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A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

April 1992

Power Reactor and Nuclear Fuel Development Corporation

This document was prepared for the IAEA/IWGFR 25th Annual Meeting (April 27 ~ 30,1992,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 43,500 hours operation in total by the end of 1991, since its first criticality in 1977.

3) On the prototype reactor "Monju", 97.6% of construction works has already been completed and the function tests are in progress aiming at the initial criticality by the end of FY 1992.(Photo)

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 Mitsubisi Heavy Industries, for selection of the basic specifications of DFBR.

The related reseach 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 the JAPAC, 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 demonstration reactor.

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 the 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 maintenace 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 managerment, 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.

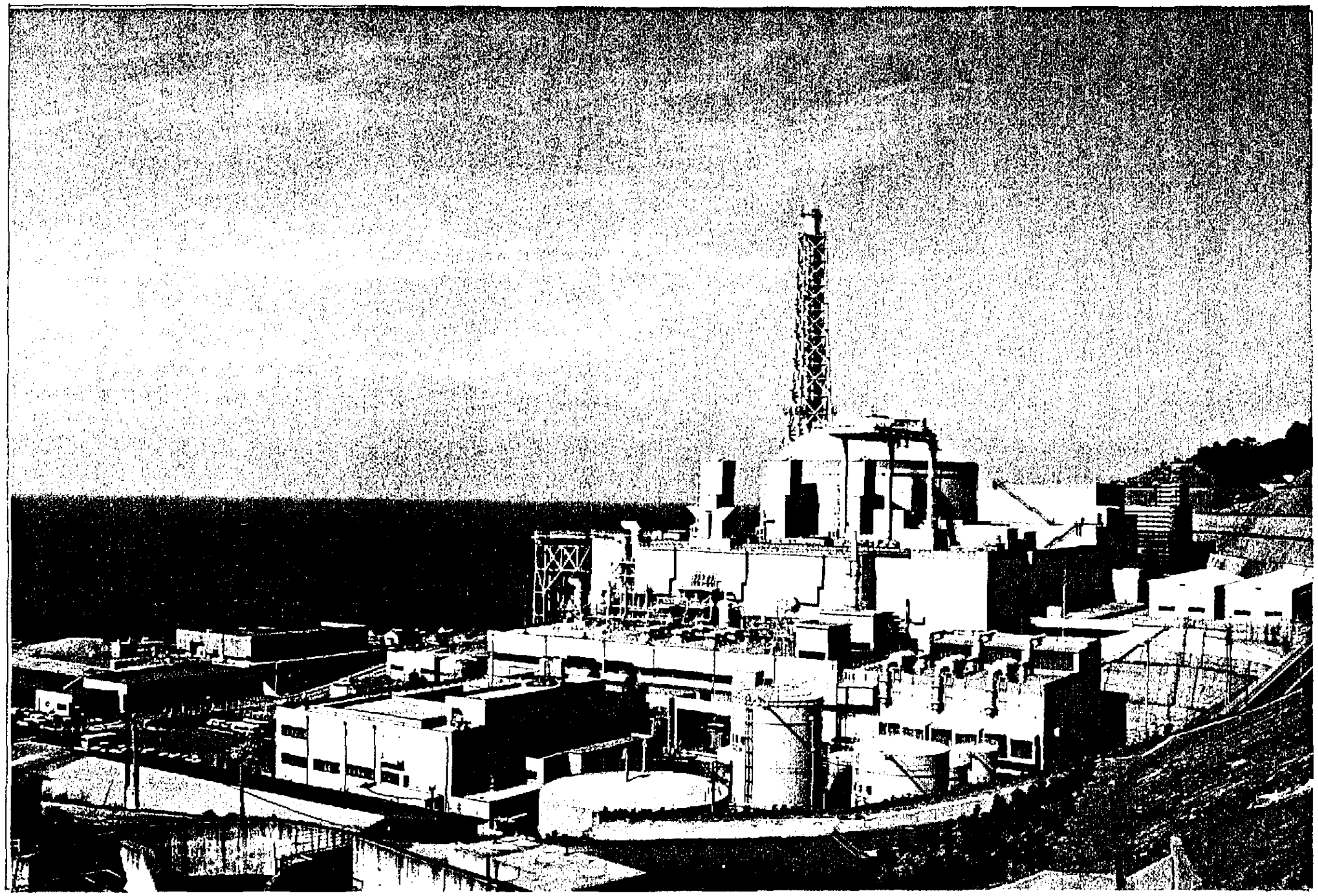


Photo. Prototype Fast Reactor "Monju" (as of 1992)

2. Experimental Fast Reactor, Joyo

2.1 General Status

This report covers the activities of Joyo from April 1991 through March 1992. The operating history of Joyo is illustrated in Fig.2.1.

The 23rd duty cycle was carried out successfully from April 6 to May 31, 1991, and the total operation time since the date of initial criticality in 1977 is more than 43,500 hrs. Fig.2.2 shows a core configuration at the 23rd duty cycles operation. After that, three kinds of the tests were performed from June through September in 1991, i.e.

1) Power to Melt Test for getting the melt data at fuels center was conducted for two days in June,

2) Detail Measuring Test of Control Rod worth was from June 29 to July 5 for the reactor core design of the MK- III program which is mentioned later, and

3) Instrumented Test Assembly Irradiation Test for observing the irradiation behavior of large diameter fuel pins was conducted from 2 to 10 in september.

Moreover, the 9th periodical inspection began from September 11 will be finished March 27, 1992. In this inspection, one control rod in the third row of the core was moved to the fifth row for preparation of upgrade from the MK- II core to the MK-III core. After the inspection, the 24th duty cycles will be started from March 28, 1992.

An irradiation test for obtaining the high burnup data of fuel pin bundle for the prototype reactor Monju (which was begun in February 1987) was completed at the end of the 23rd cycles. 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 construction of the second spent fuel storage facility was completed by December 1991. It will be available from April 1992.

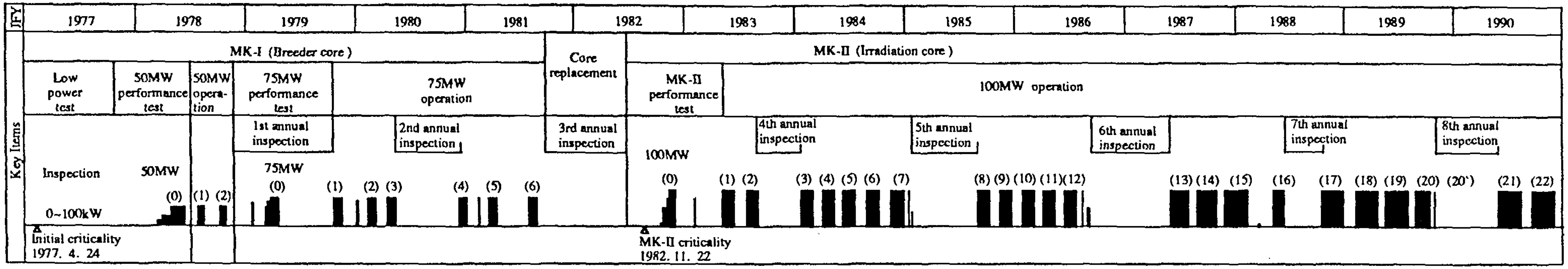
A training program for Monju operators has been conducting since 1990.

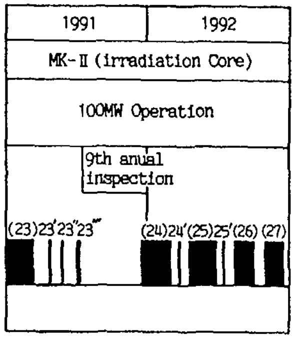
2.2 Joyo Improvement Program (Mark-III program)

The purpose of this program is to upgrade the irradiation capability of Joyo in order to accelerate the rate of the fuels development for the FBR. Higher neutron flux, modification of heat transfer system, increased number of irradiation positions, introduction of advanced irradiation technology, and increased availability factor, are being evaluated.

The principal results obtained are as follows:

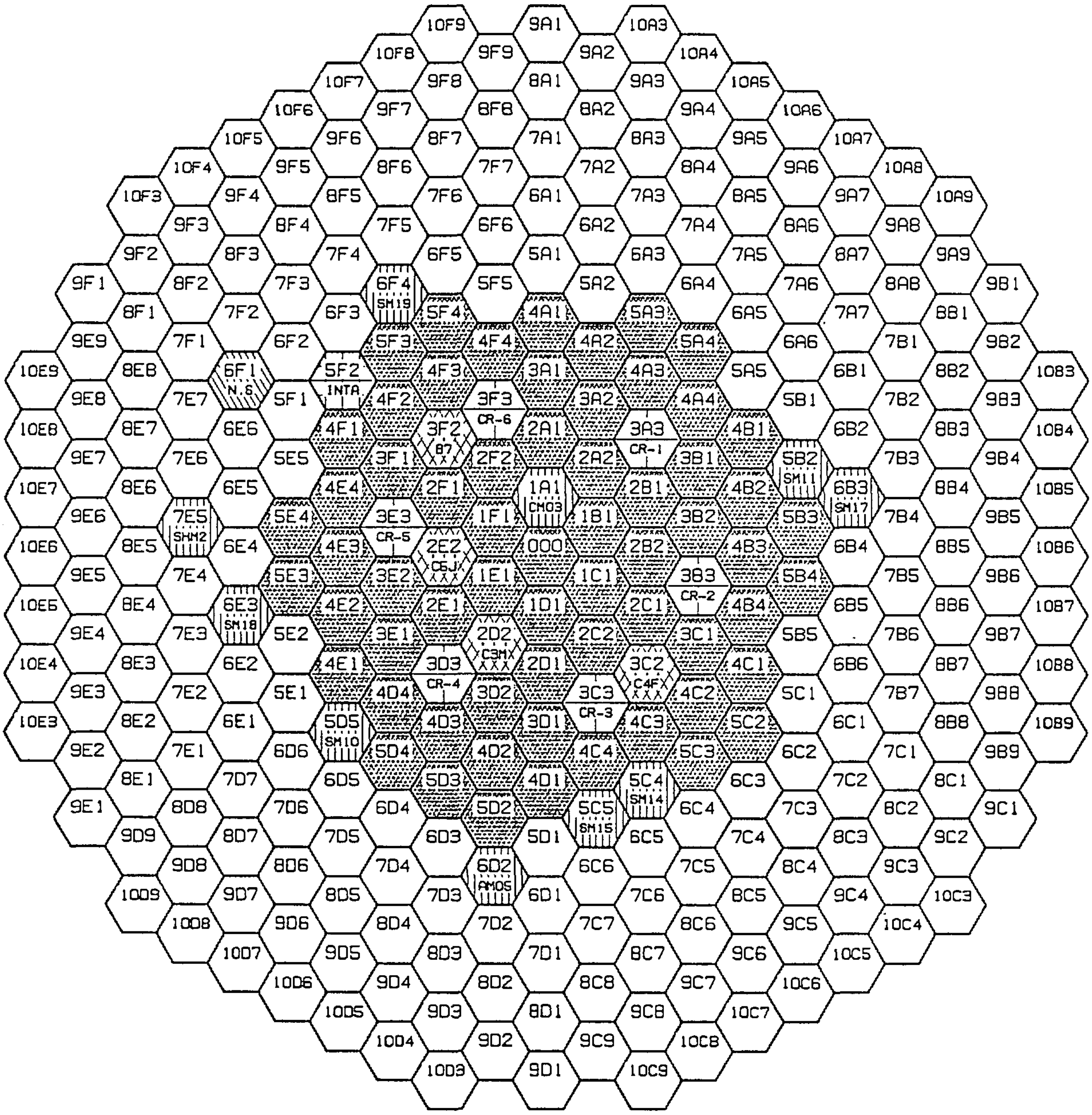
- It was confirmed that the design specifications of the MK-III core, in which the neutron fulx will be increased 30 per cent compared with the present core, could be achieved with maintaining the reactor safety.
- To raise the reactor power to 140 MWt (40 % up), the plant heat balance was determined, such as the reactor outlet/inlet (present value) temperature is 500 ℃/350℃ (500 ℃/370℃), the dump heat exchanger temperature range is 470 ℃/300 ℃ (470 ℃/340℃), and the primary and secondary coolant flow rates are 122 % and 107% , respectively.
- Fabrication of a new irradiation test equipment which can control the temperature of test specimens whithin ± 4 ℃ was begun so as to be available it from the 28th duty cycle in 1993.





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	63
Uninstrumented Irradiation Subassembly	4
INTA	1
Neutron Source	1
Materials Irradiation Rig	11
Control Rod	6
Reflector	227

Fig.2.2 Core Configuration during 23rd 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

Spring 1993 Initial Criticality

3.2 Present Status of Construction

Monju construction was 97.6 % completed as of the end of December 1991 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 are already installed.

Major civil works are also completed (about 99%).

Construction of the buildings is currently 98 % completed except for construction of solid waste storage facility.

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 Program

Function tests program is under extensive development by PNC, JAPC and fabricators. The schedule in Fig 3-2 shows the general outline which is subjected to further modification depending on the determination of detailed.

Function tests started in May 1991.

Preheating test of the reactor's primary and secondary cooling systems were carried out, followed by the continuous testing of sodium transfer to these systems.

In the fuel handling and storage facilities, test transfer of dummy fuel assemblies were carried out in air.

Sodium charging tests of the reactor's primary cooling system were completed in December 1991.

Table 3-1 Principal Monju Plant Design Characteristics

Reactor Type		Sodium cooling loop-type
Thermal Power		714 MW
Gross Electrical Power		280 MW
Core	Equivalent Diameter	1,790 mm
	Height	930 mm
	Volume	2,335 lit.
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		80,000 MWD/T
Cladding Material		SUS316
Cladding Outside Diameter/Thickness		6.5/0.47 mm
Permissible Cladding Temperature (middle of thickness)		675 ℃
Power Density		283KW/lit.
Blanket Thickness		Upper 300 mm Lower 350 mm Radial 300 mm
Breeding Ratio		1.2
Reactor in/out Sodium Temperature		397/505 ℃
Secondary Sodium Temperature (IHX inlet/IHX outlet)		325/505 ℃
Reactor Vessel (height/diameter)		17.8/7.1 m
Number of Loops		3
Pump Position (Primary and Secondary Loop)		Cold Leg
Type of Steam Genefator		Helical Coil, once-thorugh Unit Type
Steam Pressure (Turbine Inlet)		127 kg/cm² g
Steam Temperature(Turbine Inlet)		483 ℃
Refueling System		Single Rotating Plug with Fixed Arm FHM
Refueling Interval		6 Months

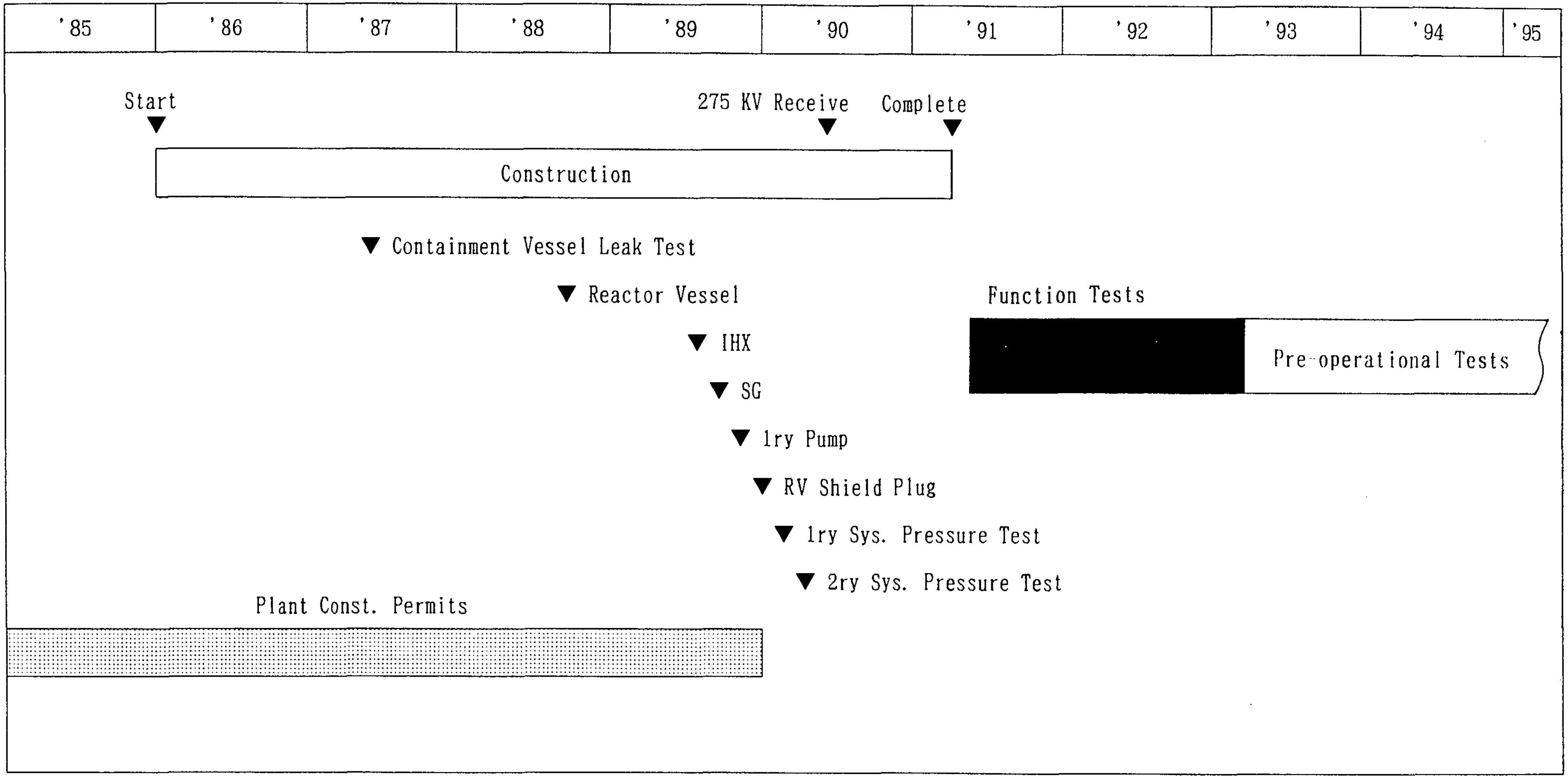


Fig.3-1 Monju Construction & Tests Schedule

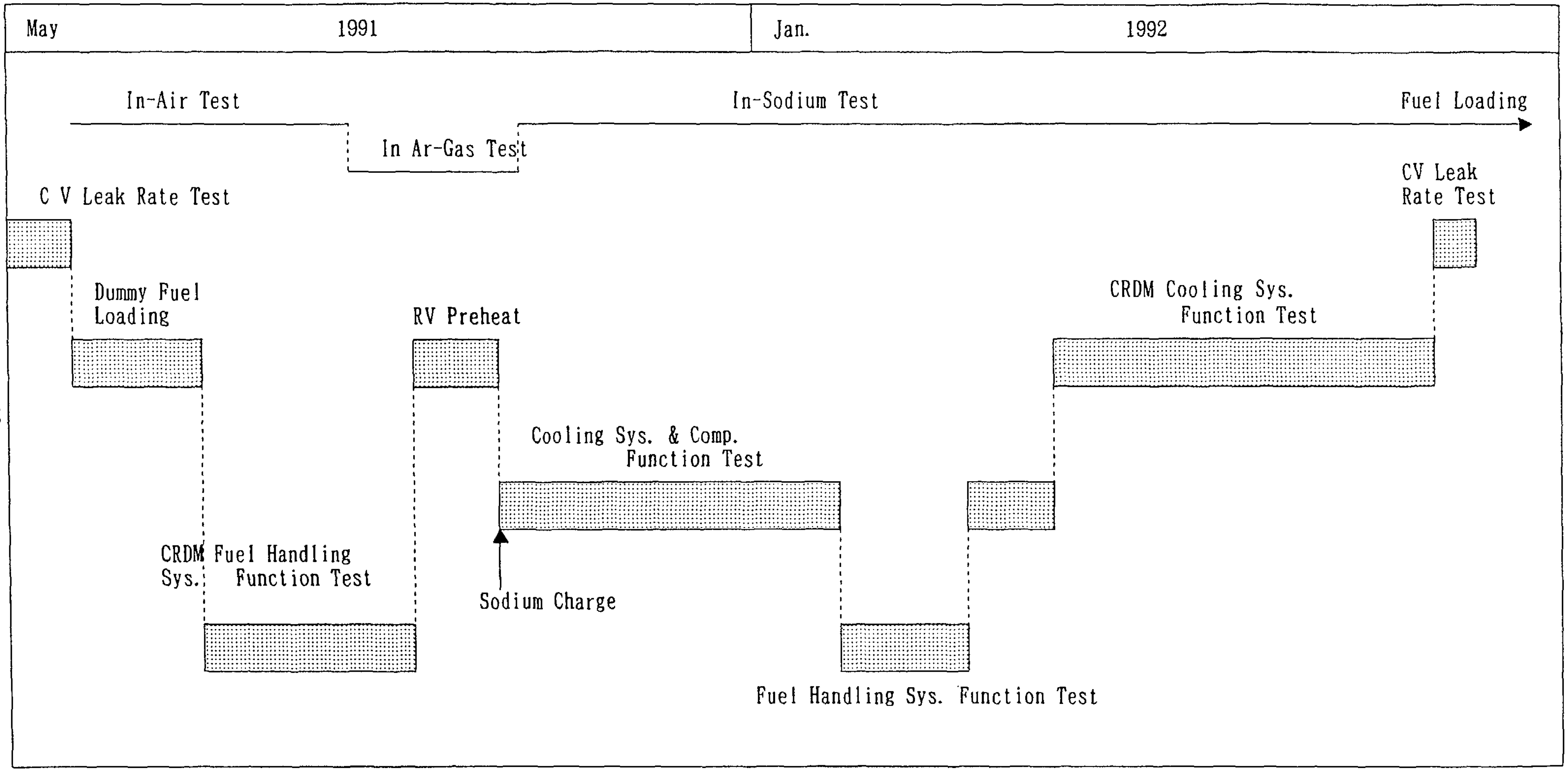


Fig.3-2 Monju Function Tests Schedule

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

The present DFBR design study by JAPC is based on the decision by the Federation of Electric Power Companies (FEPC) of Japan to endorse JAPC as the utilities instrument to develop DFBR.

In June 1990, FEPC decided to start the preliminary conceptual design study of DFBR to verify technical feasibility of Top Entry Loop type FBR. It was concluded that both Top Entry Loop type and Tank type reactors have its own potential advantages and disadvantages, and have potentials for commercialization in future, and the Top Entry Loop type reactor has more flexibilities to adopt new and innovative designs in future, and that it enables step-by-step realization of concentrated layout of primary loop components, proving reliabilities of the components likewise.

In July 1990, the JAPC started the preliminary conceptual design study on a

600MWe Top Entry Loop Type DFBR under the auspices of the FEPC. About 600MWe was selected as the plant output considering that it must be large enough to get a perspective of a full size commercial plant but as small as possible to reduce the economic burden.

This design study consists of two parts, the conceptual plant design emphasizing systems and components in primary and secondary sodium loop, and condirmation tests for investigating the phenomena important to technical feasibility.

JAPC also has performed eveluation study on commercialization scenario to commercial FBR's. These studies will be summarized by the end of FY 1991.

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 acces piping system and performed plant construction cost evaluation.

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

5. Reactor Physics

5.1 Calculation Method Development

A three-dimensional discrete-ordinates transport code 'TRITAC' has been developed for accurate calculations of eigenvalue problems and its high performance was confirmed by the NEA-CRP's three-dimensional transport code benchmark test.

Since a Hex-Z geometry is more useful and practical than a XYZ geometry for the neutronics calculations of the FBR core, the development of nodal transport code for Hex-Z geometry is in progress. Currently, three-dimensional transport equations based on analytical polynominal expansion nodal method have been derived and test calculations are being conducted.

5.2 Cross Section Adjustment

Critical assembly experiments on large fast breeder reactor were analyzed by a current neutronics analysis method for fast reactor core, using the JF-3-J2 cross section set processed from the JENDL-2 library. It was made clear that there were a tendency of C/E values for reaction rate and control rod reactivity worth to become higher along with core radius, in addition to discrepancies between calculation and experiment. A cross section adjustment was performed not only so that the radial dependence of C/E values might be solved, but also so that prediction accuracy of nuclear design of large fast reactor core might be improved at the same time. Experimental data for cross section adjustment were taken mainly from critical experiments of 600～1000 MWe class fast reactor cores which include not only homogeneous cores but also radially and axially heterogeneous cores. Furthermore, experimental data of Pu-240 and Pu-241 were added.

As the result of adjustment, the disagreement between calculation and experiment has been significantly improved. As for the prediction accuracy improvement of nuclear design, a comparison between the bias factor correction method and the cross section adjustment method was conducted for a 1000MWe homogeneous core. It was clarified that the prediction accuracy of criticality, reaction rate and control rod reactivity worth could be significantly reduced, about 50% for control reactivity worth, by employing the cross section adjustment method. Consequently, it has become clear that the cross section adjustment method is quite effective to improve the design accuracy.

5.3 Critical Experiment

As a part of reactor physics research for fast reactor with non-oxide fuel, critical experiment for metal fuel have been conducted at FCA (Fast Critical Assembly). One of the experiments is a mockup of small, about 600 liter, metal fuel core, and the nuclear characteristics of the metal fuel core, such as breeding capability and reactivity coefficients, have been experimentally verified.

Critical experiments for nitride fuel core are scheduled to be conducted from next year.

5.4 Shielding Experiment and Analysis

Power Reactor and Nuclear Fuel Development Corporation 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 Radial Shield Experiment and the Fission Gas Plenum Experiment, the third and fourth JASPER experiments, the Axial Shield Experiment and the In-vessel Fuel Storage Experiment have been completed.

The Axial Shield Expriment was aimed to evaluate the neutron attenuation characteristics of and the neutron streaming through the different axial shielding designs, such as rod bundle type, central blockage type and central Na channel type, consisting of the different shielding materials. As the results, various streaming factors were obtained design calculations. Furthermore, the accuracy of the calculational system to predict a neutron streaming effect along the fuel subassembly was estimated to be about five percent for the energy integrated flux.

The purpose of the In-vessel Fuel Storage Experiment was to evaluate the influence of in-vessel storage on shielding characteristics. The experimental results are under analyses.

6. Systems and Components

6.1 Control Rod Drive Mechanism

Research on the self-actuated shutdown system (SASS) for the 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 have been completed in November 1991. For the Monju reactor vessel, full size model tests are in progress and will be completed by March 1992. New techniques, such as remote inspection technique using optical fiber scopes for reactor vessels, electro magnetic acoustic and ultrasonic transducers for high temperature use on reactor vessels, ultrasonic tranceducers without couplant for primary piping systems are adopted after a series of performans tests at Oarai 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. To support 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. The 1MW double-wall tube steam generator model is operated at full load since November 1991.

6.4 Process Instrumentation

The calibration of fuel subassembly outlet flowmeter 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 is scheduled to 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 production. This performance test is also scheduled in 1992.

7. Fuels and Materials

7.1 Fuel Fabrication

The PFPF (Plutonium Fuel Production Facility) equipped with automated and remote handling fuel production systems started to fabricate "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 performances code for metal, carbide and nitride fuel is in progress.

7.3 Core Materials

SUS 316 stainless steel (Monju core material) irradiated over 2.5×10^{23} n/cm² (E>0.1 MeV) showed excellent swelling resistace 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 uel 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. A power-to-melt tests 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 subassembies was completed for SUS 316 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

The enhancement of the general purpose nonlinear structural analysis program FINAS has been continued since 1986, particularly with respect to inelastic constitutive models of cyclic plasticity and visco-platicity, large deformation/buckling analysis methods, shell elements, automatical computation algorithms, contact problem solution algorithms. FINAS is currently used by many research engineers at over 30 sites including fabricators and universities. The current 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, 3-D in-surface heat transfer elements, large-scale dynamic analysis, surface contact analysis is scheduled in 1992.

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

The ratchetting criteria, which are not provided explicitly in the design guide for multiaxial stress states, 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 evalution is under way.

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

A vessel model, piping bellows models and two thermal stress mitigation model tests were completed. An welded vessel model test is scheduled in 1992.

3) Plastic buckling tests (SCFT)

Buckling tests for 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 the advanced inelastic analysis methods.

8.3 Seismic Test and Analysis

A new shaking table DST, with a size of 2.5m× 3m, a loaded weight 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. A conceptual and feasibility study of a vertical seismic isolation system for FBR components is started.

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) was already completed. The step-2 program (1988-1990)are currently underway.

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

8.4 Fracture Mechanics and Structural Integrity Assessment

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

Computer codes being developed at PNC includes CANIS-J for the calculation of fracture mechanics parameters, CANIS-G for the simplified crack propagation analysis and CANIS-P for the probabilistic fracture mechanics analysis. Crack propagation tests of preflawed structural elements, like pipes, plates, and elbows which are subjected to mechanical loadings, has been performed since 1987, to validate the applicability of the computer codes in non-creep and creep regions. Crack propagation tests of a cylinder with circumferenctial and surface flaws are being conducted at the air-cooling thermal transient test facility(ATTF).

1) Tests in Air

The present Capella step 2 program includes following subjects;

* Improvement of Monju design method on creep-fatigue life, strength of weldment, inelastic constitutive equations
* Establishment of design and fabrication method of large scale structures
* Modification of material specifications including application of modified SUS 304 and 316 stainless steels
* Application of elevated temperature fracture mechanics
* Development of the material strength standard for high Cr-Mo steels

A tentative 1989' version of the Material Strength Standard including the rules for 9Cr-Mo steel and modified SUS 316, and life evaluation method was examined using test results in 1990.

2) Test in Sodium and Water/steam

A new series of sodium environmental effect tests, according to the Capella program (1985-1990), were carried out on possible candidate alloys for future FBRs. Candidate alloys were high Cr-Mo steels, and advanced type (low carbon and/or high nitrogen) SUS304 and 316 stainless steels. Corrosion and mass transfer, carbon transfer, and mechanical strength (tensile, creep, fatigue, creep fatigue) tests in sodium are still continued in the program with emphasis on modified 9Cr-1Mo steel and modified SUS316.

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 modified SUS316.

8.5 Data Banking System

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

The SMAT 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, the 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 eveluation of the thermal striping phenomena 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 susbassemblies, and plenum-channel thermohydraulic interactions in the mixed convection regime, (2) LOPI simulation experiments on the PLANDTL facility, (3) code development and validation for the subassembly and reactor core heat transfer analysis (ASFRE, SABENA), thermohydraulic analysis in plena (AQUA), 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.

The inter-subassembly heat transfer experiments were completed using the Core and Component Thermohydraulic Test Loop (CCTL) during this period.

A fundamental water experient 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 the 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 performed. The penetration of the upper plenum's fluid into the low power assembly was clearly observed. The intersubassembly flow redistribution was measured quantitatively.

The LOPI simulation experiments which used the PLANDTL facility were completed during this period. Parallel channel sodium boiling experiments in the mired convection range were initiated using PLANDTL facility. Two subassemblies are connected to upper and lower plena. Hence the thermohydraulic coupling of the plena and subassemblies can be investigated.

Efforts on thermohydraulics and safety analysis code development and validation are continued for the subchannel analysis codes ASFRE and SABENA, three dimensional thermohydraulic analysis code AQUA, and the plant system dynamics analysis code SSC.

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. During this reporting period, the inter-subassembly heat transfer experiments under low flow natural circulation condition were analyzed using ASFRE code. The code validation study is to be continued. A high-order, more accurate numerical model TVD (Total Variation Diminishing Scheme) was implemented in SABENA. Code validation was carried out by simulating an experiment of air bubble behavior on a cylindrical tank bearing water inside. The loop-type version of SSC (SSC-L) was extensively validated using LOPI simulation experiments. Also, SCC was utilized for the accident of the demonstration FBR. Improvement of AQUA code has been made by introducing an Algebraic Stress turbulence Model (ASM). Thermal striping experiments using water were analyzed to check the general performance of the ASM employed in AQUA code. Furthermore, fundamental experiment series were started to investigate the temperature fluctuation at the reactor core exit. The results are to be used for checking the applicability of AQUA code to the thermal striping evalution of the 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 severe 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 experiment, 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 analysed 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. Development of the computer code SCION was finished. The first version of the code was completed. It was used to calculate initial conditions of a local fault out-of-pile experiment at PNC, named Discharge test. Discharge test was to concentrate on a melt behavior in a triangular pin bundle geometry in gas circumstance. The first series of test were completed, and will be analysed by the code SCION.

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 almost completed. A new series of experiments centered on the thermohydraulic interactions between a molten materials jet and coolant was started. To investigate the boiling pool phenomena, a new facility named "POOL" was being constructed. The POOL facility has a 50kw microwave oven to simulate the dynamics of volumetrically heated boiling pool.

The CABRI-2 in-pile experiment activities are in progress and include pre-and post-test analysis. In this program a main concerns is to understand the high-burnup fuel behavior under mild and long-term transient condition. 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 from both the CABRI-1 and CABRI-2 are integrated into SAS3D's computational models, and the relevant knowledge obtained is also incorporated into model improvement of the SAS4A code which is considered to be a new generation reference code for initiating phase analysis. A PNC-USNRC agreement for the joint SIMMER-III code development was terminated in 1990. All the tasks were transferred to PNC,and a preliminary version of SIMMER- III has been made operational. The formulation and development of individual detailed models are in progress. For evaluation of LOHRS sequences, e.g., PLOHS and LORL, development of the APPLOHS code for the core disruption phase, were continued, including an interface routine with the SIMMER code.

9.3 Plant Accident Research

FBR plant accident research consists of two major activities. One is a study on a non-radiological sodium fire accompanied 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, with thermal radiation properties of sodium aerosol being measured. In parallel with this study the conceptual design and analysis was conducted to develop a refractory ceramic liner 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 the release 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 combution tests was initiated which is intended to quantify combustion condition in a sodium aerosol atmoshere.

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

Probabilistic Safety Assessment (PSA) began in 1982 as part of the R&D on the Monju prototype reactor.

The purpose of this reseach is to construct probabilistic safety models for the Monju plant so that an overall safety assessment can be performed. Considerations to perform this task are as follows:

(1) A systematic evaluation on plant safety will be conducted which is based on quantitative analysis.

(2) Insights on system reliability and safety enhancement measures will be provided.

(3) Operation and maintenance procedures will be established on a technical basis.

(4) Information leading to the development of a basic policy for safety design and evaluation of a large LMFBR will be given.

PNC has developing a systems analysis code network, with this network soon being able to perform level-1 PSA. Recent efforts have focused on development of PSA application softwares such as living PSA tool and an operator assistance system. The developed systems analysis codes include a human reliability analysis support probram, a Monte Carlo method phased mission analysis program, and a PC-based level-1 PSA program. The development of an accident sequence analysis system which examines the accident status during different alarm combinations is also under way. Furthermore, PNC has initiated the software development for a Living PSA System (LIPSAS).

Efforts are being made to develop LMFBR component data based on CREDO

(Centralized Reliability Data Organization), a cooperative project between PNC and the USDOE, with this work to continue in the LMFBR-related facilities of both countries. PNC also began the development of a new CREDO data base system which uses a commercial relational data base system. To analyze the data a preliminary component aging failure analysis was carried out for valves and mechanical pumps. Statistical data on Japanese natural and man-made hazards has additionally been collected and analyzed.

Systems analysis was performed to update level-1 PSA with respect to internal events. Minor design changes, updated CREDO data, and several modifications of modelling assumptions and conditions were reflected in order to requantify core damage sequences. The overall core damage frequency (CDF) was lowered by a factor of about six when compared to results obtained last year, and it is much lower than the future plant reference value of 10^{-5} /ry proposed by the INSAG of the IAEA. Uncertainty and importance analyses which include several sensitivity studies were also performed using the updated reference case results. In seismic event analysis a fragility evaluation was conducted based on both the design analyses and the test data concerning the design basis seismic event. 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.

Level-2 PSA tasks (consequence analysis) were almost completed, with in-vessel and ex-vessel physical processes being analyzed for the key core damage sequences which were identified and quantified by the sytems analysis. Phenomenological event trees were constructed to both represent complex accident progression spectra and to include various physical uncertainties. Mechanistic computer codes developed thus far were used to analyze important accident sequences in order to realistically assess their consequences and to identify risk-dominant phenomena. The results were summarizes using complementary cumulative distribution function (CCDF) curves for off-site source terms. It is concluded that the overall risk of a model plant from the core damage sequenes identified in this study is kept at a sufficiently low level. The probability of containment failure, which would result in serious radiological consequences, is also shown to be negligibly small.

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 parallell 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, produciton of Joyo fuel started in October 1988 as the first production campaign at PFPF.

The PFPF is currently fabricating fuels for Monju.

To provide MOX fuel for ATRs, PNC is planning to construct a new ATR line (40 t on MOX/year) at PFPF so that fuels for the ATR demonstration reactor will be available for startup when needed.

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.

The initial production capacity of 5 ton MOX/year of FBR line is so designed as to increase the capacity to 15 ton MOX/year by adding the process equipments, to cover the fabrication of fuels for Demonstration FBR.

About 112 tonnes of MOX fuel have been fabricated by the end of December 1991.

2) R&D on MOX Fuel Fabricaiton

Remote control and automatic operation techniques, which are indispensable for MOX fuel fabrication facilities, are being developed in PFPF. Although it has a direct maintenance system, persons do not normally have to access to nuclear materials. It was achieved through various experiences at PFPF.

At PFDF, research on the manufacturing fuels with new materials, new welding techniques and other aspects of plutonium fuel fabrication will be carried out. At PFPF, development of fabrication equipment and instruments will be continued.

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 1991 it produced about 6.6 ton of MOX powder using about 2.6 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 Facilies (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 servomanipulator, 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 94000MWD/T have been reprocessed at CPF. Through these hot tests, informations 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. Form this viewpoint, PNC is now designing Recycle Equipment Test Faciliuty (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.