The tests had the objective of evaluating the behaviour of the sealings under the nominal running conditions of SPX-1, reference conditions for the future European reactor, and under particularly critical conditions (a reduction in 50% in the supply of the lubricating oil, the presence of vibrations modelling seismic effects, the simulation of emergency scram) so as to evaluate the functioning limits.

After a total running time of 900 hours, the sealings were taken down and subjected to damage analyses. In particular the upper sealing resulted in an optimum condition with no signs of wearing of any import, whilst the lower one showed indications of blistering on the graphite ring, a sign of high local heating.

This phenomenon is a consequence of an absorption and subsequent thermal expansion of the lubricating oil in the porous graphite, and happens in the first instants of load of the component.

On the basis of the acquired experience, even though limited by the relative shortness of the testing period, a positive opinion on the design of the sealings (mechanical design, choice of the lubricating system, materials employed) was expressed.

In the framework of the same research contract, the construction of the impeller and diffuser castings was completed.

The technological development of the castings of mechanical pumps for fast reactors has provided results which may be summarized as follows:

a) the castings produced, have demonstrated the viability of producing complex castings on the basis of stringent norms (RCC-MR-3411);
b) the programme of activity has finally allowed the system of quality assurance of the company producing the castings to be tested with positive results, the producer being thus qualified as a supplier of Nº 1 category.
A special test cycle operation, called the 15'th cycle operation, was performed after the 15th duty cycle operation, starting in June 1988. The sensitivity calibration was conducted for the failed fuel detection system (FFD) which provides both precipitation system of over and the delayed neutron detecting system and also newly installed cover gas on-line ray monitoring system, in this test cycle.

Following the 15'th cycle operation, the 16th duty cycle operation was started at the beginning of August in 1988. An erradiator test rig which was composed of some French materials was also loaded in the core from this duty cycle according to the French-Japan exchange irradiation program. In addition, demonstration of the operator guidance system at the reactor emergency which utilizes AI technique was also performed in this duty cycle.

The 7th periodical inspection of the reactor was carried out from the beginning of September 1988 to the middle of January 1989, including the work to replace the test specimens in upper plug ring (UPR) with new ones. A newly developed cover gas clean-up system (CGCS) using cryogenic adsorption method was installed during the inspection period.

The interim examination during irradiation period was also performed with some of the irradiation rigs in the core, for the first time. The examination was conducted in irradiated fuel monitoring facility (FMP) adjacent to JOYO site by disassembling the test rigs which had been transferred from the reactor core. After the examination, the irradiation rigs were assembled and were loaded in the reactor core again to continue their irradiation test, avoiding any problem concerning alpha-contamination.

The reactor commenced the 17th duty cycle operation after the 7th periodical inspection, having fifteen test articles in its reactor core, and is scheduled to stop the operation at the beginning of April 1989 for refueling.

1.2 Special Test using Fission Products Source

A fission-product-source (FPS) was irradiated in the 15th cycle operation to provide data for calibrating the facility's fuel failure detection system (FFD), and for demonstrating the flux tilting and the triangulation.
methods to optimize the failed fuel location system for DFBRs.

As the fission products source, a special test assembly which was composed of U-Ni alloy pins without cladding was loaded in the core. In addition, the verification of the power tilting method and the triangular method to identify the location of a failed fuel in the core was also conducted by changing the insertion pattern of the control rods and the flow rate of the primary coolant, respectively.

The FPS test was successful and most of the objectives for it were achieved. The followings are the major results; (1) all of the cover gas (CG) and the delayed neutron (DN) monitoring systems were successfully calibrated, (2) major constants for CG & DN-activity models were determined to confirm validity of computer codes, particularly Disengagement-rate Constant of fission gas from the coolant to the cover gas was found out to depend on the flow rate of the primary coolant, and (3) failed fuel location by the flux tilting and the triangulation was demonstrated, particularly in which the flux tilting method using an online gamma-ray monitor was found out to be very useful.

Schematic diagram of the experiment is illustrated in Fig.1.2.

1.3 MK-III core program

Investigation of MK-III core program has been started. The reactor core will be expanded radially, in order to increase the irradiation test rigs which can be loaded in the core at one time. In addition, to decrease the time needed for the irradiations objectives, both the modification of the fuel subassembly to obtain higher neutron flux and the improvement of the fuel handling system to shorten the outage time for fuel exchange are investigated. R&D of innovative technologies are also planned. For instance, studies needed to eliminate the secondary heat transport system of a fast breeder reactor are picked up as a future plan.

2. Prototype Fast Breeder Reactor, MONJU

Summary

"Monju", a prototype fast breeder reactor in Japan, is now under construction. Following the erection of reactor containment vessel, the construction of reactor system was started in parallel with construction of the auxiliary systems and plant buildings. This paper describes overall Monju project and covers its project organization, key design features, manufacturing, and construction.
Introduction

The construction of the Monju FBR by the Power Reactor and Nuclear Fuel Development Corporation (PNC) is now under way at the Shiraki site located on the Tsuruga Peninsula of Fukui Prefecture, which is approximately 400 km west of Tokyo.

Monju is a 280 MWe, loop type power reactor using plutonium-uranium mixed oxide as fuel, liquid sodium as a coolant. The Monju plant incorporates the results of numerous research and development programs and the experience gained from the experimental fast reactor "Joyo".

Project Organization

The construction of Monju represents a major milestone in Japan's national FBR development plan based on the "Atomic Energy Commission's Long-Term Program on the Development of Nuclear Energy".

PNC, a semi-governmental organization under the Science and Technology Agency (STA), is responsible for the management of the project. The Japan Atomic Power Company (JAPC), on behalf of the nine Japanese electric utilities and the Electric Power Development Corporation (EPDC), is responsible for the construction of Monju, under a contract with PNC. This cooperation is a quite important factor in the construction of Monju. It utilizes PNC's experience in FBR development and JAPC's experience in LWR technology. It also serves as a vehicle to transfer the FBR technology from the government to the utilities and industries, for future commercialization of the FBR.

Many firms are participating in this project in order to gain experience and develop know-how through construction of the Monju FBR power plant.

Design

The Monju is a 280 MWe (714 MWth) cooled by liquid sodium in three loops. Monju's principal plant design and performance parameters are shown in Table 2-1. The designed average fuel burnup is 80,000 MWD/T and steam temperature and pressure at the turbine inlet are 483°C and 127 kg/cm²g respectively.

A. Reactor System

Monju's reactor core consists of 198 core fuel subassemblies, with 19 control rods.

To flatten the neutron flux distribution, the core is divided into two radial zones of plutonium enrichment defined by Pu-fissile/(Pu+U) = 16% for the inner zone and 21% for the outer zone in the equilibrium cycle core.

The reactor vessel is fabricated from twelve large seamless integrated forgings composed of type 304 stainless steel, in order to reduce the number of parts and to shorten welding length and improve reliability.

The reactor vessel is 17.8 m in height and 7.8 and 7.1 m in diameter at the upper and lower part respectively. The wall thickness is 50 mm.

A guard vessel is provided to retain any spilled sodium and assure reactor core coolability during an accident.

A closure head, placed on the sole plate of the reactor vessel, consists of a stationary plug and a rotating plug. An upper core structure and a fuel handling machine are installed through the rotating plug.

Primary sodium coolant enters the reactor vessel through three inlet nozzles provided at the lower part of the reactor vessel. The sodium temperature at the R/V inlet nozzles is 397°C, while the temperature at the upper outlet nozzles is maintained at 529°C under normal operation condition.
### Table 2-1. Principal Design and Performance Data of Monju

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Type</td>
<td>Sodium cooling loop</td>
</tr>
<tr>
<td>type/Thermal Power</td>
<td>714 MW</td>
</tr>
<tr>
<td>Electrical Power</td>
<td>about 280 MW</td>
</tr>
<tr>
<td>Fuel Material</td>
<td>PuO₂-UO₂</td>
</tr>
<tr>
<td>Core</td>
<td>Equivalent diameter 1,790 mm</td>
</tr>
<tr>
<td></td>
<td>Height 930 mm</td>
</tr>
<tr>
<td></td>
<td>Volume 2,335 lit.</td>
</tr>
<tr>
<td>Pu Enrichment</td>
<td>(Pu fissile %)</td>
</tr>
<tr>
<td>Initial core</td>
<td>15/20</td>
</tr>
<tr>
<td>Equilibrium core</td>
<td>16/21</td>
</tr>
<tr>
<td>Fuel Inventory Core (U + Pu metal)</td>
<td>5.9 Ton</td>
</tr>
<tr>
<td>Blanket (U metal)</td>
<td>17.5 Ton</td>
</tr>
<tr>
<td>Average Burn-up</td>
<td>80,000 MD/kg</td>
</tr>
<tr>
<td>Cladding Material</td>
<td>SUS316</td>
</tr>
<tr>
<td>Cladding outside Diameter/Thickness</td>
<td>6.5/0.47 mm</td>
</tr>
<tr>
<td>Permissible Cladding Temperature</td>
<td>675°C</td>
</tr>
<tr>
<td>(middle of thickness)</td>
<td></td>
</tr>
<tr>
<td>Power Density</td>
<td>283 kW/lit.</td>
</tr>
<tr>
<td>Blanket thickness</td>
<td>Upper 300 mm</td>
</tr>
<tr>
<td></td>
<td>Lower 350 mm</td>
</tr>
<tr>
<td></td>
<td>Radial 300 mm</td>
</tr>
<tr>
<td>Breeding Ratio</td>
<td>1.2</td>
</tr>
<tr>
<td>Reactor in/out Sodium Temperature</td>
<td>397/525°C</td>
</tr>
<tr>
<td>Secondary Sodium Temperature</td>
<td>505/325°C</td>
</tr>
<tr>
<td>(IHX outlet/IHX inlet)</td>
<td></td>
</tr>
<tr>
<td>Reactor Vessel</td>
<td>17.8/7.1 m</td>
</tr>
<tr>
<td>Number of loops</td>
<td>3</td>
</tr>
<tr>
<td>Pump Position</td>
<td>Cold leg</td>
</tr>
<tr>
<td>(Primary and secondary loop)</td>
<td>Helical coil, once-through unit type</td>
</tr>
<tr>
<td>Type to steam Generator</td>
<td></td>
</tr>
<tr>
<td>Steam Pressure (turbine inlet)</td>
<td>127 kg/cm²g</td>
</tr>
<tr>
<td>Steam Temperature (turbine inlet)</td>
<td>463°C</td>
</tr>
<tr>
<td>Steam Generators</td>
<td></td>
</tr>
<tr>
<td>Refueling System</td>
<td>Single rotating plug</td>
</tr>
<tr>
<td></td>
<td>with fixed arm FHM</td>
</tr>
<tr>
<td>Refueling Interval</td>
<td>6 months</td>
</tr>
</tbody>
</table>

---

**B. Heat Transport System**

The main cooling system consists of three primary and three secondary sodium loops and a steam-water loop. These loops are thermally connected through three intermediate heat exchangers (IHX) and three steam generators (SG). Each sodium loop contains a circulation pump which can be driven either by a main motor or a pony motor. The steam-water loop has two feed pumps driven by steam turbines and a feed water pump driven by steam turbines and a feed water pump driven by motor for plant startup.

The IHX is a shell-and-tube type heat exchanger in which primary sodium flows in the shell side and secondary sodium in the tubes.

Steam generating system consists of one evaporator and a superheater per loop. Both the evaporator and superheater have helically-coiled heat transfer tube bundle, whose material is 2 1/4 Cr-1Mo for the evaporator and type-321 stainless steel for the superheater. A safety system is provided to reduce build-up pressure and to release reaction products for each loop independently in the event of a sodium-water reaction. These systems are designed assuming double ended guillotine rupture of one plus three heat transfer tubes.

Three independent auxiliary cooling systems (ACS) are also installed to remove decay heat after reactor shutdown as well as during the maintenance and refueling periods.

**C. Refueling System**

The Monju fuel handling system consists of a fuel handling machine for refueling in the reactor vessel, and both ex-vessel and an in-vessel transfer machine for transferring fuel between the reactor vessel and the sodium filled ex-vessel fuel storage tank (EVST). After cleaning and cannning, the spent fuel assemblies are stored in the spent fuel storage pool after cooling in the EVST for one and a half years, then transported to a reprocessing plant.
D. Reactor Building and Other Buildings

The reactor building consists of the reactor containment vessel (C/V) and a shielding building, to prevent release of radioactive materials to the atmosphere in case of an accident. The reactor building is surrounded by the reactor auxiliary building to achieve the aseismic design.

The design of the reinforced concrete buildings, in particular the reactor building and the reactor auxiliary building, was performed incorporating high temperature design conditions in order to maintain the integrity of building under sodium fire accident conditions.

Manufacturing

Monju's main components and systems are being manufactured by Toshiba, Hitachi, Fuji Electric, and Mitsubishi Heavy Industries. FBR Engineering Company, which coordinates and integrates the work of the manufacturers, was formed by the four Monju reactor manufactures.

The reactor vessel was fabricated on schedule, then transported by ship to the site and installed in October 1988.

The IHX is currently in the final stages of assembling of heat transfer tube bundles. Assembly of the SG heat transfer tubes has been completed.

Three overflow tanks and two drain tanks for secondary sodium systems and two tanks for the EVST cooling system have also been fabricated and installed.

Construction

The Monju site is located on the north side of the Tsuruga Peninsula, facing the Sea of Japan and is surrounded by mountains of approximately 300-700m high. Since the plant is located inside the Wakasa Bay quasi-national park, its construction has been carried out with special attention to the environment.

A. Construction Schedule

The construction schedule is as follows:

<table>
<thead>
<tr>
<th>Date</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>Oct.1985</td>
<td>Start of Construction</td>
</tr>
<tr>
<td>Apr.1987</td>
<td>Completion of Construction of the Reactor Containment Vessel</td>
</tr>
<tr>
<td>Oct.1988</td>
<td>Installation of the Reactor Vessel</td>
</tr>
<tr>
<td>May.1991</td>
<td>Start of Functional Testing</td>
</tr>
<tr>
<td>Oct.1992</td>
<td>Initial Criticality</td>
</tr>
</tbody>
</table>

The critical path consists work inside the reactor containment vessel, such as the installation of the reactor cavity wall, reactor vessel, primary heat transport systems, shielding plug on the reactor vessel and control rod drive mechanism. After completion of the construction, pre-operational and startup testing will be performed.

B. Present Status of Construction

Monju construction was 63.5% complete as of the end of February 1989, including design, manufacturing, and construction at the site.

The site work has been completed, except for construction of the cooling water intake structure. Overall site work is now 89% complete.

The total area of the site is approximately one square Kilometer and the plant construction area occupies 0.36 square kilometer. At the grading of the site, 2.3 million cubic meters of soil were excavated. One million cubic meters of the soil were used in banking the hill side, reclaiming land from the sea in front of the site respectively, and the rest for creating of temporary laydown site.

Site preparation started in January 1983, including construction of an approach road and hewing trees in the construction area.
In April, 1983 grading works were commenced, followed by construction of a 490m long revetment. The construction of a 320m long breakwater and 160m long landing wharf, made of caissons, followed in 1987. The intake pump house is now under construction, and cooling water intake is to be constructed in 1989.

In March 1985, two approach tunnels were opened to transport construction materials and equipment to the site.

Monju plant construction was initiated on October 25, 1985, with the plant excavation. After inspection of the integrity of basement rock surface by JMA and MITI (the Ministry of International Trade and Industry) in March 1986, basement concrete placement was started in May 1986. Construction of the buildings is currently 60% complete.

Erection of the reactor containment vessel was initiated on July 1, 1986, with the placement of 24 temporary vessel support posts. Installation of the polar crane was completed in December 1986. The erection of the containment vessel was completed on February 24, 1987, with installation of the top hemispherical dome section. The final weld examinations were performed during March, 1987.

A one hour C/V pressure test, a three hour soap bubble test and a twenty-four hour integrated leak rate test were successfully completed in April 1987. After these tests, a temporary opening was cut in the containment vessel for transporting architectural materials and various big components such as the reactor vessel, guard vessels and reactor cavity wall blocks.

An outer shield building, in the form of cylindrical reinforced concrete structure, covers the reactor containment vessel for radiation shielding and is currently under construction.

Inside the reactor containment vessel, erection of modular reactor cavity structure, forming a biological shield wall, was started in August 1987, and completed in May 1988, with placement of concrete into the structure. The guard vessel of the reactor vessel was installed on June 18, 1988 and the reactor vessel was installed on October 24, 1988 on the intermediate floor and pedestal floors respectively. These floors are welded to the reactor cavity wall structure.

Elsewhere in the reactor containment vessel, cell liner installation is currently going on in the primary heat transfer system cells. Liner installation is closely related to the placement of concrete, and on the critical path of overall Monju construction.

C. Reactor Containment Vessel

The reactor containment vessel is cylindrical, with a hemispherical dome and a ellipsoidal bottom panel. It is fabricated from 4400 tons carbon steel plate.

The dimensions of the vessel are: 49.5m in diameter, 79.4m in height, wall thickness of 38mm for the cylinder and the bottom panel, and 19mm for the domed top panel.

The bottom panel of the vessel is embedded into the structural concrete of the basement approximately up 13.5m from the bottom of the vessel. Above that level, the containment vessel is freestanding, surrounded by a 1.5m wide annulus and a 1.0m to 1.8m thick reinforced concrete outer shield building. The domed top of the containment vessel is surrounded by a 0.45m thick reinforced concrete dome.

An annulus seal is installed between the outside of the containment vessel and the shield building. This annulus is maintained at a slightly negative pressure to prevent the direct release of radioactive material from the containment vessel during an accident.

The design pressure of the containment vessel, 0.5kg/cm² as the maximum internal pressure, maintains structural integrity under various conditions including seismic aspects, and testability of pressure and leak rate.
On the other hand, slightly negative internal pressure of 0.05 kg/cm$^2$ is considered in the design due to oxygen consumption when a fire of spilled sodium is assumed in the containment vessel. In order to prevent the buckling of the cylindrical part of the containment vessel, six stiffener rings are welded on the outer surface of the cylindrical part.

D. Lining System

In the rooms where sodium systems and equipments are installed, the concrete surfaces which may possibly be in contact with spilled sodium, are covered with steel lining plates.

Sodium spills are considered as a design basis event based upon a leak in a sodium piping having an area equivalent to one quarter the pipe diameter times the pipe wall thickness.

Monju uses two types of lining systems: a cell liner and a catch pan.

The cell liner is used for radioactive sodium cells in which the primary or EVST sodium systems' equipments are installed, while the catch pan is used for non-radioactive sodium cells in which the secondary sodium system's equipments are installed.

In the cell liner systems, following two types are used: a 'fixed liner' and a 'semi-floating liner'.

The fixed liner system is used in radioactive primary sodium cells located inside the containment vessel, in which the reinforced concrete walls and slabs are thick enough to accommodate the in-place liner forces generated by the thermal expansion of the fixed liner system. Six millimeter thick carbon steel liner plates are welded to corner anchors and frame plates, both fixed to the structure concrete. The thermal expansion of the liner plates is therefore absorbed by buckling of the liner plates between anchors and frame plates.

The liners are designed to maintain the following temperature limits: 65°C at normal operation condition, 135°C at loss of HVAC, and 530°C under DBA sodium spill condition.

The normal operating condition atmospheric leak rate for the cell liner system is 1.0% volume/day at a pressure differential of 100mm water gage. Under sodium spill accident conditions, however, the allowable atmospheric cell leak rate is 100%/day at a pressure differential of 100mm water gage.

The majority of fixed cell liners are fabricated by the 'before concrete casting method': prefabricated cell liner panels, consisting of the liner plate, air gap and insulating concrete panel, are installed, then the structural concrete is placed into the formed wall.

The remaining cells use cell liners fabricated by the 'after concrete casting method', in which the cell liners are installed after the structural concrete is placed.

The semi-floating liner systems is used for radioactive sodium cells of EVST system located in the reactor auxiliary building.

These cells are characterized by lower temperature DBA sodium spill accident conditions and by less massive structural concrete sections.

The semi-floating liner system consists of a 4.5mm thick carbon steel liner plates supported by a series of strip embedments, which are set into pockets in the structural concrete floors, walls and ceilings. A layer of precast perlite insulating concrete is placed between the face of structural concrete and the backface of the liner plates. The pockets are filled with rockwool insulation allow for the lateral deformation of the stud anchors of the strip embedent and to give the system its flexibility. The liner used curved corners to provide its flexibility and to absorb the thermal expansion of the liner plates.
The catch pan is used for non-radioactive secondary sodium cells located in the auxiliary building. All secondary spilled sodium is drained from open catch pan cells by means of 12 inch diameter downcomers into catch pans with fire suppression decks, which are located at the basemat level of the auxiliary building and provide to mitigate the sodium fire by limiting the air flow.

The catch pan seismic anchors are installed in parallel with the installation of the floor reinforcing and wall dowels, and then floor concrete is placed. The wall reinforcing is installed followed by the placement of wall concrete. After the completion of the concrete cell, liner frames used to support the liner plates, which are 6mm thick flat carbon steel plates, are placed on the base plates to provide a sliding surface for the thermal expansion of the liner plates.

Perlite boards are then placed on the structural concrete floor for thermal insulation. Finally, the liner plates are installed and welded to liner frames.

The installation of the catch pan with fire suppression decks is the same as those of the catch pan described above, except for the installation of the fire suppression deck.

E. Reactor Cavity Cell Structure

The Monju reactor cavity cell structures are quite unique, comprising the biological shield wall. The structures consist of four modular rings, a reactor vessel support pedestal ring and an intermediate guard vessel support pedestal ring.

The interior and exterior surfaces of the cavity wall are constructed from 25mm thick carbon steel plates and comprise the flanges of a plate girder. These wall plates are separated by 16mm thick and 2.25m wide vertical stiffening plates. The thickness of the structures was selected in order to reduce the radiation level of the primary cooling system room less than $10^4$ n/cm² sec.

The reactor cavity walls and the support pedestals are designed as steel structures, with the concrete placed into the structures considered as non-structural acting only as a shielding material.

The consequences of a primary sodium spill in the reactor cavity are minimized by using the guard vessel and the overflow tanks to collect the spilled sodium.

The reactor cavity design temperatures are as follows:

- 40-80°C for normal operating conditions
- 100°C for lose of HVAC conditions
- 140-150°C for DBA sodium spill conditions

Installation of the first ring of the modular reactor cavity structure was started in August 1987. The six modular wall assemblies prefabricated at a factory of manufacturer were installed on the wall anchorage embedment, welded, inspected and filled with unreinforced structural concrete.

Similarly installation of remaining three rings of the modular reactor cavity structure and the guard vessel support pedestal was completed in May 1988. On the top of the forth ring of structure, the reactor vessel support pedestal ring, consisting fo six blocks made of 50mm carbon steel, was installed and filled with serpentine concrete.

Conclusion

Monju construction work is progressing on schedule, with the cooperation of many persons and organizations and with great efforts being concentrated on keeping the schedule, quality assurance, and safety.

Construction is scheduled to be completed in April 1991, followed by commissioning tests and then initial criticality in October 1992.
3. Demonstration Fast Breeder Reactor, DFBR

3.1 Overview

The Japan Atomic Energy Commission revised Japan's "Long-Term Program for Development and Utilization of Nuclear Energy" in June 1987. In the program, it was concluded that the research and development for demonstration FBRs should be done with the cooperation of governmental and private sectors, and that utilities should play a major role in the design, construction, and operation of the demonstration FBR, aiming at the commercialization in the year from 2020 to 2030 through construction of several FBRs with a step-by-step improvement of technologies and economics.

In this respect, it is requested to clarify a long-term strategy for FBR deployment by 1990, based on the result of research and development activities made so far and to be obtained during the next few years, along with the determination of the basic specifications of the first demonstration FBR (DFBR-1). At present, the start of construction of DFBR-1 is scheduled in late 1990's and the start of operation at the beginning of the 21st century. The scenario of DFBR-1 development is shown in Fig.3-1.

3.2 Design Study

The utilities finished their three years program of the design studies in March 1987. The program was begun in 1984 by the FBR Project Office (FPO) of the Federation of Electric Power companies (FEPC) and later transferred to the Japan Atomic Power Company (JAPC). 1000MWe class loop type and pool type reactors were studied aiming at a construction cost less than 1.5 LWRs with the same level of safety and maintainability. The present study by JAPC proceeds to evaluate the above rationalized design studies adopting innovative and advanced technologies, from the viewpoint of economics and competitiveness with PWRs.

In 1988 JAPC evaluated the plant maintainability on both Pool Design and Loop Design. In 1989 JAPC is expected to start the conceptual design for the demonstration FBR.

The Power Reactor and Nuclear Fuel Development Corporation (PNC) had also carried out the design evaluation studies of large LMFBRs based on the experience of JOYO and MONJU.

In 1988 PNC started the "FBR plant design study" which includes the Large FBR plants design and the Medium and small sized FBRs design aiming at FBR commercialization in early 2000.

4. Reactor Physics

4.1 Benchmark Tests of JENDL-3T

A temporary nuclear data file JENDL-3T has been generated for testing an evaluated data file of JENDL-3. To assess the adequacy of JENDL-3T data for use in nuclear designs and applications, benchmark calculations are
required for a number of critical experiments for thermal and fast reactors. Fast reactor benchmark calculations are performed for twenty-two benchmark cores selected from the ZPR, PCA, ZEBRA, SNEAK and VERA critical experiment series. These calculations are based on one and two dimensional diffusion theories.

As the results of fast reactor benchmark tests, the $k_{\text{eff}}$ calculated with the JENDL-3 data was overpredicted for U-cores and underpredicted for Pu-cores. The reaction rate ratios of $C/9$ and $F/9$ were overestimated for the JENDL-3 data. On the other hand, Doppler and sodium void reactivities, and reaction rate distribution obtained by using the JENDL-2 data were significantly improved by using the JENDL-3 data.

The present benchmark tests of JENDL-3 showed that nuclear data to be reevaluated are $\nu$, fission cross section and fission spectrum for $^{235}$U, fission cross section and fission spectrum for $^{239}$Pu, and capture and inelastic scattering cross section for $^{238}$U until the final compilation of JENDL-3.

4.2 Analysis Method Development

A group constant set was adjusted by using integral physics parameters obtained from fast critical experiments. Radial dependences of $C/E$ (Calculation/Experiment) results were used for the adjustment. As the result of adjustment, the $^{239}$Pu fission cross section was increased by 2% for the energy range below 10keV, and the $^{238}$U capture cross section was decreased by about 6% for the 1keV-1MeV range.

4.3 Critical Experiment and Analysis

The JUPITER-III program was a joint physics large FBR core critical experiments program between the USDOE and PNC of Japan, started on January 1987 using the ZPFR facility at ANL–Idaho. From the Japanese point of view, this is a research program in the area of core for the Demonstration FBR plant in Japan, and has received technical problems raised from the design studies.

The first half of the JUPITER-III program, the ZPFR-17 program, was physics benchmark experiments to study the neutronic behavior of a large, axially heterogeneous LMR core. The ZPFR-17 assembly was of about 650MWe size, and consisted of three phases, designated 17A, 17B and 17C. ZPFR-17A was a physics benchmark with no control rods (CRs) or control rod positions (CRPs). The most extensive set of measurements was made in this configuration to investigate basic characteristics of an axial heterogeneous core ZPFR-17B and ZPFR-17C are both engineering benchmark cores containing 25 CRPs and 13 half-inserted CRs, which are simulating EOC and EOC, respectively.

The experiments have been analyzed in Japan, using the JENDL-2 library. A radial variation of $C/E$ (Calculation/Experiment) results observed for reaction rates, control rod and sample reactivity worths. A more linear axial profile of control rod integral worth was observed, compared with the result of homogeneous core. Reaction rates and small sample worths were significantly underpredicted in the internal blanket region.

4.4 Benchmark Shielding Experiment

A series of neutron penetration experiments performed at the ORNL Tower Shielding Facility as the first experiment of JASPER, the joint experimental research program between US DOE and Japan's PNC.

Neutron attenuations were measured for multilayer configurations composed of steel, sodium, graphite and boron carbide, which are representative benchmarks of the radial shield designs in Japanese advanced LMFBR concepts. Analyses were performed using Sn transport codes with two types of cross section data sets, JSD100 based on ENDF/B-IV and JSDJ2 based on JENDL-2. The calculational results were compared with the measured integral count rates of Bonner
ball detectors and the fast neutron spectra measured by NE-213 and Benjamin counters. The calculation to experiment (C/E) values for Bonner ball count rates were between 1.6 and 0.80 for JSD100 and between 1.1 and 0.60 for JSDJ2 respectively. They also showed the same decreasing trend versus depth of penetration into the multilayer shields. The newly confirmed precision accuracies of the analysis method made it possible to reduce the current margins for radial shields in Japanese demonstration LMFBR designs.

4.5 Core Performance Test at "JOYO"

At the fast experimental reactor "JOYO", feedback reactivities were investigated on the MK-II core. A feedback reactivity is composed of an effect due to fuel and an effect due to structural materials and coolant. The reactor was operated with various power levels and flow rates of coolant different from rated ones to separate each effect which contributes the feedback reactivity at power change.

Measured results have been analyzed by using the JOYO core analysis code "MAGI," and it was made clear that about 55% of the reactivity worth at the rated power was attributed to the effect due to fuel and the rest, i.e., about 45% was attributed to the effect due to structural materials and coolant.

4.6 Advanced Core Design Study

To realize very hard neutron spectrum, a fast reactor core is composed by FP gas purge/tube-in-shell type metallic fuel assemblies. They are cooled by cooling sodium flowing through cooling tubes which penetrate packed hexagonal fuel rods in a shell. The shell is filled with filling sodium which retains the most of fission products other than inert gas. By application of FP gas purge/tube-in-shell metallic fuel assembly, volume fraction of fuel is increased and those of sodium and structural material are decreased in comparison with a conventional pin-bundle type fast reactor core. Therefore, many advantages are expectable as 1) very high breeding ratio 2) low enrichment fuel 3) better neutron economy 4) unnecessary excess reactivity for burnup 5) easier FP gas purge 6) removal of limit for high burnup by internal FP gas pressure 7) shorter gas plenum 8) thinner cooling tubes. The breeding ratio of the core is very high upto 1.9 and doubling time is 6 to 7 years. Furthermore, even with low enrichment uranium metallic fuel, fuel breeding can be expected by this concept.

5. Reactor Components

5.1 Reactor Vessel and Internal Structure

5.1.1 Hydraulic Tests of Flow Distribution

Experimental studies of hydraulic characteristics in the Monju reactor vessel have been completed with an integral flow model of 1/2.14 geometric scale using water as a working fluid. Currently, the Monju design is being reevaluated using the test data obtained.

5.1.2 Transient Thermohydraulics in Reactor Plenum

A flow visualization study of an outlet plenum was conducted to examine the transient thermo-hydraulic phenomena under such an accident as check valve stick, resulting in the reverse flow in a primary heat transport system. It is important to understand these phenomena in order to evaluate the structural integrity of the reactor vessel, outlet nozzles, etc. under such off-normal conditions. The 1/6-scale transparent acrylic model of the Monju reactor vessel was used to allow flow visualization. The hot (50 C)/cold(20 C) water was used as a working fluid and the flow visualization was realized by the aid of tufts and dye injection. Flow patterns observed by the experiments well describe thermal behavior in
the plenum and outlet nozzles. These experimental results are very helpful in developing simplified mixing models in reactor plena applied to whole plant transient analyses.

5.2 Shield Plug

Sodium vapor (or mist) concentration in the argon cover gas space was measured to provide basic data used for evaluating sodium deposition rates at relatively low temperature conditions. The tests were performed in the range from 150°C to 300°C. Axial temperature difference was also a dependent variable (T=70 - 150°C) in the tests. Sodium concentration was decreased logarithmically with decreasing pool temperature, and there was almost two order of magnitude difference between the present data (3X10^-10 - 9X10^-10 g/cm) and saturated vapor concentration (1X10^-10 g/cm) at 200°C. The cause of this large difference is considered to be attributed to mist formation at just above pool surface where sodium vapor should nucleate due to steep temperature drop. The above concentration data was reflected on the evaluation of sodium deposition rates for the rotating shield plug of Monju.

An effort is being made to develop an analysis code (FLUSH-Code) for heat transfer and cover gas flow behavior above pool surface reactor vessel.

5.3 Primary Pump

Good hydraulic performances of the primary pump for MONJU were already verified by in-water and in-sodium tests with a full-size prototype.

As the final stage of the pump tests, a test with low sodium level in the pump casing and non-seal gas tests for pump shaft were carried out on the sodium pump test loop. As a result of the tests, it was not found any trouble. Endurance test of the pump are being continued. Total time of the sodium operation is about 32,000 hours.

Cavitation performance tests of a new pump impellors were carried out by in-water test with small-size impella for DFBR pump.

5.4 Intermediate Heat Exchanger

In order to evaluate the heat transfer performance of the intermediate heat exchanger of LMFBRs at low flow condition, a heat transfer experiment was carried out using a double tubed counter current heat exchanger at low Peclet numbers and unbalanced flow rate conditions between the primary and secondary sides. In the experiment, temperature distributions along the test section were measured under the following conditions: the primary and secondary inlet temperatures were about 350°C and 300°C respectively and the flow rate of primary or secondary side was varied from about 1.0 to 70 l/min. These experimental results have been used for better understanding of the heat transfer characteristics of sodium heat exchanger under low flow conditions and will be used for the design of sodium heat exchanger which can be operated both at forced low flow rate and natural circulation. Another sodium heat transfer test at low Peclet numbers is in progress for a 19-pin bare rod bundle having P/D parameters of 1.2, 1.5 and 1.8 to obtain the low Peclet heat transfer correlation.

5.5 Control Rod Drive Mechanisms

Three kinds of control rod drive mechanisms for MONJU have been tested with full scale mock-up under simulated sodium conditions. They are drive mechanisms of fine control rod (FCRD), coarse control rod (CCRD) and back-up shut down rod (BCRD). In-sodium test on the final BCRD model was finished in December 1983, and a seismic test in water was carried out in Spring of 1985. The final models of FCRD and CCRD were manufactured in March 1985 and in-water and in-sodium tests were started in Feb. 1986. The final tests are expected to be carried out on sodium test loop in Aug. 1988.
A dynamic behavior and a fatigue tests on shaft seal bellows have been carried out to establish the design basis.

The self-actuated-shutdown system (SASS) for the demo-plant FBR was initiated in spring 1987. As a first stage, the performance test of magnet for high temperature was carried out in-air. In-sodium testing is expected to be started in spring 1988.

5.6 Refueling and Fuel Storage System

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

5.7 Ex-Vessel Storage Tank

A simple geometry model test was conducted as a part of a program to study natural circulation in EVST. Experimental data obtained were temperature, velocity, and flow pattern profiles in the test vessel. Applicability of COMMIX-PNC (DRACS Version) code, which is a revised version of the COMMIX-1A code, was examined for developing the basis for its use as a design and safety analysis tool. COMMIX-PNC results indicate relatively good agreement with the basic physical effects shown by the experimental data. Both analytical flow and temperature profiles, quantitatively match with experiments. Therefore, the COMMIX-PNC code is capable of predicting thermohydraulics of in-vessel natural circulation. Following the simple geometry test, a 1/3 scale, 1/6 sector model test is being continued to demonstrate the feasibility of the decay heat removal system by natural circulation.

5.8 In-Service Inspection Equipment

An effort is being made to develop in-service inspection equipments for reactor vessels, inlet pipes and PHTS (Primary Heat Transport System). A remote inspection technique using optical fiber scopes for reactor vessels is established, with r-ray irradiation tests at R.T to 250°C.

Another effort is being directed to develop electromagnetic acoustic and ultrasonic transducers for high temperature use on reactor vessels and primary piping systems.

Preparation of full size model tests for the Monju reactor vessel, steam generator tubes and primary pipes started from the spring 1987.

5.9 Expansion joint for piping system

Feasibility studies of expansion joints for piping system have been conducted since 1985. In-sodium tests of the expansion joints started from Feb. 1986. Those expansion joints are horizontal connection and vertical connection types. They are designed for a primary system of DFBR, and the size is 42 inches in diameter.

Endurance tests of the expansion joints were finished in 1987.

5.10 Steam Generator System

6.1 Sodium-Water Reaction Studies

6.1.1 Leak Propagation Tests

Seven micro-leak tests were conducted in JFY '85 in the SWAT-4 rig. The total numbers of micro-leak tests conducted thus far in SWAT-4 since 1984 are 38 and 24, for 2-1/4Cr-1Mo ferritic steel and JIS-SUS321 (321so) austenitic stainless steel, respectively.

In a small leak range, twelve tests were conducted for SUS321 steel in SWAT-2 test loop by varying test parameters such as water injection rate, sodium temperature, and nozzle-to-target distance, to obtain an empirical formula of the SUS321 wastage rate as a function of the parameters.

A new series of wastage tests started in a small leak range for high-chrome steel which is major back-up tube
material for future heat exchangers. Wastage tests carried out in SWAT-1 revealed that mod, 9Cr-1Mo steel had more wastage-resistance than 2-1/4Cr-1Mo steel.

SWAT-3 Run 19 was conducted to clarify the possibility of overheating failure of SG tubes. In the test, water was injected from an initial leak tube at a rate of 1.9 kg/sec.

6.1.2 Development of Acoustic Leak Detection System
A background noise level in an actual steam generator was measured using 50MW Steam Generator Test Facility. Acoustic signals from sodium-water reaction of SWAT-2 wastage tests were compared with the spectrum of the 50MW SG.

Basic tests have been carried out by use of SWAT-5; a water-filled half-scale model for Monju evaporator, to obtain technical know-how of leak detection and location. In series of the tests, an influence of internals, center pipe, and outer shroud on the intensity was examined.

6.1.3 Computer Code Development
A large leak sodium-water reaction analysis code, SWACS, is being applied for various types of steam generators. Feasibility studies are under way. In addition, a one-dimensional thermal-hydraulic computer code, SWAC13E, is under development. The code was validated by comparing calculational results with SWAT-3 large leak data.

6.2 High Chrome Steel- Steam Generator
Three modified 9Cr-1Mo Steel steam generators with a 1MW or 0.5 MW heat capacity were manufactured, and in-sodium tests continue to confirm the reliability of long time operation under high temperature and thermal transient.

6.3 Double Wall Type Steam Generator
Development of double wall type steam generator started in 1985. The efforts are being made to fabricate several types of double wall tubes. Performance tests of the tubes and tube sheets started at Oct. 1986.

Work for a conceptual design of the double wall type steam generator and leak detector system continues.

7. Sodium Technology

7.1 Material Tests in Sodium

7.1.1 Core Material Tests in Sodium
For fuel cladding and duct materials of demonstration FBRs, sodium environmental tests of modified austenitic stainless steel is being performed. The following are tests' items:

* Tensile test after exposure to high temperature sodium and thermal ageing.
* Internal pressure creep rupture test
* Corrosion and mass transfer test

In addition to these tests, sodium environmental tests on possible candidate core materials for future FBRs have been commenced according to a program of core material development. The candidate core materials are high-chromium ferritic steels and oxide dispersion strengthened ferritic steels.

7.1.2 Tribology Tests in Sodium
Development and evaluation of cobalt-free hard facing materials have been performed to clarify the tribological behaviors of contacting and/or sliding parts of FBR components. The tribology test in sodium includes;

* Self-welding test
* Friction and wear test
* Sodium Compatibility test

A new test program started to evaluate the wear behaviors of high-chromium ferritic steel tubes.
7.2 Flow and Heat Transfer

The third heat transfer test at low Peclet numbers is in progress for 19-rod bundle having a P/D parameter of 1.8. The heat transfer correlation will be obtained from three sets of heat transfer data for P/D = 1.2, 1.5, and 1.8 in the final evaluation.

Measurements of pressure drop for the 19-pin wire-wrapped bundle have been conducted under isothermal and mixed convection conditions for upward and downward sodium flows, focusing on friction factor correlations. A similar test for 37-pin bundle is in progress to study bundle size effects.

Measurements of sodium mist content in argon cover gas space at relatively low temperature conditions have been completed.

A water test facility was developed to study natural circulation thermo-hydraulics in the Monju EVST. A water test for the 1/10-scale EVST model with a simple geometry have been completed. Following the above tests, 1/3-scale model tests with 1/6-sector are in progress to demonstrate the feasibility of natural circulation at the ex-vessel decay heat removal system.

7.3 Radioactive Material Behaviour and Control in Sodium

This technology area covers studies to decrease the radiation exposure to plant personnel due to radioactive corrosion and fission products (CP & FP) produced in LMFBR primary circuits, and to improve maintainability.

The objectives of the studies are as follows:

* To establish computer code for CP behaviour analysis.
* To develop CP and FP trapping methods (radioactive nuclide traps) in sodium.

(a) Development of computer code for CP behavior analysis

An analytical model for CP behaviour in the primary system of LMFBR's has been developed by using results obtained from out-of-pile studies which consisted of CP transfer experiments and examinations of metallurgical effects of sodium exposure on stainless steel test specimens, plant experience in JOYO and published data by some foreign laboratories.

A computer code "PSYCHE" has been developed to evaluate CP transfer in primary circuits of LMFBR's and radiation dose rate by CP around primary cooling piping and components. The analytical parameters in this model were obtained by using data of the out-of-pile experiments and measurements in JOYO.

The CP deposit radioactivity and dose rate distribution in the primary sodium system of JOYO was measured and compared with calculated results by PSYCHE. The radioactivity of CP calculated by PSYCHE agreeded with the measured within a factor of 0.7 - 1.7 and within a factor of 0.7 - 1.3 for the dose rate distributions.

(b) Development of CP trapping method in sodium

CP trapping materials have been developed to collect dissolved species of CP in sodium. It was found that nickel was the most effective material to trap $^{54}$Mn and $^{51}$Cr, and less effective for $^{58}$Co and $^{60}$Co. Highly effective trapping materials made of nickel have been developed and characterization tests of the materials in sodium have been carried out.

Cesium trapping tests in sodium by cold traps and RVC materials, included improvements of the trapping materials, has been carried out. RVC trap had been installed in the purification circuit of the primary system of JOYO.
7.4 Sodium Chemistry and Sodium Purification

LMFBR coolant can be by impurities like oxygen, hydrogen and carbon contaminated during operation. These impurities influence on the structural and core materials, so FBR plants must be operated under the control of these impurities. The following studies have been performed.

* To develop devices and techniques to remove impurities and keep these concentrations at a desirable level and to develop in-sodium chemical meters and techniques to monitor concentrations of impurities.
* To understand the behaviour and influence of impurities and to consider how to cope with the situation.
* To develop the regeneration methods of cold trap.

Impurity levels in sodium can be measured by the values of the impurities of metallic specimens exposed to the sodium. This method was applied for oxygen and carbon measurements. Satisfactory results for oxygen were obtained by using vanadium wire.

Tests for carbon have been carried out by using SUS304L and Fe-12Mn metallic foils.

7.5 Sodium Removal and Decontamination

Since 1982, decontamination studies for CP have been done at O-arai Engineering Center.

A series of tests has been carried out to develop the decontamination process of CP by using samples contaminated by CP. The deposition characteristics of CP in the sodium cleaning equipment of fuel assembly in Joyo and the chemical decontamination processes have been studied.

8. FBR Instrumentation

8.1 Nuclear Instrumentation

8.1.1 In-Core Fission chamber

A micro fission chamber to provide for the instrumented assembly (INTA) has been tested at JOYO since December 1985.

8.1.2 Ex-vessel BF Proportional Counter

Experiments of ex-vessel BF proportional counter for Monju have been completed in the Japan Research Reactor-4. The experimental results satisfied design specifications under the following test conditions.

Accumulated irradiation dose:
- Thermal neutron $1.6 \times 10^7$ n/cm$^2$
- Gamma ray $3.3 \times 10^3$ R

Temperature conditions
- During irradiation 30°C
- Post irradiation 160°C

(Time duration in the post irradiation experiment 30 days.)

8.2 Failed Fuel Detection and Location

A demonstration test of the tagging gas system for locating failed fuel was carried out in JOYO, as described in Chap.2 of this report.

For Monju, experiments of the cryogenic adsorption system of the tagging gas system is being carried out to determine the design specification, by changing design parameters.
8.3 Early Warning System for Fuel Failure

8.3.1 Temperature Measurement
Temperature fluctuation due to flow blockage in the pin bundle has been analysed by computer code developed. Calibration tests of the above-core instrumentation for Monju are planned to start in Jan. 1989. Fast response thermocouples have been developed for application to the above-core instrumentation.

8.3.2 Flow Measurement
New type eddy-current flow/temperature sensors were developed and tested in a sodium loop.

8.3.3 Other Systems
An acoustic detection system is being developed for purpose of detecting anomalous sound, in particular the onset of local boiling in the core.

8.4 Process Instrumentation

8.4.1 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.

A testing of on-site calibration technique using a cross-correlation technique of EM Flowmeter noise signals was carried out.

A equipment for the flowmeter calibration is being installed. The calibration tests for Monju will be started in spring 1988.

8.4.2 Sodium Level Meters
Calibration tests of the sodium level meters were conducted for Joyo reactor vessel and 50MW Steam Generator Test Facility using a sodium level meter calibration test loop. A calibration test of the sodium level meters for Monju will be started in Jan. 1989.

8.5 Sodium Leak Detection System
Sodium ionization detector and aerosol trapping filter were tested in the leak rate of order of 100g/hr with the simulated environment of Monju's primary cell. Their capability as the leak detector was also demonstrated with use of a piping model simulating Monju's secondary system.

Consequently, they have been adopted as the leak detectors of Monju's primary and secondary piping systems.

9. Fuels and Materials

9.1 Fuel Fabrication
The PFPF (Plutonium Fuel Production Facility) equipped with automated and remote handling fuel production systems has began to fabricate "JOYO" and "MONJU" fuels from last August.

9.2 Fuel Pin Performance
Fuel pin performance codes for transient state and fuel failure state have been improved with the data of operational reliability tests in EBR-II, etc. The modeling of Cesium migration was developed to evaluate the fuel performance of an axial heterogeneous core fuel.

Feasibility studies of the fuel performance code for metal, carbide and nitride fuel has has been commenced.

9.3 Core Materials
PNC 316 stainless steel (MONJU core material) irradiated until 2.07x10^{23} n/cm^2 E 70.1 MeV showed excellent swelling resistance by less than 1.5 swelling.
Out-of-reactor mechanical property and sodium corrosion tests of advanced austenitic stainless steels have been completed. Irradiation tests for the candidate steels have been conducted in JOYO and FFTF.

Two types of ferritic steel have been developed. One is high strength ferritic / martensitic steel which is considered fit to wrapper tube and another is oxide dispersion strengthened ferritic steel (ODS). The tubing technology for ODS cladding has progressed by hot working.

9.4 Subassembly

Out-of-reactor testing of bundle-to-duct interaction for large assembly has been carried out. Computer tomography techniques are applied for the investigation of bundle distortion.

The study of neutronics and thermal hydraulics for ductless subassemblies has been initiated.

9.5 Irradiation Experiments

1) JOYO

Monju fuel irradiations for high burnup core have been conducted. Irradiations of axial heterogeneous core fuel and PNC advanced austenitic stainless steel fuel have also been conducted. Fuel bundle test using CEA cladding tabs started from last August.

2) Foreign Reactors

Phase-I program of operational reliability testing in EBR-II is almost completed and Phase-II program is initiated.

Fuel subassembly tests using PNC 3/6 and advanced austenitic stainless steel cladding have been irradiated in FFTF from November, 1987.

9.6 Development of advanced fuels

Mixed carbide fuel pins have been irradiated from 1983 using the thermal reactor JRR-2, JMTR of JAERI.

Preparatory work for the development of advanced fuel (metal, carbide and nitride fuel) performance code and the irradiation test in "JOYO" open core has been commenced.

9.7 Post Irradiation Examination

Detail design and development of in-cell apparatus for large-scale PIE facility are performed to begin the examination of MONJU fuel subassembly and so on from 1995.

10. Structural Design and Materials

10.1 Development of structural Design Methods

10.1.1 Structural Analysis Methods

a. FINAS Nonlinear structural analysis program

The enhancement of the general purpose nonlinear structural analysis program FINAS has been made since 1981, particularly with respect to inelastic constitutive models of cyclic plasticity and viscoplasticity, large deformation / buckling analysis methods, shell elements, automatic computation algorithms, fracture mechanics capabilities, dynamic analysis capabilities including fluid-structure interaction and graphics options. FINAS is currently utilized by many research engineers and designers of PNC, fabricators and universities.

b. Simplified analysis method of tube-sheet-shell structures

Simplified analysis procedures combining axisymmetric and plate models have been developed for tube-sheet-shell structures.

Methods of predicting local stresses and strains in the central region of ligament as well as rim-ligament region are investigated for thermal transient loads.

10.1.2 Structural Design Guide

a. Improvement of Elevated Temperature Structural Design Guide

The following items have been discussed to improve and extend the current Elevated Temperature Structural Design Guide.

A tentative Elevated Temperature Structure Design Guide for piping bellows has been developed. It provides rules for stress limit, strain limit, creep-fatigue evaluation based on elastic analysis and buckling criteria for internal pressure. These rules have been developed based on analytical and experimental research works by PNC.

10.2 Structural Tests and Evaluations

In order to develop strength prediction methods, to evaluate the adequacy of elevated temperature design rules and also to confirm the integrity of the actual components, the following structural tests and development of associated evolution methods had been or are being performed.

a) Elevated temperature test of piping bellows in air

Creep fatigue tests, buckling tests, fracture mechanics tests, mechanical ratchetting tests and vibration tests were completed.

b) Thermal transient tests of SG tube-sheet model in air

Tests were completed.

c) Thermal creep-fatigue test with small sodium loops (SPTT and STST)

Straight pipe models made of SUS304 including weldments dissimilar joint models made SUS304 and 2-1/4Cr-1Mo steel had been tested. Structural discontinuity models are being tested for understanding of crack initiation and propagation.

d) Thermal transient tests in large sodium loop (TTS)

Two vessel model tests, piping bellows junction models tests and the thermal stress mitigation model (I) test are completed. The thermal stress mitigation model (II) test is underway.

10.3 Structural Material Tests

Structural materials tests in air, sodium and irradiation environment have been conducted to revise the "MONJU" material strength standard and prepare a new-version for demonstration FBR.

The new tests program in air and in sodium environment is called "Capella" program, and it is currently being performed. The Step-1 program (1985-1987) of the "Capella" was already completed. And the Step-2 program (1988-1990) was started in 1988.

The neutron irradiation tests are being conducted according to a neutron irradiation program.

10.3.1 Structural Material Tests in Air

In-air structural material tests for the development of the "MONJU" strength standard had been conducted since 1977 as mentioned below.

From 1977 through 1978, basic mechanical properties on typical candidate materials for "MONJU" components were tested to compare the design allowable stresses of ASME Code Case N-47 including welded metals and joints. The following tests were conducted.
* tensile test
* creep test
* relaxation test
* fatigue test
* creep-fatigue test

From 1979 through 1981, the above tests continued to prepare the material strength standard for the "MONJU" structural design guide, and some theoretical methods on material strength behavior were investigated by the following tests,
* Creep damage estimation test with strain-hold
* inelastic strain behavior test
* evaluation test on strength of welded-joint
* notch effect test
* others

The test results were evaluated to verify the validity of the descriptions of Case N-47, and were reflected to the development of the design guide.

From 1982 through 1985, the tests were carried out to refine the "MONJU" design guide and material strength standard.

From 1985, the Capella program mentioned above continues up to now.

The Capella program has two principal purposes. One is to clarify the application limits of the present evaluation methods and to improve the accuracy of these methods. The other is to select new proper structural materials for FBRs.

The improvement of the accuracy of the evaluations and the application of new proper materials will contribute to the cost reduction for the construction of demonstration FBR.

For these purposes, the Capella Step 1 (1985-1987) program involves developing the following technologies;
* Improvement of "MONJU" technology for cost-down
  * (creep-fatigue life evaluation, strength of welding, inelastic constitutive equation, and others)
* Design and fabrication of large-scale structures
* Modification of material specification (Application of advanced SUS304 and 316 Stainless Steels)
* Application of elevated temperature fracture mechanics
* Development of material strength standard for high Cr-Mo steel

The above technologies continue to the Capella Step 2 (1988-1990) program, and a new version of material strength standard including 9Cr-Mo steel and life evaluation method will be prepared tentatively in 1989.

10.3.2 Structural Material Tests in Sodium and Water

a. In-sodium Tests

Sodium environmental tests of the structural material for "MONJU" have been completed on Inconel 718 alloy which is used for the thermal stripping resistance of the upper-core structures. Tests on the material for the bellows joint and on the carbon transfer behavior in simulation loop of the secondary sodium circuits of "MONJU" were also completed.

Corrosion tests of MONJU steam generators were completed in high temperature sodium contaminated, with NaOH or NaO, supposing water leak from steam generators.

A new series of sodium environmental effect tests, according to the Capella Step 1 (1985-1987) program, were carried out on possible candidate alloys for a future demonstration FBR. The candidate alloys are high Cr-Mo steels, and advanced type (low carbon and/or high nitrogen) SUS304 and 316 stainless steels. Testing items are corrosion and mass transfer, carbon transfer, and mechanical strength (tensile, creep, fatigue, creep fatigue) tests in sodium.

These tests continue to the Capella Step 2 (1988-1990) program,

b. In-water (steam) Tests

It was confirmed that austenitic stainless steels such as SUS304, 316 and 321 were unsusceptible to S.C.C. in wet steam containing oxygen and chloride ion up to 200 ppb. As the next step, corrosion tests were completed to confirm the integrity against S.C.C. for the plugs of steam generator tubings (after plugging) under the same wet steam conditions. The corrosion tests on possible candidate alloys (high Cr-Mo steels) for a demonstration FBR under the same wet steam conditions continue to the Capella Step 2 (1988-1990) program.
10.3.3 Structural Material Tests in Irradiation Environments

a. Surveillance Tests for "JOYO" and "MONJU"

Surveillance tests for the primary components of the experimental fast breeder reactor "JOYO" have been conducted to confirm the integrity of the reactor by evaluating irradiation effects of the materials. Surveillance test data are used for the planning of "JOYO" operating program.

The second surveillance tests were completed on the materials of the reactor and safety vessels of "JOYO".

b. Research and Development Tests

Research and development tests have been conducted on the structural materials for the primary components of "MONJU" to evaluate irradiation effects on the mechanical properties up to the end of design life and to evaluate irradiation effects on the materials strength standards for "MONJU".

Both forged and rolled SUS304 steels, Inconel 718 were irradiated in "JOYO" by SMIR (Structural Materials Irradiation Rig).

Another test for a demonstration FBR has been conducted to make clear the relationship between creep rupture strength and metallurgical variables such as chemical compositions, grain size, production process etc.

c. Miscellaneous

A new facility (Material Monitoring Facility 2: MMF-2) started tests of irradiation materials from April, 1984. This facility is equipped with five uniaxial creep test machines, two creep fatigue machines and others for irradiated structural material tests.

10.3.4 Data Banking System

Material test data are compiled by specified data coding sheets, and the data inputs to the computer system for data banking, SMAT are continued.

The SMAT system was developed to meet the requirements of efficient processing of large amounts of data.

The SMAT system currently has more than 10,000 data points on 11 different kinds of mechanical tests (tensile, low cycle fatigue, creep etc.) for 10 kinds of FBR structural steels, including type 304 stainless steel, Mod. 9Cr-Mo steel, etc.

11. Safety

11.1 Thermohydraulics Related to Reactor Systems and Design

With regard to thermohydraulics related to reactor systems and design, experimental studies, code development and validation are extensively underway for demonstration FBRs.

Experimental studies are carried on decay heat removal by natural circulation, thermal striping, and heat transfer characteristics of an in-line rod array in natural convection. As for the decay heat removal experiment, the first phase water tests using a 1/6-scale reactor model of a 1000 MWe loop-type plant were completed in March 1987. The second phase water tests were started in September 1987 for a pool-type reactor in collaboration with Japan Atomic Power Co. A water test for thermal striping is continued to obtain basic mixing data on multiple turbulent jets. The data are needed to develop a rational design method for a structure placed in a fluctuating temperature field.
Furthermore, the sodium test for the in-line rod array is also in progress to determine the natural convection heat transfer characteristics of a coil-type heat exchanger immersed in a reactor plenum.

In addition to the experimental studies, code development is continued for the single-phase multi-dimensional thermohydraulics code AQUA (Advanced Quadratic Upstream Algorithm). Implementation of the fuzzy controller, which optimized time step sizes automatically based on an artificial intelligence technique, is one of the highlighted topics. Further efforts on the improvement and validation of the code were devoted to three-dimensional analyses of thermal stratification experiments conducted at OEC.

11.2 Thermohydraulics Related to Reactor Safety

This section describes thermohydraulic studies for evaluating the physical phenomena taking place in the early stage of postulated accidents such as LOPI (Loss of piping integrity), ULOF (Unprotected loss-of-flow), UTOP (Unprotected transient overpower) and LOHS (Loss-of-heat sink).

Experimental studies are conducted on fuel subassembly thermohydraulics and plenum-channel thermohydraulic interaction. As to the fuel subassembly thermohydraulics a comprehensive evaluation is now in progress for sodium mixed convection pressure drop characteristics and intra-subassembly flow redistribution characteristics. The importance of this evaluation centers on the effects of subassembly size and geometry, and on the similarity laws in the mixed convection behavior between water and sodium experiments. A water test was also continued to study the pressure drop and coolant mixing characteristics of a PNC-designed new pin bundle, whose average fuel burnup is expected to be 1.5 times higher than that of Monju.

11.3 Degraded Core Study

The objectives of this study are to prevent propagation of fuel pin failures and to mitigate consequences of whole core accidents. The major efforts in the reporting period were devoted to; (1) in-pile test analysis for SCARABEE and
CABRI ; (2) PNC out-of-pile tests on material relocation/interaction behaviors ; (3) Monju reactor analyses in the PSA activity ; and (4) code development, modification and validation studies.

The SCARABEE analysis concentrated on the post-test analysis of the first PNC test Bb+3, and the pre-test analysis of the second test (Pf-A) which treats subassembly failure and propagation to interstitial gaps. Regarding the CABRI activity, the synthesis effort on CABRI-1 results was continued, and the planning of the second stage, CABRI-2, was initiated. The CABRI-2 ad-hoc meeting was held among the partners on August 24-26, 1987 at OEC for defining program objectives and formulating base test matrix.

In the area of the PNC out-of-pile activity, VECTORS (Vapor Expansion and Condensation Tests in Out-of-Reactor Simulation) accomplished its objectives and the results are being analyzed. The JET-I tests were continued for studying molten jets and their erosive behavior in a low temperature range (500°C). In high temperature series, the MELT-I tests attained their primary goal and the facility was decommissioned; an advanced large-scale test facility, MELT-II, was constructed and system testing was completed successfully.

Reactor safety studies were continued on local faults and whole core accident analysis in the Monju PSA. Code development, modification and validation efforts were continued in connection with the above-mentioned activities.

11.4 Plant Accident Study

The main items of plant accident study are sodium fire, aerosol behavior, source term and ex-vessel accident evaluation.

In the field of sodium fire and aerosol behavior, the rates of combustion and associated sodium aerosol release in a pool fire have been determined in a low oxygen atmosphere. In addition, the development of ceramic liner was directed to the improvement of castable materials. In regard to the computer codes development, validation study of ASSCOPS was carried out using the test results from the SAPPFIRE facility. Evaluation of sodium leak accident in a primary loop in a demonstration plant was continued to extend the applicability of the CONTAIN code. The development of SOLFAS; a multi-dimensional code to analyze natural convection heat transfer during sodium fires, were continued and the turbulent model was implemented. The chemically reactive sodium combustion products may harm the respiratory organ of human. Thus, development of code to analyze aerosol behavior in the air flow paths during respiration was carried out together with performance test of respirators.

In the area of source term research, activities are concentrated on the investigation of fission products behavior related to the transport from sodium pool to gas phase. The test to determine the gas-liquid partition ratio of fission products was continued. The construction of a test rig to investigate fission products releases during a sodium-concrete reaction was completed. The test rig is used from JFY88. On the other hand, a new test was started to determine the solubility of iodine in sodium at temperatures near the reactor operating conditions.

Current activities of the ex-vessel accident study are focused on the evaluations of the Monju containment response following HCDAs. The CONTAIN code for this purpose was upgraded, and the calculations of protected loss-of-heat-sink (FLOHS) and unprotected loss-of-flow (ULOF) have been continued. From the end of November to the beginning of December, 1987, the first specialists' meeting of CONYSIN-FBR (LMR) was held at OEC. The participants were from Sandia National Laboratories (USA), GRS (FRG), KEK (FRG), and PNC (Japan), and an observer from UKAEA.
11.5 Steam Generator Safety Study

Activities related to steam generator safety during JFY87 cover the leak development study, the large leak study, and the study of the duplex-tube steam generators.

In the leak development study, tests of micro leak (self-wastage) and small leak (target wastage) for the high-chrome ferritic steels have been carried out. These steels are most promising materials as heat transfer tubes of steam generators in a demonstration plant. In parallel to the above mentioned tests, intensive studies are conducted in other areas: the study of structural integrity and the construction of materials data base.

In the area of the large leak study, the construction of a sodium-water reaction simulation test rig PEPT, and the modification of the SWACS code have been carried out. PEPT consists of a steam generator, an IHX, an expansion tank and a piping system, all of which are one-fifth scale of the secondary system of one of the demonstration plant designs. An objective of the code modification is to extend its applicability to the steam generator having no cover gas.

The study of the duplex-tube steam generations is to evaluate the feasibility of the FBR plant system which utilizes the primary steam generators. During JFY87, classification of the causes of heat transfer tube failures was conducted as well as construction and evaluation of the event tree for the tube failures. This study was conducted as one of the joint R&D program between PNC and JAPCO.

11.6 Research on Probabilistic Safety Assessment

As part of the research and development on the prototype reactor Monju, Probabilistic Safety Assessment (PSA) has been performed since 1982.

The object of this study is to construct a probabilistic model for the Monju plant so that overall safety analysis can be performed. It is expected that (1) an overall evaluation based on the quantitative analysis is performed; (2) the methods to improve the system reliability are determined; (3) the operation and maintenance procedures are established on a technical basis; and (4) the reliability, safety and rationality of the plant are improved.

The systems analysis entered the second stage in April 1987. The event trees (E/Ts) and fault trees (F/Ts), which had been the reference, were reviewed. In order to reflect the changes in design and operational procedures made since 1984, new F/Ts were developed based on the latest information. Some of the changes were made taking into account insights provided as a result of the previous systems analysis. Using the new F/T models, accident sequences leading to core damage were identified and quantified. Also, uncertainty and importance analyses including some sensitivity studies were performed. Failure data concerning sodium components leakages were obtained from the CREDO (Centralized Reliability Data Organization) database and utilized in the quantification of accident sequences. The result indicates that the core damage frequency becomes lower by one order of magnitude than of the previous analysis in the first stage. Also, probabilistic techniques based on the above models have been applied to the examination of Technical Specification. In addition to evaluation of internal events including location-dependent failures such as internal fire, assessment of seismic risk was started this year. This effort is made on a supplementary basis to estimate the effect of seismic events on the total risk.

A code network system including the SETS code has been developed to perform systems analysis. The whole system was first utilized in the second stage of systems analysis mentioned above. It was found that the use of this system can significantly reduce time and manpower for the task of level-1 PSA. More efforts have been made to improve that supporting program modules in the code network system.
Data development effort for the LMFBR components is being made. CREDO, the cooperative project between PNC and USDOE, is core of this effort. PNC continued the collection of operation and maintenance data from the various facilities at OEC including the experimental fast reactor, Joyo. Together with the data from the US facilities, the population of the data accumulated in CREDO has been growing steadily. The full application of the CREDO data to the Monju PSA was started this year. Effort is being made further to collect data on more component types and human performance. As for the data base management system (DBMS), two programs were developed, namely record verification program RECOVER and unit conversion program UNICON.

To perform value-impact analysis (VIA) with a view to examining the rationalization of safety design policy, the cost evaluation model and the cost data base SEDB (Safety Evaluation Data Base) have been developed since 1985. In addition to improving the model and data base, their application to a VIA problem was attempted this year. The results indicate the usefulness of these tools.

The plant dynamic responses in the protected accidents, PLOHS (Protected Loss-of-Heat Sink) and LORL (Loss-of-Reactor Level) have been analyzed to assess: (a) success path or potential success path that has originally been counted as a core damage path in the previous analyses; and (b) failure path to the CDA accident phase. As a result, it is found that the recovery action is most effective to reduce the total core damage frequency.

An evaluation of the in-vessel physical processes, radionuclide transport and the risk analysis was performed in view of level-3 PSA. Phase-2 of this task was started in April 1987.

Main efforts in the analyses of UTOF (Unprotected Loss-of-Flow) in-vessel physical processes were devoted to refinements of the investigation of key phenomena in the in-vessel physical processes and to integration of the in-vessel sequence analyses.

Based on the most up-to-date information on the Monju design and the previous delineation study, several representative sequences of the UTOP (Unprotected Transient Overpower) accident due to control rod withdrawal have been investigated with supports of scoping code calculations. The ex-vessel event sequences for two types of accidents, ULOF and PLOHS have been analyzed. In the ULOF analyses, the sensitivities of the physical parameters on hydrogen build-up and sodium spray in the containment atmosphere were examined carefully to assess the event sequences more realistically. In the PLOHS analyses, three preliminary calculations were performed based on the failure modes of the primary system.

Delineation study was performed for the assessment of the propagation of local fault.

12. Fuel Cycle

12.1 Policy for Nuclear Fuel Cycle

The Atomic Energy Commission (AEC) released a new policy for long-term nuclear energy development in June, 1987. The policy specifies as follows:

First, the utilization of uranium and plutonium recovered from reprocessing spent fuels shall be promoted. This will greatly contribute to save uranium resources and reduce Japan's dependence of foreign energy resources.

Secondly, plutonium utilization in LWRs and ATRs shall be promoted. It will facilitate to build a bases for plutonium utilization in FBRs which are expected to become leading nuclear power reactors in place of LWRs in the future.

Thirdly, safe and appropriate radioactive wastes treatment and disposal shall be implemented. Radioactive wastes will be treated and disposed of in suitable methods considering the nature of nuclides.

The once-through system used in LWRs is far superior to that using fossil fuels because of the stability of supply.
However, as Japan has to import all its resources, there is no fundamental difference between the two systems. On the other hand, the reprocessing-recycle system does differ in that it drastically enhances the efficient use of the uranium recovered which, according to the non-proliferation concept, requires careful monitoring. Plutonium and uranium do exist in Japan and can be recovered from these spent fuels as pseudo-domestic energy resources. Figuratively speaking, spent fuels are the 'mines' of the energy resources in Japan; reprocessing should be regarded as the means of exploiting such 'mines'. Regular use of plutonium in the FBR could greatly lessen Japan's dependence on overseas natural uranium. This may, however, take a considerable period to achieve.

Use of uranium in LWRs is expected to continue for a longer period than was formerly thought. Therefore, the long range stability of natural and enriched uranium prices will become increasingly important. As utilization of plutonium increases, the pressure of demand for natural and enriched uranium will decrease to a certain degree. This may result in the need for substitutes, and consequently contribute to stability in price.

From the standpoint outlined above, Japan should seek to establish a plutonium utilization system which is superior to that of uranium in terms of both economy and safety. The reprocessing-recycle system should become the major approach and all developments should be pointed in this direction.

12.2 MOX Fuel Fabrication

Studies and development of fabrication of uranium-plutonium mixed oxide (MOX) fuel have been carried out since 1965 at the Plutonium Fuel Development Facility (PFDF). The Plutonium Fuel Fabrication Facility (PFFF), which started operation in 1972, has two fuel fabrication lines for ATR and FBR. It supplies the fuel necessary for the operations of Fugen and Joyo. About 100 tonnes of MOX fuel will be fabricated by the end of February 1989. Of this, 1,160kg Pu are supplied from the Tokai Reprocessing Plant.

In parallel with the construction of Monju, construction of the Plutonium Fuel Production Facility (PFPF) (FBR line) (production capacity: 5t MOX/yr) started from July 1982. It was designed to develop fuel fabrication technologies as well as to fabricate fuels for Monju. The construction was completed in October 1987. The PFPF is currently fabricating the fuels for Joyo. The PFPF will start fabricating fuels for Monju in the middle of 1989.

This facility has the initial production capacity of 5t MOX/yr, but it is so designed as to increase the production capacity to 15 MOX/yr by increasing the process equipments, considering the fabrication of fuel for a demonstration FBR.

Remote control and automatic operation techniques, which are indispensable for MOX fuel fabrication facilities, are being developed in PFFF. Although it has a direct maintenance system, personnel do not normally have approach to nuclear materials. This has been achieved through various experiences carried out at PFFF.

As mentioned above, technologies related to MOX fuel fabrication have been basically established in Japan. However, to establish plutonium utilization in the future, a significant amount of technology needs to be developed for commercializing MOX fuel fabrication including safe handling of plutonium. In particular, MOX fuel for FBR use has the following specific features: it has a high plutonium content, its burnup is higher and the diameter of the pellets is smaller. Thus, there is a need to continue developing fundamental techniques. For this reason, all work concerned with pellet fabrication and process equipment that aims at automation, high throughput and remote control will be continued at PNC.
12. Plutonium and Uranium Conversion

To use plutonium and recovered uranium obtained through reprocessing of spent fuel, development of conversion technology is essential, together with reprocessing and MOX fuel fabrication technologies. PNC is the first company in the world to develop a co-conversion technology using microwave heating denitration process (MH method), which converts plutonium nitrate and uranyl nitrate solution to MOX. This method was developed from negotiations that took place between Japan and the United States of America on the operation of the Tokai Plant. Compared with the conventional method, it is a simple process and generates less liquid waste.

The Plutonium Conversion Development Facility (PCDF) (conversion capacity: 10kg MOX/d), designed for demonstrating the co-conversion technology by the MH method, was completed in February 1983. By the end of February 1989 it had produced about 4,700kg of MOX powder. About 4,200kg from the converted MOX powder were transported to PFPP and PFPF and are being used for fabrication of MOX fuel for Fugen, Joyo and Monju.

Since uranium recovered through reprocessing of spent fuel has generally a higher U235 concentration as compared with natural uranium, our country has decided to use it as LWR fuel by reenriching it and on mixing it with other enriched uranium and use it by mixing with plutonium as fuels for ATR, etc. In preparation for a large scale recovered uranium conversion facility, various technical development and design studies are now under way to establish the continuous mass production technique by the MH method.

13. FBR Fuel Recycling

In the area of FBR fuel reprocessing, PNC has developed the process and equipment with remote handling techniques, using large-scale cold mock-up tests and laboratory-scale hot tests on the basis of the experiences accumulated in the Tokai plant. PNC is planning to conduct engineering-scale equipment tests under hot conditions in order to enhance the technical level and economical efficiency.

13.1 Process Research and Development

(1) Research and Development Facilities in Tokai

In the Chemical Processing Facility (CPF) about 6.5 kg of irradiated fuel mainly from "Joyo" by 1986 was used for hot laboratory tests with burn-up between 4,400 MWD/T and 55,300 MWD/T. The fuel irradiated at Phenix in France up to about 95,000 MWD/T has been treated since October 1986. In the Engineering Demonstration Facilities, EDF-I, II, and III, many
kinds of engineering tests of process equipment have been carried out. Present status of R&D items is described below.

(2) Head-End Process

PNC has adopted a laser-beam disassembling machine. The prototype machine could effectively remove hardware of FBR fuel assembly from the fuel pins. Tests for improving remote maintainability, verifying aerosol produced by laser, and making it more compact are now performed.

PNC began to study on continuous dissolvers, considering larger capacity and higher economical efficiency which would be required in future. The development of centrifugal clarifier in EDF-II and hot fundamental dissolution tests in CPF are continued.

(3) Chemical Separation Process

A engineering scale pulsed column test apparatus using plutonium has been fabricated. This is equipped with electroreductive pulsed column. PNC has also fabricated a test apparatus of centrifugal contactor and started its performance test.

(4) Common Technology

The development of remote system technology is now being promoted to establish remote maintenance concept with rack module system for the pilot plant and the new hot test facility. A new prototype of two-armed, elbow-down type servomanipulator has been fabricated and is now being tested in EDF-III. Other remote system, such as standardized racks, remote connectors, remote sampling systems, optical fiber signal transmission systems, are now under development.

Materials of process equipment and manufacturing techniques have also been studied. In CPF, hot corrosion tests are continued. In-line analytical systems for plutonium, uranium, acid concentration, and gamma-nuclide are being developed aiming at prompt analysis, automated operation, high reliability, and reduction of waste and exposure dose.

13.2 Plant Design

(1) Recycle Equipment Test Facility

Results of recent research and development proved that highly available and economical recycling technology is essential to realize FBR fuel cycle. It needs adoption of advanced process concept and hot engineering demonstration of advanced process unit components is necessary for steady operation of the pilot plant. From this viewpoint, PNC has carried out the conceptual design of Recycle Equipment Test Facility (RETF), which will be equipped with plant-scale test components and equipment for the purpose of plant-scale demonstration with irradiated FBR fuel and collection of hot data.

In RETF, test will be performed independently for each process. Therefore, each process capacity does not need to be consistent through all processes. Remote technology with each module system will be adopted so that the test components are easily interexchangeable.

After demonstration of process components, these technologies will be applied to the pilot plant to realize high availability and economical efficiency as well as to verify cost advantage of FBR fuel recycling.
The purpose of the FBR Fuel Recycling Pilot Plant is to demonstrate the economical efficiency of FBR fuel reprocessing for recycling spent fuel from FBR such as "Monju". It is scheduled to begin operation soon after 2000.

13.3. Waste Management
The Japan Atomic Energy Commision (JAEC) initiated a high-level radioactive waste management program in 1976. Since then, the Advisory Committee on Radioactive Waste Management of JAEC made recommendations on the program in 1980 and 1984. This program provides outlines of on the radioactive waste management in Japan and covers the schemes for the development of the treatment, storage and disposal technology of radioactive waste.

The major features on the treatment of HLLW are as follows:

1. In the development of the HLLW solidification technology, emphasis should be put on the vitrification into borosilicate glass;
2. PNC and JAERI should lead the researches on the development of vitrification technology and the safety evaluation of vitrified materials.
3. Vitrification technology should be demonstrated by the early 1990's through the construction and operation of the vitrification plant.

In accordance with this program, PNC has continued the development of the vitrification technology based on the liquid-fed joule-heated ceramic melter (LFCM) process.

The geologic disposal program consists of the following four stages:

1st stage: Selection of potential geological formations (completed)
2nd stage: Selection of the candidate disposal site(s)
3rd stage: Demonstration of the disposal technology at the candidate site(s)
4th stage: Construction, operation and closure of disposal facilities

The principal goal of the program is to initiate operation of a vitrification plant in 1992 for demonstration of the high level liquid waste solidification technology, and to initiate the fourth phase of HLW disposal program, that is, actual disposal of the solidified waste into geological formations.

13.3.1 Vitrification of High Level Liquid Waste
The technology development for vitrification has been continuing since 1975 in combination of cold engineering tests, full-scale mock-up tests and hot laboratory tests. The vitrification process comprised of LFCM has been demonstrated in its performance and reliability at Engineering Test Facility, (ETF) and remote handling/maintenance techniques together with system reliability at Mock-up Test Facility. (MTF)

Vitrification test using HLLW from the Tokai Reprocessing Plant started in December 1982 at CPF where about one litter of waste glass was produced per batch. The gamma-scanning of the canister was performed to recognize the uniform distribution of some nuclides in the waste glass. Characterization of the waste glass properties is now under way.

These tests focussed on providing detailed data for designing the vitrification plant for the Tokai Reprocessing Plant.

Major development items in process technology in recent years are improvement of both the design of a ceramic melter and the performance of melter off-gass cleanup system. A new engineering-scale ceramic melter with a 915 MHz microwave heating device has been in operation since September 1985.
Melter dismantling technique, numerical flow analysis code and others are also being developed as a part of melter technology. In the off-gas technology, airfilm cooler at the outlet of the melter off-gas, high efficiency mist eliminator, scrubber, NOx absorber and adsorber are developed and their performances are checked.

The safety license of Tokai Vitrification Facility (TVF) was permitted on 9th of February, 1988. And the TVF construction was started in June 1988.

The treatment capacity of the facility is equivalent to the Tokai Reprocessing Plant (0.7 ton of HM/day), and the TVF employs fully remote operations and maintenance in a large vitrification cell. All the equipment in the cell area are designed in compliance with standardized rack-mounted modules. The reception of HLLW from the Tokai Reprocessing Plant are scheduled to start in early 1992.

The research and development on storage technology focused in cooling system, durability and a seismic character of the facility. Design study of a High-Level Vitrified Radioactive Waste Storage Plant based on these results is also under way.

13.3.2 Geological Disposal
Geological investigation had been carried out through in the first stage, achieving the results that the geological disposal is potentially feasible within geological environment combined with engineered barrier in Japan.

FIG.13.1. National programme for geological isolation of HLW.
The second stage initiated in 1985 envisions to implement (1) R&D activities on the nationwide basis and (2) study on readily available data and necessary field investigation in an attempt to select candidate disposal site(s).

In the course of the R&D scheme (Para.1 above) it is necessary to establish a basic technology which facilitates the selection of candidate disposal site(s) and to step up comprehensive researches and experiments for development and safety assessment of the geological disposal system.

A methodical approach to the R&D to the system has been implemented during the first half of the second stage with the following projects as priority items.
(1) Conceptual design of geological disposal system
(2) Performance assessment and natural analogue investigations
(3) In-site experiments in various geological environments
(4) Improvement of site characterization technology
(5) Construction of large-scale research institutes
(6) Establishment of information control system
(7) International cooperation

13.3.3 TRU Waste

Radioactive wastes containing transuranic (TRU) nuclides generated from the operation of reprocessing plants and plutonium fuel fabrication facilities are currently being safely stored in the same manner as low-level radioactive wastes. Currently, more than 40,000 drums of TRU wastes are being stored at the PNC's Tokai Works.

TRU wastes have been treated similarly to the low-level wastes. However, the importance of the research and development program on TRU waste management is intensified in the report of the Advisory Committee on Radioactive Waste Management focussing the following items:
* Reduction of the creation of wastes
* Reduction of volume of wastes
* Solidification of wastes into a form appropriate for disposal
* Segregation of non-TRU from TRU wastes
* Disposal

In order to reduce creation of TRU wastes, improvement of in-process equipment and operation techniques are being carried out.

Furthermore, efforts on the development of techniques for volume reduction of wastes are being made. These involve the incineration of combustible materials, cyclone incineration of organic chlorinated wastes, acid digestion, electroslag remelting of metal wastes and solidification of non-combustible wastes into ceramic-like waste form by microwave heating. These techniques are being demonstrated at the Pu-contaminated Waste Treatment Facility (PWTF) of PNC, which is now under operation since 1987.

The techniques for decontamination and dismantling of large-size waste contaminated with TRU nuclides have also been developed in PNC, which will be applicable to decommissioning of nuclear fuel cycle facilities. These include ice/dry ice blasting technique, electropolishing technique and redox technique for decontamination, and portable plasma cutting remote arm. These techniques have been developed and partly demonstrated at the Waste Dismantling Facility (WDF) of PNC's O-arai Engineering Center since March 1984.

13.3.4 Construction Program of the Radioactive Waste Storage Research Center

The center is to be constructed for the purpose of storing vitrified HLW and immobilized LLW, and developing waste disposal technology including utilization of heat and radiation from HLW. 4,000,000m of land is to be provided for the center which will comprise a high-level vitrified radioactive waste storage plant and low-level immobilized radioactive waste storage facilities as main facilities and a research and development laboratory, an underground research laboratory, a hot test facility and others as adjacent facilities.

80 billion yen is anticipated for overall expenditures for the first 10 years.