power physics tests were carried out until the middle of November 1977. Various core characteristics such as control rod characteristics, reaction rate distributions, temperature coefficient of reactivity, fuel subassembly worths, sodium void worths, flow distribution in core and shielding characteristics were measured. During the test period, the reactor was operated at lower than 500 kW, mostly at around 10 kW, with the coolant temperature of about 250°C. During the period of those tests, the fuel handling equipment was operated satisfactory and more than 100 subassemblies were handled by the equipment. The average loading time of a fuel subassembly into the core was approximately 6 hours.

Tests for increasing power to 50 MW was started on April 18, 1978 and the power level was achieved on July 5, 1978. The power increase was made stepwise, starting from 7.5 MW, and then increased to 15 MW, 25 MW, 40 MW and finally 50 MW. At each power level, various characteristics such as power coefficient of reactivity, burn up coefficient, plant stability, heat transport characteristics of cooling system, radiation levels at various locations, were measured. In addition, thermal transient tests such as those due to reactor scram, primary pump trip, and loss of external power supply to the plant were conducted at the power levels of 25 MW and 50 MW. Those tests were successfully completed on September 16, 1978.

Normal operation at 50 MW was started on October 27, 1978. The operation was scheduled for two operational cycles until February 26, 1979. One cycle consisted of about 45 days of operation and 2 - 3 weeks of shut down. At this power level, operation of the reactor was very stable. It was needed to manipulate the regulating rods only twice a day, once every morning and evening, in order to compensate the reactivity losses due to fuel burn up. The measured reactivity loss in one operational cycle was 0.31% $\Delta k/k$, in good agreement with the predicted value.

An application for power increase to 75 MW was filed to the regulatory body in the fall of 1977 and it was granted in September 1978. After completion of two operational cycles at 50 MW, tests for increasing power to 75 MW was started on July 3, 1979 and the power level was achieved on July 16, 1979. Various performance tests at this power level were carried out until the end of August 1979. Most of the test items conducted were similar to those made at the power increasing tests for 50 MW. The results obtained were generally satisfactory. However, an interesting phenomenon was observed as to the power coefficient of reactivity. Whenever the reactor power was increased for the first time to an unexperienced power level, a large negative power coefficient of reactivity, approximately twice the predicted value, was observed. However, such a phenomenon has never been observed in subsequent operation. The cause of this phenomenon is still under investigation, but it has been suggested that one possible cause could be rearrangement of the fuel subassemblies by thermal bowing.

After the tests at 75 MW, an annual inspection to the plant was made from September through December 1979. During the inspection, sodium was drained from the primary piping and radiation dose rates at the piping wall were measured. It was observed that the radiation level was low, at most a few mR/h. Radioactive corrosion products such as Co-58, Co-60 and Mn-54 deposited on the inner piping wall were detected. Normal operation at 75 MW was started in January 1980 and it will be continued until the end of 1981.

2.3 Topics

In the plant outage time between normal operating cycles some irradiated fuel subassemblies are served to the post irradiation examination (PIE). In the PIE of the last year some wear marks were observed on the surface of the cladding tubes. These wear marks appear at positions where the cladding tubes are in contact with the spacer wire of the adjacent fuel pins. The depth of the wear mark was approximately 60μ at most and saturated both against burn up and stay time in the core. A careful analysis of the creep usage factor for the fuel pins with wear marks showed that the complete integrity of pins can be kept during their scheduled lives.

2.4 Others

Until now 3 cycles of 75 MWt operation were conducted, resulting in the accumulated thermal output of 3,794 x 10^5 MWH and the maximum fuel burnup of 22,800 MWD/T. At the present the periodical plant inspection and some equipment modifications are in progress.

The MK-II program will be started in the beginning of 1982. The core will be converted to a core of 100 MWt power by replacing the MK-I fuels with MK-II fuels. The reactor will then be utilized as an irradiation facility for fuel and material development programs in support of FBR.

3. Prototype Fast Breeder Reactor "MONJU"

3.1 Summary

A site located in the Tsuruga Peninsula in Fukui Prefecture, approximately 400 km west of Tokyo, where several LWRs are in operation, has been proposed for constructing MONJU. Survey works in various aspects of the environment such as geological, marine, and meteorological were conducted and the results of the survey works were approved by the concerned authorities at the end of the last year. And the application for licensing was filed to the regulatory body on 10. December 1980, and the licensing procedure for commencement of construction is now being intensively conducted.

As a co-ordinator in softwares of "MONJU" construction work among manufacturers, Fast Breeder Reactor Engineering Co., Ltd. (FBEC) was established in April, 1980.

And, a special department has been set up in the Japan Atomic Power Company(JAPC) to co-operate with the PNC for MONJU construction works in the same time.

JAPC will act on behalf of the nine Japanese electric utilities and the Electric Power Development Corporation (EPDC).

3.2 Overall design

In the design of this plant particular attention has been given to safety and to achieving reliable operation. Principal design and performance data are shown in Table 1 and the plant layout is shown in Fig. 3-1.

MONJU is an about 300 MWe, loop type power reactor, fueled with mixed oxides of plutonium and uranium. The reactor inlet and outlet coolant temperatures are approximately 400°C and 530°C, respectively. The expected average fuel burn-up and the breeding ratio are 80,000 MWd/t and 1.2 respectively.

The heat transport system is shown schematically in Fig. 3-2.

Components in the primary coolant system are enclosed with guard vessels and connected with elevated pipes to prevent lowering the level of sodium in the system below the minimum required safe level.

Decay heat removal is normally accomplished by means of three parallel auxiliary cooling systems (ACS), which have the branched air cooler lines consisting of the secondary sides of the ACS. Small pony motors on the main circulating pumps can provide continued coolant circulation using emergency power in the event of loss of the main power supply. A key feature, is the use of natural circulation to remove decay heat without any reliance on emergency power.

The reactor uses a simple, top-supported reactor vessel, about 7 meters in diameter and 17.3 meters in height. The reactor vessel is a cylindrical shell with a hemispherical bottom head. The vessel is surrounded by a guard vessel and these are housed in reactor cavity which is inside the concrete biological shield structure.

The reactor vessel internals are supported at the lower part of the vessel and the core is concentric with the vessel. Each fuel subassembly in the core has 169 fuel pins being hexagonally bundled and 61 fuel rods

in the radial blanket fuel subassembly. The flow distribution in the core is controlled by fixed orifices at the bottom of the fuel assembly. The fuel assemblies are hydraulically helddown to the support plate. Cladding material of fuel pins is SUS316. The length of the sub-assembly is 4,200 mm including shielding portions. The refueling interval is fixed at about six months and the core will be fueled by five-batch scatter loading scheme.

The reactor is equipped with 19 control rods (13 regulating and safety rods and 6 back up safety rods) and B_4C is used as the absorbing material. Provision is made for instrumentation on the complete core and a portion of the radial blanket. The design provides two thermocouples and a flow meter probe for each core sub-assembly and two thermocouples for each selected innermost radial blanket sub-assembly.

Reactor fuel handling is accomplished by use of the single rotating plug and one in-vessel fuel handling machine which consists of a fixed arm and a pantograph type handling machine.

3.3 Core

The core consists of 198 core fuel subassemblies. It is surrounded by axial and radial blankets. The radial blanket consists of 172 blanket fuel sub-assemblies and its equivalent thickness is 30.6 cm.
The upper and lower axial blanket are 30 cm and 35 cm long, repsectively. The core contains 19 control rod guide tubes through which 13 regulating and safety rods and 6 back up safety rods are inserted for reactor power control and shutdown. Regulating rods are grouped into 3 fine rods and 10 shim and safety rods. Boron carbide is used as absorber.

The assemblies are supported by a grid structure consisting of two core support plates fixed to the reactor vessel. Figure 3-3 shows the core configuration.

The core has two radial zones of different plutonium enrichments to flatten a power distribution and the number of fuel subassemblies in the inner and the outer zone are 108 and 90, respectively. For the equilibrium cycle, plutonium enrichments of feed fuel are 16% and 21% Pu-fissile / (Pu + U) for the inner and outer zone, respectively.

Each fuel subassembly has 169 fuel pins with wire-wrap spacers. The fuel pin is a long stainless steel tube with a central region containing plutonium-uranium oxide fuel pellets bordered above and below by a region of uranium oxide axial blanket pellets with 0.3 w/o U-235. The region above the upper blanket contains a fission gas plenum and a fuel column hold-down device. The whole length of the fuel pin is about 2,800 mm and the outer diameter and the thickness of the fuel pin cladding are 6.5 mm and 0.47 mm, respectively. The fuel pins are arranged in a triangular array separated from each other with about 7.9 mm spacing pitch.

The duct channel assembly consists of a hexagonal duct with 11.06 cm face-to-face outer distance, handle, upper and lower pads, support nosepiece, orifice, shielding and fuel rod support hardware. The upper portion of the subassembly incorporates the handle that mates with the fuel grapple during refueling. The lower portion includes the nose-piece that along with the basic support of the subassembly, provides flow orificing. Each subassembly is supported to the support plates by hydraulic holddown force.

The diameter and the theoretical density of the mixed oxide core fuel pellet are 5.4 mm and 85% TD, respectively. The maximum linear heat rating of the fuel pellet is 466 w/cm at 116% overpower condition (including hot spot factor) and the maximum temperature of the fuel with that condition is 2600^{0}C.

Fuel inventory of Pu and U in the core is 5.9 x 10^3 kg. Refueling interval is about 6 months. Five batch refueling schemes in the core and radial blanket are planned. Power fraction of the core at the beginning of the initial cycle is 93% and that of the radial and axial blanket is the rest.

Coolant flow rate through the reactor is 15.36 x 10^6 kg/hr. The flow distribution in the core is controlled by fixed orifices at the bottom of the fuel subassembly. The flow fraction is 79.7%, 10.3%, 10.0% to the core, radial blanket and bypass, respectively.

3.4 Reactor system

As shown in Figure 3-4, the major components of the reactor system are the reactor vessel, the closure head, the guard vessel, and the reactor internal structures. The reactor vessel is supported at its upper end on the concrete ledge which surrounds the vessel, and its thermal expansion is free downward. It is about 17,800 mm high and constructed of 304 stainless steel. It has inside diameters of about 7,800 mm at the upper part (which surrounds the shielding part of the closure head) and about 7,100 mm at the lower part, with a wall thickness of about 50 mm. The vessel flange, has an outside diameter of about 8,800 mm.

Primary sodium coolant enters the reactor vessel through three 24-inch nozzles located 120^o apart in the lower plenum of the reactor vessel, and is discharged from the vessel through three 32-inch nozzles which are also located 120^o apart in the upper plenum. The reactor vessel has also an outlet nozzle of the overflow system at its upper part.

The horizontal movement of the reactor vessel in the event of earth-quake is prevented by the structure provided on the bottom of the reactor vessel pit and it works through the guard vessel.

The reactor vessel pit is usually airtight, and filled with nitrogen gas.

The closure head has a thickness of about 3,700 mm and is placed on the sole plate of the reactor vessel. It consists of the stationary plug and the rotating plug. The rotating plug is located 1,080 mm eccentric to the center of the stationary plug. The rotating plug as well as the fuel handling machine (FHM) mounted on it can rotate during fuel handling.

The reactor internal structures consist of the upper internal structure and the lower internal structures. The upper internal structure (UIS) is a cylindrical plug with an outer diameter of about 1,800 mm at its lower part and with a total height of about 13,400 mm, and its lower end is 50 mm above the top of fuel subassemblies. UIS comprises 19-CRDM, 19-CRDM guide pipes, thermocouples and flow-meters for measuring temperature and flow rate at the outlet of each fuel subassembly.

The lower internal structure consist of the core barrel, the core support plates, the flow distribution structures, storage pots for core assemblies, the fuel transfer relaying rack, 316 neutron shielding assemblies which are made of stainless steel and arranged in four layers. The core support structure transmits the whole dead weight of the reactor internal structure and the core to the core support flange of the reactor vessel. Above the core support structure, upper and lower core support plates are provided. Below the core support structure, the semi-spherical partition structure is provided for the sake of regulating the flow to the pressure plenum.

As for fuel subassemblies, coolant from the reactor inlet nozzles enters the high pressure plenum through holes between upper core support plate and lower core support plate, and is distributed to each fuel subassembly after orificing by means of orifices of each fuel assembly entrance nozzle and slits of each connecting rod which is attached to the core support plates and supports the subassembly.

The sodium level in the reactor vessel during normal reactor operation is about 6,000 mm above the top of the fuel subassemblies (in another words, about 500 mm below the lower surface of the closure head) and the all lower internal structures are submerged in sodium. The free surface of sodium is covered by argon gas and the level is kept constant by the over-flow system.

3.5 Fuel handling system

Fuel handling system consists of fuel handling facilities in the reactor vessel, the charge-discharge machine, and the auxiliary installation (in the reactor building).

Spent fuels can be withdrawn directly out of the reactor vessel after two weeks from reactor shut down.

The spent fuels are placed in the sodium filled pot at the transfer position within the reactor vessel by the pantograph mechanism of fuel handling machine. Then through the charge-discharge machine, the spent fuels being kept in the sodium filled pot are carried to the exvessel fuel storage tank. After that, nonfailed fuels are cleaned, inspected, canned, and stored in the spent fuel storage pool in the reactor building and then they are transferred to the reprocessing plants. On the other hand failed or failure suspected fuels will be transferred to the fuel monitoring facilities, being stored in the sodium filled can.

3.6 Heat transports system

The heat transport system consists of main coolant system, auxiliary core cooling system, sodium service system and cover gas system.

3.6.1 Main cooling system

The main cooling system removes heat from the reactor and transports it through intermediate heat exchangers (IHX) and steam generators (SG) to turbine generator. The main cooling system consists of three loops.

One main pump, which is of free surface centrifugal type, is located in the cold leg of each primary and intermediate loop, respectively and a check valve is set at the outlet of a primary main pump to minimize reverse flow from the operating loops for the case of one loop shutdown.

Sodium level in the reactor vessel is controlled at constant with overflow and makeup system. The sodium overflowed through the hydrostatic bearing in the pump casing is guided into pump overflow column, in which entrained gas is separated, and flows back to pump suction.

The fluctuations of steam conditions are controled by the sodium pump speed.

The relative elevation of reactor core, IHX, air cooler, and steam generator are arranged to assure natural circulation of sodium in the primary and intermediate loops to remove the decay heat from the reactor immediately after scram as a back up method.

3.6.2 Auxiliary core cooling system

The three independent auxiliary core cooling systems provide decay heat removal capability for maintenance and refueling condition as emergency conditions of such cases as loss of power and component or piping failure or main coolant system.

The primary piping of the auxiliary core cooling system is common with the main coolant piping and decay heat is transferred to IHX. The heated secondary sodium of the auxiliary core cooling system is pumped to air-cooler where heat is transferred into atmosphere.

3.6.3 Sodium service system

The sodium service system covers all of the auxiliary service system for primary cooling system, intermediate cooling system and auxiliary core cooling system.

3.6.4 Cover gas system

The cover gas system consists of primary (radioactive) argon gas system of closed cycle and secondary argon gas system of open cycle. The function of the primary argon gas system is to charge and discharge argon gas to cover sodium surface with inert gas, to seal pump shafts and to transport sodium by gas pressure.

Secondary argon gas system is provided to maintain cover gas pressure at 3,000 mmAq on the sodium surface of evaporators, pumps and sodium dump tanks.

A rare gas removal system is provided for the primary argon gas system to remove radioactive nuclides such as Xe and Kr from cover gas.

The removal efficiency of rare gas is estimated to be more than 99.99% and the decontamination factor will be more than 10^4.

3.6.5 Layout and arrangement

The following is a list of design considerations incorporated in the layout and arrangement of components and pipings related with the heat transport system.

1. The primary main sodium piping is located above the minimum safe level as a rule and guard vessels are provided arround the pipings where they are below the elevation of the safe level.
2. The high point elevation of the piping and components are restricted to be within 11 m above the emergency level to prevent Torichelli's vacuum in any emergency cases.
3. Reactor vessel cover gas pressure is maintained at about 5,500 mmAq to prevent any high points from being negative pressure.
4. The relative elevations of the reactor core, IHX, air cooler and steam generator are arranged to assure natural circulation in the primary and intermediate loops.
5. The elevation of the hydrostatic bearing of the main pump is determined to be submerged in sodium even if the sodium level in reactor vessel is below the emergency level.
6. The intermediate sodium pressure within IHX is maintained above the primary sodium at any conditions of operation and shutdown.
7. Considerations are given for the maintenance of components and instrumentations of the primary cooling system.

3.7 Steam generator system

The steam generator system consists of an evaporator and a superheater per loop, all of which have helically coiled heat transfer tubes. As shown in Fig. 3-2, an evaporator and a superheater are both operated in series.

The tube sheets of water inlet and steam outlet are provided in the cover gas region. For sodium level control, the evaporator utilizes an overflow control system and the superheater utilizes a gas control system.

A safety system for sodium-water reaction is provided for each component to release the pressure and products which are produced during the reaction. This relief system is designed on the basis of double ended fractures of four tubes. A rupture disk is installed in the cover gas region of each component and if the pressure built up is high enough to rupture, the reaction products flow through the rupture disk to a reaction product tank.

This exhaust system is maintained with inert (nitrogen) gas atmosphere and is separated from the steam generator system by the rupture disk. Hydrogen in sodium is detected by an ion pump current indication device which detects hydrogen diffused through a thin nickel memberane quickly to detect a small water or steam leakage into sodium. Hydrogen in cover gas is also monitored by a diffusion type detector.

3.8 Instrumentation and control system

Instrumentation and control systems are composed of plant protection system, neutron detection system, in-vessel detection system, failed fuel detection system, radiation monitoring system, sodium leak detection system, plant control system, and process instrumentations.

The plant protection system automatically trips the reactor by actuating reactor shutdown system to keep the fuel integrity at upset (anticipated) and emergency (unlikely) events. The plant protection system instrumentation, shown in Fig. 3-5, consists of three independent instrument channels with sensors and two logic trains to open reactor trip breakers automatically. "MONJU" has two shutdown systems, main and backup, each of them has the own ability to independently shut down the reactor.

The design effort for the in-vessel instrumentation is devoted to avoid the excessive core fuel temperature by detecting the fuel anomaly ities at early stages. Reliable detectors, which can be utilized under severe conditions in vessel, are required and now being developed. By the neutron detection system, neutron flux from the fuel loading stage to 120% of rated power output is monitored. The output signal is sent to the plant control system and to the reactor protection system.

A failed fuel detection system is based on two different principles. One is FFD which detects failed fuels and their degree of magnitude, and the other is FFDL which locates failed fuel subassemblies. The delayed neutron and cover gas method are employed for FFD and the tagging gas method is adopted for FFDL in "MONJU".

The instrumentation and control system is designed so that the heat transport system should be centrally controlled from the main control room during various stages of the plant operation in view of operational safety and simplicity.

The control system is designed to accommodate the uniform ramp change of ±5%/min and step change of ±10% without scram and activating the steam dump system. The maximum driving speed of fine regulating rods is 300 mm/min. Primary and secondary sodium flow are variable at a rate of maximum ±10%/min of rated flow between 100% and 30% of rated power output. Feedwater flow can be changed by the feedwater control valve which is located at the evaporator inlet in each loop and by the speed control of main feedwater pumps.

The plant has the set-back system, which decreases reactor power at a rate of 5%/min on anomalous phenomena of unknown origin until the set-back condition becomes cleared. If set-back conditions are not cleared before the reactor power reaches 30% of the rated power, the reactor will be scramed.

Figure 3-6 shows the plant control system of "MONJU". The plant control system includes the following systems.

1. power demand master
2. reactor power control system
3. primary sodium flow control system
4. secondary sodium flow control system
5. feedwater flow control system
6. steam pressure control system
7. steam temperature control system

Power demand master gives each subsystem demand signal corresponding to plant load demand. It sets the rate of load change, range of step load change and setback load demand. Reactor power control system uses reactor outlet temperature as a main control signal and neutron flux signal as a supplementary signal. Reactor outlet temperature is programmed as a function of power level. Primary and secondary sodium flow control systems are composed of similar scheme. The main control signal in these systems is sodium flow and the supplementary signal is the pump speed. The evaporator outlet steam temperature is controlled by feedwater flow which is regulated by the feedwater flow control valve. The feedwater control system regulates the feedwater pump speed to keep the differential pressure of feedwater control valve at a fixed value. The superheated steam pressure will be kept at a constant value by the turbine inlet control valve.

Outlet steam temperature of the superheater is kept at the fixed value for all of the power level by controlling the temperature of the reactor outlet sodium.

3-9 Radioactive Waste Processing System

The design objectives of the radioactive waste processing system are to minimize the levels of radioactive materials in the plant effluents to the environment and/or to product packages of the radioactive wastes to store appropriately in the plant.

3.9.1 Gaseous Waste Processing System (GWPS)

GWPS is designed to process gaseous wastes generated in Primary Argon Gas System (PAGS), Fuel Handling System and other systems in the plant, which consist of radioactive rare gas isotopes and fission products. These gases are collected and processed through activated charcoal beds in GWPS before release to environment.

Since gaseous fission products contained in the primary cover gas are removed by rare gas removal system and processed gas are reused as primary cover gas in PAGS, the primary cover gas is not discharged under normal operation. Practically, some in-leak gases and blow-down gases must be taken into account. To control the cover gas inventory, excessive gases are discharged from PAGS to GWPS.

3.9.2 Liquid Waste Processing System (LWPS)

Liquid wastes are classified into following five groups in accordance with their sources and properties.

(i) irradiated fuel assembly washing effluent

(ii) sodium component washing effluent

(iii) reactor building drains

(iv) radioactive waste processing system drain

(v) laundry drain

LWPS consists of two subsystems. The first is designed to process liquid wastes mentioned above in (i) to (iv), the second is designed to process laundry drain. In both subsystems, decontamination is carried out by evaporation and/or demineralization. To minimize the release activity some of processed liquids are reused for sodium component washing.

3.9.3 Solid Waste Processing System (SWPS)

SWPS is designed to process and package solid wastes listed below, to be stored in-site and/or to be transported for disposal.

(i) evaporator concentrates

(ii) spent resins

(iii) spent air cleaning filters

(iv) contaminated cloths and papers

(v) scrapped components

Evaporator concentrates and spent resins are treated by bitumen solidification unit to produce immobilized waste packages. Spent air cleaning filters are packaged at the generated place. Contaminated cloths and papers are placed in 200 ℓ drums and compacted. Scrapped components are transfered and stored in the storage pond.

3.10 Reactor containment and building

3.10.1 Reactor containment

The reactor containment consists of the primary containment cells and the containment vessel to isolate radioactivities released at a hypothetical accident. An annular space is formed by providing a cylindrical shielding of reinforced concrete surrounding the containment vessel.

(1) The primary containment cells are composed of a combination of reinforced concrete and steel lining. Atmosphere of the primary containment cells which are designed to be protected against a sodium fire is nitrogen.

(2) The containment vessel made of steel, 49.5 m in diameter and about 79.4 m in height consists of a vertical cylindrical shell with a hemispherical head and a ellipsoidal bottom shell. It has an equipment hatch, an personnel air lock and an emergency air lock. A rotary crane to move primary components is installed at the wall of the containment vessel.

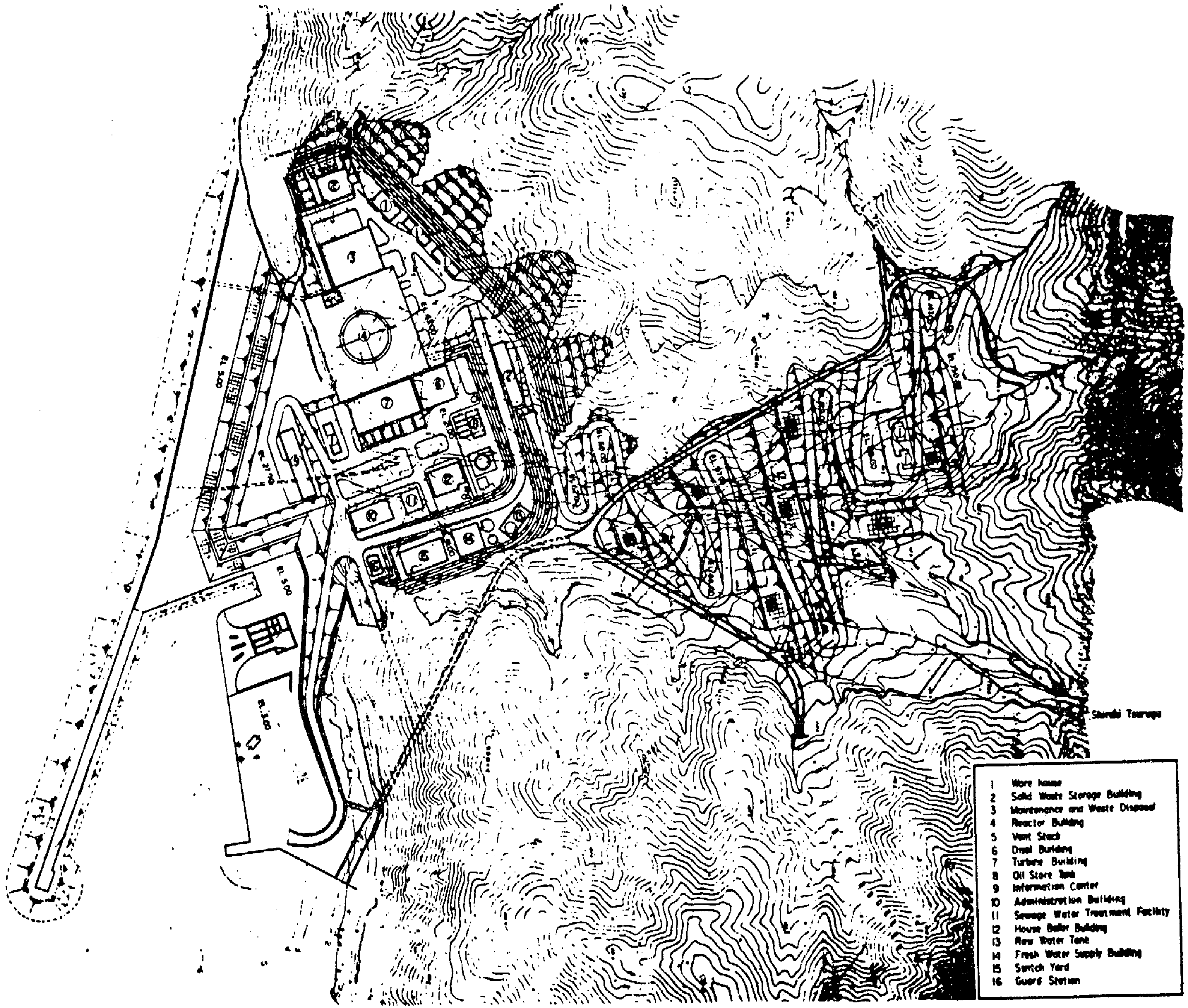
3.10.2 Building arrangement

A sectional and a horizontal cross section of the reactor building are shown in Fig. 3-7 and Fig. 3-8 respectively.

Table 3-1 Principal Design and Performance Data of "MONJU"

Reactor Type		Sodium cooling loop type
Thermal Power		714 MW
Electrical Power		about 280 MW
Fuel Material		$PuO_2 - UO_2$
Core Fuel	Equivalent diameter	1,790 mm
	Height	930 mm
	Volume	2,335 lit.

Pu Enrichment (Pufiss %)		Inner core/Outer core
	Initial core	15.0/21.1
	Equilibrium core	14.9/20.5
Fuel Inventory Core (U + Pumetal)		5.9×10^3 kg
	Blanket (Umetal)	1.71×10^4 kg
Average Burn up of Discharged Fuel		80,000 MWD/T
Cladding Material		SUS316
Cladding Outside Diameter/Thickness		6.5/0.47 mm
Permissible Cladding Temperature (middle of thickness)		675 °C
Power Density		283 kW/lit.
Blanket Thickness (axial/radial)		Upper 300 mm Lower 350 mm/306 mm
Breeding Ratio (initial/equilibrium)		1.20/1.21
Reactor in/out Sodium Temperature		397/529 °C
Secondary Sodium Temperature (IHX outer/IHX inlet)		505/325 °C
Reactor Vessel (height/diameter)		17,300/7,100 mm
Number of Loops		3
Pump Position (Primary and secondary loop)		Cold leg
Type of Steam Generator		Helical coil, once-through unit type
Steam Pressure (turbine inlet)		127 kg/cm^2g
Steam Temperature (turbine inlet)		483 °C
Refueling System		Single rotating plug with fixed arm FHM
Refueling Interval		6 months



S = 1:2,000

Fig. 3-1 "MONJU" Pilot Plant

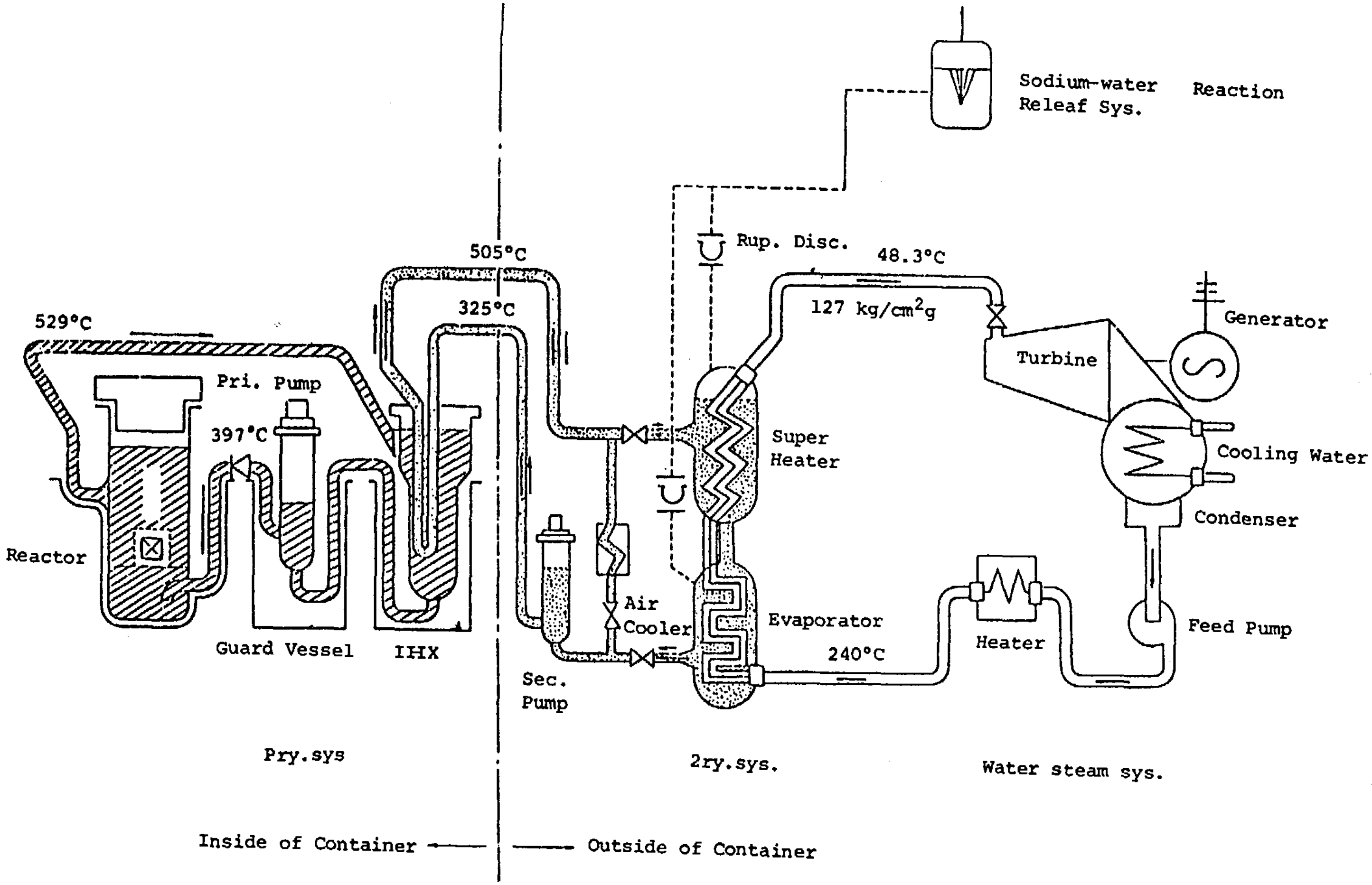
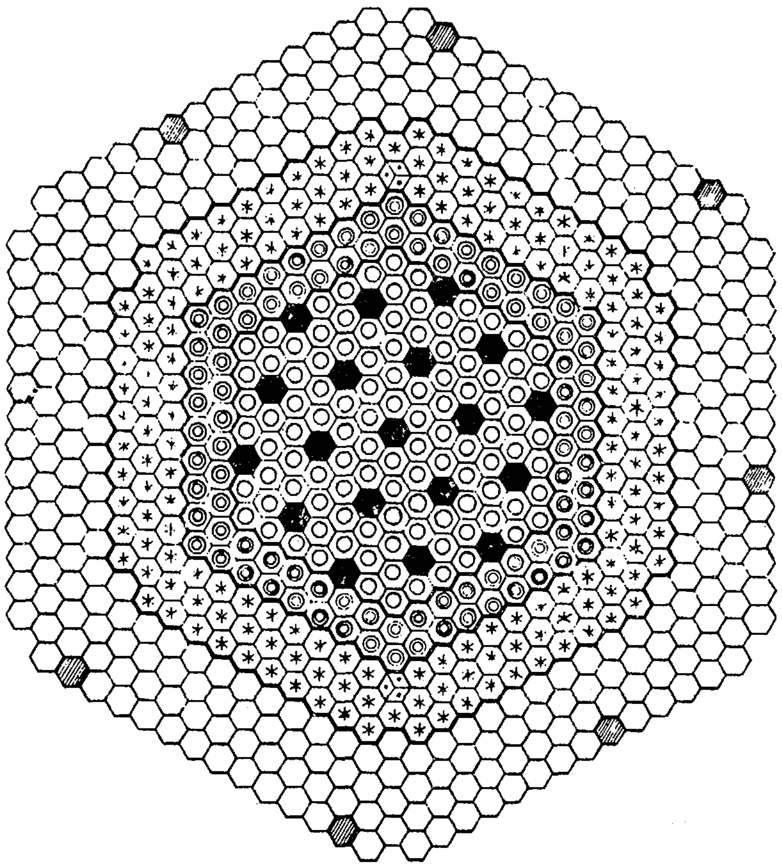

Sodium-water Releaf Sys.
Reaction
Rup. Disc.
48.3°C
127 kg/cm²g
Generator
Turbine
505°C
325°C
529°C
Pri. Pump
397°C
Super Heater
Cooling Water
Condenser
Reactor
Air Cooler
Evaporator
Feed Pump
Heater
240°C
Guard Vessel
IHX
Sec. Pump
Pry.sys
2ry.sys.
Water steam sys.
Inside of Container
Outside of Container

FIG. 3-2



core elements		marks	quantities
core fuel-S/A	zone I	⬡	108
	zone II	⬡	90
radial blanket fuel S/A		⬡	172
control rod		⬢	19
neutron source		⬡	2
neutron shielding		⬡	316
surveillance S/A		⬡	8

Fig. 3-3 Core Configuration

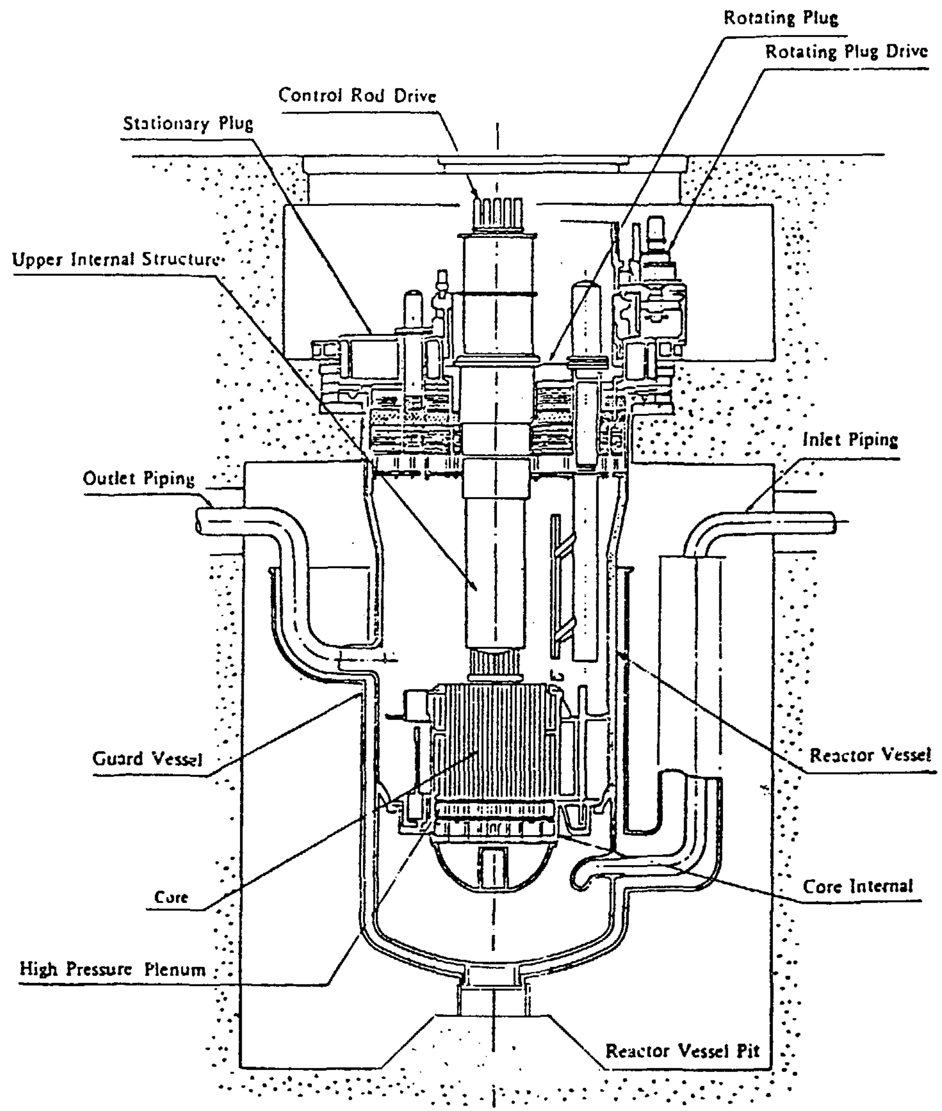


Fig. 3-4 "MONJU" Reactor System

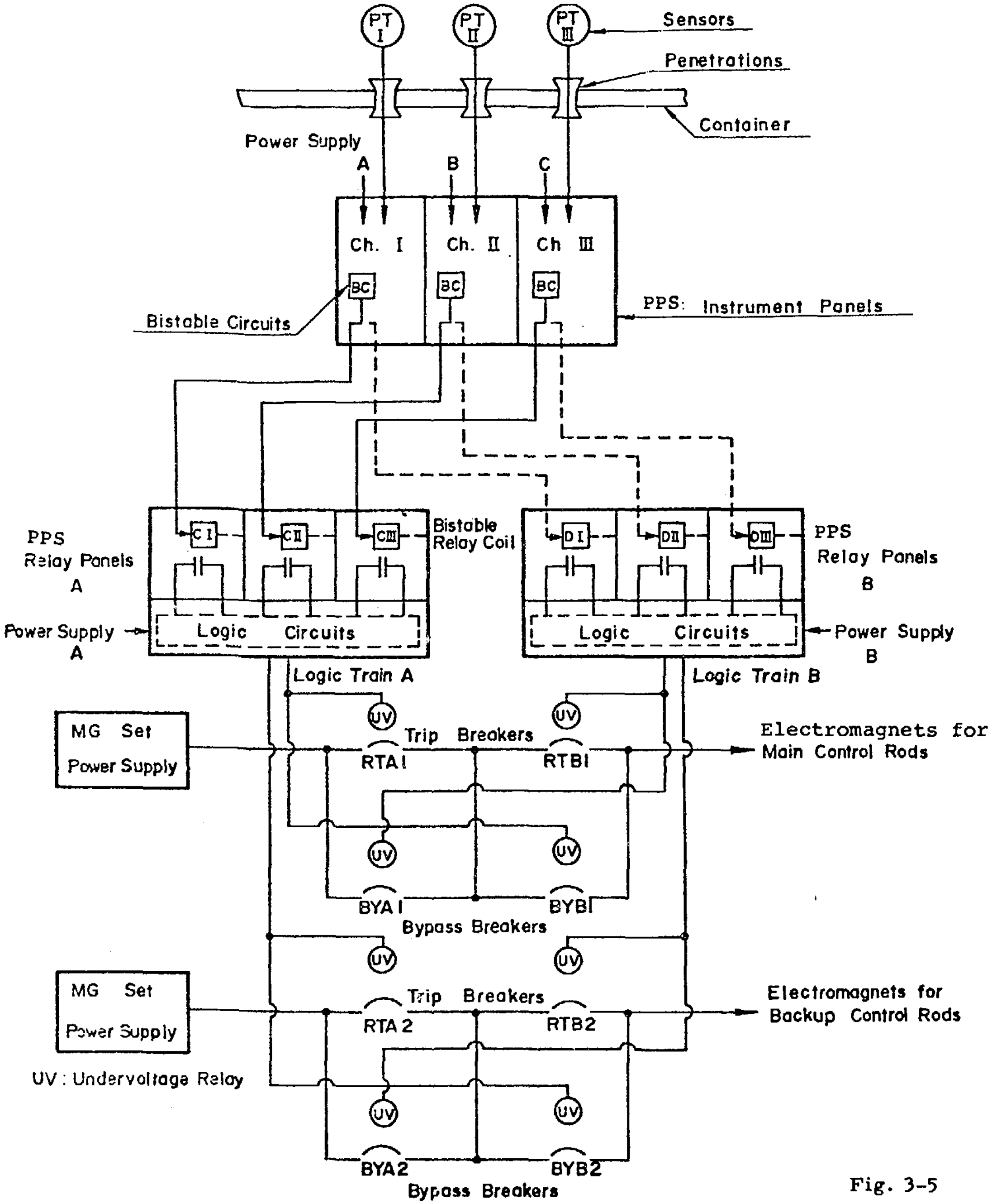


Fig. 3-5 Plant Protection System Instrumentation

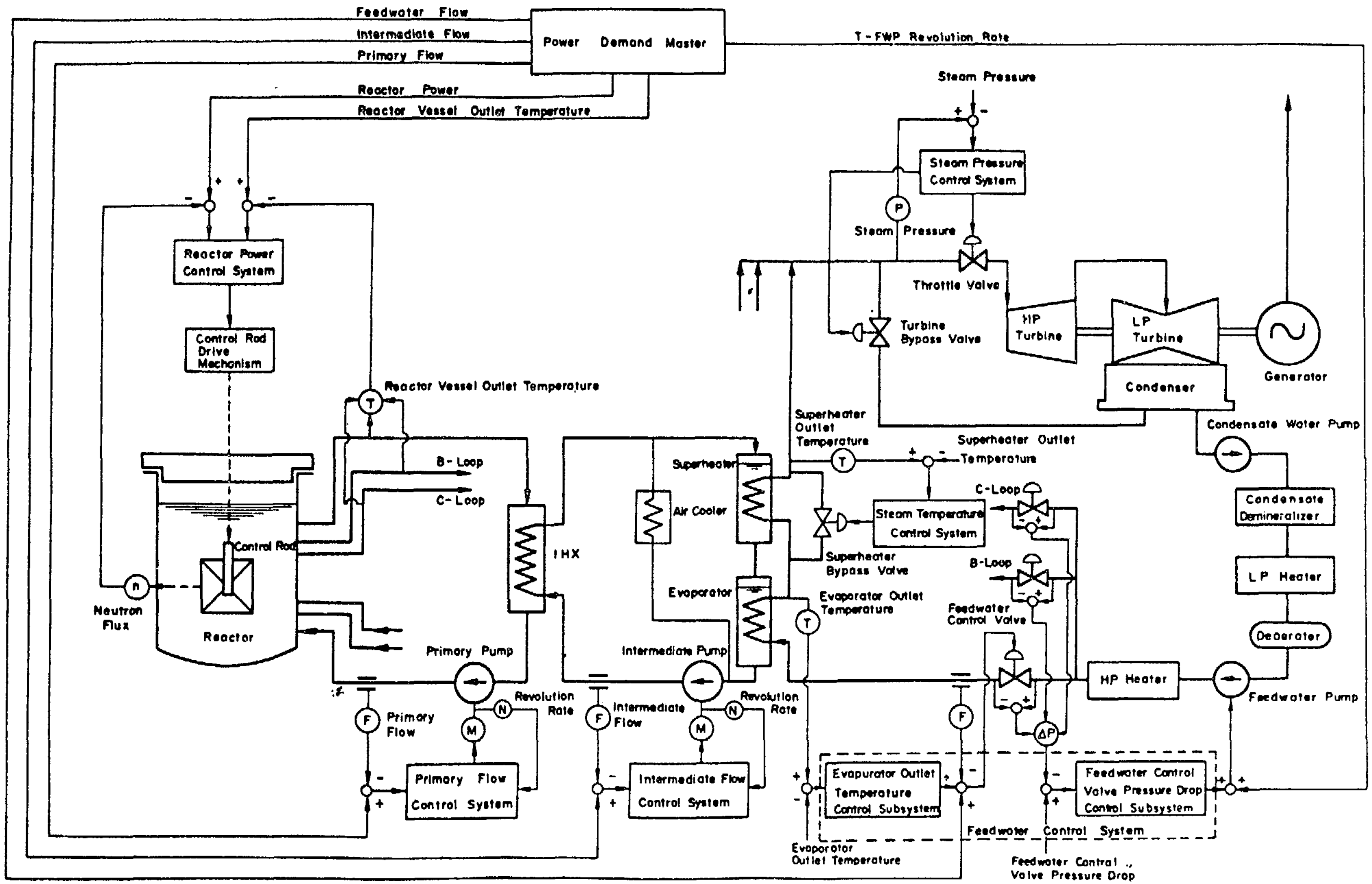


Fig. 3-6 Plant Control System

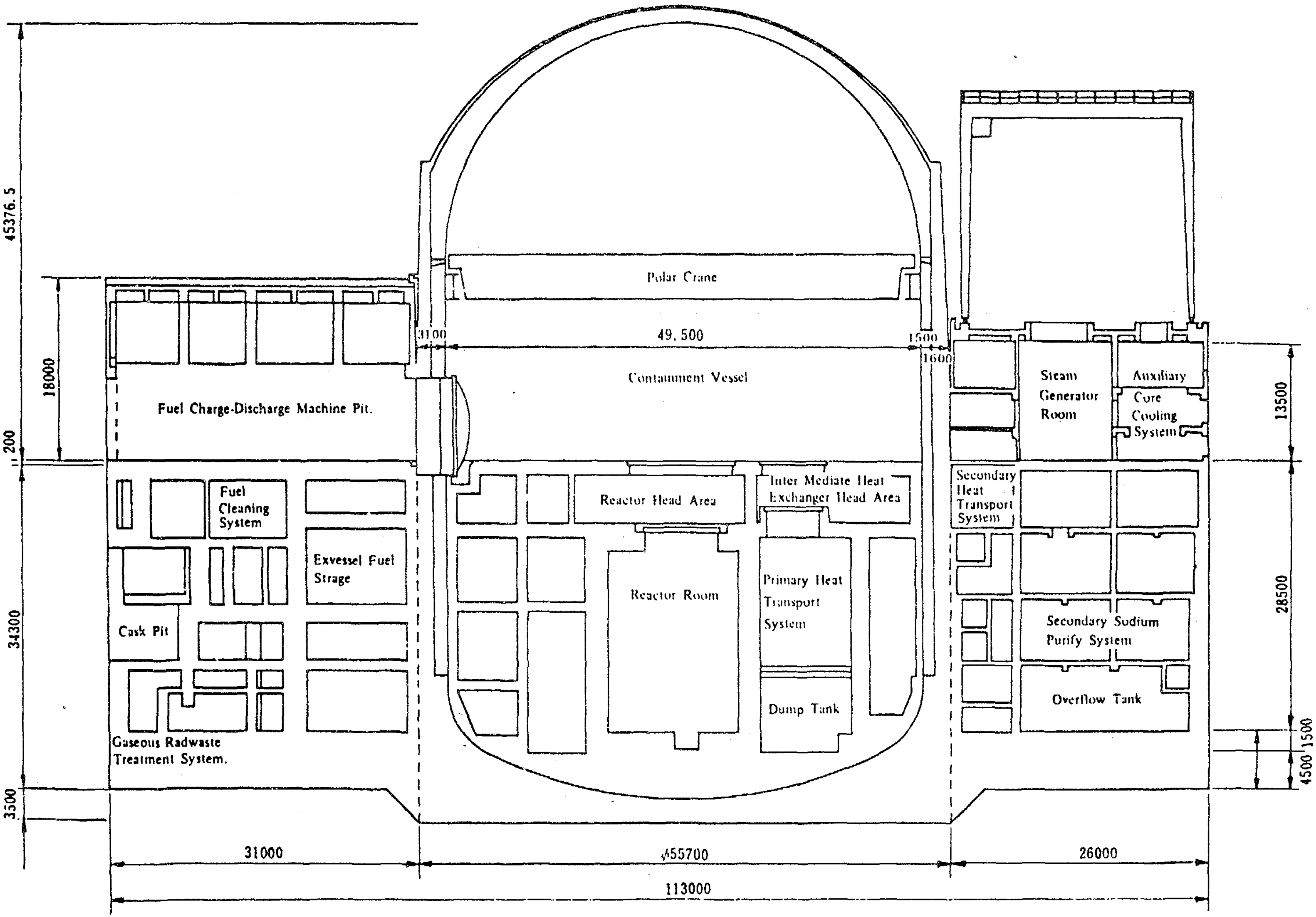


Fig. 3-7 Sectional Elevation of Reactor Building

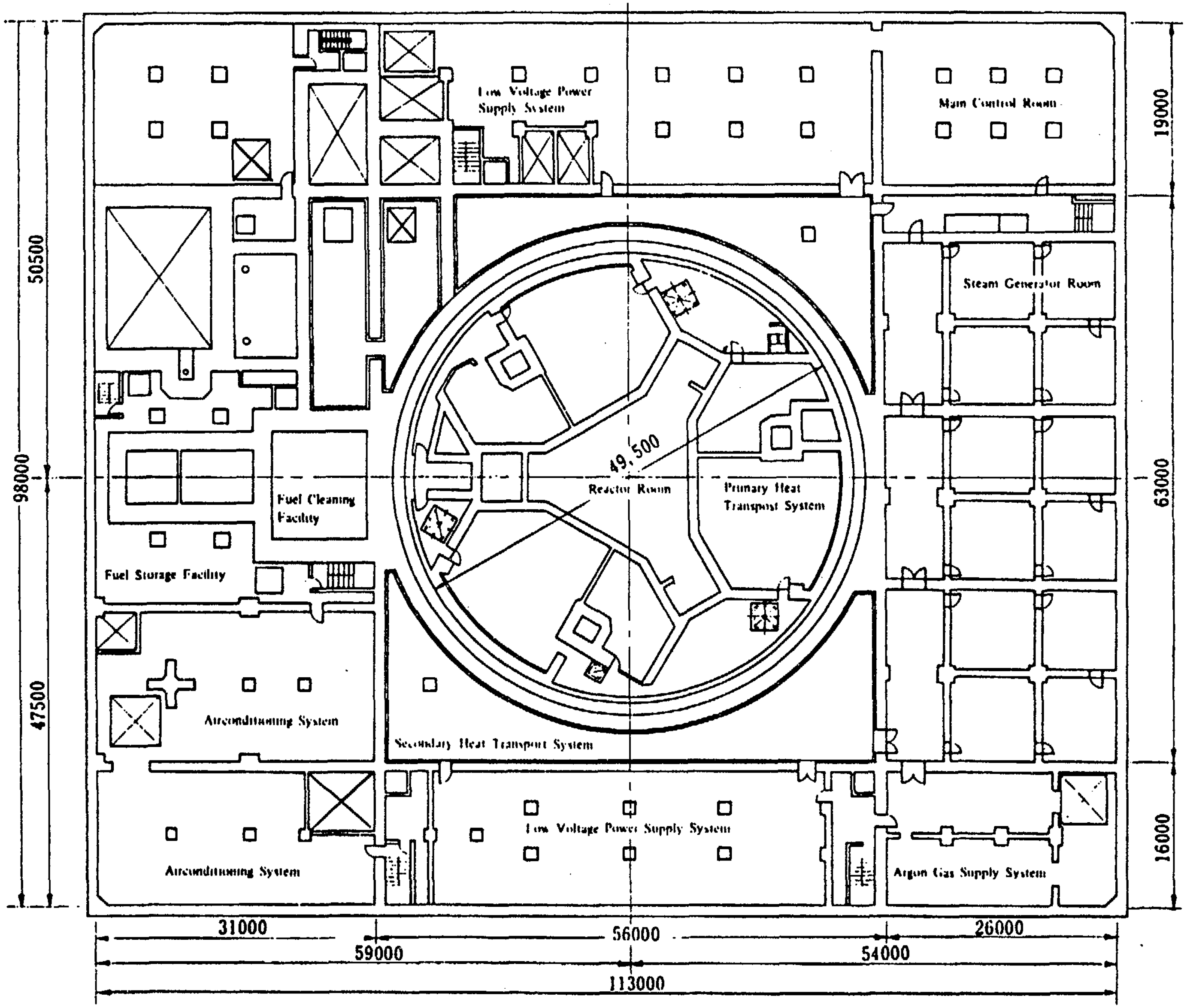


Fig. 3-8 Horizontal Cross Section Reactor Building

4. Demonstration Fast Breeder Reactor

After 4 years of preliminary design study, the new design programme started in 1979 and the first phase of a projected two-phase programme has been completed.

The first phase developed conceptual designs of 1000-MWe loop type LMFBR power plant aiming at commencing its construction in 1988.

The reactor core utilizes PuO_2-UO_2 fuel and UO_2 breeder elements in optimum configuration to improve overall breeding, fuel cycle economics and plant characteristics. Both homogeneous and hetegogeneous core configurations have been taken in consideration. The fuel pin outer diameter is 7.4 mm. Previous optimization studies have shown that this pin size gives optimum breeding characteristics and economic performance.

The breeding ratio and doubling time (C.S.D.T of 15 years or less) for these cores are based on the domestic projection of future electricity demand.

Reactor inlet and outlet temperatures of 385°C and 530°C are selected to improve plant economics. The reactor is controlled by 31 conventional B4C neutron absorber assemblies, and has two different types of safety and backup shutdown assemblies.

Two rotating plugs with offset arm are employed as refueling system, capable of handling 1/2 of a core loading in 17 days.

The primary reactor vessel, designed as small as possible, 9.5 m in diameter by 19 m deep, is top and bottom supported by an integral support ring and a circular core, respectively, is resistant to seismic disturbance and to thermal shock.

All three primary loops include single-suction-type mechanical pump with excellent low-NPSH charactèristics (100-m head requirement). The three intermediate loop each utilizes a mechanical pump with a standard centrifugal impeller.

The steam generator system employs the helical coil evaporator and superheater units, which operate in a once-through mode.

The containment system consists of a 65 m inner diameter, steel, cylindrical primary containment with a curtain wall and a 68.5 m outer diameter, annular concrete confinement.

The steam cycle conditions are 128.0 ata and 483°C at the turbine throttle. A low-speed (1,500 rpm) tandem-compound turbine is employed.

5. Physics

5.1 Evaluation of Nuclear Reactor Constants Sects.

The group constants library for fast reactor calculation, JFS-3-J2 was produced by using the processing codes PROF·GROUCH·G-II and TIMS 1 from the evaluated nuclear data file JENDL-2. The presently produced JFS-3-J2 differs from the JAERI-Fast set (JFS-2), as to concept of group constants, as follows :

(1) Group structure : Seventy energy group = 69 groups for the energy range from 10 MeV to 0.414 eV with equal lethargy of 0.25 and one thermal energy group.

(2) Weighting spectrum : REMO-correction by using collision density for a typical fast reactor core.

(3) Self-shielding factor table : Temperature = 300,800, 2,100 and 4,500 °K ; $\sigma_0 = 0, 1, 10, 10^2, 10^3, 10^4, 10^5$ and 10^6 barns ; shielding factor, f_{in} for inelastic scattering cross section; temperature dependent self-shielding factors for structural materials.

(4) Scattering mtrix : Scattering matrices for (n, n') and (n, 2n).

The benchmark test for the presently produced library, JFS-3-J2 was performed. The assessment for JFS-3-J2 was of the same degree as that for JFS-2.

5.2 Development of Core Analysis Method

For the analysis of sodium-void experiments in fast critical assemblies direction-dependent diffusion coefficients are often utilized to accurately treat the neutron streaming effect. A unified diffusion coefficient is derived which can be applied to both fuel cells and control rod positions, although the diffusion coefficients derived by Benoist and Seki are currently used for fuel cells and control rod positions, respectively. This diffusion coefficient is obtained by applying transport theory to a supercell containing few different cells, and therefore can treat the interference effect between different cells.

Theoretical and numerical comparisons among the unified diffusion coefficient and those by Benoist, Gelbard and Seki are performed for lattice cells contained in the ZPPR-3 modified phase 3. For a sodium-voided fuel cell adjacent to a voided sodium follower the interference is large and the diffusion coefficient of the fuel is increased by about 30% relative to that from the Benoist formula.

Reaction rate distribution measured in fast critical assemblies has been analysed to estimate the prediction accuracy and examine about the cause of the discrepancies between calculations and measurements. The selected assemblies are ZPPR-2, ZPPR-3 phase 2, MZB(2) and MZB(3). The analysis has been performed with use of the JAERI Fast set version 2. At first, the heterogeneity calculations were carried out to obtain cell averaged cross sections and they were used in two dimensional diffusion and transport calculations. To estimate the influence of the anisortropic elastic scattering effect, P_3 group constants have been produced and used in Sn calculations.

The results show that the reaction rate distribution could be predicted within an error of ±3% in the core regions. On the other hand, it was almost in the range of ±5% for the radial blanket, and the discrepancies between measurements and calculations seem to be systematic. It was shown that the contribution from below 1 keV was significant in the blanket region.

The effects of cell models for control rods with different ^{10}B enrichment and B_4C pin arrangements on their reactivity worths at the core centre of a fast reactor are studied. The fast reactor considered in the study is a simple cylindrical one which is provided modifying FCA VII-1 90Z, Effective cross-sections of various control rod cells (homogeneous cell, cylindrical cell and cluster cell) are produced by using a collision probability code PIGEON and JAERI-FAST V-II 25 energy group cross-section set. Diffusion and transport calculations are made using CITATION and TWOTRAN-II codes, respectively.

From a simple comparison between measurement and calculation of control rod reactivity worth, it can be expected that the difference in C/E between all the control rods with different ^{10}B enrichments would be within the expected experimental errors, when the effective cross-sections are produced so as to properly take into account of the cluster geometry of the control rods and also transport theory is applied to neutronic calculations.

5.3 Mockup Experiment and Analysis

A series of experiments has been made on FCA Assembly VIII-2 in order to refine the calculation method for the reactivity effect due to axial displacement of fuel/cladding in connection with a LMFBR core meltdown accident. FCA Assembly VIII-2 consisted of a central test region simulating the prototype FBR "MONJU" in axial dimensions and composition, and a uranium-fueled driver region.

Axially symmetric and asymmetric displacements of fuel were made in the central 3 x 3 drawers (equivalent radius 9.3 cm) with sodium expelled. Axial distribution of stainless steel reactivity worth was measured in the central 3 x 3 drawers of reference core. Stainless steel worth was also measured in the void region of each fuel displacement configuration. It was shown that reactivity change due to the fuel slumping is sensitive to the amount of stainless steel in the region where the fuel displacement takes place. The distortion in ^{239}Pu fission rate distribution was measured with a multiple chamber scanning system

(MCSS), about 60 fission chambers distributed in the test and driver regions.

Reactivity change for the respective fuel displacements was calculated by diffusion theory using JAERI-Fast Set Version II. The conventional calculation underestimates considerably the experimental values when the neutron streaming effect is large. Much improvement is achieved by using a modified diffusion coefficient for the void region. However, there still remains the tendency of underestimation which increases with expansion of the fuel slumping region to edge of the core.

The subsequent experiments on FCA have been carried out on the neutron penetration in the area of supporting grid and on the assessment of K_{eff} in the ex-vessel storage tank, in relation with the design of MONJU.

The DOE-PNC joint programme for large core critical experiments called JUPITER is in progress at ZPPR.

The Jupiter Phase 1 Program consists of ZPPR assemblies 9 and 10, intermediate-sized conventional two zone reactors. Experiments were completed on August, 1979. At the first Jupiter Analysis Meeting, each country presented detailed technical analysis of ZPPR-9 experiments. ZPRR-9 is 4,600-liter clean benchmark core with cylindrical boundary and no control rod positions. Following are the summary of review on the results of ZPPR-9 analyses.

(1) Criticality is underpredicted by ENDF/B-IV and is also slightly underpredicted by JENDL-2B

(2) Cell asymmetry configuration and half-inserted shim rods should be considered in the analysis of reaction rate distribution. C/E value of reaction rate generally shows a systematic radial bias in the core regions, and decreases sharply in the radial blanket region. Calculated reaction rate of $^{238}U\ (n,\gamma)$ is about 8% higher than measurement.

(3) UO_2 sample doppler reactivity was well predicted. Small sample worths of structural materials and U-238 were generally overpredicted, while other fuel materials were comparatively well predicted.

(4) The sodium void worth was overpredicted by roughly 10-20 percents. The current method for streaming correction, using Benoist's diffusion ocefficients, overestimates the effect.

(5) CRP (Control Rod Position) worth relative to fuel was overpredicted by 20-30 percents, which results in underprediction both of CR worth relative to fuel and of k_{eff} value. CR interaction effect was well predicted.

(6) ANL results of ZPPR-9 analyses showed same tendency as other ZPPR assemlies for all reactor parameters.

The Jupiter Phase I program is very useful in the point that it gives for the first time a thorough experimental data for the basic physics study of intermediate-sized homogeneous LMFBR. It is expected that the Jupiter experiments will make clear present status and problems of the data and methods.

5.4 Study of Large Heterogeneous Core

Large heterogeneous core (1,000 MWe) with the radial internal blanket has been studied to improve performance characteristics about the following points.

(1) Core specification

- The equivalent radius at the radial blanket outer boundary is equal to that of the homogeneous core. The assembly dimension is the same as the homogeneous core.
- The clad of the fuel pins of the heterogeneous core is thinner by 0.05 mm. (The pellet diameter of the heterogeneous core is 0.1 mm larger)

(2) Advantageous performance

- Shorter doubling time (Homogeneous : 34.5 years, Heterogeneous: 23.5 years)
- Smaller sodium void reactivity
- Uniform plutonium enrichment of the whole core fuel
- Smaller power peaking factor (XY plane power peaking factor, Homogeneous : 1.24, Heterogenerous : 1.14)

(3) Disadvantageous performance

- Large plutonium inventory (13% larger)
- Larger coolant flow rate (7% larger)

6. Research and Development for "MONJU" Components

6.1 Introduction

The design of "MONJU" has been conducted since 1968. This plant design is supported by many research and development works which has been performed since 1969. These experimental results of the research and development works has been fed very effectively "MONJU" design, and the data will be also connected closely to the detail design in future. The philosophy of research and development of plant components for "MONJU" is

1) to apply the useful R&D results of "JOYO" as much as possible and proceed to the next step for the proper items of "MONJU"

2) to have the research and development of plant components for power generation plant without such R&D for JOYO.

3) to have the R&D works increasing operational safety and reliability of "MONJU".

The numbers of the principal research and development items of plant components for "MONJU" are about thirty, and the numbers of test items are more than one hundred. These items are devided into mainly four categories.

1) Mock-up test of main components.

2) characteristic test of plant components.

3) tests for maintenance and inspection of components and piping.

4) fabrication tests of complicated plant components and check of complicated pipings layout, etc.

These experiments are carried out mainly at the O-arai Engineering Center of PNC and partly at the maker's laboratories.

6.2 Outline of R&D Works of Plant Components

6.2.1 Mock-up Tests of Main Components

Mock-up tests of main components for "MONJU" are as follows.

(1) Refueling machine.

(2) Fuel transfer mechanism.

(3) Control and safety rod drive mechanism.

(4) Back-up safety and drive mechanism.

(5) Mechanical pump.

(6) Linear induction pump.

(7) Valve.

(8) Cold trap.

(9) Failed fuel location detection system.

(10) Ex-vessel fuel storage tank.

6.2.2 Characteristic Tests of the Structural Components

These characteristic tests include the partial tests of components, small scale component tests, water and sodium flow tests, thermal shock tests by sodium and preventing tests for sodium vapour condensation, etc. These items conducted for "MONJU" are as follows.

	scale of model
(1) Water flow test in the reactor vessel.	1/2, 1/5
(2) Tube seal for rotary plug.	1/5, 1/1

(3) Preventing tests for sodium vapour condensation. 1/5

(4) Flow test for in-vessel instrumentation partially 1/1

(5) Entering gas test. 1/6

(6) Cavitation test. partially 1/1

(7) Decay heat removal test for fuel transfer machine.

(8) Integrity test for primary container equipment against sodium leak.

(9) Water flow test of IHX.

(10) Thermal shock test by sodium (Mixing tee, tube sheet of IHX, valves, outlet nozzle of RV, etc.).

(11) RV upper plenum stratification test.

(12) Nak flow test.

(13) Natural circulation.

(14) Steam generator, IMWSG, 50MWSG, Sodium-water reaction test (SWAT-1, 2, 3).

(15) Vibration test of core.

(16) Preheating test of components and piping.

(17) Sodium bearing and sealing test for EVST

6.2.3 In-service Inspection Equipment

An effort is being made to develop In-service Inspection Equipment for reactor vessel and its inlet nozzle. The equipment is required to provide functions to inspect the object surface with TV camera within high temperature guard vessel cavity. Cooling system for TV camera and some electrical parts enable to work in high temperature environment has been developed. Also, trial fabrication and testing of the equipment is underway.

6.2.4 Fabrication Test of Complicated Structural Components and Check of Complicated Piping Layout.

Partial mock-up model will be fabricated and inspected for very narrow space and difficult work-ability or very complicated arrangements of piping layout. Rotary plug of this kind is being fabricated.

6.3 Examples of R&D Works of Main Plant Components.

6.3.1 Mock-up Test of Main Circulating Pumps

Endurance test is being performed on a full scale model of the primary main circulating pump. The pump has been operated for a total of approximately 10,000 hours up to the present without any abnormality. Coast down test has been conducted to fulfill the necessity in the plant dynamics analysis including thermal transient analysis in the heat transport system. Also test on starting after hot stand-by has been conducted. Low speed driving gear using one way clutch has been trially fabricated. The gear is being mounted on the model pump to subject to running test.

6.3.2 Intermediate Heat Exchanger

As for the intermediate heat exchanger, a full scale but 1/6 sector model was fabricated to establish a favourable flow distribution so that only less number of tubes located outside the tube bundle work in tensile stress. Flow test in water is being performed on the model. In paralell to the test, thermal transient test on the upper tube sheet and shell construction has been conducted in sodium on a separate simplified model. On the other hand, in sodium life test was carried out on trially fabricated bellows to be used in top of the down comer pipe. Also on the tube to tube sheet welding, an investigation and trial fabrication has been made to select the most suitable method to enhance economy, easiness to inspection and reliability.

6.3.3 Rotating Plug Seal

Rotating plug seal of "MONJU" has both freeze seal and elastomer seal as back up.

The elastomer seal characteristics test by reduced scale model has been conducted and the seal characteristics test with freeze and elastomer seal by full scale model is being carried out. Test results by the reduced model shows that, inflatable tube seal made by Viton is recommended, and leak rate is confirmed to be about 10^{-3} standard cm^3/sec.

6.3.4 Hydraulic Test of Mock-up Core Structure

A half scale model of "MONJU" reactor vessel combined with reactor internals, fuel assemblies, control rod was constructed, and hydraulic tests have been conducted. It has gotten clear that the flow of upper plenum constists of both a main flow and two recirculation flow and that the flow mixing in under plenum is best, when the issuing angle of inlet nozzle is 15°.

Pressure and flow distributions in high pressure prenum and low pressure plenum have been investigated.

6.3.5 Mock-up Test of Fuel Handling Machine

Fuel handling for "MONJU" adopts a type of single rotating plug fixed arm pantograph. Full mock-up handling machine with fixed arm pantograph was constructed, and functional test and endurance test have been performed in sodium, since 1974. Besides, positioning device for fuel assembly which shows the position of the gripper of fuel handling machine in operation has been studied by test in sodium.

6.3.6 Steam Generator

(1) 50 MW Steam Generator

After the 3,400 hours performance tests of 50 MW SG #1 were finished in April 1975, it was disassembled for inspection.

50 MW SG #2 was constructed and the performance test began in January 1976. The accumulated operating time of SG #2 are 9,500 hours for evaporator and 3,300 hours for superheater.

Through the evaluation of experimental data and the comparison with analysis, the heat transfer and flow dynamics design method for "MONJU" steam generator was completed.

Research and development for solving some problems on materials, design and fabrication, operation and control, water leak detection, maintenance and repair of steam generator are being carried out.

A practical feasibility test of these research and development works were executed using 50 MW SG #2 evaporator under the assumption of occurrence of water leak. These works include drawing-out of heat transfer tube bundle, cleaning of sodium, in-service inspection of heat transfer tubes by eddy-current and ultrasonic method, tube-plugging by explosive technique and replacing some tubes with new one etc.

After the feasibility test was completed in October 1980, an endurance test of 50 MW SG #2 was started again with a starget of another 10,000 hours operation.

The verification test of "MONJU" auxiliary cooling system will be carried out using 50 MW SG test facility after some modification including an air blast cooler.

(2) SODIUM-WATER REACTION STUDY

1 LEAK HOLE ENLARGEMENT AND LEAK PROPAGATION STUDY

A preliminary test of leak hole enlargement was performed with 2 1/4Cr-IMo and stainless steels in the micro-leak region of Water. Metallurgical examinations were conducted in order to explain the self-wastage and self-plugging mechanisms.

Twelve intermediate leak tests in total were carried out using the SWAT-1 rig in order to observe leak propagation by the wastage and the burst due to internal pressure and tube wall softening by overheating.

Seven water injection tests were carried out using the SWAT-3 facility in order to observe the sequential phenomena of leak propagation in the tube bundle due to initial small and intermediate leak of water. Post-test examinations of leaked tubes are under way to clarify the extent and degree of failure/damage of the tube bundle.

2 LEAK DETECTOR DEVELOPMENT

The in-sodium hydrogen meter of the PNC design is being endurance-tested over 10,000 hours till now, and little trouble has been experienced during this period.

Several hydrogen meters modified to the original PNC design, and the O-H module leak detector were manufactured and are being function-tested.

The cover gas hydrogen meter (Ni membrane type) is being performance-tested using the SWAT-2 loop. And the test is also underway to obtain the data of parameters which dominate the hydrogen concentration in the covergas area of the sodium ocvergas system.

3 LARGE LEAK TESTS

Seven leak tests in total were carried out using the SWAT-3 test facility which is a 1/2.5 scale model to the MONJU SG system. The main objectives of these tests are :
to assure the integrity of the helically coiled heat transfer tube bank (six cases, different design, different leak rate, different leak position); to assure the integrity of straight tube bank in the down-comer region (one case) ; and to obtain the data of pressure and hydraulic transient during the large leak sodium-water reaction.

Using the SWAT-1 rig which is scaled to 1/8 of the MONJU SG system, five tests were performed in order to obtain the data of transient heat transfer coefficient and thermal shock generated in the pipe wall of the pressure relief line which has no preheating device and is at room temperature condition before sodium and hydrogen release.

4 COMPUTER CODE ANALYSIS

Validity check of the large leak code SWACS has been continuing using the SWAT-3 data.

The SWAC-10 code, which estimates the delay time needed to detect a water leak in the MONJU SG in the micro- and small leak regions, was revised to be able to simulate a water leak in the down commer region.

7. Sodium Technology

7.1 Material Tests in Sodium

Creep-rupture tests under internal pressure were carried out continuously for MONJU fuel cladding tubes of every recent year's trial production.

Mechanical tensile tests after exposure to high temperature sodium, as well as corrosion tests under sodium environment have been continued for MONJU fuel cladding candidate materials which have been newly developed.

A test is being carried out on the corrosion caused by reaction products formed with the sodium leaking to the outside environment.

The first step test was completed. Results showed that corrosion took place in all cases of this test.

It is likely that the dwell time of the reaction product affects to the corrosion rate of the contacting steel surface.

As the second step, a test is planned with lowered humidity (about 1,000 ppm H_2O) in the nitrogen atmosphere.

In-NaOH stress corrosion cracking test was completed. SCC susceptibility was studied on SUS 304, 316 and 321 with variables of temperature, NaOH concentration and stress.

Self-welding, friction and corrosion tests are being performed on some hard-facing materials.

Recent results of self-welding test showed that pre-exposure to sodi sodium affected to the breakaway stress and that Inconel 718 had the least dependence of breakaway stress on temperature change and dwell time among tested materials.

7.2 Flow and Heat Transfer

A study on heat transfer from the liquid sodium surface to the shield plug through cover gas space is now in progress.

The performance tests on some models of the convection barriers are now in progress.

The convection barriers are designed to prevent sodium vapor deposition on the annular gaps between the reactor vessel and the shield plug.

Water cavitation tests were completed on the entrance nozzle of fuel subassembly and the flow controlling parts of core internals of MONJU.

Tests are to be carried out on the cavitation for the whole subassembly with improved entrance nozzle in near future.

As the preliminary step of sodium cavitation test, development of an acoustic detection technique for the inception of cavitation in sodium started in November 1977. The feasibility of this technique has been confirmed.

As the next step, cavitation inception tests in sodium are being planned in order to verify the inference drawn from water test.

7.3 Behavior of Radionuclides in Sodium

The objective of this area is to study the mass transfer of radioactive corrosion products in sodium for the purpose of the evaluation of the accessibility to the LMFBR primary coolant system.

The third radioactive mass transfer test is being conducted in the Activated Material Test Loop-II. Major test conditions of this test are as follows :

cold trap temperature	: 120 °C
test duration	: 4,000 hrs
sodium temperature	: 650 °C (hot leg)
	: 400 °C (cold leg)
sodium velocity range	: 2.3m/s ~ 3.8 m/s

This test is finished in Oct. 1980.

Another test was conducted by using a small pot, containing about 300g of sodium into which Cobalt Chlolide ($CoC\ell_2$) was dissolved, in order to obtain information on the deposition behavior of radioactive corrosion products such as ^{58}Co and ^{60}Co in liquid sodium.

The deposition rates of ^{58}Co to specimens (SUS304 and Ni) in stagnant isothermal sodium were measured at the temperature between 160 and 470 °C. It was observed that the ^{58}Co diffused deeply into nickel specimen. Diffusion coefficient of ^{58}Co in nickel was obtained to be about 2×10^{-13} cm^2/s at 465°C.

A test loop containing about 4.9 kg sodium was constructed for conducting experiment on the trapping method of ^{137}Cs in sodium in July 1980.

After pretesting ^{137}Cs, trapping test will start in Oct. 1980.

7.4 Sodium Chemistry and Sodium Purification

Performance test of an experimental model of on-line gas chromotograph for JOYO cover gas monitor was done in order to determine the measuring condition and to confirm the durability.

Upon the basis of this experimental result a prototype on-line gas chromatograph was fabricated in 1979. Performance test of this type is planned to be tested in Mar. 1981.

If no difficulty is found, this will be applied the secondary cover gas system of JOYO.

Development of the other on-line impurity indicators such as plugging indicator, hydrogen, carbon, and oxygen meter, continue at O-arai Engineering Center.

A study on hydrogen behavior in sodium was performed by using a nickel membrane diffusion-type liquid metal hydrogen meter (PNC-II type).

The hydrogen solubility (C_H) in sodium and the Sievert's constant (Ks') obtained were $\log C_H = 5.98 \pm 0.18 - (2807 \pm 78)(1/T)$ in the range of sodium temperature of $100^{\circ}C$ to $250^{\circ}C$ and $Ks' = 0.72 + 22.5/T - (6.2 \times 10^{-4} - 1.7/T)$ Co in the range of sodium temperature of $250^{\circ}C$ to $550^{\circ}C$ respectively, where, T ($^{\circ}K$) indicates sodium absolute temperature and Co (ppm) indicates oxygen concentration in sodium.

Cold traps of FBR secondary system are estimated to need to be exchanged every several years because the cold traps will be plugged with the hydrogen diffused through the heat transfer tubes of the steam generator. A feasibility study on regeneration of such cold trap was conducted by evacuating gas phase over the sodium surface after dissolving the trapped hydrogen in sodium by increasing the temperature of cold trap.

The results show that "MONJU" secondary cold trap temperature is required to be about $360^{\circ}C$ in order to regenerate it within a month.

A large scale test on the regeneration system of the secondary cold trap is planned to be performed in 1981 through 1982.

A feasibility study on regeneration of "MONJU" primary cold trap is now in progress, using the heating method.

8. Development of FBR Instrumentation

8.1 Nuclear Instrumentation

8.1.1 In-Core Fission Chamber

Development of micro fission chamber has been nearly completed to provide for the instrumented subassemblies of "JOYO".

As the final stage of this development,

(i) Six chambers have been irradiated in the Japan Material Test Reactor for examination of their irradiation effects, and these chambers were being disassembled in order to investigate the irradiation effect in the hot laboratory.

(ii) High temperature and thermal cycle tests have been carried out for other six chambers for examination of their reliability.

8.1.2 Ex-vessel Fission Chamber.

High performance fission chamber having sensitivity of 0.3 cps/nv and operating up to at $550^{\circ}C$ has been developed.

8.1.3 Ex-vessel ^{10}B Lined Proportional Counter.

A ^{10}B lined proportional counter was tested under the temperature up to $200^{\circ}C$ last year. Temperature dependency was not observed except for the background pulse rate. Neutron sensitivity was about 12 cps/nv and kept the same value even when the counter was exposed to gamma flux of 200 R/h.

An improved ^{10}B lined proportional counter is now being irradiated in the Japan Research Reactor -4 for examination of reliable performance. Total neutron irradiation is to be 2.5×10^{17} nvt.

8.2 Failed Fuel Detection and Location

8.2.1 FFD

For cover gas monitoring system, a moving wire type and a fixed wire type precipitators have been developed, and the results of their performance tests were excellent.

Two types of precipitators and cover gas γ-monitoring system are being tested in the Japan Research Reactor-3 (JRR-3) for studing their reliability.

8.2.2 FFDL

The tag gas system has been developed for locating the failed fuel subassembly. The neutron irradiation test of tagging gas is to be carried out at the irradiation test subassemlies of "JOYO" MK-II core.

The following subjects are to be investigated within next two years.

(i) The membrane separating equipment which is planned to be used for concentrating tagging gas in primary cover gas system is to be tested in JRR-3.

(ii) To evaluate the yield of fission gas and cross section of tagging gas nuclides, the computer code "TAG" will be improved.

8.3 Early Warning System for Fuel Failure.

8.3.1 Temperature Measurement.

The performance and reliability of C.A. (Chromel-Alumel) thermocouples under irradiation in JMTR (Japan Material Testing Reactor) were examined and failed ones were inspected in the hot-laboratory. Thermocouple failures frequently occurred in the initial operation.

8.3.2 Flow Measurement

New type eddy current flow/temperature sensors were developed and tested in a sodium loop.
Durability tests in high temperature atmosphere and irradiation tests for these sensors will be carried out.

Flow blockage and gas bubble detection tests are being performed by using the flow sensors close to the seven mock-up outlet of MONJU subassemblies.

8.3.3 Others

An acoustic detection system is being developed for purpose of detecting some anomalous sound, in particular the onset of local boiling in the core. Experiments and analysis are being performed on acoustic propagation in subassemblies and core structure.

A reactivity meter with the Kalman filter is being studied and fabricated for application to in-core diagnoisis system of MONJU. This system is being demonstrated for evaluation of reliability by simulation study using JOYO experimental data.

8.4 Process Instrumentation

8.4.1 Sodium Flow Meters for Large Piping.

Since the adoption of the permanent magnet flowmeter was determined for the measurement of flow in the primary and secondary cold leg of MONJU, flowmeter response and calibration method became a major concern. Some tests relate to these items are in progress for their verification at the O-arai Engineering Center PNC.

The durability test for a 12-inch ultrasonic flowmeter was carried out. The 24-inch ultrasonic flowmeter is being tested, in static sodium to be applied for the calibration of the electro-magnetic flowmeter in "MONJU" plant.

8.5 Surveillance

8.5.1 Under Sodium Viewer

Development of Under Sodium Viewer (USV) is being developed for two types designed for different purposes. In-sodium characteristics test for the horizontal type USV system with the acoustic reflector assembly which functions as a acoustic sweeper has carried out using the 3/10 scale reactor core model of "MONJU".

As for the vertical type USV system which visualizes the upper surface of the core barrel, imaging technique will be demonstrated using the digital processing unit with a CRT and a transducer drive mockup. The fabrication of these equipments are now in progress.

8.5.2 Sodium to Gas Leak Detection System

A sodium ionization detector and a filter are to be fabricated and evaluated for the leak rate order of 100 g/h in the environment which simulates a primary cell of MONJU. Besides detector, other components including sampling system and sodium aerosol generation are also investigated in these tests.

8.5.3 Displacement Sensor

The sensor based on an eddy-current principle whose purpose is to measure the gap in the hydrostatic bearing of a sodium pump was fabricated and tested in the air environment of up to 450°C and in the thermal transient of -2°C/sec. The in-sodium calibration and endurance test on the pump mockup will be performed succeedingly at the O-arai Engineering Center.

9. Fuel and Materials

9.1 Fuel Fabrication

The fabrication of "JOYO" MK-II fuel is now being carried out at the modified PNC Plutonium Fuel Fabrication Facility.

The detailed design of the "MONJU" fuel assembly is almost fixed, and the construction of the PNC Plutonium Fuel Production Facility for "MONJU" fuel is scheduled to start in fiscal year 1981. PNC gained the Pu handling technology through the fabrication of "JOYO" core fuel, and those experiences are now being applied to the new technology development. The "MONJU" fuel fabrication plant utilizing as much remote technologies as possible is being designed in detail, and some of remote handling components are being developed.

9.2 Fuel Pins

Fuel pin performance code CEDER has been developed by combining and improving some performance codes which had been developed in PNC. The CEDER code is now being used for the design of "MONJU" fuel and the analysis of "JOYO" MK-II fuels.

Inner surface coating on cladding tube is currently being developed in order to prevent the f.p. -cladding chemical interaction, and 1.5 meter long tubes were successfully coated by Titanium using vacuum deposition method, and tubes were supplied for irradiation tests.

Reaction products of fuel pellet, fission products and cladding filled in the gap of irradiated fuel pins were examined by scanning electron microscope and electron probe micro-analyser to analyse FCCI (Fuel Cladding Chemical Interaction) and CCCT (Clad Component Chemical Transport) mechanisms.

9.3 Cladding Tubes

The "JOYO" MK-I fuel was composed of some 10% cold worked 316 stainless steel tubes, however, 20% cold worked 316 stainless steel tubes will be used for "JOYO" MK-II and "MONJU" fuels.

All the developmental cladding tubes were subjected to non-destructive inspections such as ultrasonic flaw detection and dimensional measurements, and destructive tests to clarify mechanical properties at room and high temperatures. Those data have been all accumulated using a computerized system (Fuel Data Banking System) for future reference use.

Modified 20% cold worked 316 stainless steel tubes have been developed for "MONJU" and "JOYO" MK-II fuels. The modification was made by controlling the concentration of Ti, Nb, B and P, in order to get better materials against creep and swelling. These materials were screened by ion bombardment and are being irradiated in Rapsodie and Phenix.

9.4 Subassembly

Test fabrications of mock-up fuel subassemblies for "MONJU" were carried out and the fabricated subassemblies were subjected to a sodium endurance test for more than 5,000 hours.

Experiments and analysis of inter-subchannel mixing and edge flow effect in wire-spaced pin bundles, bending and compression tests of partial mock-up fuel subassemblies, and seismic tests of mock-up fuel subassemblies were completed.

9.5 Fuel Irradiations

Irradiation tests of fuel pins and subassemblies are planned and being performed in some foreign fast reactors, such as DFR, Phenix, Rapsodie, EBR-II and FFTF, in order to assure the performance of "MONJU" and "JOYO" MK-II fuels.

Rapsodie PNC-10 and Phenix P-3 tests were planned for the irradiations of subassemblies of "JOYO" MK-II and "MONJU", respectively. Rapsodie PNC-6 and 7 are the pre-irradiation for safety experiments in Siloe reactor, and power-to-melt test of

"MONJU" type fuel pins were performed in EBR-II. The terget burnups of these irradiations are the same as those of "JOYO" MK-II fuels and "MONJU".

Since the startup of "JOYO", nineteen fuel assemblies were examined at the hot laboratories in O-arai for monitoring the performance of the assemblies. The post irradiation examination of "JOYO" MK-I driver fuels and blanket fuels showed no special problems.

9.6 Other Materials

Since the highly Boron-10 enriched material is used for both "JOYO" and "MONJU" control rods, researches on the recovery of Boron Carbide from used control rods and the re-enrichment of Boron-10 were conducted.

The evaluation of the vented-type control rods in the water loop and the in-sodium tests was completed.

Post irradiation experiments of reference control rods of the "JOYO" MK-I core showed good results.

At the same time, new types of control rod materials such as Gd and Eu are irradiated in JMTR.

"MONJU" design calls for the use of serpentines concrete as a shielding material adjacent to the reactor vessel. Approval tests of serpentine concrete were completed.

9.7 LMFBR Fuel Reprocessing

The principle design of the Pilot Plant for LMFBR spent fuels is now under way and the construction of the pilot plant is scheduled to start in 1986.

The Chemical Processing Facility (CPF) for research and development of LMFBR fuel reprocessing is now being constructed and is scheduled to be completed within fiscal year 1980.

The cold tests of voloxidation, rapid contactor, off gas treatment, fuel ressolution and so on have been carried out. The safety design of facilities such as the extraction apparatus, and the study on remote maintenance technique are under way.

10. Structural Design and Materials

10.1 Development of high temperature structural design analysis method

10.1.1 Inelastic structural analysis program

i) General purpose inelastic structural analysis program

The general purpose inelastic structral analysis program FINAS has been developed since 1976 and partly been used since 1978 for the design analysis and the development of simplified methods of inelastic analysis.

ii) Inelastic analysis program for straight pipes

The special purpose program TEPC which was developed for creep-ratcheting analysis of straight pipes has been utilized to make several design charts.

10.1.2 Development of simplified methods of inelastic analysis and supplementary dešign rules

Although detailed analysis method are powerful tools to evaluate stresses and strains, it is also important to develop a wide variety of simplified methods for parametric design evaluations.

On the other hand, for qualification of the results by detailed or simplified method of inelastic analysis, it is necessary to specify the analysis procedure and the material data in detail.

The following research and development were carried out during 1977 and 1979.

i) Simplified creep damage evaluation methods for structural discontinuities.

ii) Application of reference stress method for evaluation of elastic follow-up strain.

iii) Simplified inelastic analysis methods of perforated plates.

iv) Development of simplified evaluation method of local stresses at piping support.

v) Development of simplified analysis method for class 2 components.

iv) Establishment of design application procedure of inelastic analysis method

vii) Development of constitutive equations for structural materials

viii) Analysis of inelastic piping benchmark problems

ix) Analysis of inelastic nozzle-to-sphere benchmark problem

x) Development of postprocessors based on high temperature design code for Class 1 Vessels and piping of MONJU.

10.2 Structural material test

Fig. 10.1 shows the change of high temperature structural design standards and structural material test programs. From this figure, it can be seen that these tests were under way in parallel with the construction of FBR in every country. In Japan, structural material test program phase II has been conducted under coordination of PNC. Research and development on structual materials have been reflected on the design of "MONJU" as shown in Fig. 10-2.

a) In-air structural material test

Fig. 10-3 shows the summary of the in-air structural material test program. As the result of Step I plant were obtained many basic data that were available to the design of "MONJU" Now, Step II plan are under way.

According to increasing needs for structural materials data, the test program was expanded and the installation of fatigue and creep testing machines has progressed since 1977 at FBR component fabricaters. A new material strength test laboratory facilitated with ten high temperature low cycle fatigue testing machines and eighty creep testing machines will be constructed by the end of 1980 fiscal year at O-arai Engineering Center.

b) Structural material tests in Sodium
(General Purpose)
Predict nonmetallic element (mainly C) compositional changes in austenitic and ferritic steel in sodium systems and evaluate the effect of these changes on the mechanical properties of the structural materials.

The following research works are in progress :

(1) Corosion and mass transfer
Mass transfer tests on SUS304, SUS316, SUS321 and 2 1/4Cr-1Mo steel

(2) Carbon transfer
Carbon transfer test on bimetallic systems simulating the MONJU Secondary systems

(3) Mechanical strength tests in sodium
determine the effect of the sodium environment on the mechanical properties of structural materials, usch as tensile strength, creep, low-cycle fatigue and stress relaxation.

According to increasing needs for these data, the test program was expanded and so the new facility was constructed at O-arai Engineering Center in the middle of 1979.

c) Structural material irradiation - effect test

The tests on domestic 304 stainless steel have been conducted to ensure the safety of reactor vessel and internal components under the irradiated condition. The materials mainly irradiated at the Japan Material Test Reactor of JAERI are tested at PNC O-arai Eingineering Center.

Irradiation tests using Experimental Fast Breeder Reactor "JOYO" are considered.

d) Data banking and retrieval system

Material tests are conducted following the standard manual "FBR Metallic Materials Test Manual" including various data sheets and standard graphs. A data banking and receival system which handles the resulted data are under development.

10.3 Structural Element and Component Tests

In order to evaluate the adequacy of high temperature design rules and analysis methods and also to confirm the integrity of the actual components, the following structural element and component test have been performed.

a) Creep buckling test of elbow

* Clarification of elastic-plastic-creep deformation behavior of an elbow component
* Determination of the condition under which creep buckling occurs on this component

  (including the development of the simplified design rules for buckling of elbow)

* Verification of the applicability of the inelastic analysis method.

b) Creep fatigue test of elbow

* Investigation of creep-fatigue damage evaluation in a practical component

c) Thermal transient test of elbow (at ETEC)

d) Creep ratcheting test of straight pipe

* Verification of inelastic analysis method
* Confirmation of Bree's diagram
* Thermal ratcheting test finished
* Creep ratcheting test finished

e) Creep fatigue tests of elbow in sodium

f) Thermal fatigue test of circumferencial but-weld joints and dissimilar metal joints.

g) Basic structural element behavior tests

* Beam cyclinc bending
* Rod with stepped crossection in cyclic tension and compression
* Notched plate in cyclic tension and compression
* Curved beam in tension and bending

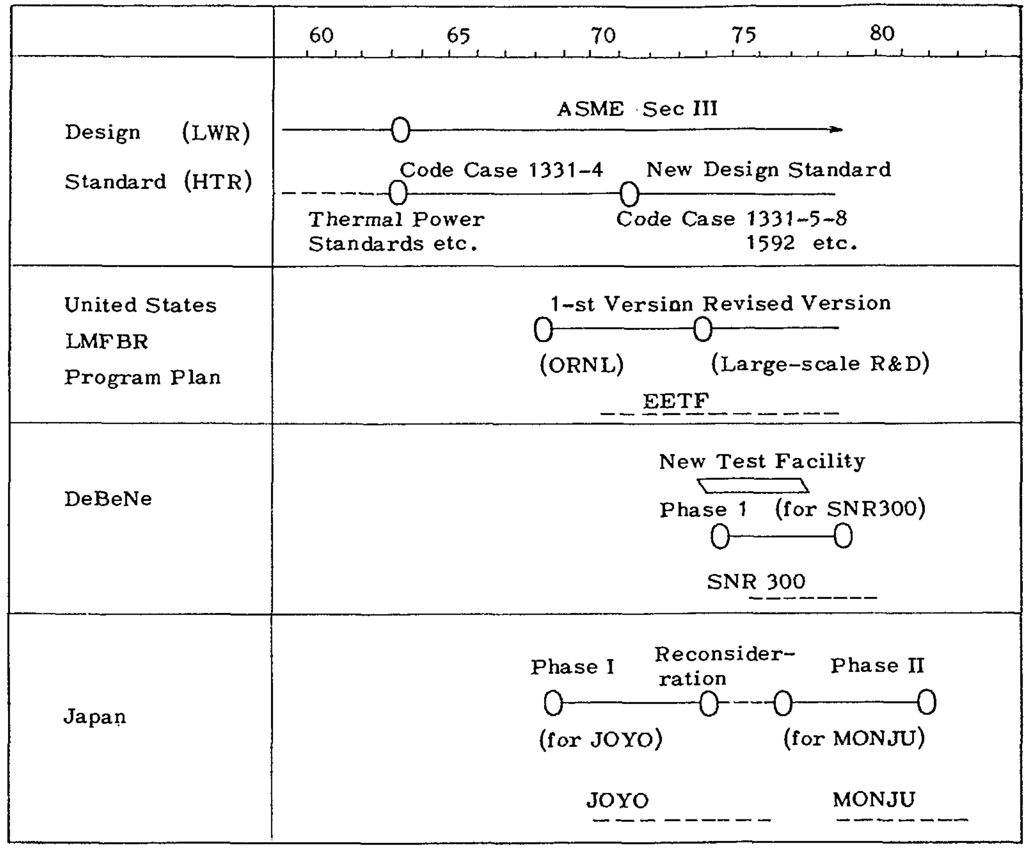


Fig. 10.1 Change of High Temperature Structural Design Standards snd Structural Material Test Program

After the new design standard was established in 1971, every country was obliged to change and expand largely the structural material test program in order to obtain back-up data. These tests are under way in parallel with the construction of FBR in every country.

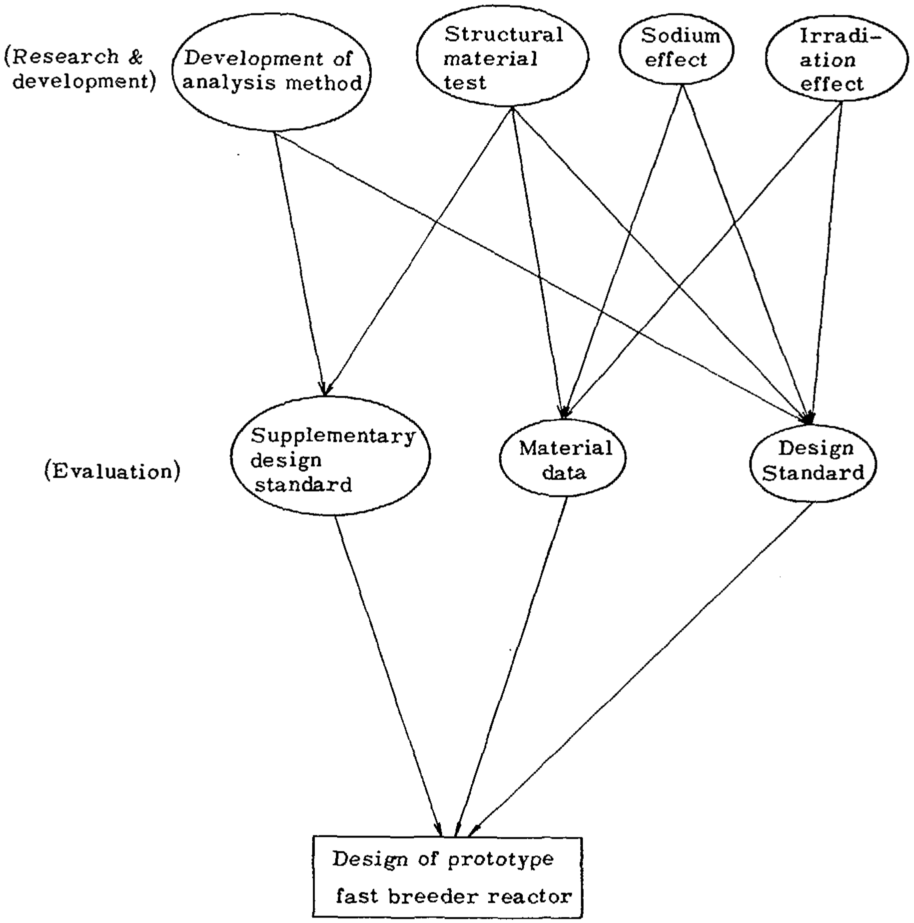


Fig. 10.2 Research and development of structural materials and their reflection on the design of fast breeder reactor

Step		Material	Product Form	Test
Step 1 (1977, 1978)	1. To obtain basic data of base metal 2. To obtain basic data of weld metal 3. Others · Test on pre-strained material · High speed tensile test on aged material · Material test on Inconel 718	SUS304 SUS316 SUS 321 and their weld metal, welded joint	Plate Forged Tube	· Tensile test · High speed tensile test · Creep test · Fatigue test · Creep-fatigue test · Relaxation Test
Step II (1979, 1980)	1. To obtain basic data of base metal 2. To investigate the test procedure on welded joint 3. To obtain basic data of welded joint 4. Others · High cycle fatigue test · Development of creep equation · Test on inelastic behavior			

Fig. 10.3 Summary of the In-air Material Test

11. Safety

11.1 Core Safety

11.1.1 Sodium Boiling

Two series of experiments were conducted in a 37-pin bundle (bundle 37G) during the period of mid 1979 to October 1980. These are confirmation experiments for the experiments performed by the previous 37-pin bundle (bundle 37F) during the period of mid 1977 to early 1979.

The first of these is LOF-type transient boiling tests. The experimental data of the bundle 37F is being analysed.

The second is low heat flux boiling tests. The experimental data of bundles 37F and 37G show that dryout did not occur in the quality range smaller than 0.5.

Water experiment has been started to investigate the behavior of large bubble which may be formed in an HCDA. A test vessel has been manufactured for an HCDA bubble experiment in sodium.

11.1.2 Fuel Failure Propagation

(1) Pin contacts

The PICO-2 code was developed to analyse the temperature rise measured at the low flow region behind a contact region.

(2) Local blockage

A series of experiments in a 61-pin bundle were performed. The bundle had a central blockage which corresponds areawise to 38%. The temperature rises at the blocked regions, local boiling and FP gas release phenomena were measured in these experiments.

The final local blockage experiments in a 91-pin bundle will start by the end of 1980. A stainless steel plate blocks a half of whole flow area.

(3) Water loop experiments

Many tests were conducted to study velocity profiles, pressure distributions, residence times and FP gas release phenomena behind permeable and impermeable blockages. These data were applied to verify the UZU code and sodium experimental results.

(4) Anomaly detection

The performance of LOCAD (Local Core Anomaly Detection) system is being examined by the out-of-pile local blockage experiments.

11.1.3 Molten Core Material Interactions

(1) Fuel Coolant Interaction Tests

Five out-of-pile tests in total were performed to investigate the FCI phenomena during the TOP accidents in an LMFBR since December 1979.

A description of a test section is shown in Ref. 1. Experimental conditions simulating the thermo-hydraulic conditions in an LMFBR are shown in Table 11.1. The power of electrically heated fuel pins was controlled at steady state of 130 or 160W/cm and then a power transient was supplied to each pin individually.

The experimental results are shown in Table 11.2. The amplitude of maximum pressure pulses was measured in the range of 0.5 to 1.3 MPa with full-width at half maximum of 0.3 to 0.6 msec. X-ray photographs in post-tests show some amount of the ejected fuel adhered to the cladding and wrapper tube around the zone of failure. In addition, its details will be clarified by the post disassembly test of the test section.

Future plans are to conduct the experiments with pre-pressurized fuel pins simulating pre-irradiated pins and to observe fuel motion in two directions by using two X-ray cinematographs and high speed cameras.

(2) Fuel EOS

High temperature states of fuel were generated by impinging a laser beam on UO_2, and Cs, Ba, Ag or Sn impregnated UO_2 pellet surfaces. The vapor pressures were measured with a torsion technique and the surface temperatures were estimated from the laser energy density absorbed by the pellets. As a result of the experiment, it was found that the saturated vapor pressure of Cs Ba, Ag or Sn impregnated UO_2 is higher than that of bare UO_2, and the pressure is highest in the case of Cs.

11.1.4 Transient Undercooling Tests

The tests were originaly aimed at confirming the integrity of "MONJU" cladding (50,000 MWD/T and 70,000 MWD/T) under a locally flow blocked (800 to 960°C, 24 hours) condition and an LOPI (160 to 200 C/sec for a few seconds) condition in Siloe Reactor.

Test pins for pre-irradiation have been already shipped to Rapsodie and the pre-irradiation is expected to begin by the end of the year.

11.1.5 Transient Overpower Tests (CABRI)

PNC is a junior partner of the Joint CABRI project and stations one engineer at Cadarache. The experimental analyses both for pre- and post- test predictions at O-arai/PNC have been greatly contributing to the code verification of both PAPAS and SAS3D.

11.1.6 Large Scale In-Pile Tests (Treat and SFSF)

PNC stations two engineers at ANL who participate precalculations for future TREAT/SLSF tests under planning. A pre-calculation is now progressing at O-arai/PNC for the SLSF W1 test by using PNC's own codes in order to prepare the forth coming post-test analyses which are scheduled to begin right upon the receipt of the data from DOE.

11.1.7 PAHR Out-of-Pile Tests

Two series of experiments are being performed.

The first is a thermal convection experiment under the high Rayleigh number region in a cylindrical electrolytic bath with internal Joule heating. The experimental heat transfer coefficients are determined.

The second is the penetration experiment of a dielectrically heated liquid layer into an underlying wax layer. The penetrating pattern and rate are investigated.

11.1.8 Accident Analysis Codes

(1) Whole core Accident Analysis

Major efforts have been forcused on developing an intergrated code system (PAPAS) for analyzing the initiating phase of a CDA, and for a better interpretation of CABRI.

The code system has been constantly applied to CABRI post-test analyses and then modified through the comparison between experimental results and calculations. Recently, the PAPAS upgrade in the area of transient fuel pin failure mechanics has been almost completed with respect to the replacement of the fuel pin mechanics module with a fast running model FLCAST which is capable of treating cracked fuel and transient FP release, etc.

Progress of stand-alone code development is summarized as follows : FCI codes (EULFCI and ACTFCI) was completed and is being used for out-of-pile FCI Test analyses; 2D space-time code 2DFEM was completed; code coupling of various PAHR codes (PARCO, PTAR and PANAC) is continuing for the completion of the first version of an integrated PAHR analysis code system.

On the other hand of domestic developmental effort, various foreign codes were imported from U.S. Those code are solely related to highly unlikely accident of MONJU reactor (SAS3D and SIMMER for HCDA and SSC for LOPI).

(2) Local Faults

A series of versatile 3D thermo-hydraulics codes in an LMFBR subassembly are being developed : ASFRE for single phase and TOPFRES for two-phase including sodium boiling. The verification of those codes has been made by comparing the calculated results with out-of-pile simulation tests for various local faults accidents both in water and sodium. A result of the ASFRE rediction is shown in Fig. 11.1 for the LMBWG international benchmark calculation on local flow blockage test. The both codes as well as a 2D sodium boiling code BOCAL are also planned to be in use for SLSF in-pile test interpretation.

(3) System Support Utilities

Large Code System Maintenance, LAXYM, was transmitted to DOE and NRC.

11.2 Structural Safety

11.2.1 Shock Structure Experiments and Analyses

(1) The second experiment on a pressure pulse propagation along a short pipe (6^B x 3^m - SUS304) with an elbow was performed in April, 1980. Mechanism on peak pressure attenuation along the pipe became clear considerably.

One of the findings in this experiment is that the peak pressure of a shock wave was not attenuated after passing an elbow part. The detailed results are now being analysed. The calculated results by PISCES-2DELK at its straight part show a good agreement with the measured data as to a pressure history and dynamic strain e.t.c.

The mechanical strength of test material at high strain rate, to be considered in calculation (Code : PISCES-2DELK) is now understudy.

Near future, shock wave attenuation test will be performed with a piping model including several elbows which simulates the hot-leg piping of a primary system in MONJU. A computer Code PISCES-3DE will partially be used in its analysis.

2) High tensile tests using explosives has been performed on SUS-316 material which was aged in 10,000 hrs at elevated temperatures. Maximum range of strain rate is about 400 1/sec. Near future, high tensile tests will be performed on the low cycle fatigue damaged stainless steel under co-operation with Joint Research Centre of ISPRA.

11.2.2 Seismic Design Tests

Safety R&D for LMFBR plant seismic design has been performed in the following two parts :

(1) R & D for soil-plant structure interaction

(a) To study the influence of topographical irregularity (shape and soil property of ground) around the site on the plant structure behavior during earthquakes, the analysis has been performed for two years using the wave propagation theory with topographical irregularity effects. The difference was studied between the dynamic behaviour of the structure with and without irregularity of the shape and soil property of ground.

(b) For studying the interaction between the ground and the plant structure due to earthquake waves in horizontal direction, a 3-D FEM dynamic analysis code has been under development since 1974.

For validating the code, observation of actual earthquake response was performed using the scale model constructed on the actual ground. The model consists of a concrete base and a steel frame structure.

For studying the interaction behaviour in the vertical direction, observation of actual earthquake response in the vertical direction is being performed now using the same model mentioned above.

(c) A next step will be to confirm that the studies which has been made so far can be used for an actual plant.

(2) Seismic Design Tests for Major Plant Components

(a) To study the dynamic behavoiur of the containment vessel due to earthquake response of the plant building structure basemat, vibration tests using scaled-down models were performed in horizontal and vertical direction from 1978 to 1979. The dynamic behaviour of the fuel assemblies and the reactor vessel structure will be confirmed by vibration tests using models after 1980. These dynamic test results will be analysed.

(b) The mechanical snubber for the "MONJU" primary piping system must withstand a high gamma radiation. Irradiation tests of the lubricating materials began in 1977 for the mechanical snubber.

High temperature tests and high frequency vibration tests will small amplitude began in 1979 for mechanical snubber.
These tests will be continued till 1982.

11.2.3 Structural Integrity Tests for PHTS Piping

The phase-II study (FY 1973 - 1980) is being performed to understand the structural integrity of the PHTS hot leg piping. The study is focused mainly on the strength of the piping components against creep-induced failures such as creep-fatigue, creep buckling and progressive deformation. The results of the study are and will be fed to establish the interim design guides for the elevated temperature components of "MONJU".

Major new findings in the phase-II study are :

(a) The structural creep and elastic follow-up behaviour of scaled down model of "MONJU" primary hot-leg piping and its seismic characteristics.

(b) Fatigue strength reduction factors for the holding times of bending moment load on elbows under elevated temperatures.

(c) Preliminary simplified analysis method predicting creep-buckling behaviour of elbows.

(d) Creep-ratcheting behaviour of 304SS straight pipe and piping elbows under sustained axial tensile force and cycle radial thermal gradient.

(e) Thermal fatigue strength of circumferential welds including similar and dissimilar metal joints.

(f) Thermal fatigue strength of piping tees.

11.2.4 Alternate Reactor Shutdown System

The R&D on this item was completed.

11.3 Radiological Consequences

11.3.1 Radionuclide Transport and Removal under Normal Reactor Operation

(1) Fission Product Loop Experiments

The second phase Fission Product Loop experimental program is started with the aim to study radioactive contamination of LMFBR primary circuits due to fission products (FPs). The principal objective of the experiment is to study non-volatile FPs (e.g. Zr/Nb-95, Ba/La-140, Ce-141/144) behaviour, especially, the nature and kinetics of their deposition on to the pipe walls from flowing sodium.

The inpile sodium loop of Toshiba Training Reactor (TTR) is going to be replaced by a completely new one.

Preliminary feasibility study was done to fix the principal specifications for the new loop. Detailed designing of the new loop is under way.

(2) Tritium Trap Development

A chemical tritium trap using a nickel canned tritium was tested in a small radioactive sodium loop. The trap effectiveness was confirmed to the tritium removal from liquid sodium. The rate determining process was found to be the tritium diffusion through the nickel. The chemical tritium trap application to the actual LMFBR plant may be limited by its low trapping rate.

11.3.2 FP and Pu Transport under Accident Conditions

(1) HCDA Bubble Behaviour Analysis

Relative importance of various physical processes involved in the fuel particle release following an energetic CDA was studied by using FTAC code. The result shows that i) the dimension, shape and non-condensable gas contents of the sodium vapor bubbles play an important role on their condensation rate; ii) the smallest size fuel particles have predominent effect on the fuel aerosol release.

Improvement of FTAC code was continued. A model for the fuel vapor bubble transport in a sodium pool was developed. The model has been included in the version III of FTAC code.

Experimental work to verify the vapor bubble behaviour model of FTAC is started at O-arai Engineering Center as is mentioned in 11.1.1.

(2) Aerosol Behaviour in Containment

Aerosol behaviour analysis code ABC was modified to afford shape factor corrections. Comparison of ABC code predictions with the former JAERI experiments was made by changing the dynamic shape factor and collision shape factor values. The result shows that the code can predict both sodium oxide and uranium oxide aerosol behaviour fairly well compared with former ABC predictions by using an apparent density correction. Improvement of numerical algorithm in ABC code is under way in order to reduce computing time.

(3) Containment Atmosphere Analysis

A computer code was developed for the analysis of the containment cell atmosphere in postualted severe accidents. The code CEDAN calculates the pressure and temperature history of up to four cells which may be inter-connected among thermselves and to the outside. The code can handle such chemical reactions as sodium pool fire and sodium concrete reactions. The code verification is under way for sodium pool fires. Sodium concrete reaction experiment is also planned to start in the near future to verify the sodium-concrete reaction model in CEDAN.

Table 11.1 Experimental conditions

Experimental No.	T070101	T070401	T070402	T070403	T070404
Bundle	7	7	7	7	7
No. of molten pins	1	4	4	4	4
Net weight of UO_2(g)	24.5	98.2	98.1	97.9	97.9
Coolant					
Inlet temperature (^oC)	450	450	500	450	450
Velocity (m/sec)	6.6	6.5	4.8	4.6	4.6
Pressure at molten zone (MPa)	0.27	0.4	0.2	0.2	0.2
Axial force on the pellets (kg/cm^2)	220	104	62	62	62
Power					
Steady state (kW)	1.3	1.3	1.58	1.58	1.58
Transient (kW)	1.3+36.5t	1.58+24t	1.58+58.5t	1.58+117t	1.58+7.9t
(sec)	(0 t 1.33)	(0 t 2.03)	(0 t 0.8)	(0 t 0.4)	(0 t 1.8)

Table 11.2 Experimental resutls

Experimental No.	T070101	T070401	T070402	T070403	T070404
Failure time of each pin from power transient initiation (msec)	805	467 - 754	259 - 414	290 - 299	1308-1626
Max. measured pressure pulse Height (MPa) Full width at half maximum (msec)	-	 1.2 0.6	 1.3 0.3	 0.5 0.4	 0.5 0.5

Reference (1) Safety Reserach in Sodium Cooled Fast Breeder Reactor in Japan, PNC D084 80-01, 2 (1980).

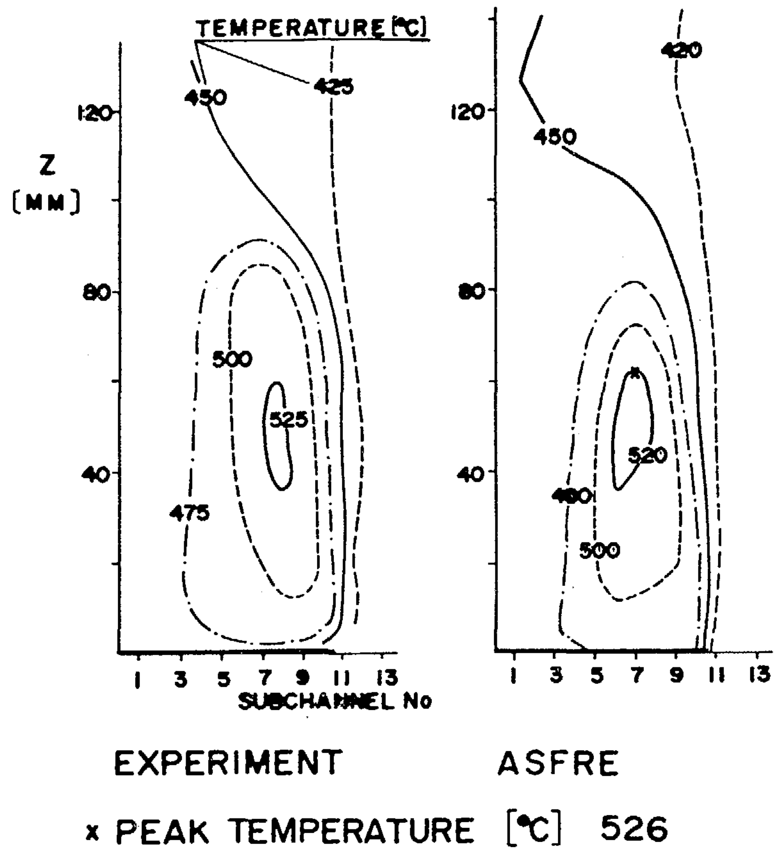


Fig. 11-1 Benchmark Calculation 49% Central Blockage Run No.1 Comparison of Temperature Distribution