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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 23rd Annual Meeting (April 24~ 27, 1990, Vienna, Austria) by International Division of Power Reactor and Nuclear Fuel Development Corporation (PNC), with contribution from;

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(IWGR: 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 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 40,000 hours operation in total by the end of 1989, since its first criticality in 1977.

3) On the prototype reactor "Monju", more than eighty percents of construction works has already been completed on schedule at Tsuruga city, aiming at the initial criticality by October 1992.

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 Toshida, Hitachi and Mitsubishi 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 cordination of the Japanese FBR R&D Steering Committee, which was established by the JAPC, PNC, Japan Atomic 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 Sub-committee on FBR Development Program of AEC.

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, thoughtful planning of the overall functional and performance tests of Monju, scheduled to start in 1991, is another important key to attain further excellency of FBR technology, with full efficient usage of the test results.

6) R&Ds on following tasks are also in progress for development of the DFBR, for excelllent 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 Joyo and Monju has started since 1988.

8) On the FBR fuel recycling, adding to the experiences at the Tokai Reprocessing Plant, R&D are underway at three Engineering Demonstration Facilities (EDF-I, II, III) and

Chemical Processing Facility (CPF), integrating the results to the design of planning 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 at Hokkaido.

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

2. Experimental Fast Reactor, Joyo

2.1 General Status

This report covers the activities of Joyo from April 1989 through March 1990. The operating history of Joyo is illustrated in Fig. 2.1.

The reactor finished the 18th, 19th and 20th duty cycles during the above peried, attaining the total operation time of 40,000 hrs from the date of initial criticality in 1977.

The average burn-up of the core was increased using increased U-235 enriched fuel assemblies. The maximum burn up of the driver fuel has reached more than 70,800 MWd/t (75,000 MWd/t licensed).

A special test operation, called the 20th Cycle Operation, was performed from January 17 to January 22, 1990, in which the power generation ratio of fuel assembly at the in-vessel storage rack (IVSR) was measured and comparad with that at the core center.

The 8th periodical inspection of Joyo began from January 1990 and will finish by September 1990.

The construction of the Third Spent Fuel Storage Facility began from December 1989 at the site of the reactor. The facility is scheduled to be completed at the end of 1991 and will be able to accommodate approximately 800 spent fuel assemblies.

A training program of Monju operators were prepared at the Joyo site. The program includes setting-up of equipments, and preparation of the textbooks and curriculums. The actual training is to start from April 1990.

2.2 Mark-Ⅲ project

A project, called MK-Ⅲ core project, is being studied at the Joyo site. The project includes the raise of thermal power level from 100 MWt to 140MWt in order to increase the neuton flux, to improve the reactor load factor by replacing fuel handling systems, and to install a double walled steam generator in the primary loop of Joyo so as to demonstrate the possibility of elimination of the secondary cooling system.

For the first step of the project, one of the six control rods of JOYO will be removed from the third row to the fifth row of the core. The preparatory work for licensing application of the Mark- III project is being carried out.

2.3 Measurement of Power Generation Rate of Fuel in IVSR

One of the major objectives of the Mark-III core program is to enhance the present irradiation capacity by increasing the maximum power level and also improving the load factor. In order to achieve the objective, present discussion is focused on the possibility of replacing the IVSR (invessel fuel storage rack) B pots, which have sodium flow holes at their bottoms, by the IVSR A pots without flow holes.

This will require the replacement of the present stainless steel reflectors by B_4C-shielding subassemblies so as to reduce the neutron flux in the vicinity of the pots, which in turn will keep the power generation rate to a comparatively low level so that natural convection cooling is sufficient to maintain the fuel temperature to a safe level.

Parametric survey calculations were carried out on some of the basic specifications of the shielding subassemblies by three dimensional Hex-Z diffusion theory. However, it is necessary to evaluate the precision of the calculated results by comparing with measured ones.

Therefore, three fresh fuel subassemblies, two of them placed in the IVSR and one in the core center as shown in Fig. 2.2, were irradiated at the rated power for three days in the 20th operation cycle. Gamma ray spectra of the three fuel subassemblies were obtained five days after the shutdown of the reactor by using a pure-germanium detection system set on the exvessel transfer machine.

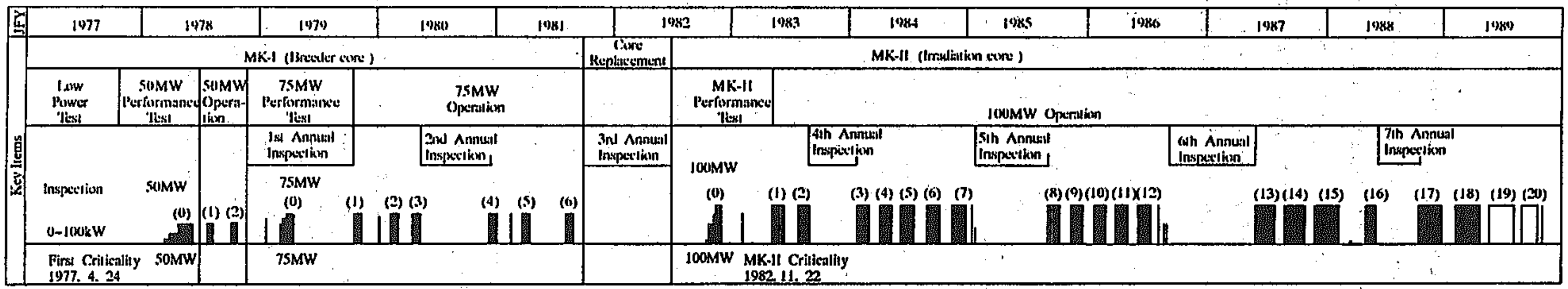

Fig. 2.1 Experimental Fast Reactor Joyo Operating History

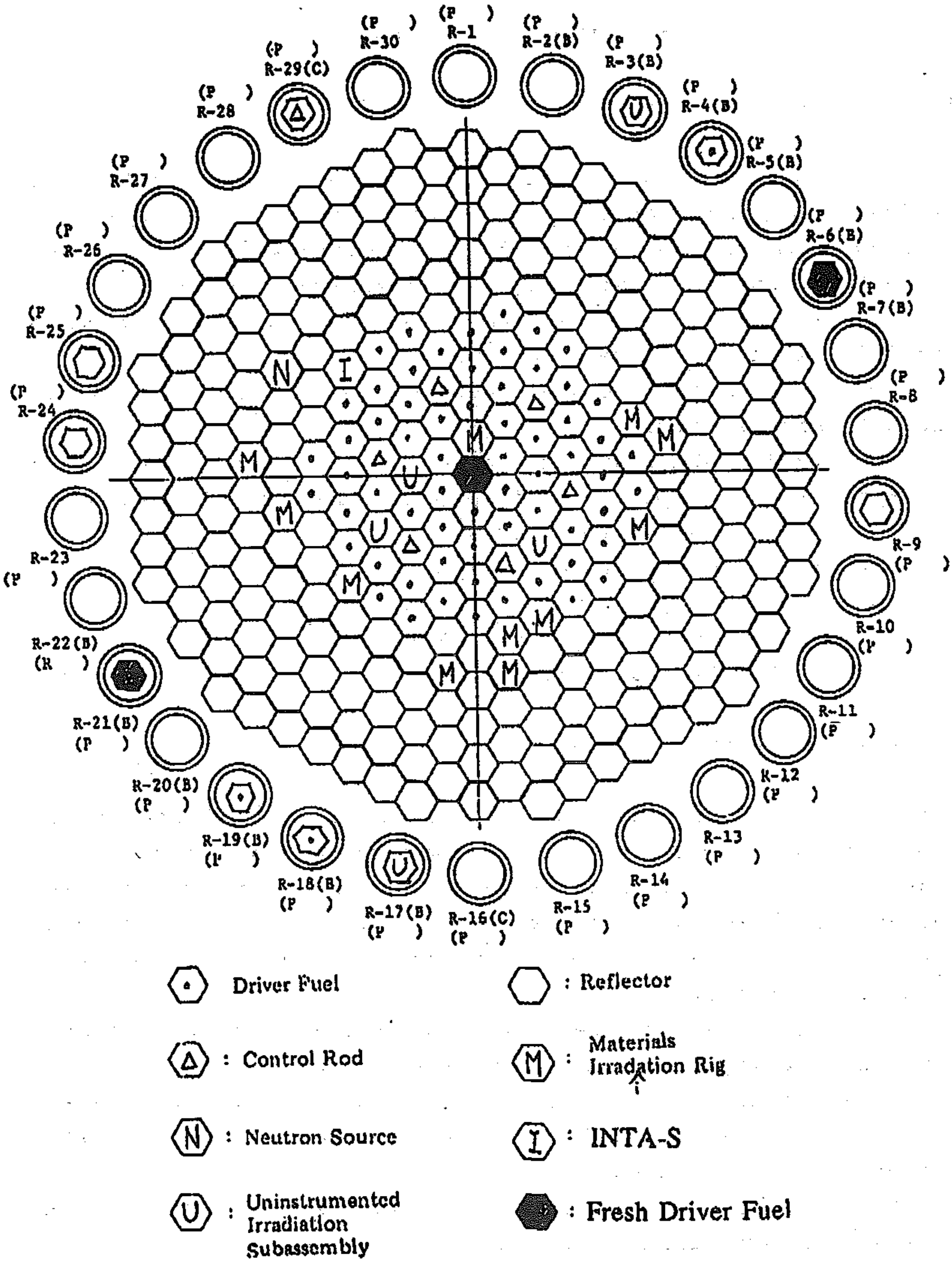

Fig. 2. 2 Core Configuration during 20th Operation Cycle

3. Prototype FBR, Monju

3.1 Construction Schedule

The Monju site is located on the north side of the Tsuruga Peninsula in the central Japan, facing the Sea of Japan and is surrounded by mountains of approximately 300-700m high. Since the plant is located inside the Wakasa Bay Quasinational 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
Oct. 1992	Initial Criticality

3.2 Present Status of Construction

Monju construction was 80.9% completed as of the end of February 1990 including design, components manufacturing, and construction works at site. Major components such as the reactor vessel, IHXs, SGs, main control consoles and various tanks are already installed and CRDs are currently in the final stage of assembling.

Major civil works are also completed (about 96%), except for construction of the cooling water intake structures.

Construction of the buildings is currently 83% completed.

Cell liner installation in the primary heat transfer system cells are completed.

Cabling, piping and other miscellaneous construction works in the reactor building and the reactor auxiliary building are now under way, with much care being taken over the salinity in the air and the cleanliness of the buildings.

The high and low pressure tests of the primary heat transfer system were successfully completed during November 1989 to February 1990.

3.3 Function Test Program

An overall Function Test 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 procedures.

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 Claddsng Temperature (middle of thickness)		675℃
Power Density		283 KW/lit.
Blanket Thickness		Upper 300 mm Lower 350 mm Radial 300 mm
Breeding Ratio		1.2
Reactor in/out Sodium Temperature		397/529℃
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 Generator		Helical Coil, Once-through Unit Type
Steam Pressure (Turbine Inlet)		127 kg/cm^2 g
Steam Temperature (Turbine Inlet)		483℃
Refueling System		Single Rotating Plug with Fixed Arm FHM
Refueling Interval		6 Months

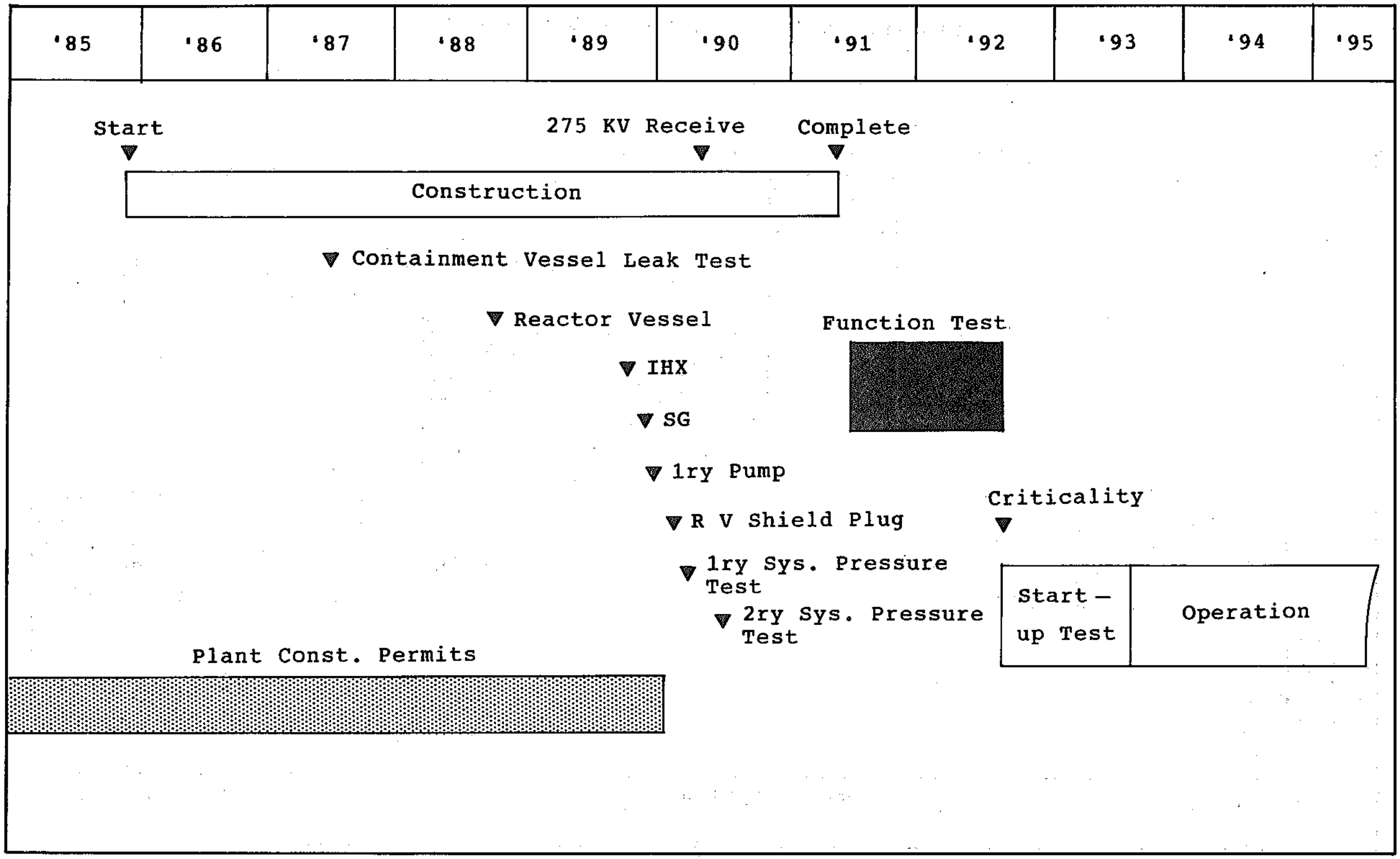

Fig. 3-1 Monju Construction & Tests Schedule

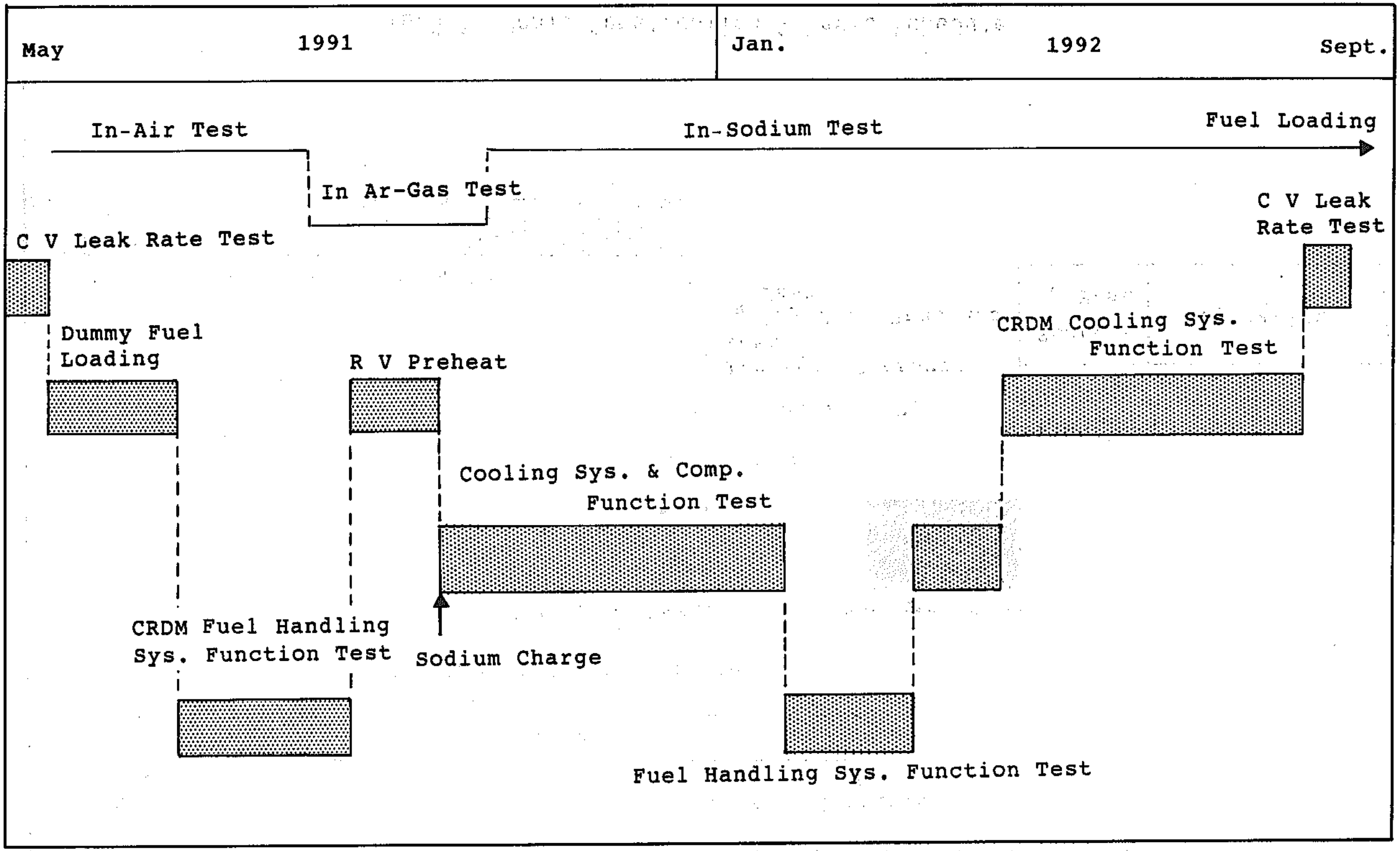

Fig. 3-2 Monju Function Tests Schedule

4. DFBR and RNC's Design Study

4.1 Overview

The Japan Atomic Energy Commission (JAEC) issued Japanese "Long-term Program for Development and Utilization of Nuclear Energey" 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 demonstration FBRs, 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 endose JAPC as the utilities instrument to develop DFBR following the basic schedule shown in Fig. 4-1.

In 1988 and 1989, the effort of JAPC was focused mainly to evaluate the plant maintenability and seismic design on the both candidate pool and loop design selected by 1988 and comparison study was performed to some details. In 1990 JAPC is expected to start the conceptual design of the DFBR with approval of the FEPC.

4.3 PNC's Design Study

In 1988, PNC started an independent FBR Plant Design Study program, which include large, medium, and small size FBR designs aiming to search the way to attain FBR commercialization in early 2000.

It includes the analysis of every technical possibility of FBRs, based on the experiences of Joyo and Monju developments.

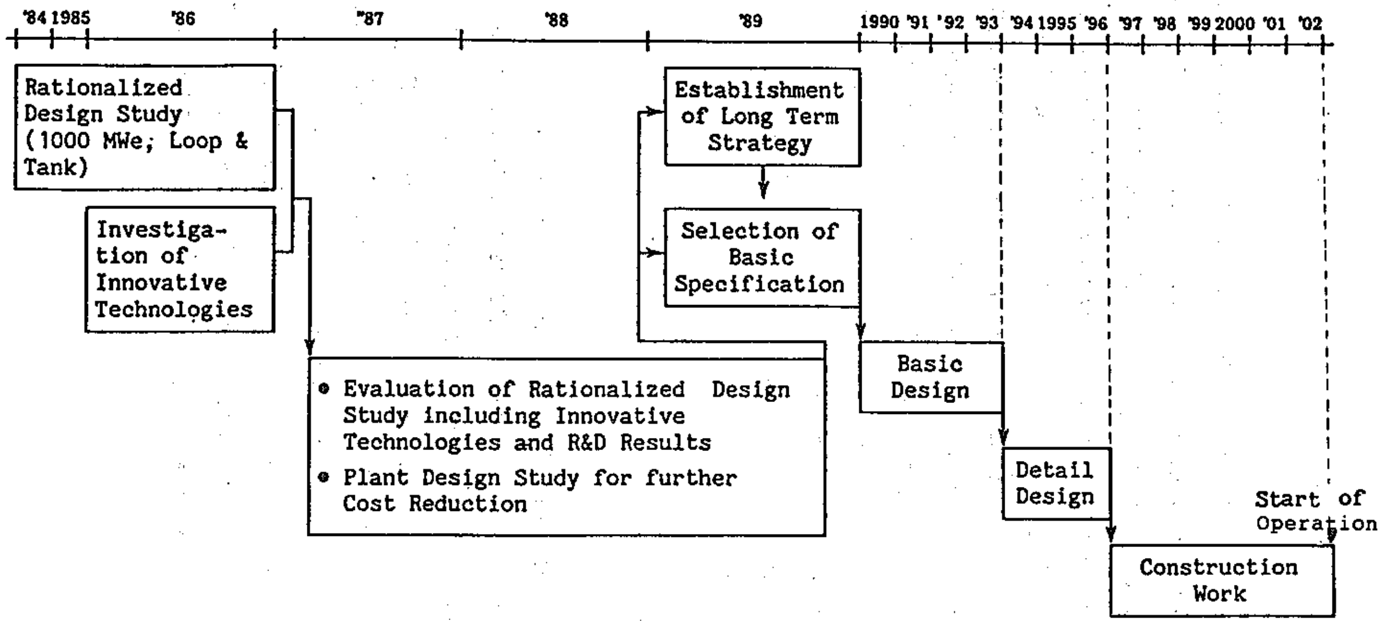

Fig. 4. 1 Scenario of DFBR-1 Development

5. Reactor Physics

5.1 Improvement of Nuclear Data Library (JENDL)

JENDL-3 is the third version of Japanese Evaluated Nucler Data Library (JENDL), which contains neutron-induced reaction data for 325 nuclides including 172 fission product nuclides. The compilation work was already completed and benchmark calculations have been done for fast critical assemblies with various core sizes and fuel compositions since 1988.

The effective multiplication factors obtained with JENDL-3 are in better agreement with the experiments for plutonium cores than those obtained with JENDL-2. However, they are underestimated for uranium cores, except for considerably hard neutron spectrum cores.

On the other items of neutron characteristics, considerable changes were observed in the analysis of reaction rate ratio of U-238 capture to Pu-239 fission and the natural UO_2 Doppler worths in a conventional 600MWe-size Mox fuel core experiment.

5.2 Development of Reactor Physics Calculation Methods

1) Evaluation of Streaming Effects Using Double Heterogenous Modeling

Neutron streaming effects in a FBR subassembly was estimated considering the "double heterogeneity", which is due to the heterogeneous structures of the pin and the wrapper tube.

The total streaming effect for reactivity caused by the double heterogeneity structure of a fuel subassembly is found to be almost twice as much as that obtained from the conventional pin-cell model.

2) Improvement of Transport Code TRITAC

Three-dimensional discrete ordinates transport code TRITAC has been developed at Osaka University and PNC since 1983.

Recently, it was improved to treat higher order of anisotropic scattering effect. The diffusion synthetic calculation procedures were also refined so that the parameters used in the diffusion calculations are optimizied automatically.

As a result, calculation time was reduced to about 1/4~1/5 in a typical FBR core calcalation, which enables the code to meet the general requirements in reactor physics and safety studies.

3) Development of Reactor Physics Calculation Codes

Some other diffusion and transport codes have been developed, based on various theories such as nodal method, SN method, and spherical harmonics method.

SIXTUS-3, an extended version of SIXTUS-2, solves 3D hexagonal-Z geometry and gives much more precise results than conventional codes in a given calculation time.

4) Improvement of Eigenvalue Separation Analysis Methods

An improved technique for inferring the eigenvalue separation, which is important in spatial stability analysis, was developed using the noise coherence function.

This technique was applied to fast reactor critical assemblies of various sizes and compositions which exhibited a wide range in spatial decoupling. The eigenvalue separation obtained by noise analysis gave good agreement with calculation.

5.3 Development of Cross-section Adjustment Method

The JUPITER program, an extensive study on reactor physics for large scale fast breeder cores, was completed in 1988.

Parallel to the experiments, experimental analyses were made using the Japanese data and methods. Through the analyses, it was made clear that there were some radial dependence of calculation and experiment (C/E) values for integral physics parameters.

A cross section adjustment was performed to minimize the radial dependence of C/E values and the C/E discrepancies from unity at the same time. As the result of adjustment, the disareement has been improved remarkably.

The uncertainty of neutronic performance parameters was evaluated for the three nuclear design methods. In calculating the uncertainty, we considered the errors due to experiment and calculation both. The derived formulas have been applied to a large homogeneous FBR core of 1000 MWe.

5.4 Research on Shielding

The JASPER program is a joint program between U.S.DOE and Japan's PNC on shielding experimental research. The shielding experiments have been performed at TSF (Tower Shielding Facility) in U.S.'s ORNL.

Items of the experiments are as follows.

a) Radial Shielding Attenuation Experiment (completed in 1986)
b) Fission Gas Plenum Experiment (Completed in 1987)
c) Axial Shield Experiment (Now underway)
d) Stored Fuel Experiment
e) Gap Streaming Experiment
f) IHX Activation Experiment
g) Flux Monitor Experiment
h) Special Materials Experiment

Experimental analyses are performed using the Japanese data (JENDL-2 and 3) and methods. The results are summarized for application to shielding design studies.

6. Systems and Components

6.1 Shield Plug

Sodium vapor (or mist) concentration in the argon cover gas space of a reactor model at OEC was measured to provide basic data used for evaluating sodium deposition rates at relatively low temperature conditions ranging from 150℃ to 300℃ . The concentration data was reflected on the evaluation of sodium deposition rates for the rotating shield plug of Monju.

Development of an analysis code (FLUSH-Code) for heat transfer and cover gas flow behavior induced by natural convection in annulus structures above pool surface of reactor vessel are continued.

6.2 Control Rod Drive Mechanism

Research on the self-actuated shutdown system (SASS) for the DFBR was initiated in spring 1987. The performance test of magnet at high temperature was carried out in-air and in-sodium testing is being performed.

6.3 In-service Inspection Equipment

Full size model tests for the Monju reactor vessel, steam generator tubes, and primary pipes have started spring 1990, aiming at the completion by December 1991. 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 OEC.

6.4 Steam Generator

Development of double-wall tube steam generators is in progress for future FBR plant under the joint R&D program between PNC and the Japan Atomic Power Company. In parallel, PNC is conducting a conceptional design study of a plant having steam generators in the primary heat transport system. To support this concept steam leakage test of double-wall heat transfer tubes was carried out to investigate detectability of failures of the inner tube walls.

6.5 Process Instrumentation

Since the permanent magnet type flowmeter was adopted to the primary and secondary system of MONJU, the off-site calibrations in sodium loop were carried out.

A calibration of the sodium level meters for Monju is being performed.

6.6 Early Warning System for Fuel Failure

Calibrations of eddy-current flow/temperature sensors which will be installed in the above-core instrumentation of Monju to detect fuel failure were carried out in a sodium loop. Accuracy of the calibration is within 1% in the flow rate range of 0 to 7 m/s.

Temperature fluctuation due to flow blockage in pin bundle has been analysed by a computer code developed. Thus, the optimum position to locate temperature sensors is clarified.

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.

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

7.3 Core Materials

SUS 316 stainless steel (Monju core material) irradiated until $2.1x10^{23} n/cm^2$ (E>0.1 MeV) showed excellent swelling resistance by less than 1.5% swelling. Out-of-reactor mechanical property and sodium corrosion tests of advanced austenitic stainless steels have been completed. Irradition tests for the future candidate steels are conducted in Joyo and FFTF.

Two types of ferritic steel were developed since 1984. One is high strength ferritic / martensitic steel which is considered to fit wrapper tube and the other is oxide dispersion strengthened ferritic steel (ODS). The tubing technology for ODS cladding has progressed by hot working.

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

A new study on neutronics and thermal hydraulics for duct-less subassemblies were initiated recently.

7.4 Irradiation Experiments

1) Joyo

Monju fuel for high burnup core, axial heterogeneous core fuel and advanced austenitic stainless steel fuel have also irradiated. Fuel subassembly using CEA cladding tubes has been irradiated since August, 1988.

2) Foreign Reactors

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

Fuel subassembies using SUS 316 and advanced austenitic stainless steel cladding have been irradiated in FFTF since November, 1987.

7.5 Development of Advanced Fuels

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

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

7.6 Post-irradiation Examination

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

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 1981, particularly with respect to inelastic constitutive models of cyclic plasticity and viscoplasticity, large deformation/ buckling analysis methods, shell elements, automatical computation algorithms, fracture mechanics capabilities, dynamic analysis capabilities including fluid-structure interaction and graphics options. FINAS is currently used by many research engineers and designers of PNC, fabricators and universities.

Recently, new functions were added including improved cyclic plasticity model, unified plasticity-creep model, buckling analysis method introducing load stiffness matrix.

2) Improvement of Elevated Temperature Structural Design Guide

The following rules are under discussion to improve and extend the current Elevated Temperature Structural Design Guide developed for MONJU.

* Design rules for welds
* Creep-fatique design methods based on elastic analysis
* Strain limit criteria (other ratchetting mechanism than the conventional Bree-type mechanism)

8.2 Structural Test and Evaluation

Following structural tests being performed to develop strength prediction methods, to evaluate the adequacy of elevated temperature design rules and also to confirm the integrity of the actual components.

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

Structural discontinuity model tests to investigate crack initiation and propagation behavior will be completed by the end of 1990.

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

Two vessel models, piping bellows models and thermal stress mitigation model tests were completed. An welded vessel model test is currently under preparation.

8.3 Structural Material Test

Structural material tests in air, in sodium 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".

1) Tests in Air

The present Capella Step 2 program includes following subjects;

* Improvement of MONJU design method on creep-fatigue life, strength of welding, inelastic constitutive equations
* Establishment of design and fabrication method of large-scale structures
* Modification of material specifications including application of modified SUS304 and 316 stainless steels
* Application of elevated temperature fracture mechanics
* Development of the material strength standard for high Cr-Mo steels

A tentative new version of the Material Strength Standard including the rules for 9Cr-Mo steel and modified SUS316, and revised life evaluation method was prepared in 1989.

2) Tests in Sodium and Water

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. Corosion and mass transfer, carbon transfer, and mechanical strength (tensile, creep, fatigue, creep fatigue) tests in sodium are still continued in the program.

3) Tests in Irradiation Environments

Surveillance tests for the primary 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 primary components of Monju to evaluate irradiation effects on the mechanical properties up to the end of design life and to evaluate irradiation effects on the 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 a DFBR is being conducted to clear the relationship between creep rupture strength and metallurgical variables such as chemical composition, grain size and production process.

8.4 Data Banking System

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

The SMAT has 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 Systems and Design

The ability of LMFBRs to remove decay heat by natural circulation is one of the important safety features of the current heat transport systems design. To support the design of passive decay heat removal systems, experimental studies are in progress using water as a working fluid. Following the first phase tests for a loop-type model, the second phase tests for a 1000 MWe pool-type FBR have been underway using a 1/8 scale reactor model since 1987 in cooperation with the Japan Atomic Power Company. The objectives of these tests are: (1) to demonstrate feasibilities of the decay heat removal systems by natural circulation in the event of loss of on-site power, and (2) to demonstrate the analytical capabilities of AQUA that computer simulation can be used to explain flow phenomena and to extrapolate the information from the experiments, where all the similarity laws cannot be fulfilled, to actual reactor conditions. Results to be obtained in the experiments will provide a norm for the selection of basic design specifications of the decay heat removal system of a 1000 MWe demonstration FBR plant. Also the sodium test for the in-line rod array is in progress for the purpose of investigating the natural convection heat transfer characteristics of a coil-type heat exchanger immersed in the reactor plenum.

Code development and validation studies are continued for the single-phase three-dimensional thermohydraulic analysis code AQUA. Improvement of the Fuzzy controller, which optimizes time step sizes automatically based on an artificial intelligence technique, has been made by implementing a learning function into the system. Further efforts have been directed toward improvement in the turbulence model and evaluation of the numerical scheme of higher order accuracy.

9.2 Thermohydraulics Related to Reactor Safety

Thermohydraulic studies have been conducted for evaluating the physical phenomena and integrity of the reactor fuel elements during 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 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 a single fuel subassembly and inter-subassemblies, and plenum-channel thermohydraulic interactions in the mixed convection regime, (2) LOPI simulation experiments on the PLANDTL facility, (3) code development for the subassembly and reactor core heat transfer analysis and iv) development of the plant system dynamics analysis code.

The inter-subassembly heat transfer experiments has started using the Core and Component Thermohydraulic Test Loop (CCTL) which was completed in December 1987. The plenum-channel interaction study is being carried out in cooperation with Massachusetts Institute of Technology. After completion of the PLANDTL facility in September 1987 and the following shakedown operation of the facility, preliminary runs for the Monju LOPI simulation experiments were carried out. The LOPI simulation experiments and their evaluation will be continued up to JFY 1991.

Efforts on thermohydraulics and safety analysis code development and validation are continued for the subchannel analysis codes ASFRE and SABENA and for the plant system dynamics analysis code SSC. The SABENA code was subject to implementation of (1) noncondensible gas two-phase flow model to assess fission gas release phenomena in an irradiated fuel subassembly and (2) a high-order accurate numerical scheme, and experimental validation was carried out with emphasis on the assessment of physical models used in the two-fluid subchannel formulation. The Super System Code SSC consists of two versions, i.e., SSC-L for loop-type FBRs and SSC-P for pool-type FBRs. Both versions were upgraded with the addition of a two-dimensional analysis capability for the reactor plenum. Also improvements were made with respect to the reactivity feedback models. The code was extensively validated against a number of out-of-pile as well as reactor experiments.

9.3 Degraded Core Study

The degraded core study addresses the fuel subassembly failure propagation in local fault accidents and the invessel physical processes of FBR severe accidents.

The local faults study focused on the SCARABEE in-pile test analysis, analysis code development for reactor application and validation with experimental data. The SCARABEE PV-A test, the last PNC test planned for September 1989 to examine failure propagation to neighboring fuel subassemblies, has been and analyzed to predict melt formation and its control for satisfying both the experimental objectives and safety requirements. The analytical code development and validation efforts were made on: (1) faulted subassembly behavior where the bundle disruption and melt formation are the key phenomena to be pursued; and (2) fuel failure propagation where the melt ejection or penetration is of primary interest to discuss the thermo-mechanical consequence in the adjacent fuel subassemblies. Regarding the faulted subassembly analysis, SABENA was applied to examine the boiling incoherency over the pin bundle and the subsequent thermal evolution to pin bundle failure. A MELTPIN model was developed to represent pin disruption and material relocaion, and MOPOS to molten pool thermal behavior and thermal loading to the hexcan wall. For the subassembly failure propagation analysis, a code, SCION is being developed to simulate melt ejection to neighboring subassemblies.

In the PNC out-of-pile test activity, VECTORS, energetics mitigation experiments for the post-disassembly expansion phase, completed the series to examine the effects of an upper core structure, In the material relocation and interaction studies, the second series of the JET-I tests focused on the molten jet/structure interaction phenomena to obtain physical understanding and analytical models to predict plate erosion behavior with varying jet temperature, diameter, flow velocity, nozzle to plate distance, impingement angle, plate material etc. The study moved to MELT-II tests in the intermediate temperature range with molten salt jet and tin plate. Substrate erosion behavior was formulated with generalized erosion depth based on the laminar and turbulent heat transfer models with consideration of crust formation and liquid film effect. This activity has moved to metal melt series with stainless steel, and ceramics melt series with Al_2O_3.

The CABRI in-pile activity focused on the international synthesis work for CABRI-I experiments. The synthesis work and related PNC analysis were completed. The follow-on CABRI-Ⅱ activity was initiated on pre- and post-test analyses.

Efforts on the whole-core accident code development and application were continued on the Monju reactor analysis, code modification and validation. The information associated with initiating phase energetics was fed to SAS3D modifications in material motion upon fuel disruption, pin failure models and fuel characterization. Regarding the core disruption phase, AFDM verification and planning of SIMMER-Ⅲ development plan were continued; a computer program called Bubble Behavior Code (BBC) was developed to simulate a CDA bubble in the post-disassembly expansion phase.

9.4 Plant Accident Study

In the area of sodium fires and aerosols, a pool combustion test in a 3% oxygen atmosphere was successfully completed. Tests on water release from low-temperature concrete were started to evaluate water release following a sodium leak accident. The released water would generate hydrogen due to sodium-water reaction. Sodium aerosol calculations using CONTAIN were carried out for the EC benchmark problem. A summary of the activities was presented at the IAEA, IWGER Specialists' Meeting on Sodium Fires and Aerosols, Obninsk, USSR, June 1988. Tests to investigate fission product release from sodium to gas and the feasibility study of hydrogen combustion test have been started in the area of source term research. The partition coefficients of iodine, cesium, and tellurium between liquid sodium and gas phases were determined.

The ex-vessel severe accident studies are focused on the evaluation of the Monju severe accident analyses. To support this, development, verification, and test of the CONTAIN code are continued.

9.5 Steam Generator Safety Study

Objectives of the R&D activities for steam generator safety can be divided into two: (1) the design and the safety evaluation of the conventional steam generators, and (2) that of an LMFBR having double-wall tube steam generators in the primary heat transport system. As to the objective (1), establishment of the wastage date base of high-chromium steels was completed. In regard to the code development, modification of the large sodium-water reaction analysis code, SWACS, was continued. In the meantime, validation of the code was carried out using the results of the LLTR test conducted in the U.S. A water test to simulate the large leak accidents was also carried out to obtain validation data of the code.

9.6 Research on Probabilistic Safety Assessment

As part of the research and development on the prototype reactor Monju, PSA has been performed since 1982. The object of this study is to construct a probabilistic model for the Monju plant so that overall safety assessment can be performed.

A code network system including the SETS code has been developed to perform systems analysis. The network was almost completed. An event tree analysis support program was developed using the technique of the expert system. Effort has been made on the development of the QUEST code which performs level-1 PSA on a PC. The other systems analysis codes developed in-clude the time-dependent unavailability analysis program, the human reliability analysis support program and so on.

Data development effort is being made for the LMFBR components based on CREDO (Centralized Reliability Data Organization), the cooperative project between PNC and USDOE. The date collection work for the PLANDTL facility was started. The CREDO file management system is being developed to manage the chronological versions of the date base. Also, some programs were developed to support the utilization of the CREDO data base. Quite recently a preliminary common cause data analysis was attempted using the CREDO data base.

As part of the external events evaluation screening analyses were performed on the spatial dependent failures such as a leak of coolant and fire. In addition, a preliminary analysis of seismic events was conducted in order to establish the methodology of seismic evaluation and to prioritize the components whose fragility is to be estimated in detail.

In the area of the systems, the modeling assumptions and conditions such as success criteria and grace time for recovery were reviewed and some of them were modified based on the new results of plant dynamics analysis in order to make more realistic evaluation. The overall utilization of the CREDO data base for the systems analysis was started this year. Failure date concerning all failure modes of sodium components were obtained from the CREDO data base and used in the quanti-fication of accident sequences. Further, uncertainty and importance analyses including some sensitivity studies were performed.

Analyses of the plant dynamic response of the Monju reactor in PLOHS, LORL and ULOHS accidents were almost completed. The results were summarized to re-evaluate the core damage frequency. The accident sequences were categorized for the in-vessel physical processes.

The accident energetics and the meltdown processes in JLOF were re-evaluated based on the improved theoretical models and computer codes. The ULOF in-vessel sequences were categorized and the primary source terms were summarized to give the initial conditions for the analyses of ex-vessel physical processes. The parametric study of the UTOP and ULOHS accidents has been continued based on refined analytical conditions and models.

The main effort is focused on demonstrating that the risk from these accidents can be sufficiently enveloped by that from the ULOF accident. The re-evaluation of the LOHS in-vessel sequences is in progress, which includes the whole in-vessel physical events such as pre-protected-meltdown, protected-meltdown, recriticality, post-accident material relocations, failure of the primary boundary, source terms and sodium leakage from a damaged boundary.

The delineation study of the ex-vessel physical processes has been continued. The ex-vessel event-trees have been constructed for the two types of severe accidents, ULOF and PLOHS accidents. The ex-vessel analyses will be continued and the results will be summarized to obtain the final risk curves.

10. Fuel Cycle

10.1 MOX Fuel Fabrication

1) PFDF and PFFF

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) (10t MOX/year) and FBR(1t MOX/year). It has supplied the fuel necessary for the operations of ATR Fugen and Joyo. About 102 tonnes of MOX fuel have been fabricated by the end of March 1990.

2) Fabrication at PFPF

In parallel with the construction of Monju, construction of the Plutonium Fuel Production Facility (PFPF) (FBR line; 5t 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.

To provide MOX fuel for ATRs, PNC is planning to construct a new ATR line (40 tonnes MOX per 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 5t MOX/year of FBR line is so designed as to increase the capacity to 15 MOX/year by adding the process equipments, to cover the fabrication of fuels for DFBR.

3) R&D on MOX Fuel Fabrication

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 approach to nuclear materials. It was achieved through various experiences at PFFF.

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

10.2 Plutonium and Uranium Conversion

PNC developed a co-conversion technology using (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: 10kg MOX/d), designed for demonstration of the co-conversion technology by the microwave heating denitration process (MH method), was constructed in February 1983. By the end of February 1990 it produced about 3,800kg of MOX powder using 1,600kg of plutonium. The converted MOX powder were transported to PFFF and PFPF, in addition to 1,791kg 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 mass production technique by the MH method.

11. FBR Fuel Recycling

In the area of FBR fuel reprocessing, PNC has developed process and equipments with remote handling technique, through large-scale cold mock-up tests at the three Engineering Demostration Facilities, EDF-I, II , and III , and laboratory-scale hot tests at the Chemical Processing Facility (CPF) in Tokai Works of PNC, on the basis of accumulated experiences 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.

11.1 Process Research and Development

About 6.5kg of irradiated fuel with burn-up from 4,400 MWD/T to 54,700 MWD/T, mainly received from Joyo by 1986, were used for hot laboratory tests at CPF. The fuels irradiated up to 94,000 MWD/T at Phenix in France has also been processed since October 1986.

Various engineering tests on process equipment have been carried out at EDFs since its establishment in 1982.

1) Head-End Process

PNC has adopted a laser-beam disassembling machine. The prototype machine removes hardware of FBR fuel assembly from the fuel pins efficiently.

Tests for improvement of remote maintainability, verification of aerosol produced by laser, and compaction of it are now proceeding.

The prototype machine of a continuous rotary dissolver, which was adopted by PNC considering larger capacity and higher economical efficiency which would be required in future will be tested in near fature.

The development of centrifugal clarifier and basic hot dissolution tests are also in progress at EDF-II and CPF, respectively.

2) Chemical Separation Process

PNC developed centrifugal contactors and continued the performance test since 1986.

In parallel with them, an engineering scale electro conductive pulsed column test apparatus, using plutonium, began to operate at the beginning of 1990.

3) Common Technology

Development of remote system technology to establish remote maintenance concept with rack module system for the Recycle Equipment Test Facility (RETF) and a future plant is underway at EDF-Ⅲ . A new prototype of two-armed, elbow-down type bilateral servo-manipulator is also being tested.

Other remote systems, such as standardized rail-in type racks, remote connectors bank for rack system, remote sampling systems, optical fiber signal transmission systems, are also under development.

Materials of process equipment, and manufacturing techniques have also been studied. Hot corrosion tests are continued in CPF.

On-line analytical systems for plutonium, uranium, acid concentration, and gamma-nuclide are being developed since 1986 aiming at quick analysis, automated operation, high reliability, and reduction of waste and exposure dose.

11.2 Plant Design of Recycling Facilities

1) Recycle Equipment Test Facility (RETF)

Results of recent analysis based on existing R&D experiences proved that high availablity and further economical technology is essential to realize commercial FBR fuel recycling.

It needs adoption of new advanced process concept with hot engineering demonstration of some process unit components to expect steady operation of the future pilot plant. From this viewpoint, PNC has carried out the design of Recycle

Equipment Test Facility (RETF), equipped with key components and process with irradiated FBR fuel for FBR fuel reprocessing.

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

After verification of process components, these technologies will be applied to the pilot plant.

2) FBR Fuel Recycling Pilot Plant

The purpose of the FBR Fuel Recycling Pilot Plant is to demonstrate the whole plant availabiliby and to evaluate the economical efficiency of FBR fuel reprocessing.
It is planned to start operation soon after 2000.

12. Waste Management

12.1 High Level Waste Management Program

High-level radioactive waste, contains both highly radioactive nuclides with short half lives and those nuclides with lower radioactivity and longer-half lives. In June 1987, the Atomic Energy Commission set forth a Long-Term Program for the Development and Utilization of Nuclear Energy. In this Program, it is stated that "high-level radioactive waste is to be solidified in a stable form and stored for cooling for appro-priate to isolation, and then finally isolated into deep underground formation."

Following sequences are to be considered for such geological isolation.

1) Solidification by borosilicate glass.
2) Store for cooling for 30 to 50 years to mitigate the effect of decay heat.
3) To isolate into deep geological formation with multi-barrier system consists of engineered and natural barriers.

The most important consideration which should be taken into account with regard to high-level radioactive waste isolation, is that the waste should eventually be isolated rather than maintained in long-term storage under human control, based on the following points fo view:

1) Technology will be established in the near future to ensure safe isolation utilizing multibarrier system.
2) High-level radioactive waste (HLW) should be isolated under the responsibility of the present generation which receives the benefits from nuclear power to minimize burden upon future generations.

The purpose of the present research on geological isolation is to prove the long-term safety of the multibarrier system to obtain public understanding.

Japanese policy for R&D on geological isolation is as follows:

1) Considering such a research and development program on geological isolation is time-consuming and comprehensive, it should be implemented in a consistent and fexible manner.
2) Research and development should be carried out on the variety of multibarrier systems which accommodate the diverse condition of geological environment in Japan.
3) Safety of geological isolation depends much upon near-field permormance. On the other hand, far field performance provides an extra margin of safety. From this standpoint, priority should be given to research and development on the near-field while steadily advancing research on the farfield phenomenona.

Power reacter and Nuclear Fuel Development Corporation (PNC), which is the central institution responsible for research and development, should compile progress reports at appropriate times to provide information to the general public for their understanding on geological isolation. These reports should be evaluated by competent authorities for further implementation of the research and development program.

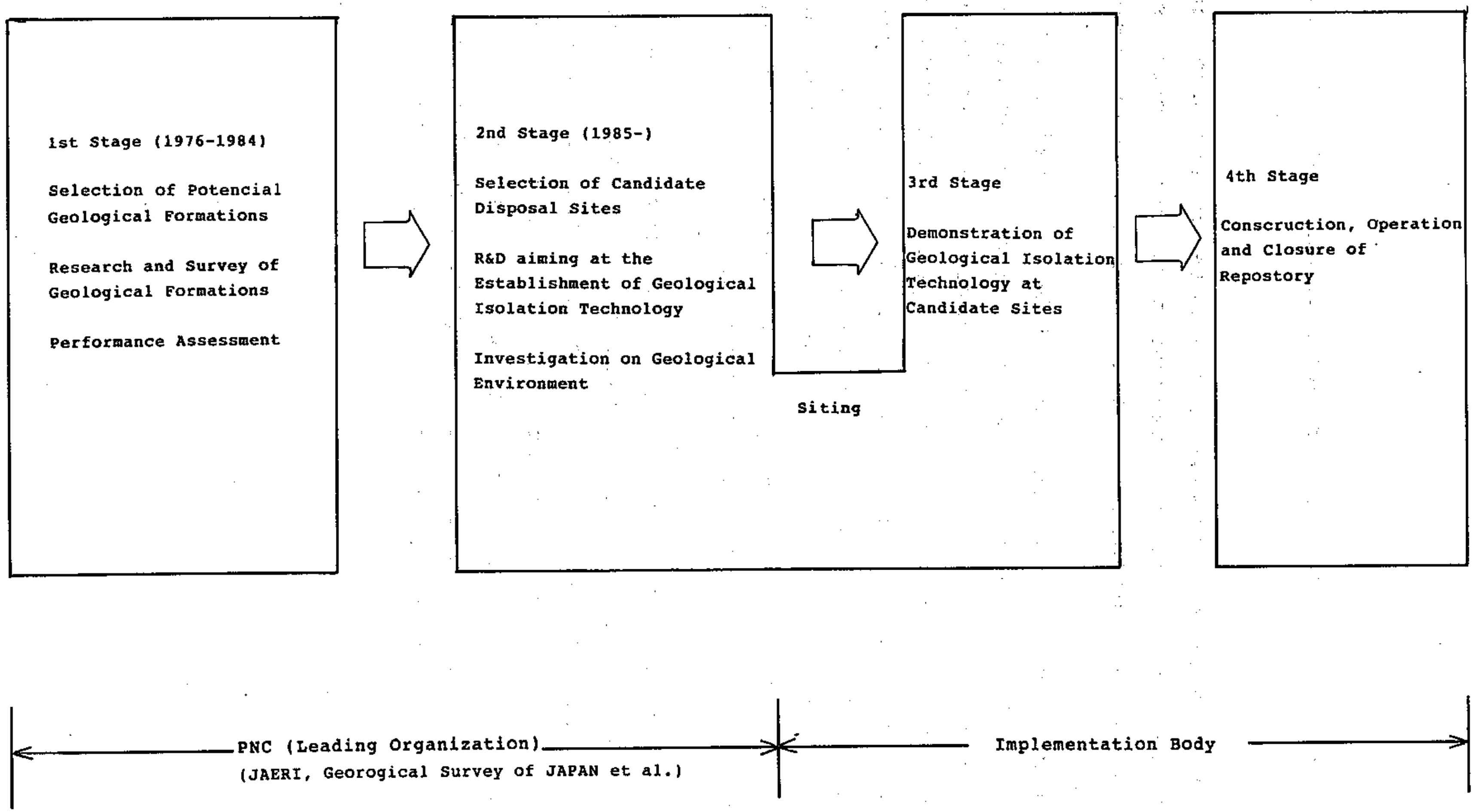

Fig. 14-1 National Program for Geological Isolation of HLW

12.2 Vitrification of High Level Liquid Waste

The technology development of vitrification has been continued sine 1975 in combination of cold engineering tests, full-scale mock-up tests and hot laboratory tests at Tokai works for PNC.

The PNC vitrification process comprised of liquid fed ceramic melter (LFCM) has demonstrated its good performance and reliability at Engineering Test Facility (ETF). The remote handling/maintenance techniques together with the system reliability was demonstrated at Mook-up Test Facility (MTF).

The vitrification test using actual HLLW from the Tokai Reprocessing Plant started in December 1982 at Chemical Processing Facility(CPF), where about one liter of waste glass was produced per batch. The gamma-scanning of the canister and the microscopic observation were performed to examine uniform distribution of some nuclides in the waste glass. Characterization of the waste glass properties is now underway.

These tests focused on providing detailed data for designing the Tokai Vitrification Facility (TVF) for the Tokai Reprocessing Plant..

The operation license of the TVF was granted February 1988, and the construction started in June 1988. The treatment capacity is equivalent to the reprocessing plant (0.7 ton of HM/day), and TVF employs fully remote operation and maintenance techniques in the large vitrification cell and all the equipment in the cell are designed in compliance with standardized rack-mounted modules.

The reception of HLLW from the Tokai Reprocessing Plant are scheduled to start in 1992.

The LFCM method developed by PNC was adopted by Japan Nuclear Fuel Services Cooperation in its commercial reprocessing plant planned to start construction in 1991. PNC will continue to provide its technical support and cooperation to the private sector.

Increased life of glass melter, which is the core of vitrification procese, and reduction of the size will lead to improvement of the economy and reduction of secondary waste genertion. In this context, PNC is developing refractory materlals of ceramic melter and a new heating process to produce a high-performance compact melter.

The research and development on the storage technology was focused in cooling system, durability and a seismic characteristics of the facility. A conceptual design of a high-level vitrified radioactive waste storge plant, based on these results, is also under way.

12.3 Major R&D Projects on Geological Disposal

The major projects being carried out by PNC are summarized as below.

1) Performance Assessment Research

· PACE program (integrated near-field performance assessment study) (Performance Assessment Center for Engineered Barriers)

The integration of assessment codes for the time dependence of geochemical conditions in the near field of radioactive waste; corresponding individual phenomena such as the corrosion of containers, the solution and migration of nuclides; and the development of an assessment code for the release of nuclides through an engineered barrier. Thermodynamic-data acquisition is underway regarding the solubility of actinides.

· ENTRY program (Engineering Scale Test and Research Facility)

The overall objectives of the ENTRY program are to conduct relatively large-scale, nonradiogenic, laboratory experiments and model development to support performance assessment for a high-level waste repository.

The ENTRY program scope is broad to include the following activities related to the performance assessment of geological disposal.

- development, evaluation, and validation of performance assessment models;
- collection of nonradiogenic laboratory data;
- development of data base for performance assessment and;
- communication of scientific results to the technical community and public.

2) Geo-scientific research

From the geo-scientific viewpoint, several subterrestrial phenomena, such as underground hydrology, geochemical phenomena, nuclide migration, rock mechanics, and thermal behavior, have been studied.

- KAMAISHI IN-SITU EXPERIMENT

An integrated In-Situ experimants in granitic rocks at the Kamaishi iron mine.

- TONO SHAFT EXCAVATION EFFECT STUDY

A study of the shaft-excavation effect on hydrology and rock mechanics in the vicinity of a shaft excavated in sedimentary rock at the Tono Mine.

3) Geological environmental research

Surface and subsurface geological investigation have been executed. Potential geological formations are classified into areas with common properties of geological environments from the viewpoint of type of rocks, age of rocks, structural features,etc.. Geological data base system has been developed for data applications obtained from these activities. The data base will facilitate effective assessment of the characteristics of the geological environment in Japan, and to provide a model data set for performance assessment.

4) International cooperative research

In addition to the above, PNC is involved in the international and bilateral collaboration as below.

- OECD/NEA International Stripa Project
- OECD/NEA International Alligator Rivers Analogue Project
- AECL Canada
- SCK Belgium
- NAGRA Switzerland
- PNL US

12.4 Low-Level TRU Bearing Waste

The principles issued by the Atomic Energy Commission (AEC) of Japan on the treatment of low-level wastes are summari-zed as follows:

Efforts must be made to minimize the generation of low-level radioactive wastes, and to process the generated wastes reducing their volume and solidifying them into stable form.

PNC, in recognition of the above AEC policy and the fact that PNC itself generates radioactive wastes, is making effort to ensure the safe storage of radioactive wastes as well as to develop the radioactive waste treatment technologies for waste volume reduction and solidification. By now, PNC has nearly achieved its preliminary goal to support stable operation of fuel cycle failities such as the Tokai Rrocessing Plant.

Transuranic (TRU) bearing liquid waste treatment technology, such as bituminization for low-level liquid concentrate, and plastic solidification for spent solvent, has been also developed through hot operation at waste trestment facilities at PNC Tokai Works. After a series of hot operation, 14,000 drums of bitumen and 500 drums of plastic have been produced by August 1989.

Effort to remove plutonium-contaminated waste from fuel fabrication facilities at PNC involves the incineration of combustible materials, cyclone incineration of organic chlorinated wastes, electroslag remelting of metal wastes, and solidification of non-combustible wastes into ceramic-like form by microwave heating. These techniques have been demonstrated at the Plutonium-contaminated Waste Treatment Facility (PWTF) which started operation in 1987.

Characterization of wastes plays an important role to study low-level waste, disposal system and PNC implemented tests for characterization of hot solidified wastes.

Development of non-destructive assay techniques for identification and measurement of radionuclides in solidified wastes, which is essential for establishment of basic data base of the source term are also underway.

The low-level TRU bearing waste are categorised as non-TRU waste after the decontamination. This leads to a considerable reduction on the amount of TRU wastes to be disposed. From this view point, PNC is now developing the technolgy to achieve low-level liquid concentrates and spent solvents from reprocessing, as well as the decontamination technology for solid wastes.

Construction of a new Low-Level Waste Treatment Facility (LWTF) is planned for stabilization of low-level TRU containing waste and for hot engineering test of nuclides separation technology.

The techniques for decontamination and dismantling of large-size wastes contaminated with TRU nuclides have also been developed in PNC, and it will be applicable to the decommissioning of nuclear fuel cycle facilities. These include mixed ice and dry ice blasting technique, electropolishing technique and redox technique for decontamination, and plasma cutting robot, laser cutting technique for dismantling.

These techniques have been developed and partly demonstrated at the Waste Dismantling Facility (WDF) at PNC O-arai Engineering Center since 1984.

12.5 Radioactive Waste Storage Research Center

PNC is promoting a program to open a radioactive waste storage research center at Horonobe in Hokkado for the purpose of storing vitrified HLW and immobilized LLW, and of development of waste management technology including utilization of heat and radiation from HLW.

The environmental survey report including the results of boring and geological investigation at candidate site was prepared and efforts to get local public acceptance are underway.