A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

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1. Introduction

The fast breeder reactor development project in Japan has been in progress in the past twelve months and will be continued this fiscal year, from April 1982 through March 1983, at a similar scale of effort both in budget and personnel to those of the fiscal year of 1981. The 1982 year budget for R & D work and for construction of a prototype fast breeder reactor MONJU is approximately 20 and 27 billion yen respectively, excluding wages for the personnel of the Power Reactor and Nuclear Fuel Development Corporation, PNC. The number of the technical people currently engaged 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 six operational cycles at 75 MWt was completed in December 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 autnorities, and the licensing of the first step was completed in the end of 1981.

Preliminary design studies of a 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.

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 Japan. Construction of JOYO was started in 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 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 was continued until the end of 1981 in order to accumulate technical data of the present core. Concerning the performance characteristics 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 satisfactorily 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, and 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 shutdown. 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% Δ 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 preidcted value, was observed. However, such a phenomenon was 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 an elongation of the fuel pellet stack.

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 was continued until December, 1981.

2.3 Topics

1) 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 m 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. In parallel with the analysis, some fuel subassemblies with various wear mark parameters were irradiated in the 6th operating cycle, and we confirmed the conditions of parameters to avoid the appearance of the wear marks.

2) A series of natural circulation tests were performed at the end of December last year. These tests were carried out stepwise from low power (~1 MWt) to full power (75 MWt) with detailed plant dynamics analysis.
   Experimenal results showed that the cooling system has the ability to remove decay heat by only the natural circulation.

2.4 Future Program

The MK-II program was started in January, 1982. The core is converted to a core of 100 MWt power by replacing the MK-I fuels with MK-II fuels.

Achievement of the criticality with new core is expected in late fall 1982 and the power ascension testings to 100 MWt will be completed in the spring of 1983.

The reactor will be utilized as an irradiation facility for fuel and material development programs.

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, was decided for constructing MONJU. Based on survey works in various aspects such as geological marine and meteorology, suitability and stability of MONJU site was studied. The application for licensing was filed to the regulatory body on December 10, 1980. Then, the safety evaluation by the Science and Technology Agency of the Japanese government continued until December last year.

The safety evaluation of MONJU by the Nuclear Safety Commission of Japan is expected to start soon as a next step. On the other hand, negotiation for the contract of the plant is under progress between PNC and manufacturers.

Fast Breeder Reactor Engineering Co., Ltd. (FBEC) was established in April, 1980 to coordinate softwares of MONJU design and construction work of manufactures. A special department has been set up in the Japan Atomic Power Company (JAPC) to co-operate with the PNC for MONJU construction works. 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 3-1 and the plant layout is shown in Fig. 3-1.

MONJU is an about 280 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 by elevated pipings to keep sodium levels above the minimum safety level, should a sodium leak to the atmosphere occur.

Decay heat removal is normally accomplished by means of three auxiliary cooling systems (ACS), each of which has a branched air cooling line . Small pony motors on the main circulating pumps can provide continued coolant circulation with emergency power in the event of loss of main power supply. A key feature is the capability of decay heat removal by natural circulation without any emergency power.

The reactor employes a simple, top-supported, cylindrical reactor vessel with a hemispherical bottom head, about 7 meter in diameter and 17.8 meter in height. A guard vessel is surrounding the reactor vessel, and the reactor is housed in a reactor cavity 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 driver fuel subassembly has hexagonally bundled 169 fuel pins, and 61 fuel pins in each blanket fuel subassembly. The flow distribution in the core is controlled by fixed orifices at the bottom of the assemblies. The fuel assemblies are hydraulically held down to the support plate. Cladding material of fuel pins is SUS316. The length of the subassembly is 4,200 mm including shielding portions. The refueling interval is 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 covering the entire core region and a portion of the radial blanket region. The design provides two thermocouples for each core subassembly and for selected innermost radial blanket subassebmblies, and a flow meter probe for some core subassemblies and a few radial blanket subassemblies.

Reactor fuel handling is done 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 reactor core consists of 198 driver fuel subassemblies. The active core region is surrounded by axial and radial blankets. The radial blanket region consists of 172 blanket fuel subassemblies, and its equivalent thickness is 30.6 cm.

The upper and lower axial blanket is 30 cm and 35 cm long, respectively. 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 material.

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 enrichment to flatten a power distribution, and the number of fuel subassemblies in the inner and the outer zone is 108 and 90, respectively. For the equilibrium cycle, plutonium enrichment 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 is 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, and shielding. The upper portion of the subassembly is an handling head which mates with the fuel grapple of the refueling machine. The lower portion includes a nosepiece which provides supporting and flow orificing. Each subassembly is fixed to the support plates by hydraulic hold-down force.

The diameter and the theoretical density of the mixed oxide core fuel pellet is 5.4 mm and 85% TD, respectively. The maximum linear heating rate of the fuel pellet is 466 W/cm at 116% overpower condition including hot spot factor, and the maximum temperature of the fuel under that condition is 2600°C.

Fuel inventory of Pu and U in the core is 5.9×10^3 kg. Refueling interval is about 6 months. Five batch refueling schemes in the core and radial balnket are planned. Power generating 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 15.36×10^6 kg/hr. The flow distribution in the core is controlled by fixed orifices of fuel subassemblies. The flow fraction is 79.7%, 10.3%, 10.0% for the core region, radial blanket region and bypass flow, 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 by the concrete ledge surrounding the vessel, and its thermal expansion is free downward. It is about 17,800 mm high and fabricated 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° 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° apart in the upper plenum. The reactor vessel has also an outlet nozzle of the overflow system at its upper plenum.

The horizontal movement of the reactor vessel in the event of earthquake 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 and the fuel handling machine (FHM) mounted on it rotate for fuel handling.

The reactor internal structures consist of the upper internal structure and the lower internal structure. 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 nineteen CRDMs, nineteen CRDM guide pipes, thermocouples and flow-meters for measuring temperature and flow rate at the outlet of each fuel subassembly.

The lower internal structure consists 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 rows.

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.

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 overflow 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 shutdown.

The spent fuels are placed in the sodium filled pots 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 in the sodium filled pots are carried to the ex-vessel fuel storage tank. After that, non-failed fuels are cleaned, inspected, canned, and stored in the spent fuel storage pool in the reactor auxiliary building and then they are transferred to the reprocessing plants. On the other hand, failed or failure suspected fuels are stored in the sodium filled can, and transferred to the fuel monitoring facilities.

3.6 Heat Transport 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 (IHXs) and steam generators (SGs) to a turbine generator. The main cooling system consists of three loops.

Each of six main pumps 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 in the case of one loop shutdown.

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

The fluctuations of steam conditions are controlled by the feed water control valve.

The relative elevation of the reactor core, IHXs, air coolers, and steam generators are arranged so as to assure natural circulation of sodium in the primary and intermediate loops as a back up method to remove the decay heat from the reactor core, should reactor scram occur.

3.6.2 Auxiliary Core Cooling System

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

The auxiliary core cooling system shares the primary pipings with the main cooling system, and decay heat is transferred to the secondary sodium through IXHs. The heated secondary sodium is pumped to air coolers which are located at the bypass line to the steam generator and heat is dumped 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 the primary (radioactive) argon gas system of closed cycle and the secondary argon gas system of open cycle. The function of the primary argon gas system is to control argon gas pressusre to cover sodium surface, to seal pump shafts, and to transport sodium by gas pressure in such case as sodium drain and charging.

Secondary argon gas system mintains cover gas pressure at 3,000 mmAq in 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 rare gas removal efficiency 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 to the heat transport system.

1) Almost of the primary main sodium pipings are located above the minimum safe level, and guard vessels are provided around the pipings wherever pipings are below the safe level.

2) The elevation of the pipings 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 negative pressure at any high points.

4) The relative elevations of the reactor core, IHXs, air coolers and steam generators are arranged to assure natural circulation in the primary and intermediate loops.

5) The elevation of the hydrostatic bearing of the main pump was determined for the bearing to be submerged in sodium even if the sodium level in reactor vessel is below the emergency level.

6) The intermediate sodium pressure within IHXs is maintained above that of 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, and they have helically coiled heat transfer tubes.

As shown in Fig. 3-2, an evaporator and a superheater are 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 of the reaction. This relief system is designed on the assumption of a double ended fracture in four tubes. A rupture disk is installed in the cover gas region of each component and when the disk is ruptured by inner high pressure, the reaction products flow through the rupture disk to a reaction product tank.

This exhaust system is maintained with nitrogen gas atmosphere, and is separated from the steam generator system by a rupture disk. A small water or steam leakage into sodium in the steam generator is detected by an ion pump current indication device which quickly detects hydrogen generated from sodium-water-interaction through a thin nickel membrane. Hydrogen in cover gas is 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 shuts down 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, and each of them has the own ability to shut down the reactor independently.

The design effort for the in-vessel instrumentation is devoted to avoid the excessive core fuel temperature by detecting the fuel anomaly 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. Ten output signals are 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 between 100% and 40% 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 setback system, which decreases reactor power at a rate of 5%/min on anomalous phenomena of unknown origin until the setback conditions are cleared. If setback conditions are not cleared before the ractor power reaches 40% 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.

a. power demand master
b. reactor power control system
c. primary sodium flow control system
d. secondary sodium flow control system
e. feedwater flow control system
f. steam pressure control system
g. 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 controlled as a function of power level. Primary and secondaly sodium flow control systems are composed of similar scheme. The main control signal in these systems is sodium flow and the supplementary signal is pump speed. The evaporator outlet steam tempeature is controlled by feedwater flow which is rgulated by the feedwater flow control valve. The feedwater control system regulates the feedwater pump speed to keep the differential pressue of feedwater control valve at a fixed value. The superheated steam pessure is 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 in principle.

3.9 Radioactive Waste Processing System

The design objective of the radioactive waste processing system is to minimize the levels of radioactive materials in the plant effluents and/or to make packages of the radioactive wastes to store them 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 releasing to environment.

Since gaseous fission products contained in the primary cover gas are removed by rare gas removal system and processed gas is 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 gas is 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.

a. irradiated fuel assembly washing effluent
b. sodium component washing effluent
c. reactor building drain
d. radioactive waste processing system drain
e. laundry drain

Evaporator concentrates and spent resins are treated by bitumen solidification unit to produce solidified waste packages. Spent air cleaning filters are packaged at the generated place. Contaminated cloths and papers are transferred and stored in the storage pond.

3.10 Reactor Containment and Building

3.10.1 Reactor Containment

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

The containment vessel fabricated 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 equipemnt 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.

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

3.10.2 Building Arrangement

A sectional and horizontal cross sections 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 (Pu fissile %)		Inner core / Outer core
	Initial core	15 / 20
	Equilibrium core	16 / 21
Fuel Inventory	Core (U+Pu metal)	5.9 Ton
	Blanket (U metal)	17.5 Ton
Average Burn up		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 / 300 mm
Breeding Ratio		1.2
Reactor in/out Sodium Temperature		397/529°C
Secondary Sodium Temperature (IHX outlet / IHX inlet)		505/325°C
Reactor Vessel (hight/diameter)		17.8 / 7.1 m
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

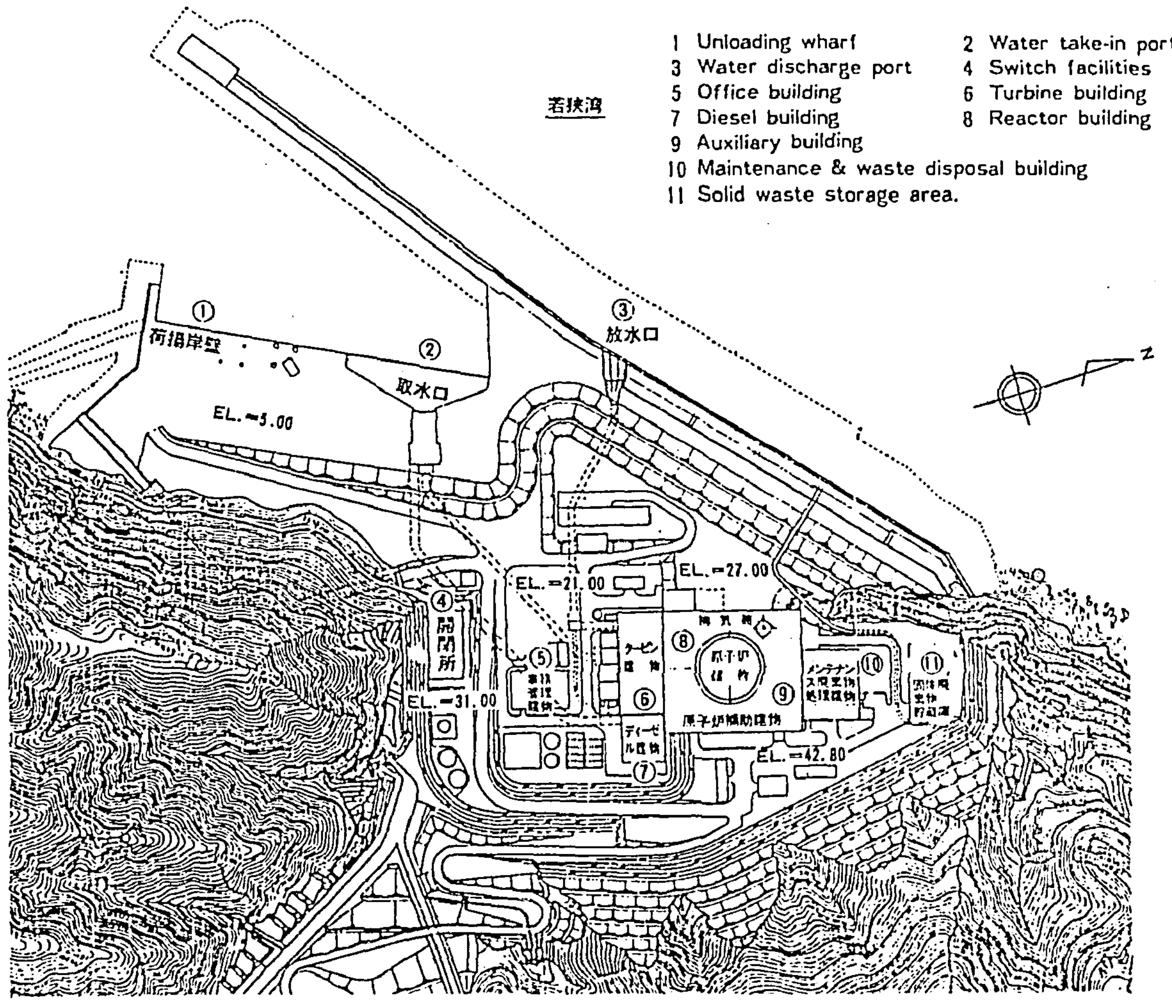


Fig. 3-1 Plant Layout of MONJU

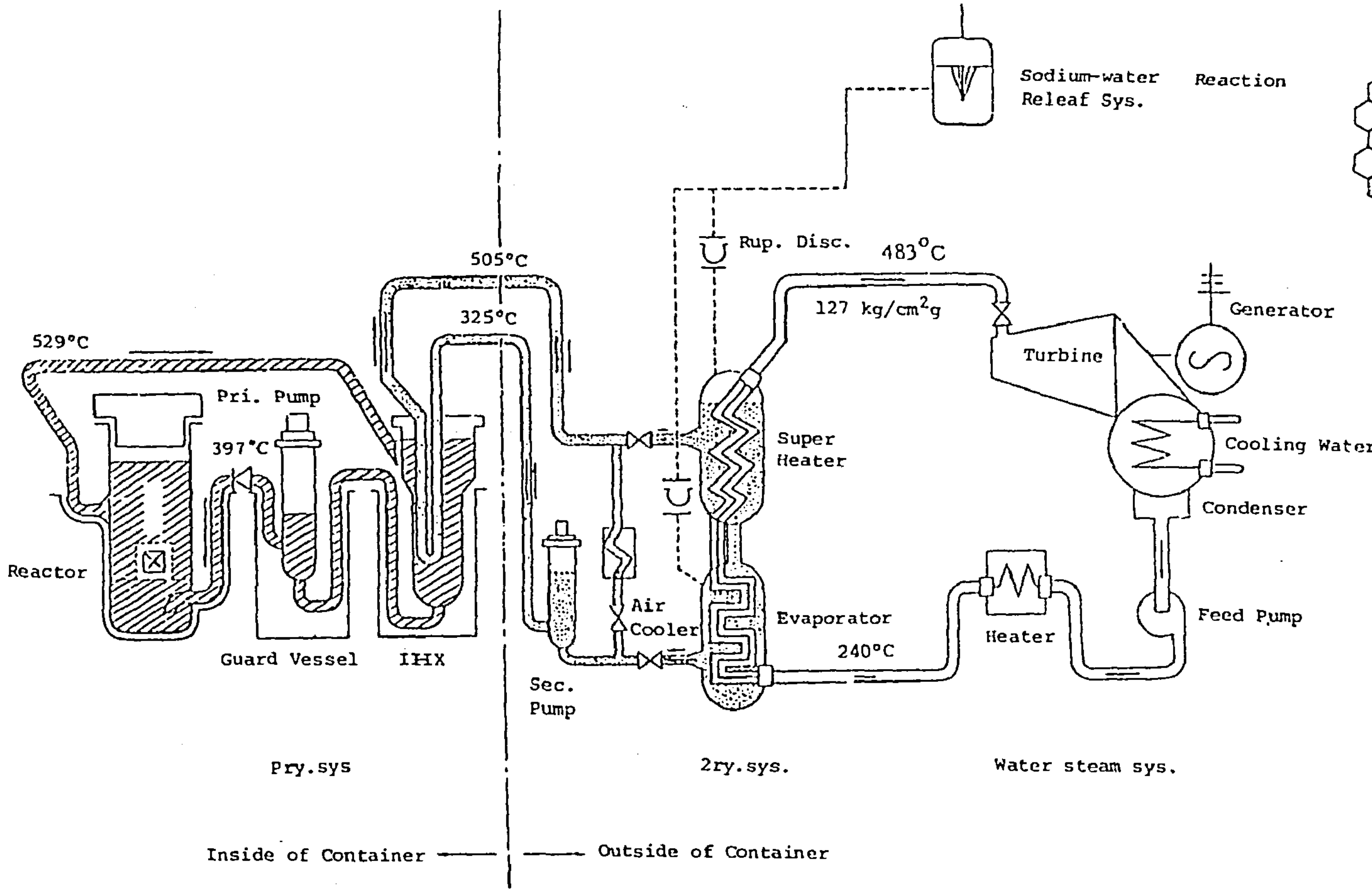
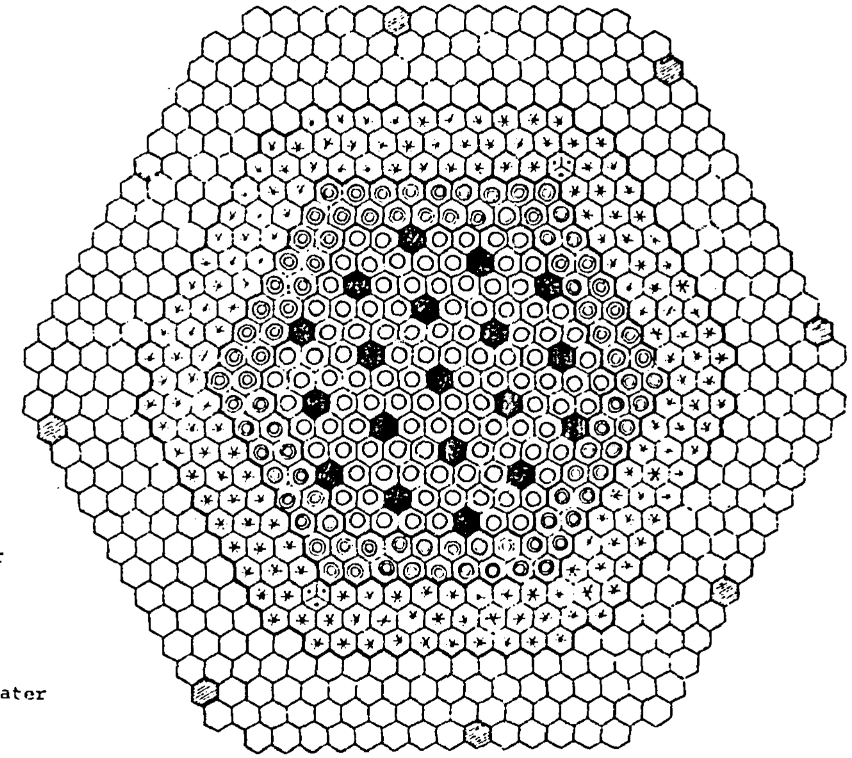


Fig. 3-2 Heat Transport System of MONJU



core elements		marks	quantity
core fuel - S/A	zone 1		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

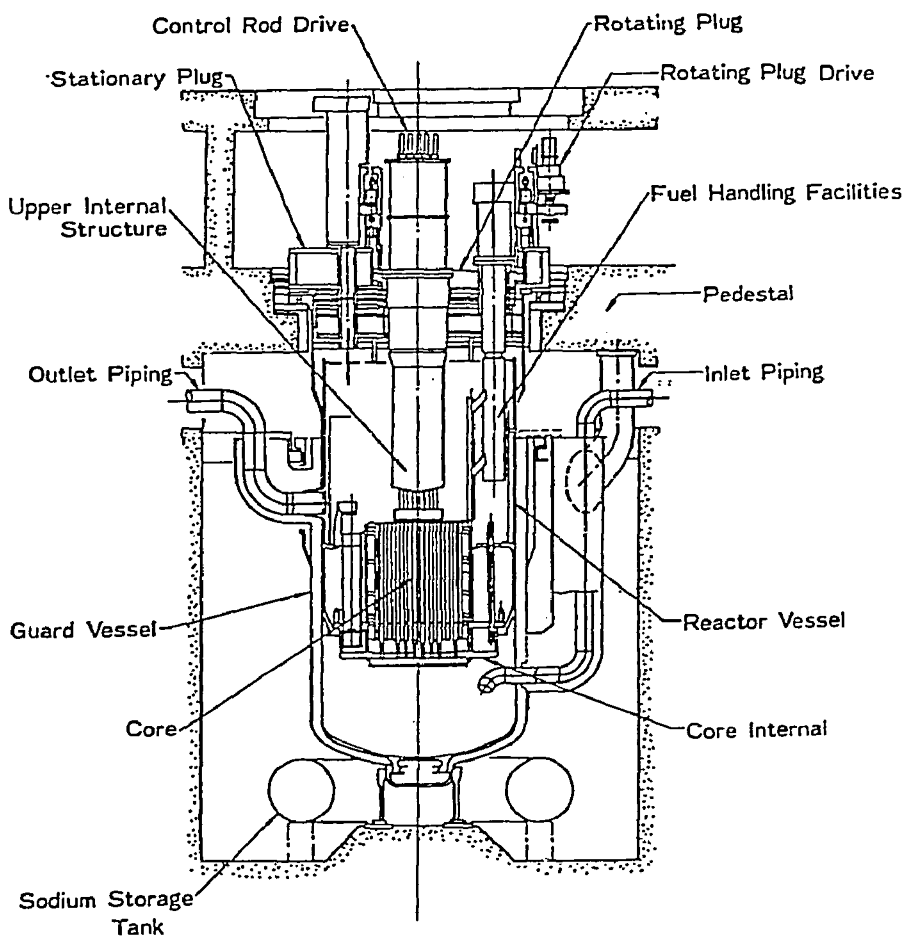


Fig. 3-4 MONJU Reactor System

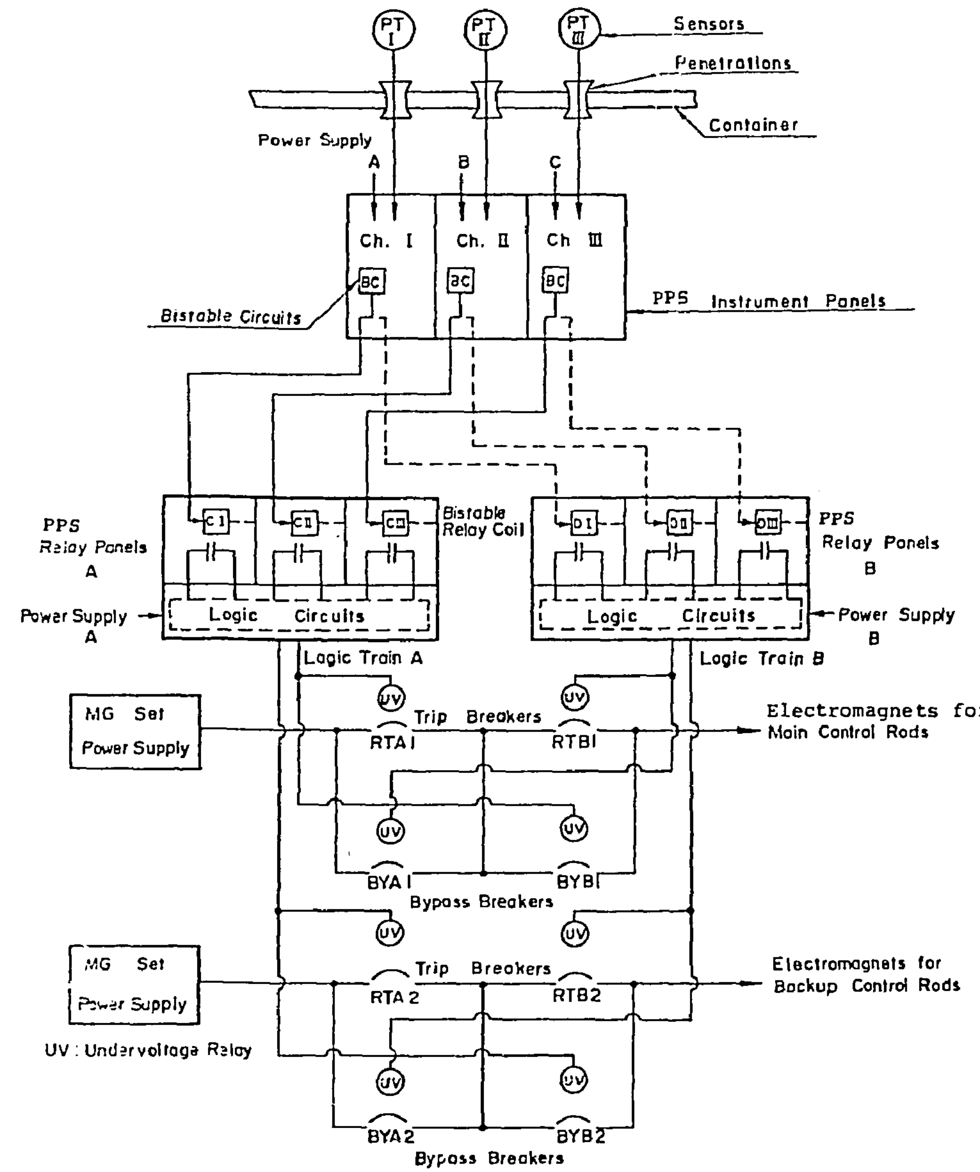


Fig. 3-5 Plant Protection System Instrumentation

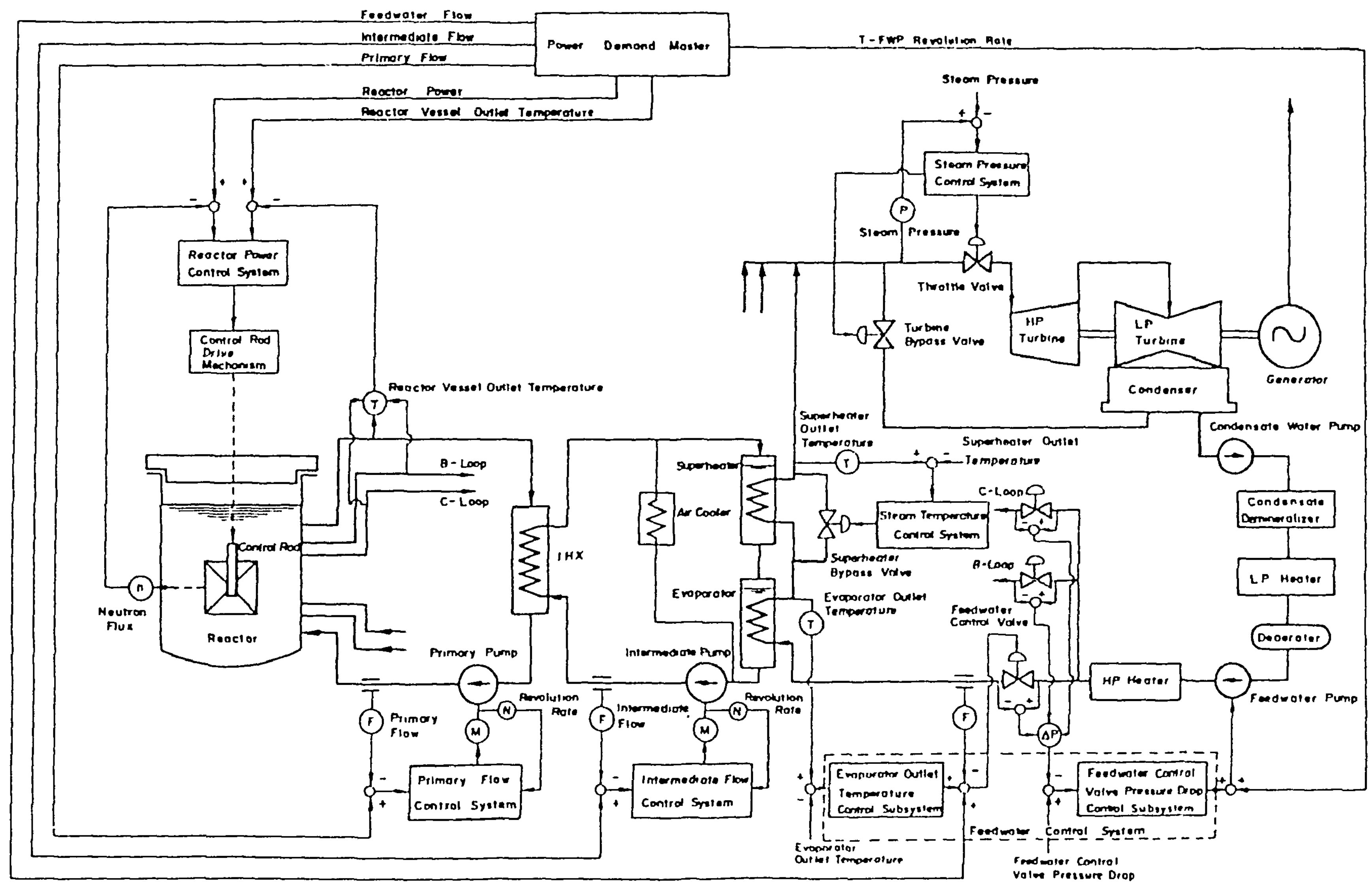
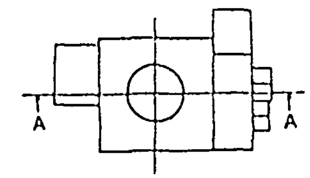
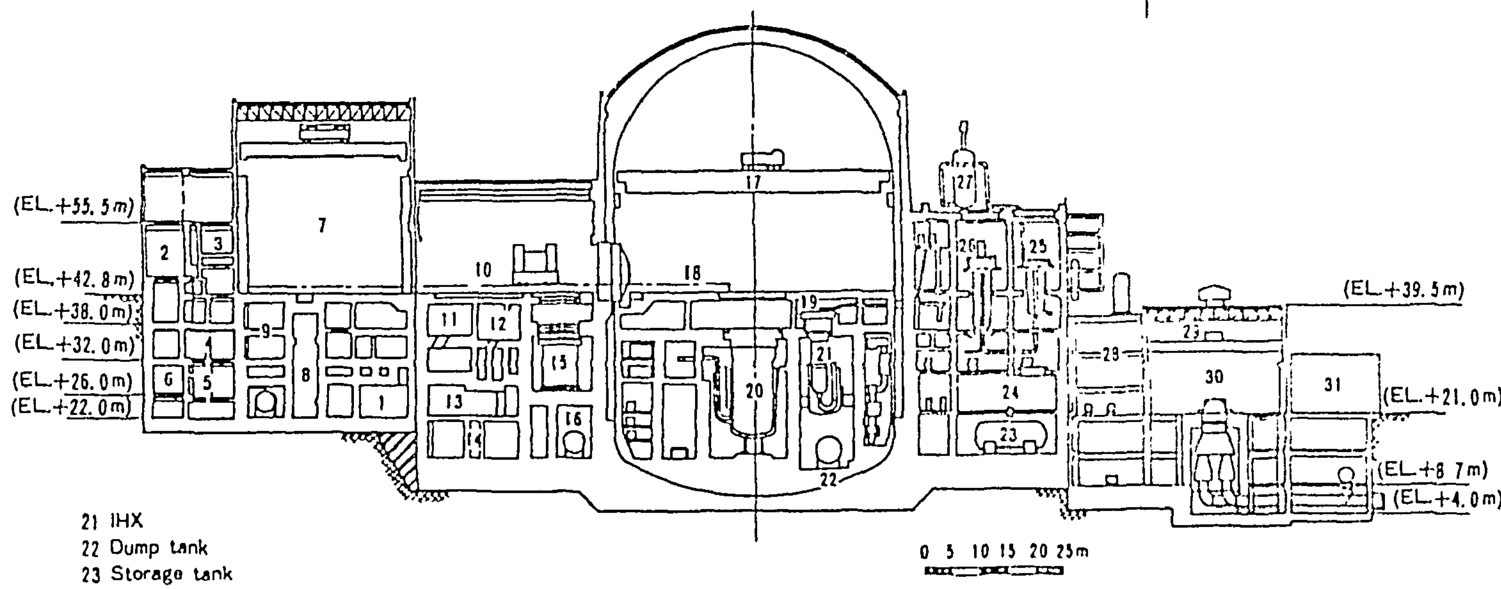


Fig. 3-6 Plant Control System

1 Ventilator and conditioner room
2 Maintenance room
3 Switchgear room
4 Monitor tank room
5 Miscellaneous waste collector tank room
6 Concentrated miscellaneous waste tank room
7 Maintenance area
8 FHM cleaning room
9 Cleaning ventilation system room
10 Ex-vessel transfer machine room
11 Fuel canning room
12 Fuel cleaning room
13 Fuel inspection equipment room
14 Gaseous radwaste processing system room
15 EVST
16 Sodium overflow tank
17 Polar crane
18 Operation floor
19 IHX head area
20 Reactor vessel

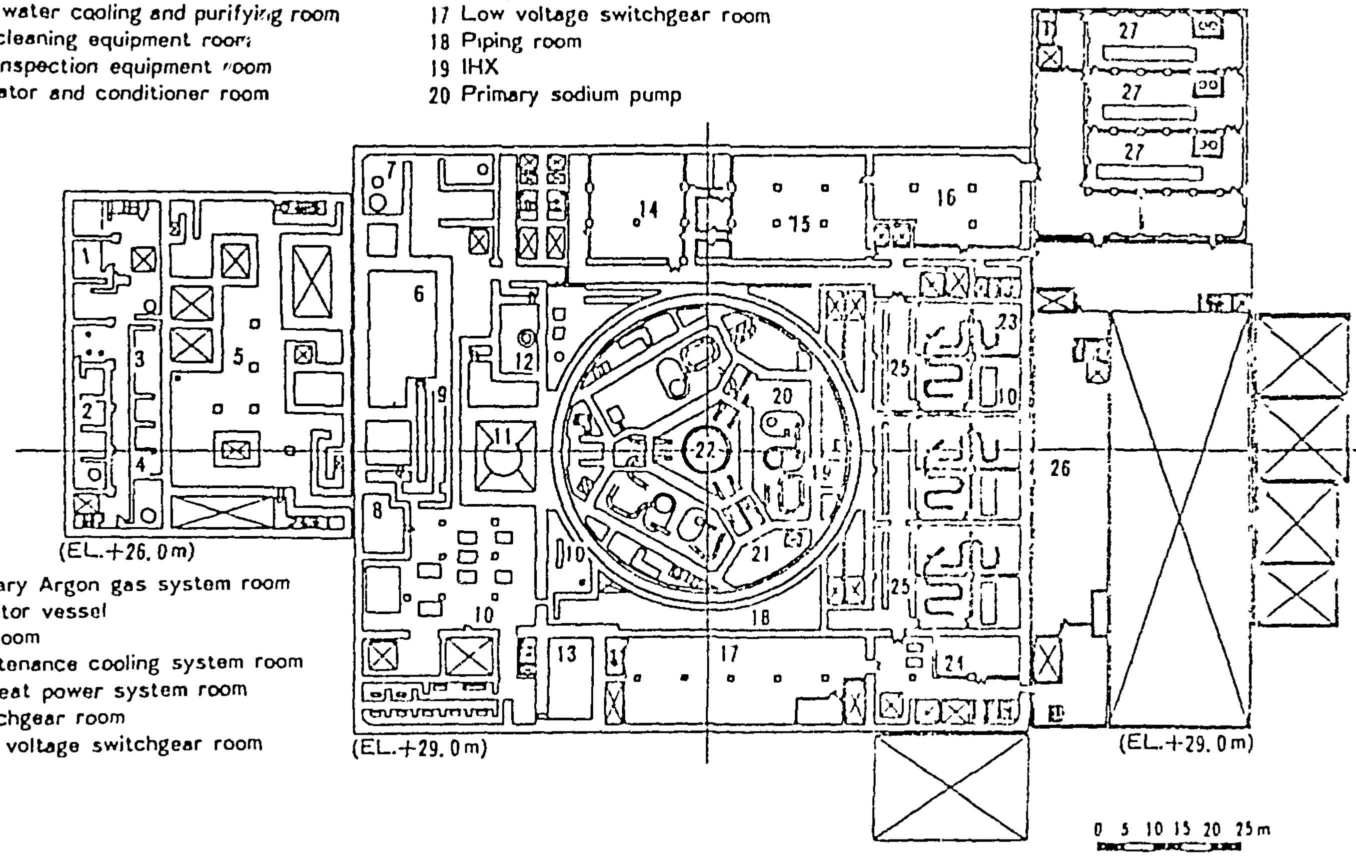




21 IHX
22 Dump tank
23 Storage tank
24 Secondary sodium purification system room
25 Super heater
26 Evaporator
27 Reaction product storage tank
28 Switchgear room
29 Crane
30 Turbine generator
31 Transformer area

Fig. 3-7 Sectional View of Main Buildings

1 Spent resin tank room
2 Concentrated miscellaneous waste tank room
3 Monitor tank room
4 Miscellaneons waste collector tank room
5 Components cleaning area
6 Fuel storage pool
7 Pool water cooling and purifying room
8 Fuel cleaning equipment room
9 Fuel inspection equipment room
10 Ventilator and conditioner room
11 EVST room
12 Cold trap room
13 FFDL room
14 Fuel handling facilities operation room
15 Main control room
16 Relay room
17 Low voltage switchgear room
18 Piping room
19 IHX
20 Primary sodium pump



21 Primary Argon gas system room
22 Reactor vessel
23 SG room
24 Maintenance cooling system room
25 Preheat power system room
26 Switchgear room
27 High voltage switchgear room

Fig. 3-8 Horizontal Cross Section of Main Buildings

4. Demonstration FBR Plant

One demonstration plant will be constructed on the way from JOYO and MONJU to commercial fast breeder reactor plants.

Design studies of the demonstration plant which is 1,000 MWe loop type plant aiming its commencing of construction in 1990, have been and are being conducted in these years.

We are now in just transition stage from a studying period to a focussing period and two matters will soon be required;

(1) Preparation of an unified specification for the fundamental design, and (2) Starting up of the research and developemnt activities for the demonstration plant.

In the conceptual design-(II) performed in FY 1980, primry pump position and reactor vessel inlet nozzle position was changed to hot-leg and to upper inlet type, respectively. And, chute type fuel-charge-and-discharge system was adopted in stead of cask-car type. These are being kept in the conceptual design-(III) which is in progress at present.

In consequence, the reactor vessel diameter becomes larger, and the reactor cover gas pressure is lowered.

The core consists of 420 fuel assemblies which contain 217 pins each. The number of cooling loops are three.

The building is approximately one-hundred and ten meters square with a dome of 64 meters in diameter.

The above descriptions are based on the design study sponcered by PNC.

Also the utility group is conducting design study of a four-loop type demonstratin plant separately.

Both streams are just flowing into a specification unification stage. The unification effort will be a key for the early realization of the demonstration plant in Japan. A development schedule of FBR in Japan is shown in Table 4-1.

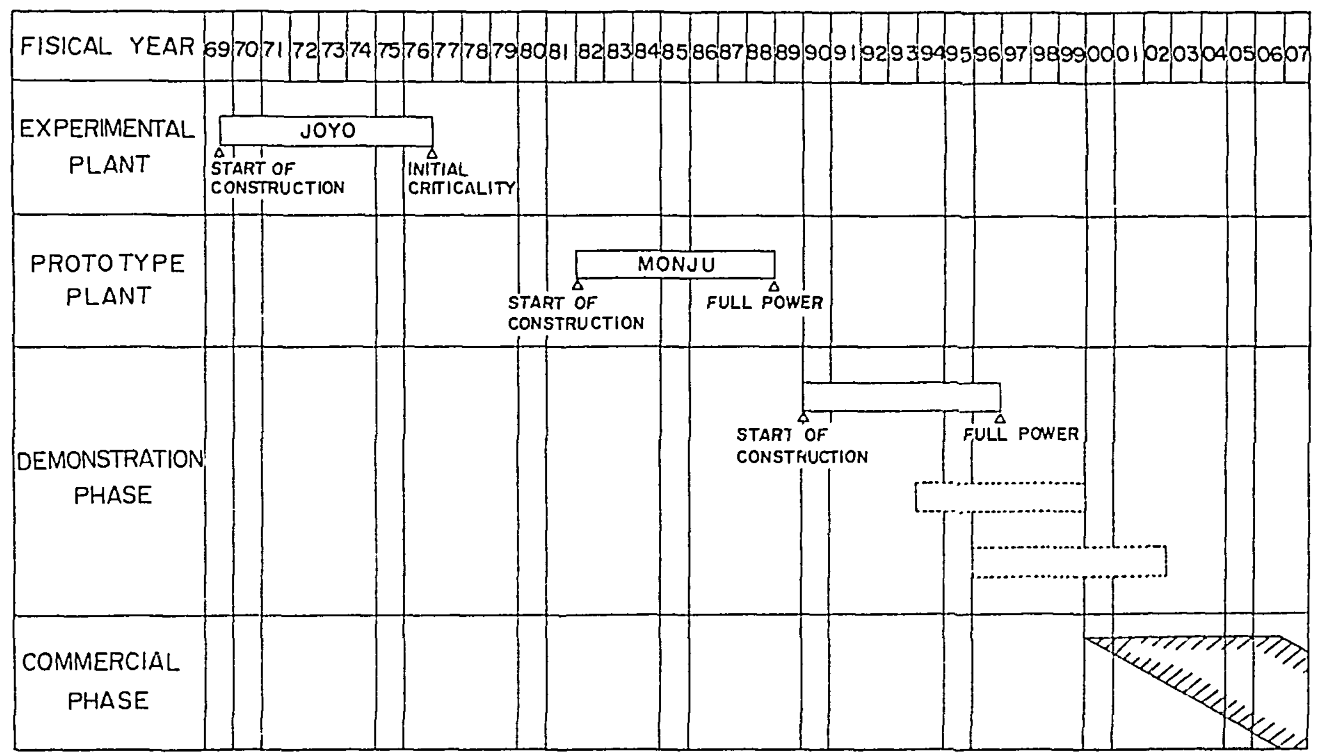


Fig. 4-1 Development Schedule of FBR in Japan

5. Physics

5.1 JUPITER Phase II Program

The DOE/PNC Joint Physics Large Heterogeneous Core Critical Experiments Program, called JUPITER Phase II Program, would involve 1 year and 8 months of measurements. The experiments start as ZPPR-13 program about May, 1982, using the ZPPR facility of Argonne National Laboratory, located near Idaho Falls, Idaho, U.S.

The present status of research and development on heterogeneous core in Japan is in the preliminary stage, with an emphasis on basic research to provide data for the assessment of overall viability of the concept of the heterogeneous core relative to that of conventional homogeneous core, including both the radially and axially heterogeneous core configurations. The primary Japanese objectives of the JUPITER-II Program are to obtain a better understanding of the basic characteristic features of heterogeneous cores and to improve the obtain calculation and design methods for them.

The current plan of JUPITER-II Program is to obtain important physics and engineering data for heterogeneous cores in the 700MWe size range. The emphasis will be on data of special interest to large heterogeneous cores such as core zone decoupling, sodium-void reactivity worths, control rod interactions, Doppler coefficients and reaction rate distributions. A series of measurements is planned to focus on these issues of particular importance.

The experiments will be analyzed using the JENDL-2 library. The experimental data will be used for the evaluation and improvement of the data and methods for the large LMFBR nuclear core design.

5.2 FCA Experiments

Mock-up experiments will be done to support the JOYO MK-II Program, using the fast critical assembly FCA of JAERI. In order to verify the reliability of the supervisory core performance code and to support the irradiation and refueling plans of MK-II cores, reaction rate distributions and other reactor parameters will be measured in detail in the mock-up systems with reflectors.

5.3 Data and Methods

The neutron nuclear data for higher plutonium nuclides will be evaluated in order to support the compilation work on the JENDL-3 (Japanese Evaluated Nuclear Data Library), which is one of the activities of the JNDC (Japanese Nuclear Data Committee).

The effective resonance cross sections for U-238 is planned to be produced by taking account of both the chemical binding and multi-level effects.

A set of benchmark problems, based on 2D diffusion calculation, will be compiled to assess the applicability of the fast reactor group constants.

To treat the interference effect of neutron streaming between different cells on diffusion theory, we extended the Bonoist formula to a super cell model and derived the unified diffusion coefficient. It will be applied to analyses of critical experiments, especially sodium void and neutron streaming experiments.

An effective cell homogenization procedure has been developed, which preserves group-wise reaction rates. The method will be applied to calculations of control rod worths with various in-cell structures.

A 3D transport code will be developed for detailed analyses of critical assemblies and for tests of 2D calculational models.

A new method of subcriticality monitoring is planned to be developed, which uses the statistical data processings and adjustments of the count rate changes from LLFM (Low Level Flux Monitor), installed outside the system. The method will be developed and tested using JOYO experiments and critical experimentrs. It will be applied to subcriticality monitoring of JOYO and to safety evaluation of the External Vessel Storage Tank.

5.4 Research for Design

Design calculation methods will be reviewed and the accuracies of nuclear core design parameters will be assessed for the large homogeneous and heterogeneous cores, based on the current data and methods.

The safety characteristics will be assessed by using the SAS3D and VENUS codes, and will be compared between the large homogeneous and heterogeneous cores.

5.5 Research on Shielding

For the shielding design and analysis of FBR, researches have been carried out in three items;

a. Developements of Numerical Analysis Methods and Programs,
b. Shielding Experiments, and
c. Shielding Analyses of the Fast Reactor Plants.

For the shielding calculations of fast reactors, two numerical methods, discrete ordinates and Monte Carlo, are the most important, and the applicabilities of these methods have been studied for years.

The two dimensional discrete ordinate transport codes have been widely used in the shielding analyses of fast reactors, and the method of these codes was proved to be very effective. Based upon these experiences, the two dimensional albedo Sn code has been developed; the code is able to adopt the differential albedo boundary conditions. This albedo Sn code is expected to save the computing time in "boot-strap" calculations.

The appliacbility of this code will be studied by the analysis of the shielding experiment this year.

The evaluation of the neutron streaming through the complex geometry requires the Monte Carlo method. We have developed the albedo Monte Carlo code system, and adopted the code to the streaming calculations. Combinations with the forward and/or adjoint two-dimensional discrete ordinate calcualtions, and the adjoint Monte Carlo calculations were used for the estimations of the neutron streaming.

For these calculations, the group constants are being compiled from the Japanese Evaluated Nuclear Data Library (JENDL).

To obtain the experimental data for evaluation of applicabilities of these codes, the streaming experiments have been carried out in Fast Neutron Source Reactor YAYOI in Tokyo University. Streaming experiment for the control rod drive mechanism of the fast reactor is being planned in YAYOI.

Computer code systems for the shielding calculations should be examined finally by applying them to the reactor plants. Reuslts of analyses are able to be referred directly in the design calculations of the new FBR plant. Two and three dimensional discrete ordinate codes, and Monte Carlo code have been applied to the abalyses of JOYO shielding measurements. In 1981, the streaming data along the ducts in JOYO were analyzed, and we are now planning the shielding analyses of FFTF.

6. Research and Development of Reactor Components

6.1 Reactor Vessel and Internal Structure

6.1.1 Hydraulic Tests of Flow Distribution

To investigate the flow dynamics of coolant in the reactor vessel of prototype FBR MONJU, simulation tests have been performed at Hydrodynamic Test Facility, O-arai Engineering Center, PNC with a half-size flow model of the vessel using water as model fluid. Among these tests, experiments on pressure distribution in plenums, pressure loss between plenums, and flow rate distribution through the core and blanket regions have been conducted. It became clear that the flow in upper plenum consists of both a main flow and two recirculation flow and that the flow mixing in lower plenum is best when the issuing angle of inlet nozzle is 15°.

6.1.2 Thermal Transient Tests

Structural design of the reactor components and analysis of plant dynamics require an accurate understanding of the thermal and hydraulic behavior of the coolant in the reactor upper plenum. This led to the sodium transient test at O-arai Engineering Center in 1/6 and 1/10-scale model of MONJU upper plenum, which were conducted focussing on the improvement of a previously developed two dimensional thermo-hydraulic analysis code SKORT, and on the evaluation of the thermo-hydraulic behavior of the prototype reactor. In order to make further clear the hydraulic problems around the upper plenum and to get the design basis of the reactor structure, flow experiments were carried out in manufacturer's laboratories with 1/1 (1/3 sector) and 1/6-scale model using water.

6.1.3 Entrained Gas Test

Accumulation of gas dissolved in sodium and gas bubbles released from failed fuels may raise a variety of problems in fast reator. Hydraulic tests for investigating gas bubbles behavior in plenums of the reactor vessel and around the gas-vent-holes have been performed.

6.1.4 Structural Test of Reactor Vessel

MONJU reactor vessel has an inner weir structure which is designed to alleviate the temperature gradient growing on the reactor vessel wall just above the liquid surface level. To demonstrate the effetiveness of the structure and to evaluate thermal stresses in the vessel, approximately 1/4 scale model was installed accompanied with a sodium loop at O-arai Engineering Center. The testing began in April, 1982.

6.2 Shield Plug

A temperature distribution test is being carried out on a simulated reactor upper shield plug which has a scale of approximately 1/3 in diameter and 1/1 in height. It is operated under elevated temperature using sodium. In connection with this test, thermal emmisivities of liquid sodium surfaces, natural convection and sodium deposition within the annular gap have been explored using scale models.

6.3 Primary Pump

Prototype primary pump of MONJU was constructed to confirm the performance of the actual pump, and tests have been performed both in water and sodium.

The pump was operated up to full flow rate in water and showed good hydraulic characteristics.

In sodium, one-fifth flow rate impeller, of which delivery pressure is equivalent to the actual pump is applied to the prototype pump in lieu of regular impeller because of limitation of the loop capacity.

At the early stage of the test in sodium, stick of the bearing occurred due to the deformations of inner casing which had non-uniform temperature distribution in operations.

The cause of this incident was presumed to the natural convection of argon gas in the gap between inner and outer casings. So thermal shielding plates were attached on the inner casing to prevent the gas convection.

After the modification, the pump has been running in good condition for over 20,000 hrs by now. Coast down test was conducted to fill the requirement in the plant dynamics analysis.

Start up test after long hot stand-by, low speed operation test with low sodium level im pump casing and none seal gas operation test have also been conducted to demonstrate the various pump operation modes of MONJU.

6.4 Intermediate Heat Exchanger

A full scale and 1/6 sector model of the MONJU intermediate heat exchanger was fabricated to establish favourable flow distribution, and the study was made using water.

Also 1/2 scale model of the IHX was fabricated both to make sure the established flow distribution in primary side and to optimize the flow control device in the secondary side. The test is being performed also using water.

In parallel with these tests, a thermal transient test on the upper tube sheet and shell structure has been conducted in sodium on a simplified model.

Further sodium flow test under transient condition is planned to investigate the stratification characteristics in the rising flow region after inlet nozzle which might affect thermal shock rate to the upper tube sheet section.

On the other hand, in-sodium life test was carried out on the trially fabricated bellows to be used at the top of the down-commer pipe.

Furthermore, on the tube-to-tube sheet welding, trial fabrications have been made to select the most suitable method considering the reliability and economy.

6.5 Sodium Valves

Several large sodium valves are fabricated to develop the heat transfer systems of MONJU.

The 12 and 16 in. valves were designed for flow control service, 22 in. valve for isolation service and 16 in. valve for anti-reversal flow service.

Endurance tests and thermal shock tests were applied on these valves and provided good results as expected.

To demonstrate a applicability of a large sodium valve to MONJU, two prototype valves were fabricated and tested in sodium.

It was found that the valves showed good performance under the actual service condition.

There are a 22 inch butterfly type valve for use of SG insulation and a 24 in. swing check valve with dash pot for use of anti-reversal flow at outlet of primary pump.

6.6 Control Rod Drive Mechanisms

Three kinds of control rod drive mechnisms have been tested with full scale mock-up under simulated sodium conditions. These have fine control, coarse control and back-up functions respectively. These three test articles are designed and fabricated by three separate manufacturers. Bellows for shaft seals have been tested to establish the design basis. As dynamic behavior affects on the life of bellows, movement of bellows in case of scram has been studied in detail in both experimental and analytical method. Besides, fatigue life data have been stored through in-sodium and in-argon gas test on bellows made of 316L type stainless steel and of Inconel.

6.7 Refueling and Fuel Storage System

Refueling system of MONJU consists of in-vessel fuel handling machine (FHM), in-vessel transfer machine and ex-vessel transfer machine (EVTM). Testing of prototype FHM in sodium and characteristic test of the shaft seal were completed. Testing of prototype EVTM in sodium has been performed since 1981.

The ex-vessel fuel storage tank (EVST) is designed to hold the fuel assemblies by a rotating rack in sodium. Testing of the bearing and the shaft seal in air was carried out and testing in sodium will begin in 1982.

6.8 In-Service Inspection Equipment

An efforts is being made to develop In-Service Inspection Equipment for reactor vessel and its inlet nozzles. Basic examination technique is visual examination.

The equipment is required to provide functions to inspect remotely the object surface with ITV camera. The camera must be moved through the narrow annulus between reactor vessel and guard vessel.

Cooling system for ITV camera and several eletrical parts to operate at high temperature are being developed. The remotely controlled ISI equipment for inlet nozzle is under testing with a mock-up at about 200°C.

Another effort is being directed to develope the ultrasonic method at high temperature for weldings of austenitic stainless steel as one of volumetric examination technique.

7. Steam Generator System

7.1 50 MW SG Test Facility

After the 3,400 hour operation with steaming condition, the testing of No. 1 50 MWSG was completed in April, 1975. It was disassembled for inspection and the feasibility of the MONJU SG design concept was confirmed.

Then, No. 2 50MW SG was constructed and the performance test began in January, 1976. Test items and schedule is shown in Fig. 7-1. The accumulated operating time of No. 2 50MW SG is 11,500 hours with steaming condition for the evaporator and 3,800 hours for the superheater. Total operating time for secondary sodium loop with sodium is 26,200 hours (as of end of 1981).

Research and development on the operating characteristics, materials, operation and control techniques, water leak detection and the maintenance and repair of the steam generator are under way.

The spacial characteristics of the evaporator were evaluated taking into account of the plugged tube effect. Evaluation of the effect of the water side failure is under way. No performance change has been observed after 11,000 hours operation.

Water and hydrogen are injected into the evaporator or the sodium inlet piping and hydrogen behavior in the loop is being studied.

Acoustic leak detection system is also tested at the 50MW SG Test Facility.

Demonstration of maintenance and repair techniques for SG was carried out in 1981 using the evaporator. Works conducted are shown in Fig. 7-2 and these techniques were confirmed to be applicable to the real plant.

The verification test of MONJU auxiliary cooling system will be carried out using the 50MW SG Test Facility in 1982 and 1983. For this purpose, several modifications of the test facility such as the installation of an air cooler in the secondary sodium loop are under way.

Detailed tests on water flow instability, DNB and thermal transient on tubesheet were carried out using LMW SG Test Facility.

7.2 Sodium Water Reaction Study

7.2.1 Leak Hole Enlargement and Leak Propagation Study

Eight self-enlargement tests on micro crack defect of 2 1/4Cr-1Mo and stainless steel were conducted under the MONJU's operation conditions using SWAT-4.

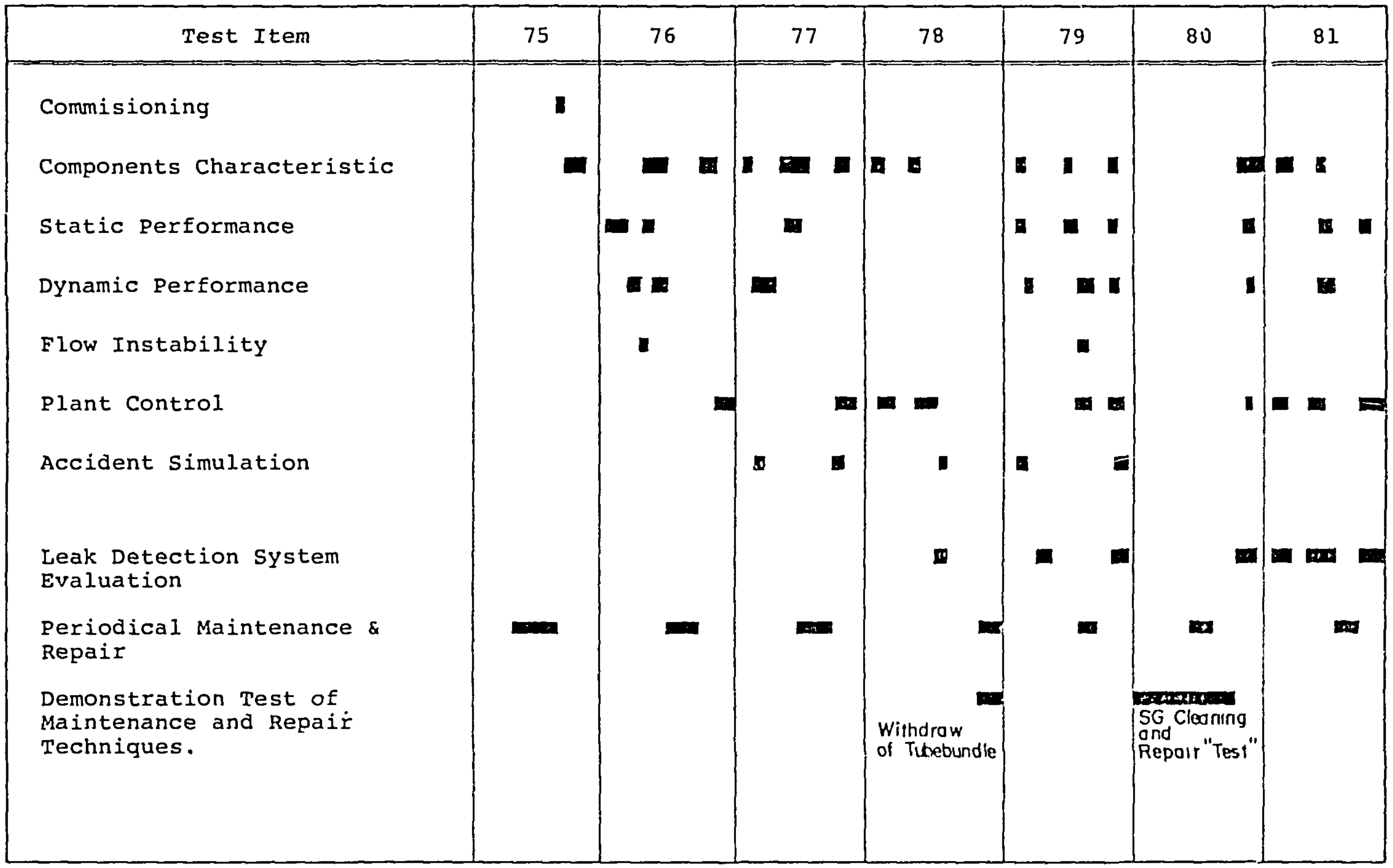


Fig. 7-1 No. 2 50MW Steam Generator Test Program

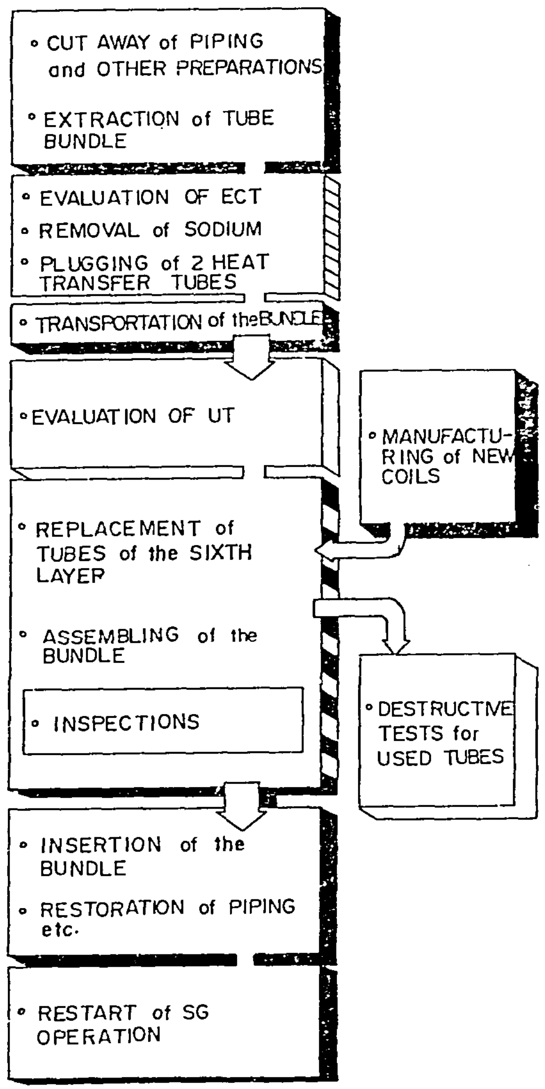


Fig. 7-2 Maintenance and Repair Test Items on 50MW SG Test Facility

Two of them grew rapidly from non-detectable leaks to damagable leak for adjacent tubes.

Three intermediate leak tests were conducted using SWAT-01 in order t look for the severest condition for the selection of MONJU's DBL, and th present DBL was proved to be valid.

Two failure propagation tests were carried out using SWAT-3 under mor prototypical conditions. One was full term simulation test, where all tubes wer filled with saturated water and the leak propagated up to fifth step from th initiation of the water injection to the termination of the water dump. Anothe was an overheating proof test in the large leak region, where some data on th heat transfer coefficient by chemical reaction was acquired. Even considerin these test results, the DBL remains unchanged.

7.2.2 Leak Detector Development

Three PNC type in-sodium hydrogen meters are being operated to study th long-term performance, and maximum time is over 20,000 hours without an troubles.

An in-cover gas hydrogen meter (Ni membrane type) was tested usin SWAT-2, and the hydrogen content range of 1 -10,000 Vppm was measured. Th meter was also used to measure hydrogen detection rate in cover gas, solvin rate and rising velocity of hydrogen bubble in sodium.

7.2.3 Computer Code Analysis

The SWACS code, which predicts the large leak phenomena, has been use for the validation study with SWAT-3 tests data of RUN-3 through RUN-7 Analysis showed that the predicted pressures agreed reasonably well with test data using appropriate input parameters.

The LEAP code, which predicts the leak-propagation phenomena, has bee developed, and miscellaneous code modification have been made by means o comparison with SWAT-1 and SWAT-3 tests data. The code tends to over estimate the extent of tube failures.

The SWAC-10 code has been used for the validation study with the wate injection tests data of the 50MW SG facility. Analysis showed that the predicte detection characteristics agreed with the test data.

8. Sodium Technology

8.1 Material Tests in Sodium

Creep-rupture tests under internal pressure are carried out for modified SUS316 fuel cladding tubes.

Mechanical tensile tests after exposure to high temperature sodium, as well as corrosion test in sodium are continued for modified SUS316 and advanced alloy for future fuel cladding tube.

Caustic corrosion tests of SUS304 and SUS316 are continued in the primary environment of nitrogen including oxygen of 0 - 2% and H_2O of 0 - 14,800 ppm. Also, caustic corrosion tests of SUS304 have been started in the secondary environment of air at the temperatures of 325°C and 505°C.

Stress corrosion cracking tests of SUS304, SUS316 and SUS321 are continued in the wetted steam including oxygen and chlorine.

Self-welding, friction and corrosion tests are continued on Inconel 718, Stellite No. 6, Colmonoy No. 5 and Chronium carbide LC-1H. The tests are carried out under the variable condition as the parameter test.

8.2 Flow and Heat Transfer

A study on heat transfer from the liquid sodium surface to the reactor shield plug through cover gas space was completed.

The performance tests on some models of the convection barriers are now in progress. The 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 subassemblies and the flow controlling 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 tests in sodium are being made in order to verify the interrelation with the water test.

8.3 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 evaluating the radiation dose of the 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 temperautre	:	120 °C
test duration	:	4,000 hrs
sodium temperature	:	600 °C (hot leg)
		400°C (cold leg)

A computer code for corrosion product behavior in sodium is to be developed.

The basic studies for inhibition of corrosion product were started in 1979. A model test for inhibition of Mn- 54 will be made soon. Trapping of corrosion products using metals like wickel is also investigated.

8.4 Sodium Chemistry and Sodium Purification

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

A prototype on-line gas chromatograph was fabricated in 1979 and performance test was made in 1981.

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

A typical full size cold trap for FBR primary cooling system has been tested on an improved model based on data obtained from past experiences. The test includes quickened life test and regeneration test as well as ordinary characteristics test.

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

The results show that MONJU secondary cold trap is required to be heated to about 360°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 peformed in 1982 through 1983.

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

8.5 Sodium Removal and Decontamination

Since 1971, sodium removal tests have been carried out for the various sodium components at O-arai Engineering Center. Sodium removal experiences were obtained in the past two or three years about the following components.

a. MONJU fuel handling machine (mock-up)
b. MONJU intermediate heat exchanger (mock-up)
c. MONJU primary coolant circulation pump (mock-up)
d. JOYO primary coolant circulation pump
e. Various JOYO fuel subassemblies
f. Grapples of JOYO Fuel Handling Machine

The mock-up of JOYO reactor vessel with its internals and rotating plugs will be dismantled within next two years and new experiences will be obtained.

A feasibility study is being made on removing sodium from crevices.

Since 1976, study of radioactivity decontamination of primary system components are being made in laboratories of manufacturers.

The program consists of the followings;

a. Researches in the chemical decontamination method
b. Researches in the physical decontamination method
c. Studies on treatment techniques of the decontamination waste water
d. Miscellaneous studies.

8.6 Miscellaneous

Full size cold trap of MONJU primary cooling system has been tested on an improved model based on data obtained from past experiences. The test includes forced life test and regeneration test as well as ordinary characteristics test.

9. Development of FBR Instrumentation

9.1 Nuclear Instrumentation

9.1.1 In-Core Fission Chamber

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

At the final stage of this development, six chambers were irradiated in Japan Materials Testing Reactor (JMTR), and these chambers were disassembled in a hot laboratory to investigate the irradiation effect and high temperature, and thermal cycle tests were carried out on six other chambers for examination of the reliability.

9.1.2 Ex-vessel Fission Chamber

High performance fission chamber having sensitivity of 0.3 cps/nv and operating up to 550°C has been developed.

9.1.3 Ex-Vessel ^{10}B Linear Proportional Counter

A ^{10}B linear proportional counter was tested under the temperature up to 200°C. Temperature dependence of the characteristics was not observed except for the background pulse rate. Neutron sensitivity was about 12 cps/nv and it was kept constant even when the counter was exposed to gamma flux of 200 R/h.

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

9.2 Failed Fuel Detection and Location

9.2.1 FFD

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

9.2.2 FFDL

The tag gas system has been developed for locating the failed fuel subassembly. The adoption of the cryogenic adsorption system was made for MONJU.

A simulation test of the cryogenic adsorption system for MONJU is to be investigated within next two years.

The neutron irradiation test of tagging gas is to be carried out in the test subassemblies of JOYO MK-II core.

9.3 Early Warning System for Fuel Failure

9.3.1 Temperature Measurement

The performance and reliability of C.A. (Chromel-Alumel) thermocouples under irradiation condition by JMTR were investigated and failed thermocouples were inspected in a hot-laboratory. Thermocouple failures occurred frequently in the initial operation stage, but the numbers were fairly decreased later.

9.3.2 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 of the sensors are being carried out.

Flow blockage test and gas bubble detection tests are being performed using the flow sensors and seven mock-up subassemblies of MONJU fuel.

9.3.3 Other Systems

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 the acoustic propagation in subassemblies and through core structure.

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

The reactivity balance method and reactor noise analysis are being studied on the TCA (Thermal Neutron Critical Assembly) in JAERI since 1979. A correlation method using the reactivity and noise signals are being investigated.

9.4 Process Instrumentation

9.4.1 Sodium Flow Meters for Large Piping

Since the permanent magnet type flowmeter was adopted for the flow measurement of the primary and secondary system of MONJU, flowmeter response and calibration method became a major concern. Some tests related to these items are in progress at O-arai Engineering Center.

The durability test for a 12-inch ultrasonic flowmeter was carried out. A 24-inch ultrasonic flowmeter is being tested for the appliation to the calibration of the electro-magnetic flowmeter in MONJU.

A testing of on-site calibration technique using cross-correlation technique of EM Flowmeter noise signals was carried out in JOYO cooling system.

9.5 Surveillance

9.5.1 Under sodium viewer (USV) is being developed in two types designed for different purposes.

In-sodium characteristics test of the horizontal type USV system, which functions as a acoustic sweeper, has been carried out using a 3/10 scale model of MONJU with acoustic reflector assemblies.

A vertical type USV system, which visualizes the upper surface of the core barrel, demonstrated an imaging technique using the digital processing unit and CRT.

9.5.2 Sodium Leak Detection System

A sodium ionization detector and aerosol trapping filter were tested at the leak rate of the order of 100 g/h in the simulated environment of a primary cell of MONJU.

Testing in the simulated environment of secondary system of MONJU is being planned. In relation to this system, aerosol generation and diffusion are also investigated.

9.5.3 Displacement Sensor

A sensor based on the eddy-current principle was tested to measure the gap in the hydrostatic bearing of a sodium pump 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 mock-up has been performed at O-arai Engineering Center.

10. Fuel and Materials

10.1 Fuel Fubrication

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 soon. 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.

10.2 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 FP -cladding chemical interaction, and 1.5 meter long tubes were successfully coated by titanium using vacuum deposition method, and tubes were applied 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 analyze FCCI (Fuel Cladding Chemical Interaction) and CCCT (Clad Component Chemical Transport) mechanisms.

10.3 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 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 in France.

10.4 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.

10.5 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 was completed in 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.

11. Structural Design and Materials

11.1 Development of High Temperature Structural Design Method

11.1.1 Inelastic Structural Analysis Program

a. General purpose inelastic structural analysis program

The general purpose inelastic structural 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.

b. 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.

11.1.2 Development of Simplified Methods of Inelastic Analysis and Supplementary Design Rules

Although detailed analysis method is 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.

a. Simplified creep damage evaluation methods for structural discontinuities.

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

c. Simplified inelastic analysis methods of perforated plates.

d. Development of simplified evaluation method of local stresses at piping support.

e. Development of simplified analysis method for class 2 components.

f. Establishment of design application procedure of inelastic analysis method.

g. Development of constitutive equations for structural materials.

h. Analysis of inelastic piping benchmark problems.

i. Analysis of inelastic nozzle-to-sphere benchmark problem.

j. Development of postprocessors based on high temperature design code for class 1 vessels and piping of MONJU.

11.2 Structural Material Test

Fig. 11-1 shows the change of high temperature structural design standards and structural material test program. From this figure, it can be seen that these tests were under way in parallel with the construction of FBRs in every country. In Japan structural material test program phase II has been conducted under coordination of PNC. Research and development on structural materials have been reflected on the design of MONJU as shown in Fig. 11-2.

11.2.1 In-Air Structural Material Test

Table 11-1 and 11-2 show the summary of the in-air structural material test program. As the result of step I, many basic data were obtained which were used in the design of MONJU. Now, step II plan is in the final stage.

Next, we shall have the step III, and the first purpose is placed on the confirmation and verification of the Design Guide and the Material Strength Standard for MONJU.

The second purpose of step III program is to broaden the scope of research works to the demonstration plant.

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 in FBR component fabricaters. A new material strength test laboratory facilitated with ten high temperature low cycle fatigue testing machines and eighty creep testing machines was constructed in O-arai Engineering Center in 1981.

11.2.2 Structural Material Tests in Sodium

The following research works are in progress :

a. Corrosion and mass transfer

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

b. Carbon transfer

Carbon transfer test on bimetallic systems simulating the MONJU secondary systems

c. Mechanical strength tests in sodium
Determine the effect of the sodium environment on the mechanical properties of structural materials, such 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 in O-arai Engineering Center in 1979.

11.2.3 Irradiation - Effect Test on Structural Material

The tests on domestic 304 stainles steel have been conducted to ensure the safety of reactor vessel and internal components under the irradiated condition. The materials are mainly irradiated at JMTR of JAERI and are tested in O-arai Engineering Center.

Irradiation tests using JOYO are in progress.

11.2.4 Data Banking and Retrieval System

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

11.3 Test of Structural Elements and Components

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 or are being 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 ratchting 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 butt-weld joints and dissimilar metal joints.

g. Basic structural element behavior tests
   * Beam cyclic bending
   * Rod with stepped cross setion in cyclic tension and compression
   * Notched plate in cyclic tension and compression
   * Curved beam in tension and bending

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 FBRs in every country.

Table 11-1 Outline and Characteristics of In-Air Test Programme for FBR Structural Materials

Year	Outline and characteristics	Reactor	Material	Production Form	Material test item
Step I (1977, 1978)	1. Acquirement of basic strength data for base materials. 2. Acquirement of basic strength data for weld deposited metal. 3. Other tests • Pre-strain effect test on SUS304 materials • High-speed tensile test on SUS316 aged materials • Strength test on inconel 718 base materials	Prototype reactor	SUS304 SUS316 SUS321 2 1/4 Cr-1Mo (Inconel 718) and their deposited metal, welded joint	Plate Forged tube	Tensile test High-speed tensile test Creep test Fatigue test Creep fatigue test Relaxation test Materials behavior test, etc.
Step II (1979~ 1981)	1. Acquirement of basic strength data for base materials. (Supplement of Step I) 2. Examination of material strength test procedures for welded joint. 3. Acquirement of basic strength data of welded joint. 4. Other tests • Creep constitutive equation confirmation and preparation test. • Plastic and elastic behavior tests, etc.				
Step III (1982~)	1. Supplemental test to Steps I and II. 2. Various effect tests. 3. Tests in improved base material and welded material.	Demonstration reactor			

Table 11-2 Summary of the In-Air Material Test

<table>
<tr><th>Step</th><th>Purpose</th><th>Material</th><th>Product Form</th><th>Test</th></tr>
<tr><td>Step 1
(1977, 1978)</td><td>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</td><td rowspan="2">SUS304
SUS316
SUS321
and
their weld
metal,
welded joint</td><td rowspan="2">Plate
Forged
tube</td><td rowspan="2">· Tensile test
· High speed tensile test
· Creep test
· Fatigue test
· Creep-fatigue test
· Relaxation test</td></tr>
<tr><td>Step II
(1979, 1980)</td><td>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</td></tr>
</table>

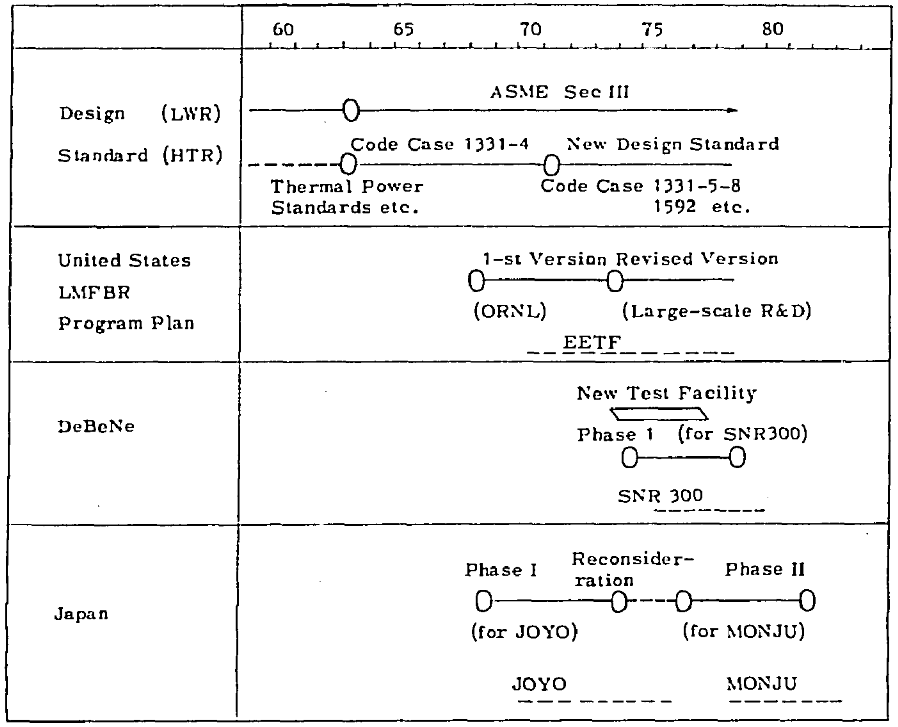


Fig. 11-1 Change of High Temperature Structural Design Standards and Structural Material Test Program

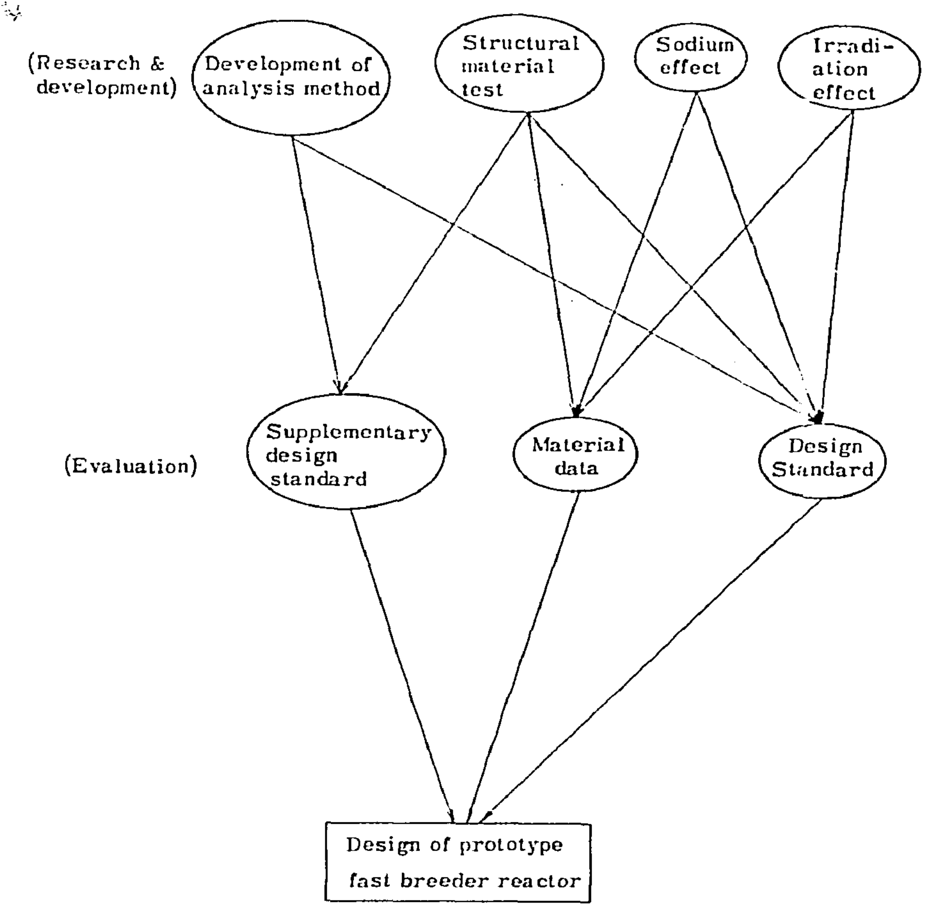


Fig. 11-2 Research and Development of Structural Materials and Their Reflection on the Design of Fast Breeder Reactor

12. Safety

12.1 Core Safety

12.1.1 Sodium Boiling

Two series of experiments were conducted with 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 analyzed.

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.

12.1.2 Fuel Failure Propagation

1) Pin contacts

The PICO-2 code was developed to analyze 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 has a central blockage which corresponds areawise to 38%. The temperature increase in the blocked regions, local boiling and FP gas release phenomena were investigated in these experiments.

The final local blockage experiments with a 91-pin bundle began in 1980. A stainless steel plate blocks a half of whole flow area.

3) Water loop experiments

Many test were conducted to study velocity profiles, pressure distribution, 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.

12.1.3 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 12-1. The power of electrically heated fuel pins was controlled at steady state of 130 or 160 W/cm and then a power transient was supplied to each pin individually.

The experimental results are shown in Table 12-2. 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 tubes 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 prssures 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.

12.1.4 Transient Undercooling Tests

The tests were originally aimed at confirming the integrity of MONJU cladding (50,000 MWD/T and 70,000 MWD/T) under a locally flow blocked (800 to 900°C, 24 hours) condition and an LOPI (160 to 200°C/sec for a few seconds) condition in Siloe reactor in France.

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

12.1.5 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.

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

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

12.1.7 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 intrnal 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 is investigated.

12.1.8 Accident Analysis Codes

1) Whole core accident analysis

Major efforts have been focused on developing an integrated 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 ad 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 codes 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 prediction is shown in Fig. 12-1 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.

12.2 Structural Safety

12.2.1 Shock Structure Experiments and Analyses

1) The second experiment on a pressure pulse propagation along a short pipe (6B x 3m - SUS304) with an elbow was peformed 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 analyzed. 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, etc.

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

In the 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 have 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. In the near future, high tensile tests will be performed on the low cycle fatigue damaged stainless steel under co-operation with Joint Research Center of ISPRA.

12.2.2 Seismic Design Tests

Safety R & D for LMFBR plant seismic des'gn 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 beavior during earthquakes, the analysis has been performed for two years using the weight propagation theory with topographical irregularity effects. The difference was studied between the dynamic behavior of the structure with and without irregularity of the shape and soil property of ground.

b. For studying 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 behavior 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 behavior 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 behavior of the fuel assemblies and the reactor vessel structure was confirmed by vibration tests using models after 1980. These dynamic test results are being analyzed.

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 with small amplitude began in 1979 for mechanical snubber. These tests will be continued till 1982.

12.2.3 Structural Integrity Tests for PHTS Piping

The phase-II study (FY 1973 - 1980) was performed to understand the structural integrity of the PHTS hot leg piping. The study is focussed 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 findings in the phase-II study are ;

a. The structural creep and elastic follow-up behavior 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 behavior of elbows.

d. Creep-ratcheting behavior of 304SS straight pipe and piping elbows under sustained axial tensile force and cyclic radial thermal gradient.

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

f. Thermal fatigue strength of piping tees.

12.3 Radiological Consequences

12.3.1 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) behavior, especially, the nature and kinetics of their deposition to the pipe walls from flowing sodium.

The in-pile 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 with 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.

12.3.2 FP and Pu Transport under Accident Conditions

1) HCDA bubble behavior 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 rates; 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 behavior model of FTAC is started at O-arai Engineering Center as is mentioned in 12.1.1.

2) Aerosol behavior in containment

Aerosol behavior 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 collosion shape factor values.

The result shows that the code can predict both sodium oxide and uranium oxide aerosol behavior 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 postulated severe accidents. The code CEDAN calculates the pressure and temperature history of up to four cells which may be inter-connected among thermosleeves 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 12-1 Experimental Conditions for Molten Core Material Interactions

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 (°C)	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 to 1.33)	(0 to 2.03)	(0 to 0.8)	(0 to 0.4)	(0 to 1.8)

Table 12-2 Experimental Results for Molten Core Material Interactions

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)	-	1.2	1.3	0.5	0.5
Full width at half maximum (msec)		0.6	0.3	0.4	0.5

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

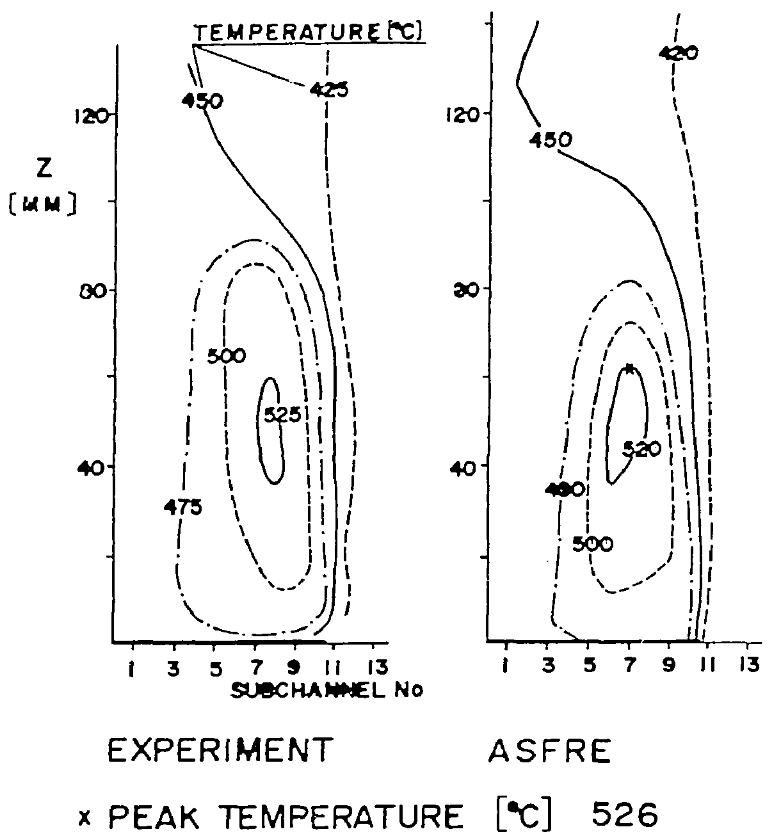


Fig. 12-1 Benchmark Calculation 49% Central Blockage
Run No.1 Comparison of Temperature Distribution

A REVIEW OF THE UK FAST REACTOR PROGRAMME

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United Kingdom Atomic Energy Authority,
Risley Nuclear Power Development Establishment,
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ABBREVIATIONS USED IN THIS REVIEW

AEA	United Kingdom Atomic Energy Authority
AEEW	Atomic Energy Establishment, Winfrith
AERE	Atomic Energy Research Establishment, Harwell
AGR	Advanced Gas Cooled Reactor
BNL	Berkeley Nuclear Laboratories (CEGB) Gloucester
BPD	Burst Pin Detection
BWR	Boiling Water Reactor
CDFR	Commercial Demonstration Fast Reactor
CEGB	*Central Electricity Generating Board*
CERL	Central Electricity Research Laboratory, Leatherhead
CFR	Commercial Fast Reactor (The future series of LMFBR following CDFR)
COVA	COde VAlidation Experiments (Containment studies)
CSNI	Committee on the Safety of Nuclear Installations
DFR	Dounreay Fast Reactor
DNE	Dounreay Nuclear Power Development Establishment
DPA	Displacements per Atom
EDTA	Ethylene diamene tetra acetate
ERA	Electrical Research Association
HALIP	Helical Annular Linear Induction Electromagnetic Pump
HCDA	Hypothetical core disassembly accident
IHX	Intermediate Heat Exchanger(s)
IRD	International Research and Development Co, Newcastle
KfK	Kernforschungszentrum, Karlsruhe
KNS	Compact Sodium Boiling Loop, KfK
LDE	Low density explosive
MEL	Marchwood Engineering Laboratory, CEGB
MFCI	Molten Fuel/Coolant Interaction
MMA	Manual metal arc (welding)
NII	Nuclear Installations Inspectorate
NNC	National Nuclear Corporation
NRT	Norgett-Robinson-Torrens (model of neutron displacement cross-sections)
PFR	Prototype Fast Reactor, Dounreay
PIE	Post Irradiation Examination
RNL	Risley Nuclear Power Development Laboratories
SNL	Springfields Nuclear Power Development Laboratories
SRD	Safety and Reliability Directorate, UKAEA
TIG	Tungsten-Arc Inert Gas (Welding)
TREAT	Transient REActor Test Facility, Idaho
ZEBRA	Zero Energy Breeder Reactor Assembly, Winfrith