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A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

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Edited by Y. Mi[illegible]

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Power Reactor and Nuclear Fuel Development Corporation

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1. Introduction

The fast breeder reactor development project in PNC has been in progress steadily in these nineteen years. Concerning the experimental fast reactor "JOY", continuous operation on the Mark-II core has been succeeded, attaining the accumulated power of 745,203 MWH (8,823.84 hours) since the first criticality of the Mark-II core on 22. Nov. 1982. The maximum burnup of an test fuel subassembly is approximately 60,000 MWD/T. Until now no fuel failer was observed.

With respect to the prototypte fast breeder reactor "MONJU", the first contract on reactor component fabrication was closed in January 1984 between the PNC and component manufacturers and the activity for fabrication permit has recently started.

Testings of some sodium components are being stil continued to verify their expected performance for "MONJU" Plant in Oarai Engineering Centre/PNC.

Conceptual design of a demonstration reactor and elementary technology development including general design studies for large LMFBR are being carried out at the Federation of Electric Power Compaies (FPO) and at PNC, respectively.

2. Experimental Fast Reactor "JOYO"

2.1 General Status

This report covers the activity on the experimental fast reactor "JOYO" in the period from 1984. 4 through 1985. 3. In this period five cycles of operation at 100 MWt were carried out with no fuel failure and other plant troubles. Parallel to the normal operation with several irradiation test rigs in the core and reflector region an effort has been continued to obtain license for carring out the following items:

a) Enlargement of flexibility of Pu-fiss ratio in the fuel,

b) Extension of burn-up of test fuels from 130,000 MWD/T to 150,000 MWD/T, including a possibility of cladding breach (RTCB) and

c) Extension of linear heat rate of test fuels from 600 W/cm to 640 W/cm (at over power condition).

2.2 Topics

2.2.1 Control Rod Vibration Analysis

During JOYO Mark-II operation, control rods vibration analysis was conducted by using noise analysis technique, neutron flux, reactivity signal, primary flow rate, load weight of each control rod, and acoustic signal of control rod driving mechanism were measured and analyzed to obtain their r.m.s. values, auto power spectral densities, and coherences. The estimation of control rods vibration mode as well as the identification of neutron flux noise source in the Mark-II core was carried out.

These signals were measured under various plant conditions, changing primary flow rate, excess reactivity, and control rods pattern. Dependence of neutron flux noise characteristics on those factors was analyzed, and it is confirmed that the main source of neutron flux fluctuation is flow induced vibration of control rods.

Furthermore, the analysis of measured data under the condition of different flow rate distribution in the core, the comparison of reactivity fluctuation between the calculated and the measured, and the review of post irradiation examination on control rods were conducted, the mode and driving force of control rods vibration were estimated.

Schematic view of JOYO control rod is shown in Fig. 2.1. After detailed measurement and analysis the conceptual vibration mode was identified as Fig. 2.2.

This control vibration corresponds to $\pm 2\%$ of power fluctuation, which should desirably be reduced. Considering these results and desire, modification of vibration restriction mechanism of control rod is in progress in order to reduce neutron flux fluctuation during power operation. (cf. Fig. 2.3)

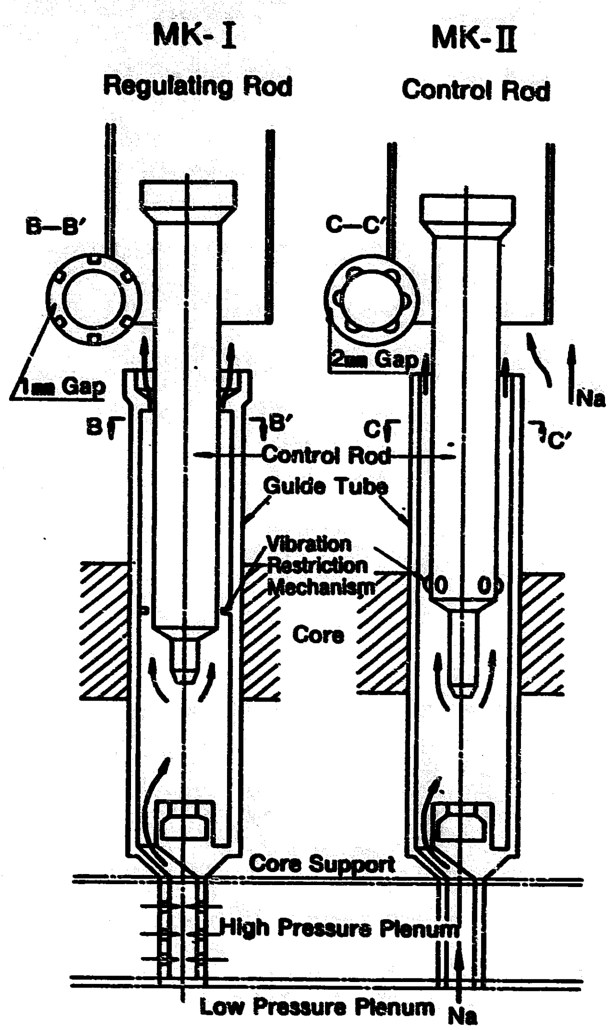


Fig. 2.1 Schematic view of JOYO control rod.

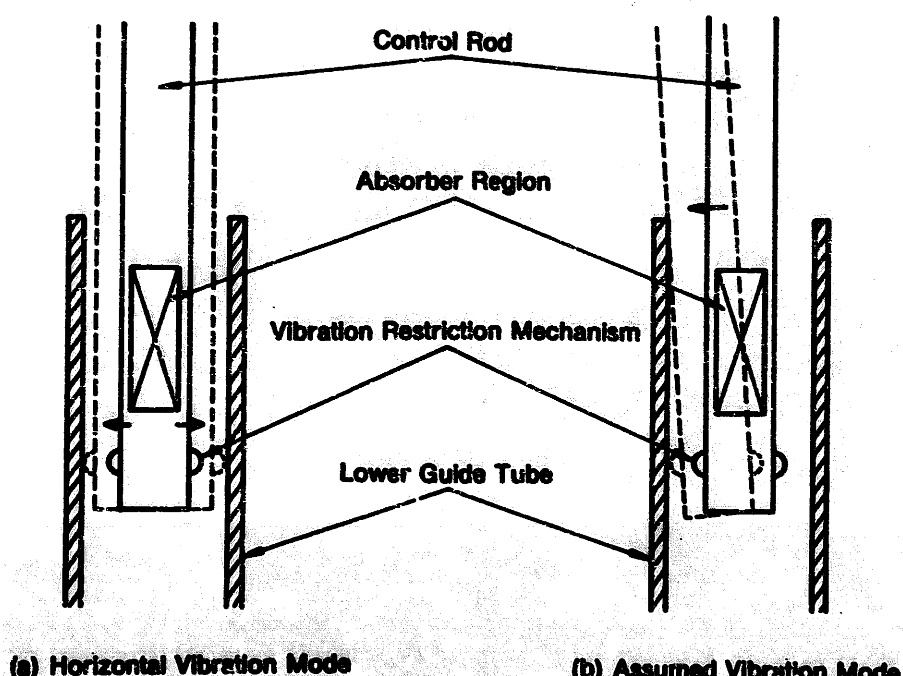
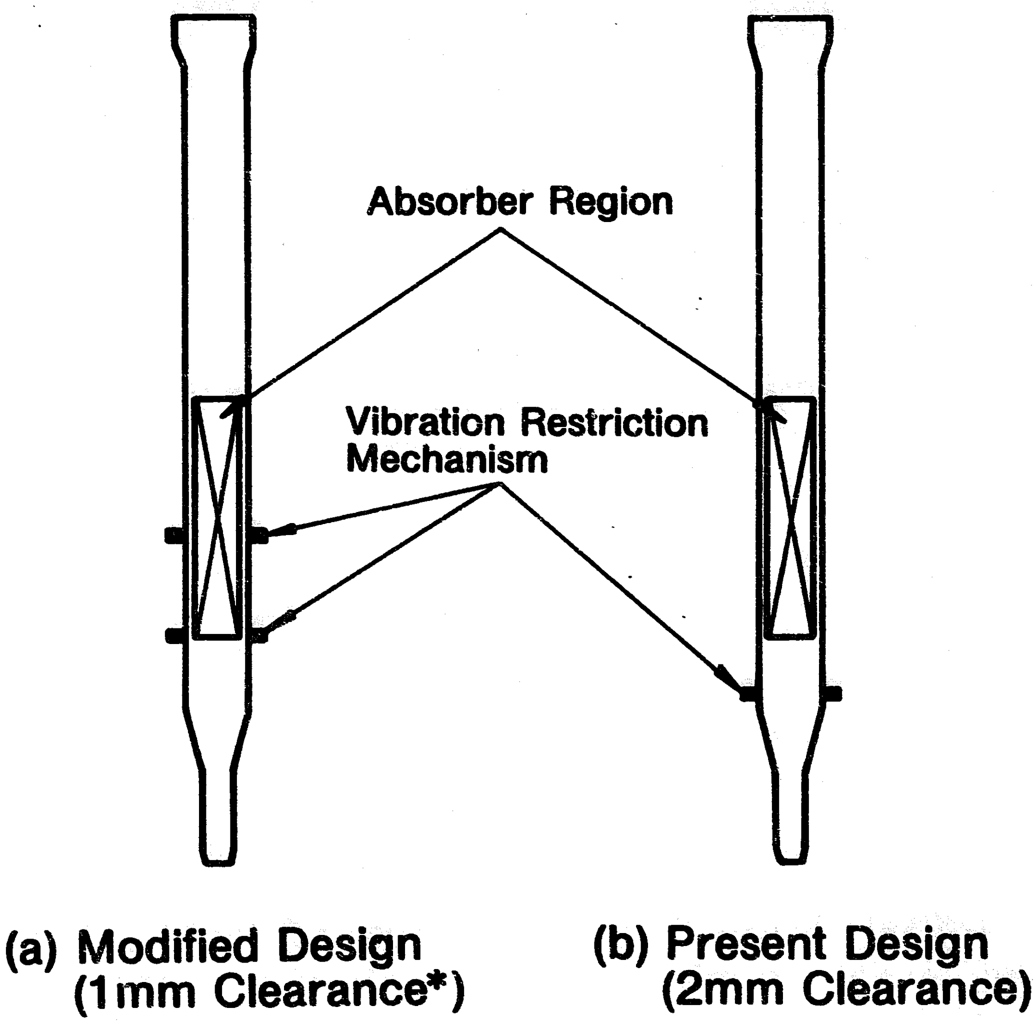


Fig. 2.2 Conceptual view of vibration mode of control rod.



* Distance between Vibration Restriction Mechanism and inner Surface of Lower Guide Tube

Fig. 2.3 Modified design of vibration restriction mechanism of control rod.

2.2.2 Bowing behavior of subassemblies

In JOYO, the measured power coefficients in the beginning of the operation cycle of MK-I and MK-II cores showed power dependence, while the calculation without taking account of bowing predicted little power dependence (Fig. 2.4)

The bowing analysis was performed in order to investigate the power dependence observed in the measured power coefficients and the following conclusions were obtained.

(1) The evaluated power coefficients taking account of bowing effect agree better with measured ones than the calculated ones without taking account of bowing effect in MK-I core. (Fig. 2.5)

(2) In MK-II core, although the analytical results show not so good agreement quantitatively with the measured power coefficients, it is suggested that they agree better depending on the uncertain parameters such as the heat generation in the reflector region, the threshold moment for leaning and the stiffness of the inner reflector. (Fig. 2.6)

(3) It becomes clear from these results that the power dependence observed in the measured power coefficients in JOYO is due to the bowing effect.

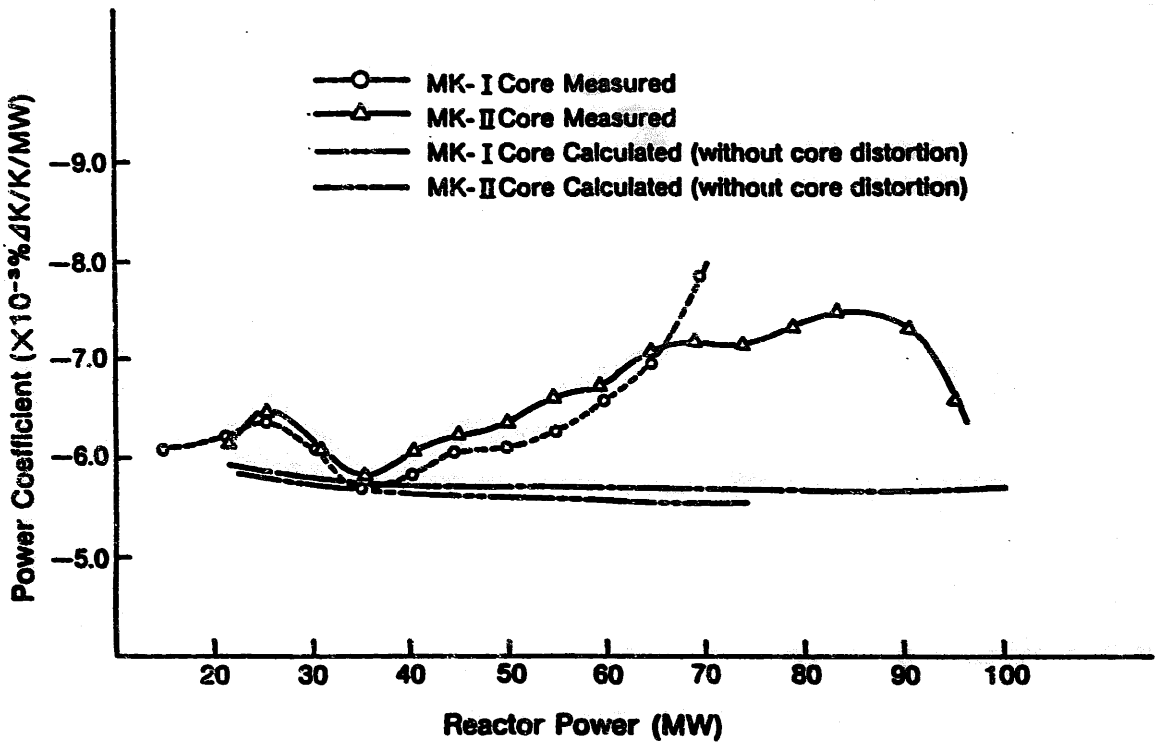


Fig. 2.4 Power Coefficient Measured in MK-I and MK-II Cores

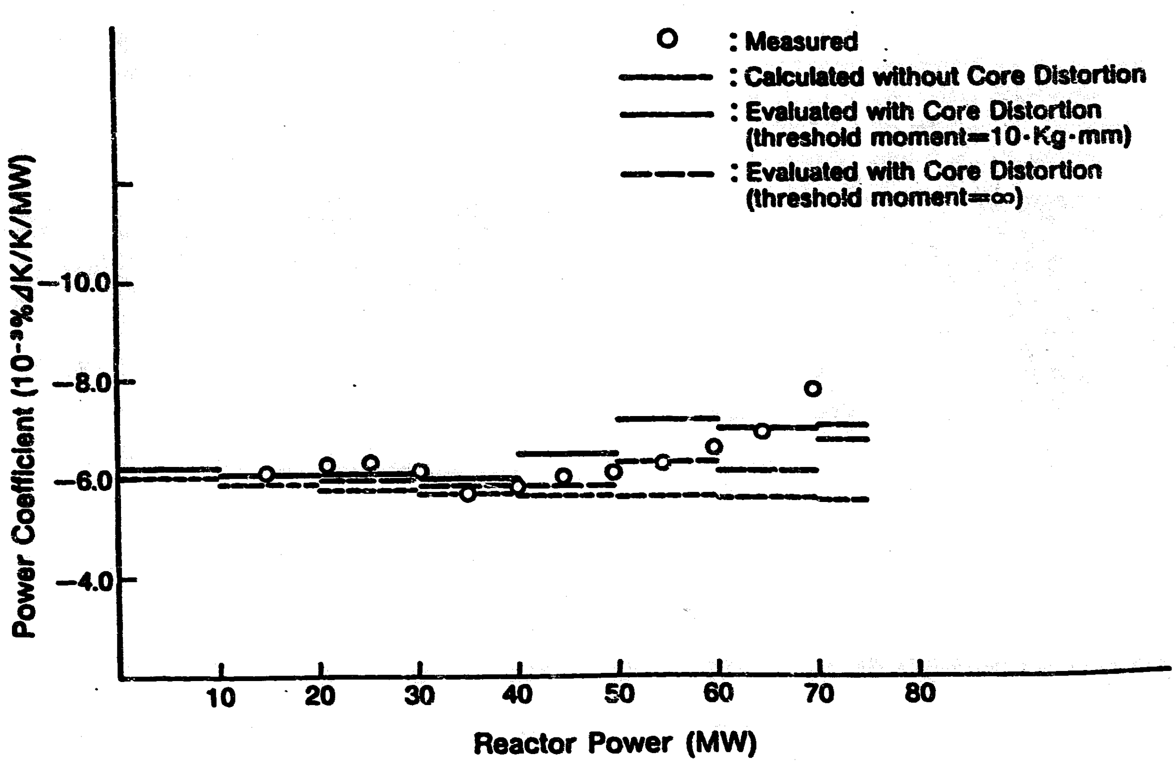


Fig. 2.5 Comparison of Measured and Calculated Power Coefficient (MK-I Core)

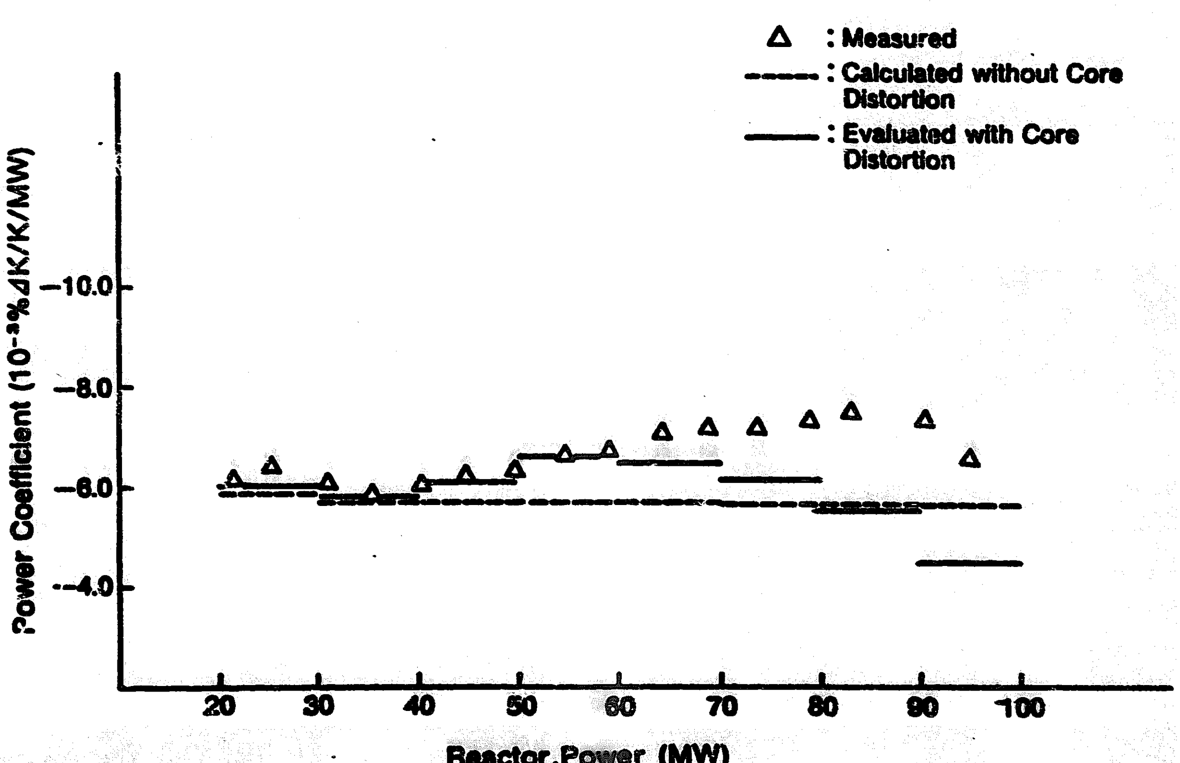


Fig. 2.6 Comparison of Measured and Calculated Power Coefficient (MK-II Core)

Absorber Region

Vibration Restriction
Mechanism

(a) Modified Design
(1mm Clearance*)

(b) Present Design
(2mm Clearance)

MK-I Core Measured
MK-II Core Measured
MK-I Core Calculated (without core distortion)
MK-II Core Calculated (without core distortion)

: Measured
: Calculated without Core Distortion
: Evaluated with Core Distortion
(threshold moment = 10.kg.mm)
: Evaluated with Core Distortion
(throshold moment = ∞)

: Measured
: Calculated without Core Distortion
: Evaluated with Core Distortion

3. Prototype Fast Breeder Reactor Monju

3.1 Summary

Tsuruga Peninsula in Fukui Prefecture, appoximately 400km west of Tokyo, was selected as a construction site of Monju in 1970. A site suitability study was conducted on the basis of geological, marine, meteorological and other survey results.

Sefety Review process by the regulatory body was initated on December 10, 1980, when licensing application was filed to the Science and Technology Agency (STA) of the Japanese government. Reviews by STA and by the Nuclear Safety Commission were completed in December 1981 and April 1983, respectively. Reactor establisment was permitted by the Prime Minister in May 1983 on the basis of these review results, encouraging site preparotay work as well as detailed design work. The first contract on component fabrication was closed in January 1984 between the PNC and component manufacturers.

Software coordinating company, Fast Breeder Reactor Engineering Co., Ltd. (FBEC), was established in April 1980, with contributions from manufacturers. A special department was organized within the Japan Atomic Power Company (JAPC) to support the PNC as well as to serve as liason with the nine Japanese electric utilities and the Electric Power Development Corporation (EPDC).

Reviews on the design and method of construction by STA and on the construction plan by MITI were initiated at almost the same time in December 1984.

4. Demonstration Fast Breeder Reactor DFBR

4.1 Introduction

Design studies of DFBR have been carried out by the Power Reactor and Nuclear Fuel Development Corporation (PNC) and the private utilities in parallel with development of the experimental and prototype reactors. PNC is focusing mainly on R&D aspects and related design studies in accordance with the role of PNC indicated in "the Long Term Programme of Development and Use of Nuclear Energy" (1982) and in "The Interim Report on the Development of the Fast Breeden Reactor" (1984) issued by the Japan Atomic Energy Commission in 1982. The utilities, from their own standpoints are focusing mainly on proper incorporation of users' needs into the design; such needs as achievement of economic goals as well as of reliability and availability goals and improvement of operability and maintenability.

The studies have been promoted under close cooperation of PNC and the utilities including activity to determine the R&D programme and to develop technical bases needed for selection of specifications of the basic DFBR design to be planned in near future.

4.2 Design Study of DFBR

The utilities has carried out the conceptual design studies under cooperation of all of the private ten electric power companies until 1983. The study consisted of three phases. Regarding the study of the loop-type reactor, key concepts of the design were selected in the phase I and the design was reviewed mainly from the standpoint of operation and maintenance in the phase II. Design specifications were established on the basis of the further design of the total system and components in the phase III from FY 1981 through 1983. Regarding the study of the pool-type reactor, a preliminary concept definition was carried out studying the design of preceeding plants in phase I and it was reviewed mainly from the stand-point of seismic characteristics in phase II. The key subsystems of the pool-type reactor were designed in phase III in parallel with model tests studied by the Central Research Institute of Electric Power Industries (CRIEPI).

Presently, utilities are carrying out conceptual studies on both loop type and pool type plants with the emphasis on cost reduction.

PNC has continued the design study since 1975. The design studies being done presently by PNC are on a loop type plant with compact primary piping layout, as well as on some individual key technologies of FBR systems and components. Design studies on some critical areas of pool type reactors are also being made. Fig. 4.1 shows an example of the compact primary piping layout.

5. Physics

5.1 JUPITER-II Program

The JUPITER-II program is the Joint Physics Large Heterogeneous Core Critical Experiments program between the U.S. DOE and PNC, Japan. The experiments began in May 1982 and ended in April 1984, as a part of the ZPPR-13 program. The ZPPR-13 is a series of critical assemblies to study the fundamental neutronic behavior of large, radially-heterogeneous LMFBR cores. The ZPPR-13 series consists of six assemblies of approximately 650 MWe-size, containing a large central blanket zone, two internal blanket zones and three fuel zones. The analysis of the JUPITER-II experiments has been done in Japan since 1982, using the JENDL-2 library. The analytical works have been performed by PNC in cooperation with the eight other organizations listed below, which are under contracts with PNC:

Osaka University
Japan Atomic Energy Research Institute (JAERI)
Toshiba Corporation (TOSHIBA)
Fuji Electric Co., Ltd. (FUJI ELECTRIC)
Mitsubishi Atomic Power Industries, Inc. (MAPI)
Hitachi Ltd. (HITACHI)
FBR Engineering Company, Ltd. (FBEC)
Japan Information Service, Ltd. (JAIS).

The first joint analysis meeting of the JUPITER-II program was held on September 11-14, 1984 at the ZPPR facility in ANL-Idaho, between the U.S. and Japan. The objectives of the meeting were to review analytical results from ZPPR-13A and -13B and to discuss future activities of the JUPITER program. As the result of discussion, a strong correlation was noted between the U.S. and Japanese analyses of the detailed technical results of the ZPPR-13A and -13B experiments. The two delegations mutually conclude that the results of the JUPITER-II experiments raise serious questions relative to the choice of certain radially heterogeneous cores with a high fraction of internal blankets as a reference design for large LMFBRs. For reasons that are reasonably well-understood, the planar power distribution may be difficult to control satisfactorily under normal operating and manufacturing practices. A more definitive statement can be made after the ZPPR-13C results have been analyzed.

5.2 Axial Heterogeneous Core Experiments at FCA

A series of critical experiments have been performed at FCA to investigate nuclear characteristics of the axial heterogeneous core concept. The FCA XI-2 assembly has a cylindrical test region surrounded by a driver region, and has a disk-shape internal blanket of $58.8\ cm^{\phi}$ x $20.4\ cm^{h}$ at the mid-plance of the cylindrical core. Criticality, reaction rate distributions and ratios, sample reactivity worths and sodium void reactivity worths were measured. As the result of analysis, ^{238}U caputure rate and sodium void worth were underpredicted at the internal blanket.

5.3 Development of 3D Sn Code "TRITAC"

The TRITAC is a three dimensional neutron transport code based on the discrete ordinate method. It has been developed for the purpose of salution of the reactor core eigen value problem. Efforts were made to improve the acceleration technique, and the diffusion synthetic acceleration method was applied to the code. The preliminary result of numerical calculation showed good agreement with the experimental value.

5.4 Multidrawer Effect for ZPPR-13A Analysis

A multidrawer model was applied to the analysis of ZPPR-13A, in order to take into account the interference effect between fuel and flanket regions adjacent to each other. The multidrawer effects obtained are as follows.

k_{eff}	:	0.43 % $\Delta k/k$ increase
^{238}U (n, f) distribution	:	4 % increase in fuel compared to blanket region
Na void worth	:	small effect (-3 ~ 5 %), but relatively large space dependence

5.5 Shielding Analyses of "JOYO"

Neutron dose rate distribution and neutron flux spectrum were measured in the pit-room of "JOYO" by using Bonner ball detector system. Numerical calculations were performed by two-dimensional discrete ordinate code DOT3.5 and the results were compared with the measured data. The spectrum of the numerical results was shown to be softer than that of the measured data. The effect of the fuel assemblies in the in-vessel storage rack and that of the concrete pit-cover were analyzed in detail. These numerical and measured data will be reflected to the operation and the radiation exposure control of "JOYO".

5.6 Shielding Analyses of "FFTF"

Shielding characteristics of "FFTF" were calculated by the DOT3.5 code and albedo Monte Carlo code MORSE-ALB. The results were compared with the measured data which were offered in FFTF/JOYO Shielding Data Exchange. Comparing with the data of "JOYO", it is distinctive that the fuel assemblies in the in-vessel storage rack have much more effects on the dose rate in the maintenance deck area.

5.7 Shielding Experiment in "YAYOI"

Neutron streaming through the narrow and long annular gap in the steel was measured by using the fast neutron source reactor "YAYOI" of Tokyo University. The measured geometries were selected to be as typical gaps in the control rod drive mechanism of FBR. The data is now being analyzed.

5.8 Group Cross Section Sets for Shielding Analyses

Neutron transport cross section set for shielding analyses was prepared from JENDL2 library by using SUPERTOG and MINX codes. Several Benchmark calculations for FBR plant have shown that the new set estimates the neutron flux after penetration smaller than that from the ENDF/B-IV file. It was also shown that the sepctrum by the JENDL2 was softer than that by the ENDF/B-IV.

6. Research and Development of 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. Currently, the Monju design is being evaluated using the test data obtained.

6.2 Shield Plug

A temperature distribution test was carried out on a simulated reactor upper shield plug which has a scale of approximately 1/3 in diameter and 1/1 in height. It was operated under elevated temperature using sodium. Test result showed existence of natural convection of argon cover gas in the annular gap around the plug. In order to suppress this effect covection restraining plates were employed. After setting of restraining plates, in-sodium testing of the shield plug was carried out from May 1983 to December 1984, and the shied plug was pulled out from test tank and was observed in February 1985. Analysis and evaluation of test results have being continued.

6.3 Primary Pump

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

As the final stage of the pump tests for MONJU an operation test with 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, it was not found any trouble.

6.4 Intermediate Heat Exchanger

Water flow tests of 1/6 sector model and 1/2 scale full model of the MONJU intermediate heat exchanger has been completed. The uniform flow distributions in both primary and secondary side were established with low pressure losses by the flow control devices devoloped in these tests. Further sodium flow test under transient condition was performed to investigate the stratification characteristics in the rising flow region after inlet nozzle which might affect thermal shock rate to the upper tube sheet section. On the other hand, in-sodium life test was carried out on the trially fabricated bellows to be used at the top of the down-commer pipe. Non-destructive inspection method was also developed for the tube-to-tube sheet welding.

6.5 Control Rod Drive Mechanisms

Three kinds of control rod drive mechanisms for MONJU have been tested with full scale mock-ups 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 will be carried out in spring of 1985. The final models of FCRD and CCRD will be manufactured till March in 1985 and in-water and in-sodium tests will start from that time on.

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

6.6 Refueling and Fuel Storage System

Refueling system of MONJU consists of in-vessel fuel handling machine (FHM), and ex-vessel transfer machine (EVTM). Testing of prototype FHM in sodium and characteristic test of the shaft seal were completed. Testing of prototype EVTM in sodium has been performed since 1981. In-sodium testing of prototype EVTM was completed in December, 1984.

6.7 In-Service Inspection Equipment

An effort is being made to develop In-Service Inspection Equipment for reactor vessel, its inlet pipes and PHTS (Primary Heat Treatment System). Basic examination technique is "Visual". Remote inspection technique with optical fiber scope for reactor vessel are now being developed. As a step-I test, γ-ray irradiation tests of fiber scope were completed at R.T to 250°C.

Another effort is being directed to develop ultrasonic transducers for high temperature use as one of volumetric examination technique.

7. Steam Generator System

7.1 50 MW SG Test Facility

After the 3,400 hour operation with steaming condition the testing of No. 1 50 MW SG were completed in April 1975. It was disassembled for inspection and the feasibility of the "MONJU" SG design concept was confirmed.

Then No. 2 50 MW SG was constructed and the performance test began in January 1976. The accumulated operating time of No. 2 50 MW SG is 14,000 hours with steaming condition for the evaporator and 4,570 hours for the superheater. Total operating time for secondary sodium loop with sodium is 31,300 hours as of the end of 1984. Research and development on the steam generator characteristics, plant controlability and thermal transient test were almost completed.

The spatial characteristics of the evaporator were evaluated taking into account of the plugged tube effect. Evaluation of the effect of the water side fouling is under way. No performance change has been observed after 14,000 hours operation.

Demonstration of maintenance and repair techniques for SG was carried out in 1981 using the evaporator.

Performance test of water leak detection system was carried out in 50 MW SG Test Facility. Water, hydrogen and argon gas were injected into the evaporator or the sodium inlet piping and detected by in-sodium hydrogen detectors, in-gas hydrogen detectors or acoustic detectors. Characteristics of hydrogen behaviour in sodium or cover gas were evaluated. Performance of acoustic leak detection system are now being studied.

Monju auxiliary cooling system (ACS) test equipment was installed on secondary sodium loop of 50 MW SG Test Facility, simulating Monju system and, operation and testing of this system was started from July 1982, and completed in Dec. 1983.

Heat removal ability, controlability, transient behaviour, natural circulation characteristics and so on become clear. Computer codes about plant dynamic analysis and natural circulation analysis were verified. Many useful data were obtained for the design of MONJU, about heat removal ability, system controlability, transient behaviour, natural circulation characteristics and so on. Computer codes for plant dynamic analysis and natural circulation analysis were verified.

Endurance test of No. 2 50 MW SG is now being continued to demonstrate the reliability of MONJU SG. Future test is planned for steam generators of post-MONJU FBR.

7.2 Sodium-Water Reaction Study

7.2.1 Leak Hole Enlargement and Leak Propagation Study

The self-enlargement test on a micro-crack defect is under continuation with the SWAT-4 test rig in an attempt to accumulate data of 2-1/4Cr-1Mo and stainless steels for various sodium temperatures and leak rates. In 1984, tests with alloy 800 and weld of 2-1/4Cr-1Mo have also been conducted.

With SWAT-1, one wastage test was conducted in the intermediate water leak region under the Monju superheater conditions.

With SWAT-3, failure propagation test under the superheater conditions was conducted. At this test, feasibility test to remove sodium-water reaction products by operating a cold trap was also conducted.

7.2.2 Computer Code Development

Improvement of the SWACS code is under continuation. During this period, new function to analyze thermal response of fluid and structures was added to the code.

7.2.3 Leak Detector Development

Basic tests of an acoustic-type leak detection system have been performed using a water test rig of a half scale model of the Monju evaporator. In the tests, sound generated by a sodium-water reaction was simulated by injecting nitrogen gas into water through a small hole. Leak detection as well as leak location technique are under development by processing the sound signals with a micro-computer and a large computer.

8. Sodium Technology

8.1 Material Tests in Sodium

8.1.1 Core Material Tests in Sodium

In-sodium test of the fuel cladding (and duct) materials for "MONJU" has been conducted for the development and preparation of the fuel assemblies design according to the program of core materials development. The 20% cold-worked modified SUS316 is used for the fuel cladding tube, and the material tests in sodium have been conducted as following items. Such as;

- Mechanial tensile test after exposure to high temperature sodium and thermal aging
- Internal pressure creep rupture test
- Corrosion and mass transfer test

The modified stainless steels and advanced alloys are expected to be used in demonstration FBR, etc. and R&D shown above will be conducted.

The new R&D program (for "MONJU" and especially for demonstration FBR) is now under preparation.

8.1.2 Tribology Tests in Sodium

For the development and licencing of "MONJU" components, the R&D on some hard facing materials (especially cobalt-free alloys) have been performed to clarify the tribological behaviors of structural materials applied for contracting and/or sliding parts.

The tribology tests in sodium conducted are as follows;

- Self-welding test
- Friction and wear tests
- Sodium compatibility test

The new R&D program, mainly for the development of cobalt-free hard facing alloys, is to be prepared for "MONJU" and especially for demonstration FBR.

8.2 Flow and Heat Transfer

In order to evaluate the performance of reactor vessel shield plug, tests have been conducted to determine the sodium mist content in cover gas space at relatively low temperature condition.

Tests on flow distribution in the Monju reactor vessel have almost been completed using water test models. Experimental data will be used to validate the computer code for prediction of adequate flow distribution in the Monju reactor vessel.

Experimental study on overflow of the Monju reactor vessel has been completed using water flow model. The test results have been effectively utilized for the design of the overflow line.

Heat transfer tests at low Peclet numbers have been completed using two heater pin bundles at extremely low flow velocity of sodium. A third bundle test is planned in near future before final evaluation of this important test.

8.3 Radioactive Material Behaviour and Control in Sodium

This technology area has studied to decrease the radiation exposure to plant personel due to the contaminated primary sodium systems by the radioactive corrosion and fission products (CP & FP) introduced into the LMFBR primary systems and therefore to improve the maintenance operation safety.

The objectives of Research and Development (R&D) in this tehcnology area are as follows;

- To establish a computer code for CP behaviour analysis.
- To develop methods of CP trapping.

These studies are controlled by the program plan of the task work named as ALPHABET PROJECT which was set up to get the conclusion of the CP problem in LMFBR systems and to develop its decreasing methods.

(a) Development of computer code for CP behaviour analysis

The analytical model for CP behaviour in the primary systems of LMFBR has been developed by using the results obtained from the out-pile studies which were consisted of the CP transfer experiments and the examinations of metallurgical effects of sodium exposure on the stainless steel test specimens, the plant experiences in JOYO and the published data by some foregin laboratories. This is called the solution-precipitation model, which involves the surface movement of steel wall by the thinning and precipitation on the boundary surface and the diffusion in steel and would be expected to be able to explain quantitatively many kinds of phenomena. The computor code named as "PSYCHE" has been developed to evaluate the CP behaviour in LMFBR systems (JOYO and MONJU) and the radiation dose rate around the primary cooling piping systems. The analytical parameters in this model have been obtained by the out of pile experiments and moreover would been examined by using the data in JOYO and then the evaluation for MONJU and a large scale power breeder reactor would be performed. The aim of this task is the establishment of a general useful and higher precision computer code for plant planning.

(b) Development of CP trapping method in sodium

The effective result would be obtained by a CP trapping method which can decrease the CP concentration in sodium. The CP trapping materials have been developed to collect the soluble species of CP in sodium. It was found so far that nickel was the most effective material on gettering ^{54}Mn and ^{51}Cr, and one would be expected to be able to collect ^{58}Co and ^{60}Co a little. The characterization test of the trapping material has been conducted. The optimum settling position of the getter in LMFBR systems must been designed on the basis of CP trapping and behaviour characteristics. The CP trap installed in a fuel assembly has been developed which makes up possible to replace the nickel getter. Both types of a collector and a coating on the plenum and blanket region of fuel cladding tube have been designed. However the temperature for using the CP trap should be limitted in order that the dissolved nickel in sodium from the getter would influence the CP behaviour for wrong and the nickel getter behaviour in core must also be understood.

8.4 Sodium Chemistry and Sodium Purification

Such some kinds of impurities as oxygen, hydrogen and carbon etc. will be introduced into LMFBR sodium coolant during all phases and types of operations. These impurities influence the structural and core materials of LMFBR systems and the normal operation of components for wrong. Therefore a LMFBR plant must be operated at any definite concentration levels of these impurities. For the purpose the following studies have been performed.

- To develop the devices and techniques to remove injurious impurities and keep the concentration at any definite desirable levels and to develop the in-sodium chemical meters and the techniques to monitor if the concentration of impurities is kept at the desirable level.
- 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.

The test items which have been performed latest are as follows.

(a) Sodium impurity measurement by metallic specimen equilibration methods

The impurity level in sodium can be calculated from analyzed value of the impurity in the metallic specimen by using the equilibrium paritition ratio to the impurity level in sodium. This is the metallic specimen equilibration method. These methods for oxygen and carbon have been performed. The satisfying result was obtained by using the vanadium wire method for oxygen.

The tests for carbon have been performed by using the metallic foil of SUS304L and Fe-12Mn. The exposure tests of these materials in gas mixture have been performed at the same time to remeasure the foil carbon concentration vs activity relationship.

(b) Regeneration test of sodium purification system

The conceptual design study has been completed to regenerate the cold trap in the secondary sodium purification system in MONJU and after, that was the method of thermal decomposition of hydride and sweeping gas from the liquid surface.

8.5 Sodium Removal and Decontamination

Since 1982, decontamination studies for CP in fast breeder reactor are being made in the laboratories at O-arai Engineering Center.

A series of elementary tests has been carried out to develop a decontamination process of radioactive corrosion products (CP) in FBR by using the sample gained from the sodium loop which was contaminated by CP. The deposition characteristics of CP in sodium system and the physical and chemical decontamination processes have been studied. The results obtained in the present study will be utilized effectively for developing the decontamination process of sodium components and sodium cleaning equipments in JOYO and MONJU.

8.6 Miscellaneous

Full size cold trap of MONJU primary cooling system has been tested on an improved model based on data obtained from past experiences. The test includes forced life test and regeneration test as well as ordinary characteristics test.

9. Development of FBR Instrumentation

9.1 Nuclear Instrumentation

9.1.1 In-Core Fission Chamber

Development of micro fission chamber to provide for the instrumented subassemblies of JOYO has been completed.

The instrumented subassemblies will be loaded in JOYO MK-II core at the end of 1984.

9.1.2 Ex-vessel Fission Chamber (FC) and compensated Ionization chamber (CIC)

High performance FC and CIC for Nuclear Instrumentation of MONJU have been developed.

Endurance tests in JOYO Mark I have been carried out for each chamber and tests results were excellent.

9.1.3 Ex-vessel BF_3 Proportional Counter

Ex-vessel BF_3 proportional counter was tested under the temperature up to 250°C. Neutron sensitivity was about 4.5 cps/nv and it was kept constant even when the counter was exposed to gamma flux of 200 R/h.

Reliability tests of an improved BF_3 proportional counter are now being carried out in Japan Research Reactor -4.

9.2 Failed fuel Detection and Location

9.2.1 FFD

For cover gas monitoring system, a moving-wire-type and a fixed-wire-type precipitators have been developed, and the results of the performance tests were excellent.

9.2.2 FFDL

The tagging gas system has been developed for locating the failed fuel subassembly.

A simulation test of the cryogenic adsorption system for Monju is now being carried out.

The neutron irradiation test of tagging gas and tagging gas transfer test are to be carried out in the test subassemblies of Joyo MK-II core to verify the tagging gas system.

9.3 Early Warning System for Fuel Failure

9.3.1 Temperature Measurement

The performance and reliability of C.A. (Chromel-Alumel) thermocouples under irradiation condition by JMTR were investigated and failed thermocouples were inspected in a hot-laboratory.

9.3.2 Flow Measurement

New type eddy-current flow/temperature sensors were developed and tested in a sodium loop. Furability tests in high temperature atmosphere are completed.

A flow blockage test is being performed using the flow sensors and seven mock-up subassemblies of MONJU fuel.

9.3.3 Other Systems

An acoustic detection system is being developed for purpose of detecting some anomalous sound, in particular the onset of local boiling in the core. Experiments and analysis are being performed on the acoustic propagation and sodium boiling sound characteristics. A correlation method using the reactivity and acoustic signals were investigated on the TCA (Thermal Neutron Critical Assembly) in JAERI.

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. Some tests related to these items are in progress at the O-arai Engineering Center.

A 24-inch ultrasonic flowmeter was tested, for the application to the calibration of the electro-magnetic flowmeter in JONJU.

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

9.5 Surveillance

9.5.1 Under Sodium Viewer

Two types of USV, horizontal and vertical, are being developed. Under-Water Imaging test of the horizontal type USV system which functions as an acoustic sweeper was finished.

Under-Sodium Imaging test of the vertical type USV system, which visualizes the upper surface of the core barrel, was finished using a 3/10 scale model of MONJU reactor vessel.

9.5.2 Sodium Leak Detection System

A sodium ionization detector (SID) and aerosol trapping filter has been tested at the leak rate of the order of 100 g/h in the simulated environment of a primary cell of MONJU and test results were excellent.

Radiative Ionization Detector (RID) has been developed to apply the secondary gas to sodium leak detector of Monju.

Testings in the simulated environment of secondary system for MONJU are being carried out from september, 1984.

10. Fuel and Materials

10.1 Fuel Fabrication

The fabrication of "JOYO" MK-II fuel is now being carried out at the PFFF (Plutonium Fuel Fabrication Facility).

The detailed design of the "MONJU" fuel assembly was almost fixed and the building construction of the PFPF (Plutonium Fuel Production Facility) for "MONJU" fuel was completed. Fabrication of equipment for PFPF is in progress and the installation was initiated.

10.2 Fuel Pins

Irradiation of "MONJU" reference fuel pin was completed in Phenix. Preliminary postirradiation examination indicated superior performance of the reference cladding at peak neutron fluence of 1.9×10^{23} n/cm^2 (E>0.1 MeV).

Steady state fuel pin performance code CEDAR was successfully developed. Transient performance code CEDAR-T is under development incorporating the data from EBR-II operational transient testing.

Melting point and thermal diffusivity measurement apparatus for irradiated fuel were developed and installed in the extended hot-cell in AGF (Alpha-Gamma Facility). The melting point of mixed oxide fuel was confirmed up to the burnup level of 110,000 MWD/MTM.

10.3 Core Materials

Optimized type 316 stainless steel had been finalized for "MONJU" and "JOYO MK-II" applications. Out of reactor testings for design data evaluation were completed and those cladding specimens are being irradiated in FFTF.

For backup materials to the 316 modified stainless steel, six candidate austenitic stainless steels had been selected. Out of reactor evaluation for mechanical properties and FCCI (fuel cladding chemical interaction) screening was completed. Irradiation testing of two of those candidate materials was initiated in FFTF.

As the third generation of core materials, development of precipitation-strengthened high Ni alloys and ferritic steels are under way. Cladding fabrication testing of 12% Cr ferritic steel was initiated.

Construction of MMF-2 (extended Materials Monitoring Facility) had been completed and hot operation was started for expanded postirradiation examination of core materials as well as reactor structural materials.

10.4 Subassembly

Postirradiation examination and evaluation of "JOYO" MK-I driver fuel subassemblies were completed. The fuel subassemblies exhibited satisfactory performance without detrimental subassembly deformation or without any indications of fuel pin breach. Based on these experience, the MK-I fuel designing and fabrication techniques were totally confirmed and established a mile stone to the next step of fast reactor fuel development.

Fabrication techniques for "MONJU" integrated spacer pad and cromium-carbide coating were established.

Fabrication study of cladding and wrapper tube for the large fuel subassembly is in progress.

10.5 Irradiation Experiments

1) JOYO MK-II

Performance of irradiation test vehicles were confirmed by the postirradiation examination of four types of rigs discharged during the initial five cycles.
Irradiations were completed for fuel fabrication parameter study using "JOYO" MK-II fuel pins and are under examination.
The "MONJU" fuel pins and subassemblies irradiation are being conducted.
Instrumented fuel pins for the measurements of fuel temperature, fission gas pressure, etc. were fabricated and is ready for assembling as the first irradiation experiment in MK-II.

2) Foreign Reactors

Safety related irradiation experiments in Siloe reactor were successfully completed.
Bundle irradiation in FFTF for wear mark investigation was completed. Although the detailed examination is in progress, the preliminary examination indicated the wear mark suppression as predicted from the results of "JOYO" MK-I experiments.
Operational reliability testing in EBR-II made a significant progress in fuel pin overpower capability and RBCB (run beyond cladding breach) operation. The preliminary results indicated relatively large design margin of overpower-to-breash level and the fuel pin RBCB behaviour had gentle characteristics.

11. Structural Design and Materials

11.1 Development of Structural Design Methods

11.1.1 Structural Analysis Methods

a. Inelastic structural analysis program

Extension of the general purpose inelastic structural analysis program FINAS has been made since 1981, particularly with respect to inelastic analysis options, dynamic analysis capabilities and output options. FINAS is currently used by users in PNC and allied fabricators.

b. Simplified analysis method of tubesheet-shell structures

Simplified analysis procedures combining axisymmetric and plate models are being developed for tube-sheet-shell structures.
Methods of predicting local stresses and strains in the central region of ligament as well as rimligament region are explored for thermal transient loads.

11.1.2 Structural design guides

a. Improvement of MONJU design guides and supplements

The following items are discussed on task force basis to establish the improved MONJU design guides and supplements.

- Material strength standard
- Inelastic analysis methods
- Stress reports
- High temperature sodium valves

b. Establishment of application guidelines

As application guidelines of MONJU design guides, the following items are established.

- Design methods of welds
- Buckling safety factors
- Design methods of tubesheet-shell structures
- Bolts etc.
- Cell liner bolt

c. Improvement of design post-processor

The design post-processor POST-DS based on the MONJU structural design guide for elevated temperature service has been developed and improved. The analysis results obtained by FINAS are easily incorporated into POST-DS.

11.2 Structural Test and Evaluations

11.2.1 Structural element and component tests

In order to evaluate the adequacy of high temperature design rules and analysis methods and also to confirm the integrity of the actual components, the following structural element and component tests have been or are being performed.

a) Creep fatigue tests of elbows in sodium
Tests and evaluation are completed.

b) Fatigue tests and creep buckling tests of T-joints
Fatigue tests and creep buckling tests are completed.

c) Thermal and creep ratcheting tests of pipes and elbows
Creep rachtting tests on 2¼ Cr-1Mo pipes and elbows are completed.

d) Creep tests of cylindrical shell with axial temperature gradient
Test completed.

e) Creep tests of beam with primary and secondary stresses
Tests under way

f) Biaxial stress relaxation tests of cylinder
Tests under way

g) Elevated temperature tests of piping bellows
Creep fatigue test rig was completed in Jan, 1985.
Buckling test ring will be completed in June, 1985.

h) Thermal transient tests of SG tubesheet Model
Testing facility is completed and the first test is under way.

i) Thermal transient test by TTS
The thermal transient test facility for structures (TTS) was constructed, and the trial test on vessel was completed. The first test on a uessel model was started.

11.2.2 Thermal and hydraulic tests in reactor components

Thermal and hydraulic tests are being performed to capture complex thermal boundary conditions for structural design of reactor components.

a. Thermal stratification tests of 1/6 - scale and 1/10 - scale models of MONJU upper plenum Tests completed.

b. Structural integrity tests of reactor vessel with sodium level Phase 1 test and Phase 2 test completed.

c. Thermal striping tests for UCS
Water test completed
Sodium test under way

11.3 Structural Material Test

Research and development (R&D) on structural materials has been conducted to refine and/or revise the material strength standard for "MONJU" in air, sodium and irradiation environment. In Fig. 11-1 are shown the R&D items in air and in sodium environment necessary to the design of fast breeder reactor.

The new R&D program for the tests in air and in sodium environment (called "CApella" program) is started. Step-1 program (1985-1987) includes both R&Ds for "MONJU" and Capella future reactors (especially demonstration FBR).

The neutron irradiation test is conducted according to the neutron irradiation program.

11.3.1 Structural Material Test in Air

In-air structural material tests for the development of "MONJU" strength standard have been conducted since 1977 as mentioned below.

In Step-1 program (1977-78), 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. Tested properties were as follows;

- tensile test,
- creep test,
- relaxation test,
- fatigue test,

and
- creep-fatigue test.

In Step-2 program (1979-81), the above tests were continued to prepare material strength standard for "MONJU" structural design guide, and some theoretical methods on material strength of behavior were investigated by the following tests,

- creep damage estimation test during strain-hold,
- inelastic strain behavior test,
- evaluation test on strength of welded-joint,
- notch effect test,

and
- other specified test.

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

Step-3 program (1982-84) has been completed in March, 1985. The objective of this program was to refine the above design guide and standard.

And the prototype FBR "MONJU" is designed in accordance with this standard.

Now is under the way of the R&D in Capella program.

This study have two principal purposes.

One is to clarify the application limits of the present evaluation methods and to improve the accuracy of these methods. Another is to select new proper structural materials for FBR.

The cost reduction for the construction of demonstration FBR will be derived from the rationalization of the design due to the improvement of the accuracy of the evaluations and the application of new proper materials.

For the purpose of mentioned above, Capella program is planned to develop the following technologies;

- level-up of "MONJU" technology for cost-down, (creep-fatigue life evaluation, strength of weldment, inelastic constitutive equation, and others)
- design and fablication of large-scale structure,
- modification of material specification,

and
- application of fracture mechanics.

11.3.2 Structural Material Test in Sodium and Water

a. In-sodium Test

Sodium environmental tests on structural materials for "MONJU" have been performed on Inconel 718 alloy which will be used for the thermal striping resistance for above-core structures. Tests on bellows material and carbon transfer behavior in the secondary sodium circuit of "MONJU" will be also continued.

A new sets of sodium environmental effect tests according to Capella program will be started on possible candidate alloys for future demonstration FBR, such as high chromium ferritic steels, alloy 800, and modified type (low carbon and/or high nitrogen) austenitic stainless steels. Testing items included are corrosion/mass transfer tests, carbon transfer tests, and mechanical strength (tensile, creep, fatigue, creep fatigue) tests in sodium.

And corrosion tests of materials for steam generator were started in the contaminated high temperature sodium added NaOH or Na_2O, supposing that water leak in the steam generator.

b. In-water (steam) Test

Stress corrosion cracking tests of SUS304, 316 and 321 were carried out in the wetted steam to confirm the unsusceptibility after water leak in the steam generator of "MONJU". It is confirmed that above materials are unsusceptible to S.C.C in the wetted steam cluding dissolved oxygen and chloride ion up to 200ppb. The next objective is to confirm unsusceptibility of the plugs of the steam generator tubings (after plugging) in the same wetted steam.

11.3.3 Structural Material Test in Irradiation Environment

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

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

Surveillance test data are used to make the operating program of "JOYO".

The first time surveillance tests were completed on the materials of core barrel, core support plate, reactor vessel and safety vessel fo "JOYO".

It was not deduced from the evaluation of first time surveillance tests that the operating schedule of "JOYO" and the withdrawal schedule of surveillance test specimens should be changed.

As for "MONJU", its surveillance test program is under final refinement.

b. Research and Development Tests

Research and development tests have been conducted on the structural materials (such as SUS304) steel for primary components of "MONJU" to evaluate the irradiation effects on the mechanical properties up to the end of design life time and to introduce the irradiation effect rationally to the materials strength standards for "MONJU".

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

Uniaxial in-pile creep test on rolled SUS304 steel has been conducted in JMTR (Japan Material Test Reactor) to compare it's results with those of post-irradiation creep.

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

c. Miscellaneous

The new facility (Material Monitoring Facility 2: MMF-2) started to test irradiation material in April, 1984. This facility is equipped with five uniaxial crrep test machines, two creep-fatigue machines and others for irradiated structural material tests.

11.3.4 Data Banking System

Material test data were compiled by the specified data coding sheets, and inputted to computer system for data banking. If we refer these data, these system is available and refered data can be stocked in user's file of computer. This data banking system is called SMAT, and has been already developed preliminarily. Furthermore, in order to avail this system effectively, data analysis system including statistical analysis and graphic processor was developed preliminarily.

These total system will be improved by user's requirements in the process of operation.

12. Safety

12.1 Thermohydraulics

12.1.1 Thermohydraulic analysis Codes

1) Multi-Dimensional In-Vessel Thermohydraulic Analysis

The work using COMMIX code in this period includes: JOYO MK-I (75MWt) natural circulation test analysis, analysis of invessel thermohydraulic test conducted at CRIEPI, and other analyses involved in the experimental works currently underway. Continuous efforts are also being made on the improvement of numerics on COMMIX code.

2) Plant System Transient Analysis

A Monju auxiliary cooling system (ACS) model has been implemented into SSC code and the ACS test conducted at 50 MWSGTF is being evaluated for the purpose of code validation. Also, SSC was used for evaluating JOYO MK-I natural circulation test. A preliminary study is underway to apply the code to the consequence analysis of possible events such as LOPI and PLOHS of Monju plant system.

3) Fuel Subassembly Analysis

ASFRE, a single-phase subchannel analysis code with a wirewrap model, has been applied to a number of in-pile as well as out-of-pile pin bundle tests including: (a) steady-state 19-, 61-pin isothermal water experiments, (b) steady-state 37-pin heated sodium experiments with/without central blockage, and (c) 37-pin sodium flow rundown experiments. SPIRAL (a new name for SPOTBOW-3D), a three-dimensional, single-phase subchannel analysis code, is currently under development aiming at predicting detailed temperature and turbulent flow velocity fields around distored fuel pin.

For two-phase flow, SABENA, a three-dimensional sodium boiling analysis code, proved itself an excellent capability by providing a good prediction of KfK L-22 loss-of flow experiment using a KNS 37-pin bundle as the PNC's participation in the 11th LMBWG benchmark exercises.

12.1.2 Decay Heat Boiling Test

The modification work of the Sodium Mixing Test Facility to accomodate a DHB test loop has been completed and currently preliminary tests are underway. A new test bundle (37H) is a 37-ing bundle simulating the full length of Monju fuel subassembly with a chopped-cosine heating profile. Objectives of DHB test are (a) examination of steady dryout conditions under wide parameter regions including reversal flow conditions and (b) examination of boiling suppression and/or excursion produced by the interacting buoyancy and two-phase pressure drop effects.

12.1.3 LOPI Test

The construction of new test facility called PLANDTL (Plant Dynamics Test Loop) is planned in the coming two year period, which is capable of providing variety of thermal and flow transient conditions. The first test scheduled to begin in 1987 is a demonstration of the decay heat removal capability under the simulated LOPI conditions of Monju plant. A preliminary design was completed.

12.1.4 Decay Heat Removal Tests by Natural Convection

A 19-pin bundle is used to study natural convection under such conditions as LOPI and loss-of-power. The test has just started in the Sodium Mixing Test Facility. The data will be used to validate codes such as ASFRE, COBRA-IV, etc. Some of the DHB/37H tests will provide similar data.

The decay heat from irradiated fuel subassemblies stored in the ex-vessel storage tank (EVST) may be removed by means of natural convection in the tank in which a cooling coil is inserted. The confirmatory test is planned for this scheme using a simple water model. The test rig is almost completed and the test will begin soon.

12.2 Reactor Safety

12.2.1 Accident Analysis Codes (Whole Core Accident Analysis)

Several computer codes are used and maintained in a study of whole core accidents in Monju and future reactors. These include: SAS3D for initiating phase; SIMMER-II for core disruption phase; and SAME for SAS3D/SIMMER-II data transfer. SAS4A was introduced from Argonne National Laboratory and the verification effort is in progress through comparing with SAS3D and analyzing the CABRI and STAR experiments. Several graphics codes also have been developed for post-processing SAS4A and SIMMER-II.

The PAPAS-1S code was upgraded to PAPAS-2S, which incorporated the EULFCI (fuel-coolant interaction) and REMUS (transient fission gas release and fuel swelling) modules.

The CALIBRE code using the response surface method was developed for sensitivity analyses with accident analysis codes.

The NEUSAFE system (Standard Neutronics Code System for FBR Safety Analysis) has been developed for providing consistent neutronics data to the accident analysis codes, and it is being used in various accident and reactor physics analyses at PNC.

12.2.2 Fuel Failure Propagation

PNC negotiated with the Japanese Government to join the SCARABEE program of the French CEA and obtaind financial permission in the end of 1984. PNC is going to join the BE series tests which deal with failure propagation from a failed subassembly completely blocked at the inlet to adjacent subassemblies.

12.2.3 Molten Core Material Interactions

1) Out-of-pile Experiments on Molten Fuel Behavior in a Pin Bundle (completed)
   Out-of-pile tests to investigate the molten fuel behavior and the fuel sodium interaction (FSI) phenomena under conditions simulating transient overpower (TOP) and loss-of-flow (LOF)-driven TOP accidents have been completed.

   The final results indicated that the FSI is mild and that the most of released fuel is swept out without any significant blockage formation in the flow area.

2) Out-of-pile Experiments on Molteu Fuel Jet-Material Interactions

   New project of out-of-pile experiments on molten fuel jet-material interactions was initiated to investigate Post-Accident Debris Relocation Behaviors in HCDA.

   A test facility to simulate the molten fuel jet by low melting alloy is under construction to get the fundamental informations on the break up behaviors of the molten fuel jet in the coolant and the thermal attack behaviors of the molten fuel jet on structural materials such as stainless steel.

   The large facility to melt UO_2 ($\sim$10kg) by induction heating was also designed for the future experiments on the molten fuel jet-material interactions.

12.2.4 Transient In-Pile Tests

1) CABRI

   PNC has participated in the joint CABRI project as a junior partner since 1975, stationing a delegate at Cadarache, France. The tests with a fresh or a low burn-up fuel pin were started.

   Useful information has been accumulated on the behavior of fresh and irradiated fast reactor fuel pins under rapid power transient conditions. The PAPAS and SAS3D codes have been validated through the experimental analyses (pre- and post-test calculations).

   The latest activity was devoted to the post-test analyses of TOP-type tests, focusing on the fuel failure mechanism and post-failure MFCI phenomena, while the pre-test analysis concerning driving mechanisms of the disrupted fuel was made for the LOF-driven-TOP type tests.

2) STAR

   PNC has joined the STAR program (Sandia Transient Axial Relocation) at Sandia National Laboratories. The program intends to observe directly the behavior of fuel and cladding during simulated unprotected loss-of-flow accident sequences. The results will be used to improve the initiating phase analysis.

12.2.5 Large Scale In-Pile Tests (T

PNC joined the PNC-NRC joint study of fuel removal potential during the late initiating and transition phases (TRAN Program at Sandia National Laboratories, Albuquerque).

12.2.6 PAHR In-Pile Tests

PNC joined the Joint Debris Bed Program since 1980 which are conducted under USNRC programmatic management st Sandia National Laboratories.

A two-dimensional boiling and dryout model for the debris bed was developed and coupled with a heat conduction code, TAC2D. The TAC2D code with the mocel was applied for analysing the debris bed cooled from top and side boundaries.

12.2.7 Core expansion tests

An out-of-pile experimental program has been planned to study the phenomena of core expansion during an HCDA due to a neutronic power burst or disassembly. The first series of the tests will be performd by injecting high pressure steam into water through a pin bundle to observe the thermal-hydraulic effects in the upper structure of the core and the upper plenum of the reactor vessel. The test results will be compared with thermal-hydraulic calculations of the SIMMER-II for the code validation. The design and construction of the test apparatus are underway.

12.2.8 Shock Structural Experiments and Analyses (completed)

The investigations on the Structural response of the reactor vessel against HCDA energetics for Monju have been completed both experimentally and analytically. Many reports, documents and data were put in order for future needs.

12.3 Sodium Fires and Aerosol Behavior

12.3.1 Basic Test

During 1984, a simulation test of a sodium leak accident in the secondary building of the Monju plant was conducted. In the test, small models of a sodium piping equipped with thermal insulation cover, a floor liner, and a smothering tank were used, and phenomenology of a full sequence of the accident was studied.

A aerosol proof test of the Monju prototype pony motor of the primary pump and prototype mechanical snubbers was also conducted.

12.3.2 Water Simulation Test

Water simulation test of a sodium leak from secondary piping of the reactor was conducted. Size distribution of the water droplets generated by a splash flow from the defected piping was measured for the Monju prototype straight, T-type, and elbow pipes, all of which were equipped with thermal insulation covers.

12.3.3 Large Scale Test

Construction of a new facility (SAPFIRE) for the large sodium fires and aerosols tests is under progress and will be completed in May, 1985. In the facility, test rigs of a $100m^3$ vessel, a $3m^3$ vessel, and a reactor boilding simulating two story high concrete cell are to be installed. Performance test of the Monju fire mitigation systems, aerosol test, large pool and spray fires tests will be started in June 1985.

12.3.4 Computer Code Development

During this period, a new sodium pool fire code SPM (Sodium Pool Fire Model) has been developed. This code calculates heat and mass transfer in the presence of a flame sheet at just above the pool surface.

Development of a natural covection code was started. Prime objective of the development is to improve the heat and mass transfer calculation of the atmospheric gas under the turbulent natural convection conditions for the new pool fire code.

Validation of ASSCOPS, combined pool and spray fires code, is in progress. During 1984, recalculations of the test results obtained with the FAUNA facility, KfK, Federal Republic of Germany, have been conducted.

12.4 Radiological Consequences

12.4.1 In-Pile Fission Products Behavior Test

In-pile tests with FPL-II (in-pile sodium loop) to study the fission products behavior in the flowing sodium system had been completed.

During this period, the loop was disassembled and the detailed examination of the fission products deposits on the stainless steel walls have been conducted by using the test pieces from different locations in the loop. In the meantime, fuel particles that had been irradiated in the loop were sent to a PIE facility for their examination and to determine their roughness factor. Analyses of gamma-ray spectra of the test pieces were conducted and the results obtained revealed that the deposition rate constants of the non-volatile fission products so far investigated depended upon sodium flow rate, temperature, and oxygen concentration of sodium. Additional analyses of the gammaray spectra in regard to the other non-volatile and volatile fission products are under progress.

Ultimately, the deposition rate constants obtained and to be obtained will be used as input data for a computer code that calculates radioactivity distribution within a primary system of the fast reactor.

12.4.2 Containment Response and Consequence Analysis

The CONTAIN code developed by Sandia National Laboratories, USA has been tested, and its application to the Monju consequence analysis was started.

During this period, a modification was made to the sodium fire module to treat unburnt sodium. Verification tests of the modules for sodium fires, aerosol behavior, fission products behavior, condensation and chemical reactions in the atmospheric gas were conducted. A new module for sodium-concrete reaction is under development.

12.4.3 Sodium-Concrete Reaction

Third series of the test has been conducted during this period.

An objective of the test was to investigate the effect of concrete thickness on the reaction as well as to study the interaction of evolved hydrogen with sodium and oxygen impurity in nitrogen cover gas over sodium. The latter study was conducted with the cover gas oxygen concentration in the range from 0 to 21%.

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