A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

Power Reactor and Nuclear Fuel Development Corporation

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(Contents)

Abstract

The main R&D results of Japanese activities are summarized as follows: (1) the experimental 140 MW(th) sodium cooled fast reactor "Joyo" provided abundant experimental data and excellent operational records, attaining more than 50,000 hours of operation since its first criticality in 1977; (2) the prototype 280 MW(e) fast reactor "Monju" reached initial criticality on 5 April 1994, presently Monju is under the cold shutdown state because of secondary sodium leak on 8 December 1995, and multiple cause investigations of the sodium leak are being performed; (3) the Japan Atomic Power Company is promoting design studies for demonstration fast reactor (DFBR) with a power output of 600 MW(e) and R&D for DFBR are being conducted under the cooperation of governmental and private sectors.

1. General Review

1) The Japan Atomic Energy Commission (JAEC) promulgated Japanese "Long-term Program for Development and Utilization of Nuclear Energy" in June 1994. In the program, followings were concluded:

a) because of the scarcity of energy resources, plutonium utilization is the key development for long-term energy supply and energy security of Japan; Japan intends to steadily and systematically carry forward development of fast breeder reactors,

b) the research and development for demonstration FBRs (DFBRs) should be done with the cooperation of governmental and private sectors,

c) utilities should play the major role in design, construction and operation of the DFBR aiming at the commercialization by the year about 2030 through construction of two DFBRs with a step-by-step improvement of technologies and economics; the start of construction of DFBR-1 is expected in the early 2000's in the program.

d) Power Reactor and Nuclear Fuel Development Corporation(PNC) should play the central role in technological development over the entire period to commercial commissioning of FBRs through continuous long-term independent R&D efforts aiming at the establishment of technological system particular to FBRs.

2) The experimental reactor Joyo, located in the O-arai Engineering Center(OEC) of PNC, has provided abundant experimental data and excellent operational records attaining 51,200 hours operation in total by the end of March 1996, since the first criticality in 1977.

3) On the prototype reactor Monju, the initial criticality was achieved in April 1994. The nuclear heating test was started in February 1995, and the first connection to the grid was

succeeded in August 1995. In the power -up test, however, a sodium leak occurred in the C-loop of the secondary heat transport system on December 8, 1995. Presently Monju is under the cold shutdown state and multiple cause investigations of the sodium leak are being performed.

4) As for the demonstration fast breeder reactor (DFBR) of Japan, the Japan Atomic Power Company (JAPC) conducted conceptual design studies for the past several years, and confirmed the feasibility of top entry loop type reactor concept. Based on the results of the design studies, the Federation of Electric Power Companies (FEPC) decided in January 1994 to start construction of the DFBR plant at the beginning of the 2000's. FEPC also decided the basic specifications of the DFBR plant.

The related research and development (R&D) works are underway at several organizations under the discussion and coordination of the Japanese FBR R&D Steering Committee, which was established by JAPC, PNC, Japan Atomic Energy Research Institute (JAERI) and Central Research Institute of Electric Power Industry (CRIEPI).

Progress of the design study and the related R&D are reported to the Subcommittee on FBR Development Program of JAEC.

5) Recent major emphases on the PNC's R&D are placed on the development of technologies specific to FBRs as well as the integrated feedback of all existing R&D results and experiences to the development of DFBR.

Furthermore, the overall functional and performance tests of Monju, is another important key role to attain further excellency of FBR technology, with full efficient usage of the test results.

6) R&D on following tasks are also in progress for development of DFBR, for excellent technology to attain FBR commercialization, and for technological breakthrough.

a) improvement of reactor core safety
b) improvement of plant safety
c) development of measures for probabilistic safety assessment
d) development of high performance core and fuel
e) study for various fast reactor core
f) development of structure and material for high temperature system
g) improvement of sodium technology
h) synthesis assessment of Monju data
i) synthesis assessment and improvement of plant technology using Monju
j) design study of plant system

7) In addition to the MOX fuel fabrication at the Plutonium Fuel Fabrication Facility for Joyo, Fugen (ATR), and BWRs in Japan, a new Plutonium Fuel Production Facility (PFPF) was constructed at Tokai Works of PNC. PFPF started production of initial core fuel of Monju in October 1989 and completed in January 1994.

8) On the FBR fuel recycling, adding to the experiences at the Tokai Reprocessing Plant, R&Ds are underway at three Engineering Demonstration Facilities (EDF-I, II, III) and Chemical Processing Facility (CPF), integrating the results to the design of Recycling Equipment Test Facility (RETF) and future FBR Fuel Recycling Pilot Plant. The construction of RETF was initiated in January 1995.

9) Following the national program on waste management, PNC is also actively contributing to the area of vitrification of high level liquid waste, geological disposal of it, and low level transuranium bearing waste treatment, and promotion of construction of a storage engineering center in Hokkaido.

10) Aiming to the age of future FBR commercialization, further extensive and effective collaboration with foreign institutions will also have to play an important role.

2. Experimental Fast Reactor Joyo

2.1 General Status

This report covers the activities of Joyo from April 1995 through March 1996.

Various tests of nuclear and plant characteristics of the MK-II core were carried out successfully during this period. Figure 2.1 shows the core configuration for nuclear and plant characteristics test. Major items of the experiments are as follows :

1) Power reactivity coefficient
2) Isothermal reactivity coefficient
3) Heat transfer characteristics of dump heat exchangers

After this test operation, the 11th periodical inspection has begun in May 1995.

The operating history of Joyo is illustrated in Fig.2.2. The total operation time as of March 31, 1996 is about 51,200 hours since the initial criticality in 1977, and also the accumulated thermal power is about 4,160 GWh.

A large number of irradiation tests, such as for the nitride fuel and the carbide fuel, are now in progress. The creep test of fuel cladding materials under irradiation with MARICO* is also in progress.

* Materials Testing Rig with Temperature Control

2.2 Upgrading Project of Joyo(MK-III project)

The Joyo modification project named the MK-III project is planned to upgrade its irradiation capability. The safety review for the MK-III project by government has been completed in September 1995. The plant modification schedule will be discussed because of the influence of the Monju secondary sodium leak accident.

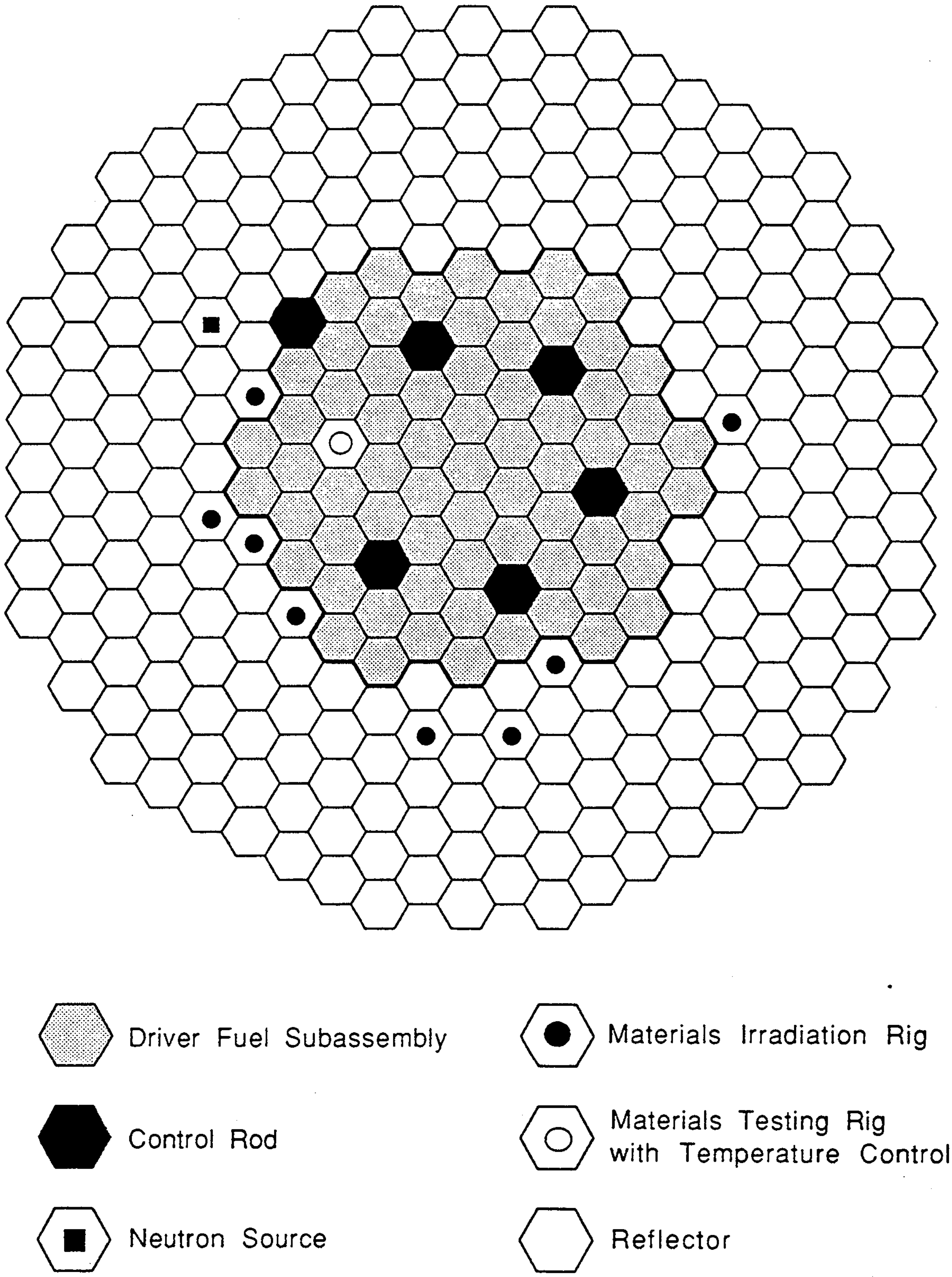


Fig.2.1 Core Configuration for Nuclear and Plant Characteristics Test

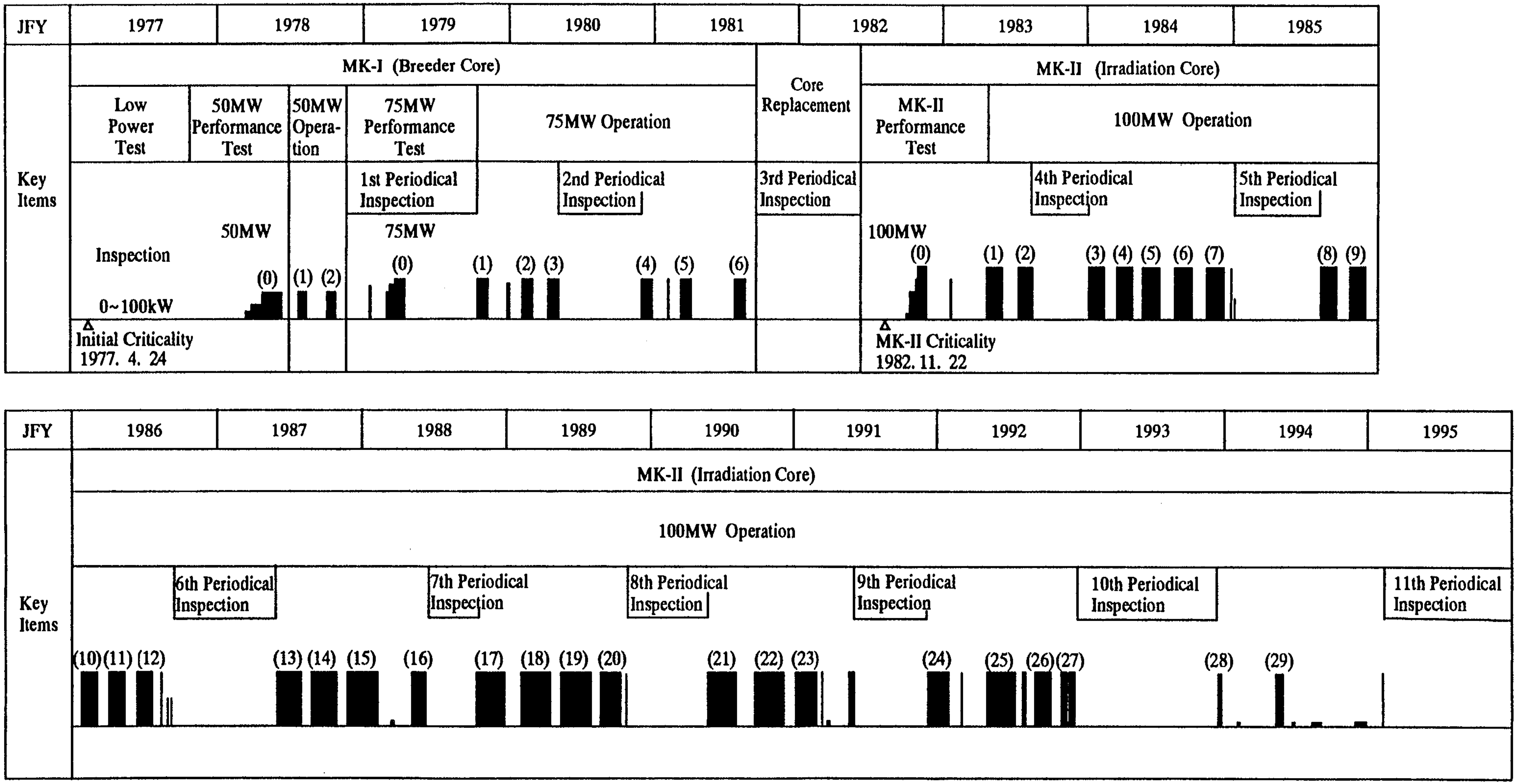


Fig. 2.2 Operating History of JOYO (As of March, 1996)

3. Prototype FBR, Monju

3.1 Construction Schedule

The Monju site is located on the north side of the Tsuruga Peninsula in the central Japan, 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 works have been carried out with special attention to the environment.

Major milestones of the construction schedule (shown in Fig. 3.1) are as follows;

Oct. 1985	Start of Construction
Apr. 1987	Completion of Construction of the Reactor Containment Vessel
Oct. 1988	Installation of Reactor Vessel
Apr. 1991	Completion of Construction

3.2 Present Status

The equipment installation was finished and the construction was completed in Apr. 1991. The function tests were performed from May 1991 to Dec. 1992. After that, the start-up tests was performed. The reactor power was increased gradually from Feb. 1995. Generating electricity and connecting to the grid was started in Jul. 1995. During 40% start-up test, the reactor was manually shut down due to the sodium leak in the secondary heat transport system (SHTS), Loop C in Dec. 1995.

3.3 Pre-operational Tests

Pre-operational tests were composed of Function Tests and Start-up. The schedule in Fig. 3.2 shows the general outline of the Pre-operational tests.

1) Function Tests

Function tests were carried out from May 1991 to December 1992.

Function tests were conducted to confirm the function and performance of the plant systems, and were composed of various tests and inspections during fabrication and installation of the components in Monju.

2) Start-up Tests

The purpose of the start-up tests is to confirm and to evaluate the performance of the core, the plant systems and the components.

The start-up tests consist of criticality test, reactor physics test, power-up test, etc.

First of all, the pre-performance test of plant system was conducted to examine plant characteristics under high temperature condition without nuclear heating.

Initial criticality was achieved with 168 fuel assemblies in Apr. 1994.

After the criticality test, the reactor physics test was performed and core reactivity worth, core reaction rate distribution , core flow rate distribution, etc. were measured.

The nuclear heating test was started in Feb. 1995 and the reactor power was increased gradually. Monju was connected to the grid in Aug. and then power-up test was started. In the power-up test, plant characteristics under power operation and transient condition was confirmed up to 40% power operation.

When power was being increased for the plant trip test, as a part of 40% power test, a sodium leak occurred in SHTS, Loop-C in Dec. 1995. The leak took place near the outlet of the secondary side of the intermediate heat exchanger (IHX). (Fig. 3.3)

The reactor was manually shut down to cold shutdown state, and the sodium in the primary and the secondary loops of the affected Loop C was drained.

Presently the examinations of the leak part of the temperature sensor have been performed, and also multiple investigations of the cause of the accident are being performed.

Table 3-1 **Principal Monju Plant Design Characteristics**

Reactor Type		Sodium cooled FBR, loop-type
Thermal Power		714 MW
Gross Electrical Power		280 MW
Core	Equivalent Diameter	Approx. 180 cm
	Height	93 cm
	Volume	2340 lits.
Fuel		PuO_2 - UO_2
Pu Enrichment	(Pu fissile)	(Inner core/outer core)
	Initial Core	15/20%
	Equilibrium Core	16/21%
Fuel Inventory	Core(U+Pu metal)	5.9 ton
	Blanket (U metal)	17.5 ton
Average Burn-up		Approx. 80,000 MWD/T
Cladding Material		SUS316
Cladding Outside Diameter/Thickness		6.5/0.47 mm
Permissible Cladding Temperature (middle of thickness)		675 °C
Power Density		275KW/lit
Blanket Thickness		Upper 30 cm
		Lower 35 cm
		Radial 30 cm
Breeding Ratio		Approx. 1.2
Reactor inlet/outlet Sodium Temperature		397/529 °C
Secondary Sodium Temperature (IHX inlet/outlet)		325/505 °C
Reactor Vessel (height/diameter)		18/7m
Number of Loops		3
Pump Position (Primary and Secondary Loop)		Cold Leg
Type of Steam Generator		Helical Coil, once-thorough Unit Type
Steam Pressure (Turbine Inlet)		127 kg/cm^2 g
Steam Temperature (Turbine Inlet)		483 °C
Refueling System		Single Rotating Plug with Fixed Arm FHM
Refueling Interval		Approx. 6 Months

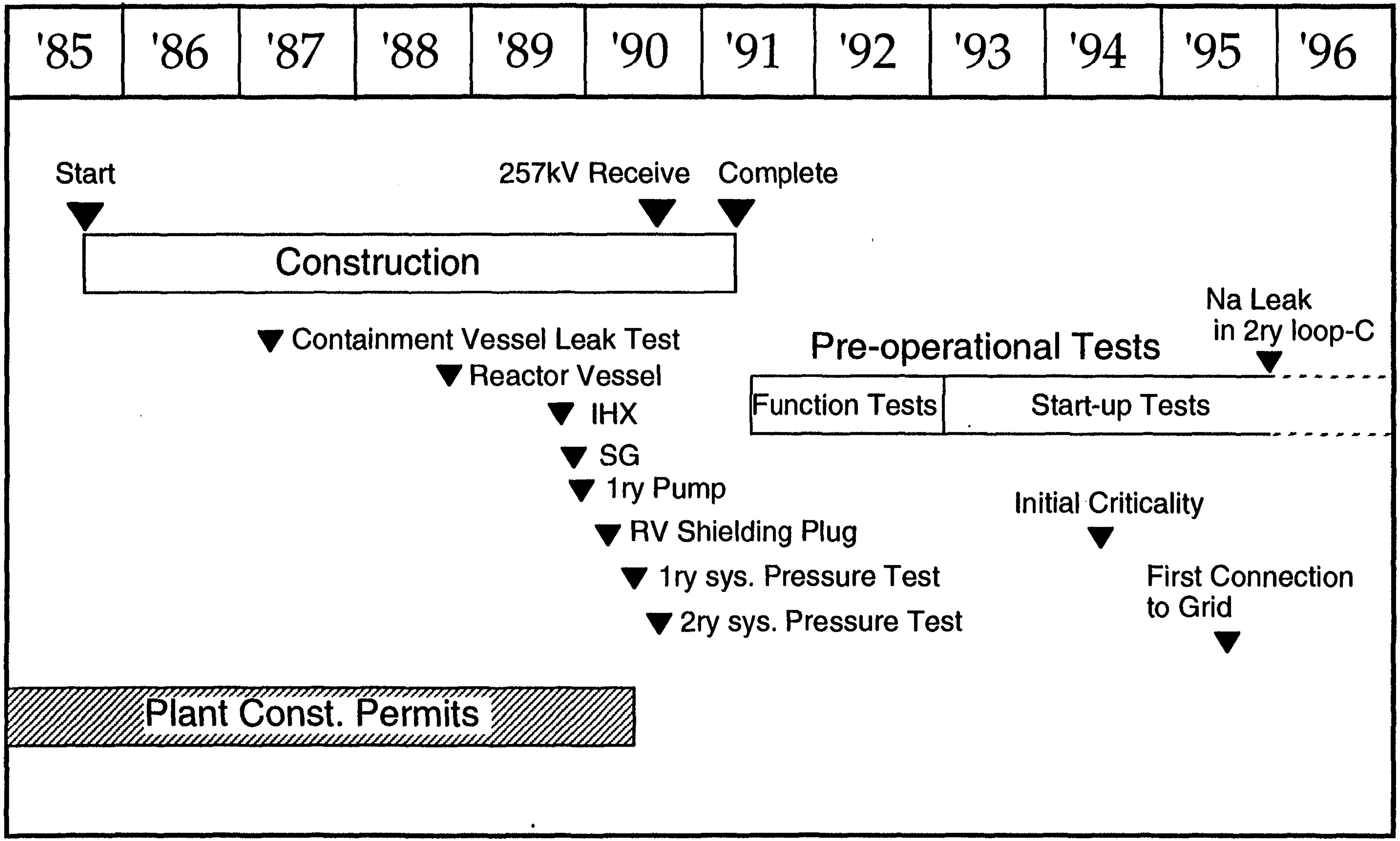


Fig. 3.1 Monju Construction & Tests Schedule

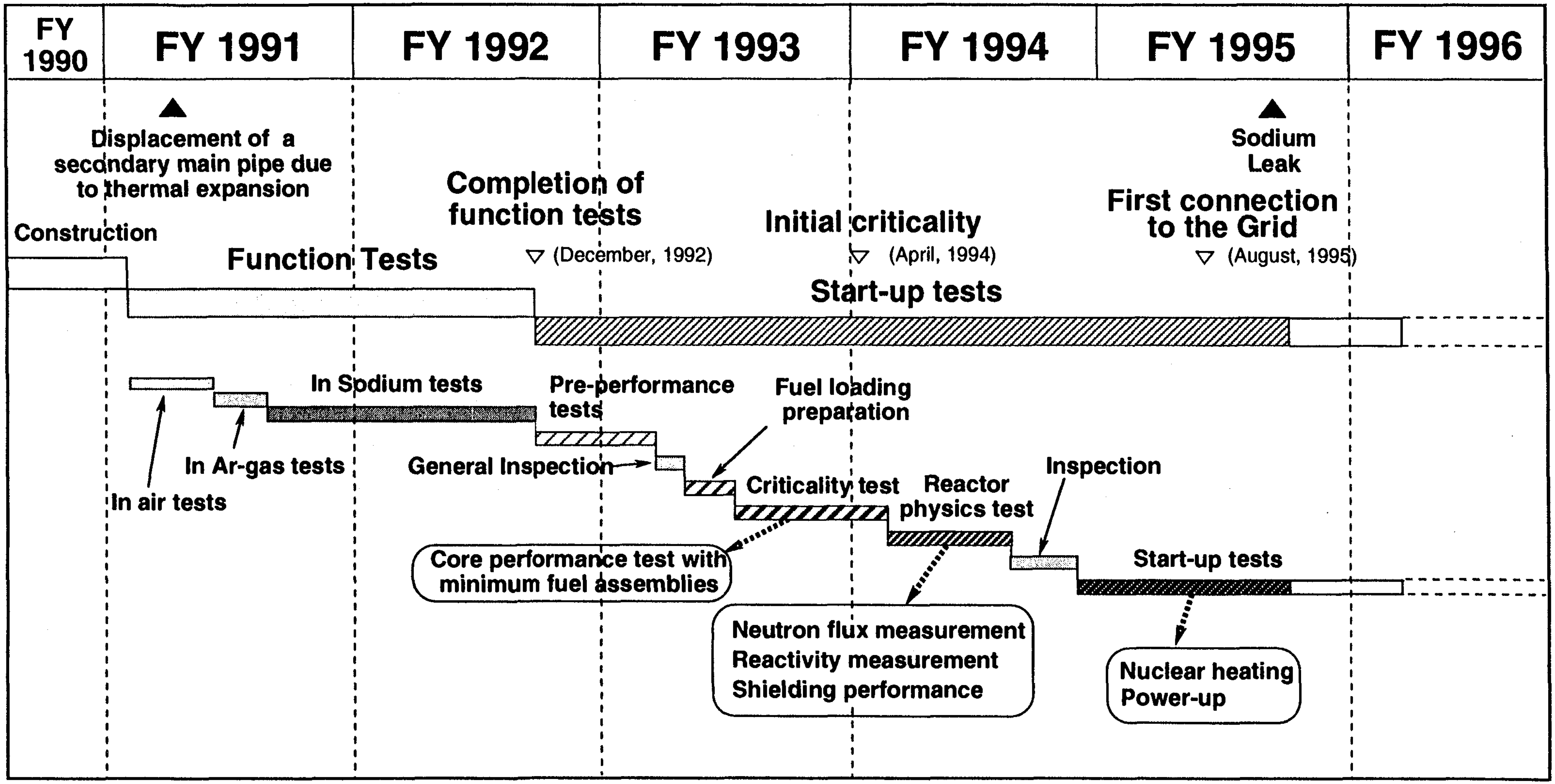
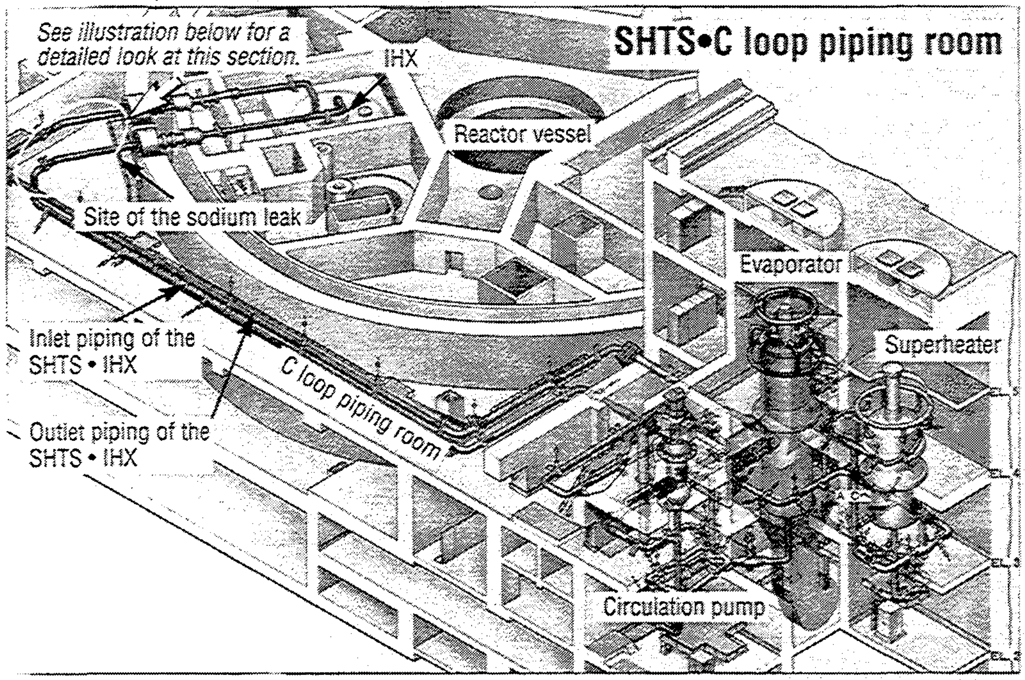


Fig 3.2 Schedule of Monju Pre-operational Tests



As seen from the direction of the arrow above

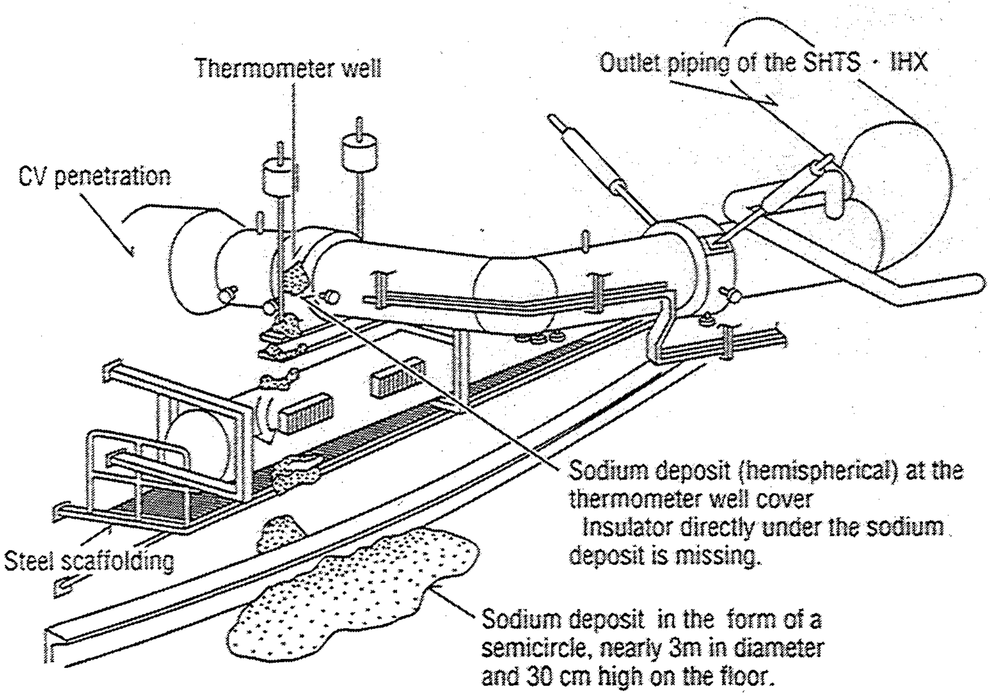


Fig 3.3 Schematic view of a sodium leak in SHTS C loop piping room

4. DFBR and PNC's Design Study

4.1 Overview

The Japan Atomic Energy Commission (JAEC) promulgated Japanese "Long-term Program for Development and Utilization of Nuclear Energy" in June 1994. In the program, it was concluded that the research and development for demonstration FBRs (DFBRs) should be done with the cooperation of governmental and private sectors, and that utilities should play the major role in design, construction and operation of the DFBR aiming at the commercialization by the year about 2030 through construction of two DFBRs with a step-by-step improvement of technologies and economics.

The start of construction of DFBR-1 is expected in the early 2000's in the program.

4.2 Design Study of DFBR

The Japan Atomic Power Company (JAPC) has conducted for the past several years conceptual design studies of the Demonstration Fast Breeder Reactor (DFBR) in accordance with the basic policy of the Federation of Electric Power Companies (FEPC) and has confirmed the feasibility of top entry loop type reactor.

Based on the result of these design studies, FEPC decided in January 1994 to start the construction of the DFBR plant in the early 2000's, and the major specifications.

The optimization study of the DFBR has been conducted by JAPC for three years since FY1994, based on the major specifications and those design studies. These studies have been focused on core safety enhancement including the application of gas expansion modules (GEM), increase of design margins for the structural tolerance of the containment facilities, feasibility and licensability of seismic isolation plant for introduction to the DFBR. Through these studies, the goal to make the design of the whole DFBR plant harmonious with both safety and economic viability, and to prove preparation for the basic design.

4.3 PNC's Design Study

In 1988, PNC started plant design study applying key technologies such as reactor vessel head access concept and performed plant construction cost evaluation.

PNC conducted design study on a 600MWe-size plant for 1990 - 1991. In 1992, JAPC and PNC discussed on the mutual design results to improve the demonstration reactor design study. Design study on a 1300MWe-size plant featured with passive safety using nitride fuel was completed at PNC in 1995.

PNC is now studying advanced recycling system using new type fuel (nitride, metal) and minor actinide, and is conducting conceptual design study of Recycle Test

Reactor focusing on core design in order to study FBR core variety such as plutonium burning, minor actinide burning and use of new type fuel. Necessities of the Recycle Test Reactor are being discussed at the Subcommittee on Advanced Recycling Program of JAEC.

5. Reactor Physics

5.1 Development of Analytical Method

The effort to develop a neutron transport code based on an improved nodal method has been continued in order to treat the Hex-Z geometry of FBR cores more accurately. In the previous version of the nodal code, the axial leakage from a node was approximated by a second order polynomial expansion of the flux, which occasionally leads to the difficulty of numerical convergence, especially in the case of control rod inserted core. A method to evaluate the boundary flux distribution interpolating the fluxes of axially adjacent nodes was newly developed to overcome such steep flux-gradient cases. A benchmark test of an FBR core showed the new treatment of axial leakage could obtain the flux convergence within the error of o.1%dk for keff, even in the case where the previous method failed.

The continuous-energy Monte Calro method, which has no analytical modeling errors in principle, has become more realistic in Japan because of both the release of an efficiently vectorized code and the rapid improvement of computer hardware. PNC is now utilizing the Monte Carlo method as a tool to verify the standard deterministic analytical system which uses ABBN-type 70-group constant sets and three-dimensional diffusion and/or transport codes. A comparison between the Monte Carlo method and the standard deterministic method agreed well within the difference of 0.1%dk for the keff of ZPPR-9 FBR benchmark core.

5.2 Critical Experiment and Analysis

In order to improve the design method and accuracy of large FBRs, extensive work has been performed to accumulate and evaluate many kinds of experimental and analytical results from the past fast reactor physics study. As a part of efforts to develop the standard database for FBR core nuclear design, a latest version of nuclear data library in Japan, JENDL-3.2, was applied to the large FBR core critical experiment JUPITER, and the results were compared with those of the previous library, JENDL-2. One particular feature of JENDL-3.2 is the good prediction of sodium void reactivity within 10%, on the other hand, the former JENDL-2 largely overestimated it by 20-40%. From a sensitivity study, it is found that sodium inelastic and Pu-239 capture reactions, which had not been considered important before, mainly contributed to the great improvement of JENDL-3.2, although many other nuclide-reactions had opposite effects and canceled with each other.

5.3 Cross-Section Adjustment

Compared with critical experiments, the power reactors have some special characteristics such as changes of fuel compositions with core burnup and increase of fuel temperature due to heat generation. About the burnup characteristics, PNC has successfully developed a code system to calculate the cross-section sensitivity of those, based on a generalized perturbation theory including burnup chain reactions. On the other hand, the present cross-section adjustment method cannot treat the Doppler reactivity which is dominated by resonance-peak broadening of cross-sections. An effort has been continued to extend the applicability of the cross-section adjustment and design accuracy evaluation system to the Doppler effect. This year a cross-section set including self-shielding factors was preliminarily adjusted by Doppler data of ZPPR experiment as well as other general core parameters. As a result, the C/E (Calculation/Experiment) values of the Doppler experiment were found to become close to unity after the adjustment without any harm to prediction of other core characteristics.

5.4 Shielding Experiment and Analysis

Post-analysis of Japanese-American Shielding Program for Experimental Research (JASPER) was almost completed. Analytical results of JASPER were effectively utilized in the Demonstration FBR shielding design review, which was conducted in the joint study of Japan Atomic Power Company.

As the next step of shielding study, a new work was initiated to develop a standard database for FBR shielding design this year. In the database, the experimental and analytical information from JASPER and other shielding experiments will be evaluated and compiled in a systematic and consistent manner. As a part of the efforts, typical radial shielding experiments of JASPER is now being re-analyzed with the latest version of nuclear data library, JENDL-3.2, and compared with the results of the former JENDL-2.

6. Systems and Components

6.1 Reactor Shutdown System

Research on the self-actuated shut-down system (SASS) by using a curie point magnet is in progress. A system performance test on SASS by use of Joyo is under planning based on various out pile characteristic tests conducted in air and sodium environment.

6.2 Process Instrumentation

A development of ISI test equipment for Monju was completed in O-arai engineering center(OEC). Mock-up test rigs and ISI test equipments at OEC were disassembled and transferred to Monju from OEC.

A development of the visualization technique of heat transfer tube inner side by using Laser is in progress.

6.3 Steam Generator

PNC precedes a conceptual design study for a future FBR plant having a steam generator in the primary heat transport system. The studies on a double-wall tube(DWT) steam generator have been in progress regarding structural integrity and thermal hydraulic performance of the DWT steam generator.

The 1MW scale DWT steam generator test was completed and post test examination is underway.

6.4 Sodium Component

An in-sodium performance test of a magnetic flux concentration type electro magnetic pump (EMP) was completed. An evaluation of the EMP was completed.

6.5 Thermal Hydraulics

A verification of plant dynamics simulation code (Super-COPD) by using Monju functional tests was carried out and the accuracy of the models was validated. Predictions for the Monju start-up test were carried out beforehand and the verification by using the test is planned.

7. Fuels and Materials

7.1 Fuel Fabrication

The PFPF (Plutonium Fuel Production Facility) equipped with automated and remote handling fuel production systems started fabricating Joyo and Monju fuels from October 1988.

7.2 Fuel Pin Performance

Fuel pin performance codes of MOX type pellet for analyzing the steady state and transient conditions have been improved since 1984, with the data of PNC/DOE operational reliability testing program in EBR-II, etc. The modification of codes for annular MOX fuel is emphasized with the irradiation data base. A three dimensional FEM code for an analysis of defected fuel pin behavior has been modified with PIE results of Run Beyond Cladding Breach (RBCB) pin in EBR-II. Development of nitride fuel performance code is also in progress.

7.3 Core Material

PNC1520 advanced austenitic steel was developed for high burnup FBR fuels. Improved performance than current PNC316 steel is demonstrated by the out-of-reacter and material irradiation testing programs. Design base standard of PNC1520 is completed and evaluation study is in progress to apply Monju and DFBR cores.

7.4 Irradiation Experiments

1) Joyo

Fuel pin irradiation continues with advanced austenitic cladding fuel pins and high strength ferritic cladding fuel pins. Large diameter annular pellet fuel pins for Monju high burnup and DFBR cores are also included in the test.

The fuel subassembly with CEA austenitic stainless steel cladding tubes has also been irradiated since 1988 and reached the final stage at 125GWd/t.

2) Foreign Reactors

Phase-II program of PNC/DOE collaborative operational reliability testing in EBR-II has been successfully completed by the end of 1995. Large amount of data base on TOP (transient over power) and RBCB (run beyond cladding breach) tests is summerized for typical MOX fuel pins.

The PIE of fuel subassemblies of PNC316 steel and PNC1520 advanced austenitic stainless steel irradiated in FFTF is in progress. The shipment of irradiated fuel materials to PNC O-arai was continued to take a further investigation of high burnup fuel behavior.

7.5 Development of Advanced Fuels

Feasibility study of advanced fuels (nitride, metal, carbide) has been conducted since 1986. PNC/JAERI irradiation test of nitride and carbide pins has been conducted in Joyo since 1994.

Mixed carbide fuel pins were also irradiated using the thermal reactor JRR-2 and JMTR of JAERI.

7.6 Post Irradiation Examination

Construction of PIE facility of Fuels Monitoring Facility (FMF-2) adjacent to existing FMF-1 at OEC has completed to handle the large fuel assemblies irradiated in FBR Monju and ATR Fugen. Hot operation of FMF-2 will start from late 1997.

8. Structural Design and Materials

8.1 Development of Structural Design Method

1) FINAS nonlinear structural analysis program

Effort to entire the capavility of the general purpose nonlinear structural analysis program FINAS is continued, particularly with respect to adoptive mesh generation based oann r-method and h-method. FINAS was mounted as the solver in CAE systems such as CADAS, ATLAS, FEMAP and so on. FINAS is currently used by many engineers at about 40 sites including fabricators and universities. The latest version, V12.0, was translated into English.

Personnal computer version of FINAS was developed.

2) Improvement of Elevated Temperature Structural Design Guide

The following rules are investigated to improve and extend the current Elevated Temperature Structural Design Guide.

i) Creep-fatigue design methods based on elastic analysis

A new creep-fatigue method, which is based on the concept of a general elastic follow-up model, is being developed. The elastic follow-up equations to predict strain magnification and creep relaxation for structural discontinuities are established.

ii) Design rules for weldment

A new design approach, taking into account the metallurgical and geometrical discontinuities inherent in weldment, is being pursued.

iii) Strain limit critera

A ratchetting criteria for multiaxial stress state, which are not provided explicitly in the design guide, was developed, and is being examined the applicability of the criteria to general components by FE Analysis.

8.2 Structural Test and Evaluation

Structural tests are being performed to improve strength prediction methods, to evaluate the adequacy of elevated temperature design rules, and also to verify advanced nonlinear structural analysis methods.

1) Thermal creep-fatigue test with small sodium loops (STST)

A thermal creep-fatigue test, whose specimen was a new test model "a cylindrical shell with cross-section gradually step-changing" is started and is continued with use of the test facility (STST).

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

A thermal transient test of a vessel with fillets model is now under way. A new test model, made of FBR grade 316 stainless steel, was fabricated.

3) Distortion tests of fuel sub-assembly duct (CMDT, WFT)

A new R&D, which is to investigate the distortion behaviors of internal structures, was started. Firstly a fundamental distortion tests of ducts of a fuel sub-assembly were performed with use of new equipment, CMDT and WFT, and then a distortion tests of several fuel sub-assemblies will be stared in near future.

8.3 Seismic Test and Analysis

1) A conceptual study on the vertical seismic isolation system for FBR components is underway. A series of shaking table tests and analytical works are included to assess the feasibility of the system.

2) Seismic analysis method development and verification on FBR core in the framework of "IAEA/IWGFR Coordinated Research Program on Intercomparison of LMFBR Seismic Analysis Codes" has completed in 1995.

8.4 Fracture Mechanics Tests and Evaluation

Both deterministic and probabilistic fracture mechanics methodologies are being development for the integrity assessment of flawed or cracked structure.

Computer codes developed at PNC as CANIS-J for calculation of fracture mechanics parameters, CANIS-G for simplified crack propagation analysis were modified further.

Crack propagation test with a cylinder with an axial temperature gradient is continued.

8.5 Structural Material Tests and Evaluation

Structural material tests in air, in sodium, in water/steam, and under post-irradiation condition have been conducted to revise the Monju Material Strength Standard and to prepare a new version for DFBR.

The test program in air and in sodium environment is called "Capella" program and the step-1 program(1985-1987) , the step-2 program (1988-1990) and the step-3(1991-1993) were already completed. The step-4 are currently underway with emphasis on long-term extrapolation.

The post-neutron irradiation tests are underway within the scope of neutron irradiation program "Spica".

1) Tests in Air

The present Capella step-4 program includes following subjects;

- Validation of long-term extrapolation of a new criterion for creep-fatigue failure strength of weldment, inelastic constitutive equations on new materials(modified 9Cr-1Mo steel and FBR grade 316 stainless steel)
- Improvement of LBB evaluation method for FBR plant.
- Development of the material strength standard for modified 9Cr-1Mo steel and FBR grade 316 stainless steel

A new concept creep-fatigue failure criterion was proposed using secondary creep basis ductility exhaustion for stainless steels. Applicability of the present criterion to modified 9Cr-1Mo steel was validated by the mechanical and metallurgical studies. The equations of cyclic plasticity of FBR grade 316 stainless steel were revised based on the data-base consisted with low cyclic fatigue and tensile test data.

2) Test in Sodium and Water/steam

Mechanical strength (tensile, creep fatigue, creep fatigue) tests on 316FR (nitrogen controlled) in sodium are still continued in the program to evaluate the carbon and nitrogen transfer effects. Low-cycle fatigue and creep fatigue tests on modified 9Cr-1Mo steel in water/steam environment were conducted.

3) Tests in Irradiation Environments

Surveillance tests for the Class 1 components of Joyo were conducted to confirm the integrity of the reactor by evaluating irradiation effects of the same materials.

The test data were used for the planning of Joyo operating program.

Tests for the Class 1 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 Material Strength Standard for Monju are also in progress.

Both forged and rolled SUS304 steels were irradiated in Joyo using SMIR (Structural Material Irradiation Rig).

Another test for DFBR was conducted to clear the relationship between creep rupture strength and metallurgical variables such as chemical composition, grain size and production process.

Several post irradiation material tests and in-pile creep tests on FBR grade 316 stainless steel were continued in Joyo and JMTR (Japan Material Test Reactor of JAERI) in accordance with the R&D program Spica step-2.

4) Data Processing System

Material data are complied using specific data coding sheets, and the data inputs to the computer data processing system SMAT are still continued.

Entry data in SMAT are currently more than 12,000 data points on 11 different kinds of mechanical tests (including tensile, low cycle fatigue, creep) for 10 kinds of FBR structural steels.

9. Safety

9.1 Safety Evaluation for Normal and Abnormal Events

Safety evaluation studies have been conducted for confirming the physical phenomena and integrity of the fuel subassemblies, the core internal structures and the heat transport systems during the normal operation, scram transients and the early stage of postulated accidents. On this account, thermohydraulic experiments related to the decay heat removal by natural circulation have been carried out, and the development and validation of the thermohydraulic safety analysis codes is also in progress.

1) Thermohydraulic Experiments

An integral sodium experiment has been carried out with a partial core model composed of seven subassemblies, inter-wrapper gaps, an upper plenum and dipped cooler. The tests for inter subassembly heat transfer and core-plenum interaction were completed. A series of tests is under way focusing inter-wrapper flow under conditions of natural circulation decay heat removal.

2) Development and Validation of Analysis Codes

Subchannel analysis codes, ASFRE for single phase flow and SABENA for two-phase flow, have been developed for the purpose of predicting fuel element temperatures and thermohydraulic characteristics in the FBR fuel subassembly. ASFRE has the detailed wire-spacer model called distributed flow resistance model, which calculates the effect of wire-spacer on thermohydraulics, and the planer and porous blockage models are also implemented in ASFRE to evaluate the accident in the fuel subassembly. In this reporting period, the improvement of the thermal conduction model was carried out to be able to estimate the azimuthal temperature distribution on the fuel cladding. Furthermore, the analyses of the 37-pin and 61-pin bundle data obtained from the sodium experiments were conducted as a code validation study.

SABENA is the two-phase flow analysis code based on the two-fluid model. The correlation equations of wall friction and two-phase friction corresponding to the change of the two-phase flow pattern were implemented in SABENA, and the effects of improvement were confirmed by the validation analyses using sodium boiling test data at decay heat level.

Concerning about multi-dimensional thermohydraulic codes, the improvement of the turbulence model in AQUA was carried out for simulating temperature fluctuations by thermal striping at the core exit region. The peak-to-peak levels of temperature fluctuations were well simulated by the turbulence model based on an algebraic stress model (ASM).

A fluid-structure thermal interaction analysis code FLUSH have been developed to evaluate the temperature distribution in the shield plug. As for the pre-test analysis of the MONJU performance test, the temperature distribution in the shield plug was successfully predicted by the interactive calculation between structural components and cover gas fluid regions, and the applicability of FLUSH to practical systems have been identified.

A three-dimensional arbitrary-Lagrangian-Eulerian (ALE) finite element code SPLASH-ALE has been modified and applied to a simulation of free surface behavior. A capillarity model based on the cubic Spline function has been developed and incorporated in SPLASH-ALE. Free surface sloshing and capillarity phenomena analyses have been performed to validate the model. The simulation results are in excellent agreement with the theoretical results regarding sloshing frequencies and the shape of the free surface. It was confirmed that capillarity should be adequately considered in the free surface dynamics analysis.

Regarding the whole plant simulation, a module type plant dynamics code Super-COPD has been developed and validated using MONJU plant data. In this reporting period, a cover gas module has been implemented in Super-COPD. The cover gas module is considered to be quite useful to evaluate the effect of cover gas pressure fluctuation on coolant surface levels in plant components. The performance test analysis of MONJU is in progress to validate the effectiveness of the model to practical problems.

A space-dependent plant dymamics code system has been developed to apply to the performance evaluation of passive safety features of FBRs. Three-dimensional calculations on neutronic characteristics have been perfomed to investigate the local power distorsion that is expected to occure in the case of asymmetric functioning of passive shut-down devices. The development of interface program between neutronic and thermohydraulic calculations is sheduled to be started.

9.2 Degraded Core Research

The degraded core research at PNC addresses: the fuel failure propagation during local-fault accidents, and physical phenomena during core disruptive accidents (CDAs).

The local fault studies include out-of-pile experiments on local coolant blockages by porous media with water to confirm fuel pin integrity through analyses using a detailed sub-channel code, ASFRE. A new series of sodium experiments is also planned. To focus on an intra-subassembly failure propagation behavior and to establish a termination scenario, a synthesis study is in progress by reviewing and interpreting the past in- and out-of-pile experimental data base, such as SCARABEE, MOL 7C, SLSF and TREAT.

The current out-of-pile experimental program at PNC consists of various simulant melt experiments using the MELT-II facility. A series of experiments to investigate the erosion behavior of solid structures by a high-temperature molten jet was completed. Experiments to study thermal interactions between a molten jet and coolant are in progress. A low-temperature series of tests with Woods metal and water has been completed. From the experiments, four distinct modes of interaction behaviors were observed. Conditions necessary for energetic interactions were identified, based on consideration of the minimum film boiling temperature. From extrapolation to reactor materials, it is predicted that such conditions are unlikely to be met. High-temperature experiments with steel and and alumina are planned for the next few years.

On the in-pile experiments jointly conducted with French CEA, all the planned tests have completed for the CABRI-FAST program, in which slow and fast transient tests have been conducted, mainly with high burnup, annular fuel pins. A joint synthesis work is underway. Starting in 1996, a next joint in-pile test program, CABRI-RAFT, is inihated, where the total of 8 tests are planned in the CABRI and SCARABEE reactors through 2000.

The CDA analysis code development and validation studies have continued extensively: SAS4A for CDA's initiating phase and SIMMER-III for the transition phase. For SAS4A, under collaboration with FZK and CEA, a reference code version REF-96 has been developed for common use in the future. A special single channel code PAPAS-2S, which models detailed fuel pin mechanics, is also being elaborated through CABRI analyses. Version 2 of SIMMER-III has almost completed with coupling with the neutronics model. A code assessment program is jointly participated by FZK and CEA. Also underway is the development of a new interface code SAME-II that couples SAS4A and SIMMER-III. The system of new-generation codes are being applied to CDA studies of D-FBR and future fast reactors.

Finally, a long-term research program SERAPH (Safety Experimental Reactor for Accident Phenomenology), has been undertaken over the last several years at PNC. This program aims at identifying long-term research needs of integral in-pile safety experiments, which are essential to further advance safety technologies towards FBR commercialization in the next century. The areas of primary interest include: eliminating recriticality concerns during CDAs, demonstrating advanced fuel design and types, and establishing local-fault scenarios. A feasibility study with related R&D's on elementary technologies is in progress to define required features and performance of the in-pile test facility, and to establish a basic design concept of a reference core.

9.3 Plant Accident Research

FBR plant accident research consists of two major activities. One is a study on a non-radiological sodium fire caused by sodium leakage from the intermediate heat transport system (IHTS), and the other is a study on the radiological source term, with emphasis being placed on quantifying various mitigation factors of fission product (FP) release and transport from failed fuels to the environment. The latter study also includes an integrity assessment study of the reactor containment with respect to FP leakage during a severe accident.

Most topical activity in FY1995 is the Monju sodium leak reproduction test and analysis. A series of prototypical sodium fire experiments were scheduled to clarify the sequence of the Monju SHTS sodium leak accident on December 8th, 1995 by use of the sodium fire test facilities such as SAPFIRE. Two tests were conducted so far to evaluate the sodium leak rate in the accident. Monju accidental analysis as well as pretest and post test analyses are under way using sodium fire code such as ASSCOPS, SOLFAS and CONTAIN.

In the source term study, a new experimental rig has been constructed for investigation of FP release behavior from high temperature fuel under an accidental condition. Various performance tests are in progress to validate experimental procedures and measurements. Experiments using irradiated fuels of Joyo are planned in near future. The FP bubble behavior in sodium and the FP release behavior from sodium pool surface are studied with the SABER and the START rigs, respectively. The TRACER code is under development to describe various FP behavior in an in-vessel sodium system. For an ex-vessel study, the hydrogen burn experiments are continued in atmospheric conditions containing sodium aerosols or mists. The CONTAIN-LMR code is continuously improved for analyses of overall severe accident progression in a containment.

9.4 Steam Generator Safety Research

Current steam generator (SG) safety researches consist of two major activities. The one is for improvement of the sodium-water reaction evaluation method for the large-scale demonstration FBR plant. For this purpose, an overheating failure propagation process is studied in detail because this may potentially determine the design base leak (DBL) of future plant SG's. A series of simulation experiments in the TRUST rig has been conducted to examine structural performance of high-temperature tube under accident conditions. Obtained data are analyzed using the computer code, FINAS, and resonable agreement was found between the data and the code. Present TRUST rig which simulates the failure of the heat transfer tube in the nitrogen gas pressurization is remodeled and TRUST-2 rig is newly manufactured in order to simulate the failure of the heat transfer tube in the superheated steam flow condition. Since these TRUST experiments simulate reaction heat by means of induction heating, large scale sodium-water reaction experiments are planned in the modified SWAT-3 facility. Effects such as sodium flow and tube cooling by water system blowdown will be simulated in the large scale experiments.

Another study for the SG safety is aimed for the FBR design concept which eliminates the secondary sodium loop. To support this conceptual feasibility study, double-wall tube SG design is proposed and examined. Since a reliable leak detection system is essential for this design, detailed investigation of tube failure detectability is performed. A inert gas system in the gap of double-wall tube is used for detection. In case of an inner tube failure, steam in the inert gas system is detected. On the contrary, inert gas contents in sodium is detected in case of an outer tube failure. Both experimental studies and code development are in progress to validate this detection system.

9.5 Research on Probabilistic Safety Assessment

PNC has been performing the research on Probabilistic Safety Assessment (PSA) for more than ten years as part of the R&D of a fast reactor.

The purpose of this research is to construct probabilistic safety models for a typical loop-type FBR plant so that an overall safety assessment can be performed. It is expected that (1) a systematic evaluation on the plant safety is conducted based on the quantitative analysis, (2) the insights on measures to enhance system reliability and safety are provided, (3) the operation and maintenance procedures are established based on a risk-based consideration, and (4) useful information is given to the development of basic policy on safety design and evaluation of a large LMFBR.

PNC has been improving the systems analysis code network which is able to perform a level-1 PSA. Recent efforts have focused on the development of a system configuration control program that evaluates the risk during the shutdown, a dynamic reliability analysis program (DYANA) dealing with the Emergency Operating Procedures, and a Living PSA System (LIPSAS). The LIPSAS has been installed at the site of Monju plant to examine the applicability of the system to safety management of a real plant. Furthermore, PNC has been developing a new software SAGE (Severe Accident Guidance using Expertise) which is able to quantify the consequence of core damage quickly with the expert system that models accident phenomena based on the experience of full scope PSA.

Efforts are being made to develop a new relational data base system of LMFBR component reliability data on an engineering workstation. The system is based on CREDO (Centralized Reliability Data Organization), a cooperative project between PNC and the USDOE, which ended in 1992. As part of the data analysis, failure rates of external leakage mode were quantitatively estimated for various valves in sodium system. Additionally, risk-related data on energy production systems such as solar photo voltaic energy system and LMFBR nuclear fuel cycle have been collected and health risk to the public is compared among various energy production systems.

A detailed shutdown PSA is underway. The reliability of the decay heat removal system (DHRS) during the plant shutdown has been evaluated based on the latest information on the inspection schedule. PSA is being applied to a large LMFBR with three IRACS loops, one DRACS loop, and a water steam system as the DHRS. The importance measures of each DHRS to the PLOHS frequency were evaluated. Also the study on the classification of safety evaluation events (SEEs) is ongoing. The failure probability of PS (prevention system) and MS (mitigation system) for a typical large LMFBR was examined and the occurrence frequency of the SEEs were quantified taking into consideration the multiple failure of PS and MS.

Level-2 PSA tasks (consequence analysis) are underway. The current effort includes a preliminary analysis of an E/T of ULOF accident in a large LMFBR particularly reflecting recent experimental and analytical knowledge. For example, it has been becoming apparent that the molten fuel could be discharged from the core region through an escape path with a large hydraulic diameter such as control rod guide tubes. Since this phenomenon could be highly effective to mitigate an energetic event in the transition phase, it is anticipated based on the preliminary assessment that the most probable sequence of ULOF is a mild one with neutronic shutdown due to fuel escape and thus non-mechanical energy release. For the ex-vessel physical process, an analysis of source term has been continued postulating a reactor vessel melt-through event. A debris-concrete interaction has been mainly investigated by means of a sensitivity analysis of the relation between the initial conditions such as debris mass and temperature and the quantity of radiological nuclides to be released.

10. Fuel Cycle

10.1 Mox Fuel Fabrication

1) Construction and Fuel Fabrication

R&D on fabrication of uranium-plutonium mixed oxide (MOX) fuel has been carried out since 1965 at the Plutonium Fuel Development Facility (PFDF) in Tokai works of PNC.

The Plutonium Fuel Fabrication Facility (PFFF), which started operation in 1972, has two fuel fabrication lines for Advanced Thermal Reactor (ATR) (10 ton MOX/year) and FBR (1 ton MOX/year). It has supplied the fuel necessary for the operation of ATR Fugen and FBR Joyo.

In parallel with the construction of Monju, construction of the Plutonium Fuel Production Facility (PFPF) (FBR line; 5 tons MOX/year) started in July 1982. It was aimed to develop fuel fabrication technologies with larger scale and automated equipment as well as to fabricate fuels for Monju and Joyo. The construction was completed in October 1987. After testing operation, production of Joyo fuel started in October 1988 as the first production campaign at PFPF. Production of the initial core fuel of Monju started in October 1989 and was completed in January 1994 at PFPF.

The PFPF is currently fabricating fuels for Joyo.

PNC is planning to extend the FBR line at PFPF so as to meet a Monju improving program.

About 145 tons of MOX fuel have been fabricated by the end of March 1996.

2) R&D on MOX Fuel Fabrication

Remotely controlled operation technology is one of the most important key element to achieve a large scale production of MOX fuel.

PFPF equipments including material transfer system are designed and manufactured so as to realize the fully automated operation.

Through the operation of PFPF FBR line so far, PNC has been accumulating experience for it.

10.2 Plutonium and Uranium Conversion

PNC developed a co-conversion technology utilizing the microwave heating direct denitration process (MH method) which converts plutonium nitrate and uranyl nitrate solution to MOX powder. Compared with the conventional method, it is a simple process and generates less liquid waste.

The Plutonium Conversion Development Facility (PCDF) (conversion capacity: 10 kg MOX/d), designed for demonstration of the co-conversion technology by MH method, was completed in February 1983. By the end of March 1996, it produced 12.6 tons of MOX powder containing about 5.2 tons of plutonium. The converted MOX powder were transported to PFFF and PFPF, in addition to about 1.8 tons of MOX powder processed at PFPF, and are being used for fabrication of MOX fuel for Fugen, Joyo and Monju.

Since depleted uranium recovered from reprocessing of spent fuel has generally higher U235 concentration compared to natural uranium, our country has decided to try to use it as LWR fuel by re-enriching and mixing it with other enriched uranium and by mixing with plutonium as fuels for ATR, etc..

Reconstruction of pilot scale reprocessed uranium conversion facility was completed in June, 1994.

Conversion test for reprocessed uranium was started in August, 1994.

11. FBR Fuel Recycling

In the area of FBR fuel reprocessing, PNC has developed process and equipment with remote handling technique, through large scale cold mock-up tests at the three Engineering Demonstration Facilities (EDFs) and laboratory scale hot tests at the Chemical Processing Facility (CPF) in Tokai Works, taking into account of accumulated experience in the Tokai reprocessing plant for LWR fuels.

The construction of Recycle Equipment Test Facility (RETF), where advanced process and equipment tests in engineering scale under hot conditions are conducted in order to enhance the technology and economical efficiency, is initiated in January, 1995.

Almost all of R&D activities are oriented towards the RETF project.

11.1 Process Research and Development

1) Head End process

In order to remove the hexagonal wrapper tube efficiently prior to fuel chopping, a disassembly system with CO_2 laser has been developed and tested. A reference cutting scenario has been established through tests with dummy fuel assemblies. A prototype test equipment of geometrically safe continuous rotary dissolover was fabricated and tested.

Test results on both equipment have been reflected on the RETF design.

2) Chemical Separation Process

Major effort of solvent extraction contactor development is paid on centrifugal contactor. The design of the prototype contactor for RETF has been completed successfully.

In order to eliminate the generation of secondary salt-bearing waste in the purex process, studies and tests on solvent cleanup with salt-free reagents and electro-reoxidation process for Pu have been continued.

3) Common Technology

To establish remote maintenance concept with rack system, key technologies such as bilateral servo manipulator (BSM), roll-in type rack, remote connector bank, and remote sampling system have been developed.

Materials of process equipment and advanced analytical system have been developed.

4) Hot Tests at CPF

Irradiated fuel from Joyo with burnup up to 99,800 MWD/T have been used for hot tests at CPF.

Through these hot tests, information of dissolution characteristics and nuclides behavior in the solvent extraction process have been obtained. Especially, PNC has developed the Np co-extraction process with U and Pu to enhance non-proliferation aspect.

11.2 Plant Construction and Design of Recycling Facilities

1) Recycle Equipment Test Facility (RETF)

Verification of high availability and economical prospects of FBR fuel reprocessing are essential for deployment of FBR and its fuel cycle. In order to accomplish them at future pilot plant, hot engineering demonstrations of important process and equipment are necessary in advance. From this viewpoint, PNC has planned the RETF project to provide a test bed for advanced equipment and process.

RETF features a large remote cell which accommodates both head-end and chemical process equipment test areas. Most of the chemical processes will be mounted on the racks installed along either cell wall. The maintenance of these chemical process equipment as well mechanical components will be conducted by using overhead crane and bilateral servo-manipulator (BSM).

The construction of the main test building was initiated in January 1995 and it is scheduled to start hot tests in the beginning of 2000s.

It is expected that important data and experience for designing of future FBR fuel reprocessing plants will be compiled through the RETF operation.

2) FBR Fuel Recycling Pilot Plant

PNC has a plan to construct the FBR Fuel Recycling Pilot Plant to verify the whole plant system and it will be the first step for achieving the cost requirement.

The operation will be initiated in the middle of 2010s.