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A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

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Power Reactor and Nuclear Fuel Development Corporation

This document was prepared for the IAEA/IWGFR Consultancy Meeting(May 4 - 7, 1993, Vienna) by Reactor Development Project of Power Reactor and Nuclear Fuel Development Corporation(PNC), with contribution from;

Nuclear Fuel Cycle Development Division

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(IWGFR: International Working Group on Fast Reactors)

A Review of Fast Reactor Program in Japan
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1. General Review

1) In accordance with the Long-term Program for Development and Utilization of Nuclear Energy defined by the Japan Atomic Energy Commission (JAEC), Power Reactor and Nuclear Fuel Development Corporation (PNC) is playing the key role in the development of a plutonium utilization system by fast breeder reactor (FBR), which is superior to the uranium utilization system by light water reactor, aiming to achieve future stable long-term energy supply and energy security of Japan.

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 50,000 hours operation in total by the end of FY 1992, since its first criticality in 1977.

3) On the prototype reactor Monju, 99.7% of construction work has already been completed and the start-up tests are in progress aiming at the initial criticality in 1993.(Photo 1)

4) As for the demonstration fast breeder reactor (DFBR) of Japan, the Japan Atomic Power Company (JAPC) is promoting design study under the contracts with several leading Japanese fabricators, including Toshiba, Hitachi and Mitsubishi Heavy Industries, for selection of the basic specifications of DFBR.

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 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.

① development of advanced fuels

② development of advanced large core

③ higher plant operating temperature

④ simplified advanced piping and components

⑤ development of rational confinement facilities

⑥ development of seismic isolation structures

⑦ development of simplified system without secondary loops

⑧ development of highly reliable decay-heat removal system

⑨ development of advanced operational and maintenance technology

⑩ establishment of rational safety logic

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 and MOX fuel production for Monju is in progress.

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.

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.

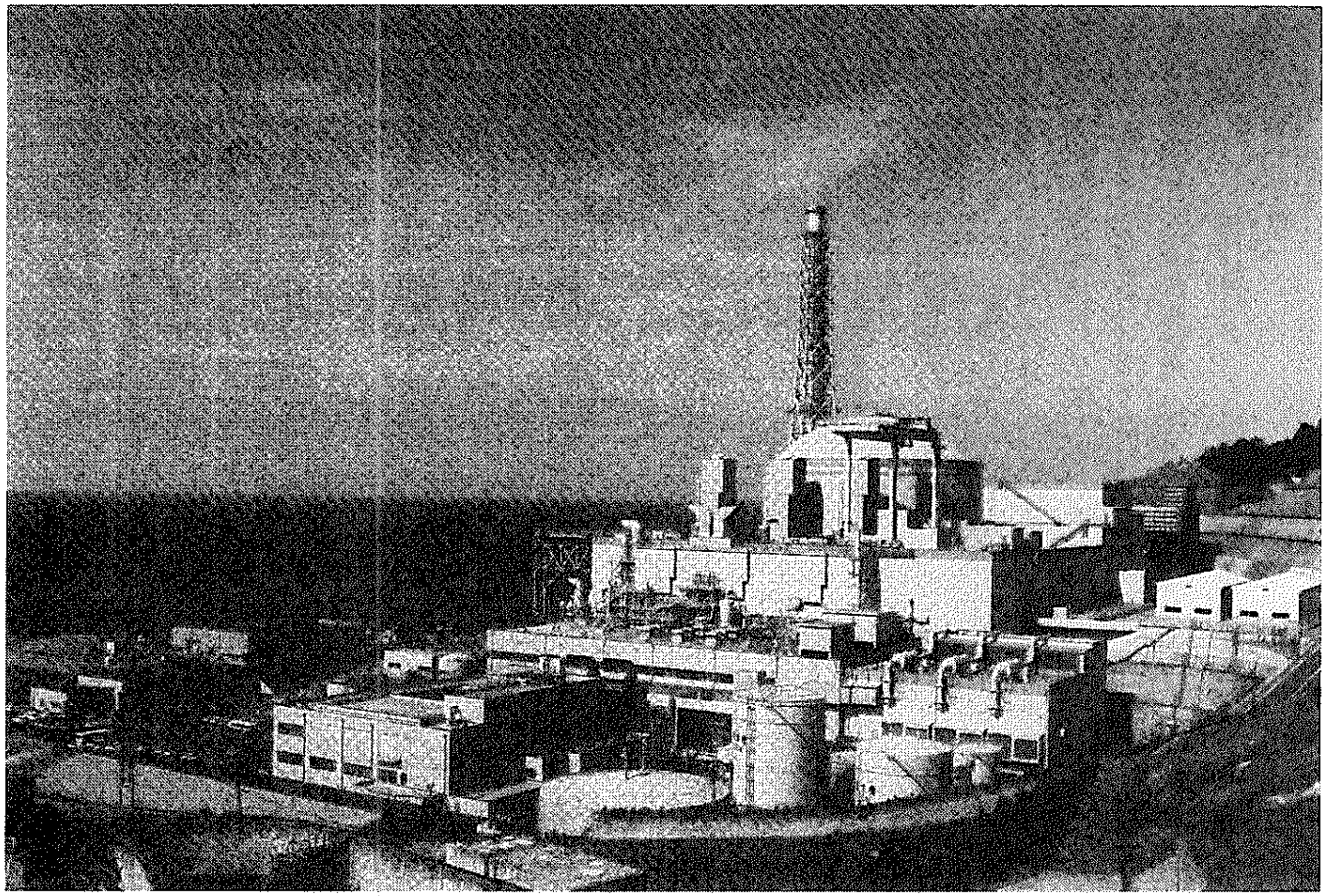


Photo1 Prototype Fast Reactor Monju (as of 1992)

2. Experimental Fast Reactor, Joyo

2.1 General Status

This report covers the activities of Joyo from April 1992 through March 1993. The operating history of JOYO is illustrated in Fig. 2.1.

From the 24th to the 27th duty cycles, operation and two special tests were carried out successfully during this time. Fig. 2.2 shows a core configuration at the 27th duty cycle as a representative core.

As of March 1993, the total operation time since the date of initial criticality in 1977 is attained approximately 50,000 hours, and the accumulated thermal output was more than $4x10^6$ MWh.

About the irradiation tests, almost irradiation tests for development fuels of the prototype reactor Monju were finished at the end of the 27th cycle.

The other many kinds of the irradiation tests, such as for development of high performance fuels and materials for the demonstration reactor, etc., are in progress.

The other hand, the special tests performed are as follows:

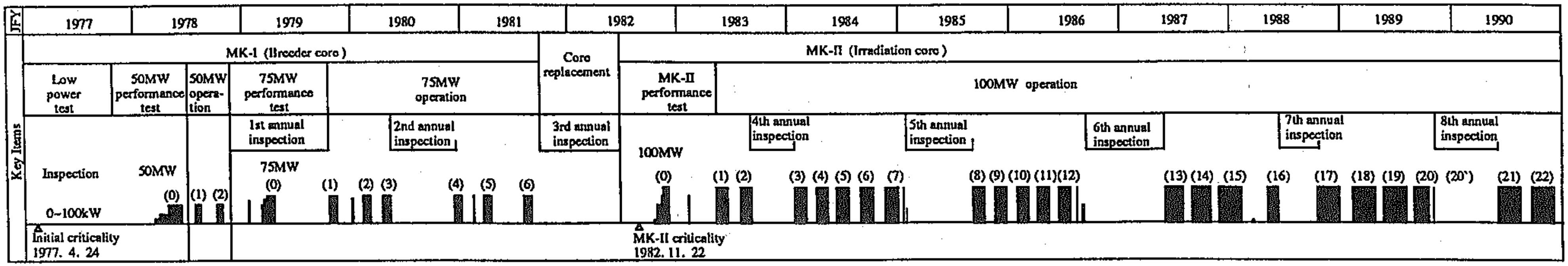
1) The second power-to-melt test for getting fuel melt data was conducted followed last year, for two days in June after the 24th cycle.
2) Trial examination of fuel failure for development technology concerned the Failed Fuel Detection and Location was carried out for five days in November after the 25th cycle.

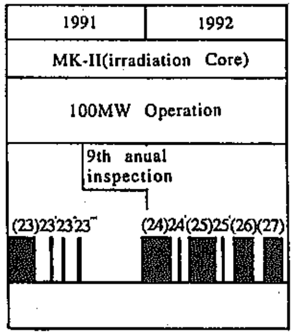
   After the 27th cycle, the 10th periodical inspection is just started from the end of March in 1993.

   A training program for Monju operators was completed by December 1992.

2.2 Joyo Improvement Program(MK-III program)

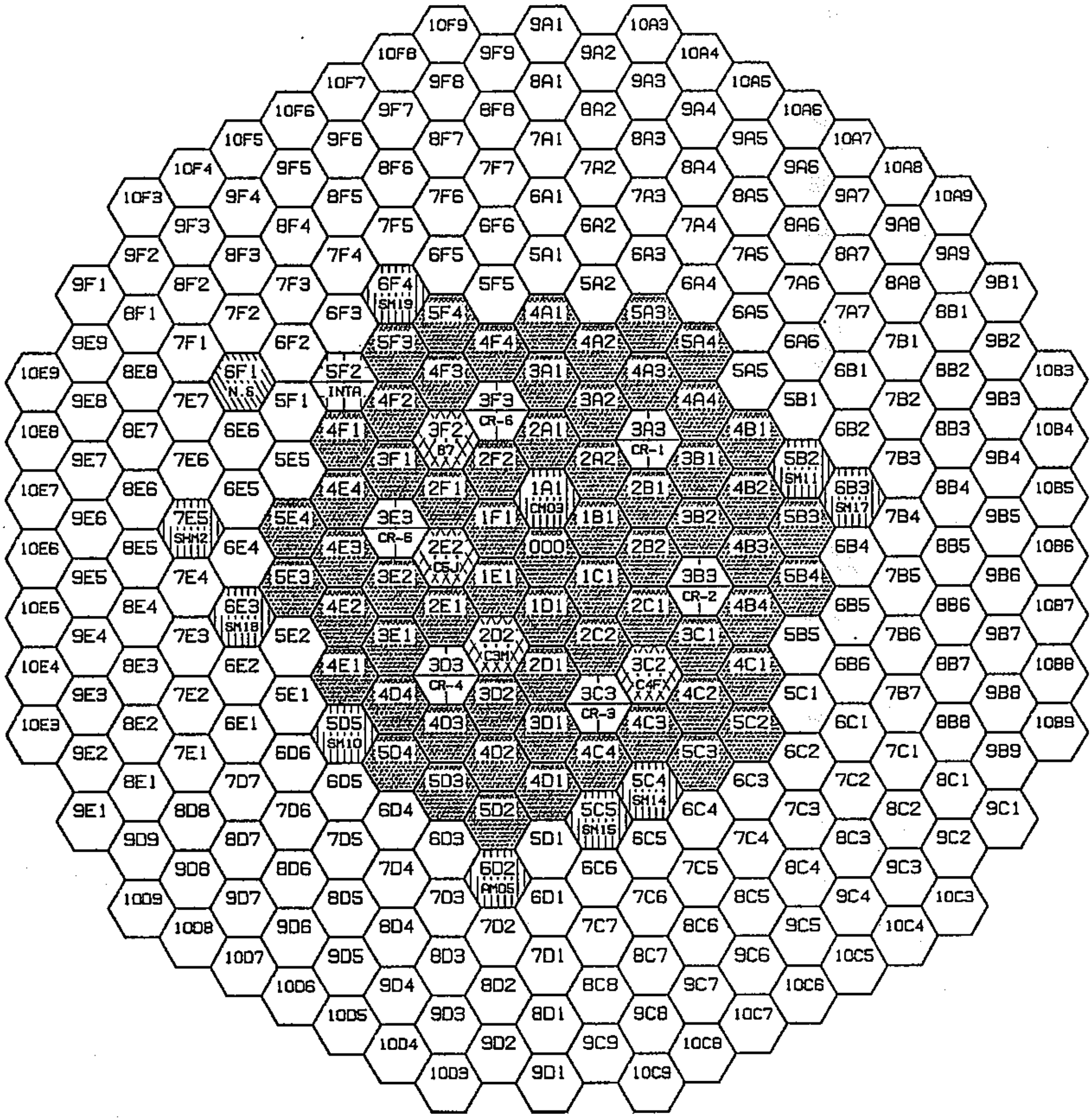
By this program, the irradiation capability of Joyo will be upgraded to four times compared with the present that ; that is, higher neutron flux is increased 30 percent, availability factor is improved from about 40 percent to about 60 percent, and number of irradiation test positions is increased approximately double. The preparations of safety review for approval of installation which will be applied to the government in next year, are in progress.





Operation hours ; 45,393hr
Accumulated
thermal output ; 3,645,999MWh
(as of March, 1992)

Fig. 2.1 Experimental Fast Reactor Joyo Operating History



Driver Fuel	64
Uninstrumented Irradiation Subassembly	3
Neutrn Source	1
Materials Irradiation Rig	11
Control Rod	6
Reflector	228

Fig. 2.2 Core Configuration during 27th Operation Cycle

3. Prototype FBR, Monju

3.1 Construction Schedule

The Monju site is located on the northside 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. 1989	Installation of Reactor Vessel
Apr. 1991	Completion of Construction
May. 1991	Start of Function Test
Dec. 1992	Start of Start-up Test
Autumn 1993	Initial Criticality

3.2 Present Status of Construction

Monju construction was 99.7 % completed as of the end of March 1993 including design, components manufacturing, and construction works at site. Major components such as the reactor vessel, IHXs, SGs, CRDs, main control consoles, and various tanks were already installed.

Major civil works were also completed.

Construction of the buildings was completed.

Sodium deliveries started in March 1991, and transfer of 1700 tons of sodium was completed in November 1991.

Equipment installation was completed in April 1991.

3.3 Function Tests

The schedule in Fig. 3.2 shows the general outline of 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, following various tests and inspections during fabrication and installation of the components in Monju.

The tests are divided into three phases:

1) Testing the fuel handling system and control rod drive mechanism in air at room temperature prior to sodium charging.
2) Tests in argon gas before loading sodium into the systems. Argon is used for preheating and heatup.
3) Testing the cooling systems, control systems, fuel handling systems etc. after sodium loading.

There were about 300 system function tests, of which 240 are specific to the FBR. Included in these are:

1) The configuration of a dummy core
2) Confirmation of the operation of fuel handling components in gases and sodium
3) Confirmation of the movement of in-service inspection and pre-service inspection equipment
4) Preheating of the system components and sodium charging
5) Leak rate measurement of the reactor containment vessel after loading the sodium

3.4 Start-up Tests

The schedule in Fig. 3.3 shows the general outline of startup tests.

Major start-up tests will be carried out after fuel loading, and will confirm and evaluate the performance of the core, each system of the plant, and the plant as a whole. They will be conducted along with various phases of criticality tests, reactor physics tests and power-up tests, somewhat similar to the phases for an LWR. Safe operation of the plant in accordance with the design will be confirmed at full power.

After the leak rate test of the reactor containment vessel, which is the final system function test, a criticality approach test will be performed as the first performance test. The minimum critical mass will be measured, replacing dummy elements with core fuel stored in the ex-vessel storage tank one by one using the fuel handling system (whose function was already verified in the course of the system function tests).

First criticality is scheduled for October 1993. After that, reactor physics, nuclear heating, power-up and load tests will be 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

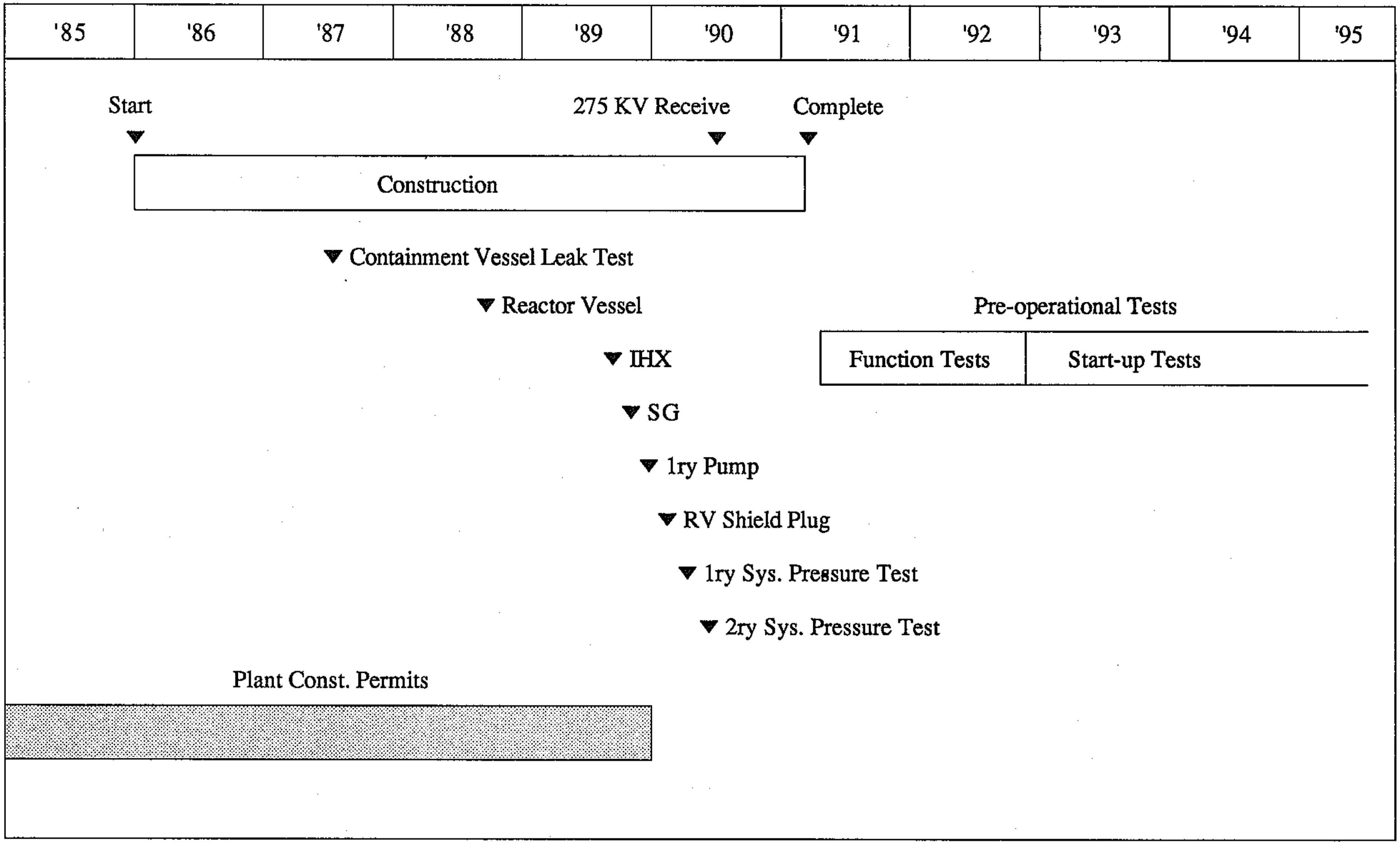


Fig. 3.1 Monju Construction & Tests Schedule

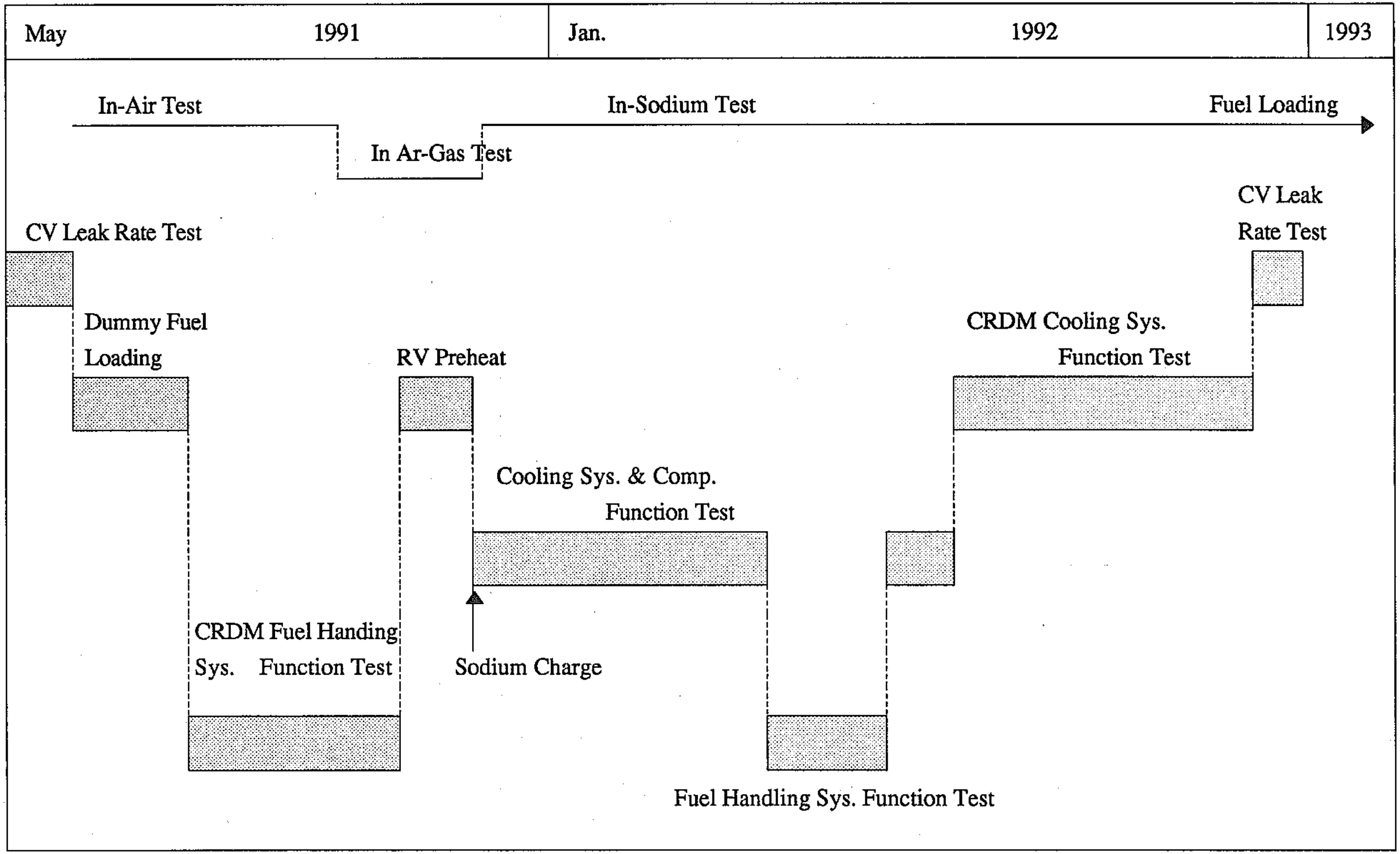


Fig. 3.2 Monju Function Tests Schedul

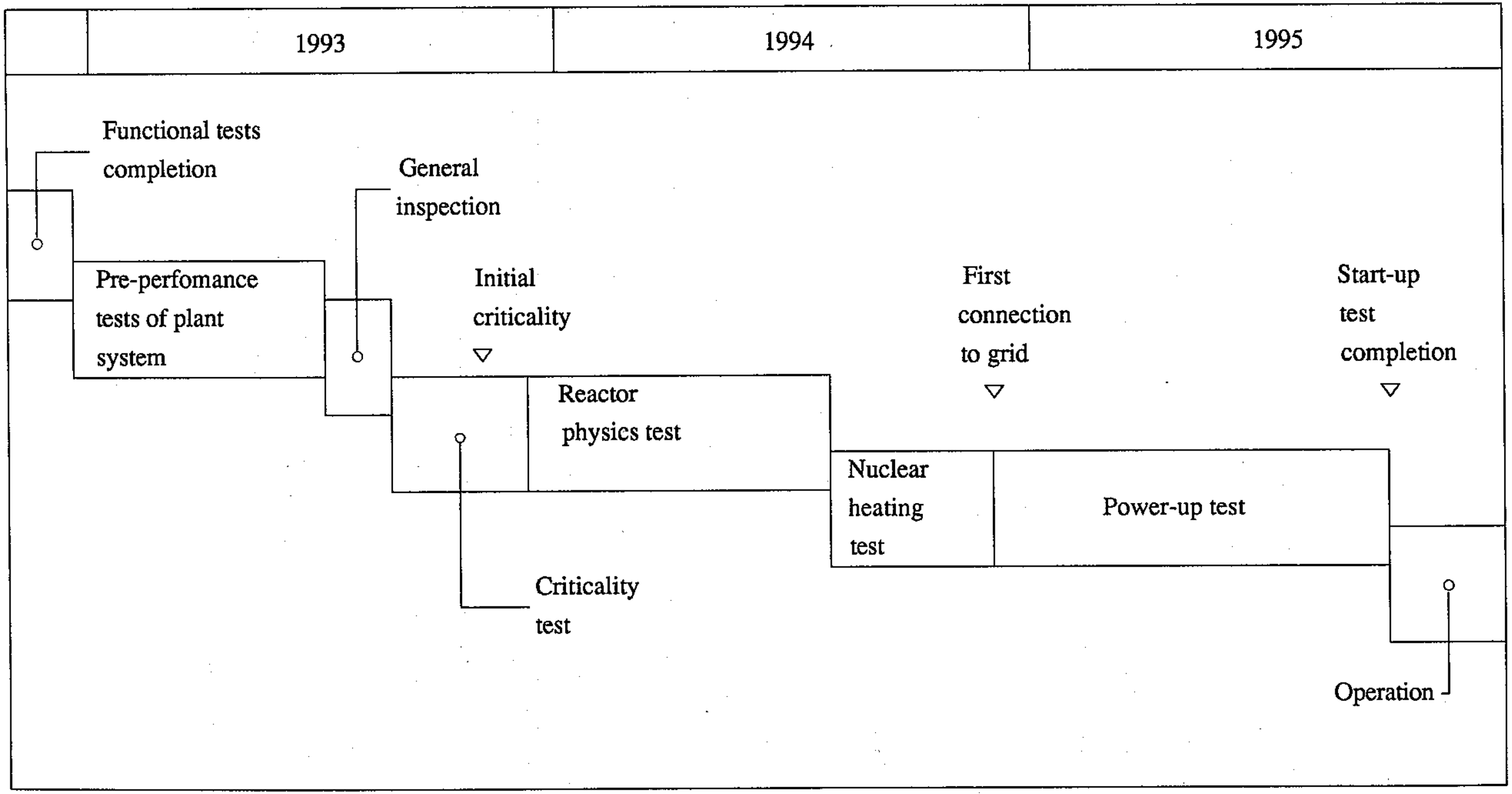


Fig. 3.3 Main schedule of Monju start-up test

4. DFBR and PNC's Design Study

4.1 Overview

The Japan Atomic Energy Commission (JAEC) issued Japanese "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 (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 in the year from 2020 to 2030 through construction of several FBRs with a step-by-step improvement of technologies and economics.

The start of construction of DFBR-1 is expected in the late 1990's in the program.

4.2 Design Study of DFBR

In July 1990, JAPC started the preliminary conceptual design study on a 600MWe top entry loop-type DFBR under the auspices of the Federation of Electric Power Companies(FEPC). In this study of the general plant design of a demonstration reactor, major systems and components in primary and secondary sodium loops of a top-entry loop-type reactor were designed. Plant safety, structural integrity and fuel integrity were evaluated, confirming that safety could be secured during various anticipated events, that integrity of fuel and component structures would be maintained. Operability, maintainability and repairability are also examined, confirming that system could be operated and maintained appropriately. The technical feasibility was also confirmed applying appropriate measures based on the results of confirmation test. In terms of economy, the economical target is confirmed to be satisfied.

From the aforementioned results, the technical feasibility of the top-entry loop-type reactor is confirmed.

4.3 PNC's Design Study

PNC has defined 10 key technical issues to be attained for commercialization of FBRs. In 1988, PNC started plant design study applying the key technologies such as reactor vessel head access piping system and performed plant construction cost evaluation.

For 1990 - 1991, design study on a 600MWe-size plant has been conducted.

In 1992, JAPC and PNC discussed on the mutual design results to improve the demonstration reactor design study.

5. Reactor Physics

5.1 Calculation Method Development

Since a Hex-Z geometry is more useful and practical than a XYZ geometry for the neutronics calculations of the FBR core, a nodal Sn method and a code based on this theory have been developed. In contrast with the conventional coarse-mesh method, this nodal Sn method considers the spatial dependency inside a node analytically, so it can reduce the truncation error arisen coarse-mesh calculation. As a result of applying this method to the Three-Dimensional Neutron Transport Benchmarks, this method agreed with Monte Carlo method under the error of 0.5 %Δk, and this result shows high calculational precision of the nodal Sn method.

In the analysis of fast reactor cores, it is important to evaluate self-shielding of resonance heavy nuclide exactly. Therefore the multiband method has been applied to the cell calculation code. Multiband method divides any energy group not into more energy groups but into total cross section band, based on the idea that self-shielding effect depends on the total cross section which is inversely proportional to neutron flux.

5.2 Cross Section Adjustment

In order to analyze burn-up properties of fast reactor core, a burn-up sensitivity analysis code SAGEP-BURN is under development. The code can treat four burn-up properties of keff, burn-up reactivity loss, number density at the end of a cycle and breeding ratio. Also, the refueling method of adjoint number density which is an important problem in multicycle burn-up sensitivity analysis has been studied.

5.3 Critical Experiment

As a part of reactor research for fast reactor, critical experiments for non-oxide fueled core have been conducted at FCA (Fast Critical Assembly) since 1989. After completing the experiments for metal fueled core, the experiments for oxide fueled reference core which should be compared with the non-oxide fueled core are in progress together with manufacturing aluminum nitride plates to be used in the next experiments.

5.4 Shielding Experiment and Analysis

PNC and United States Department of Energy started the cooperative shielding experiments program designated JASPER(Japanese-American Shielding Program for Experimental Research) at TSF(Tower Shielding Facility) of Oak Ridge National Laboratory in 1985. Following the In-vessel Fuel Storage Experiment, the Intermediate Heat Exchanger(IHX) Experiment, the Gap Streaming Experiment, the Flux Monitor Experiment and the Special Materials Experiment have been conducted, and all of the scheduled JASPER experiments have been successfully completed.

Analyses for the experimental results are in progress. The completion of the analyses are scheduled in 1995.

6. Systems and Components

6.1 Control Rod Drive Mechanism

Research on the self-actuated shutdown system (SASS) for DFBR is in progress. Based on the partial model test results in-air and in-sodium, a prototypical test using an actual reactor is being planned.

6.2 In-service Inspection Equipment

Full size model tests for the Monju steam generator tubes and primary pipes were completed in November 1991. For the Monju reactor vessel, full size model tests were completed in March 1992. New techniques, such as remote inspection technique using optical fiber scopes for reactor vessels, electromagnetic acoustic and ultrasonic transducers for high temperature use on reactor vessels, ultrasonic transducers without couplant for primary piping systems are adopted after a series of performance tests at O-arai Engineering Center (OEC).

6.3 Steam Generator

PNC is conducting a conceptual design study for a future FBR plant having steam generators in the primary heat transport system. In support of developing this concept, the studies on a double-wall tube steam generator have been in progress to evaluate the leak detectability, failure probability and the heat transfer characteristic on the double-wall tubes. A 1MW double-wall tube steam generator model has been operated at full load since November 1991.

6.4 Process Instrumentation

The calibration of fuel subassembly outlet flow meter of Monju is in progress.

An out-pile calibration of EMF for Joyo Instrumented Test Assembly was performed in sodium and a calibration curve was obtained with sufficient accuracy.

For an out-pile calibration of EMF to verify Monju core flow allocation, production of the EMF and conversion of sodium test loop are in progress. The test was performed in 1992.

For an EMP for Joyo subassembly test rig, coil temperature characteristic test of 1/4 scale model was performed. And 1/2 scale model EMP for performance test is under way in 1993.

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 for transient state and fuel failure state have been improved since 1984, with the data of operational reliability tests in EBR-II, etc. The modeling of cesium migration has been developed since 1986 to evaluate the fuel performance of an axial heterogeneous core fuel.

Development of fuel performance code for metal, carbide and nitride fuel is in progress.

7.3 Core Materials

SUS316 stainless steel (Monju core material) irradiated over 2.5×10^{23} n/cm^2 (E>0.1 MeV) showed excellent swelling resistance by less than 1.0% volume swelling under the Monju irradiation condition. 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 were developed since 1984. One is a high strength ferritic/martensitic steel which is considered as a good material suitable for a wrapper tube and the other is an oxide dispersion strengthened ferritic steel (ODS). The tubing technology for ODS cladding has progressed by hot working process.

Sodium environmental tests of core materials including hard facing materials for fuel assembly pads, out-of-reactor tests to evaluate bundle to duct interaction for large assembly were also conducted.

7.4 Irradiation Experiments

1) Joyo

Advanced austenitic stainless steel cladding fuel pins have been irradiated. On fuel subassembly using CEA cladding tubes has also been irradiated since August, 1988. Several irradiation tests for a large diameter fuel pin have started. The second power-to-melt test was conducted.

2) Foreign Reactors

Phase-I program of operational reliability testing of FBR fuel in EBR-II was completed and Phase-II program is in progress.

The irradiation of fuel subassemblies was completed for SUS316 and the irradiation of advanced austenitic stainless steel cladding fuel pins has been continued in FFTF since November, 1987.

7.5 Development of Advanced Fuels

Study of advanced fuels (nitride, metal, carbide) has been conducted in technological evaluation of the availability.

Mixed carbide fuel pins have been irradiated since 1983 using the thermal reactors JRR-2 and JMTR of JAERI.

Carbide and nitride fuel irradiation test is planned in Joyo.

7.6 Post Irradiation Examination

Construction of PIE facility is in progress to begin the examination of Monju fuel subassembly and so on from 1995.

8. Structural Design and Materials

8.1 Development of Structural Design Method

1) FINAS nonlinear structural analysis program

Effort to enhance the capability of the general purpose nonlinear structural analysis program FINAS has been continued since 1986, particularly with respect to inelastic constitutive models for cyclic plasticity, large deformation/buckling analysis methods, dynamic fluid-structure interaction analysis methods, and contact problem solution algorithms. FINAS is currently used by many research engineers at about 30 sites including fabricators and universities. The version V11.0 was released in 1989. The new version, V12.0, which has more advanced and flexible capabilities in terms of inelastic constitutive models, user defined elements, material data library, dynamic condensation and surface contact analysis, was released in January 1993.

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 design method, which is based on the concept of a generalized elastic follow-up model, is being developed.

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 criteria

A ratchetting criteria for multiaxial stress states, which are not provided explicitly in the design guide, are being investigated.

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 (SPTT and STST)

Structural discontinuity model tests to investigate crack initiation and propagation behavior was completed by the end of 1990 and their evaluation is under way.

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

Tests with a vessel model, piping bellows models and two thermal stress mitigation models were completed. Test with an welded vessel model is in progress.

3) Plastic buckling tests (SCFT)

Buckling tests on cylindrical shells subjected to shear loads are being conducted.

4) Inelastic behavior tests (BHAT)

Inelastic behavior tests of simple structures such as notched plates and three-bar structure are being performed to verify advanced inelastic constitutive methods.

8.3 Seismic Test and Analysis

A new shaking table DST, with a size of 2.5m x 3m, a loading capacity of 10 tons and a maximum acceleration of 3G, was constructed at OEC in 1990. Sloshing and fluid-structure interaction tests for horizontal and vertical excitations are being conducted. Conceptual and feasibility study of a vertical seismic isolation system for FBR components is started.

8.4 Fracture Mechanics and Structural Integrity Assessment

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

Computer codes being developed at PNC include CANIS-J for calculation of fracture mechanics parameters, CANIS-G for simplified crack propagation analysis and CANIS-P for probabilistic fracture mechanics analysis. Crack propagation tests of pre flawed structural elements, like pipes, plates, and elbows which are subjected to mechanical loadings, have been performed since 1987, to verify validity and applicability of the computer codes in non-creep and creep regions. Crack propagation tests of a cylinder with circumferential and axial flaws are being conducted by the Air-cooling Thermal Transient Test Facility (ATTF).

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) and the step-2 program (1988-1990) were already completed. The step-3 program (1991-1993) are currently underway.

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

1) Tests in Air

The present Capella step 3 program includes following subjects;

- Improvement of Monju design method on creep-fatigue life, strength of weldment, inelastic constitutive equations considering the application of new materials (modified 9Cr-1Mo steel and FBR grade 316 stainless steel)
- Application of elevated temperature fracture mechanics

- Development of the material strength standard for modified 9Cr-1Mo steel and FBR grade 316 stainless steel

  A tentative 1989' version of the Material Strength Standard including the rules for modified 9Cr-1Mo steel and FBR grade 316 stainless steel(316FR), and damage mechanism of creep-fatigue failure was examined in detail using test results in 1992.

2) Test in Sodium and Water/steam

A new series of sodium environmental effect tests, according to the Capella program were carried out on modified 9Cr-1Mo steel and FBR grade 316 stainless steel(316FR). Corrosion and mass transfer, carbon transfer tests have been already finished. Mechanical strength (tensile, creep, fatigue, creep fatigue) tests and nitrogen transfer of 316FR (nitrogen controlled) in sodium are still continued in the program.

3) Tests in Irradiation Environments

Surveillance tests for the Class 1 components of Joyo have been 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, Inconel 718 were irradiated in Joyo using SMIR (Structural Materials Irradiation Rig).

Another test for DFBR has been already conducted to clear the relationship between creep rupture strength and metallurgical variables such as chemical composition, grain size and production process. Now new R&D program Spica step 2 was started with emphasis on 316 FR.

4) Data Processing System

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

Entry data in the 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 Thermohydraulics Related to Reactor Design and Safety

Thermohydraulic studies have been conducted for evaluating the physical phenomena and integrity of the reactor fuel elements and the structures in primary system during the normal operation, scram transients, and 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). Major emphasis has been placed on evaluation of the thermal striping phenomena and free surface behavior under normal operation condition, on evaluation of the sodium boiling phenomena under accidental conditions, and on clarification of the mixed to natural convection phenomena. The subjects covered are: (1) experimental studies for thermohydraulics of single and inter-fuel subassemblies, and plenum-channel thermohydraulic interactions in the mixed convection regime, (2) code development and validation for the subassembly and reactor core heat transfer analysis (ASFRE, SABENA), thermohydraulic analysis in plena (AQUA, SPLASH, DINUS), and the plant system dynamics analysis(SSC).

The ability of LMFBRs to remove decay heat by natural circulation is one of the important safety features of the current heat transport system design. Especially, understanding of the inter-subassembly heat transfer, the intra- and inter-subassembly flow redistribution, the inter-wrapper natural convection, and the cold fluid penetration into the subassembly from the upper plenum is essential for an evaluation of the natural circulation decay heat removal capability. To support the design of passive decay heat removal systems, experimental studies are in progress using water and sodium as working fluids.

A fundamental water experiment has been done to investigate the cold fluid penetration phenomena. The test section has slab geometry. A vertical flow channel is connected at the top to an upper plenum which has a vertical cold wall simulating the dipped cooler. The hot water was supplied into the upper plenum from the heated flow channel and exits at the top of the plenum. The onset condition of the penetration flow was obtained as an empirical formula of Re and Gr number. It has been found that AQUA code predicts the penetration flow generally and that turbulence model constants of AQUA should be modified for the situation of low Re number flow.

A sodium experiments has been carried out with three parallel bundles model for investigating the intra- and inter-subassembly flow redistribution and the cold fluid penetration phenomena. The test section, which consists of three parallel pin bundles connected to the upper and lower plena, was installed in the CCTL. The first phase of experiments, which is a series of steady state experiments, has been continued. The effects of the main subassembly's transverse temperature distribution cased by heat transfer (heating/cooling) across the wrapper tube wall were investigated in order to develop a procedure to estimate the transverse peak temperature and inter-subassembly heat transfer. In addition, numerical simulations of the test cases were carried out using the multi-dimensional analysis code AQUA. The resultant calculated and experimental transverse temperature distributions in the subassemblies showed good agreement.

An integral sodium experiment has been carried out with a partial core model comprised of seven subassemblies, inter-wrapper qaps, an upper plenum and a dipped cooler. The first series of the experiments showed that cold sodium provided by the dipped cooler could reduce hot spot temperature in the pin bundle mainly via the inter-wrapper qaps.

Parallel channel sodium boiling experiments in the mixed convection range were completed using PLANDTL facility during the reporting period.

Efforts on thermohydraulics and safety analysis code development and validation are continued for the subchannel analysis codes ASFRE and SABENA, three dimensional thrmohaydraulic analysis code AQUA, and the plant system dynamics analysis code SSC. Furthermore, new computer codes SPLASH for fluid dynamics analysis using finite element method and DINUS-3 for direct numerical simulation have been developed.

ASFRE is a subchannel code which calculates a fuel subassembly's transient single-phase fluid flow and temperature distributions. The code contains a distributed resistance model which accounts for the wire-wrap spacers, and also has the capability of calculating the velocity and temperature fields in the presence of a subassembly partial blockage. The code validation study is to continued using the PLANDTL-CCTL experiment data. SABENA code validation was carried out by simulating the parallel channel sodium boiling experiment. The loop-type version of SSC (SSC-L) was improved so that dynamics reactivity feedback effects are simulated. Passive safety analysis in accidents without scram of FBRs was performed during this reporting period. Also, SSC was utilized for the accident of the demonstration FBR and natural circulation test analysis of Monju. AQUA code has been utilized for the design of thermohydraulic measurement system and transient system and transient analysis of Monju. Thermal striping experiments using water were analyzed to check the general performance of the newly developed DINUS-3 code. The codes are to be used for the thermal striping evaluation of the demonstration FBR. SPLASH can deal with the free surface motion in the reactor plenum. It is to be applied to the sloshing and qas entrainment analysis of demonstration FBR.

9.2 Degraded Core Research

The degraded core research addresses the fuel subassembly failure propagation in local fault accidents and the in-vessel physical processes of FBR core disruptive accidents.

The local fault studies focused on the SCARABEE in-pile test analysis and reactor application code development/validations. PNC participated in three shots of SCARABEE in-pile experimental program, and has conducted the data analysis. The two tests, BE+3 and PI-A, have been analyzed to simulate molten pool thermal behaviors and thermal loading of the hexcan wall using the computer codes SABENA and FUMES, respectively a subchannel sodium boiling analysis code and a molten fuel thermal behavior analysis code. The last test, PV-A, has been analyzed to simulate molten fuel behaviors penetrating into an intact subassembly using the computer codes TAC and SCION, respectively a general heat conduction analysis code and a molten material behavior analysis code.

The out-of-pile experiments using the MELT-II facility are in progress. A series of experiments to investigate the erosion behavior of the solid plate by the high temperature liquid jet was completed. A new series of experiments to study the thermohydrailic interactions between a molten materials jet and water is in progress. In the next step, it is planned to perform similar experiments with sodium, To investigate the boiling pool phenomena, a new facility named "POOL" was constructed. The POOL facility has a 50u/c microwave oven to simulate the dynamics of volumetrically heated boiling pool. Preliminary and calibration experiments are underway.

The CABRI-2 in-pile experimental program, jointly conducted with the European partners, is in its final stage. International joint evaluation of the experimental data is in progress. The next CABRI-FAST program with higher burnup fuel has initiated. The whole core accident analysis code development and applications continued for the initiating phase using SAS3D, PAPAS-2S and SAS4A, and for the core disruption phase using SIMMER. The results and knowledge from the in-pile experiments are incorporated into model improvement of the SAS4A code. The first version of the SIMMER-III code was completed, which encompasses all the fluid-dynamics models. Presently, the code development and assessment program is participated by the European partners.

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.

In the sodium fire study a three-dimensional code, SOLFAS, is under development to analyze the thermochemical processes of sodium fire and aerosol behavior. In parallel with this, the conceptual design of a refractory ceramic liner was completed which can withstand sodium leak accident.

In the source term study, several experiments using FP simulants are in progress in order to investigate the physical and chemical forms of FPs and the attenuation factor of FP bubble in liquid sodium system after FPs being released from fuel. For the containment analysis, CONTAIN-LMR has continuously been improved by collaboration with other U.S. and European users. A series of hydrogen combustion tests is under way, which is intended to quantify combustion conditions in a sodium aerosol atmosphere.

9.4 Steam Generator Safety Research

Current steam generator (SG) safety researches consist of two major activities, i.e., the improvement of the evaluation method for large demonstration plant SGs and the development of analytical models for future commercial LMFBRs which will use a double wall tube SG in a primary heat transport system instead of a conventional one in an intermediate heat transport system. Overheating failure mechanism of heat transfer tubes are investigated using 3-D structural analysis code for the former item and an analytical model, HYBAC, of hydrogen bubble behavior has been developed for the latter one.

9.5 Research on Probabilistic Safety Assessment

PNC has been performing the research on Probabilistic Safety Assessment (PSA) for nearly 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 based on the Monju plant information 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 conversion of the developed code network into a PC-based system, improvement in a Monte Calro method phased mission analysis program, and development of a Living PSA System(LIPSAS). LIPSAS consists of three modules; PSA update, risk monitor and risk management modules. The prototype version of this system has been completed and 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 system which is able to quantify the consequence of core damage quickly with the expert system which models accident phenomena based on the experience of full scope PSA.

Efforts are being made to develop LMFBR component reliability data based on CREDO(Centralized Reliability Data Organization), a cooperative project between PNC and the USDOE, with this work to continue till 1992 in the LMFBR-related facilities of both countries. PNC also has been developing a new CREDO data base system on a engineering work station which uses a commercial relational data base system. As a part of the data analysis preliminary component aging failure and common cause failure analyses were carried out for valves and mechanical pumps. Additionally, risk-related data on Japanese energy production systems such as fossil-fueled power plants and nuclear power plants have been collected and preliminary comparisons were made between various levels of risk from alternative power sources.

Level-1 and 2 tasks with respect to internal events were completed. Current efforts are focusing on evaluation of external events and a preliminary PSA application to large LMFBR plant.

In seismic event analysis a detailed fragility evaluation was conducted based on the probabilistic response analyses using the SMACS code for the important structures and components such as reactor vessel, primary heat transport system and so on. Several event trees were developed to delineate seismic event sequences. Component failures from seismic events were incorporated into fault tree models using Boolean transformation equation techniques. Then the seismic-induced core damage frequency is being quantified based on the fragility data and the developed event tree and fault tree models. Space dependent failures(external impact on the structures and components) such as coolant effluents and general fires were evaluated and they were found to be less contributor to core damage in comparison with the internal events.

Preliminary application of PSA to a large LMFBR is under way to provide basic information in developing safety design and evaluation policy. The fifteen initiating event categories were identified and the associated event trees were developed. The reactor shutdown system and decay heat removal system were modeled based on fault trees and phased-mission analysis approach. Also the study on classification of safety function importance for an LMFBR is ongoing. Reliability goal of PS(preventing system) and MS(mitigating system) has been examined based on the insights obtained from the PSA application.

In parallel level-2 PSA tasks(consequence analysis) are under way. The quantitative results obtained for the medium-sized LMFBR has been examined and the analysis of key core damage sequences was started for a large LMFBR. Regarding the in-vessel physical process preliminary analysis of ULOF(unprotected loss of flow) event was conducted to compare with the behavior of medium-sized core. Also the dynamic behavior of molten core pool was studied based on the analysis of transition phase. The insights on adequate suppression of energetics were obtained. Regarding the ex-vessel physical process the various parametric analyses of key event sequences in containment were performed to comprehend the sensitivity of phenomenological and design-related parameters such as containment volume, design pressure, leakage rate, amount of ejected sodium, etc..

10. Fuel Cycle

10.1 Mox Fuel Fabrication

1) Construction and Fuel Fabrication

R&D on fabrication of uranium-plutonium mixed oxide (MOX) fuel have 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 ton MOX/year) started in July 1982. It was designed to develop fuel fabrication technologies 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. The PFPF is currently fabricating fuels for Monju.

PNC is planning to construct a new ATR line (40 ton MOX/year) at PFPF so as to produce fuels for the ATR demonstration reactor before its startup.

The present Japanese suppliers of uranium fuel and PNC will also cooperate to make increased use of PFPF to manufacture MOX fuel, for large scale demonstration of plutonium use in LWRs in Japan.

About 119 tons of MOX fuel have been fabricated by the end of December 1992.

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 equipment including material transferring system were designed and manufactured so as to realize the fully automated operation except hand-on maintenance.

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

10.2 Plutonium and Uranium Conversion

PNC developed a co-conversion technology using 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 December 1992, it produced about 8.1 ton of MOX powder using about 3.3 ton of plutonium. The converted MOX powder were transported to PFFF and PFPF, in addition to about 1.8 ton of MOX powder processed at another small scale facility, and are being used for fabrication of MOX fuel for Fugen, Joyo and Monju.

Since recovered uranium through reprocessing of spent fuel has generally higher U235 concentration compared to natural uranium, our country has decided 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..

In preparation for a large scale recovered uranium conversion facility, various technical development and design studies are now under way to establish the continuous production technology by the MH method.

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, on the basis of accumulated experience in the Tokai reprocessing plant for LWR fuels.

PNC is also designing Recycle Equipment Test Facility (RETF) to conduct engineering scale equipment tests under hot conditions in order to enhance the technology and economical efficiency.

PNC and USDOE entered into a joint collaboration agreement where the US shares in the R&D effort.

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 now tested at ORNL. The design is based on the past experience accumulated at ORNL and the criticality control requirement set by PNC.

2) Chemical Separation Process

Significant information on pulsed-column technology has been obtained through engineering scale uranium and plutonium tests. Now major effort of solvent extraction contactor development is paid on centrifugal contactor. Developmental efforts at ORNL and PNC merged and the design of the prototype contactor for RETF has been completed in joint effort.

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

Development of remote system technology to establish remote maintenance concept with rack system is now underway. Advanced servo manipulator, roll-in type rack, remote connector bank, and remote sampling system have been developed.

Materials of process equipment and on-line analytical system are also under development.

4) Hot Tests at CPF

Irradiated fuel from Joyo, Phenix, and DFR with burnup up to 94000 MWD/T have been reprocessed at CPF. Through these hot tests, information of dissolution characteristics dependent on many factors and nuclides behavior in the off-gas have been obtained.

11.2 Plant Design of Recycling Facilities

1) Recycle Equipment Test Facility (RETF)

Verification of high availability and economical prospects of FBR fuel recycling 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 is now designing Recycle Equipment Test Facility (RETF) to provide a test bed for advanced 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).

RETF is scheduled to start hot tests in the late 1990's.

2) FBR Fuel Recycling Pilot Plant

The purpose of the FBR Fuel Recycling Pilot Plant is to demonstrate the whole plant availability and to evaluate the economical efficiency of FBR fuel reprocessing.