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A REVIEW OF FAST REACTOR PROGRAM IN JAPAN

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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 eighteen years. Concerning the experimental fast reactor, JOYO, the MK-II core attained criticality on November 22, 1982 with 51 fuel assemblies, and received the "Certificate of Inspection before Operation" from Government Authority on March 31, 1983, after 100 hours operation with the rated output of 100 MW. Since then, the core has been utilized to implement irradiation bed characteristics test, and to irradiate fuels and structural materials especially for the prototype reactor MONJU.

With respect to the prototype reactor MONJU, the installation permit was issued on May 27, 1983, from the prime minister, and the contracts of the first stage between PNC and fabricators were made recently. At the same time, almost all the licenses of preparatory construction works were issued by March 1983, and preparatory construction works were started in April 1983.

On the other hand, conceptual design of a demonstration reactor is now under way in a close cooperation with concerned authorities and utilities, as well as investigations of the way of conducting necessary research and development.

2. Experimental Fast Reactor "JOYO"

2.1 General Status

This report covers the activity of the experimental fast reactor in the period from 1983.4 through 1984.3. After completion of the power ascension testing on the Mark-II core the reactor has been operated at low power level for four months to perform characteristic tests as an irradiation bed. From August the normal cycle operation was started at the power level 100 MWt. In the case of "JOYO" one operating cycle consists of 45 operating days and 15 refueling days. Legal annual inspection was started from the beginning of December, which continues until middle of April 1984.
For reference main items of Mark-I and II cores are shown in Tab. 2-1.

2.2 Test, Operation and Maintenance Experiences

2.2.1 Tests

Test items for core characterization are as follows:

- o Criticality

- o reactivity characteristics
  control rod calibration, core excess reactivity, scram margin, temperature, power and burn-up reactivity coefficients, fuel subassembly reactivity worth

- o thermo-hydraulics
  core flow distribution, subassembly outlet temperature

- o core stability
  step disturbance test, reactor noise, pseudo random binary disturbance test

Test items for plant characterization are as follows:

- o thermo-hydraulic
  reactor thermal power calibration, heat teransfor characterization,

- o Plant stability
  step temperature and reactivity disturbance test, pseudo binary disturbance test

- o Plant characterization and monitoring
  decay heat removal, core and loop pressure drop, piping distorsion measurement, primary sodium sampling analysis, argon cover gas sampling analysis, rad-waste gas monitoring, radiation dose distribution, fuel failure monitoring and rotating component vibration monitoring.

Some of main results of testings are given in Tab.2-2,2-3 and Fig.2-1 and 2-2.

2.2.2 Operation

The Mark-II core consists of 64 driver fuel subassemblies and 3 irradiation test rigs. At the full power operation the reactor outlet temperature reached 500°C with 370°C at the inlet. The highest sodium temperature at the fuel subassembly outlet was 554°C which agrees well with the prediction based on measured flow rate. It was also confirmed that all four dump heat exchangers could remove the heat with an air flow margin of 20%.
Operating history of "JOYO" is shown in Fig.2-3 and the irradiation program is shown in Fig.2-4.

2.2.3 Maintenance

Sodium Vapor Deposition on the Rotating Plug

A jack down failure of the rotating plug occurred during refueling on January 11, 1982, for the first time. At this time, the large rotating plug became incapable of completing the jack-down movement, remaining stack 13mm short of a full vertical stroke of 20mm. After repeated plug rotation and jack down operations, the plug settled in the bottom position.

Investigation revealed that presumably, some material found its way onto the rubbing part of the rotating plug's metal ground surface, thereby resulting in strange noise and movement while jack-down operations were being made. Judging from its behavior, it was believed that the interfering matter was soft sodium, and the adhesion spots were the socket and spigot section with the smallest gap, exposed to a gaseous atmosphere in the reactor vessel. To confirm this belief an observation of this section by insertion of a fiber scope into the inspection hole was made, and sodium deposition was observed.

On October 25, 1982, an observation of the socket and spigot section was carried out. Observation was done with a T.V. camera connected to a fiber scope inserted into the inspection hole. The large rotating plug was rotated, and the full circle observation was completed. As the result of this observation, a large amount of sodium deposition was discovered. Sodium deposition in greater proportions was witnessed within the 0 to 90 degree range. Smaller sodium deposition has been observed in the 180 degree direction, most of which has resulted in rolled form after having been scraped off by the plug rotation.

It is believed that the sodium deposition is created by natural convection of the cover gas generated in the reactor vessel, as observed in numerous tests, and that the ascending current of the argon cover gas containing sodium mist takes place from an area in the 0 degree direction in this natural convection. The mist gradually deposits on the socket section of the rotating plug. Presumably, the cover gas with a much decreased mist flows along the socket section in the 180 degree direction, changing into a descending current.

The temperature distribution was also investigated, on the outside of the large rotating plug. The result indicates higher temperatures on the 0 degree side and lower on the 210 degree side, giving rise to a temperature difference of 60 °C. at the thermal shielding plates that are attached to the lower side of the plug body in the case of 50 MWt reactor output. The difference is 66 °C at a reactor output of 75MWt, and 76 °C in the case of 100MWt. Thus the increased output raises temperature differences in the peripheral directions. Therefore, if the scale of natural convection is proportional to the temperatures and temperature differences of the peripheral direction, sodium deposition is considered to increase in accordance with the increased output of JOYO at 100 MWt.

To cope with this, plans are to enlarge the inspection hole at which observations were performed, from 15 mm to 40 mm diameter. At the same time, new devices are being developed to scrape off deposited sodium as an effective means of sodium removal. The deposion place is shown in Fig.2-5.

Overhaul of the Primary Main Pump

The overhaul and inspection of the primary main pump contaminated with activated sodium and corrosion products was successfully conducted after reactor operation of 13,000 hours.

The pump inner casing was removed using the pump maintenance cask in order to prevent the leakage of radiation and primary argen cover gas. After removal, the pump was sodium cleaned and decontaminated. In the steam and water cleaning method, activated sodium adhering to the surface of the pump inner casing was completely removed although some sodium remained in the bolt holes. The decontamination of corrosion products was mainly accomplished by the brushing and wiping method. Through the decontamination, 89 % of ^{60}Co and 27 % of ^{54}Mn were removed, and this difference is considered to be that the decontamination work by these methods was not effective for ^{54}Mn since it diffused into the steel. The radiation level of the pump inner casing surface was reduced from 600 mR/h to 50 mR/h.

In addition to the inspection work, baffle plates were attached to the inner casing of the pump. The function of the baffle plates was to prevent the pump cover gas from circulating in the annulus between the pump inner and outer casing. The pump characteristic tests demonstrated the effect of the natural convection barrier, and resulted in the circumferential temperature difference of the pump inner casing decreasing from the range of 20 to 50 ^{o}C into the range of 0 to 4 ^{o}C.

The experience of this overhaul and inspection demonstrated the ability to handle large scale equipment contaminated with activated sodium and corrosion products, and established a criterion for maintenance of the pump.

Corrosion Product Deposition on the Primary Cooling System Piping and Components

The JOYO Mark-I core operation was completed at the end of 1981 without any fuel failures, and the accumulated thermal power and the maximum fuel burn-up were about 28000 MWD and 40100 MWD/T, respectively. JOYO has not been contaminated by fission products. The dose rate of the primary piping cell areas, however, gradually increased due to the deposition and build-up of corrosion products inside the primary cooling system piping and components with continued operation. Although the primary coolant sodium is drained to the dump tanks during the annual inspection and maintenance period, deposited corrosion products remain on the inside of the primary cooling system piping and components.

Radiation measurements (radiation dose rate and activity measurements) have been conducted to define the behavior and distribution of corrosion products in JOYO plant since 1978. These measurements are performed during the annual inspection and maintenance period with the coolant drained. The measurements of radiation dose rate are taken at about 100 locations (about 800 points) of the surface on the primary cooling system piping and components using a portable G-M tube type surveymeter and Thermo-Luminesence Detectors (TLD). The measurement of corrosion product activity is performed at 16 piping locations (A-loop 13 locations, B-loop 3 locations) using a Germanium Solid State Detector (Ge-SSD).

Along with these measurements, a corrosion product behavior analysis code has been developed and verified with the measured radiation data, in order to analyze the distribution of corrosion product in the primary cooling system. Further, to estimate the radiation exposure of maintenance personal, a three dimensional analysis code for the primary piping cell radiation distribution has been also developed.

Summary of results are as follows;

(1) The radiation source of the primary cells is deposited corrosion product, and the main nuclides are ^{54}Mn and ^{60}Co.

(2) Deposited corrosion products are built-up with the reactor operation time, and the activity is increasing together with the accumulated thermal power (Fig.2-7).

(3) The calculations by the analysis code agreed well with the measurements of corrosion products which deposited on the piping in the order of 10^{-2} Ci/cm^2 for ^{54}Mn and 10^{-1} Ci/cm^2 for ^{60}Co.

(4) After 18-year operation, radiation dose rate at the primary cold leg piping is estimated about 100 mR/h by the analysis code (Fig.2-8).

2.3 Future Plans

Establishment of Preventive Maintenance Techniques

To evaluate the operating and maintenance experience of JOYO, data on incidents and troubles occurring at JOYO have been collected from the first critical operation. The total numbers of incidents recorded but not resulted in unexpected reactor shutdown during the period from April 1977 to March 1982 is shown below.

TOTAL NUMBER OF INCIDENTS AT JOYO

Fiscal year	Period		Number of incidents(1)
1977	April 1977	March 1978	357
1978	April 1978	March 1979	312
1979	April 1979	March 1980	251
1980	April 1980	March 1981	229
1981	April 1981	March 1982	245

Figure 2-9 shows the comparison of incident frequencies in those five years for several system of the plant.

(1) excluding the fuel handling systems

For those five years, the service building air-conditioning system had the highest frequency of incidents, accounting for 14 % of the total. The most frequent incidents for those five years were caused by process instruments. They accounted for about 30 % of the total during the period from FY 1977 to FY 1979, and the percentage decreased to about 15 % during the period from FY 1980 to FY 1981. However, the incident frequencies of recorders, which have the majority of process instrument incidents, did not change much. According to a more detailed analysis, it appears that stronger human involvement incidents are gradually increasing over the five years.

These data are analyzed in more detail to establish the component reliability data base for PRA (Probabilistic Risk Assessment).

To do a more detailed analysis efficiently on the plant operation and maintenance, a computer-aided preventive maintenance system is under development at JOYO.

This consists of the following three subsystems.

(1) Data banking subsystem
Collecting and maintaining system components' engineering data, operation history and maintenance history, this subsystem edits the components' maintenance history, and analyses components' failures and evaluates components' reliabilities.

(2) Predictive maintenance subsystem
Dealing with noise signals, acoustics and vibrations computationally , this subsystem picks up some statistical characteristics and can detect a component anomaly by the change of such characteristics. Also, the system can detect a component anomaly by referring to the dynamic model of the object component. Furthermore it can diagnose an anomaly by means of the EXPERT SYSTEM based on knowledge engineering.

(3) Maintenance control subsystem
By computerizing maintenance schedules and maintenance standards, this system can provide the information service needed for routine maintenance work, for example, printing the data when each maintenance work should begin, listing the components which need inspections or repares and so on.

When a component anomaly is discovered in the plant, the operator and maintenance engineer would infer or determine the cause, the effect and the counter-measure according to their knowledge and gained experience of the plant. Therefore, building the knowledge base of an "Expert" in to the system and inputting anomaly findings off-line and process signals on-line to the knowledge base, the "Expert system", which gives the cause, the effect and the countermeasure of component failure using "Expert knowledge" is defined into the preventive maintenance system.

As a first step to building the knowledge base, the maintenance experience gained so far is being collected on failure data sheets, which show the flow from the birth of the anomaly to the countermeasure.

Tab. 2-1 Main Core Parameters of "JOYO"

		MK-I		MK-II
		First	Second	
Reactor Output	MWt	50	75	100
Primary Coolant Flow Rate	t/h	2,200	2,200	2,200
Reactor inlet Temperature	°C	370	370	370
Reactor Outlet Temperature	°C	435	470	500
Core Stack Length	cm	60	60	55
Core Volume (max.)	ℓ	294	304	250
Linear Heat Rate (max.)	W/cm	210	320	400
Fuel Pin Diameter	mm	6.3	6.3	5.5
$PuO_2/(PuO_2 + UO_2)$	W/O	18	18	30
U^{235} Enrichment	W/O	23	23	12
Neutron Flux (max.)	n/cm²sec	2.1×10^{15}	3.0×10^{15}	5.1×10^{15}
Neutron Flux (Core av.)	n/cm²sec	1.2×10^{15}	1.9×10^{15}	2.6×10^{15}
Max. Excess Reactivity	% $\Delta K/K$	~4.5	~4.5	~5.5
Control Rod Worth	% $\Delta K/K$	Safety Rod ~5.6 Regulating Rod ~2.8	Safety Rod ~5.6 Regulating Rod ~2.8	~9
Max. Burn Up (pin av.)	MWD/t	25,000	42,000	50,000
Operation Cycle		45 days Operation 15 days Outage		

Tab. 2-2 Control Rod Worth (MK-II)

	Experiment (%ΔK/KK')			Calc. (%ΔK/KK')	C/E
	Measured	Correction*	Corrected		
#1 (3A3)	2.27	0.957	2.18	2.33	1.07
#2 (3B3)	2.28	0.957	2.19	2.31	1.05
#3 (3C3)	2.43	0.900	2.19	2.33	1.06
#4 (3D3)	2.44	0.900	2.19	2.34	1.07
#5 (3E3)	2.45	0.900	2.20	2.36	1.07
#6 (3F3)	2.32	0.957	2.22	2.36	1.06

*Shadowing Effect Correction

Tab. 2-3 Comparison of the Measured Temperatures with the Calculated Temperatures on 100MW Power Ascension Test

Orifice Zone	Number of S/A	Mean Temperature (°C)			ⓐ − ⓑ	$\frac{a-b}{b}$ X100	Memo
		Measured	Meas.ΔT:ⓐ	Cal.ΔT:ⓑ			
Core 0,1	4	551.8	181.8	176.9	4.9	2.8	
2	12	537.3	167.3	170.9	−3.6	−2.1	
3	12	546.7	176.7	168.6	8.1	4.8	
4	24	529.3	159.3	153.5	5.8	3.8	
5	12	536.7	166.7	147.8	18.9	12.8	
Inner 1	18	486.4	116.4	84.6	31.8	37.6	
2	30	478.8	108.8	61.0	47.8	78.4	

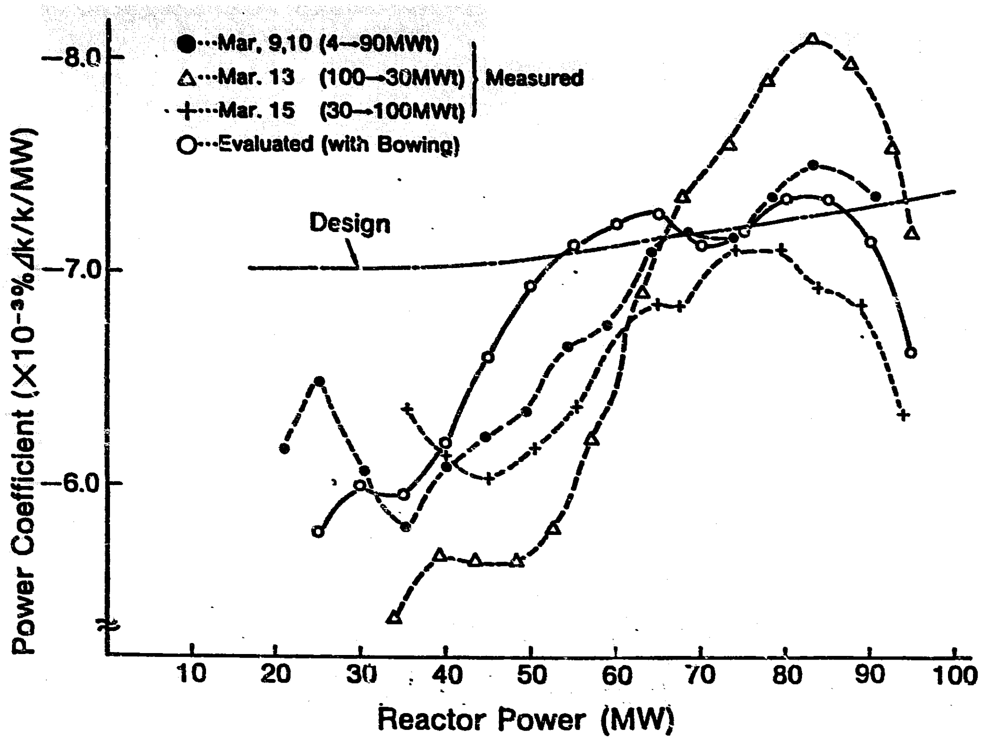


Fig. 2-1 Comparison Between Measured and Evaluated Power Coefficient

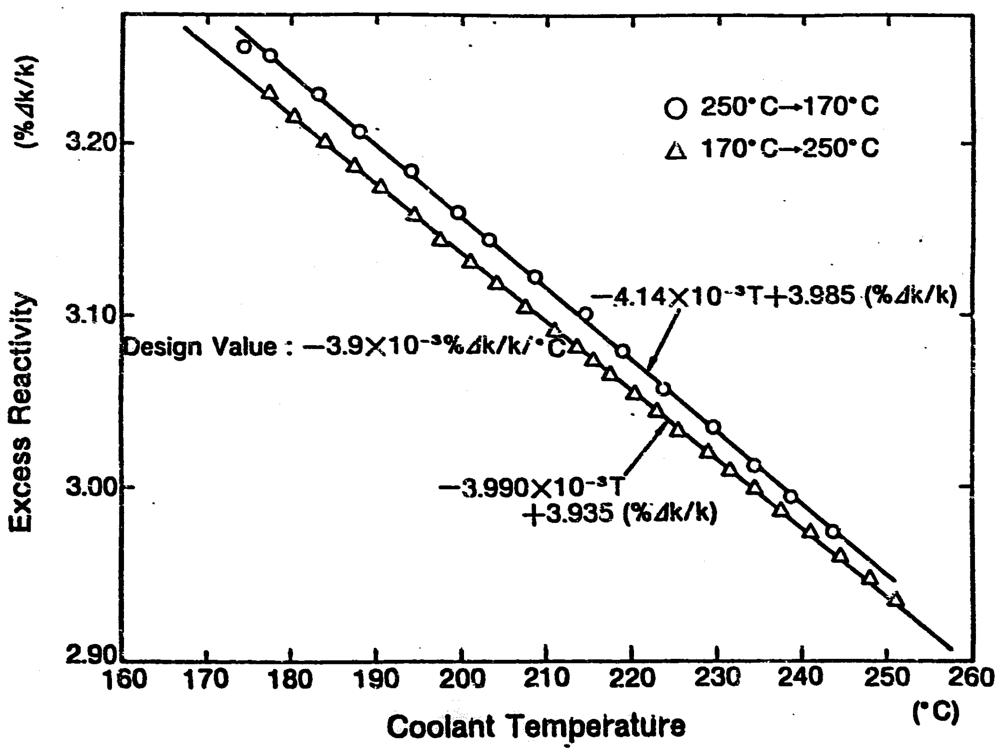


Fig. 2-2 Isothermal Temperature Coefficient

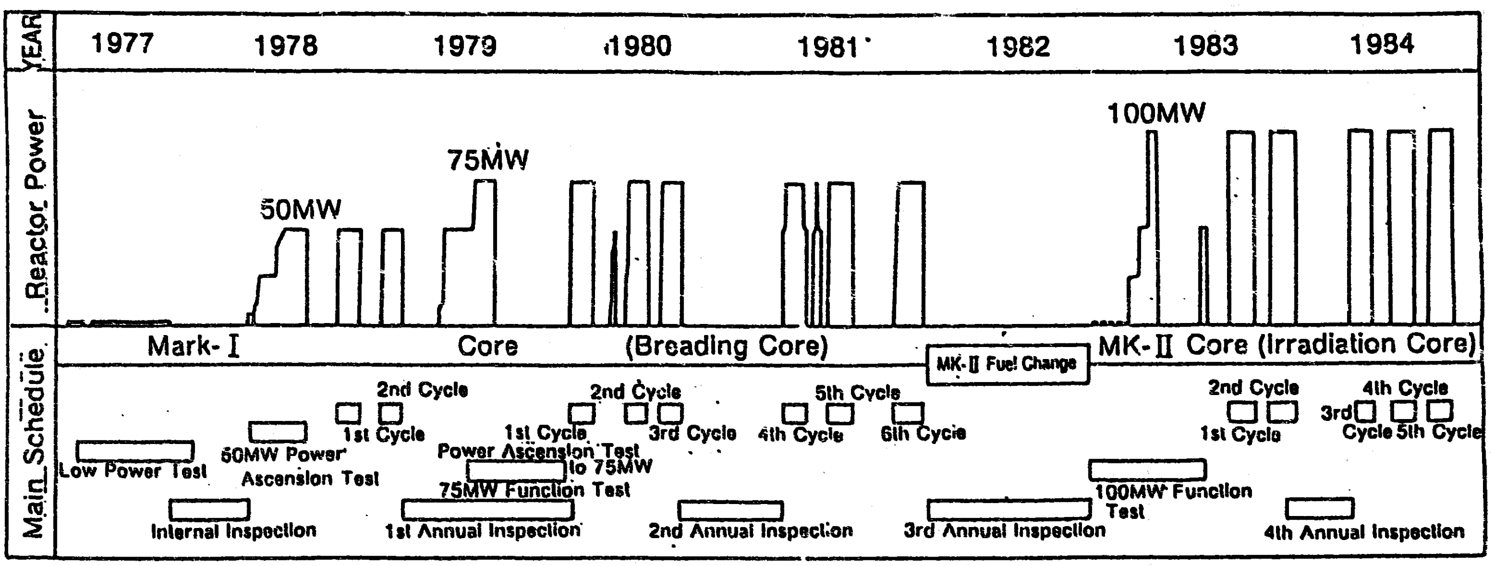


Fig. 2-3 EXPERIMENTAL FAST REACTOR JOYO OPERATING HISTORY

Fig. 2-4 Irradiation Test Programs in Joyo MK-II (Phase 1)

Irradiation Objects	Irradiation Purpose		Irradiation Schedule 1982	1983	1984	1985	1986	1987	1988
Fuel Element	Examining the performance of the fuel elements with high burn up, high power and high temperature.	UNIS-A UNIS-B	B0J CMIR-0 B1J		B2M A1M		B3M CMIR		
	Fuel failure detection methods	UNIS-B							
	Examining the behavior of the core materials under high neutron fluence	CMIR							
Subassembly	Examining interactions between fuel elements and interactions between fuel element and duct	UNIS-C	CMIR-0	C1J	C2M	INTA-1	CMIR-0 C3M		
	Behavior of the core materials	CMIR							
	Continuous monitoring of the irradiation conditions	INTA							
Control Rod and Neutron Absorber Materials	Examining the interaction between B_4C and the cladding material and performance of new absorber materials	AMIR			AMIR-1 AMIR-2				
Structure Materials	Verification of the Integlity of the structure materials	SMIR UPR		SMIR-4 SMIR-1 SMIR-3	SMIR-2	SMIR-8 SMID-5 SMID-7 UPR-1	SMIR-6 UPR-2	SMIR-12 SMIR-9 SMIR-11 UPR-3	SMIR-10 UPR-4

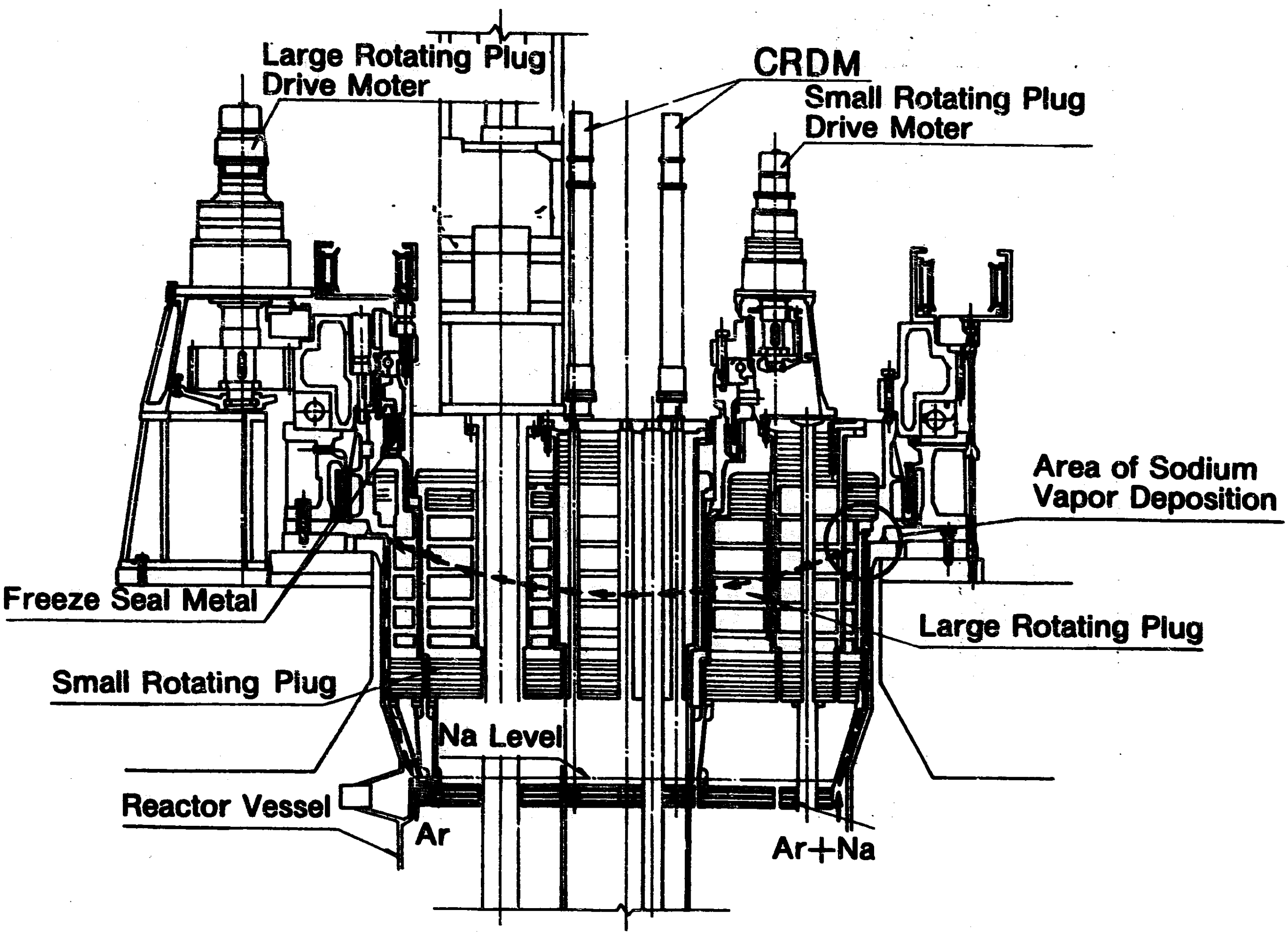


Fig. 2-5 Sodium Vapor Deposion on the Large Rotating Plug

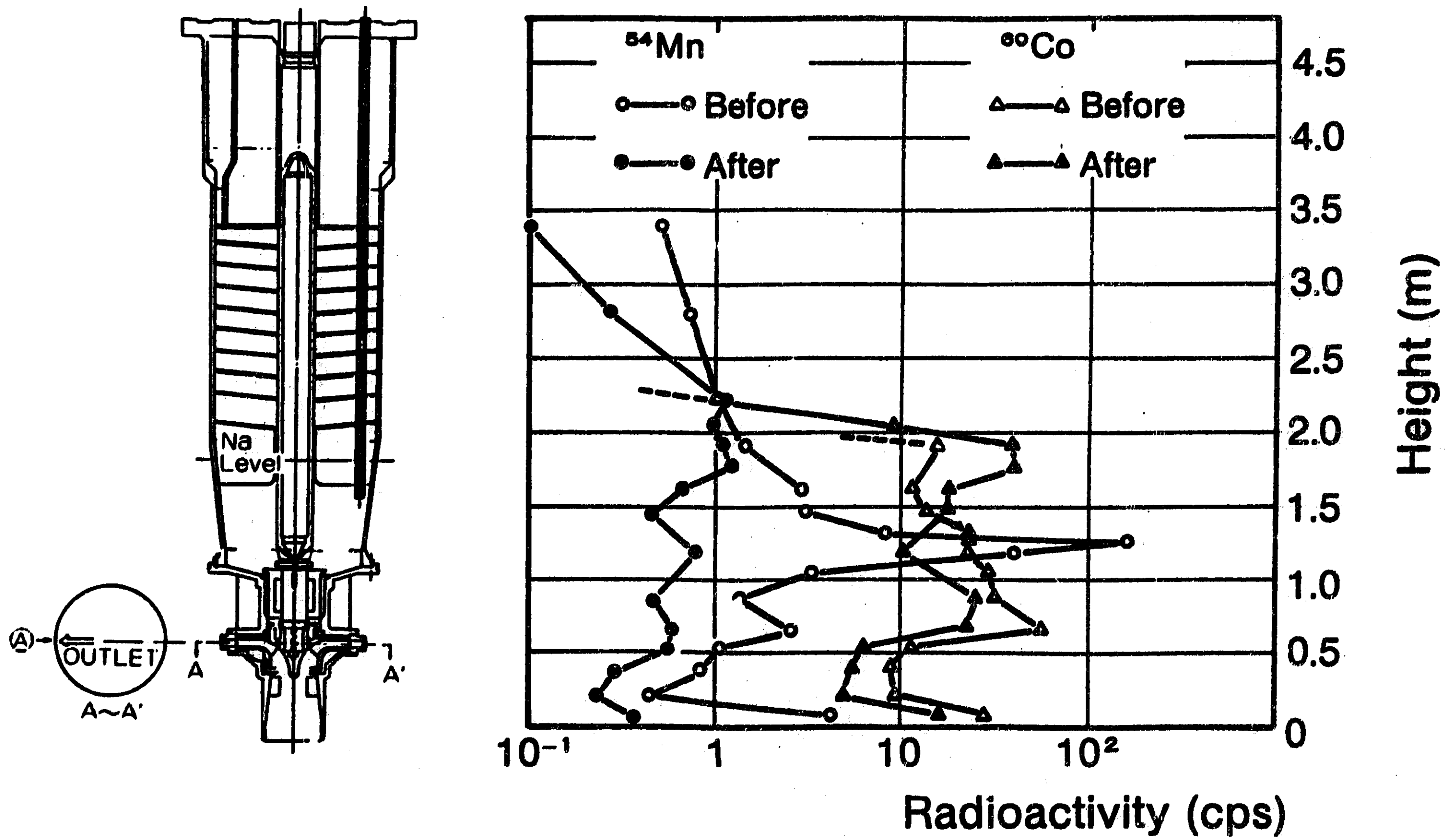
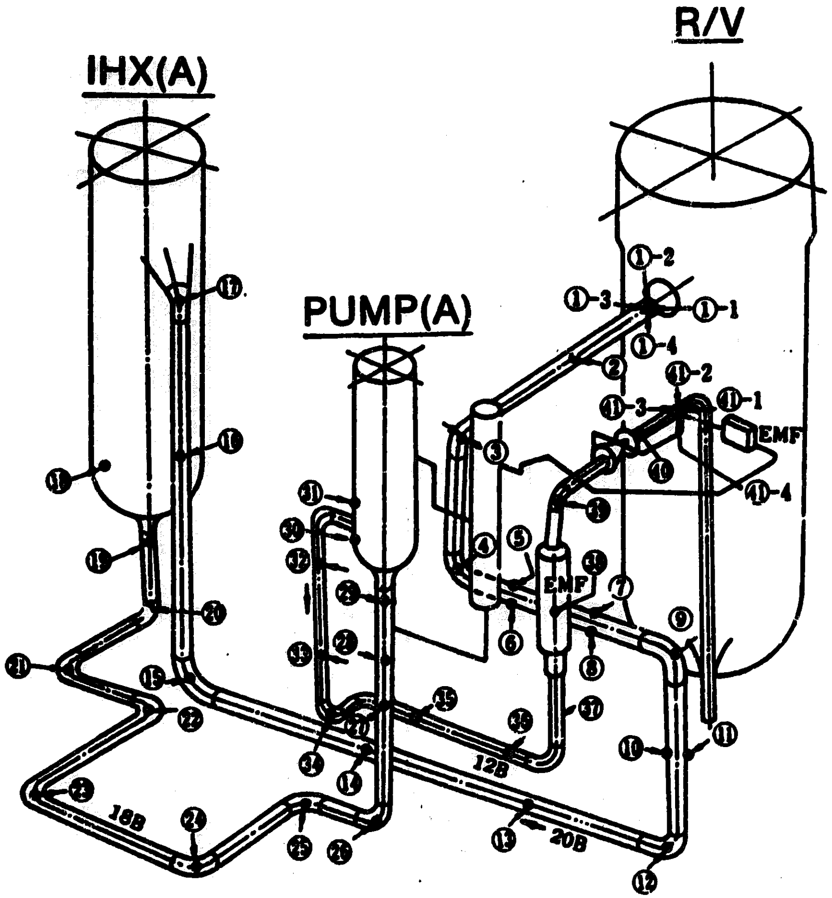
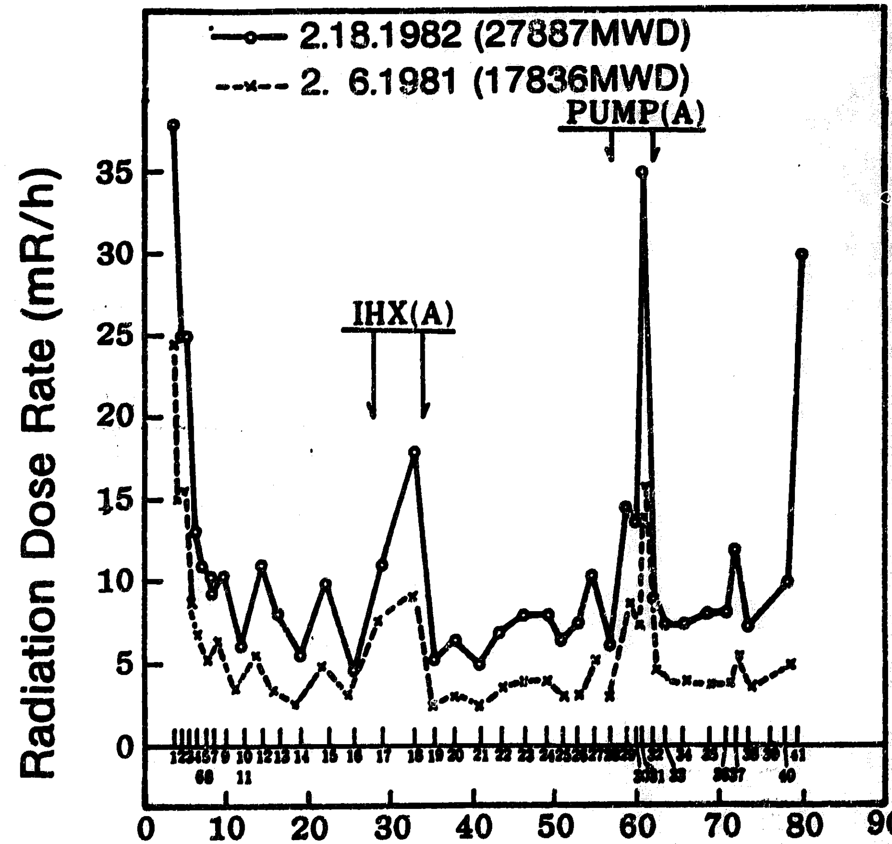


Fig. 2-6 Corrosion Products Distribution Before and After Decentamination



Location of A-Loop



Measuring Point of A-Loop

Fig. 2-7 Radiation Dose Rate at Primary Cooling System

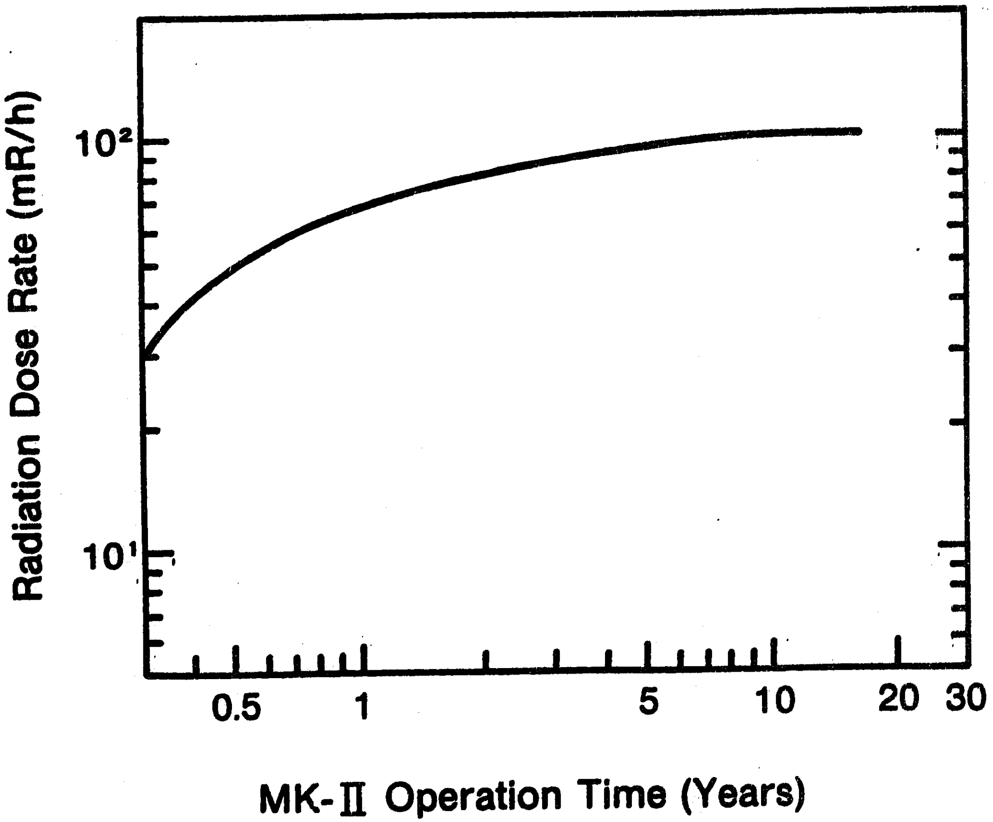


Fig.2-8 Trend of Cold Leg Piping Radiation Pose Rate

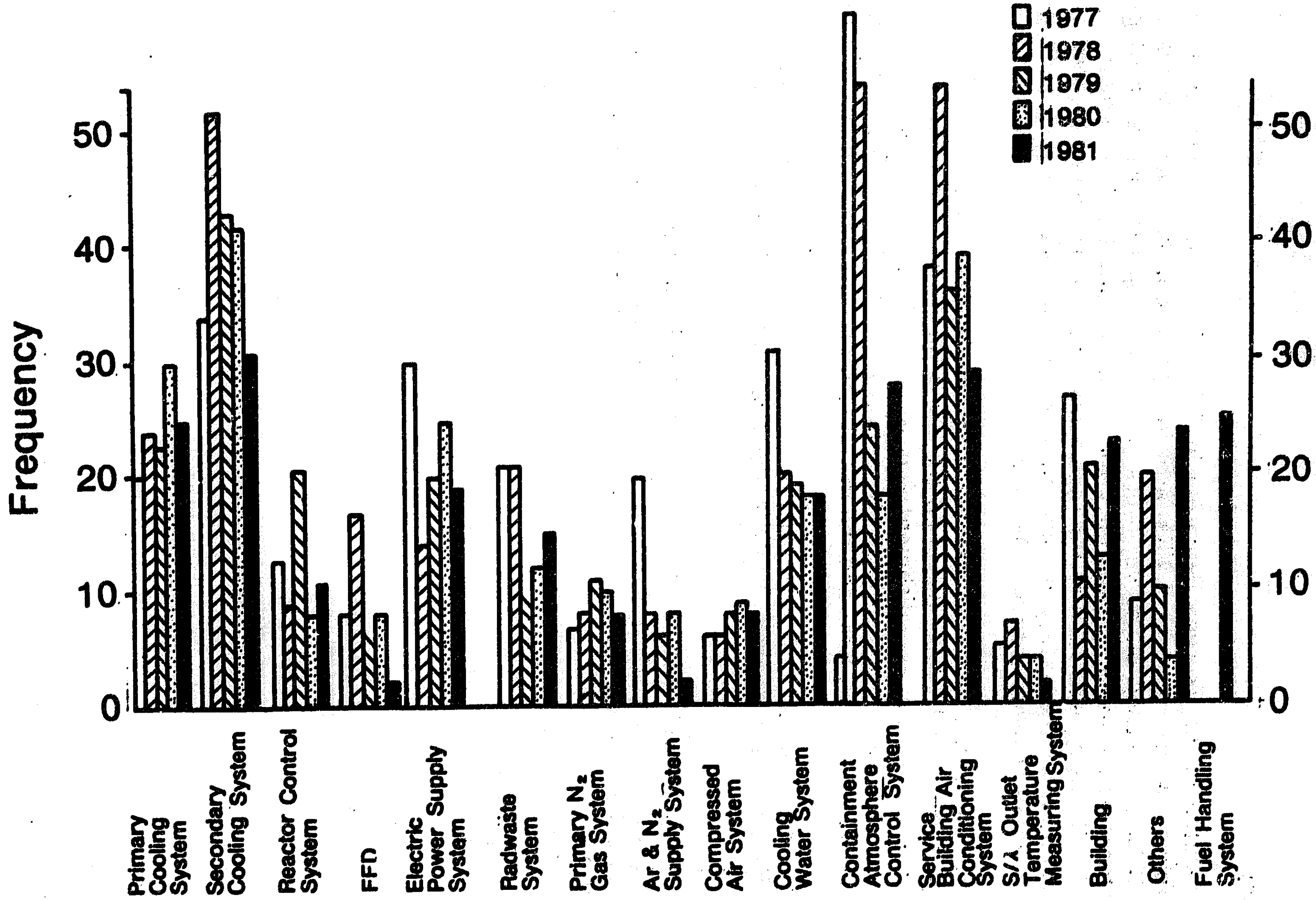


Fig.2-9 Comparison of Incident Frequencies at JOYO

3. Prototype Fast Breeder Reactor "MONJU"

3.1 Summary

A site located in the Tsuruga Peninsula in Fukui Prefecture, approximately 400 km west of Tokyo, where several LWRs are in operation, has been decided for constructing "MONJU". Based on survey work in various aspects such as geological, marine, and meteorological aspects, suitability for "MONJU" site has been studied. And the application for licensing was filed to the regulatory body on December 10, 1980, then MONJU was subjected to the safety evaluation work by the Science and Technology Agency of the Japanese government until December 1981.

The safety evaluation work for "MONJU" by the Nuclear Safety Commission of Japan was completed in April 1983. Based on these examinations, the permission of the Prime Minister for the reactor establishment was issued in May 1983. Since then, the civil work at the site has been intensively conducted by PNC. The components manufacturing contract between PNC and the manufacturers was signed in January 1984.

As a co-ordinator in software of "MONJU" construction work among manufacturers, Fast Breeder Reactor Engineering Co., Ltd. (FBEC) was established on April, 1980.

And, a special department has been set up in the Japan Atomic Power company (JAPC) to co-operate with the PNC for MONJU construction work at the same time. JAPC will act on behalf of the nine Japanese electric utilities and the Electric Power Development Corporation (EPDC).

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 identification of R&D programme in accordance with the role of PNC indicated in "the Long Term Programme of Development and Use of Nuclear Energy" 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 reliability and availability goals and improvement of operability and maintainability.

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.

Some of key specifications on the loop type reactor examined and extracted by PNC and the utilities so far are shown in Figure 4.1. The key specifications such as reactor-type (loop vs. pool), the number of loops and reactor outlet temperature will be selected in accordance with national needs for the DFBR.

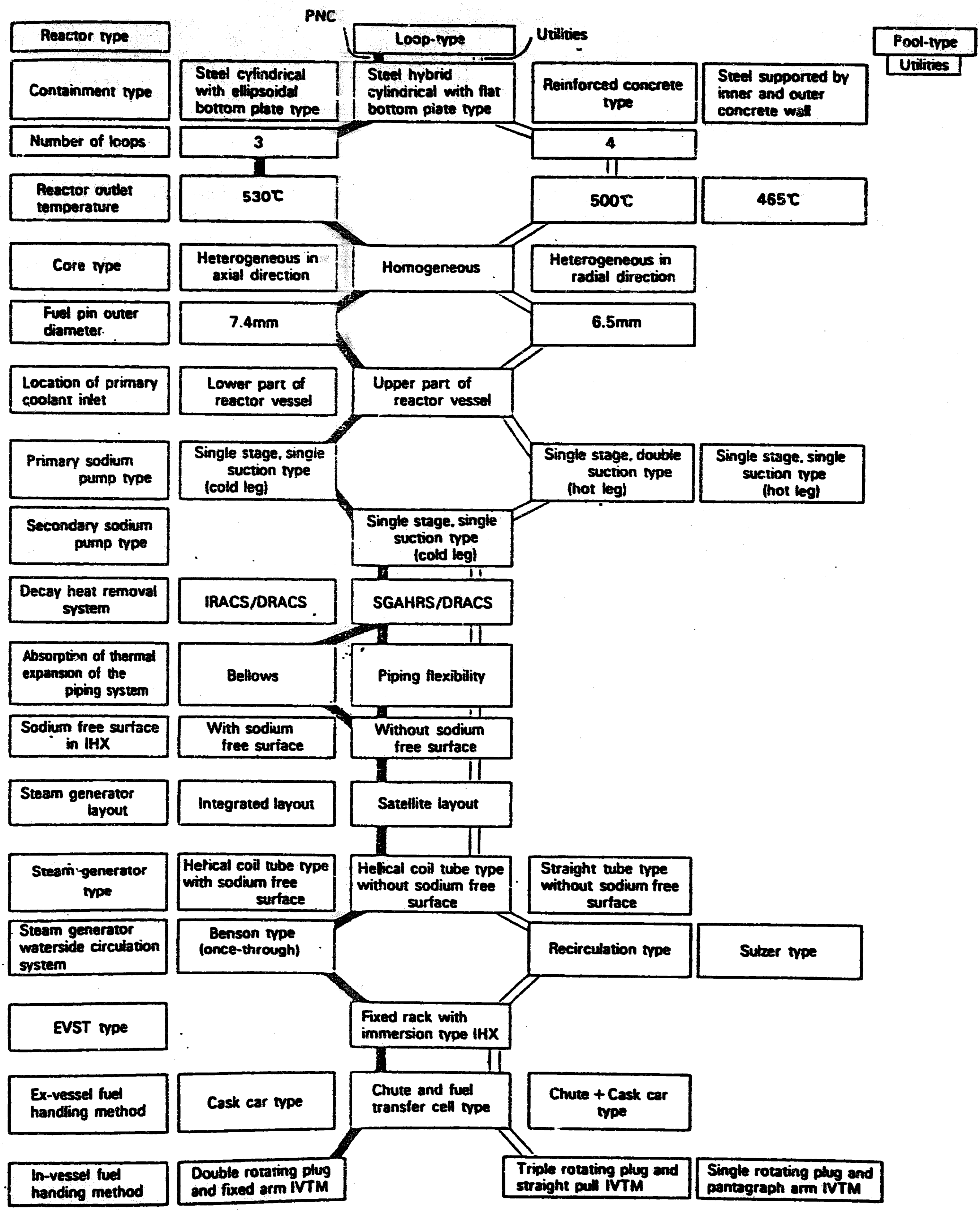


Figure 4.1 Main trade-off study on loop-type plant

4.2 Design Study of DFBR

The utilities have carried out the conceptual design studies under cooperation of all of the private ten electric power companies since 1978. The study consists 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 are to be 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 precursor plants in the phase I and it was reviewed mainly from the stand-point of seismic characteristics in the phase II. The key subsystems of the pool-type reactor have been designed in the phase III in parallel with model tests studied by the Central Research Institute of Electric Power Industries (CRIEPI).

At the end stage of phase III a preliminary design aiming the cost reduction has been made. It is scheduled to carry out the cost down design for three years.

PNC has continued the design study concentrated in the method of initial cost reduction,maintaining safety in these years.

Many potential methods has been comparatively studied such as follows:

1 Reduction of control rods and radial blanket rows, linear heat rate increase with large diameter fuel pins and elimination of handling of the outer row of shielding assembly by fuel handling machine which could make core compact resulting in diameter decrease of reactor vessel and outer shielding concrete of approximately five percent.

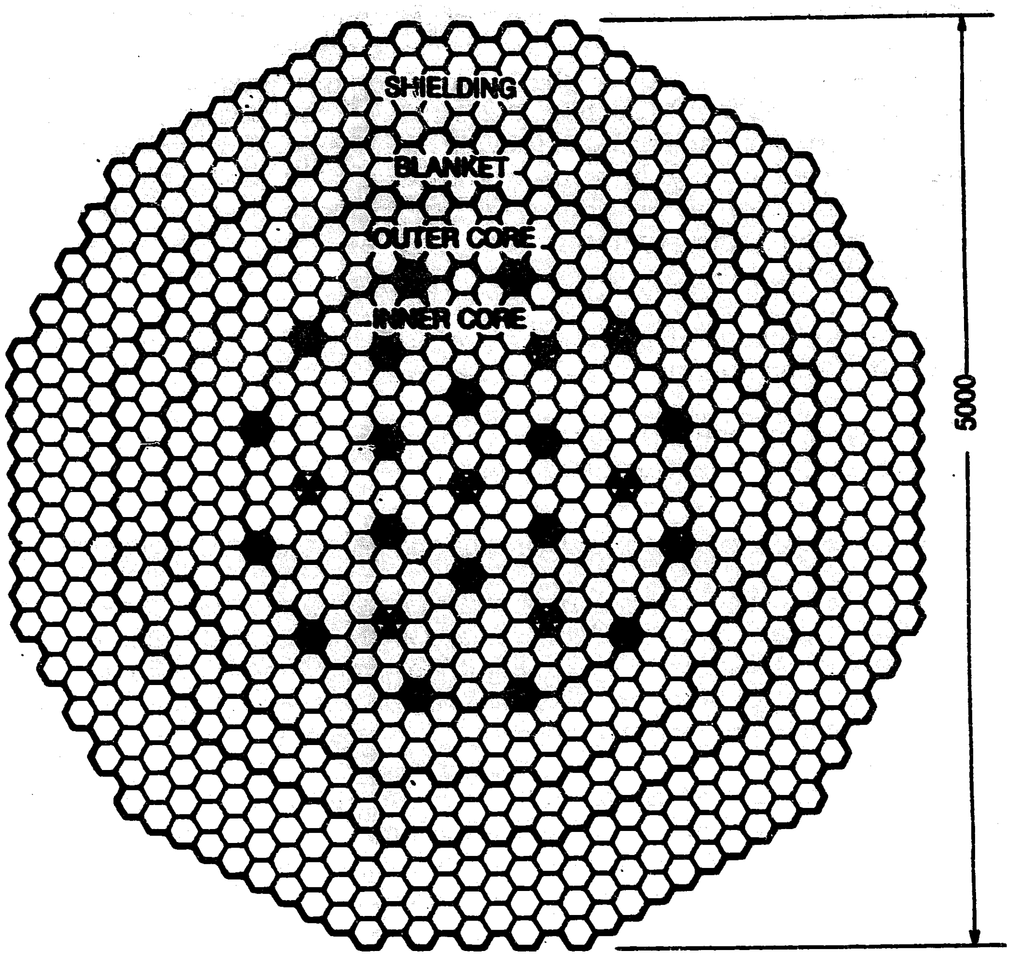
2 Introduction of piping bellows joint into main heat transport system, direct reactor auxiliary cooling system and immersion type cold trap which could make the heat transport system compact resulting in dimension decrease of the reactor and auxiliary buildings of approximately twenty percent.

3 New-type reactor shut down system and direct reactor auxiliary cooling system which could bring more diversity in reactor safety engineering system.

4 Other examples are thermal isolation for reactor vessel wall, in-water storage of spent fuel assemblies, reduction of seismic floor response, inclined chute type fuel charge and discharge system and so on.

Further, a design study including broard parametric study, conceptual trade-off study and optimization study will be continued.

The core arrangement, reactor structure sectional view and heat transport system arrangement are shown in Figure 4.2, 4.3 and 4.4, respectively.



	NUMBER OF ASSEMBLIES
CORE FUEL	INNER FUEL 174 } (TOTAL) OUTER FUEL 162 } 336
RADIAL BLANKET-FUEL	150
CONTROL ROD	PRIMARY CONTROL ROD 18 } (TOTAL) BACK-UP CONTROL 7 } 25

(NOTE : CORE FUEL ASSEMBLY WITH 271 FUEL ELEMENTS)

Figure 4.2 Core Arrangement

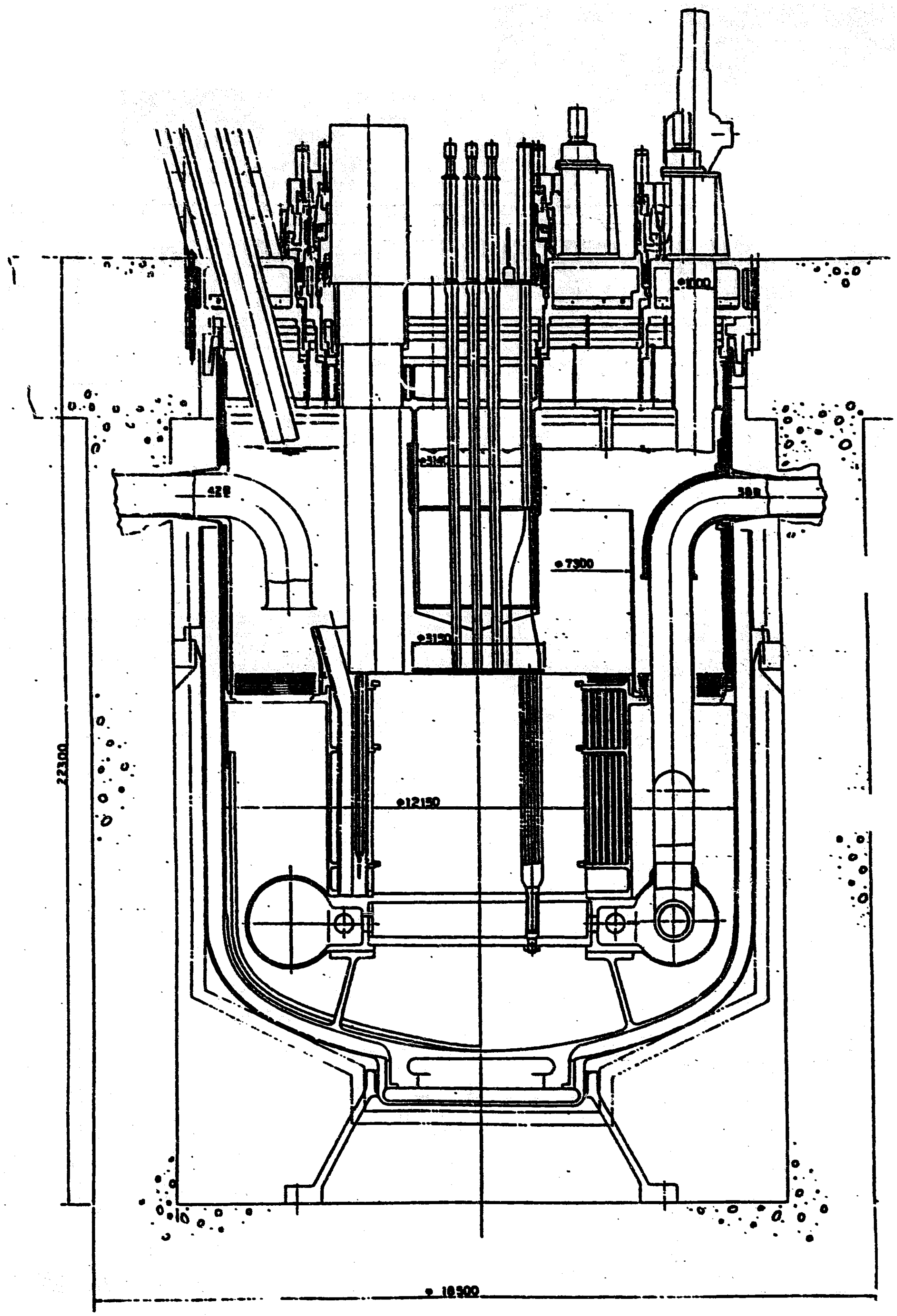


Figure 4.3 Sectional View of reactor structure

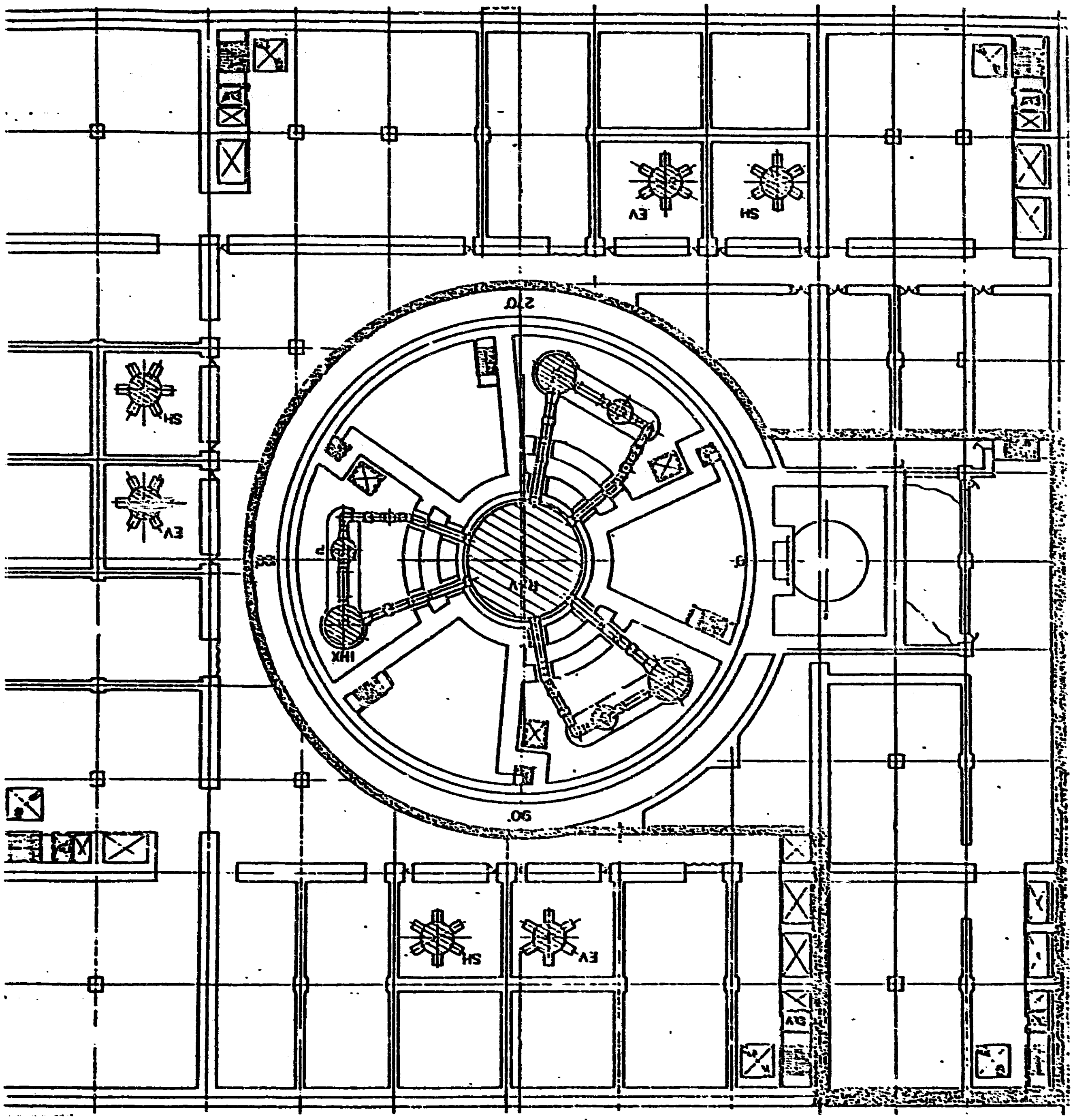


Figure 4.4 Heat Transport System Arrangement

5. Physics

5.1 JUPITER Phase II Program

The DOE/PNC Joint Physics Large Heterogeneous Core Critical Experiments Program, called JUPITER Phase II Program, started as ZPPR-13 experiments on May, 1982, using the ZPPR facility of Argonne National Laboratory.

The whole experiments of more than 20 months will be completed by March, 1984. The analysis of experiments is under way in Japan, using the JENDL-2 library. The first assembly of the JUPITER-II Program, ZPPR-13A, has almost been analyzed. The analytical results are to be compared with the U.S. results at the U.S./Japan JUPITER Analysis Meeting, which will be held on September 1984, at ANL-W.

5.2 Experiments at FCA

According to the modification of the Fast Experimental Reactor "JOYO" to MK-II, a series of mockup experiments were carried out on FCA Assembly X from April 1982 to February 1983. The major modifications considered on the mockup experiments were: (1) increase of plutonium content in fuel material and (2) replacement of the uranium blanket by the stainless-steel reflector.

Experimental studies on fundamental physics aspects of conventional large fast reactor cores were done on FCA Assembly XI. The first version of the assembly (Assembly XI-1) went critical at the end of February 1983. The assembly has a central test region of 60cm x 90cm height simulating the core composition of a homogeneous fast reactor and a driver region mainly fuelled with ^{235}U.

5.3 Burnup Analysis of JOYO MK-I Core

The burnup characteristics of JOYO MK-I core were calculated based on the actual operational data of the reactor, and were compared with the post-irradiation data. In case of a low burnup of 20,000 MWD/T, the C/E values are summarized as follows.

Effective multiplication factor	:	0.993
Control rod worth	:	1.024
Burnup reactivity loss	:	$0.96 \sim 1.02$
Burnup	:	$0.93 \sim 0.98$

5.4 Group Constant Set JFS-3-J2

The group constants for ^{181}Ta, ^{151}Eu, ^{153}Eu, ^{237}Np, many fission products and the lumped nuclides of ^{235}U, ^{238}U and ^{239}Pu were generated to be accommodated in the JFS-3-J2 set.

5.5 Development of Calculational Method

An effective homogenization method of control rods, which preserves the integrated reaction rates in a heterogeneous channel by iteratively changing the cross sections used in a homogeneous super-cell calculation, has been extended to treat off-center control rod channels in FBR. An albedo at the super-cell surface was combined with collision probabilities to treat the neutron leakage. The method has been applied to the central rod worth calculation in a typical demonstration LMFBR, and to the 1-D off-center rod worth calculation.

5.6 Reactivity Analysis of Pin and Plate Cores

The reactivities of plate and pin ZEBRA-CADENZA cores were analyzed at Osaka University and JAERI, and the results were presented at the NEACRP specialist meeting held on 21-23 June, 1983 at Winfrith.

5.7 Effect of Cell Model on Heterogeneous Core Parameter Calculation

Effect of cell model on predicting the nuclear characteristics of a heterogeneous critical assembly has been investigated by analysing the physics experiments made at ZPPR-7A. The principal effects of cell model for ZPPR-7A are 0.43% Δ k/kk' for criticality, 5% for ^{238}U (n,f) reaction rate distribution and 15% for Na-void reactivity effect.

5.8 Double Heterogeneity Effect of Fuel Pin and Subassembly in a Fast Power Reactor[5)]

The double heterogeneity effect due to the fuel pin and the subassembly is estimated for neutronics parameters of a prototype fast power reactor. The effect is found to be 0.5%Δk for k_{eff}, the positive sodium-void worth is reduced by 26%, and the negative Doppler reactivity increases by 7% for a prototype fast breeder reactor.

5.9 Analysis of Heterogeneity Effect in FCA-VI-2 by Monte Carlo Code VIM

The analysis of heterogeneity effect for pin and plate cell in FCA-VI-2 was performed by using the VIM code. The results calculated with VIM were in a good agreement with those calculated with the SRAC and/or SLAROM code based on the collision probability method.

5.10 Shielding Analyses of "JOYO"

After the first criticality of "JOYO" with the MK-II core, several shielding measurements were performed to obtain the shielding characteristics data of the plant. The measured data are now being analyzed especially for the effects of the fuel assemblies in the in-vessel storage tank.

Radiation streaming through the gas-plenum of the fuel pins have been calculated by the albedo-Monte Carlo code MORSE-ALB, and the neutron flux at the upper core structure was investigated.

5.11 Shielding Analyses of FFTF

FFTF/JOYO Shielding Data Exchange Meeting was held in Tokyo on May 12, 13, 1983, between US DOE and PNC. Following the meeting, we are now analyzing the shielding characteristics of FFTF, and the calculations and measurements will be compared to predict the reliability of shielding calculations. Our calculational results have been partly compared with those of ORNL, and the effects of difference of calculational conditions were examined.

5.12 Improvement of Albedo Monte Carlo Code "MORSE-ALB"

The albedo Monte Carlo Code "MORSE-ALB" was improved to treat the deep penetration of radiation in shielding configuration of large scale geometry. The arbitrary coupling surfaces with the discrete ordinate calculations were made to be applicable to forward and adjoint calculations.

The improved code was applied to teh analyses of the shielding experiment on the fast neutron source reactor "YAYOI" of University of Tokyo and the streaming measurement on the primary coolant pipe of "JOYO".

5.13 Preparations of Group Cross-Section Sets from JENDL File

Cross-section sets prepared from ENDF/B-IV file for shielding calculation were revised by using JENDL file. SUPERTOG code in RADHEAT system and MINX code were checked for JENDL file and infinite dilution cross sections and resonance self-shielding factors were calculated by those codes. By the benchmark problems (such as NEA shielding benchmark), we are now checking the data and estimating the effects of the difference of nuclear data files.

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 performed with an integral flow model of 1/2.14 geometric scale using water as a working fluid. The model was designed to simulate the hydraulic characteristics. Among these tests, measurements were made under isothermal and steady flow conditions; (1) pressure distribution in the each plenum, (2) pressure loss characteristics of the plenum inlet flow holes, and (3) flow rate distribution through the reactor core, blanket, control rod and neutron shield regions. Since the design of the reactor core support structure was changed due to the result of the Monju HCDA analysis, the experiments have been started to obtain data focussing on new features of the design. The construction of test facility has been completed. Currently, the preliminaly analysis and experiments are being carried out.

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 convection restraining plates were employed. A performance test was carried out from May to October 1983 on the above-mentioned model with the convection restraining plates. The data of this test are now being analized. In connection with this test, measurement of evaporation rate of sodium mist in the temperature range from 150°C to 300°C have been started using a small sodium pot. This basic data will be used for the calculation of sodium mist deposition rate on the relatively low temperature walls of the annulus.

6.3 Primary Pump

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

As the final stage of the pump test series, some special tests are planned to demonstrate its sound functions under emergency conditions: An operation test with low sodium level in the pump casing will be soon started, and some additional tests will follow under the condition of failure of seal gas feed.

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. Further, on the tube-to-tube sheet welding, the filet/but welding was selected through some trial fabrications. And 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 has been tested with full scale mock-up under simulated sodium conditions. These have fine control, coarse control and back-up functions respectively. In-sodium test on the back-up control rod drive mechanism is finished in December 1983. Bellows for shaft seals has been tested to establish the design basis. As dynamic behavior affects on the life of bellows, movement of bellows in case of scram has been studied in detail in both experimental and analytical method. Besides, fatigue life data have been stored through in-sodium and in-argon gas test on bellows.

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.

The ex-vessel fuel storage tank (EVST) is designed to hold the fuel assemblies by a rotating rack in sodium. Testing of the bearing and the shaft seal in air and in sodium were completed in 1982.

6.7 In-Service Inspection Equipments

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.

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 contructed 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 1983. Research and development on the steam generator characteristics, plant controlability and thermal transient test were almost completed.

The spacial characteristics of the evaporator were evaluated taking into account of the plugged tube effect. Evaluation of the effect of the water side fauling 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.

Water and hydrogen are injected into the evaporator or the sodium inlet piping and hydrogen behavior in the loop is being studied.

Acoustic leak detection system is also being studied at the 50 MW SG test facility.

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. Flow diagram and test items of ACS is shown in Fig. 7-1 and Fig. 7-2, respectively. Verification efforts of COPD-50 & NATURAL-SG Code (Plant dynamics analysis code & natural circulation analysis code of 50 MW SG Test Facility) are now made through analysis of ACS test results.

7.2 Sodium Water Reaction Study

7.2.1 Leak Hole Enlargement and Leak Propagation Study

40 self-enlargement tests on micro crack defect of 2-1/4Cr-1Mo and stainless steels were conducted in SWAT-4 under various operating conditions of Monju from 1981 to 1983.

Six tests were carried out using SWAT-1 from 1981 to 1983 to investigate wastage phenomena under various conditions of a intermediate leak to support the selection of a Monju DBL, which were a wastage test for weld joint of heat transfer tubes, a steam leak test in the cover gas region, etc. They proved to be inessential for the DBL for Monju steam generators.

Two failure propagation tests were performed by use of SWAT-3 in 1981/82 to clarify the potential of tube burst due to overheating. A plenty of information was obtained on the conditions and the mechanism of the overheating. In consideration of all the failure propagation tests, the DBL for the Monju steam generators was determined.

7.2.2 Computer Code Development

The SWAC10 code has been developed for evaluating effectiveness of the leak detection system. Data obtained by the 50 MW SG test facility have been compared with code results.

The LEAP code has been developed for estimating the extent of leak propagation. This code aims at evaluating the design basis leak (DBL), and has been validated by the SWAT-3 data.

The SWACS code has been developed for computing pressure and fluid-flow dynamics of the large-scale sodium-water reaction accidents. Data obtained by the SWAT-3 facility were used to validate SWACS. An improved version of the long-term module is being developed.

7.2.3 Leak Detector Development

Tests of nitrogen gas injections into water were carried out for studying a usefulness of the acoustic detectors using a half scale model of the Monju evaporator.

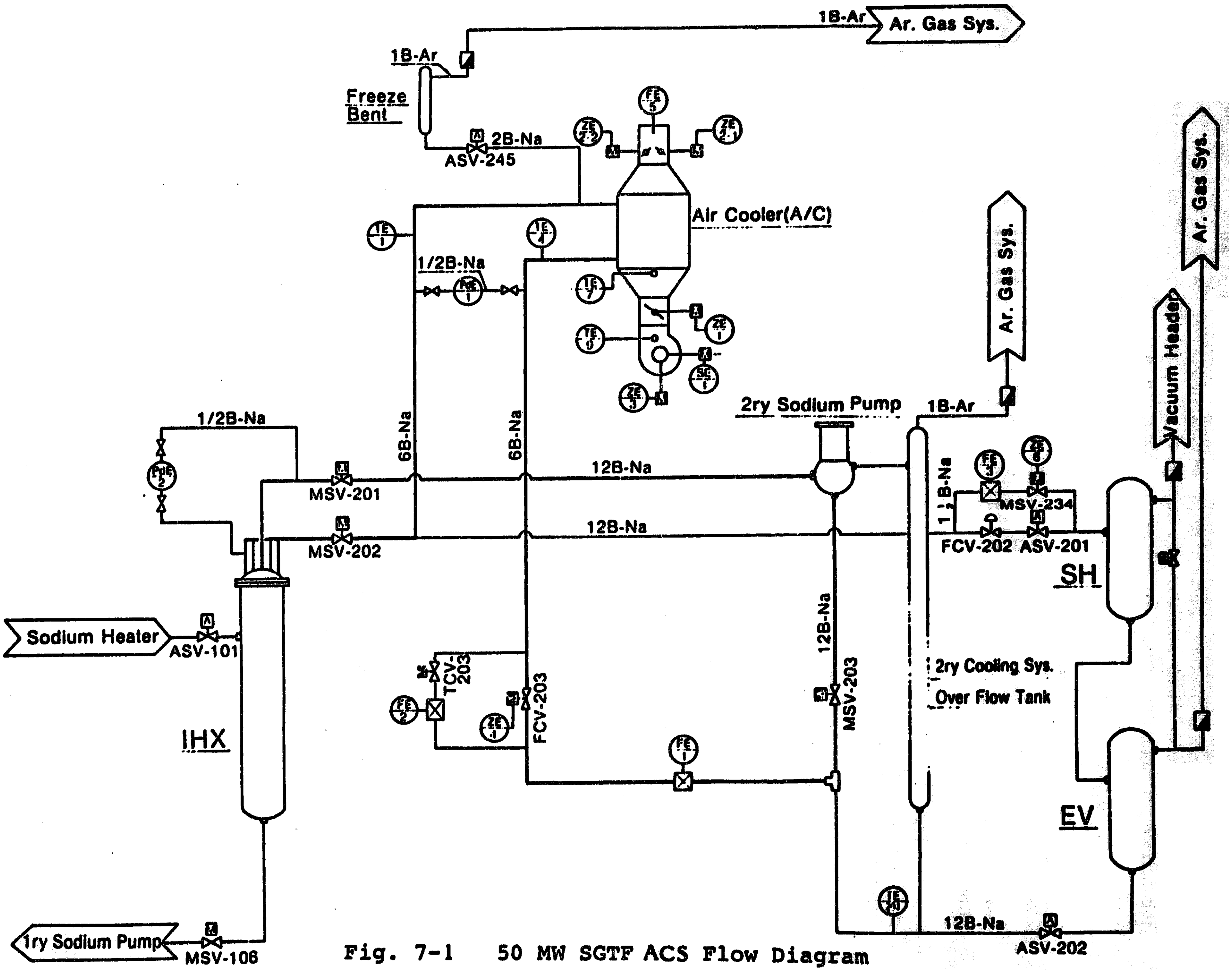


Fig. 7-1 50 MW SGTF ACS Flow Diagram

I. Component Unit Charc., A/C Static Test

- o ACS Press. Loss. Heat Loss, Valve Perf.
  Chara Pump Coast Down Char.
- o A/C Static Char. (ACS Operating & Waiting)

II. Dynamic Char., Controlability Test

- o PI Servery, Open Loop Char.
- o ACS Outlet Sodium Temp. Cont.
  ACS Sodium flow Cont.
- o Natural circulation Char.

III. Accident simulated Test

- o Plant Trip Simulated Test
- o Loss of all Power Supply Sim. Test
- o ACS Related Compoment Failure Simulated Test

Fig. 7-2 ACS Test Items

8. Sodium Technology

8.1 Material Tests in Sodium

Creep-rupture tests under internal pressure are carried out for modified SUS316 fuel cladding tubes.

Mechanical tensile tests after exposure to high temperature sodium, as well as corrosion test in sodium are continued for modified SUS316 and advanced alloy for future fuel cladding tube.

Caustic corrosion tests of SUS304 and SUS316 were carried out in the primary environment of nitrogen including oxygen of 0 - 2% and H_2O of 0 - 14,800 ppm. Also, caustic corrosion tests of SUS304 are continued in the secondary environment of air at the temperatures of 325°C and 505°C. Various materials are tested in off-normal condition including high C or NaOH. Long term creep test is carried out SUS321 for super heater material.

Stress corrosion cracking tests of SUS304, SUS316 and SUS321 are continued in the wetted steam including oxygen and chlorine.

Self-welding, friction and corrosion tests are continued on Inconel 718, tribaloy 700, Colmonoy No.5, Chronium carbide LC-1H and FR452. The tests are carried out under the variable condition as the parameter test.

8.2 Flow and Heat Transfer

In order to evaluate the deposition of sodium vapor in the annular gap between reactor vessel and shield plug, it is required to obtain the experimental data on sodium mist content in cover gas space at relatively low temperature condition. Modification of apparatus has been started to conduct the experiments.

Tests on flow distribution in the Monju reactor vessel have recently started by using water test model. Experimental data will be collected late in this FY 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 started by using water flow model. As a result of preliminary test, minor modification will be required on the design of the overflow line.

Heat transfer tests at low Peclet numbers have been continued by using heater pin bundle at extremely low flow velocity of sodium. As the result of the test, it is expected to obtain the accurate data on Nusselt numbers at low flow conditions.

8.3 Behavior of Radionuclides in Sodium

The objective of this area is to study the mass transfer of radioactive corrosion product in sodium for the purpose of the radiation dose evaluation in the primary coolant system and the development of the method to decrease radioactive corrosion product.

The 5th radioactive mass transfer exposure test is being progressed in the Activated Material Test Loop-II in order to develop CP trap. Major Test conditions of this test were as follows:

Cold trap temperature	:	140 °C
Test duration	:	5,000 hrs
Sodium temperature	:	400 °C (hot leg)
	:	400 °C (cold leg)

A computer code for corrosion product behaviour and dose evaluation in the reactor plant, "JOYO" has been developed.

Trapping of corrosion products using metals like Nickel is also investigated.

8.4 Sodium Chemistry and Sodium Purification

Performance test of an experimental model of on-line gas chromotograph for JOYO cover gas monitor was done in order to determine the measuring condition and to confirm the durability.

A prototype on-line gas chromatograph was fabricated in 1979 and performance test was made in 1981.

Development of the other on-line impurity indicators such as plugging indicator, hydrogen, carbon, and oxygen meter, continue at O-arai Engineering Center.

Vanadium-Wire equilibration method for determination of oxygen in sodium was developed and completed. Steel foil equilibration method for carbon was studied preliminarily and is being developed.

A typical full size cold trap for FBR primary cooling system has been tested on an improved model based on data obtained from past experiences. The test includes quickened life test and regeneration test as well as ordinary characteristics test.

Cold traps of the secondary cooling system are estimated to be exchanged every several years because the traps will be plugged with the hydrogen diffused through the heat transfer tubes of the steam generator. A preliminary study on regeneration of the plugged cold trap was conducted by evacuating gas phase over the sodium surface after dissolving the trapped hydrogen into sodium by raising the temperature.

The results show that MONJU secondary cold trap is required to be heated to about 360°C in order to regenerate it within a month. A large scale test on the regeneration system of the secondary cold trap is planned to be performed in 1984 through 1987.

A feasibility study on regeneration of MONJU primary cold trap was completed.

8.5 Sodium Removal and Decontamination

Since 1971, sodium removal tests have been carried out for the various sodium components at the O-arai Engineering Center. Sodium removal experiences were obtainted in the past two or three years about the following components.

a. Monju fuel handling machine (mockup)

b. Monju intermediate heat exchanger (mockup)

c. Monju primary coolant circulating pump (mockup)

d. Joyo primary coolant circulating pump

e. Joyo various fuel subassemblies

f. Grapples of Joyo Fuel Handling Machine

The mock-up of JOYO reactor vessel with its internals rotating plug will be dismantled within this year and new experiences will be obtained.

A feasibility study was made on removing sodium from crevices.

Since 1976, study of radioactivity decontamination of primary system components are being made in laboratories of manufactures.

The program consists of the following from parts;

a. Researches in the chemical decontamination method

b. Researches in the physical decontamination method

c. Studies on treatment techniques of the decontamination waste water

d. Miscellaneous studies

One of Joyo primary mechanical pumps was dismantled and decontaminated in 1982 and the other pump will be dismantled in 1984.

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 subasseblies 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. Durability 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 MONJU.

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 of MONJU is being planned. In relation to this system, aerosol generation and diffusion are also investigated.

10. Fuel and Materials

10.1 Fuel Fabrication

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

The detailed design of the "MONJU" fuel assembly is almost fixed, and the construction of the PNC Plutonium Fuel Production Facility for "MONJU" fuel is in progress. Plutonium handling technology gained through the fabrication of "JOYO" core fuel are now being applied to the new technology development. The "MONJU" fuel fabrication plant utilizing as much remote technologies as possible is being designed in detail, and some of remote handling components are being developed.

10.2 Fuel Pins

Inpile fuel instrumentation techniques such as inpile measurements for fuel center temperature and fission gas pressure are being developed.

Metalic coated cladding tubes, developed to reduce the fuel cladding chemical interaction, is fabricated and planned in irradiation in "JOYO" reactor.

Out of pile tests to clarify the basic mechanism of CCCT (Clad Component Chemical Transport) and FCCT were conductes.

10.3 Cladding Tubes

Optimized type 316 stainless steel, controlling the concentration of C, Si, Mn, P, Ni, Cr, Mo, N, B, Ti, Nb and solution temperature in order to get better creep strength and swelling resistance, has been developed for "MONJU" and "JOYO" MK-II fuel cladding tubes. These tubes are being irradiated in Phenix and in "JOYO" MK-II where the interim examination showed the good results.

For back up materials to type 316 stainless steel, 6 modified austenitic stainless steels were chosen. Further-more, development of precipitation-strenghened high Ni alloys and ferritic steels has been initiated.

10.4 Subassembly

Sodium endurance test of mock-up fuel subassembly fabricated for final testing was carried out and the inspection of assembly parts after disassembling was finished.

Wrapper tube with integrated spacer-pad has been developed for "MONJU" subassembly and pad coating technique is under development.

Tag gas capsule has been developed with PNC design.

10.5 Fuel irradiations

MK-II irradiations started. Several subassemblies are being modified and irradiated to verify the design criteria for "MONJU" fuel claddings. Irradiations of fuel pins and subassemblies are being performed in some foreign reactors, such as, phenix, FBR-II and FFTF in order to assure the performance of "MONJU" fuels. Safety irradiation experiments in Siloe reactor, using the pre-irradiated fuel pins in Rapsodie, started and will be completed in 1984.

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
- Relaxation damage evaluation
- Buckling safety factors
- Design methods of tubesheet-shell structures
- Bolts etc.

c. Improvement of design post-processor

The design post-processor POST-DS based on the MONJU structural design quide 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 are completed and evaluation is under way.

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

c) Thermal and creep ratcheting tests of pipes and elbows
Creep rachtting tests on 2¼ Cr-1Mo pipes and elboes 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
Test apparatus is under construction for the first test in 1985.

h) Thermal transient tests of SG tubesheet Model
Testing facility is completed and the first test is scheduled in 1984.

i) Thermal transient test by TTS
The thermal transient test rig for structures (TTS) is under construction and the first test on vessel is scheduled in 1984.

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 completed and
Phase 2 test scheduled.

c. Thermal striping tests for UCS
Test apparatus is made in 1984.

11.3 Structural Material Test

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

11.3.1 In-air Structural Material Test

In table 11-1 are shown the R & D schedule of FBR structural material test. Now is under way the R & D for the design and licencing for MONJU. Furthermore were begun from last year, 1982, the preliminary investigations on the structural materials for demonstration reactor.

According to increasing needs for structural materials data on "MONJU" components, the installation of fatigue and creep testing machines has progressed since 1977 in FBR component fabricators. And new test laboratory for structural material strenghts, which was facilitated ten high temperature low cycle fatigue testing machines and eighty creep testing machines, was constructed in O-arai Engineering Center in Aug., 1980.

11.3.2 Structural Material Test in Sodium

The following research works are in progress:

a. Corrosion and mass transfer
Mass transfer tests on SUS304, SUS316, SUS321 and 2¼ Cr-1Mo steel

b. Carbon transfer
Carbon transfer test on bimetallic systems simulating the MONJU secondary systems

c. Mechanical strength tests in sodium
Determine the effect of the sodium environment on the mechanical properties of structural materials, such as tensile strength, creep, low-cycle fatigue and creep-fatigue.

The new facility was constructed on O-arai Engineering Center in 1979.

11.3.3 Structural Material Test in Irradiation Environment

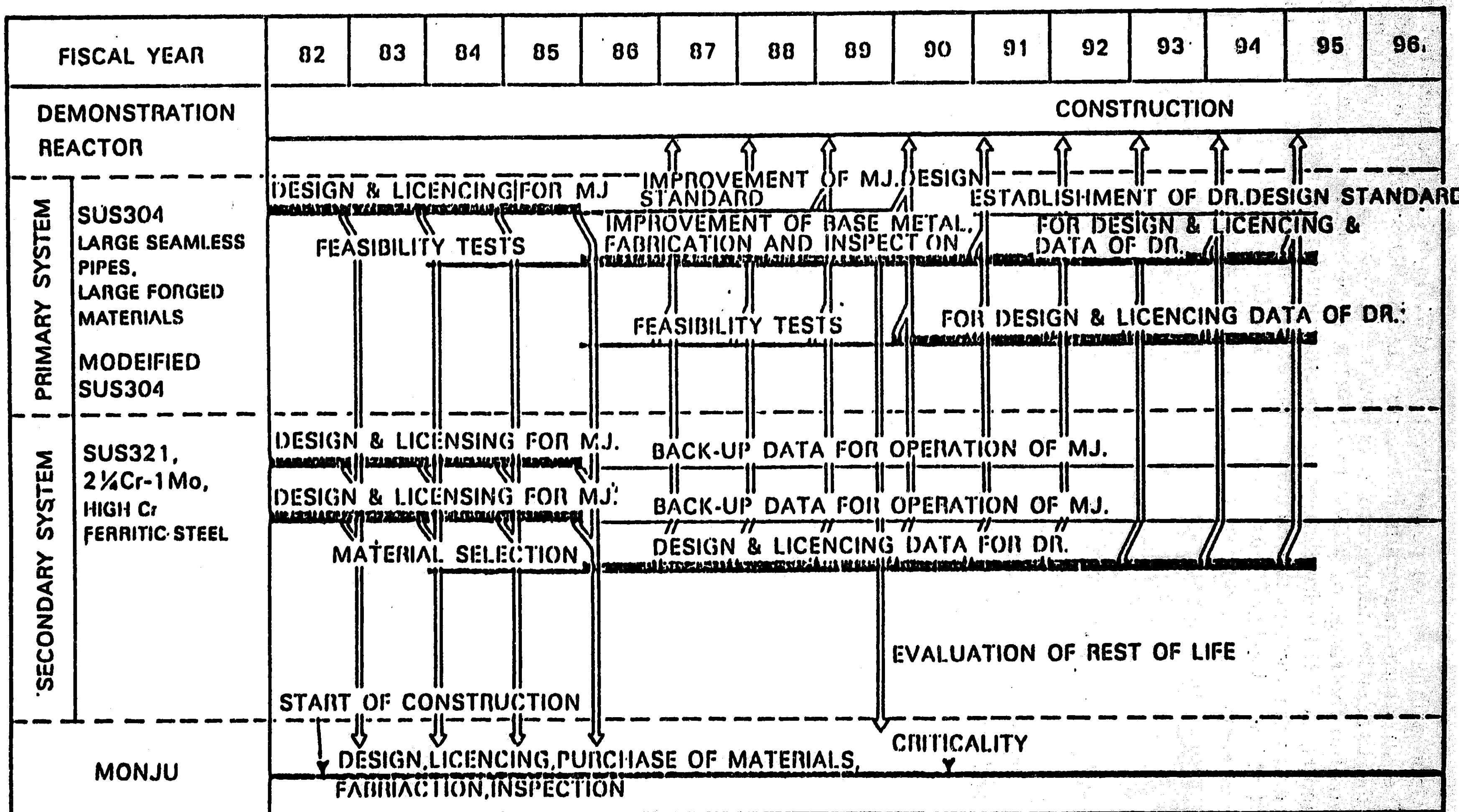
The tests on domestic 304 stainless steel have been conducted to ensure the safety of reactor vessel and internal components under the irradiated condition.

Upon attainment of power level 100 MW, Joyo is to be operated for "irradiation bed". The structural materials irradiation rig (SMIR) which is designed to be able to control the irradiation temperature, is now being fabricated. With SMIR, the temperature control in the range of 400°C to 610°C is to be enabled. The upper core structure irradiation rig (UPR) for irradiation of structure irradiation rig under construction, which is designed to get the irradiation conditions at high temperature (530°C) in low neutron fluence ($15 \times 10^{18} n/cm^2$ - ϕth, 5×10^{16} $1 \times 10^{18} n/cm^2$ - E 0.1 Mev.) The post-irradiation tests are conducted in Material Monitoring Facility at O-arai Engineering Center.

11.3.4 Data Banking and Retrieval System

Material tests are conducted following the standard manual "FBR Metallic Materials Test Manual" including various data sheets and standard graphs. A data banking and retrieval system which handles the resulted data are under development.

Table 11-1 R & D Schedule of FBR Structural Material Tests



MJ.:MONJU , DR.:DEMONSTRATION REACTOR

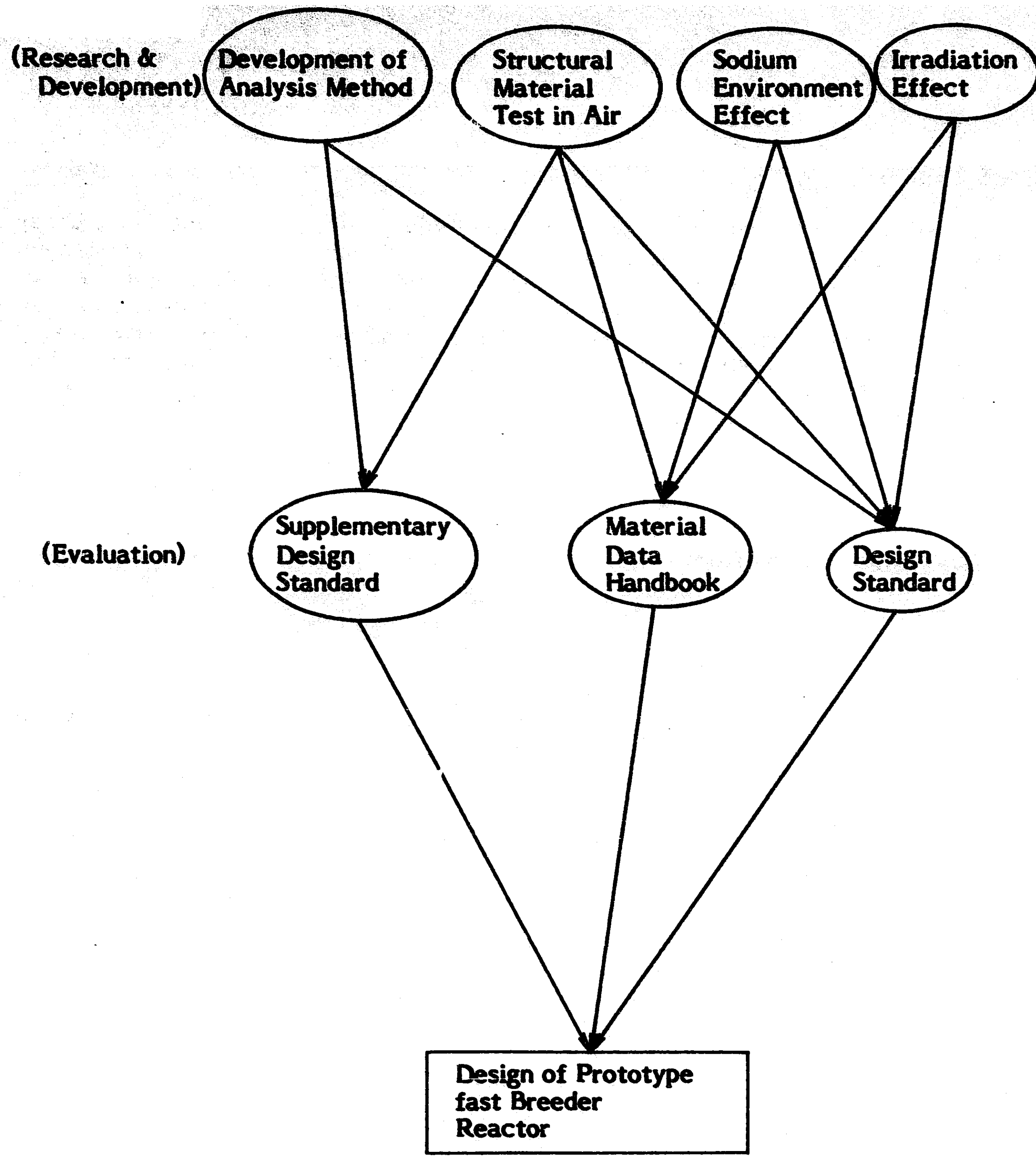


Fig. 11-1 Research and Development Items of Structural Materials Necessary to the Design of Fast Breeder Reactor

12. Safety

12.1 Reactor Safety

12.1.1 Sodium Boiling

Summary work of the past two series of sodium boiling experiments at decay power levels, i.e. 37F and 37G experiments using 37-pin bundles, have been performed. A next series boiling experiment 37H is planned to be started at the earlist by the end of 1984. Teh 37H test section differs from the 37F/37G test sections in simulating the full length of MONJU fuel subassembly with a chopped-cosine heating profile. The Mixing Test Facility is under modification to have a capability of conducting the boiling experiment with the 37H test section. The appended new part of the facility is called by the name DHB (Decay Heat Boiling).

Based on the first series experiment of large bubble behavior in water pool, a new test plan of the second series has been designated. Emphasis will be directed to examining the bubble expansion process. A visualization method using water as a working fluid will be taken again.

12.1.2 Fuel Failure Propagation

The local blockage experiments using a 91-pin bundle which was blocked a half of whole flow area by a stainless steel plate were conducted till the end of 1982. The analyses of the test results have been performed based on the idea introduced thus far.

Also, accoustic noise and bundle-exit-flow/temperature meter signals were analized for anomaly detection study.

The temperature increases in the recirculating flow region behind the blockage were evaluated by empilical correlations. Taking into account the effect of FP gas release possible to occur under high temperature conditions, a local fault event scenario which is consievable as best from obtained information was drawn. This project will be finished summarizing these results.

12.1.3 Molten Core Material Interactions

Six out-of-pile tests were performed to investigate the molten fuel behaviors and FSI phenomena during TOP and LOF-d-TOP accidents since February 1983. In the tests, the molten fuel behaviors in flowing sodium were observed by dual X-ray cinematographies. The data analysis of the tests is under way.

The preliminary results are as follows. The maximum height of pressure pulses was 5.3 MPa. The conversion ratio from thermal to mechanical energy evaluated on the basis of the conservative assumptions was less than 0.2 %. These results indicate that the FSI phenomena were mild. It was observed that the most of the released fuel was dispersed due to the pressure pulses, and effective channel blockage was not observed in the tests. The particle size distributions of fuel debris were measured.

12.1.4 Transient Overpower Tests (CABRI)

PNC has participated in the joint CABRI project as a junior partner since 1975, stationing delegates at Cadarache, France.

Useful information has been accumulated on the behavior of fresh and irradiated fast reactor fuel pins under rapid power transient conditions.

Code verifications for the PAPAS and SAS3D codes have been done through the experimental analyses (pre-and post-test calculations).

12.1.5 Large Scale In-Pile Tests (TRAN)

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).

12.1.6 PAHR In-Pile Tests

PNC joined the Joint Debris Bed Program since 1980 which are conducted under USNRC programmatic management at Sandia National Laboratories, Albuquerque, New Maxico. A technical staff was sent for participation in the conduct and analysis of the debris bed coolability and dry capsule tests.

These results were used in the PAHR analysis of the Monju reactor.

12.1.7 Accident Analysis Codes

1) Whole core accident analysis

Several computer codes are used and maintained in a study of HCDA events in the Monju reactor. These include: SAS3D; VENUS-PM2; SIMMER-II; SAVE for SAS3D/VENUS data transfer; DIF3DS and SPLINT for preparing reactivity worth data to SAS3D; SASGRIP and SASMOVIE for improved graphical representation of the bulky output from SAS3D. SAS4A recently introduced from ANL is being installed and implemented.

Verification efforts were continued for the SAS3D and PAPAS-1S codes mainly through the analysis of the CABRI test data.

New efforts are undertaken for upgrading PAPAS-1S and completing as PAPAS-2S: These include incorporation of EULFCI (fuel-coolant interaction), FLCAST (fuel deformation) and a model for transient fission gas release and swelling of fuel pellets.

2) Local faults

Single-phase subchannel analysis code ASFRE has been applied to a number of in-pile and out-of-pile tests using a focing function in a wire wrap model.

Results prevail the thermo-hydraulic modeling is reasonable in the range of high Reynolds number. The ASFRE code was also applied to the 10th LMCWG Benchmark wall blockage cases with and without leakage flow. The code is being updated to include the mixed connection and natural circulation analysis capabilities for a wire wrapped fuel subassembly.

Single subassembly sodium boiling 2D analysis codes, REDNEC and BOCAL, have been applied to LOF experiments and SLSF-W1 tests to verify their capabilities in view of physical modeling and numerics. REDNEC, using drift flux formulation, was found to be fast in running time and will be used as a tool for parametric studies of low heat flux boiling phenomena.

Based on the assessment of 2D codes, needs for a robust two-fluid model code were called. A 1D two-fluid model analysis code SABENA with primary loop system coupling has been completed. The SABENA code has been applied to a wide spectrum of boiling experinents, including KfK 7 pin LOF experinents, SLSF-W1, and PNC sodium boiling experinents. Currently 2D version is finishing and 3D version development is underway.

3) System safety analysis

PNC started the cooperation with Brookhaven National Laboratory for the Super System Code (SSC) development and application in 1980. Since the introduction of the SSC into PNC, most of the efforts have been directed to applying the code to LOF and LOPI accidents of Monju plant system. For the PLOHS events and other long time transients, SSC will by extensively used to study the consequences of many possible events. In this connection, the ACCS model was developed and is being implemented into SSC.

Regarding the in-vessel thermal-hydraulics, COMMIX-1A was introduced to PNC in 1982. In 1983, PNC organized a benchmark project and round-robin comparison of Japanese multi-dimensional thermo-hydraulic analysis codes and COMMIX-1A was carried out against the Monju upper plenum stratification tests. Benchmark test results have been compiled and will be published. Parallel to the benchmark test, applications of COMMIX-1A to JOYO In-Vessel natural circulation test and thermal stratification test at CRIEPI are now underway.

4) Code system support utilities

Large Code System Maintenance, LAXYM, was upgraded to treat FORTRAN-77. The resultant new version, LAXYM-77 can process CALL COMDECK statements, and refer statement cards either by internal statement numbers or by card identifiers in the source program.

In addition HISTORIAN-processed source programs can be analysed by the new version.

12.2 Shock Structural Experiments and Analyses

Structural response of the reactor vessel against HCDA energetics have been studied for Monju both experimentally and analytically.

1) Scaled model reactor vessel tests (1981-83)

All experimental results were summarized and examined very carefully for the evaluation of Monju reactor vessel integrity.

Some of the experimental results were analyzed numerically of with PISCES2DELK to evaluate the capability of the prediction of PISCES-2DELK code.

2) Analytical studies with PISCES (1981-83)

Sensitivity studies were performed by PISCES-2DELK for the uncertainties of input data and modeling. Effects of the internal structures on the reactor vessel response to HCDA energetics were studied by PISCES-2LK.

These results were summarized and examined very carefully for the evaluation of the mechanical margin of the Monju reactor vessel.

3) Tensile tests on materials (1981-)

Basic material tests on high-speed tensile properties have been performed jointly by JRC/Ispra and PNC.

12.3 Radiological Consequences

1) Fission product loop

The FPL-2 sodium in-pile loop has been installed in the Toshiba Training Reactor in June, 1982. The loop has a delay line of about 25 meters and equipped with eleven gamma collimators and two DN detectors. The test series is aimed at measuring the transfer and deposition of fission products, and ultimately understanding the signals from failed fuel elements.

Sixteen experiments were carried out from September 1982 to March 1983. The major parameters were sodium flow rate, sodium temperature, and reactor power. Many nuclides were detected for the first time in sodium loop. The first analyzed FP elements were Sr, Ba, and Zr, and further analysis is under way. A second series of experiments has been initiated since September 1983.

2) Sodium spray fire experiments

For the purpose of validating the SPRAY code used in Monju licensing and of checking the influence of water vapor in the atmosphere, six spray tests were conducted in a 21 m^3 cell covered internally by steel liners. The spray nozzle on the ceiling ejects about 50 kg of sodium at 530°C, the mean diameter of the sprayed sodium particles is about 1 mm. The temperature and pressure of the gas, the change of concentrations and the generation rate of aerosols were measured. The experiments revealed that the humidity has practically no influence on the peak temperature and pressure of the gas and that no hydrogen is generated when the oxygen content is 21 volume percent.

SPRAY predicts gas temperature and pressure in the case of low oxygen concentration and further analysis is being continued for high oxygen tests.

3) Sodium fire analysis

To enable analysis of simultaneous sodium fires (spray and ensuing pool) in the secondary system, an attempt was made in coupling SPRAY and SOFIRE into a unified code, ASSCOPS. This utilizes the 2-cell model of SOFIRE and a spray fire is assumed in the upper cell. Because of the differences in the geometry and compositions in the two cells, the coupling was not easy. We have reached the stage that the calculation results seem reasonable.

4) Containment analysis

CONTAIN developed by Sandia for US-NRC has been checked out on a CYBER-176. Simultaneous conversion of the code to the IBM-compatible form has been completed.

The applicability of the CONTAIN-1B code has been studied to verify sodium combustion models and to examine their capability of sodium fire analyses in the Monju plant. The code was compared with the SPRAY-3 code and the SOFIRE-M II code.

5) Sodium concrete reaction studies

Thirteen experiments were conducted as a first series using graywacke concrete which is planned for Monju. Sodium of 0.75 - 8 kg at 430 - 630°C was spilled onto the concrete specimens of 20 cm diameter and 30cm length. The penetration rate into the concrets and hydrogen generation rate were measured. For temperatures higher than 530°C, the violent reaction occurred as well as basalt concrete.

A second series test have been carried out to study the effect of vertically faced concrete and sodium/perlite concrete reactions. A third series will be initiated at the end of this year.

12.4 Sodium Fires

1) Basic test

The basic tests are to study the performance of funneling floor, draining pipe and fire suppression system in a small scale. In this tests, the maximum of 180 kg of sodium used. Three tests were carried out and the basic characteristics of sodium fire and effectiveness of fire protection systems were revealed.

2) Large scale tests

The large scale test are to study the sodium fires with a half scale model of Monju secondary building, where a fire suppression system, funneling floors, and draining system are installed. The facility construction starts from early 1984, and the test will be started from early 1985. In the test, the maximum of 2o tons of sodium in total weight will be spilled in the model.

3) Water simulation test

Seven tests were carried out using water to observe the splash pattern from the defected pipe of IHTS. Three parts of pipe such as the straight pipes, bends and elbows were simulated and effect of inner jackets, outer jackets and thermal insulating material was studied. Quantitative study will be continued.