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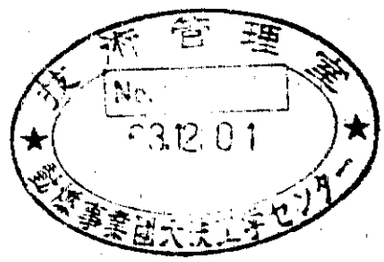
A REVIEW OF FAST BREEDER REACTOR PROGRAM IN JAPAN



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日本の高速増殖炉開発レビュー

編集

沢井　定*
堀　雅夫*　柚原　俊一**

要　旨

我が国の原子力委員会は昭和62年6月，5年振りに「原子力開発利用長期計画」を改訂した。本計画では高速増殖炉の基本的な開発政策についても見直しし，以下の主要政策を明示している。

(1) 軽水炉に引き続く主要な原子力発電プラントとしての高速炉の使用

(2) 高速増殖炉を中心とするプルトニウム利用体系の確立

高速増殖炉開発の展開については，(1)もんじゅ開発の推進，(2)実証炉1号開発の推進，(3)高度技術の開発及び(4)革新技術の開発を主要な柱としている。

これらの開発戦略に基づき，「もんじゅ」及び「常陽」を中心とする高速増殖炉の研究開発は動燃を主体として進められている。

高速実験炉「常陽」は順調に運転を継続し，高速原型炉「もんじゅ」の建設は計画通りの進捗で，昭和63年10月末で57.1%の進捗率である。

高速実証炉DFBR-1については，設計及び建設の主体である原電，研究開発の主体となる動燃，電中研，原研，これらの4機関の協力のもとに設計研究及び所要の研究開発が推進されている。

高速増殖炉の要素技術となる研究開発については，安全，燃料，構造・材料，炉物理，プラント機器及びシステム等の種々の分野で進められている。

本レビューは，第6回，日-仏，独，高速炉技術協力レビュー会議(The 6th PNC-CEA, KfK/Interatom FBR Technical Cooperation Review Meeting, Nov. 28～30, 1988 at CEA HQ/Paris)及び第11回，日英高速炉協力レビュー会議(The 11th JAERI, PNC-UKAEA FBR Cooperation Review Meeting, Dec. 8, 1988 at Risley)で紹介される。

*動力炉研究開発本部
**大洗工学センター

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A REVIEW OF FAST BREEDER REACTOR PROGRAM IN JAPAN

Editors:
Sadamu Sawai*,
Masao Hori* and Shunichi Yuhara**

Abstract

The "Long-Term Program for Development and Utilization of Nuclear Energy" was revised in five years by the Japan AEC in June 1987, in which the basic development policy for FBR in Japan was also renewed. The following policy is clearly shown in the Program.

(1) Use of FBR as Principal Nuclear Power Generation System following LWR

(2) Establishment of Pu Utilization System Mainly Using FBRs

The evolution of FBR development is aimed as follows; (1) promotion of "MONJU" Project, (2) promotion of Demonstration FBR-1 and associated R&D, (3) R&D of improved technologies for FBR and (4) R&D of innovative technologies.

On basis of the development strategy above, FBR's R&D associated with "MONJU" and "JOYO" are promoted mainly by PNC.

The Japanese Experimental Fast Reactor "JOYO" has been continuing the successful operation since it achieved the rated power of 100 MWt in 1982. Regarding the prototype fast breeder reactor "MONJU", plant construction is progressing in accordance with the schedule. Its overall progress attainment in construction is 57.1% at the end of October in 1988.

The Japanese Demonstration FBR-1 is under design studies and preparations for the selection of the plant basic specifications. The Japan Atomic Power Company (JAPC) was decideded to play a principal role in design and construction. Design studies of the Demonstration FBR-1 and associated R&D are promoted under cooperation of JAPC, PNC, the Central Research Institute of Electric Power Industry (CRIEPI) and the Japan Atomic Energy Research Institute (JAERI).

Efforts on R&D of the common technologies for FBRs have been continued for solution of reactor key issues in such fields as safety, reactor physics, fuel, structural design and material, plant component and system, etc.

This review was perpared for the 6th PNC-CEA, KfK/INTERATOM FBR Technical Cooperation Review Meeting held at the CEA headquarter in Paris on Nov. 28~30 in 1988 and for the 11th JAERI, PNC-UKAEA FBR Cooperation Review Meeting held at Risley on Dec. 8 in 1988.

* Reactor Research Development Project
** Oarai Engineering Center

A Review of Fast Breeder Reactor Program in Japan

Contents

1. Introduction 1
2. Experimental Fast Reactor, "JOYO" 3
2.1 Overview 3
2.2 Irradiation Test with INTA-S 4
2.3 Special Test using Fission Product Source 4
2.4 MK-III Core Program 5
3. Prototype Fast Breeder Reactor, "MONJU" 10
3.1 Overview 10
3.2 Licensing and Construction 10
4. Demonstration Fast Breeder Reactor, "DFBR-1" 15
4.1 Overview 15
4.2 Design Study 15
5. Reactor Physics 18
5.1 Benchmark Tests of JENDL-3T 18
5.2 Analysis Method Development 18
5.3 Critical Experiment and Analysis 19
5.4 Benchmark Shielding Experiment 20
5.5 Core Performance Test at "JOYO" 21
5.6 Advanced Core Design Studies 21
6. Reactor Components 24
6.1 Reactor Vessel and Internal Structure 24
6.2 Shield Plug 24
6.3 Primary Pump 25
6.4 Intermediate Heat Exchanger 25
6.5 Control Rod Drive Mechanisms 26
6.6 Refueling and Fuel Storage System 26
6.7 Ex-Vessel Storage Tank 26
6.8 In-Service Inspection Equipment 27
6.9 Expansion Joint for Piping System 27
7. Steam Generator System 28
7.1 Sodium-Water Reaction Studies 28

7.2 High Chrome Steel Steam Generator 29
7.3 Double-Walled Tube Type Steam Generator 29
8. Sodium Technology 30
8.1 Material Tests in Sodium 30
8.2 Flow and Heat Transfer 30
8.3 Radioactive Material Behaviour and Control in Sodium 31
8.4 Sodium Chemistry and Sodium Purification 32
8.5 Sodium Removal and Decontamination 32
9. Instrumentation 34
9.1 Nuclear Instrumentation 34
9.2 Failed Fuel Detection and Location 34
9.3 Early Warning System for Fuel Failure 34
9.4 Process Instrumentation 35
9.5 Sodium Leak Detection System 35
10. Fuels and Materials 37
10.1 Fuel Fabrication Development 37
10.2 Fuel Pin Performance 38
10.3 Core Material (Improvement) 39
10.4 Subassembly 40
10.5 Irradiation Experiments 41
10.6 Development of Advanced Fuels 41
11. Structural Design and Materials 45
11.1 Development of Structural Design Methods 45
11.2 Structural Tests and Evaluations 46
11.3 Structural Material Tests 46
12. Safety 51
12.1 Thermohydraulics Related to Reactor Systems and Design 51
12.2 Thermohydraulics Related to Reactor Safety 51
12.3 Degraded Core Study 53
12.4 Plant Accidents Study 53
12.5 Steam Generator Safety 54
12.6 Research on Probabilistic Safety Assessment 56
References 59

List of Tables

Table	Title	Page
2.1	Major Milestones of "JOYO" and R&D Activities	6
2.2	"JOYO" Operational Experience Data (OCT. 1988)	6
3.1	Major Milestones of "MONJU" and Activities	11
3.2	Present Status of "MONJU" Construction	12

List of Figures

Figure	Title	Page
2.1	Experimental Fast Reactor "JOYO" Operating History	7
2.2	Instrumented Test Assembly (INTA-S)	8
2.3	Schematic Diagram of Experiment using Fission Product Source	9
3.1	General View of "MONJU" under Construction. (Oct., 1988)	14
4.1	Schedule for DFBR-1 Development	17
10.1	Aerial View of Plutonium Fuel Production Facility (PFPF) ...	44

1. Introduction

The "Long-Term Program for Development and Utilization of Nuclear Energy" was revised in five years by the Japan AEC in June 1987, in which the basic development policy for FBR in Japan was also renewed. The following policy is clearly shown in the Program.

(1) Use of FBR as the Principal Nuclear Power Generation System Following LWR,

(2) Establishment of Pu Utilization System Mainly Using FBRs

The evolution of FBR development is aimed as follows; (1) promotion of "MONJU" project, (2) promotion of the Demonstration FBR-1 (DFBR-1) and associated R&D, (3) R&D of improved Technologies for FBR and (4) R&D of innovative technologies.

On basis of the developmental strategy above, FBR's R&D associated with "MONJU" and "JOYO" are promoted mainly by the Power Reactor and Nuclear Fuel Development Corporation (PNC).

The Japanese Experimental Fast Reactor "JOYO" has been continuing the successful operation since it achieved the rated power of 100 MWt in 1982. The period of a duty cycle operation and the maximum burn-up of the driver fuels have been raised to 70 days and 75,000 MWd/t, respectively. MK-III program is planned for up-grade of irradiation core characteristics, innovative R&D and others.

Regarding the prototype fast breeder reactor "MONJU", its plant construction is pregressing in accordance with the schedule. The overall progress attainment in construction is 57.1% at the end of October in 1988. Major milestones achieved since the beginning of 1987 are as follows; completion of the containment vessel set-up, the reactor guard vessel and the reactor vessel installations and others.

In licensing of "MONJU", the approval on the safety examination for the reactor establishment was made in May 1983 by the Nuclear Safety Commission in Japan. Following the safety examination, applications for the licensing approval of the designs and construction methods have been made step by step.

DFBR-1 is under design studies and preparations for the selection of the plant basic specifications. The Japan Atomic Power Company (JAPC) was decideded to play a principal role in design and construction. Design studies of the Demonstration FBR-1 and associated R&D have been conducted under co-

operation of JAPC, PNC, the Central Research Institute of Electric Power Industry (CRIEPI) and the Japan Atomic Energy Research Institute (JAERI).

Efforts on R&D of the common technologies for FBRs have been continued for solution of reactor key issues in such fields as safety, reactor physics, fuel, structural design and material, plant component and system, etc. At the same time innovative technologies are also being developed. These efforts are mainly focused to reflection for large-scale fast breeder reactors in future.

In this report, current status of FBR development in Japan is reviewed in eleven major R&D fields; the experimental fast reactor "JOYO", the prototype fast breeder reactor "MONJU", the demonstration fast Breeder reactor "DFBR-I", reactor physics, reactor components, steam generator system, sodium technology, instrumentation, fuels and materials, structural design and materials, and safety.

2. Experimental Fast Reactor, "JOYO"

2.1 Overview

The operation history of "JOYO" is illustrated in Fig. 2.1. This report covers the activities of the experimental fast reactor "JOYO" from January 1987 to October 1988.

The 6th annual inspection was finished in September 1987. Work for maintenance or modification of the plant has been done, such system as cecium trap in order to enable the loop to trap cecium which will be released from failed fuel during the reactor operation, the cold trap (replacement) of the primary coolant system in order to get higher trapping efficiency and 2nd set-up (reloading) of on-line irradiation test assembly (INTA-S).

Following the 6th periodical inspection of the reactor, the 13th duty cycle operation started at the beginning of September in 1987. The reactor continued the operation of the 14th and the 15th duty cycles, starting at the end of November in 1987 and at the end of February in 1988, respectively. From the 13th duty cycle operation, both the period of a duty cycle operation and the maximum burn-up of the driver fuels, of which maximum were increased in new license from 45 days to 70 days and from 50,000 MWd/t to 75,000 MWd/t respectively, had been gradually extended by using modified driver fuels, named J2 fuel. Consequently, 70 days operation was attained at the 15th duty cycle, where the maximum burn-up of a driver fuel reached over 70,000 MWd/t.

A special test cycle operation, called the 15th cycle operation, was performed after the 15th duty cycle operation, starting in June 1988. The sensitivity calibration was conducted for the failed fuel detection system (FFD) which provides both the precipitation system of cover gas and the delayed nuetron detecting system and also newly installed cover gas on-line γ ray monitoring system, in this test cycle.

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

operation. The 7th periodical inspection of the reactor was begun in September 1988, and is now in progress. A newly developed Cover Gas Clean-up System (CGCS) will be installed during the inspection period. The reactor will commence the 17th duty cycle operation at the end of December in 1988.

Major milestones of "JOYO" and R&D activities are shown in Table 2.1 and its operational experience data (Oct. 1988) in Table 2.2.

2.2 Irradiation Test with INTA-S

An on-line monitored material irradiation test using Instrumented Test Assembly (INTA-S) has been successfully conducted since the 13th cycle in September 1987. It measures center temperaturs of metallic material test specimens due to gamma heating, boron carbide pellet etc. during irradiation.

The structure of INTA-S is shown in Fig. 2.2. This irradiation test is to be continued untill the 20th cycle in 1989.

2.3 Special Test using Fission Products Source

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

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

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

the flux tilting method using an online gamma-ray monitor was found out to to be very useful.

Schematic diagram of the experiment is illustrated in Fig. 2.3.

2.4 MK-III core program

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

Table 2.1 Major Milestones of "JOYO" and R&D Activities

- Milestones
  - ○ First Criticality 1977
  - ○ Installation of 2nd Core 1982
  - ○ 100 MWt Operation 1983
  - ○ MK-III Program (1992~)
- Present and Future R&D Activities
  - ○ Irradiation of Fuels and Materials
    Target Burnup ~ 200,000 MWd/t
  - ○ On-Line Irradiation Instruments
  - ○ Computer-Acided Operation Support System
  - ○ MK-III Program

Table 2.2 "JOYO" Operational Experience Data (OCT. 1988)

• Accumulated Reactor Operation time	34,001 hrs
• Accumulated Heat Generation	2,031,000 MW•hr
• Fuel Irradiation	2,554,662 MW•hr
Maximum Fuel Burn Up Achieved	70,000 MWd/t (Fuel element Average)
Number of Fuel Assemblies Discharged from Reactor Vessel	418 (Core Fuel)
• Number of Start-Ups	423 (Including Critical Test)
• Number of Annual Inspections	7

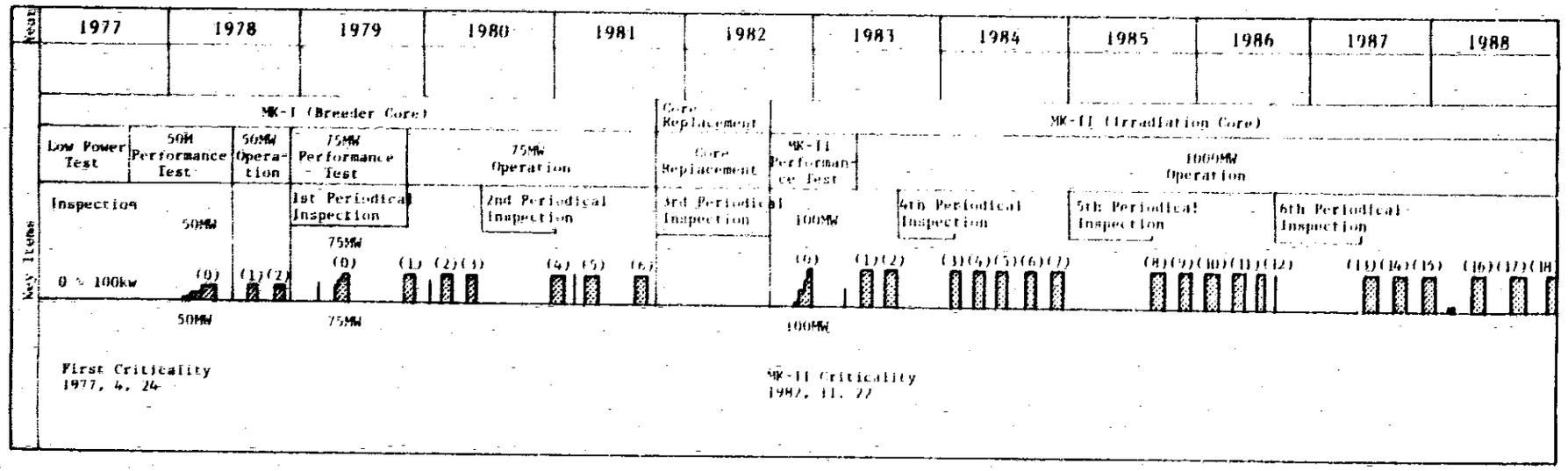


Fig. 2.1 Experimental Fast Reactor JOYO Operating History

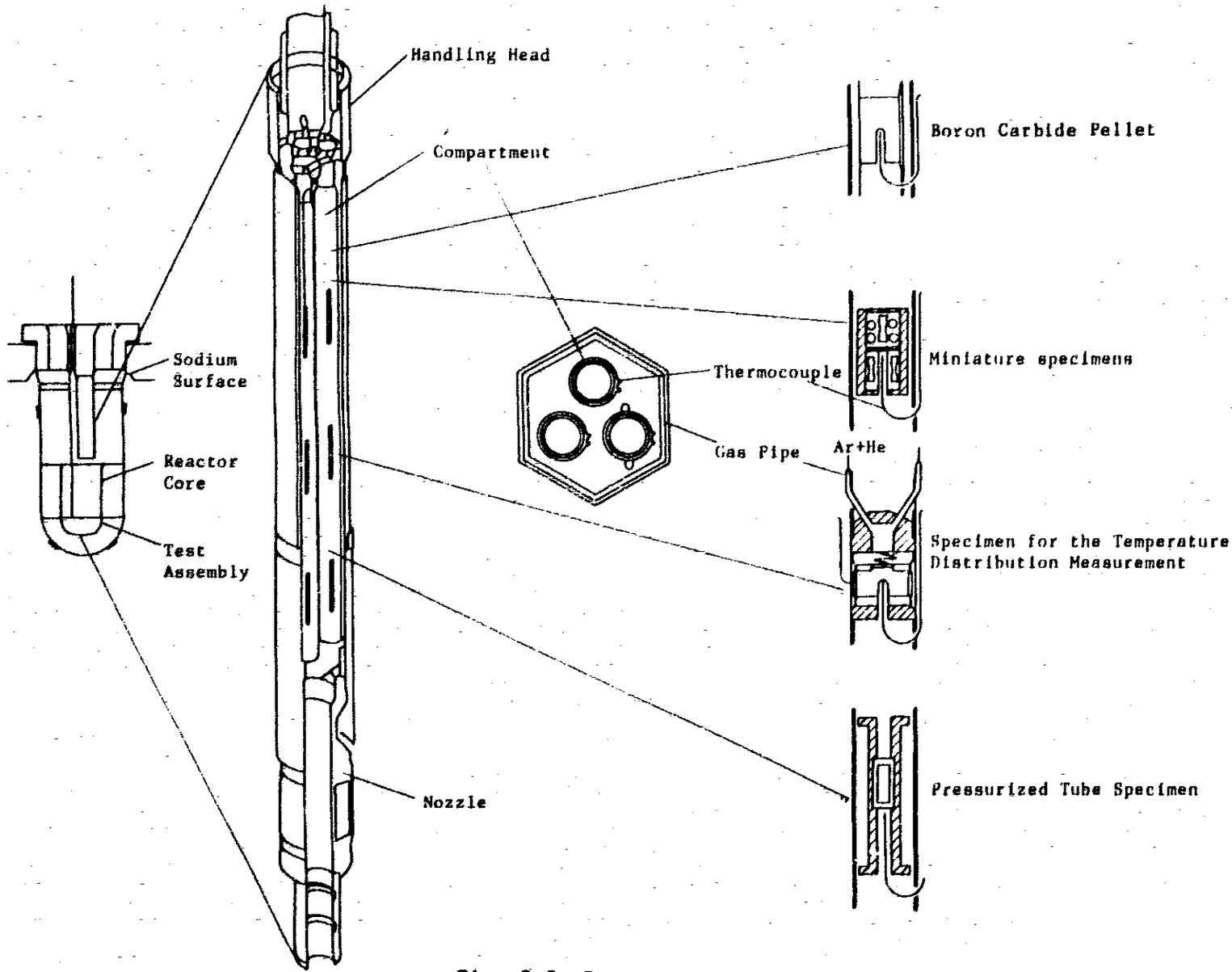


Fig. 2.2 Instrumented Test Assembly (INTA-S)

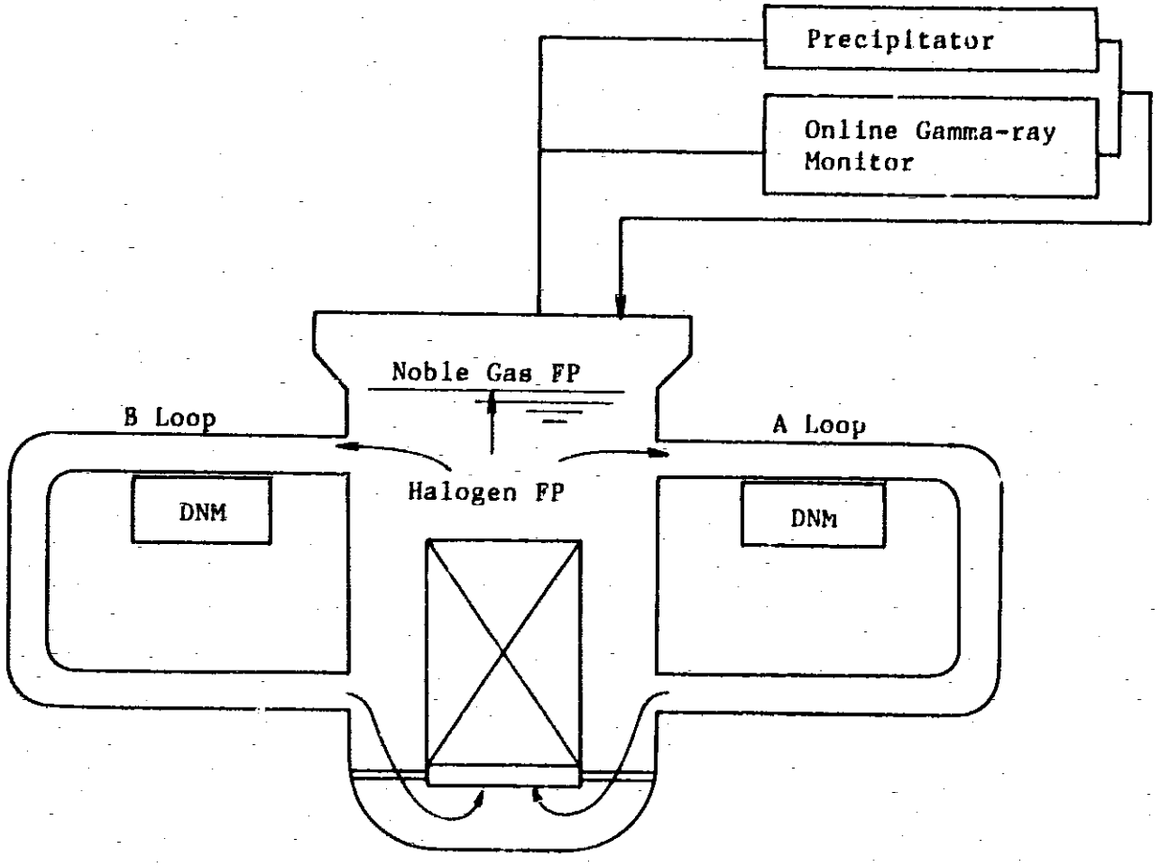


Fig. 2.3 Schematic diagram of experiment using fission product source

3. Prototype Fast Breeder Reactor, "MONJU"

3.1 Overview

The prototype fast breeder reactor "MONJU" is a nuclear power plant of medium size, which is being built at Shiraki in Tsuruga City, Fukui Prefecture, approximately 400 km west of Tokyo. "MONJU" is a prototype reactor, which stands between the experimental fast reactor and the demonstration fast breeder reactor in the development program of fast breeder reactors, and aims to attain technological advancement and economic prospect towards the establishment of commercial viability of future nuclear power plants.

The Power Reactor and Nuclear Fuel Development Corporation (PNC) undertook the design of "MONJU" and the selection of its site in 1968, and in 1970 shiraki area was chosen as a prospective site. Since then PNC conducted an environmental assessment and safety evaluation. The environmental assessment was completed by the local authorities of Fukui Prefecture in June, 1982, and by the government in July, 1982. The first stage of the safety examination was made by the reactor regulatory section of the Science and Technology Agency (STA) and the second by the Nuclear Safety Commission. The approval was granted for the reactor establishment in May, 1983. Preparatory construction work began early in 1983 such as preparations of the building in the site and the roads around the site.

In January, 1984 PNC contracted with four firms of the main component manufacturers (Toshiba, Hitachi, Fuji Electric and Mitsubishi Heavy Industry) for design, fabrication, and installation of a containment vessel, design and fabrication of a reactor vessel, and design of primary and secondary heat transport systems, steam generators, electrical systems, fuel handling system, etc., as the first stage. For implementing the project the manufacturers consigned the engineering and project management works to their joint firm, FBR Engineering Co., Ltd.

Major milestones and activities of "MONJU" project are shown in Table 3.1.

3.2 Licensing and Construction

PNC made applications for the licensing approval of the designs and

construction methods step by step. Reviews on the design and mthods of construction by STA and on the construction plan by the Ministry of International Trade and Industry (MITI) were initiated almost at the same time in December, 1984.

With the approvals of STA ROUND 6 & MITI ROUND 8 in August, 1987, all major heat transport systems and safety related systems & components have completed their licensing reviews. The final rounds of STA and MITI hearings are scheduled to resume in November, 1989 and expected to be completed in early 1990.

The overall progress of "MONJU" construction is 57.1% at the end of October in 1988. Major milestones achieved since January in 1987 are as follows; completion of the containment vessel set-up, the modular reactor cavity construction, and of the reactor guard vessel installation. The installation of the reactor vessel was started at the end of October in 1988. Present status of "MONJU" construction is shown in Table 3.2 and the general view of "MONJU" under construction in Fig. 3.1.

Table 3.1 Major Milestones of "MONJU" and Activities

Milestones

- 1st Safety Examination completed 1981
- Consent of Governments 1982
- 2nd Safety Examination completed 1983
- Preparatory Work Start 1983
- Excavation Start 1985
- Completion of Plant Installation (1991)
- Initial Criticality (1992)

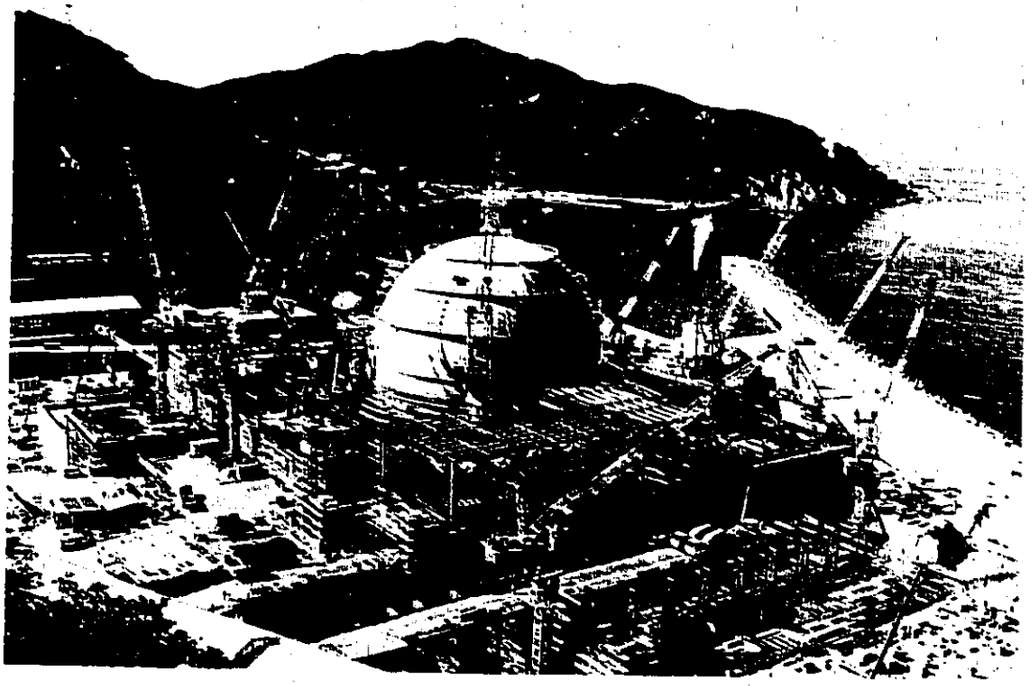
Activities

- Construction Works
- Licensing Works for Design and Construction
- Plant Functional Testing
- Commissioning
- R&D Performance Tests
  - Fuels, Control Rods, Structural Materials, Components
  - ISI System, Instrumentation
  - Others

Table 3.2 Present Status of "MONJU" Construction

Oct. 31, 1988

Item	Activity	Progress (%)
Civil Engineering Work	• Site Work • Landing Wharf and Revetment Construction Work • Cooling Water Intake Construction	88.7
Building	• Design and Construction of Reactor Bldg. and Reactor Auxiliary Bldg.	51.6
Reactor	• Design, Fabrication and Installation of Components • Construction of Reactor Cavity and of Liner System	55.8
Turbine Generator	• Design, Fabrication and Installation of TG Systems	35.5
Electric System	• Design, Fabrication • Construction of Electrical Systems	47.6
Total Work		57.1



3.1 General View of "MONJU" under Construction.
(Oct., 1988)

4. Demonstration Fast Breeder Reactor, "DFBR-1"

4.1 Overview

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

In this respect, it is requested to clarify a long-term strategy for FBR development by 1990, based on the result of research and development activities made so far and to be obtained during the next few years, along with the determination of the basic specifications of the first demonstration FBR (DFBR-1). At present, the start of construction of DFBR-1 is scheduled in late 1990's and the start of operation at the beginning of the 21st century. Schedule for DFBR-1 development is shown in Fig. 4.1.

4.2 Design Study

The utilities finished their three years program of the cost reduction design studies in March, 1987. 1000 MWe class loop type and pool type reactors were studied aiming at a construction cost less than 1.5 LWRs with the same level of safety and maintenability. The program was begun in 1984 by the FBR Project Office (FPO) of the Federation of Electric Power Companies (FEPC) and later transfered to the Japan Atomic Power Company (JAPC). The JAPC evaluated the cost reduction design studies adopting innovative and advanced technologies from the viewpoint of economies and competiveness with LWRs and has begun, based on the above results, the conceptional plant design study for further cost reduction of DFBR-1.

PNC has carried out the parametric design studies for large scale FBRs. Those works have been conducted not only to assist JAPC's DFBR-1 design studies, but also to extract the R&D subjects for the development of large scale FBRs including DFBR-1. A plant image of a large scale FBR was created,

which was composed of the key technologies for FBR cost reduction. These works are mainly based on the experiences of "JOYO", "MONJU" and R&D works.

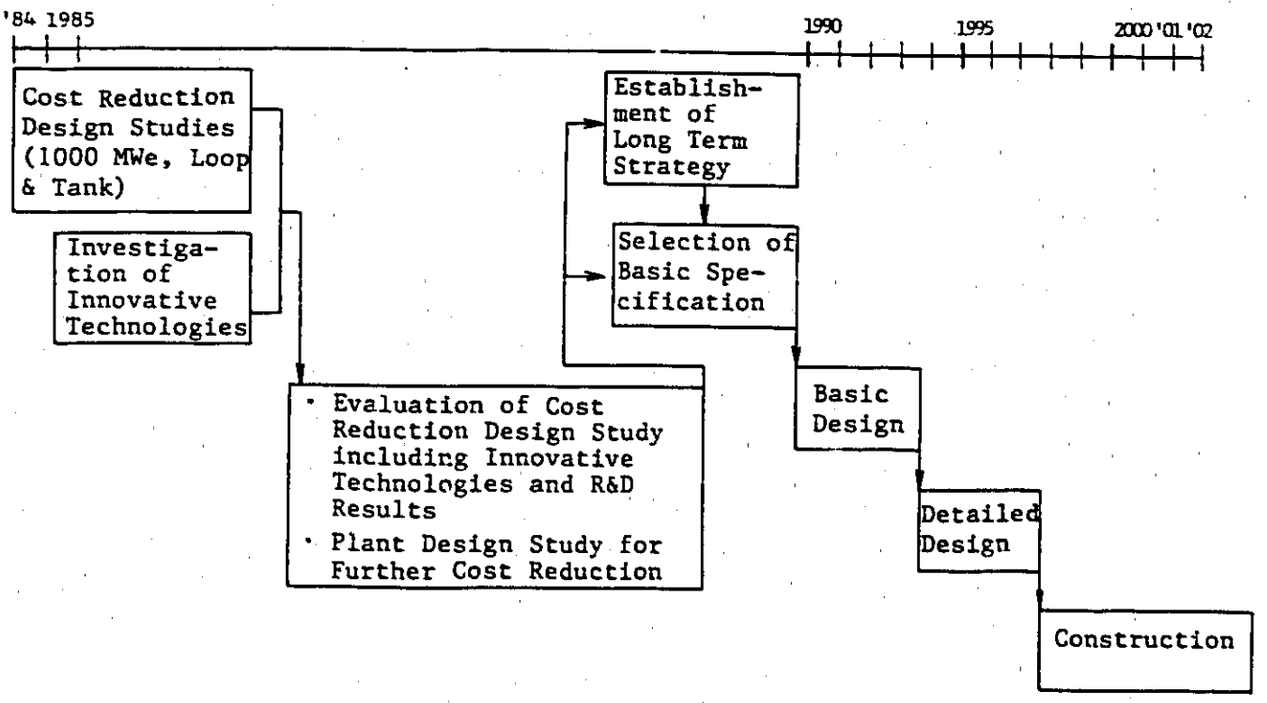


Fig. 4.1 Schedule for DFBR-1 Development

5. Reactor Physics

5.1 Benchmark Tests of JENDL-3T

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

As the results of fast reactor benchmark tests, the k_{eff} calculated with the JENDL-3T data was overpredicted for U-cores and underpredicted for Pu-cores. The reaction rate ratios of C8/F9 and F8/F9 were overstimated for the JENDL-3T data. On the other hand, Doppler and sodium void reactivities, and reaction rate distribution obtained by using the JENDL-2 data were significantly improved by using the JENDL-3T data.

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

5.2 Analysis Method Development

Two-dimensinal heterogeneity effects on cell-averaged cross sections and diffusion coefficients have been assessed. Both of the heterogeneity effects caused by the flux fine structure and the resonance shielding are taken into account. The two-dimensional cell heterogeneity effects on core performance parameters have been evaluated for the large fast critical assembly ZPPR-9 and the Zebra CADENZA plate-pin experiments.

A neutron streaming effect in an FBR fuel assembly was evaluated by using a double heterogeneous modeling and multigroup (70-group) calculations. The effect for a typical medium-size FBR was evaluated to be $-0.2\%\Delta k/k'$, while the effect was evaluated to be $-0.1\%\Delta k/kk'$ by a conventional modeling.

A three-dimensional discrete ordinates transport code TRITAC (Hex, 2) in hexagonal-Z geometry has been developed for solving eigenvalue problems in reactor cores. The code treats configurations which have 120 degree rotational symmetry and consist of equilateral hexagonal right prismatic cells. A simple test problem was solved and the results were compared with a two-dimensional R-Z calculation.

A three-dimensional transport/diffusion code TRITON has been developed. The TRITON code has been applied to transport and diffusion calculations in a simple 3-D (XYZ) fast reactor core model. The computing time was markedly reduced, compared with the CITATION/TRITAC system.

A new treatment of sodium void coefficient in the fast reactor transient analysis is proposed. The method is based on the multigroup exact perturbation theory and utilizes the void reactivity map constracted by the multigroup first order perturbation theory. The change in the spatial distribution of neutron flux is corrected by the ratio of the fluxes in the transient and the steady states calculated with few-energy groups.

The prediction uncertainty of neutronic performance parameters of a large FBR core has been estimated using the three methods, the bias factor method, the adjustment method and the combined method. The prediction uncertainties obtained are compared to know the method which is preferable to reduce the uncertainty.

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

5.3 Critical Experiment and Analysis

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

The first half of the JUPITER-III program, the ZPPR-17 program, was

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

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

5.4 Benchmark Shielding Experiment

A series of neutron penetration experiments were performed at the ORNL Tower shielding Facility as the first experiment ofJASPER, the joint experimental research program between US DOE and PNC.

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

The newly confirmed prediction accuracies of the analysis method made it possible to reduce the current margins for radial shields in Japanese DFBR designs.

5.5 Core Performance Test at "JOYO"

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

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

5.6 Advanced Core Design Studies

5.6.1 Heterogeneous Core

The axial heterogeneous core concept has been studied in Japan as a candidate for large LMFBR core. Trade off studies to optimize a core design and critical experiments at FCA were conducted on this concept. The concept was investigated in 1981 in the large LMFBR conceptional design study of PNC, and was also selected in 1984 to be a reference core of the cost reduction design study of the Japan Atomic Power Company (JAPC) on the pool type DFBR. In these studies, a neutronic performance of an-axial heterogeneous core, optimized with respect to breeding ratio, burnup characteristics, power flattening and so on, was calculated and compared with that of an equivalent homogeneous counterpart. As a result, it was observed that the axial heterogeneous core had a potential to have advantages over the homogeneous core in burnup reactivity loss, maximum fast neutron fluence, power peaking factor and the reactivity response to the movement of control rods in earthquakes, though it was accompanied by some disadvantages in other respects at the same time.

Design studies were performed to optimize the 1000 MWe AHC configuration with a view of obtaining more compact core systems, longer fuel lifetimes and better safety characteristics. In the present core design studies, a disk-shape internal blanket is located at the mid-plane of the core, or at the

lower part of the core. the purpose of the latter asymmetric arrangement of the internal blanket is to make the power distribution flat throughout the whole burn-up cycle, even when the control rods are partially inserted from the top of the core.

5.6.2 High Burn-up Core

High burn-up potential of various core concepts has been studied from the viewpoint of fuel cycle cost reduction and reactor availability factor improvement. In order to achieve a high burn-up core of 150-200 MWd/kg, it is necessary to develop long life fuels with low-swelling and high creep-strength materials. The Super Long Life Core (SILC) or Ultra Long Life Core (ULLC) concept has been evolved as one of the targets of the innovative approaches for FBR cost reduction. A neutronic feasibility study was done for the SLLC concept, where core life time was extended up to the plant life of about 30 years, by applying the radially and axially multi-zoned core concept with mixed oxide fuels.

5.6.3 Metal, Carbide and Nitride Fuelled Cores

Feasibility studies were started for the advanced fuel loaded FBR cores, i.e., metal, carbide and nitride fuelled cores. The nuclear and safety characteristics of a large metallic fuel core have been evaluated in comparison with the mixed oxide core. Preliminary safety calculations indicate that higher thermal conductivity of metal fuel enhances the inherent safety of the reactor with respect to loss of flow without scram and loss of heat sink without scram. design and safety evaluation will be continued for the metal fuelled core.

Carbide and nitride fuels have also potentials to be used in future FBR cores. High thermal conductivity and high content of heavy metal of these fuels yield high linear heat rate and high breeding ratio. Nuclear and safety characteristics of carbide and nitride fuels, as well as fuel cycle and economic implications, have been evaluated, and will be compared with those of oxide metallic fuels.

5.6.4 FP Gas Purge/Tube-in-Shell Type Metallic Fuel Fast Reactor

To realize very hard neutron spectrum, a fast reactor core is composed by FP gas purge/tube-in-shell type metallic fuel assemblies. They are cooled

by cooling sodium flowing through cooling tubes which penetrate packed hexagonal fuel rods in a shell. The shell is filled with sodium which retains the most of fission products other than inert gas. By application of FP gas purge/tube-in-shell metallic fuel assembly, volume fraction of fuel is increased and those of sodium and structural material are decreased in comparison with a conventional pin-bundle type fast reactor core.

6. Reactor Components

6.1 Reactor Vessel and Internal Structure

6.1.1 Hydraulic Tests of Flow Distribution

Experimental studies of hydraulic characteristics in the "MONJU" reactor vessel have been completed with an integral flow model of 1/2.14 geometric scale using water as a working fluid. The "MONJU" design is being reevaluated using the test data obtained for further improvement of the evaluation method.

6.1.2 Transient Thermohydraulics in Reactor Plenum

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

6.2 Shield Plug

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

steep temperature drop. The above concentration data was reflected on the evaluation of sodium deposition rates for the rotating shield plug of "MONJU".

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

6.3 Primary Pump

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

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

Cavitation performance tests of a new compact impeller were carried out in water for the DFBR pump.

6.4 Intermediate Heat Exchanger

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

6.5 Control Rod Drive Mechanisms

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

A dynamic behavior and a fatigue tests on shaft seal bellows have been carried out to establish the design basis.

Experimental tests of the self-actuated-shutdown system (SASS) for DFBR was initiated in the spring of 1987. As a first stage, the performance test of magnet for high temperature was carried out in-air. In-sodium testing was started in spring 1988.

6.6 Refueling and Fuel Storage System

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

6.7 Ex-Vessel Storage Tank

A simple geometry model test was conducted as a part of the program to study natural circulation in EVST. Experimental data obtained were temperature, velocity, and flow pattern profiles in the test vessel. Applicability of COMMIX-PNC (DRACS Version) code, which is a revised version of the COMMIX-1A code, was examined for developing the basis for its use as a design and safety analysis tool. Analysis results by COMMIX-PNC indicate relatively good agreement with the basic physical effects shown by the experimental data. Both analytical flow and temperature profiles fit with experiments quantitatively. The COMMIX-PNC code is well capable of predicting thermo-hydraulics of in-vessel natural circulation. Following the simple geometry

test, a 1/3 scale, 1/6 sector model test is being continued to demonstrate the feasibility of the decay heat removal system by natural circulation.

6.8 In-Service Inspection Equipment

An effort is being made to develop in-service inspection equipments for reactor vessels, inlet pipes and PHTS (Primary Heat Transport System). A remote inspection technique is being established by using optical fiber scopes for reactor vessels. An endurance test was finished under γ-ray irradiation in the temperature range of R.T to 250°C.

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

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

6.9 Expansion Joint for Piping System

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

Endurance tests of the expansion joints were completed in 1987.

7. Steam Generator System

7.1 Sodium-Water Reaction Studies

7.1.1 Leak Propagtion Tests

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

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

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

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

7.1.2 Development of Acoustic Leak Detection System

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

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

7.1.3 Computer Code Development

A large leak sodium-water reaction analysis code, SWACS, is being applied for various types of steam generators. Feasibility studies are under way. In addition, a one-demensional thermal-hydraulic computer code SWAC13E is under development. The code was validated by comparing cal-

culational results with SWAT-3 large leak data.

7.2 High chrome Steel Steam Generator

Three modified 9Cr-1Mo steel steam generators with a 1 MW or 0.5 MW heat capacity were manufactured, and in-sodium tests have been continued to confirm the reliability of long time operation under high temperature and thermal transient. 1 MW steam generator will be dismantled from the test loop, in December 1988.

7.3 Double-Walled Tube Type Steam Generator

Development of double-walled tube type steam generator started in 1985. The efforts are being made to fabricate several types of double-walled tubes. Performance tests of the tubes and tube sheets started in Oct. 1986.

Works for a conceptual design of the double-walled tube type steam generator and leak detector system are continuing. In-sodium testing of the 1 MW Double-Walled Tube Type Steam Generator will start in 1989.

8. sodium Technology

8.1 Material Tests in Sodium

8.1.1 Core Material Tests in Sodium

For fuel cladding and duct materials of demonstration FBRs, sodium environmental tests of advanced austenitic stainless steel is being performed. The following are tests' items;

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

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

8.1.2 Tribology Tests in Sodium

Development and evaluation of cobalt-free hard facing materials have been performed to clarify the tribological behaviors of contacting and/or sliding parts of FBR components. The tribology tests in sodium include;

* Self-welding test
* Friction and wear test
* Sodium compatibility test
* Thermal cycling test

Wear and friction tests of high-chromium ferritic steels as the candidate material for DFBRs' steam generator have been performed.

8.2 Flow and Heat Transfer

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

Measurements of pressure drop for the 19-pin wire-wrapped bundle have

been conducted under isothermal and mixed convection conditions for upward and downward sodium flows, focussing on friction factor correlations. A similar test for 37-pin bundle is in progress to study bundle size effects.

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

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

8.3 Radioactive Material Behaviour and Control in Sodium

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

The objectives of the studies are as follows;

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

8.3.1 Development of Computer Code for CP Behavior Analysis

An analytical model for CP behaviour in the primary systems of LMFBRs has been dveloped by using results obtained from out-of-pile studies which consisted of CP transfer experiments and examinations of metallurgical effects of sodium exposure on stainless steel test specimens, plant experiences in "JOYO" and published data by some foreign laboratories. A computer code "PSYCHE" has been developed to evaluate CP transfer in primary circuits of LMFBRs and radiation dose rate by CP around primary cooling piping and components. The analytical parameters in this model were obtained by using data of the out-of-pile experiments and measurements in "JOYO". The CP deposit radioactivity and dose rate distribution in the primary sodium system of "JOYO" was measured and compared with calculated results by PSYCHE. The radioactivity of CP calculated by PSYHE agreed with the measured data within a factor of 0.9~1.3 and within a factor of about 1.2 for the dose rate dis-

tributions.

8.3.2 Development of CP Trapping Method in Sodium

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

Cesium trapping tests in sodium by cold traps and RVC materials (Reticulated Vitreous Carbon), which were aimed for improvements of the trapping materials, has been carried out. RVC trap had been installed in the purification circuit of the primary system of "JOYO".

8.4 Sodium Chemistry and Sodium Purification

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

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

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

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

8.5 Sodium Removal and Decontamination

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

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

9. Instrumentation

9.1 Nuclear Instrumentation

9.1.1 In-Core Fission chamber

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

9.1.2 Ex-vessel BF_3 Proportional Counter

Experiments of ex-vessel BF_3 proportional counter for "MONJU" have been completed in the Japan Research Reactor-4. The experimental results satisfied design specifications under the following test conditions.

Accumulated irradiation dose:

Thermal neutron	1.6×10^7 n/cm^2
Gamma ray	3.3×10^9 R

Temperature conditions

During irradiation	30°C
Post irradiation	160°C

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

9.2 Failed Fuel Detection and Location

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

For "MONJU", experiments of the cryogenic adsorption system of the tagging gas system is being carried out to determine the design specifications, by changing design parameters.

For DFBR-1, innestigations of delayed neutron method, shipping method for failed fuel detection etc. are being carried out to determine the design specifications.

9.3 Early Warning System for Fuel Failure

9.3.1 Temperature Measurement

Temperature fluctuation due to flow blockage in the pin bundle has been analysed by computer code developed.

Calibration tests of the above-core instrumentation for "MONJU" are planned to start in January 1989.

Fast response thermocouples have been developed for application to the above-core instrumentation.

9.3.2 Flow Measurement

New types of eddy-current flow/temperature sensors were developed and tested in a sodium loop.

A calibration test of the eddy-current flow/temperature for "MONJU" will be started in the spring of 1989.

Investigation of the flow meter for the DFBR-1 main coolant has been started.

9.3.3 Other Systems

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

9.4 Process Instrumentation

9.4.1 Sodium Flow Meters for Large Piping

Since the permanent magnet type flowmeter was adopted for the flow measurement of the primary and secondary system of "MONJU", flowmeter response and calibration method became a major concern.

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

A equipment for the flowmeter calibration was installed. The calibration tests for "MONJU" were started in the spring of 1988.

9.4.2 sodium Level Meters

Calibration tests of the sodium level meters were conducted for "JOYO" reactor vessel and 50 MW Steam Generator Test Facility by using a sodium level meter calibration test loop.

A calibration test of the sodium leve meters for "MONJU" will be started in March, 1989.

9.5 Sodium Leak Detection System

Sodium ionization detector and aerosol trapping filter were tested in the leak rate of order of 100 g/hr with the simulated environment of the primary cell in "MONJU".

Their capability as the leak detector was also demonstrated with use of a piping model simulating the secondary system of "MONJU".

Consequently, these have been adopted as the leak detectors of the primary and secondary piping systems of "MONJU".

Work of the analysis code for sodium leak detection system was started in the spring of 1988.

10. Fuels and Materials

10.1 Fuel Fabrication Development

In 1972, PNC began the first operation at the Tokai Works to fabricate driver fuel pins for two developmental reactors, i.e. the experimental fast reactor "JOYO" and the prototype advanced thermal reactor "FUGEN". By the end of March in 1988, 345 subassemblies for "JOYO" had been produced using about 4.6 tons of MOX fuel. In addition, a large number of irradiation test fuel pins were fabricated and irradiated in various FBRs.

The maximum burnup achieved on PNC fuel pins exceeds 130,000 MWd/t, which corresponds to the target value of "MONJU" fuel. Over the ten year operation of "JOYO", there have been neither fuel failure nor anomalies in fuel behavior. The successful performance of "JOYO" fuel can be attributed to modification of fuel design based on detailed analyses and evaluations on irradiated fuels by postirradiation examination (PIE), production quality control through improvements and automation of equipment, modification of fabrication methods, rigorous inspection discipline and standarization. Recently, PNC established a new FBR fuel fabrication line for "MONJU" fuel in the Plutonium Fuel Production Facility (PFPF) at the Tokai Works. This facility will also provide fuel for a demonstration type ATR. Fig. 10.1 shows an aerial view of PFPF in the Tokai Works.

Based on the twenty year experiences on MOX fuel development and production, the equipment with high reliability and productivity has been provided. All the processes of pellet fabrication, pin working, component assembling, etc., are systematically and automatically controlled together with the nuclear criticality control, and radiation protection management.

Operation of the PFPF are expected to demonstrate techniques for the cost reduction of MOX fuel for a future commercial plant, reduction of human exposure to radiation, and in-line material accountability and safeguard function.

The PFPF line can produce 5 tons MOX fuel annually, which corresponds to 150 subassemblies for both "MONJU" and "JOYO". "MONJU" fuel fabrication is scheduled to start in 1989. PFPF is also expected to supply the fuel for DFBR-1 in Japan.

10.2 Fuel Pin Performance

Fuel pin performance analysis code "CEDAR" is successfully used for the performance prediction of "JOYO" fuels. A particular emphasis is paid for the precise description of temperature of fresh fuel pellet during a reactor start up for both "MONJU" and DFBR-1 in Japan, based on the experience of INTA-1 and UNIS-B rig tests. The validity of code in this subject will be proven in 1991 by two PTM tests in "JOYO". As for the transient behavior, the EBR-II TOP experiments have been contineously conducted. Data evaluation, framing and coding of the PCMI Model are being made.

PNC has recently developed a Cs migration code "MINER7A" in order to apply it to DFBR-1 in Japan. Based on the thermo-dynamical calculation and mass transfer mechanism of Cs under temperature gradient both radial and axial directions in fuel pin, the code can express the behavior and calibrated with irradiated fuel pins such as Phenix P-3, Rapsodie PNC-4 and "JOYO".

A failed fuel performance analysis code is being investigated in oder to obtain a better understanding of failed fuel behavior, especially the temperature evaluation of failed fuel pellet at breach and RBCB operations. EBR-II RBCB Experiments can yield necessary informations.

A control rod performance analysis code "CORAL" has been established. Based on the data on a lot of irradiated control rod pins of "JOYO", thermal and mechanical behaviors of pins B4C/modified 316 SS was revealed. It was concluded that ACMI is not life determing process. A feasibility study of designing the C/R of DFBR is going on using the "Coral" and "Vortex" codes.

Feasilility studies of the fuel performance codes for metal, carbide and nitride fuels are commenced.

"JOYO" MK-II driver fuel subassembly have achieved the average burnup of 64,000 MWd/t (peak 83,000 MWd/t) (corresponds to the target burnup of "MONJU" initial core) and highest fluence of fast neutron to modified 316 SS cladding reached 1.0×10^{23} n/cm (50 dpa). No indication of anomaly and/or failure have been ditected by DN and CG monitors so far. It's PIE is going on.

Reloading technique has been established in 1988. Irradiated UNIS-B test assembly and CMIR (Cladding Material Irradiation Rig) were discharged from "JOYO". After internal inspection of pins, and materials in FMF and MMF, the rigs were reconstructed and charged again to "JOYO" successbully.

As for the control rod performance maximum neutron capture of B4C pellet is 7.8×10^{21} cap/cm^3.

The modified 316 stainles steel for "MONJU" core material was irradiated up to 2.1×10^{23} n/cm (105 dpa) in FFTF MOTA and Phenix P-4 and revealed the swelling less than 2%.

10.3 Core Material (Improvement)

10.3.1 Modified 316 Stainless Steel

The development of modified 316 stainless steel was carried out to optimize the standard SUS316 within the limit of it's compositional specifications prescribed in JIS. Through minor modification of composition, PNC achieved to make a superior core material exhibiting highest swelling resistance and excellent creep rupture strength among Type 316 stainless steels in the world. In 1987, fabricaion of claddings and ducts made of modified 316 for the prototype fast breeder reactor "MONJU" were started.

10.3.2 Advanced Austenitic Alloy

Advanced austenitic alloys are also being investigated as a candidate core materials for DFBRs.

The optimized composition of an advanced austenitic alloy (PNC1520) for a three year-core application was tentatively selected with a nominal composition of Fe-15Cr-20Ni-2.5Mo-0.75Si-0.25Ti-0.1Nb-0.025P-0.04B in the 20% cold-worked condition. The creep rupture strength of this alloy is almost 30% larger than that of modified 316 stainless steel as the result of introducing fine precipitates of $M_{23}C6$, TiC and NbC carbides. Complementary results from ion and neutron irradiations suggest the PNC1520 will exhibit a higher incubation fluence for swelling than the modified SUS316.

10.3.3 High Strength Ferritic/Matensitic Steel

Excellent swelling resistance of the ferritic alloys has been confirmed by nickel ion bombardment at 150 dpa, although neutron irradiation data for ferritic alloys were not yet obtained. A particular elaboration for these alloys was made to improve high temperature strength at 650°C, without loss of fracture toughness. As for the ferritic/martensitic type, the alloy composition has been adjusted through out-of pile tests. PNC obtained a high

strength ferritic/martensitic steel by the addition of effective elements of 1.5-2.5 wt% tungsten, 0.1-0.5 wt% Cr martensitic steel in the normalized and tempered condition.

But it is still difficult for those claddings to attain the target strength in this series of alloys, so PNC placed the ferritic/martensitic steel as a candidate long life duct material. Substantial improvement will be demonstrated on ODS steels for long-life cladding applications.

10.3.4 Oxide Dispersion Strengthened Ferritic Alloy

Mechanical alloying techniques are being used to fabricate the oxide dispersion strengthened ferritic alloy. The increase of finely dispersed yttria (Y203) particles improves creep rupture strength; this effect is enhanced by small addition of titanium. Molybdenum and Tungsten addition can make dispersion strengthening more effective due to solution hardening effect. Optimization of the cladding fabrication process is necessary to obtain a homogeneous grain size; otherwise an axially elongated fine grain structure is unexpectedly formed which leads to decrease of creep rupture strength under biaxial stress condition.

10.4 Subassembly

10.4.1 Large Bundle Deformation Analysis

The purpose of this research is to decide the restriction value of Bundle-Duct-Interaction (BDI) for DFBR. PNC is carrying out a compressive test of large-scale 271 fuel pin bundle and measuring fuel pin displacements under compressed bundle conditions by using CT imaging computer system. A new code for Large Bundle Deformation Analysis is under development.

10.4.2 Ductless Fuel Assembly Development

In order to develop ductless fuel assemblies, a water flow test of a small scale core model was performed and prospect of thermal hydraulic feasibility was obtained.

In the next stage, PNC intends to obtain detailed performance of large fuel assemblies from water fine flow test.

10.5 Irradiation Experiments

10.5.1 JOYO

Irradiation program is now focused to a development of long-life and advanced fuels from standard type fuels such as SUS316 "MONJU" fuel. Material and fuel irradiations of advanced austenitic steel (PNC1520) cladding are being conducted.

CEA 15-15Ti fuels subassembly (UNIS-C) irradiation has started in November 1988.

Pin and bundle performance tests of large pin diameter fuels will be initiated. Irradiation tests of power-to-melt and on-line fuel analysis by using an instrumented assembly for a DFBR large diameter fuel pins are at detailed planning stages.

The wrapper fuel tube of high strength ferritic/martensitic steel is applied to large diameter axial heterogeneous fuel pin irradiation to demonstrate a long life feasibility.

10.5.2 Foreign Reactors

Phase-I program of operational reliability testing in EBR-II has completed and Phase-II program is initiated. Advanced pins are being manufactured with three types of cladding from PNC 1520, PNC high strength ferritic/martensitic steel and DOE D9 steel.

"MONJU" fuels irradiations in FFTF started in 1987. Phenix irradiation of "MONJU" fuels are ready to start.

10.6 Development of Advanced Fuels

In JAERI, post irradiation examinations of four uranium-plutonium mixed carbide fuel pins irradiated at low and medium linear heat rates (450 and 650 W/cm) in the thermal reactor JRR-2 were completed at the Reactor Fuel Examination Facility in 1987. The other four mixed carbide fuel pins are under irradiation in the Japan Material Test Reactor, JMTR, aiming at medium and high burn-ups (30 and 60 GWD/T).

The uranium-plutonium mixed nitride fuel pellets have been also prepared successfully by the carbothermic reduction of the mechanical blended dioxides with graphite under nitrogen-hidrogen mixed gas stream followed by crushing,

milling, cold-pressing and sintering. The irradiation tests of four uranium-plutonium nitride fuel pins will be started in next year using JMTR.

PNC and JAERI are planning to perform cooperatively the irradiation tests of the carbide and nitride fuels in "JOYO" in 1992.

In CRIEPI, the following studies have been continued in relation with the development of advanced fuels.

1) Feasibility study of metallic fuel (1986 ~ 1989)
   (1) Evaluation of metallic fuel irradiation behavior
      • Performance prediction code development
      • Metallic fuel safety evaluation
   (2) Core characteristics
      • Evaluation of optimum core concept depending on power scale
      • ATWS analysis of above cores
      • Ultra long life core for non-refueling reactor
      • Applicability of vent fuel for metallic fueled long life core

2) Feasibility study on pyro-process (1986 ~ 1989)
   (1) Evaluation of feasibility of each significant processes such as Electrorefining, injection casting and waste management etc.
   (2) Evaluation of process feasibility of facility summarizing (1)
   (3) Evaluation of fuel cycle economy

In PNC, the following studies have been conducted and are planned in aiming at future evolution.

Feasibility studies of metal, mixed carbide and mixed nitride fuels for FBR fuel systems has been intiated. Basic principles for the studies are to compare the key technology and cost evaluation of entire fuel cycle of advanced fuels with that of existing mixed oxide fuel. PNC comparatively lays stresses on metal and MN fuels at present.

As for the metal fuel research, fabrication test of U-Zr, U-Pu-Zr alloys has been initiated. Anticipated irradiation behavior i.e. migration of Zr and the formation of eutectic with cladding materials will be investigated in out-of-pile experiments. Pyrometallurgical studies for reprocessing will be performed. Waste management is also taken into accout.

If al the feasibility studies go on without any difficulties, fuel pins will be irradiated in "JOYO" in 1993.

Studies on MN fuel was also started in 1987. MOX fuel techniques may be applicable to MN fuel without great modification of fuel cycle system. Collaboration with European organizations is to be expected in this study. The fuel pin irradiation in "JOYO" is scheduled also in 1993.

Preliminary survey of the core design using advanced fuels has been conducted.

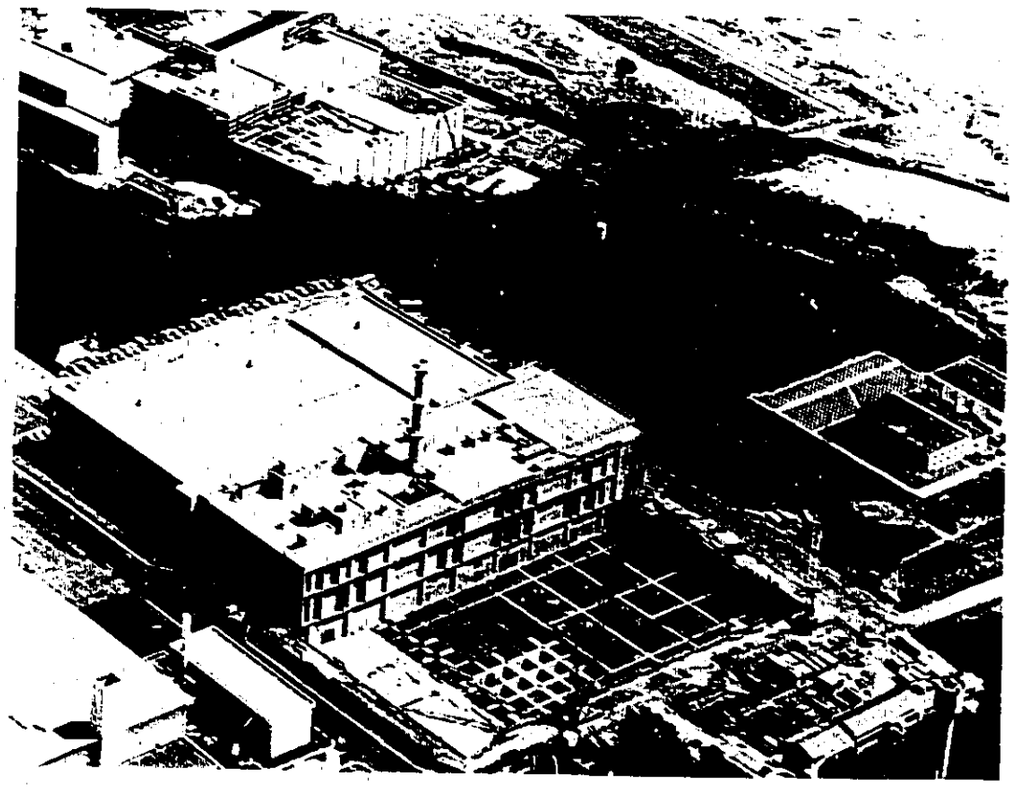


Fig. 10.1 Aerial View of Plutonium Fuel Production Facility (PFPF).

11. Structural Design and Materials

11.1 Development of Structural Design Methods

11.1.1 Structural Analysis Methods

a. FINAS Nonlinear structural analysis program

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

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

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

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

11.1.2 Structural Design Guide

a. Improvement of Elevated Temperature Structural Design Guide

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

* Design rules for welds
* Creep-fatigue design methods based on elastic analysis
* Strain limit criteria (ratchetting which differs from the Bree-type mechanism)

b. Development of a Elevated Temperature Structural Design Guide for Piping Bellows

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

11.2 Structural Tests and Evaluations

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

a) Elevated temperture test of piping bellows in air

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

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

Tests were completed.

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

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

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

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

11.3 Structural Material Tests

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

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

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

11.3.1 Structural Material Tests in Air

In-air structural material tests for the development of the "MONJU"

strength standard have been conducted since 1977 as mentioned below.

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

* tensile test
* creep test
* relaxation test
* fatigue test
* creep-fatigue test

From 1979 through 1981, the above tests had been continued to prepare the material strength standard for the "MONJU" structural design guide, and some theoretical methods on material strength behavior were investigated by the following tests,

* Creep damage estimation test with strain-hold
* inelastic strain behavior test
* evaluation test on strength of welded-joint
* notch effect test
* others

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

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

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

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

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

For these purposes, the Capella Step 1 (1985-1987) program involves developing the following technologies;

* Improvement of "MONJU" structural design technologies for cost-reduction (creep-fatigue life evaluation, strength of weldment, inelastic con-

stitutive equation, and others)

* Materials and fablication of large-scale structures
* Modification of material specifications for improvement of creep damage and revision of creep damage factos for structure desiga (Application of advanced SUS304 and 316 Stainless Steels)
* Application of elevated temperature fracture mechanics
* Development of material strength standard for high Cr-Mo steels

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

11.3.2 Structural Material Tests in Sodium and Water

a. In-sodium tests

Sodium environmental tests mainly on high cycle fatigue behavior for "MONJU" have been completed on Inconel 718 alloy which is used as liner material of the upper-core structures for increasing resistance to the thermal striping. Post exposure tensile, in-sodium creep and fatigue tests on the material for the bellows joint for piping systems and the carbon transfer behavior tests in the simulation loop of the secondry sodium circuits of "MONJU" were also completed.

Corrosion tests of "MONJU" steam generators were completed in high temperature sodium contaminated with NaOH or Na_2O, supposing water leak from steam generators.

A new series of sodium environmental effect tests, according to the Capella Step 1 (1985-1987) program, were carried out on possible candidate alloys for DFBRs. These alloys are high Cr-Mo steels, and advanced type (low carbon and/or high nitrogen added) SUS304 and SUS316 stainless steels and those weldments. Testing items are as follows; corrosion and mass transfer, carbon transfer, and mechanical strength (tensile, creep, fatigue, creep fatigue) tests in sodium. Sodium compatibility tests on the weldment of carbon steels were also carried out for the Step-1.

These tests have been continued in the Capella Step 2 (1988-1990) program.

b. In-water (steam) tests

It was confirmed that austenitic stainless steels such as SUS304, 316 and 321 were unsusceptible to S.C.C. in wet steam containing oxygen and

chloride ion up to 200 ppb. As the second step, corrosion tests were completed to confirm the integrity against S.C.C. for the plugs of stream generator tubings (after plugging) under the same wet steam conditions. As the third step, the corrosion tests on possible candidate steels (high Cr-Mo steels) for DFBRs under the wet steam conditions of FBR have been performed in the Capella Step 2 (1988-1990) program.

11.3.3 Structural Material Tests in Irradiation Environments

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

Surveillance tests for the primary components of the experimental fast reactor "JOYO" have been conducted to confirm the integrity of the reactor by evaluating irradiation effects of the materials.

Surveillance test data are used for the planning of "JOYO" operating program.

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

b. Research and development tests

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

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

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

11.3.4 Data Banking System

Material test data are compiled by specified data coding sheets, and the data are inputted into the computer aided Structural Material Data Processing System, SMAT.

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

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

Mod. 9Cr-Mo steel, etc. Efforts for improvement of the SMAT system have been continued to increase its function and to shorten the response times.

12. Safety

12.1 Thermohydraulics Related to Reactor Systems and Design

With regard to thermohydraulics related to reactor systems and design, major R&D work for "MONJU" has been substantially completed. Current experimental studies and code cevelopment are under way mainly for DFBRs.

Experimental studies have been carried on decay heat removal by natural circulation, thermal striping, and heat transfer characteristics of an in-line rod array in natural convection. As for the decay heat removal experiment, the first phase water tests using a 1/6-scale reactor model of a 1000 MWe loop-type plant were completed in March 1987. The second phase water tests were started in September 1987 for a pool-type reactor and are in progress. In these tests, emphasis was put on determination of flow and temperatue fields under transient low-flow conditions. A water test for thermal striping is being continued to acquire basic mixing data on multiple turbulent jets. The data are needed to develop a rational design method for structures placed in a fluctuating temperature field. Furthermore, the sodium test for the in-line rod array is also in progress to determine the natural convection heat transfer characteristics of a coil-type heat exchanger immersed in a reactor plenum.

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

12.2 Thermohydraulics Related to Reactor Safety

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

Experimental studies are conducted on fuel subassembly thermohydraulics

and plenum-channel thermohydraulic interaction. As to the former, the mixed convection pressure drop characteristics and intra-subassembly flow redistribution have been investigated using four types of test sections. Comprehensive evaluation of these experiments is now in progress. A water test is also continued to find out the pressure drop and coolant mixing characteristics of a PNC-designed new pin bundle planned for the DFBRs, whose average fuel burnup is expected to be 1.5 times higher than that of "MONJU" fuel subassemblies. Besides the experiments on an individual fuel subassemblies, another water test was conducted to clarify the thermo-hydraulic behavior of a parallel vertical fuel channel array interacting with plena. This test concerns the development of thermohydraulic methods for assuring the effectiveness of decay heat removal in the demonstration plant.

The other thermohydraulic subject related to reactor safety is LOPI simulation experiments. Three different experimental programs, SIENA/37F and 37G, DHB/37H and PLANDTL/37I&37J have been in progress in parallel. Experiments related to low-power dryout characteristics were completed by the DHB/37H boiling series. Our present interest is focused PLANDTL/37I&37J, which aims at understanding of thermohydraulic behavior of a bundle linked with the plant system response following an inlet-pipe break. The construction of the PLANDTL facility for the LOPI experiments was completed in 1987, and the experiments were started in January, 1988.

Efforts on safety analysis code development and validation have been continued. The SPIRAL code for the three-dimensional distributed analysis of fuel subassemblies has been upgraded with respect to a turbulence model, which enabled us to calculate local characteristics of turbulent momentum transport in a rod bundle. Vectorization of the subchannel analysis code ASFRE-III and the two-phase flow subchannel analysis code SABENA was accomplished during the fiscal year of 1987. The SABENA code was subject to experimental validation with emphases on the assessment of physical models used in the two-fluid subchannel formulation. For the thermohydraulic analysis of a FBR plant system, the Super System Code Loop Version (SSC-L) was applied to the analysis of the "JOYO" MK-II core natural circulation experiment. Improvements of the Super System Code Pool version (SSC-P) are continued with emphasis on refinement of the plenum model which is essential for predicting pool-type LMFBR plant dynamics.

12.3 Degraded Core Study

The objectives of this study are to concerned with preventing propagation of fuel pin failures and mitigating consequences of whole core accidents.

Examination and quantification of local faults were continued in the framework of probabilistic safety assessment of the "MONJU" plant. Regarding the subassembly-to-subassembly failure propagtion, the PI-A test in the SCARABEE Program was conducted, and a PNC attache at CEA/Cadarache performed post-test analysis.

The simulation experiments VECTORS (Vapor Expansion and Condensation Tests in Out-of-Reactor Simulation) were continued to study the energetics mitigation to effects during CDA bubble expansion and condensation and to validate the SIMMER-II fluid-dynamics model. Other experimental efforts to study the materials motion and interactions related to the accident- and post-accident phases have proceeded to: (1) initiation of the JEF-I tests to study the behavior of molten jets and erosive behavior of sturctural materials in a low temperature range (<500°C), (2) completion of the MELT-I tests in an intermediate temperature range (1000-2500°C), and (3) facility construction and initiation of an advanced large-scale test series (MELT-II).

Regarding the reactor and the code validation studies, much effort was devoted to the probabilistic evaluation of the in-vessel physical processes for the "MONJU" PSA. Main emphasis was placed on establishing methods to probabilistically represent accident trends, analyzing accident sequences and guarantee the results in the light of available experimental and theoretical information. Hence validation and reactor studies were performed mostly in support of the "MONJU" consequence analysis while the progress of the PSA activities will be reported in Section 12.6.

12.4 Plant Accidents Study

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

In the field of sodium fire and aerosol behavior, the rates of combustion and associated sodium aerosol release in a pool fire have been determined in a low oxygen atmosphere. In addition, the development of ceramic linar was directed to the improvement of castable materials. In regard to

the computer codes cevelopment, validation study of ASSCOPS was carried out using the test results from the SAPFIRE facility. Evaluation of sodium leak accident in a primary loop in DFBRs was continued to extend the applicability of the CONTAIN code. The development of SOLFAS; a multidimensional code to analyze natural convection heat transfer during sodium fires, were continued and the turbulent model was implemented. The chemically reactive sodium combustion products may harm the respiratory organ of human. Thus, development of code to analyze aerosol behavior in the air flow paths during respiration was carried out together with performance test of respirators.

In the area of source term research, activities are concentrated on the investigation of fission products behavior related to the transport from sodium pool to gas phase. The test to determine the gas-liquid partition ratio of fission products has been continued. The construction of a test rig to investigate fission products releases during a sodium-concrete reaction was completed. The test rig is used from 1988. On the other hand, a new test was started to determine the solubility of iodine in sodium at temperatures near the reactor operating conditions.

Current activities of the ex-vessel accident study are focused on the evaluations of the "MONJU" containment response following HCDAs. These analyses are used for the accident consequent evaluation of the "MONJU" PRA which be described in Section 12.6. The development of CONTAIN code has been also continued through the information exchange between PNC/SNL/KfK/GRS.

12.5 Steam Generator Safety

In the area of steam generator safety, major subjects are the leak development study, for the new materials, the large leak study, and the study of the double-walled tube steam generators.

12.5.1 Leak Development Study on 9 Chrome Steels

Tests of micro-leak and small leak have been carried out using 9 Cr steels, the most promising materials as heat transfer tubes of a steam generator in a DFBRs.

Twenty-one micro-leak tests were conducted to clarify the self-wastage behavior for Mod.9Cr-1Mo and 9Cr-2Mo steels using the SWAT-4 test rig. In the tests, a fatigue crack type and a slit type nozzles were used to proto-

typically simulate the actual micro-crack.

The small leak wastage tests were continued using he SWAT-2 test rig to determine the wastage rate of the Mod.9cr-1Mo, 9cr-2Mo and 9Cr-1Mo-Nb-V steels. One hundred wastage data were obtained from those three materials in total and they were compared with the conventional tube materials.

The SWAT-1 failure propagation tests started using the Mod.9Cr-1Mo steel in an intermediate leak range under the prototypical condition simulating an internal tube cooling effect.

12.5.2 Large Leak Study

In the area of the large leak study, the Pressure Effluence System Performance Test Rig, PEPT, was constructed to simulate a large leak sodium-water reaction event using water instead of sodium and explosives as a pressure source. The PEPT consists of simulated IHTS, which are one-fifth scale model of DFBR designs. Plenty of information has been obtained from the test results on bursting behaviors of in liquid type rupture disks and on the effects of pressure effluence system design on the pressure wave behavior.

In parallel with the simulation tests, the large scale sodium-water reaction analysis code, SWACS, is under modification. There are two main items to be modified in the present code. One is the coupling of a rupture disk model and the other is the application of the boundary tracking model (BTM).

12.5.3 Development of Double-Walled Tube Steam Generator

The present study is to evaluate the feasibility of the FBR plant system which utilizes the primary steam generators. Firstly, all the steam generator tube failure incidents of FBRs, PWRs, and fossil power plants were surveyed and analyzed. Secondly, the mechanisms that might cause either one-sided or both-sided failure of the double-walled tube were picked up after the rearrangement of the failure data. The result is applied to the construction of the fault-tree and the quantitative evaluation of the tube failure occurrence probability of the double-walled tube steam generator.

12.6 Research on Probabilistic Safety Assessment

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

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

In Phase V, which began in April 1986, importance and uncertainty analyses were performed for safety-related systems and components and for the major event sequences. Also efforts were made to develop computer codes and data-bases.

The systems analysis entered the second stage in April 1987. The event trees (E/Ts) and fault tress (F/Ts), which have been the reference, are being reviewed. In order to reflect the changes in design and the technical specification made since 1984, new F/Ts are being constructed based on the newest information. Examples of changes are the inter-system dependencies between the component cooling system and the cooling seawater service system, and the interlock signals for the primary sodium overflow system (PSOS) and the maintenance cooling system (MCS). The interactive modular F/T constructing program MODESTY, which was developed in 1986, is utilized for this task. Another program FAUST is used in preparing the input data for the SETS code. More support programs have been developed for expeditions and efficient systems analysis. Namely, an uncertainty propagation analysis program UNIFORM, an importance measure calculation program IMPROVE, and E/T drawing program ETD. For commercialization of fast reactors, evaluation of a rational safety design policy is needed. Thus a simplified systems analysis program, QUST is being developed. This program enables quick and easy case studies on various system designs.

Data development effort for the fast reactor components is being made. CREDO (Centralized Reliability Data Organization), the cooperative project between Japan and the USA, is the core of this effort. The operation and maintenance data on the 25 type, 2500 components have been obtained mainly

from the experimental fast reactor "JOYO". And these data have been entered into CREDO through FREEDOM, the primary database management program. Also the data from the 50 MW Steam Generator Test Facility (SGTF), 2 MW Sodium Flow Test Facility, Sodium Exposure Test Loop, and Control Rod Drive Mechanism Test Loop were entered into the CREDO data-base. The data accumulated in CREDO by now are 1×10^9 component hours from PNC. Together with the data from the US fast reactors EBR-II and FFTF, and sodium test facilities, more than 20,000 components, 1,300 failure data, 1.5×10^9 component hours are accumulated.

More efforts will be made to collect reliability data for the fast reactor components and to develop programs for database management and statistical analysis. Efforts will be made further to collect data on moe component types and human performance.

To actualize commercializing fast reactor plants, its economy needs to be improved while maintaining the same level of safety as light water reactors. Value-Impact Analysis (VIA) is effective in solving these problems. Development of the cost evaluation model as well as the PSA is needed for this purpose. Thus, the cost database SEDB is under development, which can be used in the evaluation of the rationality of a plant safety design policy of fast reactors.

A preliminary evaluation of the in-vessel physical processes, radionuclide transport and risk analysis was performed in view of the level-3 PSA. Phase-1 of this task was finished in March 1987. The event sequences were divided into the following phases: Initiating; Transition; Core Disruption; Core Materials Relocation; Post-Accident Heat Removal; Structure Response; Containment Response; and Fission Product Transport. Phenomenological E/Ts with multiple branches were developed for each of the above phases based on the result of the analyses. ULOF and PLOHS were mainly investigated in Phase 1, UTOP and ULOHS are added in Phase 2.

Event progression and anomaly detectability of the local fault were analyzed. The result of the analyses using E/Ts shows that the probability of a local fault propagating into a whole core accident is very small. The detailed analyses reflecting the in-pile test result are planned on Blockage Formation, Subassembly-to-subassembly Failure Propagation, and Delayed Neutron Emitter Transport Phases.

The containment response analysis is being performed using the CONTAIN

code. It is planned to develop an analysis model for debris-concrete interaction in the dry cavity and to construct containment E/Ts based on the analyses on each event sequence.

A best-estimate based on plant dynamics analysis was performed using the SSC-L code. As a result, the failure criteria which had been used in the earlier systems analysis could be relaxed for the decay heat removal system (DHRS) such as the Auxiliary Cooling system (ACS) and MCS.

In the present year, sensitivity studies are to be executed for the PLOHS, LOPI, and ULOHS sequences to estimate the safety margin of the plant. Also the analyses on the seismic event which could cause a significant consequence are to be performed. These are, namely, the hazard analysis, the fragility analysis of the structures and components, and the screening analysis of seismic consequence.

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This review was edited by PNC with the assistance of JAPC, CRIEPI and JAERI.