PNC
TN241 85−03

PNC INFORMATION
FOR
USDOE ADVANCE TEAM MEETING WITH PNC
ON ADVANCED REACTOR COOPERATION

MARCH 28, 1985

POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION

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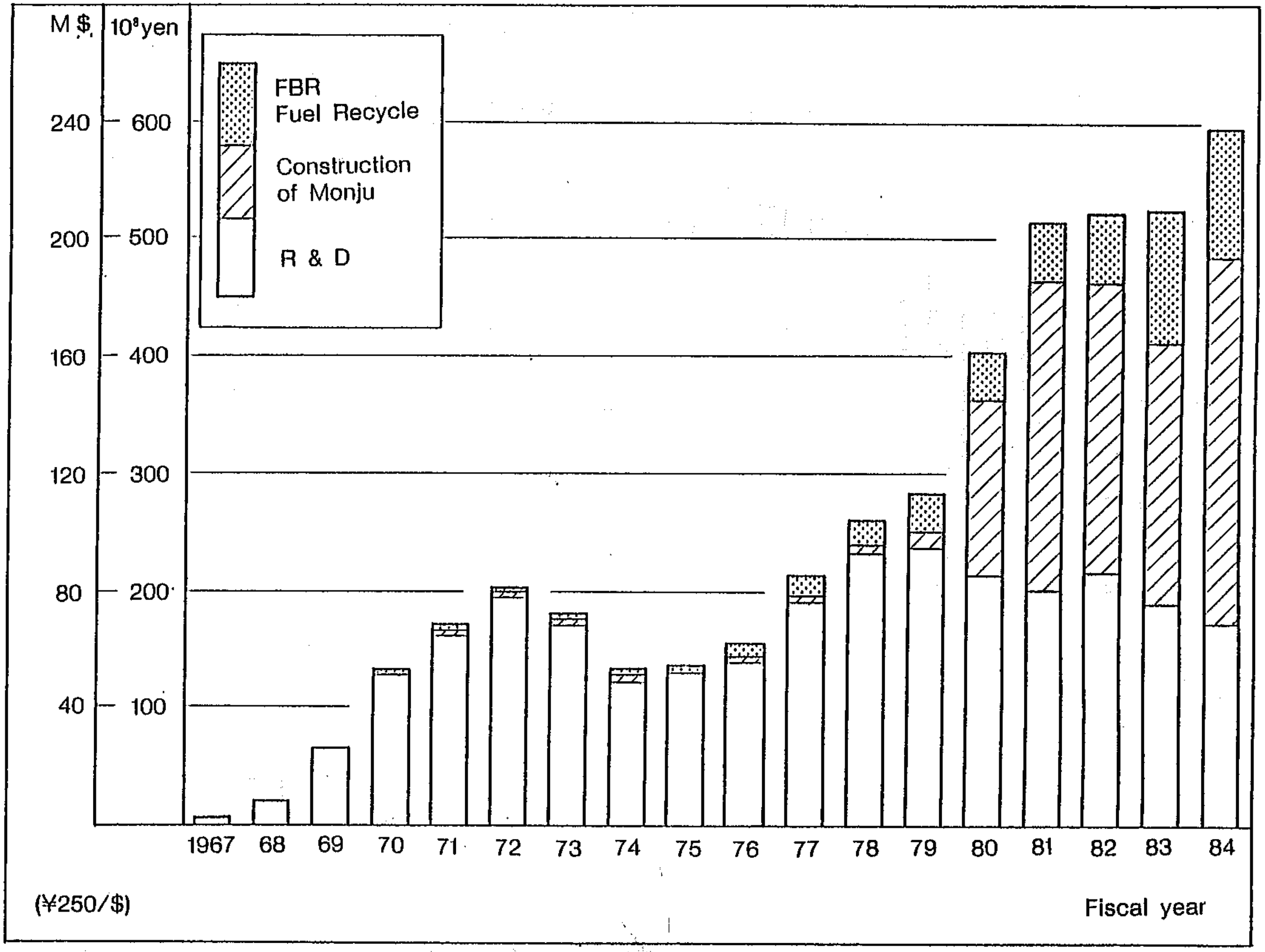
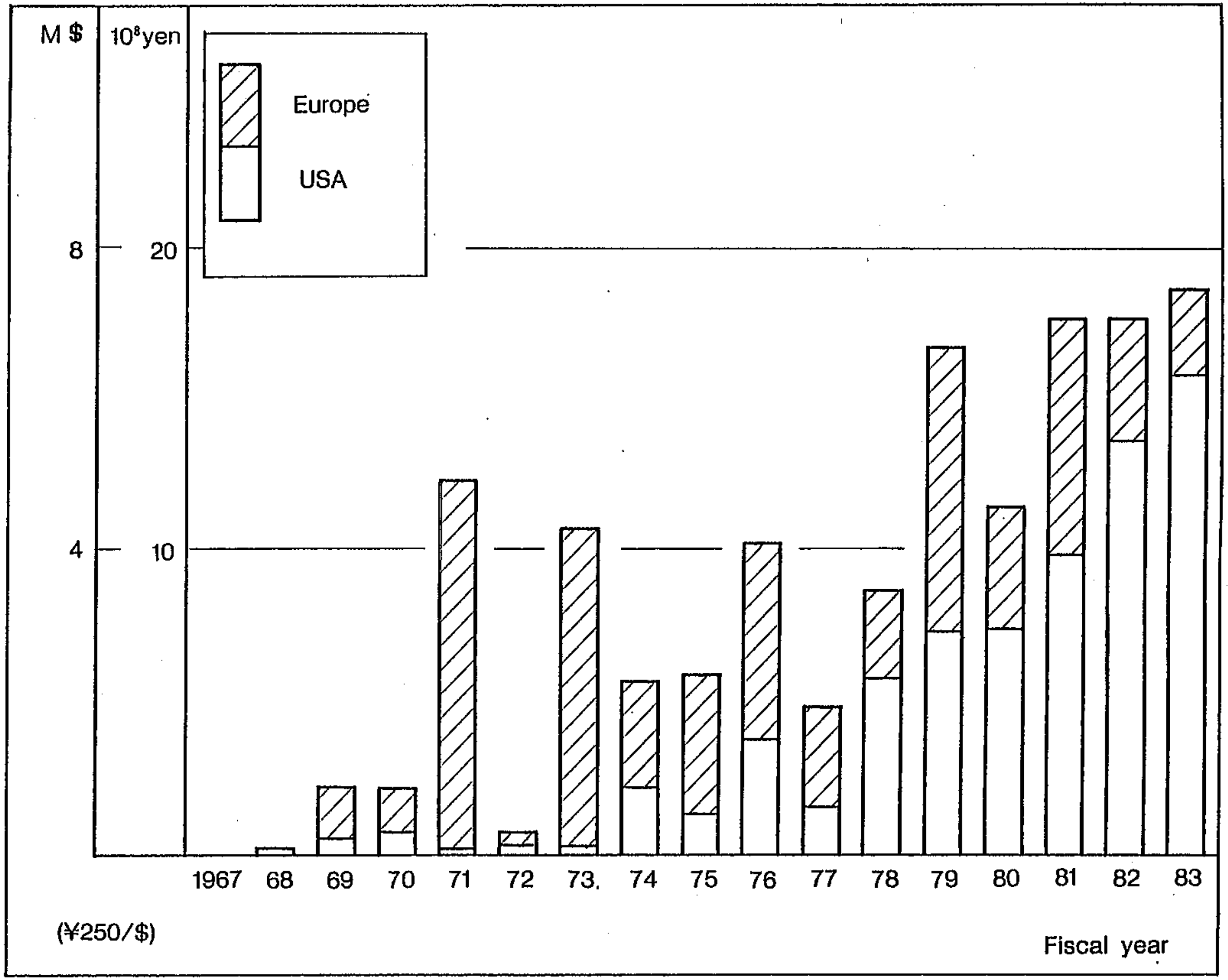
- FBR Steam Generator Development
- Fuel and Material Division
- FBR Safety
- FBR Fuel Reprocessing Technology Development
- Chemical Processing Facility

Budget (FBR Development)



Payment for International Cooperation (FBR Development)



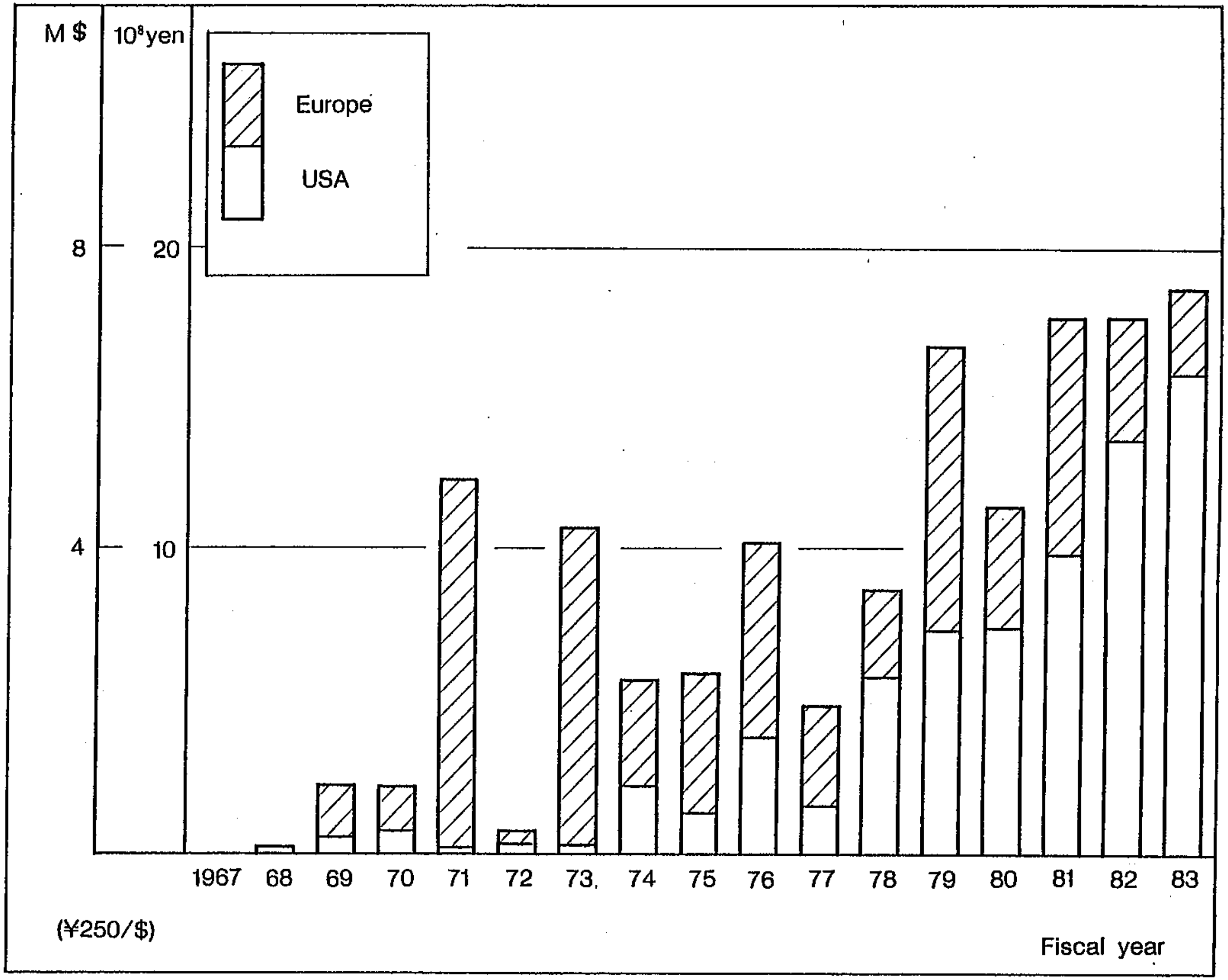
I. FBR Development Program of PNC

FBR DEVELOPMENT PROGRAM OF PNC

1. OPERATION OF EXPERIMENTAL REACTOR "JOYO"
2. CONSTRUCTION AND OPERATION OF PROTOTYPE REACTOR "MONJU"
3. DESIGN AND RELATED R & D OF DEMONSTRATION REACTOR AND SUCCEEDING ONES
4. RESEARCH AND DEVELOPMENT ON FBR TECHNOLOGY
5. FBR FUEL CYCLE

2. Budgets

Payment for International Cooperation (FBR Development)

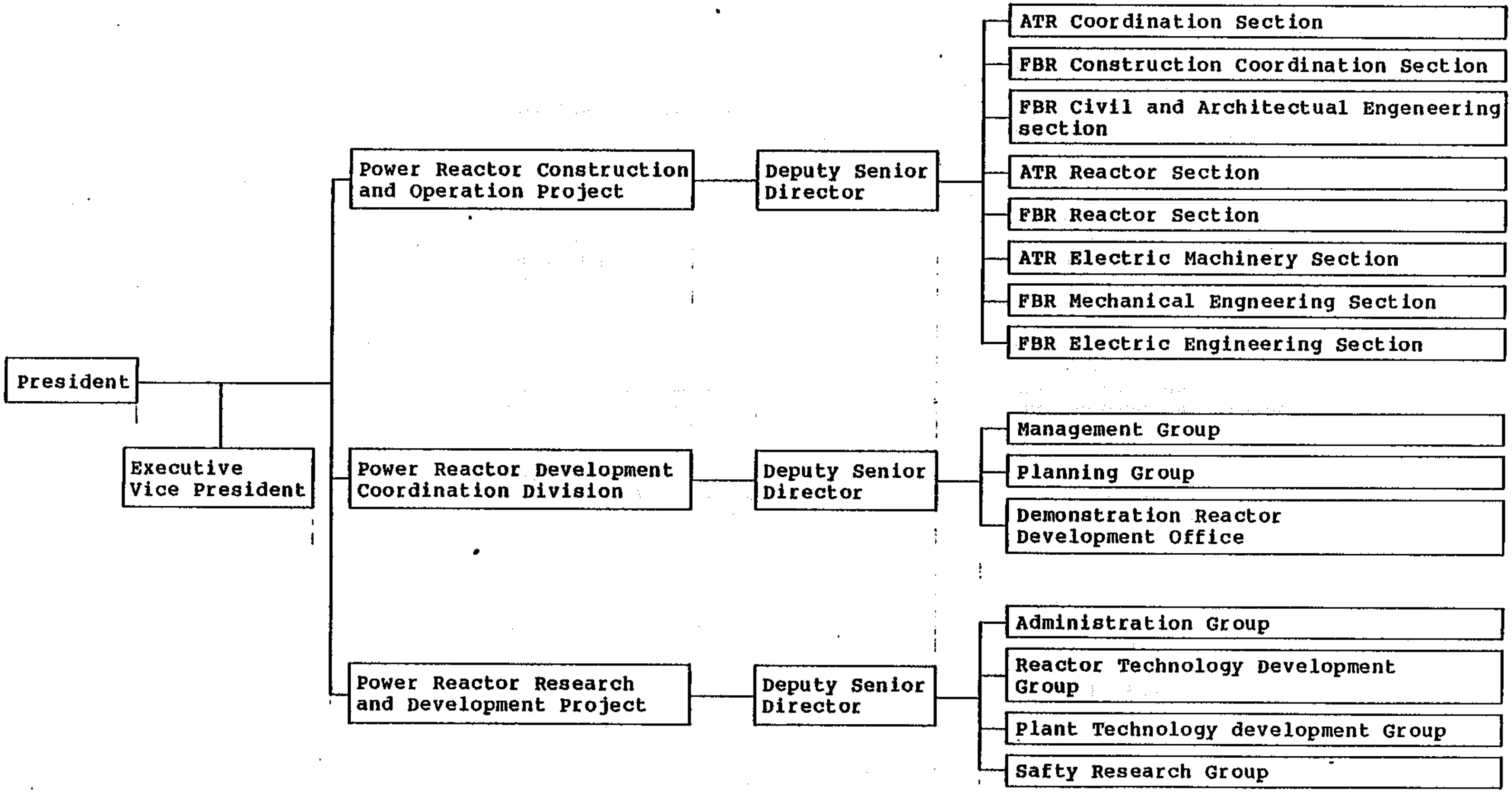


3. Organization and Number of Employee

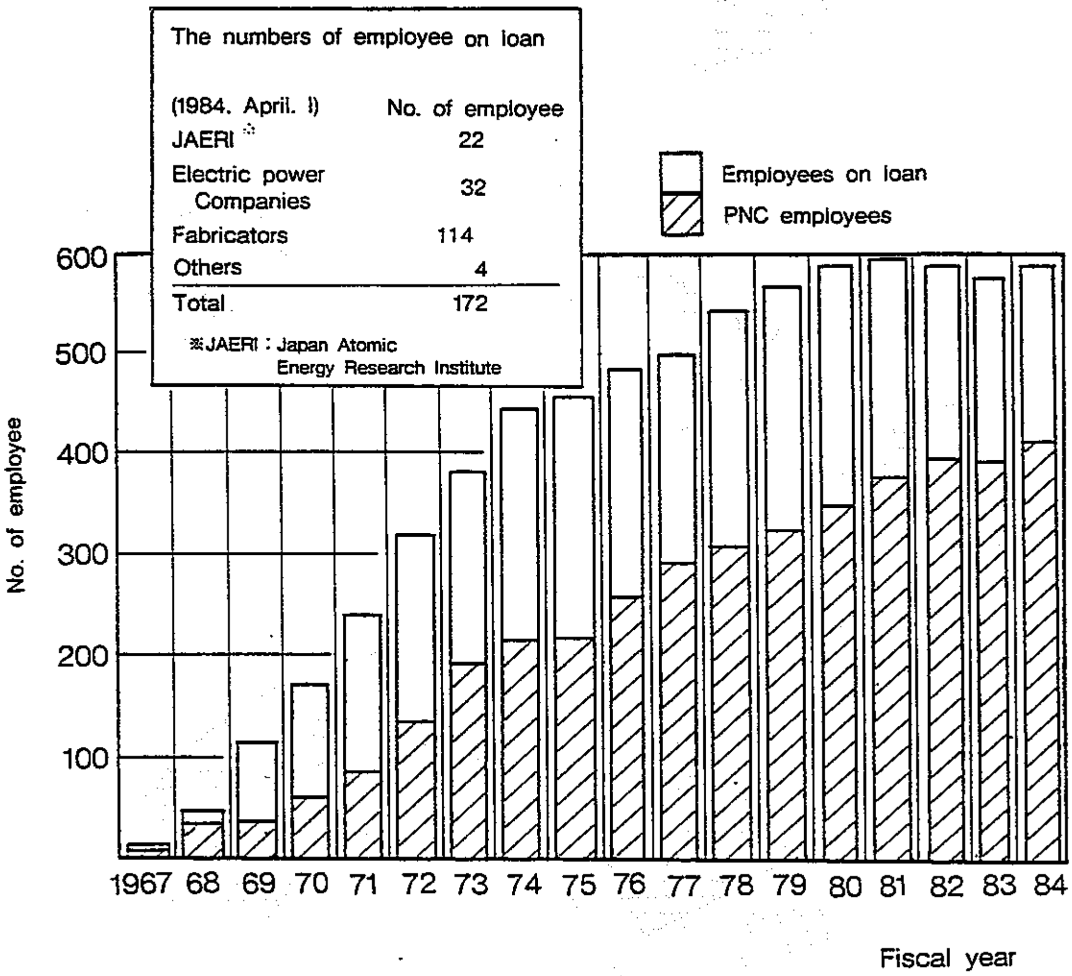
O-ARAI ENGINEERING CENTER

- Director
  - Vice Director
    - Health and Safety Division
      - Safety Administration Section
      - Radiation Control Section
    - Administration Division
      - General Affairs Section
      - Labor and Personnel Section
      - Accounting Section
      - Purchase Section
      - Maintenance Section
      - Waste Management Section
    - Technology Development Division
      - Research Management Group
      - Plant Engineering Office
      - Technology Management Office
    - Systems and Components Divisions
      - FBR Components Development Section
      - Structural Engineering Section
      - Materials Development Section
      - ATR Components Development Section
    - Fuels and Materials Division
      - Analysis and Evaluation Section
      - Alpha-Gamma Section
      - Materials Monitoring Section
      - Fuels Monitoring Section
    - Experimental Reactor Division
      - Coordination Section
      - FBR Operation Section
      - FBR Maintenance Section
      - Reactor Technology Section
      - Irradiation Section
      - Deuterium Critical Experiment Section
    - Safety Engineerig Division
      - FBR Engineering Section
      - FBR Reactor Safety Section
      - FBR Plant Safety Section
      - ATR Reactor Safety Section
      - ATR Heat Transfer Section

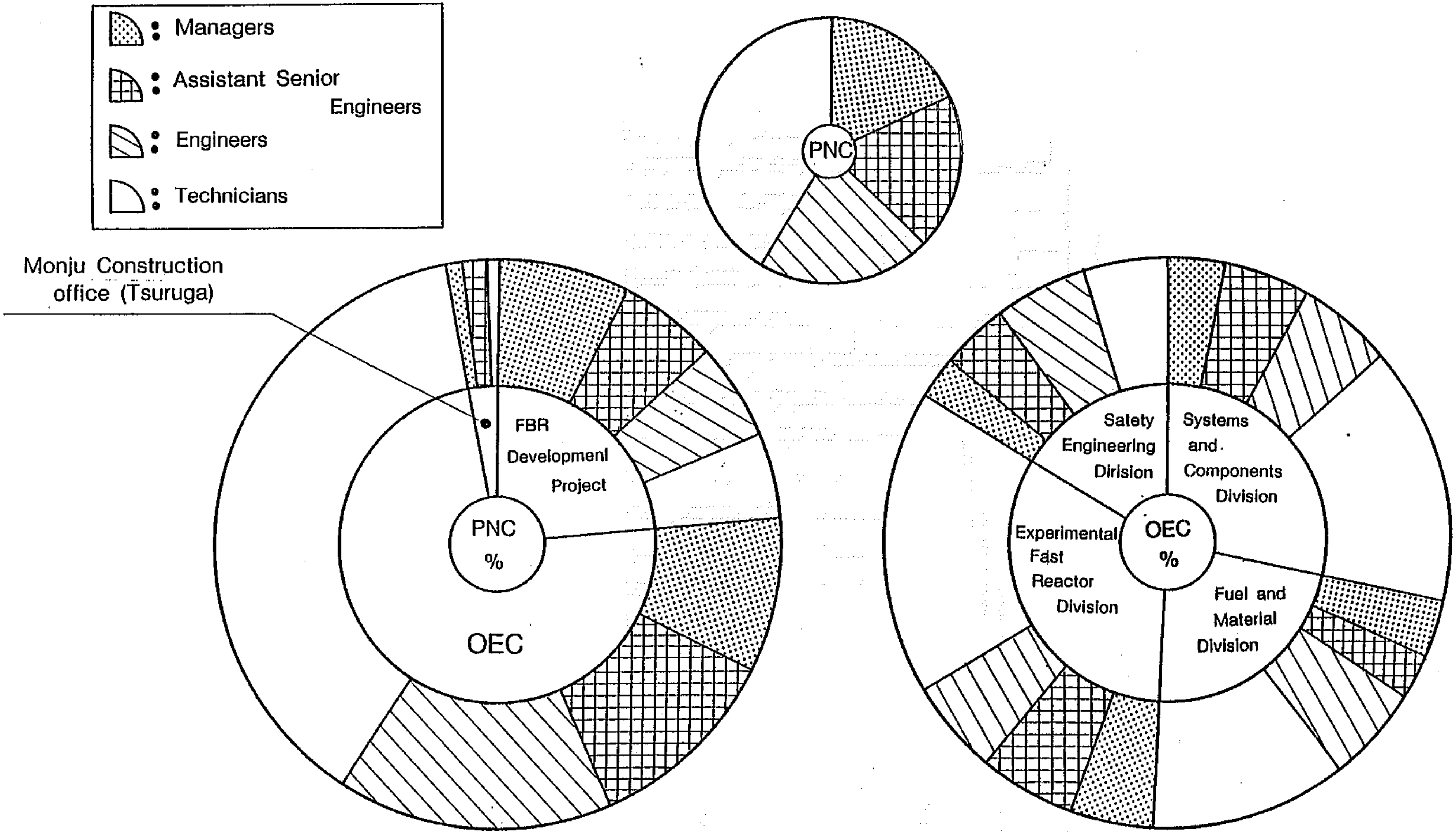
POWER REACTOR PROJECT ORGANIZATION



Numbers of Employee for FBR Development (1967~1984)



PNC EMPLOYEE OF FBR DEVELOPMENT (1984)



OEC: O-arai Engineering Center

4. List of Major Facilities

LIST OF MAJOR FACILITIES

1. O-ARAI ENGINEERING CENTER

(1) EXPERIMENTAL FAST REACTOR "JOYO"

(2) SODIUM COMPONENTS TEST FACILITY–1

(3) SODIUM COMPONENTS TEST FACILITY–2

(4) HYDRODYNAMICS TEST FACILITY–1

(5) HYDRODYNAMICS TEST FACILITY–2

(6) FLUID DYNAMICS TEST FACILITY

(7) SODIUM TECHNOLOGY TEST FACILITY–1

(8) SODIUM TECHNOLOGY TEST FACILITY–2

(9) SODIUM TECHNOLOGY TEST FACILITY–3

(10) SODIUM ANALYSIS LABORATORY

(11) 1-MW STEAM GENERATOR TEST FACILITY

(12) 50-MW STEAM GENERATOR TEST FACILITY

(13) FAST REACTOR SAFETY TEST FACILITY–1

(14) FAST REACTOR SAFETY TEST FACILITY–2

(15) FAST REACTOR SAFETY TEST EACILITY–3

(16) FAST REACTOR SAFETY TEST FACILITY–4

(17) ALPHA-GAMMA FACILITY

(18) MATERIAL MONITORING FACILITY

(19) FUEL MONITORING FACILITY

(20) RADIOACTIVE WASTE TREATMENT FACILITIES

2. TOKAI WORKS

(1) FUEL INSPECTION DEVELOPMENT LABORATORY

(2) FUEL ANALYSIS DEVELOPMENT LABORATORY

(3) PLUTONIUM FUEL DEVELOPMENT LABORATORY

(4) CHEMICAL PROCESSING FACILITY

(5) ENGINEERING TEST FACILITY

(6) APPLICATION TEST FACILITY

5. Major FBR Development Activities

5. Major FBR Development Activities

5.1 Experimental Fast Reactor "Joyo"

5.2 Prototype Reactor "Monju"

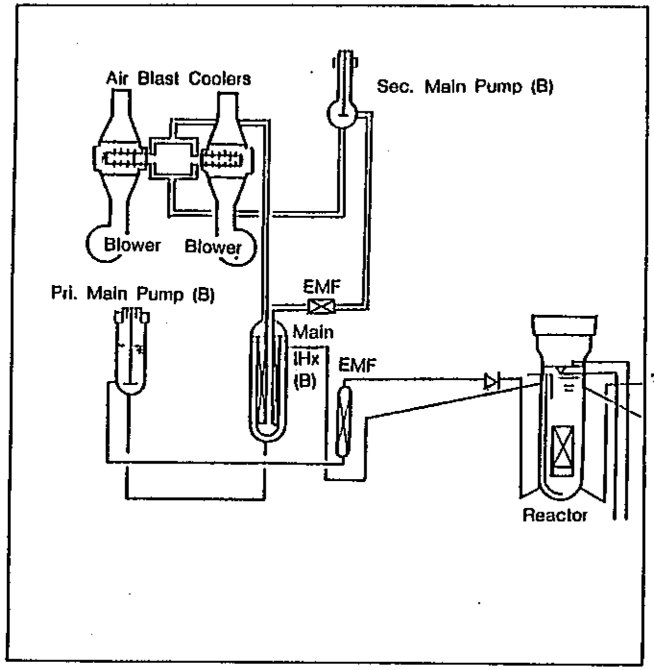
5.3 Demonstration Reactor

5.4 Research & Development on FBR Base Technology

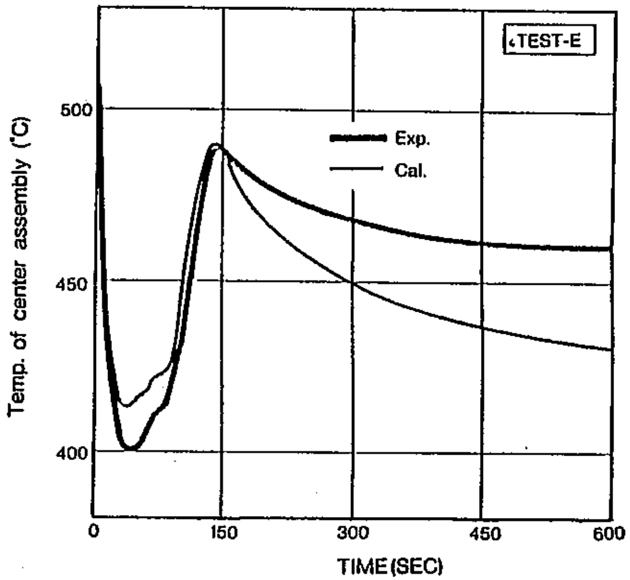
5.5 R & D Activities

5.1 Experimental Fast Reactor "Joyo"

Test at "JOYO" → Analysis → Evaluation of code → Extrapolation for "MONJU" design

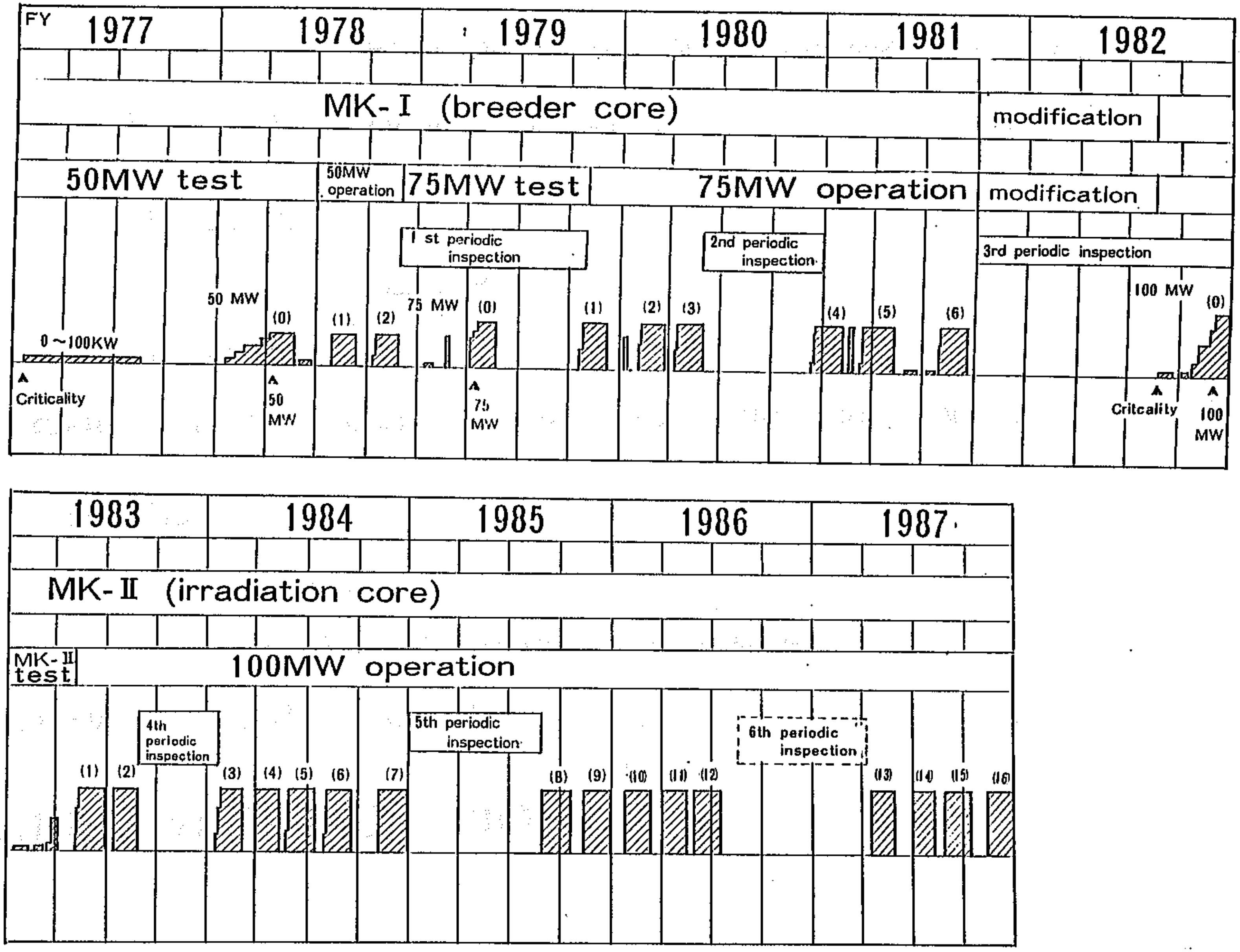


JOYO plant system



Natural Circulation Test

"JOYO" Operation Schedule



FUTURE TARGET OF "JOYO"

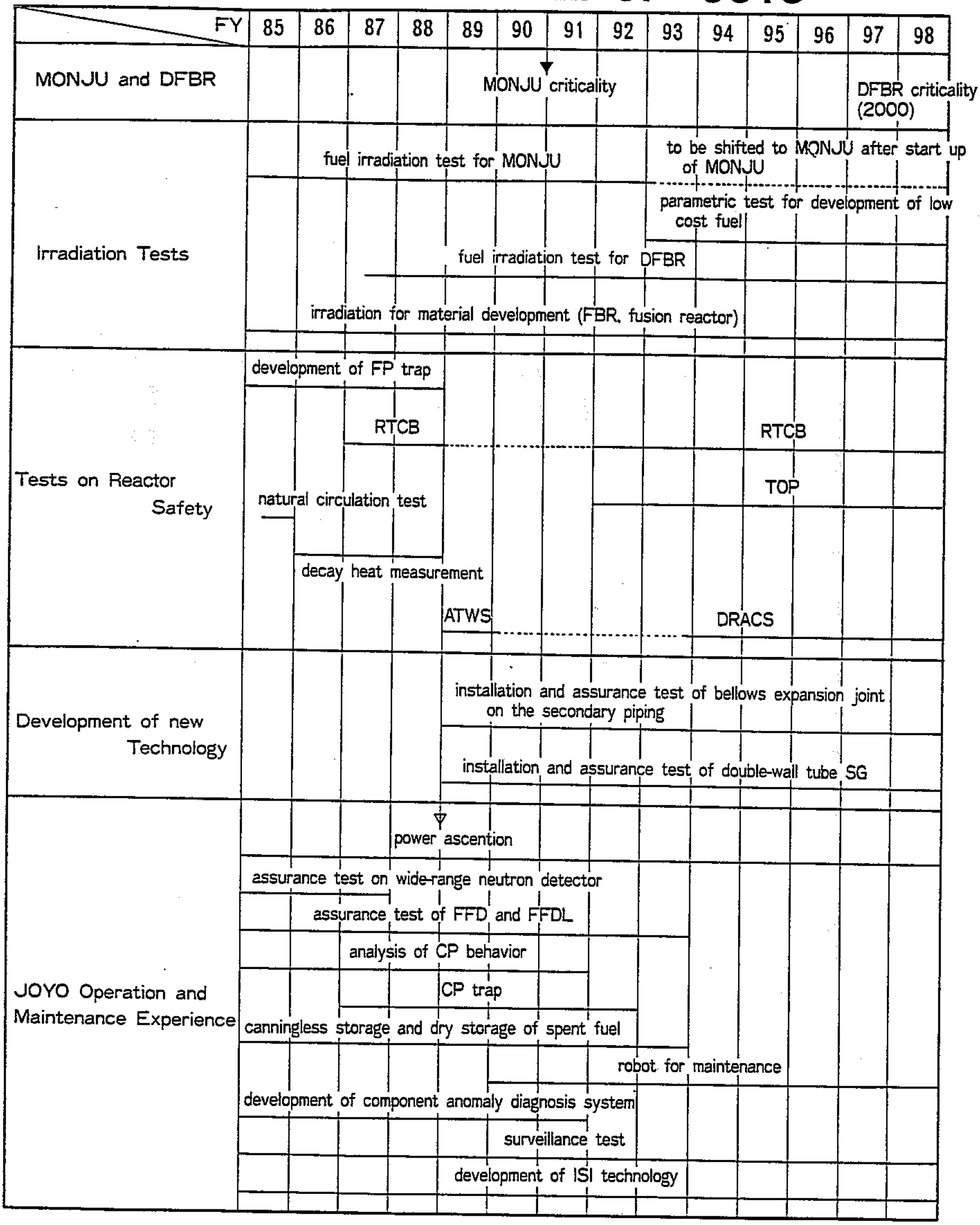
1. IRRADIATION TESTS
   - ○ High Quarity Irradiation Technique
   - ○ Power to Melt Test
   - ○ RTCB Test

2. GENERALIZATION OF OPERATION AND MAINTENANCE EXPERIENCE
   - ○ Reduction of CP Level
   - ○ FFD, FFDL

3. PREVENTIVE MAINTENANCE
   - ○ Analysis of Failure Data and a Component Reliability Evaluation

4. CONTRIBUTION TO FBR COST REDUCTION

LONG TERM PROGRAM OF "JOYO"



Irradiation Test in JOYO for Fuel and Materials

- Examine Performance of MONJU Design
  - Fuel Pin Irradiation
    - High Power (465W/CM) — A 1 M
    - Middle Burn-up (62.4GWD/T) — B 2 M
    - Maximum Burn-up (91GWD/T) — B 3 M
  - Fuel Bundle Irradiation
    - Wear Mark — C 3 M
    - Middle Burn-up (67GWD/T) — C 2 M
    - Maximum Burn-up (105GWD/T) — C 3 M
  - Material Irradiation
    - Maximum neutron fluence (2.3×10^{23}nut) — CM I R
- Basic Test
  - Fuel Pin Irradiation
    - Tag Gas Behaviour in Fuel Pin — B 0 J, B 1 J, B 2 M, B 3 M
    - Anti-FCCI — A 1 M
    - Plenum Pressure — B 1 J, B 2 M, B 3 M
    - Fuel Temperature (On-line) — INTA
    - F.P Gas Pressure (On-line) — INTA
  - Fuel Bundle Irradiation
    - Wear mark — C 1 J, C 2 M, C 3 M
    - Thermo-Hydraulic Performance (On-line) — INTA
  - Material Irradiation
    - Swelling and Creep — CMIR

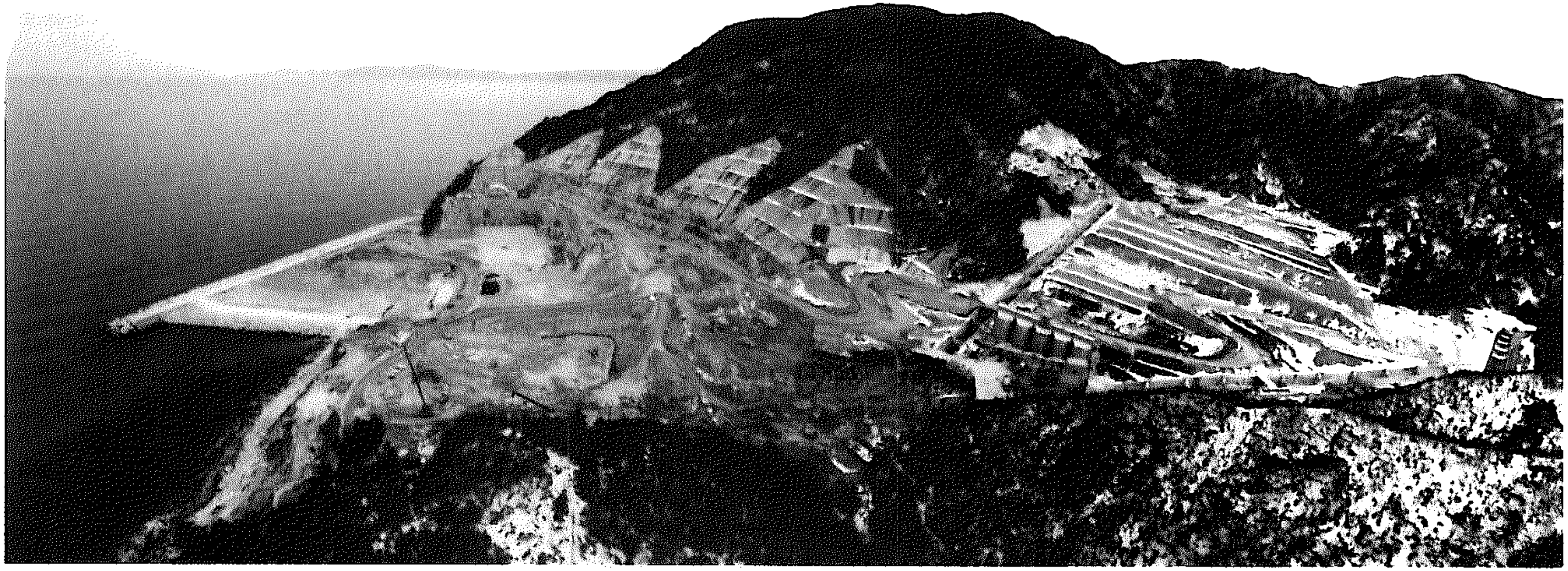
Irradiation Test in JOYO for Fuel and Materials (Cont'.)

- Absorber Materials — MONJU Confirmation Test — B_4C — He Release, Swelling — AMIR-1, AMIR-2
- Structural Materials
  - MONJU Surveillance Backup Test
  - MONJU CRD Confirmation Test
    - Spring — Spring Coefficent — UPR-1
    - Magnet-Holding Power — UPR-1
- Na Technology — CP Trap Test — Ni Plate — Ni Behavior — A1M, UPR-1, 2

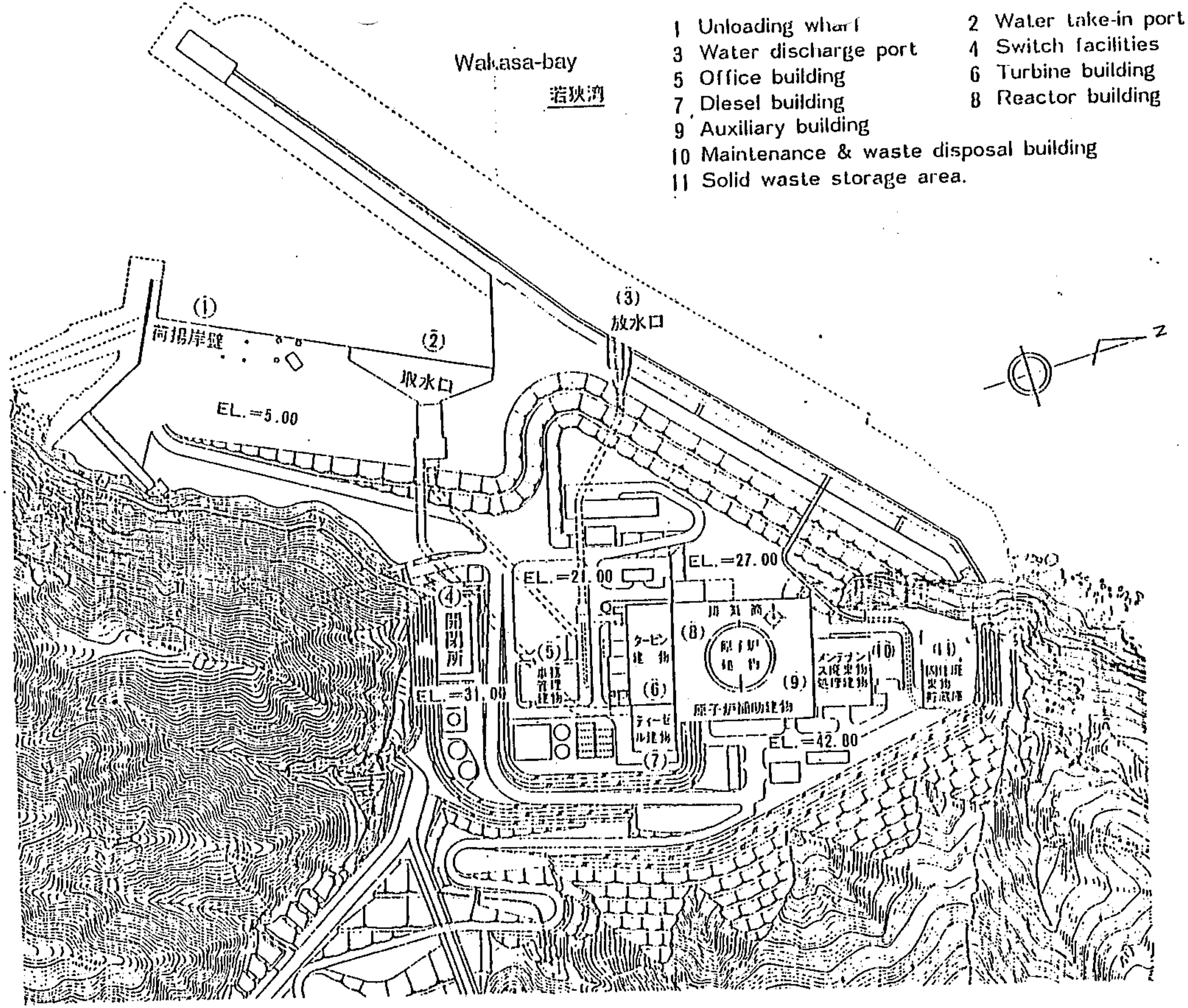
- JOYO
  - Irradiation Technology — Irradiation Device
    - Irradiation Rig — A1M,B0J,CMIR-0, INTA,UPR-1,C1J
    - Detector (On-line) — INTA-1,UPR-1
    - Off-line Monitor — Every Test
  - Reactor Characteristic — Irradiation Device
    - Reactivity — B0J,C1J,CMIR-0
    - Temperature Fluctuation around "Upper Core" Structure — UPR-1
    - Temperature Distribution around Na Liquid Level — UPR-1
    - Temperature Distribution in Shield Plug — UPR-1
    - Simulation of Failed Fuel Pin — F1B
  - Monitoring
    - Core Fuel Assembly — Fuel Behavior — Driver Fuel Assembly
    - Control Rod — Control Rod Behavior — Driver Control Rod
    - Structural Material Sureillance — Structural Material Behavior — Sureillance Assembly

5.2 Prototype Reactor "Monju"

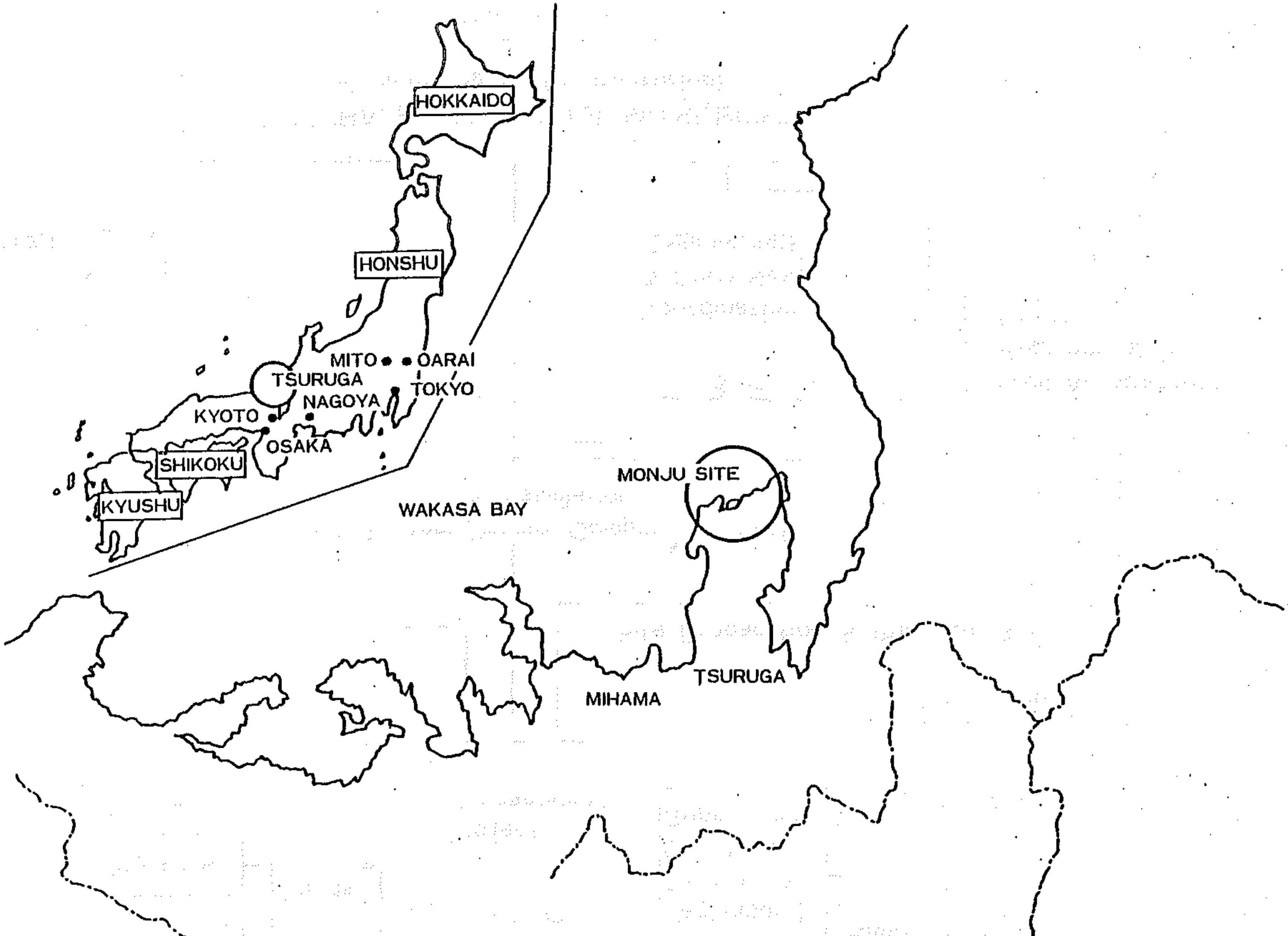
5



"MONJU" SITE TSURUGA FEB, 1985

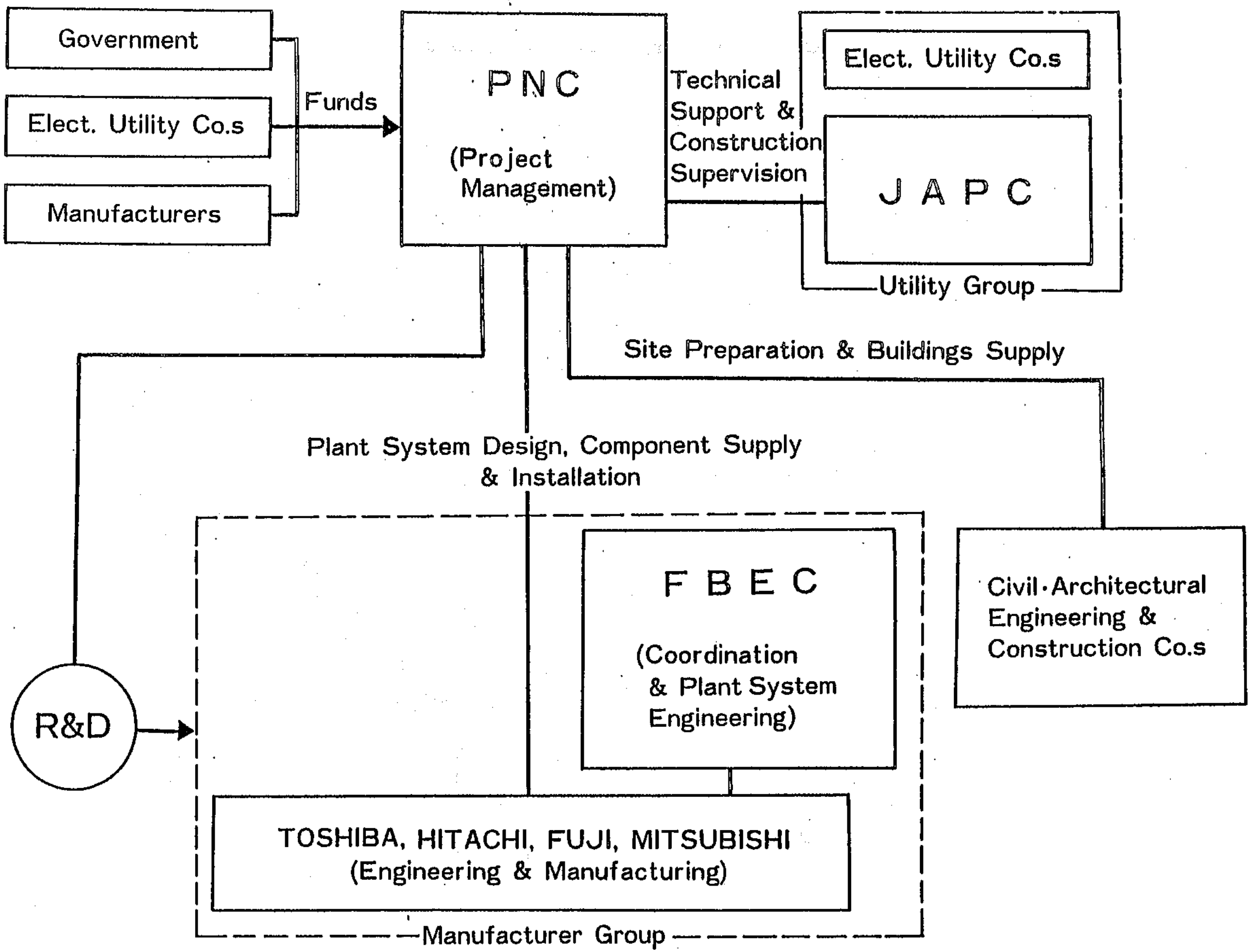


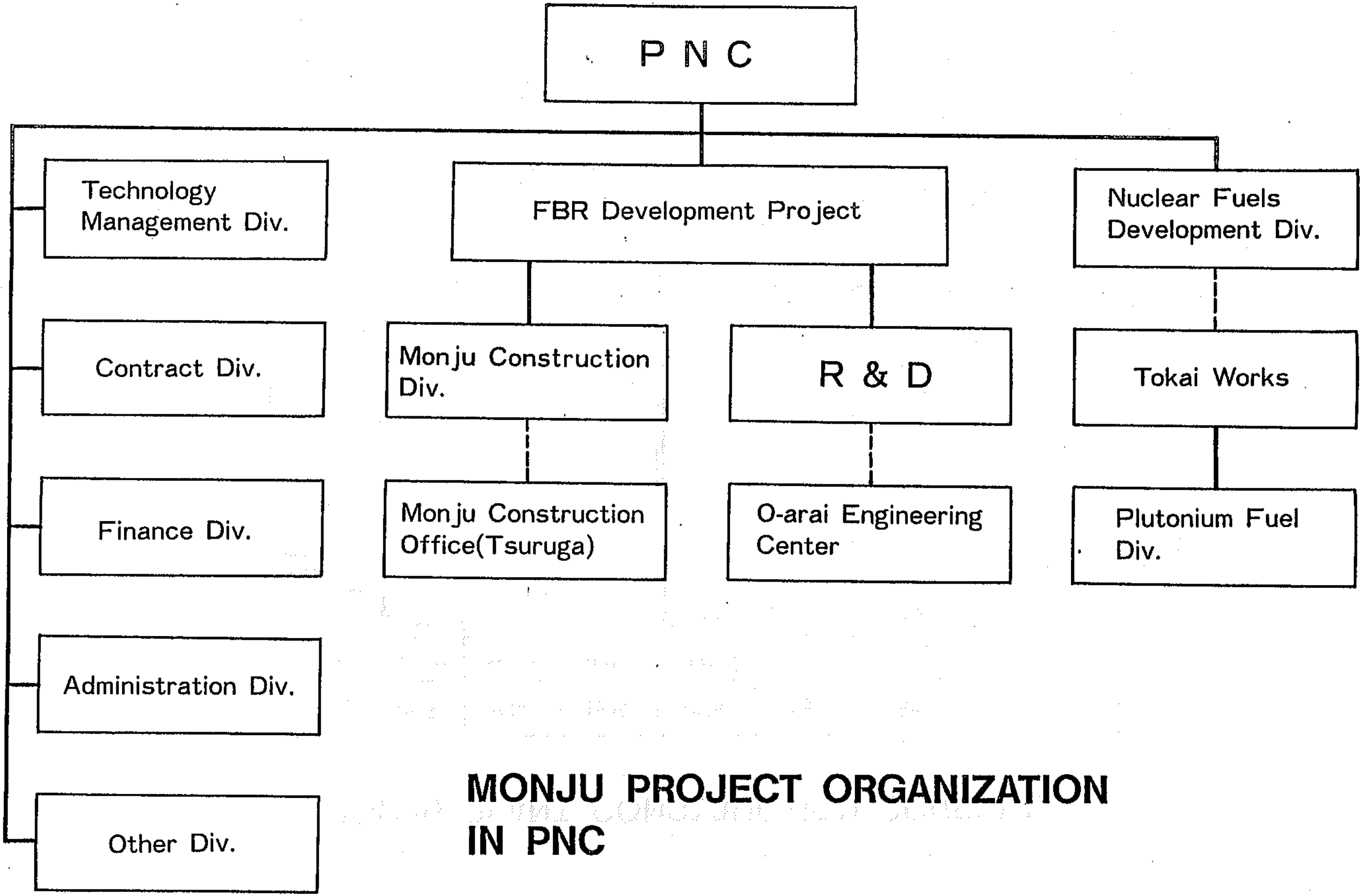
Plant Layout of Monju



LOCATION OF MONJU SITE

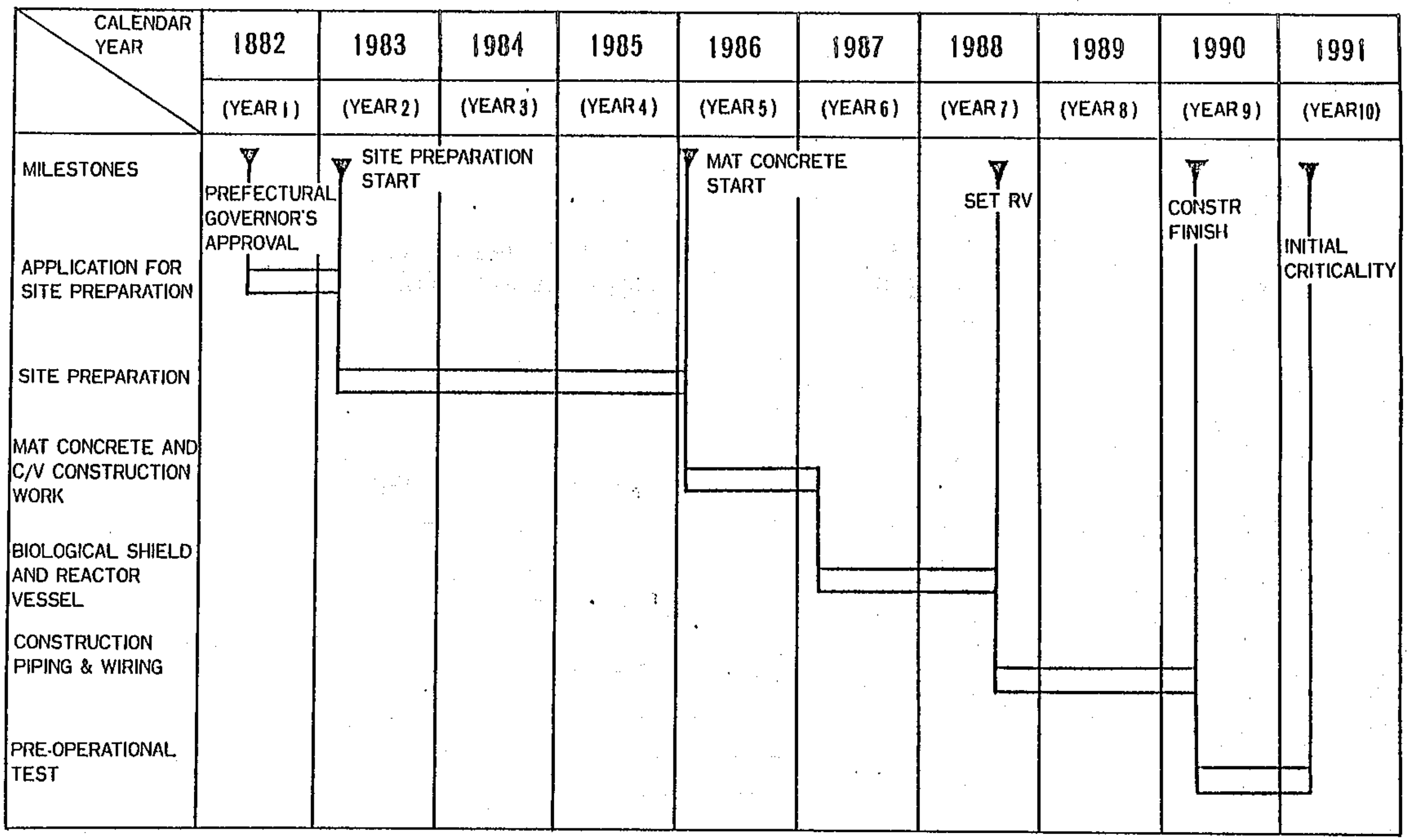
"Monju" Project Organization





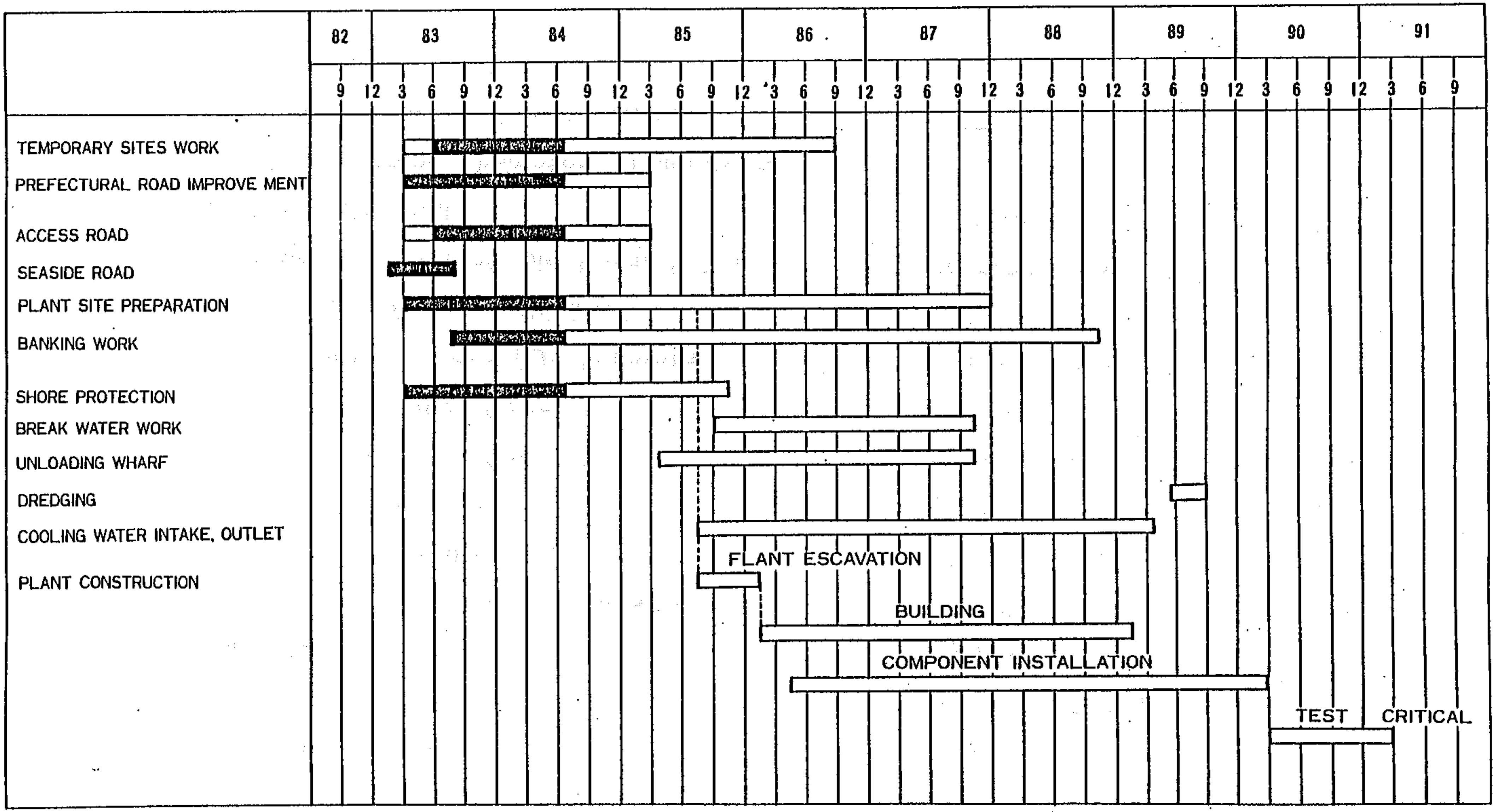
MONJU PROJECT ORGANIZATION IN PNC

MONJU PLANT CONSTRUCTION SCHEDULE



SITE PREPARATION

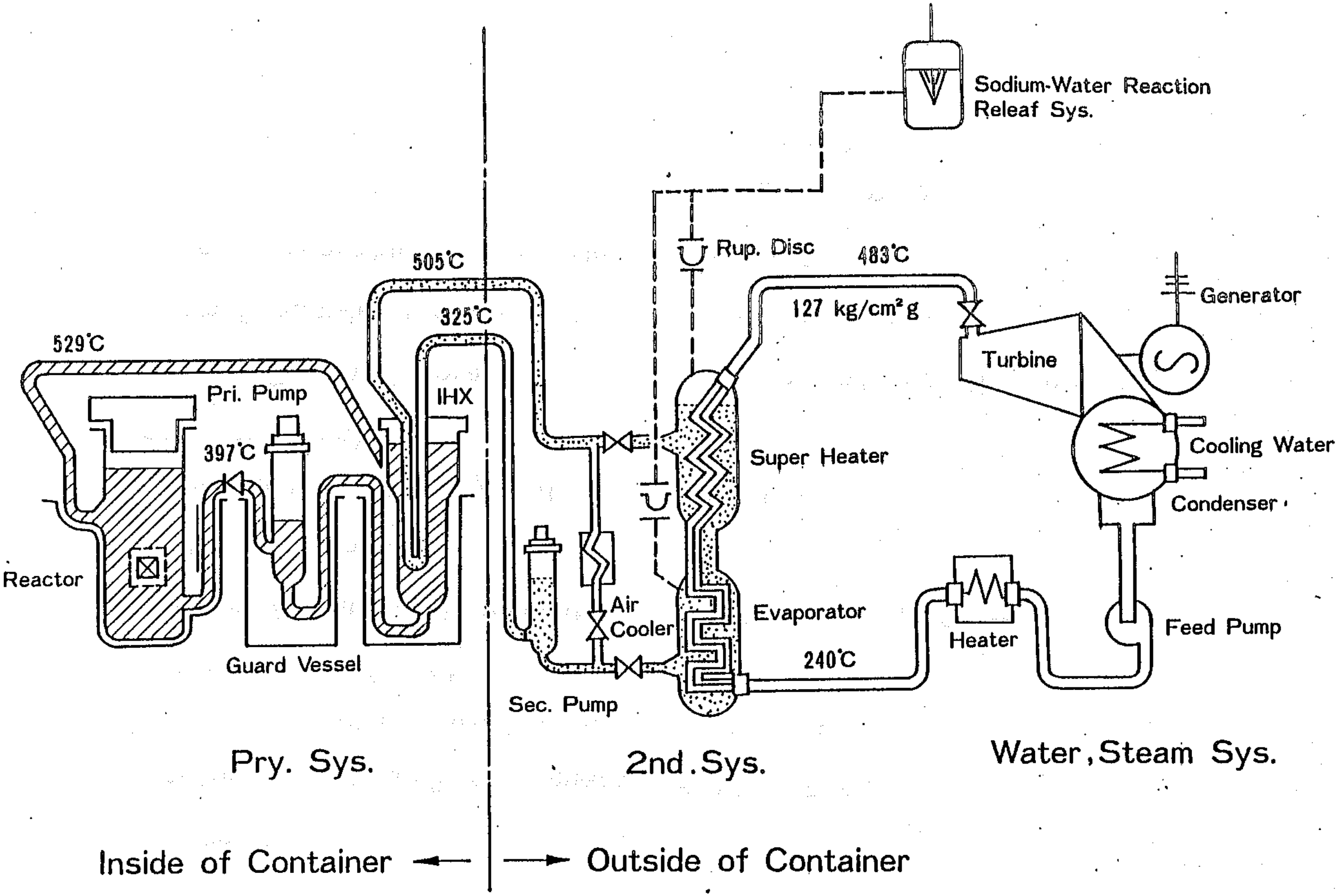
JULY 31. 1984



Principal Design and Performance Data of "MONJU"

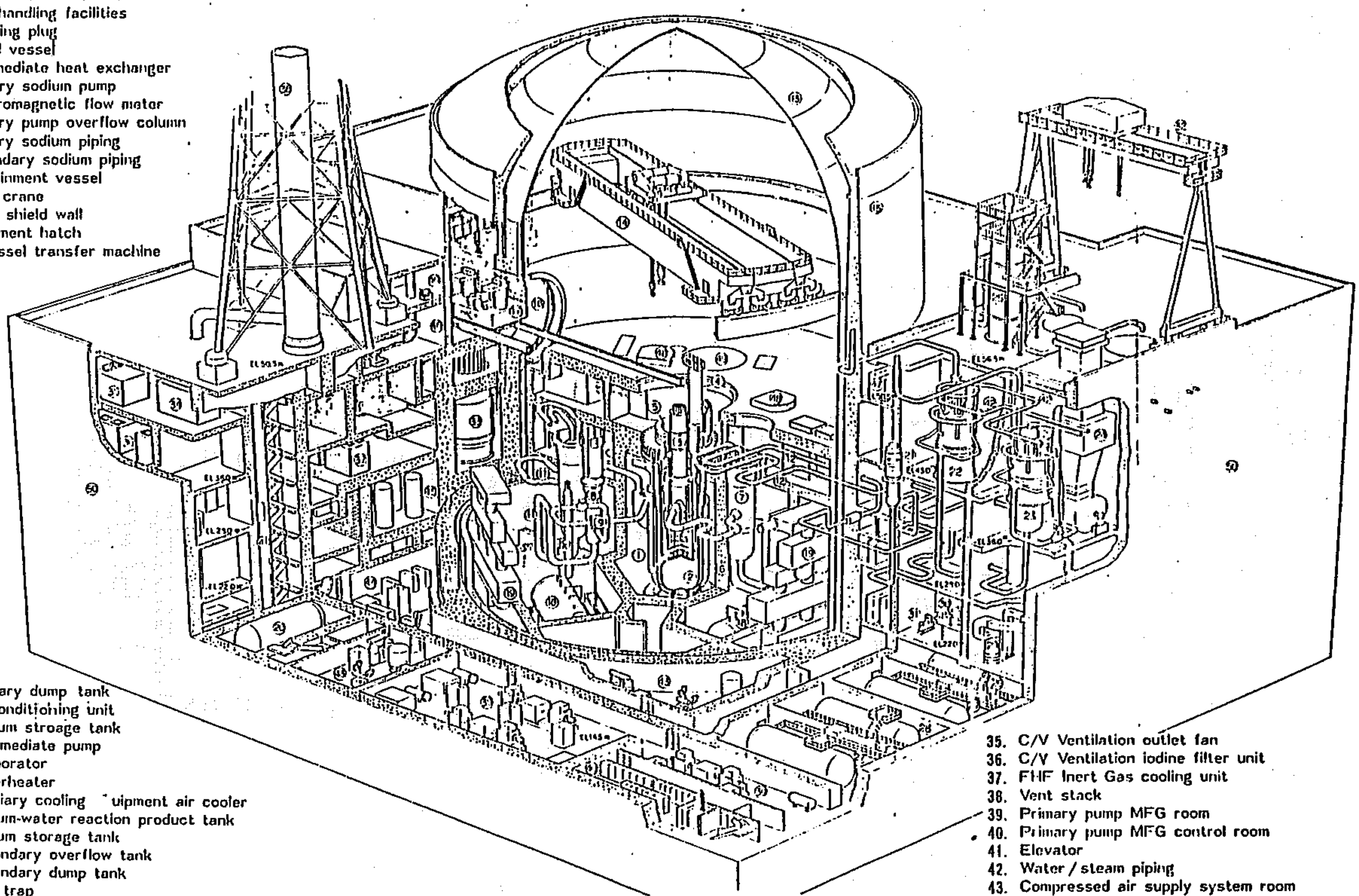
Reactor Type		Sodium cooling loop type
Thermal Power		714 MW
Electrical Power		about 280 MW
Fuel Material		PuO_2-UO_2
Core Fuel	Equivalent Diameter	1,790 mm
	Height	930 mm
	Volume	2,340 lit.
Pu Enrichment (Pufiss %)		Inner core / Outer core
	Initial Core	15./20
Fuel Inventory	Core (U+Pu-metal)	5.9×10^3 kg
	Blanket (Umetal)	1.75×10^4 kg
Average Burn up of Discharged Fuel		80,000 MWD / T
Cladding Material		SUS 316
Cladding Outside Diameter / Thickness		6.5 / 0.47 mm
Permissible Cladding Temperature (mid-thickness)		675 ℃
Power Density		280 kW / lit.

Blanket Thickness (axial / radial)	Upper 300 mm Lower 350 mm / 300 mm
Breeding Ratio	1.2
Reactor in / out Sodium Temperature	397 / 529 ℃
Secondary Sodium Temperature (IHX outlet / IHX inlet)	505 / 325 ℃
Reactor Vessel (height/diameter)	17,800 / 7,100 mm
Number of Loops	3
Pump Position (Primary and secondary loop)	Cold leg
Type of Steam Generator	Helical coil, once-through unit type
Steam Pressure (turbine inlet)	127 kg / cm^2g
Steam Temperature (turbine inlet)	483 ℃
Refueling System	Single rotating plug with fixed arm FHM
Refueling Interval	6 months



Main System of MONJU

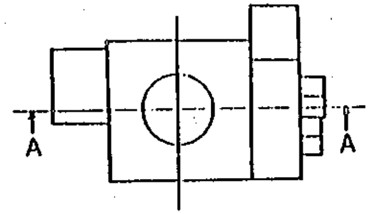
Core
2. Reactor vessel
3. Control rod drive mechanism
4. Fuel handling facilities
5. Rotating plug
6. Guard vessel
7. Intermediate heat exchanger
8. Primary sodium pump
9. Electromagnetic flow meter
10. Primary pump overflow column
11. Primary sodium piping
12. Secondary sodium piping
13. Containment vessel
14. Polar crane
15. Outer shield wall
16. Equipment hatch
17. Ex-vessel transfer machine

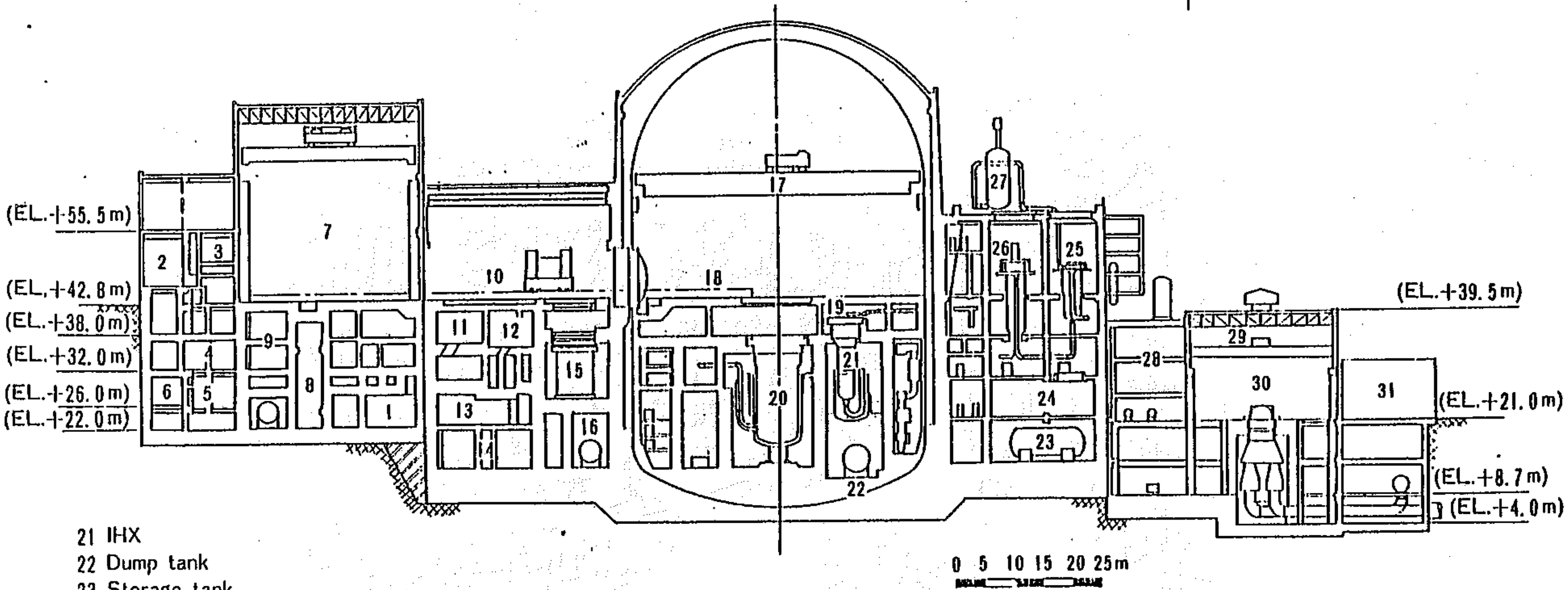


18. Primary dump tank
19. N_2-conditioning unit
20. Sodium stroage tank
21. Intermediate pump
22. Evaporator
23. Superheater
24. Auxiliary cooling uipment air cooler
25. Sodium-water reaction product tank
26. Sodium storage tank
27. Secondary overflow tank
28. Secondary dump tank
29. Cold trap
30. Economizer
31. Electromagnetic pump
32. SG crane
33. Ex-vessel storage tank
34. C/V Ventilation Unit

35. C/V Ventilation outlet fan
36. C/V Ventilation iodine filter unit
37. FHF Inert Gas cooling unit
38. Vent stack
39. Primary pump MFG room
40. Primary pump MFG control room
41. Elevator
42. Water / steam piping
43. Compressed air supply system room
44. Radioactive Argon proessing subsystem
45.
46. EVST cooling system room
47. Secondary pre eat power control room
48. Hatch for main pump
49. Hatch for IHX
50. Reactor auxiliary building

1 Ventilator and conditioner room
2 Maintenance room
3 Switchgear room
4 Monitor tank room
5 Miscellaneous waste collector tank room
6 Concentrated miscellaneous waste tank room
7 Maintenance area
8 FHM cleaning room
9 Cleaning ventilation system room
10 Ex-vessel transfer machine room
11 Fuel canning room
12 Fuel cleaning room
13 Fuel inspection equipment room
14 Gaseous radwaste processing system room
15 EVST
16 Sodium overflow tank
17 Polar crane
18 Operation floor
19 IHX head area
20 Reactor vessel

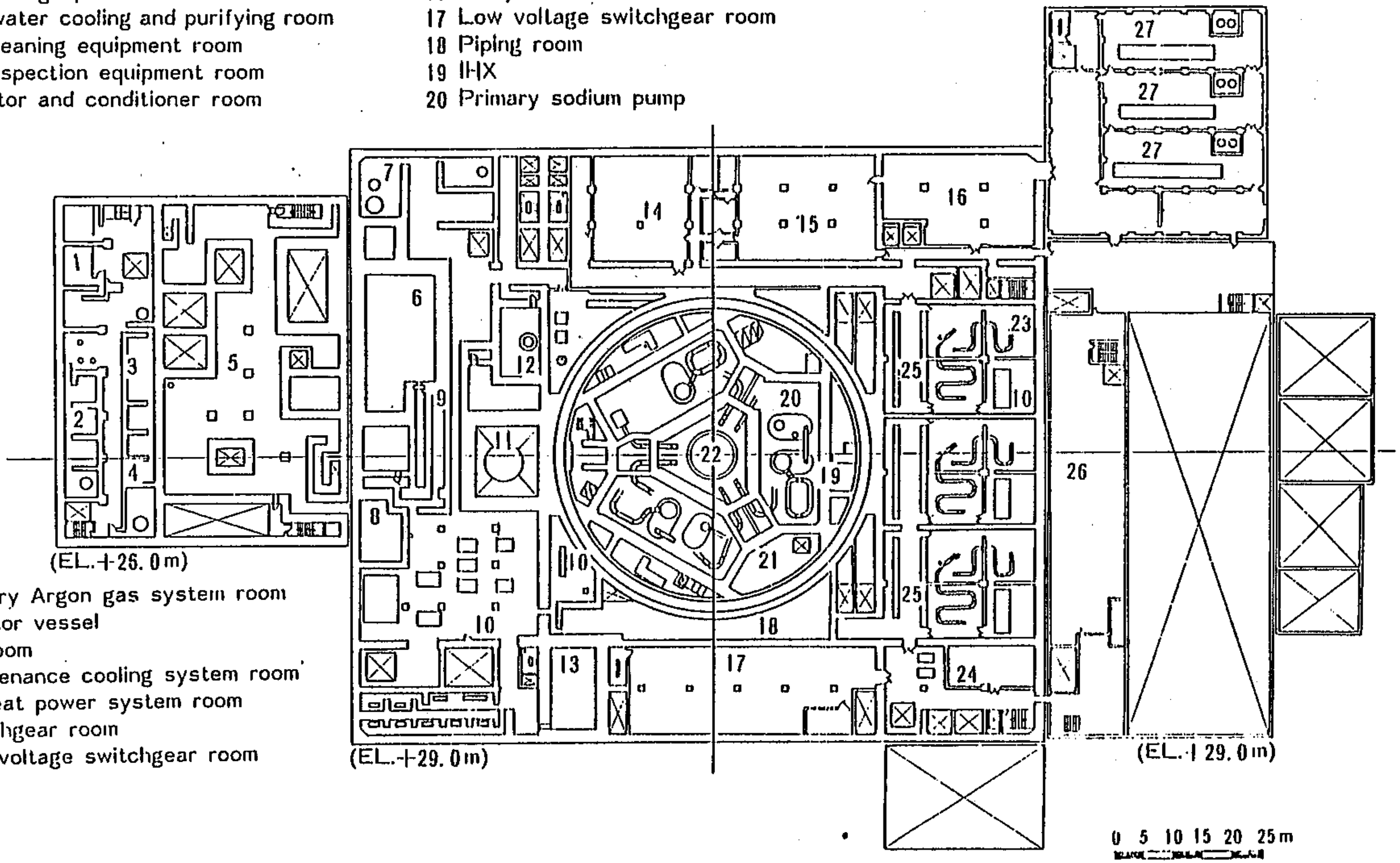




21 IHX
22 Dump tank
23 Storage tank
24 Secondary sodium purification system room
25 Super heater
26 Evaporator
27 Reaction product storage tank
28 Switchgear room
29 Crane
30 Turbine generator
31 Transformer area

Sectional View of Main Buildings

1 Spent resin tank room
2 Concentrated miscellaneous waste tank room
3 Monitor tank room
4 Miscellaneons waste collector tank room
5 Components cleaning area
6 Fuel storage pool
7 Pool water cooling and purifying room
8 Fuel cleaning equipment room
9 Fuel inspection equipment room
10 Ventilator and conditioner room
11 EVST room
12 Cold trap room
13 FFDL room
14 Fuel handling facilities operation room
15 Main control room
16 Relay room
17 Low voltage switchgear room
18 Piping room
19 IHX
20 Primary sodium pump



21 Primary Argon gas system room
22 Reactor vessel
23 SG room
24 Maintenance cooling system room
25 Preheat power system room
26 Switchgear room
27 High voltage switchgear room

Horizontal Cross Section of Main Buildings

5.3 Demonstration Reactor

Long Term Program for Development and Utilization of Nuclear Energy

Government (1) Suppont operation and construction
(2) Establish regime of develepmant for demonstration FBR
(3) adopt appropriate measures for expediting siting work

P N C (1) Promote mainly related research and development
(2) Provide the data for establishing safety design criteria
(3) Provide the data for safety review
(4) Cooperate with the private sector for education and training of the personnel operating the plants

Utilities (1) Take positive role for construction and operation
(2) Work to secure suitable sites for construction
(3) Promote related research and development.

Manufactures (1) Establish responsible manufactures who can manage construction of plant
(2) Promote related research and development

October 23, 1984

The interim report on the development of the fast breeder reactor

——by the advisory committee under the Japan Atomic Energy Commission——

THE MAIN POINT

I. BASIC PHILOSOPHY

(1) • PROMOTE EARLY COMMERCIALIZATION OF FBR.

(2) • TRASITIONAL PHASE TO COMMERCIALIZATION FOLLOWS AFTER DEMO-PLANT PROJECT WHICH SUCCEEDS PROTOTYPE "MONJU".

(3) • DEVELOPMENT BY DOMESTIC TECHNOLOGY OF PNC AND OTHERS WITH INTERNATIONAL COOPERATION.

- PROMOTE TO TRANSFER PNC'S TECHNOLOGY TO PRIVATE SECTOR
- IMPROVE ECONOMIC EFFICIENCY KEEPING SAFETY AND RELIABILITY.

(4) • PROMOTE DESIGIN, CONSTRUCTION, & OPERATION AS WELL AS R&D OF DEMO PLANT

- ESTABLISH SAFETY DESIGN CRITERIA.
- PROMOTE BASE TECHNOLOGY

(5) • COOPERATION OF GOVERNMENT AND PRIVATE SECTOR IS ESSENTIAL, NEAR TERM PROJECT IS DEFINED.

(6) • TARGET START OF CONSTRUCTION YEAR IS EARLY 1990'S

2. NEAR TERM PROGRAM OF THE DEMONSTRATION REACTOR DEVELOPMENT

(1) NEAR TERM PROGRAM ARE AGREED AS FOLLOWS

① UTILITIES LEAD CONCEPTUAL DESIGN WORK, WITH SUPPORT OF MANUFACTURERS, UNDER COOPERATION WITH PNC TO REFLECT R&D.

② PNC LEADS CONCEPTUAL DESIGN RELATED R&D (INCLUDING DESIGN STUDY), WITH SUPPORT OF MANUFACTURERS, UNDER COOPERATION WITH UTILITIES.

EFFECTIVE UTILIZATION OF PNC'S LARGE TEST FACILITIES SHOULD BE PLANNED

PNC LEADS R&D ON BASIC AND COMMON TECHNOLOGIES.

③ SAFETY DESIGN CRITERIA ARE TO BE ESTABLISHED.

④ UTILITIES PREPARE FOR THE ESTABLISHMENT OF RESPONSIBLE BODY, AND FOR THE SITE SELECTION.

(2) FOR THE NEAR TERM, R&D TO DETERMINE PRINCIPAL SPECIFICATIONS WILL BE PROMOTED.

PRICIPAL SPECIFICATIONS, CONCLUDED BY UTILITIES ARE CHECKED AND RIVIEWED BY JAEC TO DETERMINE NATION'S PROGRAM.

3. FUEL FABRICATION AND REPROCESSING

(1) PNC HAS STARTED R&D ON MOX FUEL

(2) SUBCOMMITTEE STUDIED ON REPROCESSING.
IMPORTANCE OF DOMESTIC TECHNOLOGY SHOWN.
PNC'S PRESENT TECHNOLOGY IS CAPABLE, BUT COORDINATION WITH REACTOR SIDE NEEDED.

4. INTERNATIONAL COOPERATION

TOO EARLY TO JOIN JOINT CONSTRUCTION PROGRAM.
BILATERAL COOPERATION IS REALISTIC.
MULTINATIONAL COOPERATION DESIRED FOR SAFETY PHILOSOPHY AND CRITERIA

SHORT TERM PROGRAM REGARDING ASSIGNMENT AND COOPERATION FOR RESEARCH & DEVELOPMENT IN DEVELOPMENT OF DEMONSTRATION FBR

PRESENTED TO THE ADVISORY COMMITTEE ON FBR DEVELOPMENT (JAEC)

December 9, 1983

Power Reactor & Nuclear Fuel Development Corp.
Federation of Electric Power Companies

To promote the development of the demonstration FBR (hereafter referred to as "demonstration reactor"), as indicated in the "Long Term Program for Development & Utilization of Nuclear Energy" of 1982, "the Federation of Electric Power Companies (FPEC) will take a positive role with the support of the government" and "related research ar.d development (R & D) will be made mainly by the power Reactor and Nuclear Fuel Development Corporation (PNC), in parallel with the growing role of the private sector."
From these points of view, considering the present transient situation to establish a development organization, assignment and cooperation for the current R & D are arranged.

1. R & D of the Demonstration Reactor

A goal of the demonstration reactor is to demonstrate techniques regarding reliability and safety of the commercial-scale power plant, and simultaneously to have an economical perspective for the commercialization. In the development of the demonstration reactor, R & D aiming at 3 fields with countermeasures,
for "making sure reliability and safety,"
for "capacity-up" in accordance with the increased output,
and
for "improvement of economical efficiency"
is necessary to be conducted, utilizing experiences hitherto and many results of R & D as well as international cooperation efficiently, in order to attain the goal of demonstration reactor specified above.
The development of the demonstration reactor is divided into 2 phases, i.e. the 1st phase includes until provision of the basic specification and the 2nd phase follows hereafter.

(I) Items of R & D in the 1st phase are composed of that regarding development of the basic technique and that required at selection of the design concept.

The former is a basic general purpose R & D item including a study of technique improvement, which is obtained by extrapolation of techniques accumulated since the time when experimental reactor was started, to be applied for development of the demonstration reactor.

The latter is an item to study compatibility of new technique and design concept in order to provide the basic specification for the demonstration reactor.

(2) In the 2nd phase, in compliance with the decided basic specification for the demonstration reactor, the related practical R & D, e.g. performance test of trial product and confirmation test of system compatibility, are conducted.

2. Basic Idea about Assignment and Cooperation for R & D

Regarding R & D in the 1st phase of the transient period, basic idea about assignment of role and cooperation toward PNC and Electric power companies is as follows :

(1) PNC, as the major responsible parent body in R & D of the demonstration reactor, plays a main role assigned to the comprehensive planning and adjustment of R & D based on experiences of the experimental reactor and the prototype reactor, and conducts R & D including related to fuel cycle with the cooperation of makers,

For the time being, arrangement of standards, data and calculation codes necessary to evaluate safety, research of development and modification of equipment system based on experiences of construction, operation and maintenance of FBR, as well as arrangement of large-scale test facility are conducted.

Regarding the research of design, participating in the concept design plan conducted by Electric power companies, PNC simulteneously studies various kinds of design including survey with wide-range parameter and their optimization, based on the results of R & D andexperiences of plant operation, to be able to give appropriate technical support on the occasion of evaluation and selection of the basic specification for the demonstration reactor.

(2) As for Electric power companies , to play a prime role for construction, operation and maintenance of the demonstration reactor and to fulfil the responsibility, concept design of the demonstration reactor and the related preliminary study with the cooperation of makers as well as participation in program for development of the demonstration reactor in regard to pick-up of requisite R & D item and its orientation are conducted.

In the 2nd phase, the period after the basic specification is provided, since main frame of the business and construction plan etc. are clarified, appropriate assignments of R & D among PNC, governmental research organizations, Electric power companies and makers with proper support of the government will be decided and achieved.

In order to paly the roles specified above effectively, a close cooperative relationship between PNC and Electric power companies is established and the plan of development for the demonstration reactor and proceeding of the implementation are discussed.

An Idea regarding Assignment of the Current R & D for the Demonstration FBR

In the 1st phase, the period until the basic specification is provided, indicated by the "Short Term Program regarding Assignment and Cooperation for R & D in Development of Demonstration FBR", considering that various factors, such as investment and organization etc., of the demonstration FBR are transient, the prime themes to be resolved in R & D fields are assigned as follows :

	Classification	PNC	FPO
(1)	R & D of risky new technique	◎	○
(2)	Arrangement of large scale equipment and test facility	◎	○
(3)	Succession and development of experiences hitherto and extended result	○	
(4)	Arrangement of basic policy of safety design and standards, and offer of various data necessary to safety inspection	○	○
(5)	Arrangement of bbasic for development of analysis codes	○	
(6)	Development of technique in common with the light water reactor		○
(7)	Improvement of operation and maintenance feasibility	○	◎
(8)	Manufacturing method for economical improvement and new construction method	○	◎
(9)	Establishment of teqhnicue regarding the quality assurance, such as inspection technique to improve reliability	○	○

Note : ◎…… major assignmented

REGARDING CURRENT R & D OF DEMONSTRATION REACTOR

PRESENTED TO THE ADVISORY COMMITTEE ON FBR DEVELOPMENT (JAEC)

April 20, 1984

Power Reactor & Nuclear Fuel Delopment Corp.
Federation of Electric Power Companies

Introduction

The basic idea of R & D of demonstration reactor was presented in the information, "An Idea regarding Assignment of the current R & D for the Demonstration FBR" (information at the 4th conversational meeting of FBR development, Dec. 9, 1983), which is proposed after discussion between PNC and FEPC, but details will be talked elsewhere.

In December 1983, the FBR liaison conference was organized for smooth proceeding of the demonstration reactor by FEPC and PNC and current R & D necessary to the demonstration reactor is discussed and materialized as follows.

1. Basic Idea of the Discussion

(1) Period

Aiming that provision of the basic specification is feasible in 1987, period of the current 3 years (1984 through 1986) is discussed.

(2) Subject

Current R & D necessary until the basic specification is provided are divided into,

1) item necessary to select the design concept, and

2) item of basic technique.

In this discussion, 1) was chosen and summarized deliberatively between FEPC and PNC.

(3) Proceeding

Current R & D proceeds in order to provide the basic specification smoothly as follows :

Plan & adjustment — Charge : FEPC／PNC

- R & D necessary to selection of the design concept is discussed and planned.
- Proceeding of R & D is adjusted according to each fiscal years of requirements.

Ⅰ. Concept design
Ⅱ. Evaluation research
Ⅲ. Test research

Charge : FEPC／PNC／CRIEPI／makers (in joint research)

• Refer to the attached Table

Overall evaluation   Charge : FEPC／PNC

Following items are evaluated as a whole :

• Proceeding of R & D in Ⅰ through Ⅲ above specified

• Draft of the basic specification for basic design

2. Result

Essential items necessary in these 3 years are classified, arranged and shown in the attached Table where result of discussion regarding assignment is described.

3. Proceeding Hereafter

Based on the summary, materialization of the plan is scheduled according to the individual assignments and liaison meeting between FEPC and PNC is conducted.

Attached Table

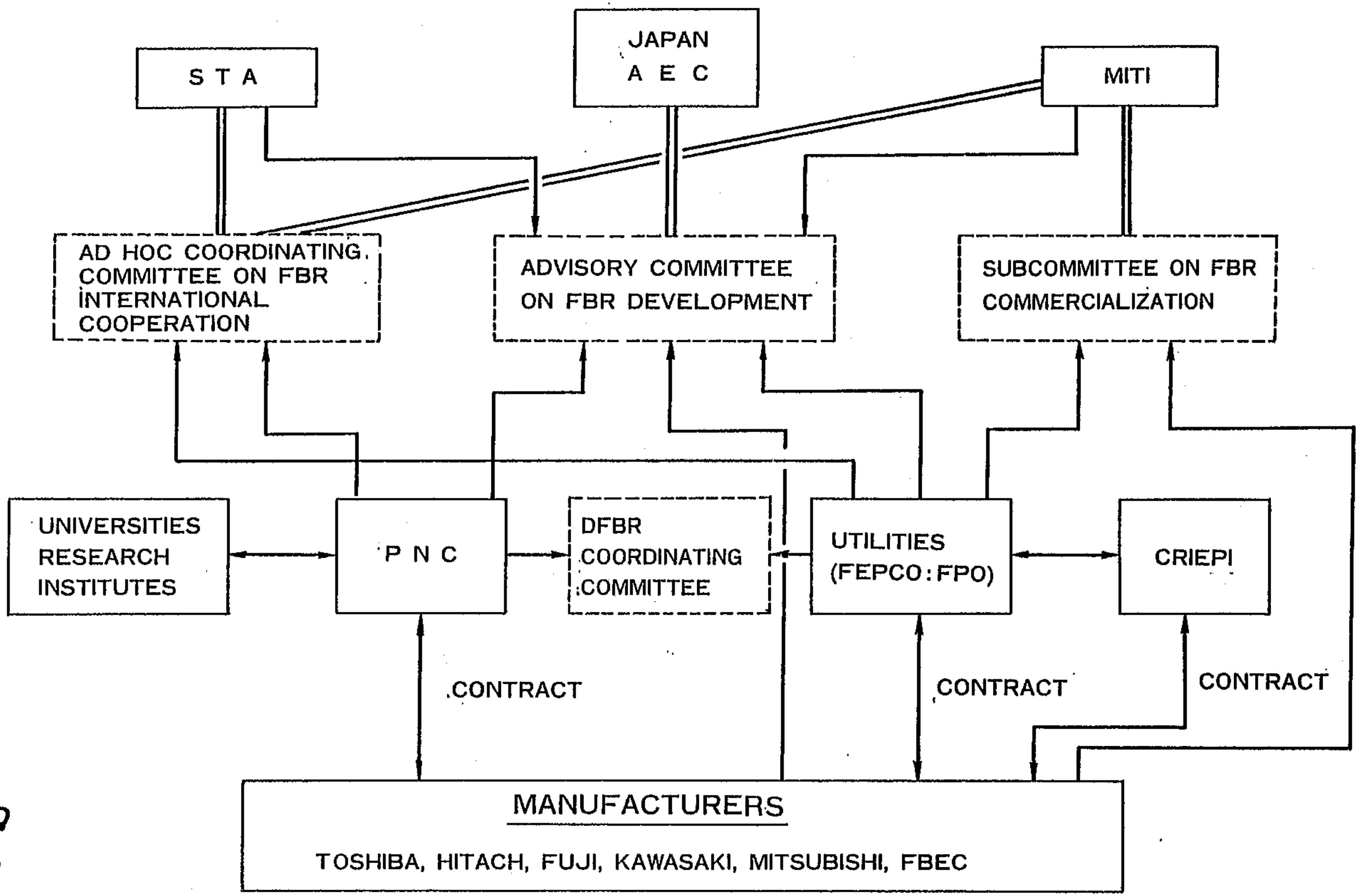
R & D Items Necessary to Selection of Design Concept

Item	Assignment
I. Concept design	
1. Rationalized design of loop and tank reactor	FEPC *
II. Evaluation research	
1. Study on various kinds of design including survey with wide-range parameter and their optimization	PNC
2. Economical evaluation	
(1) Study on estimation of construction cost regarding reactor-type evaluation	FEPC
(2) Study on economical evaluation including nuclear fuel cycle	PNC
3. Adjustment of safety design standards	FEPC. PNC
4. Information collection of experiences for design, construction, operation and maintenance	FEPC. PNC
III. Test research	
1. Research of the tank reactor	
(1) Rationalization of earthquakeresistant structure and improvement of structural reliability	CRIEPI *
(2) Miniaturization of equipment and evaluation of flow & heat transfer inside reactor	CRIEPI *
(3) Reliability of structure and material	CRIEPI *
2. Research of the loop reactor	
(1) Technique of shortening pipings * *	PNC
3. Common research of the both reactors	
(1) Shut-down device for new type reactor	PNC
(2) Direct cooling system for reactor core	PNC
(3) Structure of intermediate heat exchanger	PNC
(4) Axial heterogenious reactor core	PNC
(5) Rationalization of allowances for shieling design	PNC
(6) Preparation of test facility for assurance of major equipment	PNC

* mark indicates joint research with maker

* * mark includes applicability to 2ndary system of the tank reactor

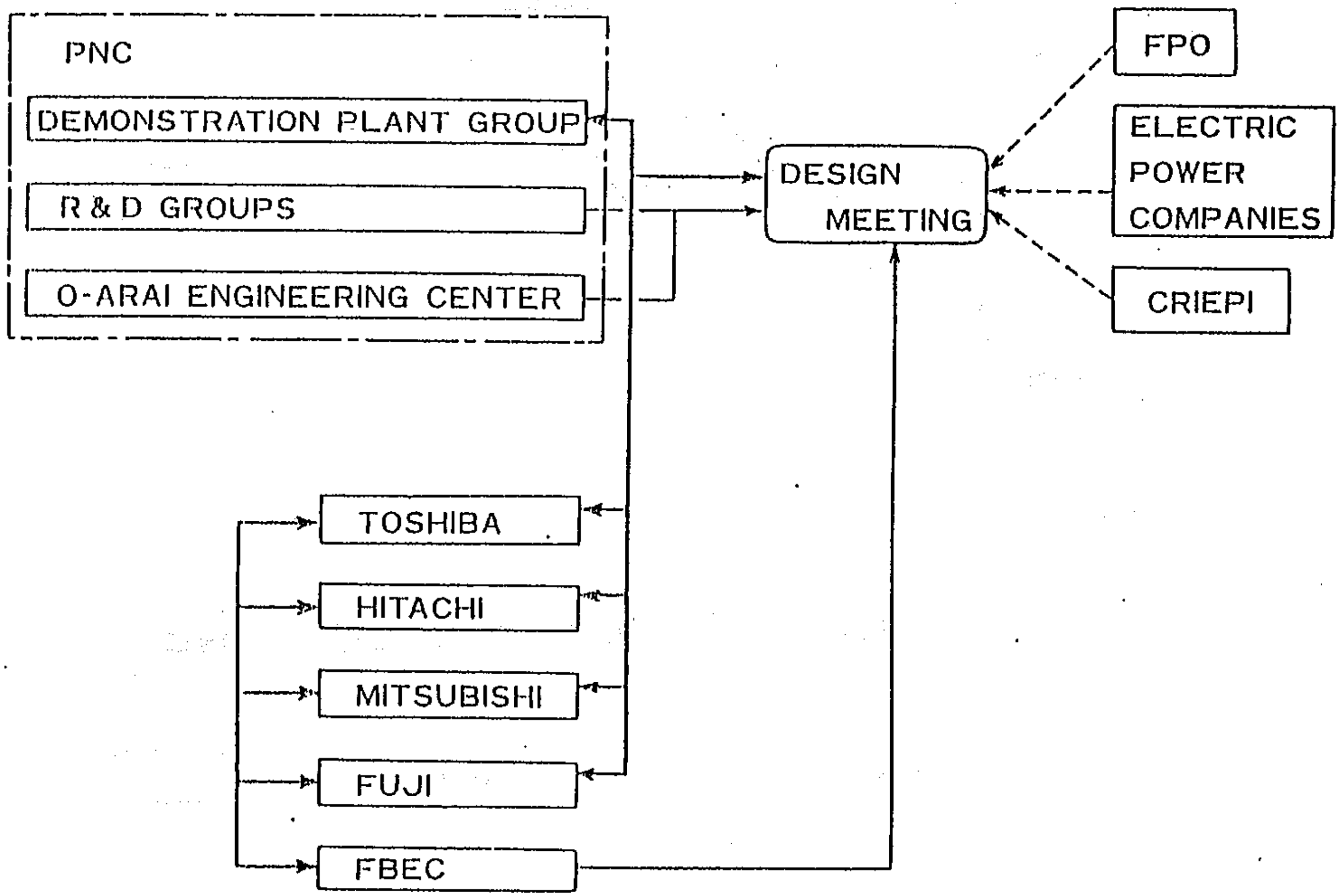
ORGANIZATIONS FOR DFBR DEVELOPMENT (March 1985)



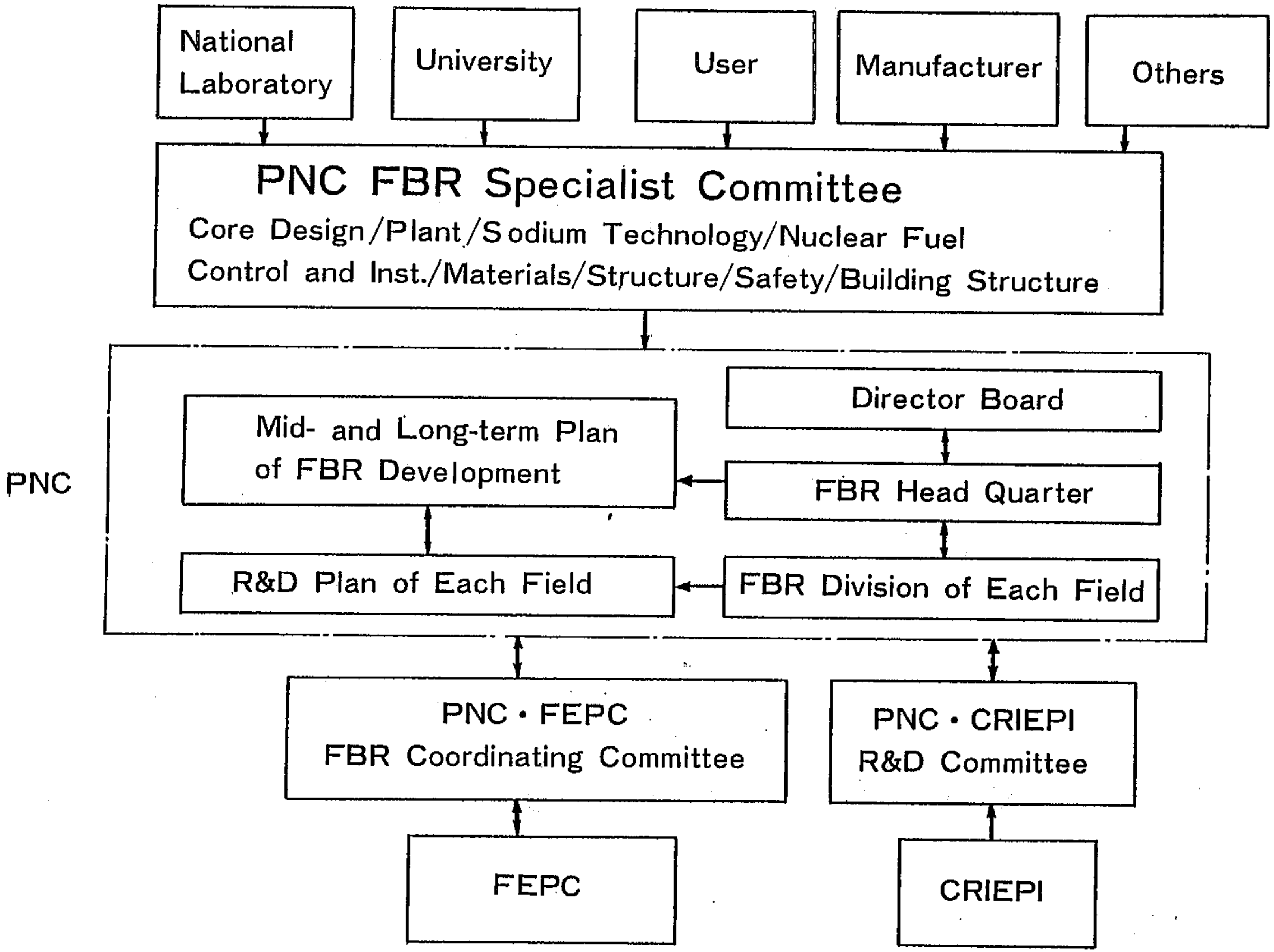
PNC's attitude toward the R & D on Demonstration Reactor

Regarding developmeht of the demonstration reactor, the proceeding was discussed at the Atomic Energy Commission along with the Long Term Program for Development and Utilization of Nuclear Energy (decided by the Atomic Energy Commission on June 30, 1982).
In the 1st phase, R & D necessary to provide the basic specification is conducted from 1984 through 1986 concentrated on the items which are agreed on assignment and cooperation of R & D with Electric power companies.
In the 2nd phase, on and after 1987, R & D necessary to construction and operation (in around 2000) of the demonstration reactor along with the basic specification is conducted through overall evaluation of the basic specification and check-and-review by the Atomic Energy Commission.
A prime object of R & D for the demonstration reactor is economical efficiency because of its high construction cost compared to the light water reactor. Reflection and thorough analysis of results for the prototype reactor is conducted so as to contribute to capacity-up and efficiency-up of the plant. policy-making of safety design exhibiting the characteristics of FBR and corresponding new concept design of plant system and equipment are proposed and test research is conducted to confirm them.
For the time being, besides supporting research jointly with Electric power companies and discussion of necessary and important themes in order to optimize design of the large-scale reactor by extrapolating the experiences obtained by the prototype reactor, system evaluation research to obtain knowledge for reactor-type comparison, new reactor-core concept, and test research of reliability of equipment system and material, and furthermore, preparation of large-scale test facility, are conducted.

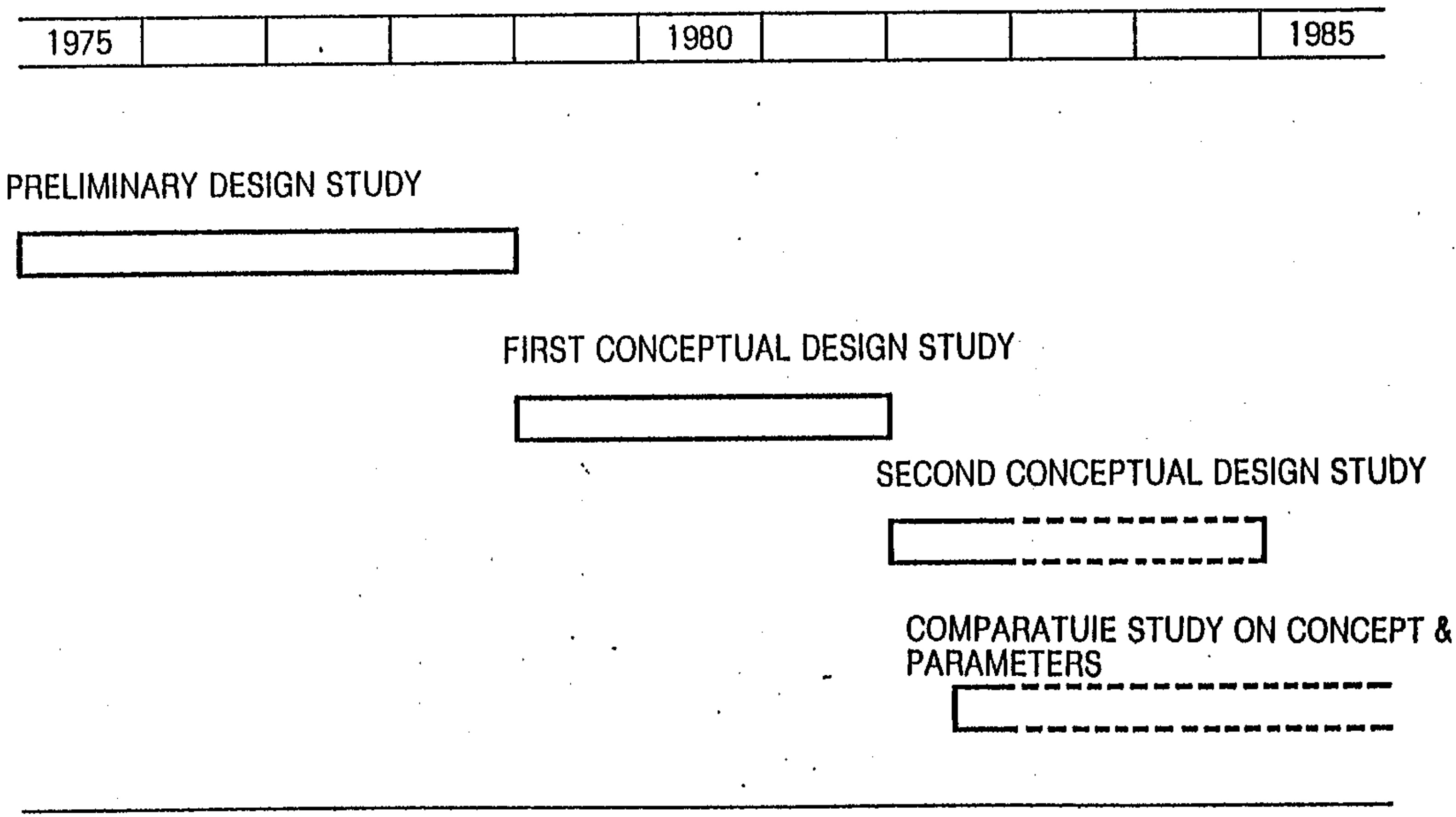
ORGANIZATION OF DEMONSTRATION PLANT DESIGN STUDY



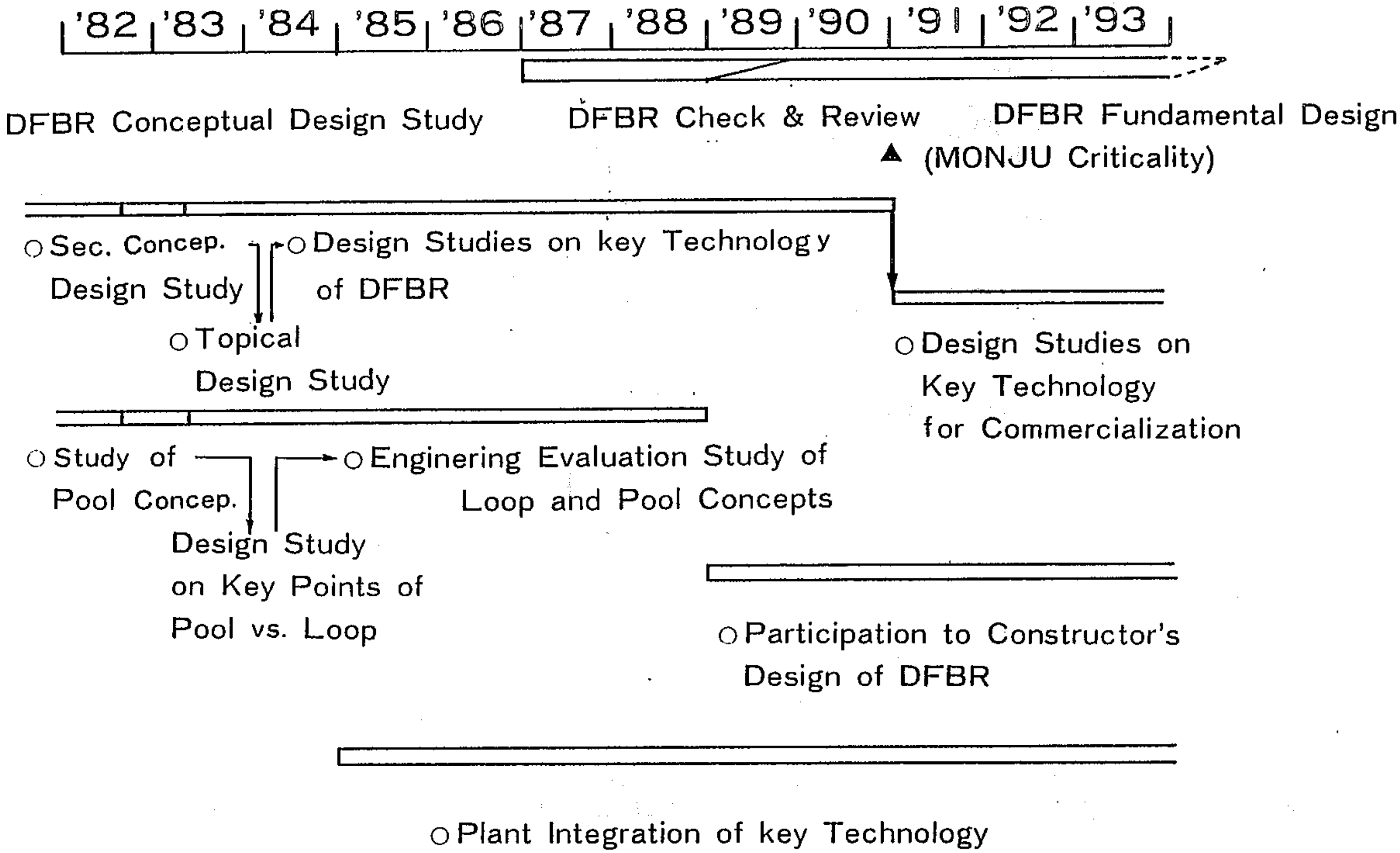
ORGANIZATION OF R&D PLANNING



TIME TABLE OF DESIGN STUDY FOR DFBR



SCHEDULE OF PNC DESIGN STUDY



Overall Flow Regarding Research of Design for Large-scale Reactor

throuqh 1983 fiscal year	1984 fiscal year	1985 fiscal year	1986 fiscal year
the 1st phase concept design (1979-1981) Study on feasibility of large-scale plant by applying conventional pipings **the 2nd phase concept design (1982-1983) (research of design for the demonstration reactor)** Study on feasibility of plant aiming at low construction cost by applying bellows-pipings	**Research of key technology design (I)** Study on elemental technique which has perspective regarding realization and restriction of extended tech nique, based on the independently developed result up to "JOYO" and "MONJU". Selection of R & D and analysis of the result Study on the plant image to adjust tebhnique	**Research of key technology design (II)** Study on detailed (I) and supplemental items Study on the plant image to adjust technique	**Research of key technology design (III)** Overall study Study on detailed (II) and supplemental items Study on the plant image to adjust technique
	Research of design for the demonstration reactor Study on partial supplement to items in 1982 & 1983		
Research of design regarding selection of the basic specification Preliminary study on technique necessary to the tank reactor	**Research of design for selection of reactor-type and basic specification (I)** Study on technique necessary to the tank reactor succeeding the previous fiscal year.	**Research of design for selection of reactor-type and basic specification (II)** Study on the tank-type image for comparison Preliminary study on comparison of the loop-and the tank reactor	**Research of design for selection of reactor-type and basic specification (III)** Overall study on comparison of the loop-and the tank reactor

PNC'S NEAR TERM R&D FOR DFBR

- CONDUCT DESIGN STUDY AND RELEVANT TEST AND EXPERIMENT BASED ON EXPERIENCE OF JOYO AND MONJU AND NEW CONCEPTS.

- STUDY FEASIBILITY OF PLANT CONCEPT AND NEW TECHNOLOGIES TO BE EMPLOYED.

- SUGGEST BASIC SPECIFICATION OF DFBR AND GUIDELINES TO COMMERCIALIZATION EFFORT.

- PREPARE LARGE TEST FACILITIES INSTALLATION.

5.4 Research & Development on FBR Base Technology

RESEARCH & DEVELOPMENT ON FBR BASE TECHNOLOGY

1) OBJECTIVES

R & D OF FBR KEY TECHNOLOGY BASED ON A LONG-TERM STRATEGY OF FBR COMMERCIALIZATION

2) THEME

- LONG-TERM STRATEGY FOR FBR R & D (ENERGY SYSTEM, FUEL CYCLE, ECONOMICAL ANALYSIS)
- R & D TO UTILIZE FBR INTRINSIC PROPERTIES
- R & D FOR FBR PLANT SAFETY & RELIABILITY
- FBR DATA BASE, CODES, STANDARDS
- R & D OF NEW TECHNOLOGY, NEW REACTOR CONCEPT

5.5 R&D Activities

5.5 R & D Activities

1) Reactor Physics and Shielding

2) Systems and Components

3) Fuels and Materials

4) Structures & Materials and Sodium Technology

5) Safety

6) FBR Fuel Reprocessing

7) Waste

8) Profile of Major Test Facilities

I) Reactor Physics and Shielding

JASPER Program

(Japanese-American Shielding Program of Experimental Researches)

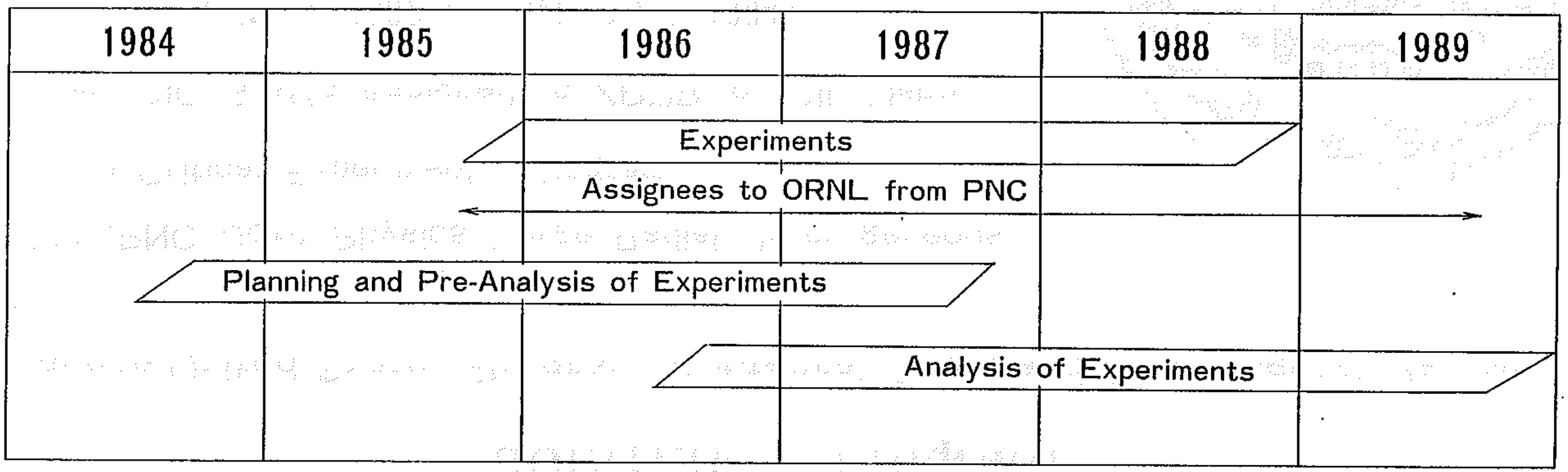
- Goal of Program

  Acquisition of Experimental Data on Large LMFBR Shielding
  Improvement of Shielding Design Method
  Prediction of Shielding Design Accuracy
  Optimization of Shielding Design

- Experiments

  DOE / PNC Joint Experiments Program using TSF (Tower Shielding Facility) in ORNL

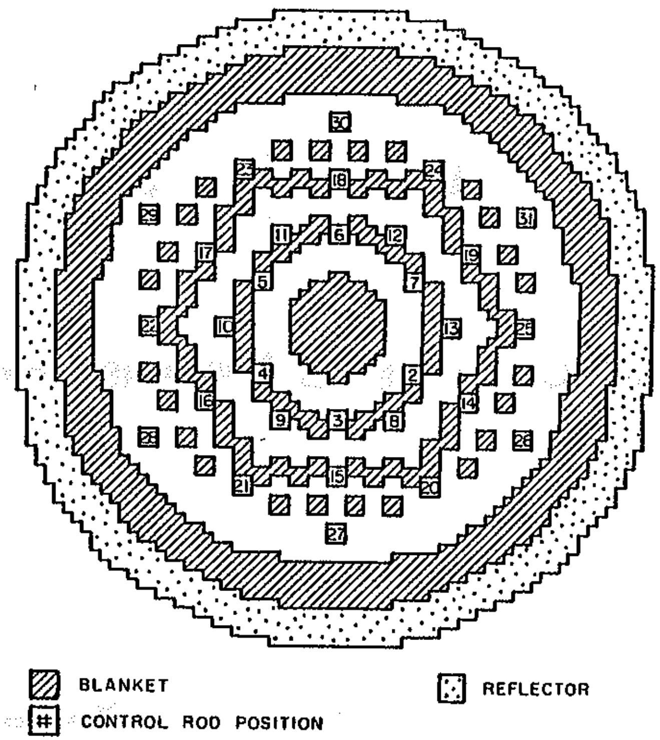
- Tentatively Expected Time Schedule



JUPITER-Ⅱ Program

(Japanese-United States Program of Integral Tests and Experimental Researches)

- DOE / PNC Joint Physics Large Radial Heterogeneous Core Critical Experiments Program
- Three PNC Staffs Assigned to ZPPR in ANL-Idaho
- Experiments (May, 1982 through April, 1984)
  Six Assemblies (ZPPR-13A,B/1, B/1, B/2, B/3, B/4, C)
  650 MWe-Size Radial Heterogeneous Benchmark Cores
  Central Blanket and Three Fuel Rings
- Analysis: With ENDF/B-IV (U.S.) and JENDL-2 (Japan)
  The First Joint Analysis Meeting was held on September 11-14, 1984 at ANL-Idaho. 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.



ZPPR-13B/4

JUPITER-Ⅲ Program

○ DOE / PNC Joint Physics Benchmark Experiments Program
for Axial Heterogeneous LMFBR Cores, and
for 1000～1300 MWe-Size Homogeneous LMFBR Cores

○ Under Negotiation Between DOE and PNC

○ Preliminary Proposed Time Schedule

Planning and Pre-Analysis	:	1985～1986
Experiments	:	1986～1988 (Two Years)
Assignees	:	1986～1989 (Four People)
Analysis of Experiments	:	1986～1989
Joint Analysis Meeting	:	1987～1989 (Two Times)

2) Systems and Components

SYSTEMS AND COMPONENTS (1/2)

○ GENERAL VIEW : MONJU MAJOR COMPONENTS DEVELOPMENT IS COMING TO FINAL STAGE, BUT SOME TESTS NECESSARY TO FULFIL REQUIREMENTS ON CONSTRUCTION PERMIT REMAINS FOR FOLLOW-ON LARGE SCALE PLANT, DEVELOPMENT OF KEY ELEMENTS IS BEING INITIATED.

○ COMPLETED

REACTOR : COOLANT DISTRIBUTION TO CORE, STRUCTURE, SEISMIC TESTS

PRIMARY PUMP : FUNCTION AND PERFORMANCE TESTS UNDER NORMAL CONDITION

INTERMEDIATE HEAT EXCHANGER : FLOW DISTRIBUTION, THERMAL TRANSIENT TESTS

CONTROL ROD DRIVE MECHANISMS : FUNCTION AND RELIABILITY TESTS

FUEL HANDLING SYSTEM : FUNCTION TEST OF IN-VESSEL TRANSFER MACHINE

VALVE : CLOSING CHARACTERISTICS TESTS OF CHECK VALVE, FUNCTION TESTS OF STEAM GENERATOR ISORATION VALVE

STEAM GENERATOR : TESTS ON PERFORMANCE AND RELIABILITY, MAINTENANCE, RUPTURE DISC FUNCTION, WATER LEAK DETECTION SYSTEM, WATER SODIUM REACTION, TUBE PLUGGING TECHNIQUE, IN-SERVICE NONDESTRUCTIVE TUBE EXAMINATION.

SYSTEMS AND COMPONENTS (2/2)

○ CURRENTLY UNDERGOING

- STRUCTURAL INTEGRITY OF MONJU REACTOR VESSEL AT NEAR SODIUM LEVEL
- EFFECT OF THERMAL STRIPING REACTOR UPPER CORE STRUCTURE
- PRIMARY PUMP TEST UNDER OFF NORMAL CONDITION INCLUDING SEISMIC TEST
- STEAM GENERATOR : INTEGRITY OF PLUGGED TUBE, I S I OF TUBE
- CONTROL ROD DRIVE MECHANISMS : RELIABILITY DEMONSTRATION INCLUDING SEISMIC TEST
- IN-SERVICE INSPECTION EQUIPMENT
- COOLING BY NATURAL CIRCULATION IN EX-VESSEL FUEL STORAGE TANK
- FLEXIBLE PIPING JUINT : FATIGUE TEST OF BELLOWS

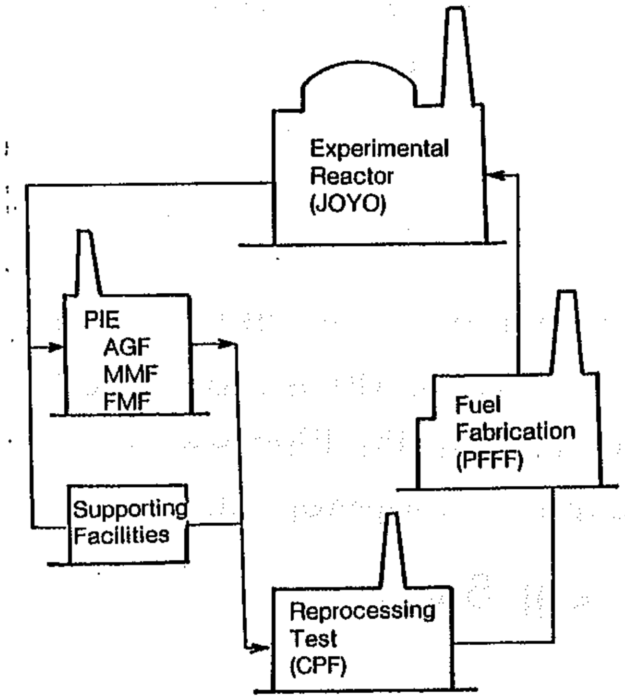
○ FUTURE PLANS

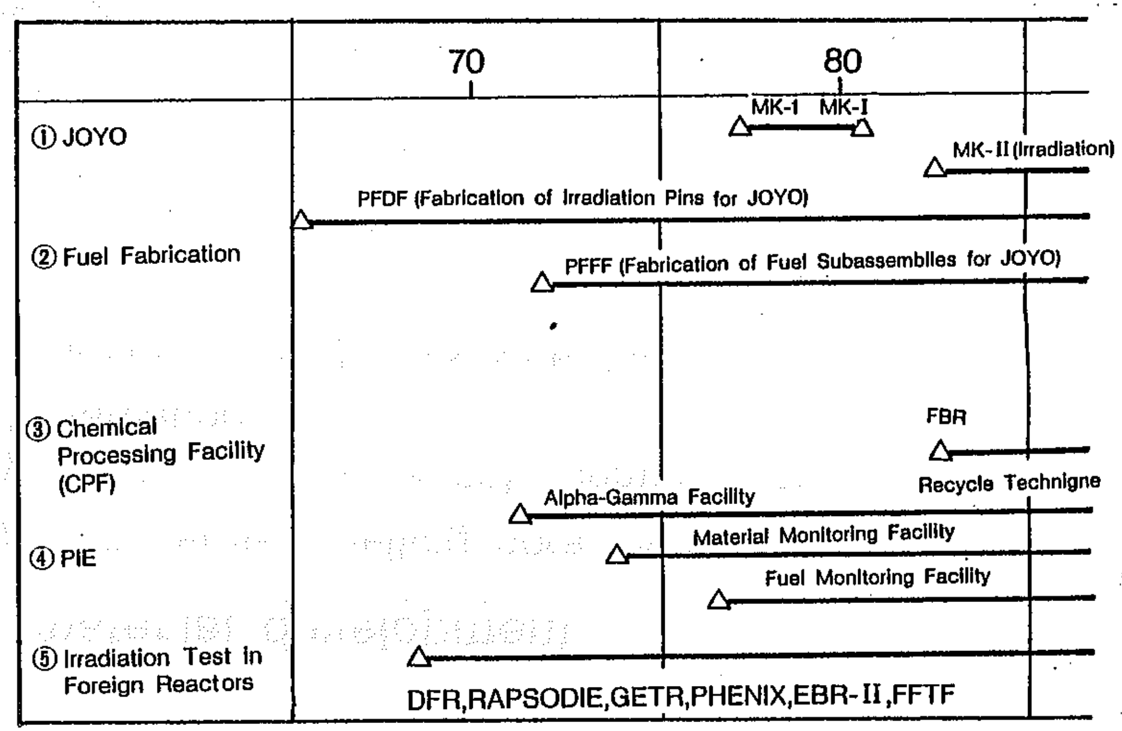
- ADVANCED MATERIAL, INTEGRATED ONCE THROUGH STEAM GENERATOR
- DOUBLE WALLED STEAM GENERATOR
- SELF ACTUATED REACTOR SHUT DOWN SYSTEM
- MITIGATION OF SEISMIC FLOOR RESPONCE INCLUDING BUILDING ISOLATION
- DEVELOPMENT OF PLANT WITHOUT INTERMEDIATE HEAT TRANSFER SYSTEM

3) Fuels and Materials

FBR FUEL DEVELOPMENT STATUS IN JAPAN

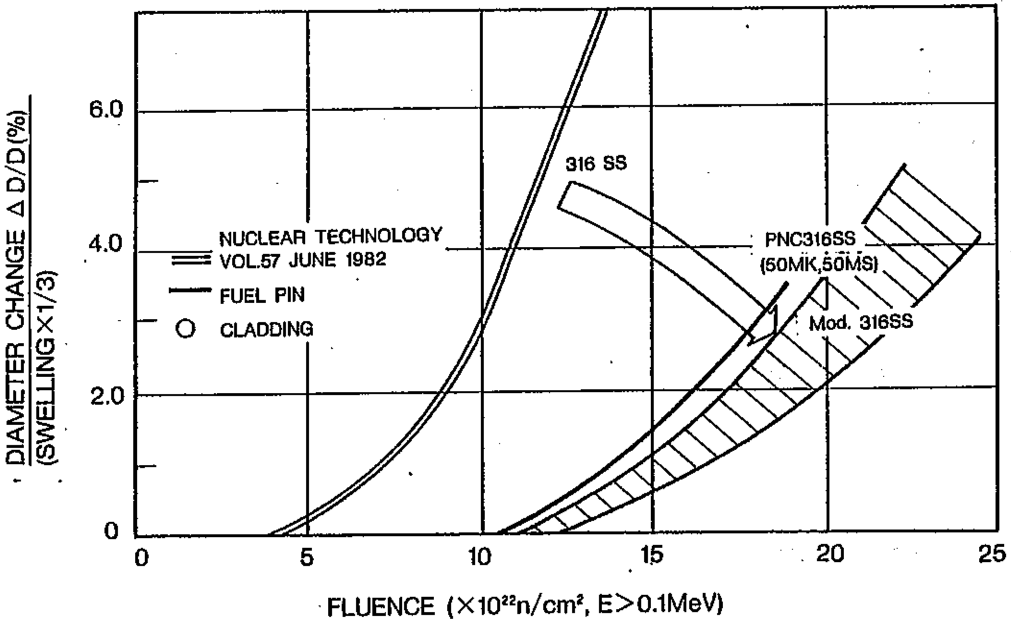
- PNC has constructed all necessary facilities for fuel development and is operating them.
- Fuel development is focused at present on proving the integrity of "MONJU" fuel and to develop long-life core.
- International collaboration is greatly effective in fuel development and PNC will continue it in the future.



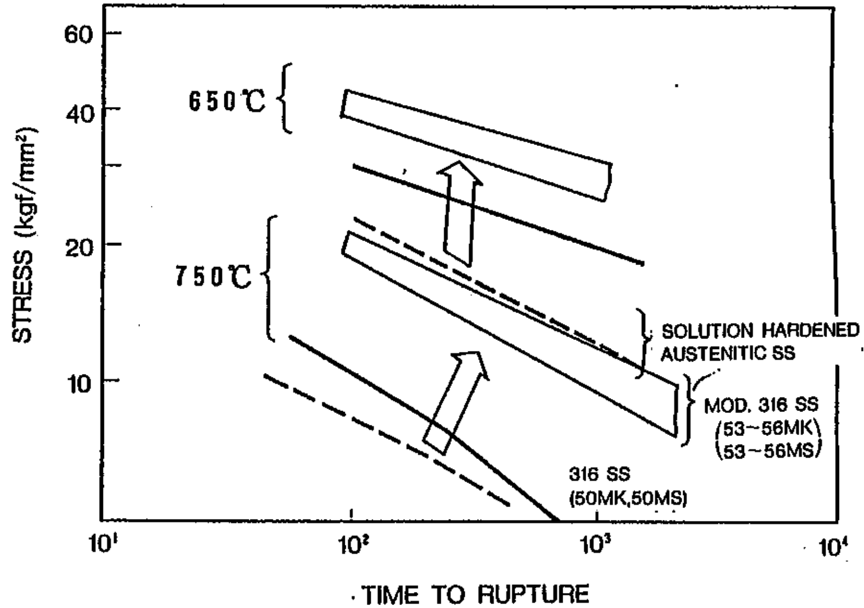


Long life core material development

- PNC has developed high strength, moderate swelling core material
- Core material fabricated at early stage of development demonstrated the excellent performance up to ~2×10[23] nvt irradiation
- PNC continues development of various kind of core materials



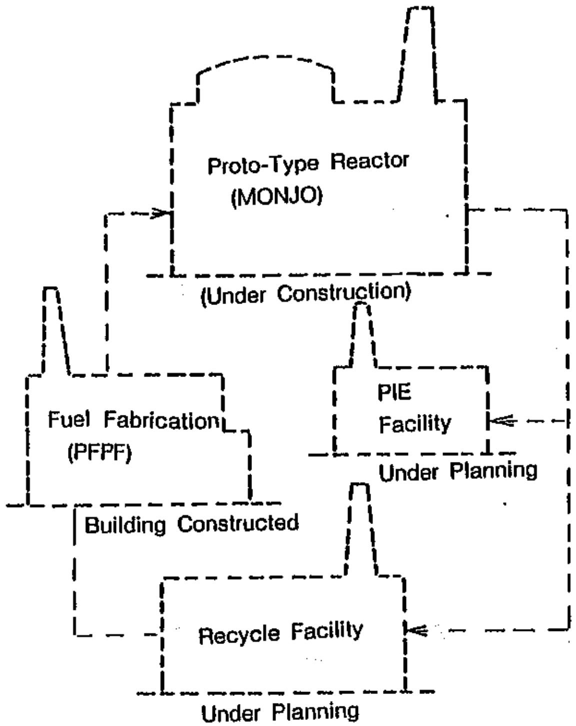
SWELLING RESISTANCE OF PNC'S ADVANCED CORE MATERIALS

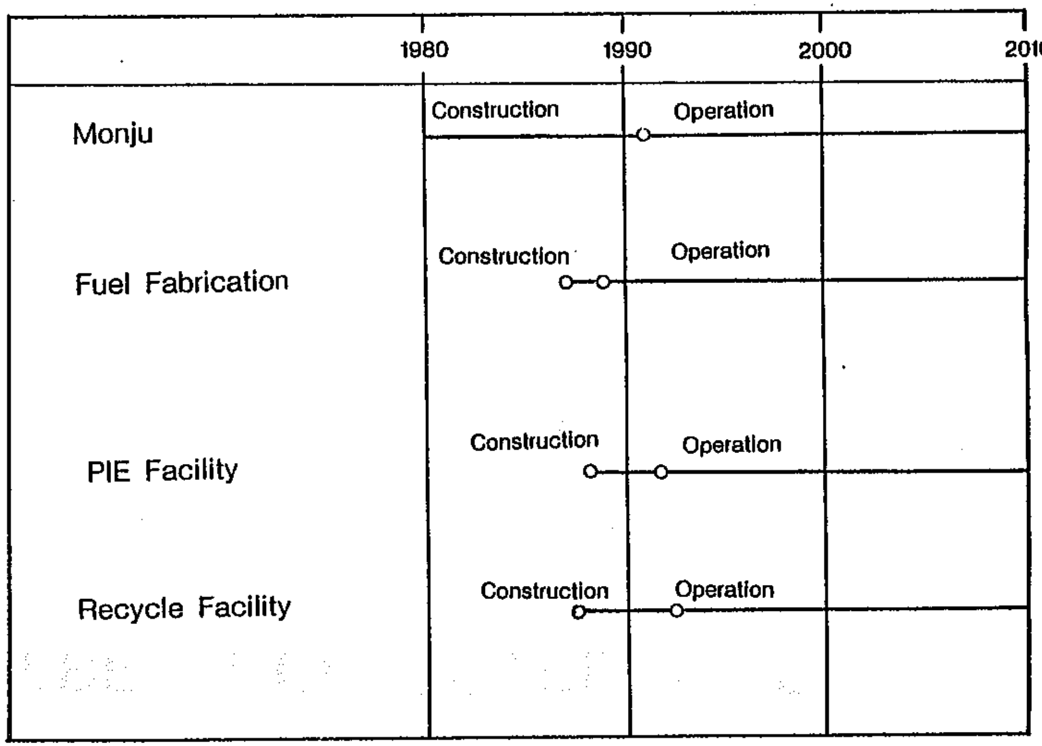


CREEP RUPTURE STRENGTH OF PNC'S ADVANCED CORE MATERIALS

FBR FUEL DEVELOPMENT PROGRAM IN JAPAN

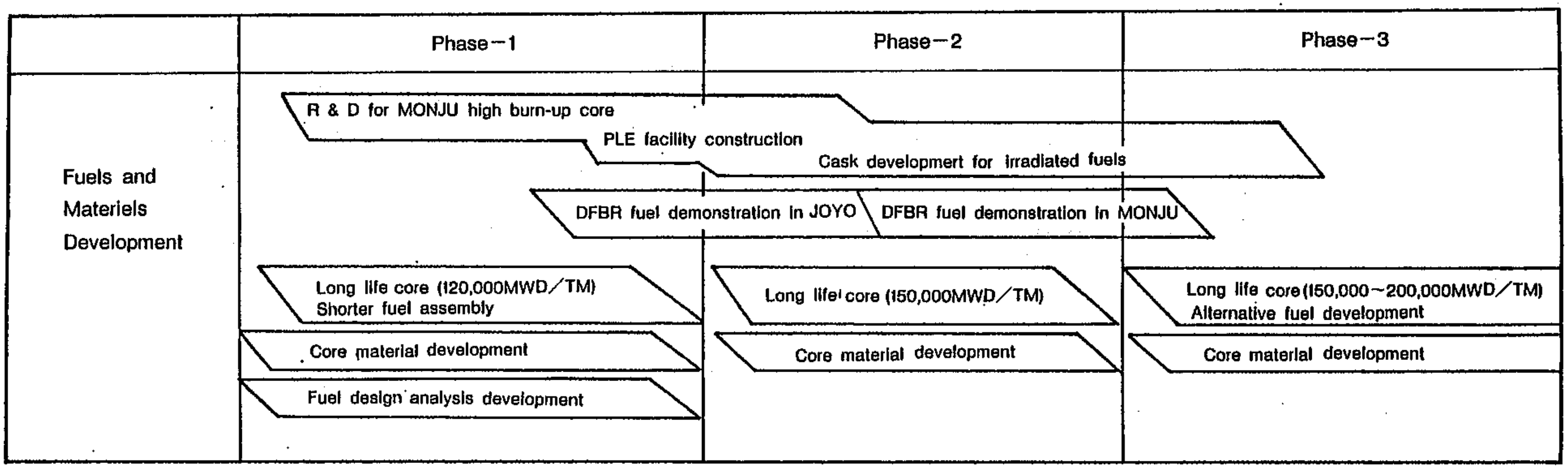
- PNC is to construct and operate new large-scale facilities necessary for developing "MONJU" and DFBR fuel.
- PNC could enhance further active fuel development ability using "JOYO" and "MONJU".
- International collaboration will be continued to effectively proceed common goals of FBR fuel development.





TARGET OF THE FBR FUEL DEVELOPMENT

1. Fuel performance —— long life, high linear heat rate, and safe
   - Core materials with enough resistance to high dosage
2. Rationalization of the fuel design concept
   - Larger fuel pin diameter and shorter fuel assembly
   - Review of the design margin

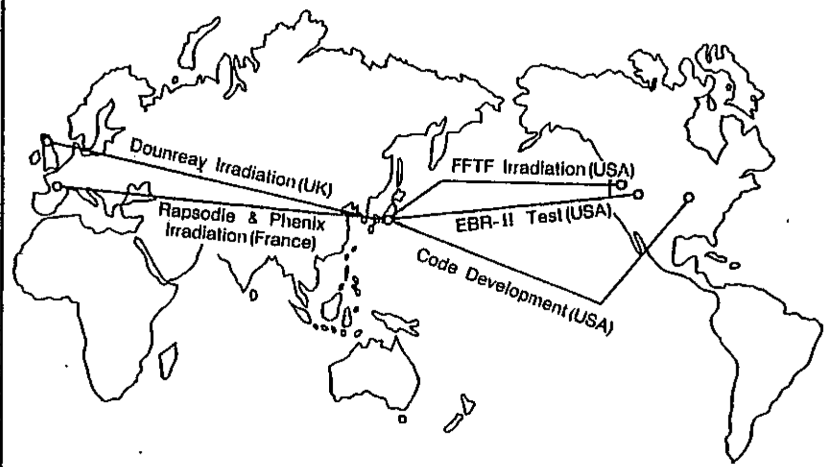


INTERNATIONAL COLLABORATION ON FBR FUEL DEVELOPMENT

Collaboration is greatly effective in realizing the common goal of long life core materials development for FBR through mutual exchanges of experiences and with minimum developmental cost.

PRESENT STATUS

Test Designation	Reactor
Operational Reliability Testing of Oxide Fuel in EBR-II (EBR-II ORT)	EBR-II
FFTF Bundle Porosity Test (FFTF BUND-1)	FFTF
FFTF Creep/Creep-Rupture Test (FFTF CCR-2)	FFTF
Proto-Type Reactor Fuel Irradiation Test (Phenix PNC P3)	Phenix
Material Irradiation Test in Phenix (Phenix PNC P4)	Phenix



FUTURE SCOPE

Test Designation	Reactor
Operational Reliability Testing of Oxide Fuel in EBR-II (Continued)	EBR-II
Collaborative Testings for Long Life Core Materials development	FFTF
Exchange of Irradiations between PHENIX and JOYO	Phenix

4) Structures & Materials and Sodium Technology

STRUCTURES AND MATERIALS DEVELOPMENT

◎Structural materials Test

The "CAPELLA" Program : In-Air, In-Sodium and In-Water/Steam Test

The "SPICA" Program : Neutron Irradiation Test

◎Structural Integrity Test

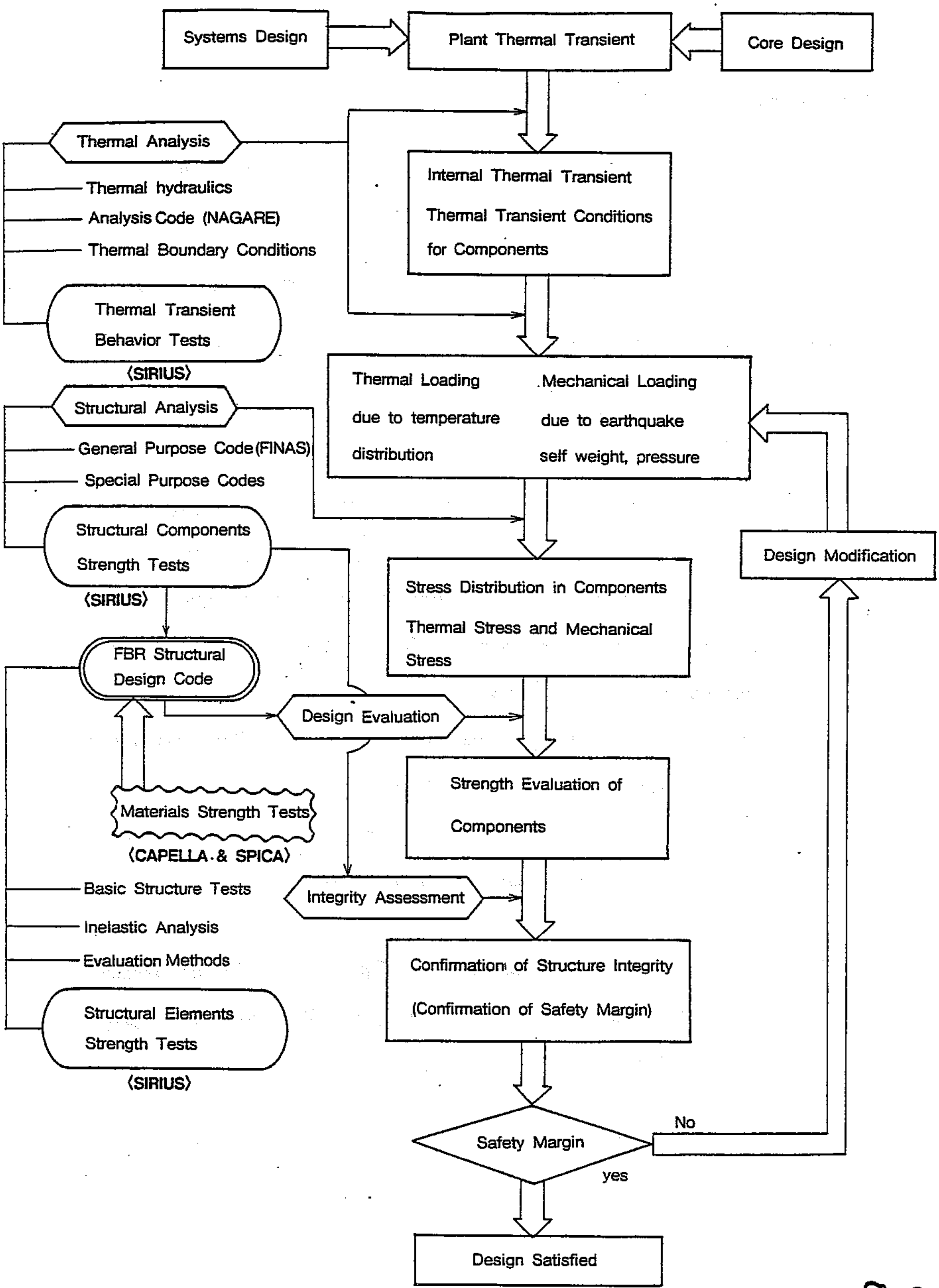
The "SIRIUS" Program

CAPELLA : The reason why the program is designated as the CAPELLA program is as follows. The development of structural materials is one of the most important R&D items necessary to develop the nuclear reactor and leads to the commercial fast breeder reactor. This resembles closely the role of the Charioteer who leads the destination of the Chariot, which corresponds to the nuclear reactor itself.

SPICA : Scheme of Irradiation Test and Post-Irradiation Examination on Candidate Structural materials for FBR.

SIRIUS : Structural Integrity under Severe Loadings.

DESIGN FLOW AND PROGRAM



BASIC CONCEPT OF "THE CAPELLA PROGRAM"

(1) For the Primary Cooling System (SUS304)

◎Reflection and Enlargement of R&D Work for "MONJU"

- Solution of Items Pointed out in the Course of "MONJU" Safety Review and Evaluation
- Improvement of Residual Life Evaluation Method (Creep-Fatigue Evaluation)
- Environmental Effect under Off-Normal Conditions

◎Development of Large Size Forgings and Thick Plates

◎Development of Large Diameter Seamless Pipe

(2) For the Secondary Cooling System (High Cr-Mo Steel)

◎Screening Test for High Cr-Mo Steels

◎Preparation of material Strength Standard

◎Development of Welding Procedure and Strength Evaluation Method of Weldment

◎Environmental Effect (in Sodium and in Water/Steam)

FBR STRUCTURAL MATERIAL TESTS FLOW AND THE TEST ITEMS NECESSARY FOR THE DEMONSTRATION FBR

JOYO

Preparation of the High-temperature Standards with the Literature Both in Japan and abroad

Small Scale Research and Development on Base Metals and Weldments of Type 304 and 316 Stainless Steels and 2 1/4Cr-1Mo Steel

Evaluation Including Environmental Effect, with Backup Research and Development and Surveillance Test during Operation.

MONJU

Development of "MONJU" Material Strength Standard Based on Japanese Material Data, Referring to ASME Code ("MONJU" Design Standard)

Extended Tests Performed for 304, 316, 321SS and 2 1/4Cr-1Mo Base Metals and Weldments

- Basic Material Properties
- Creep-Fatigue Evaluation Method
- Constitutive Equation
- Welding Method, Inspection, Strength Evaluation Tests

Demonstration Reactor

《Improvement of Present Design Standard》

《For Improvement of Reliability and Economy》

R&D for Improvement of present Design Evaluation Methods

- Creep-Fatigue Evaluation (304SS)
- Constitutive Equation (304SS)
- Strength Evaluation of Weldments (304SS)

R&D for Modified Materials

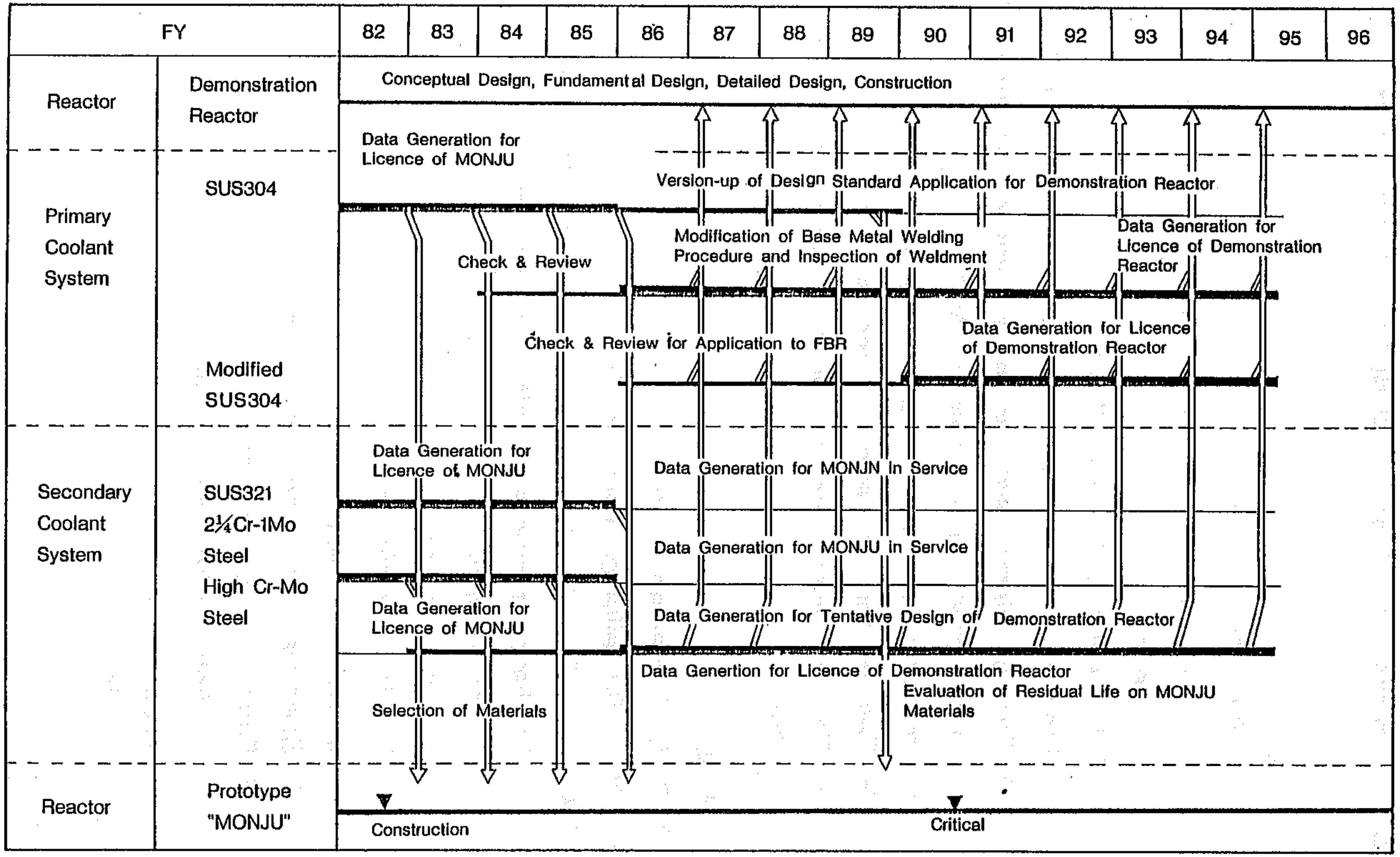
- High Cr-Mo Steel
- Modified 304 Stainless Steel
- Large Diameter Seamless Pipe Material (304SS)

《For Larger Size》

R&D for Larger Stainless Steel Products

- Improvement of Base Material
- Welding Procedures, Inspection Techniques, Strength Evaluation Tests

SCHEDULE ON THE CAPELLA PROGRAM



R&D Items of the "SPICA" Program

◎For "JOYO"

- JOYO Surveillance Test
- JOYO Surveillance Confirmation Test (for unirradiated material)
- JOYO Surveillance Back-up Test
  - (1) Irradiation of the reactor vessel materials in JMTR.
  - (2) Thermal history tests on reactor vessel materials (Part 1)
  - (3) In-pile creep rupture test in JMTR.
  - (4) Irradiation of reactor vessel materials in "JOYO".
  - (5) Test on the size effect of test specimens.
  - (6) Thermal history test on the core supporting plates.
  - (7) Thermal history test on the reactor vessel materials (Part 2)

◎For "MONJU"

- MONJU Surveillance Test
- MONJU Surveillance Back-up Test
- R&D Test (Test for the Large Scale Reacter Included)
  - (1) Irradiation temperature effect test on material strength characteristics.
  - (2) In-pile creep test.
  - (3) Irradiation effect test on SUS304 forged materials.
  - (4) Irradiation effect test on Inconel 718.
  - (5) Specimen size effect test (in-air).
  - (6) Fracture toughness test.
  - (7) Irradiation effect test on welded joints.
  - (8) Tests on neutron fluence.
  - (9) Irradiation effect test on modified SUS304.
  - (10) Irradiation test of high Cr-Mo steel

FIG IRRADIATION TEST SCHEDULE OF "MONJU" STRUCTURAL MATERIAL IN JOYO MK-II

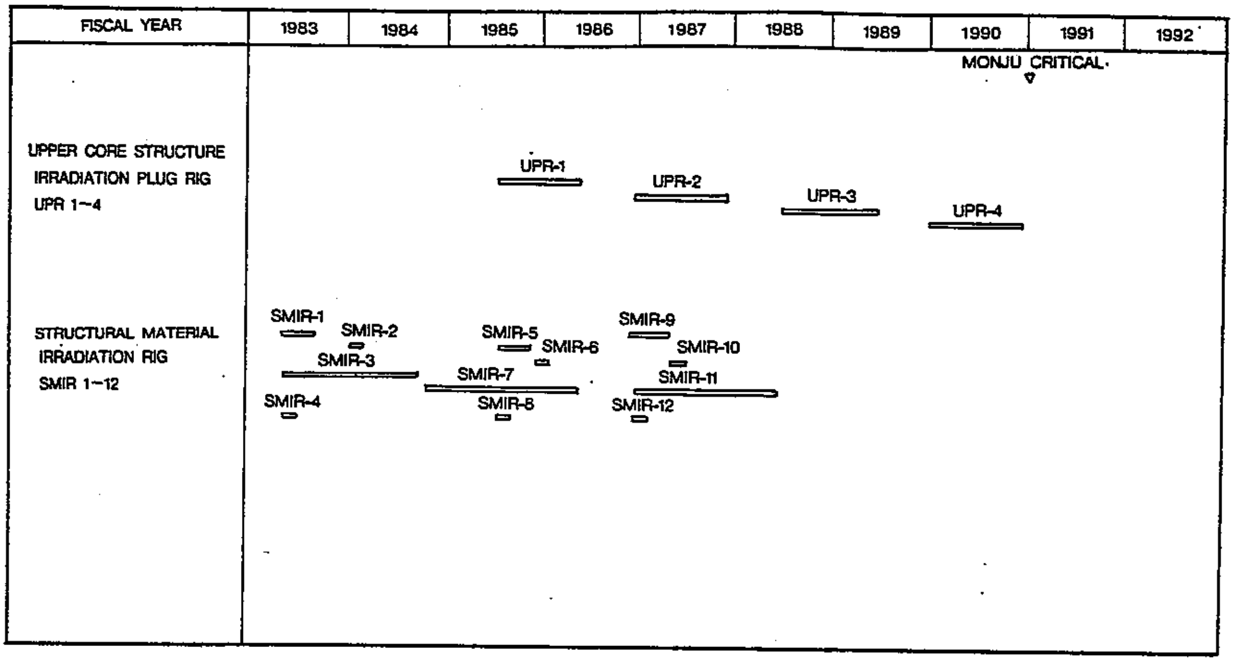
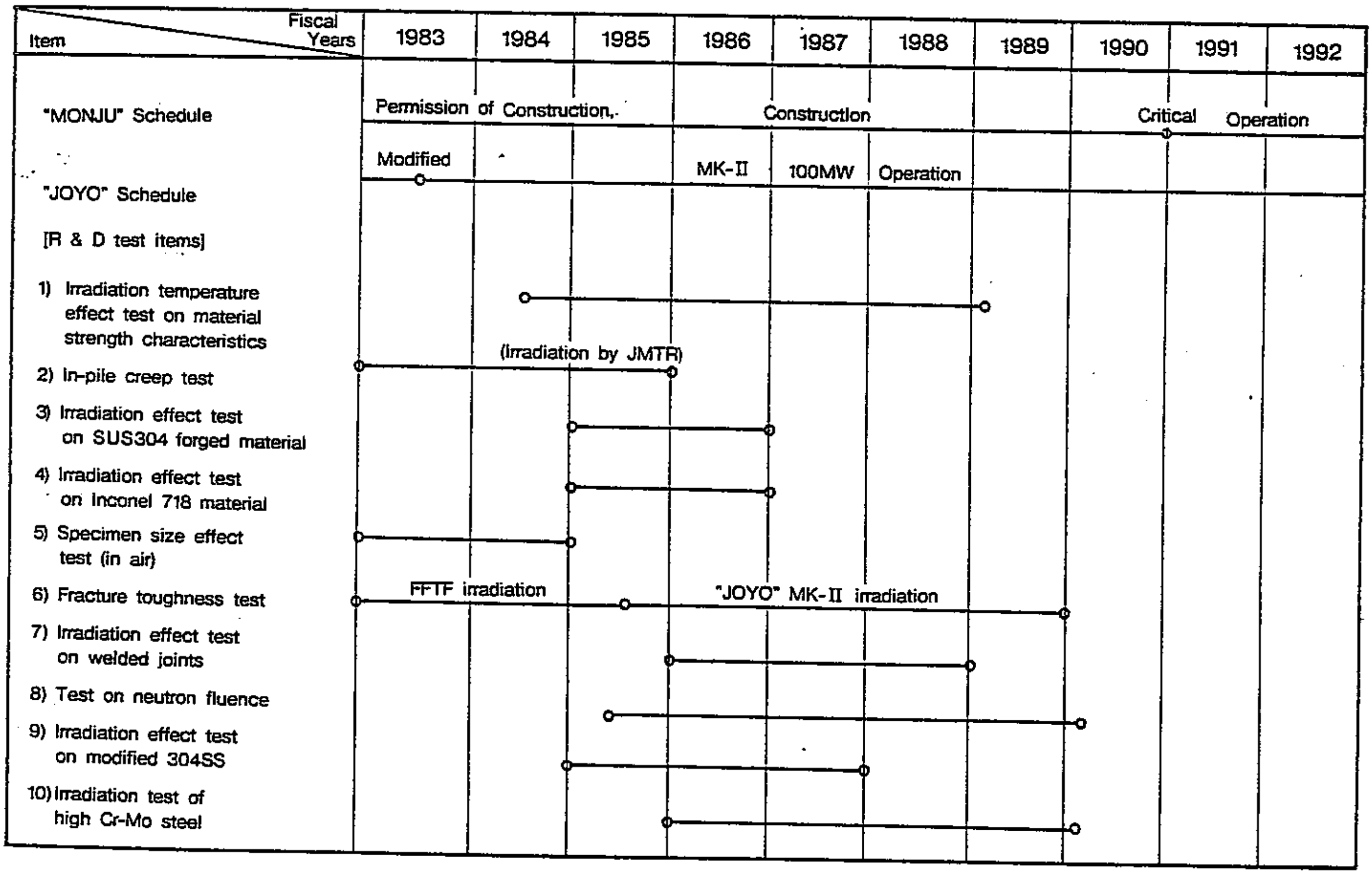
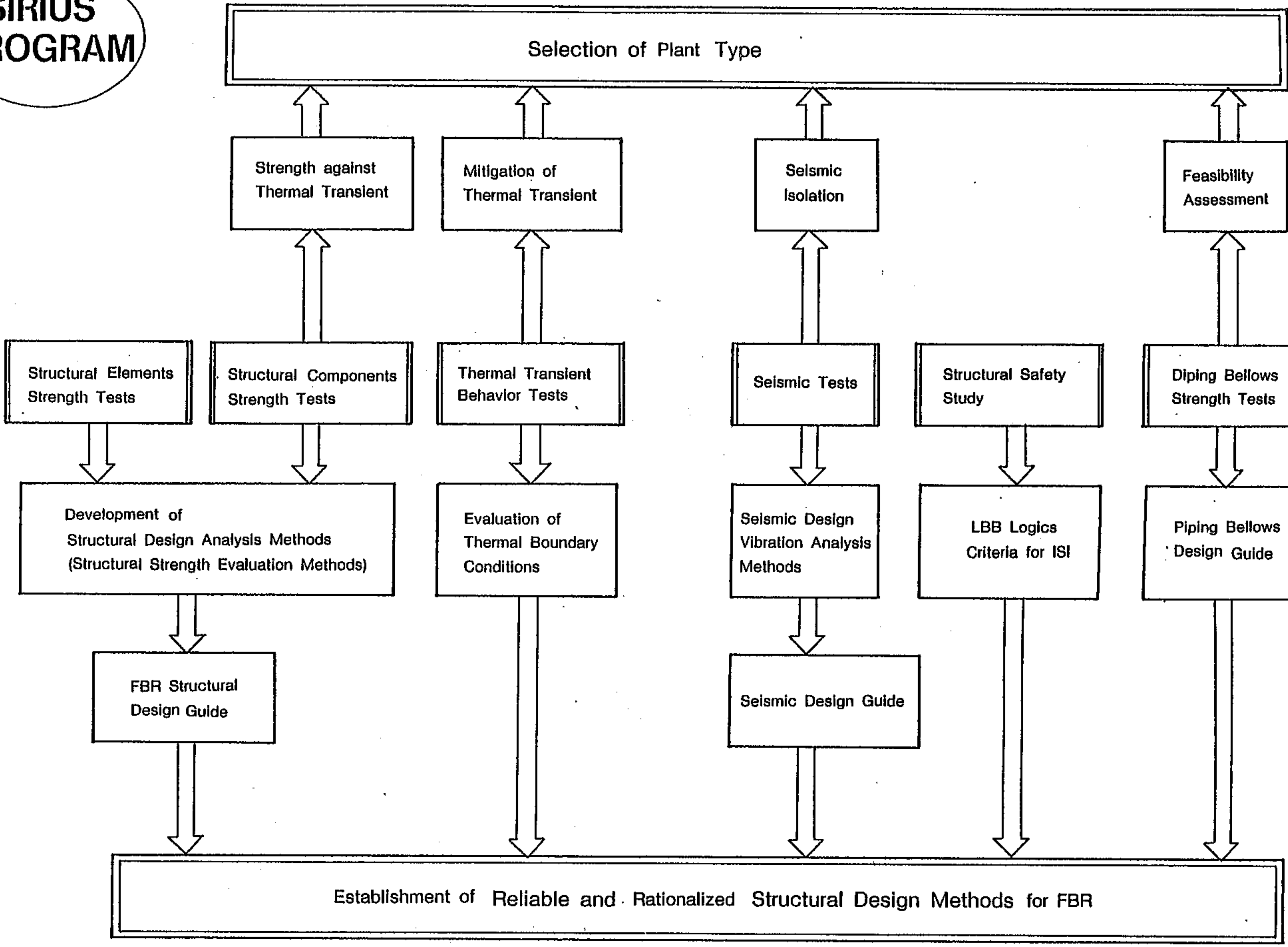


FIG R&D SCHEDULE (TENTATIVE)



SIRIUS PROGRAM



R&D Items of the "SIRIUS" Program

○ Structural Design Analysis Method
- Improvement and Rationalization of "the Elevated Temperature Structural Design Guide for MONJU"
- Improvement of structural strength Evaluation Method
- Improvement of General Purpose Structural Analysis Program (FINAS)

○ Structural Elements Strength Tests
- Thermal Transient Tests and multiaxial Tests of Piping Elements, etc., Including Structural Discontinuiety and Weldment.

○ Structural Component Strength Tests
- Structural Integrity Tests to Failure of Components (Reactor Vessel, SG Tube to Tubesheet, Large Diameter Piping Bellows, etc.)

○ Thermal Transient Behavior Tests
- Thermal Striping Tests of Various Configuration Types of UCS.
- Development of Thermal Insulation System for Near-Sodium Level

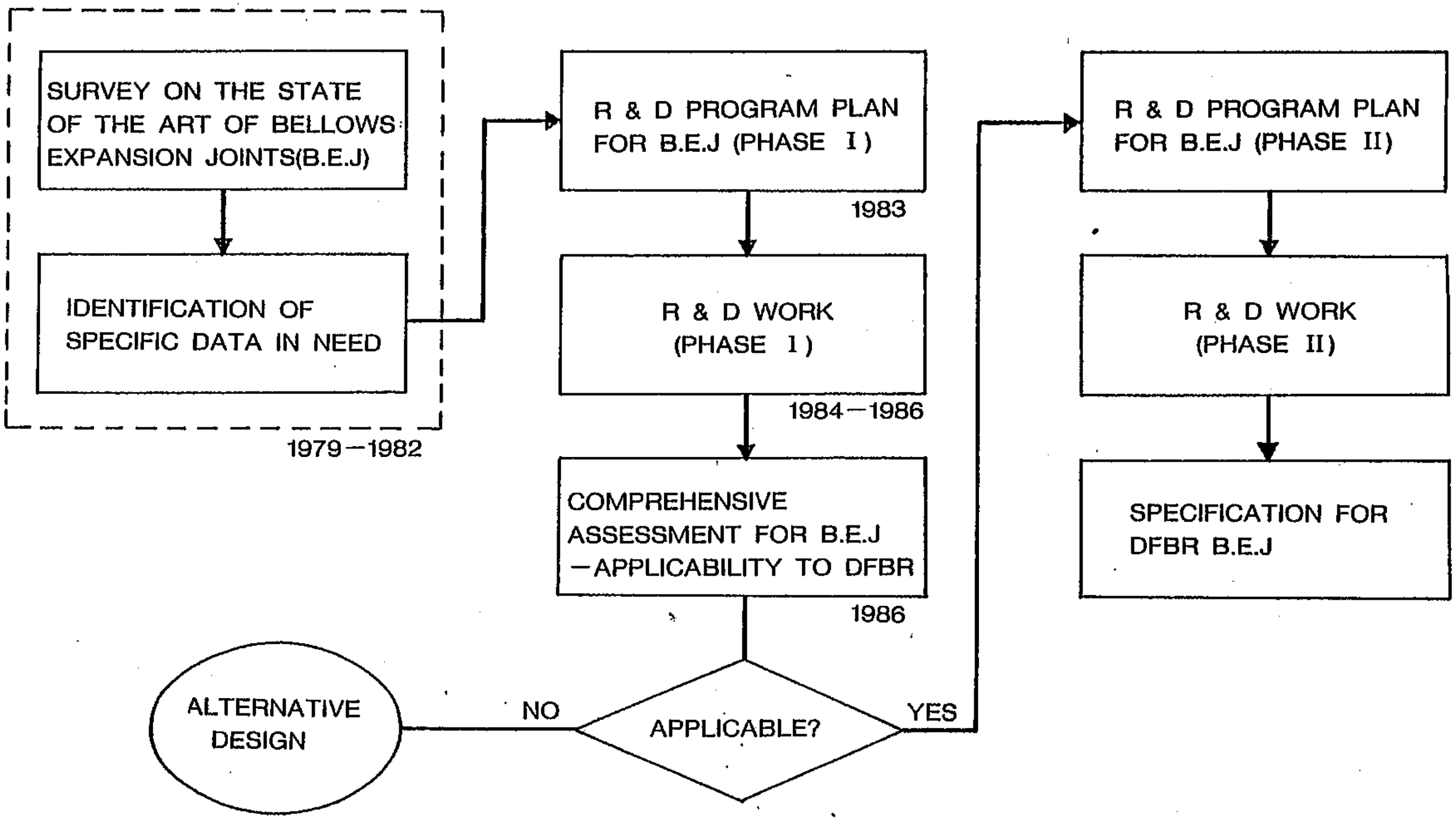
○ Seismic Tests
- Feasibility Study of Isolation System
- R&D Works for Development of Seismic Analysis Method (Core Assembly, Piping System, Fluid Structure Interaction, etc.)
- Preparation of Seismic Design Guides Specific to LMFBR.

○ Structural Safety Tests
- Crack Propagation Tests of Structural Element
- Establishment of Fracture Mechanics Approach and Flaw Evaluation Method
- Extension of LBB Theory by Fracture Mechanics Approach.

○ Piping Systems Rationalization Study (Piping Bellows)
- Feasibility Study of Piping Bellows
  Creep-Fatigue, Thermal Transient, Ratchet, Buckling, Vibration Tests, etc.
- Preparation of Design Guide for Piping Bellows.

GUIDE LINE FOR THE DEVELOPMENT OF BELLOWS EXPANSION JOINTS AT PNC (Included in the "SIRIUS" Program)

R & D FLOW



SODIUM TECHNOLOGY

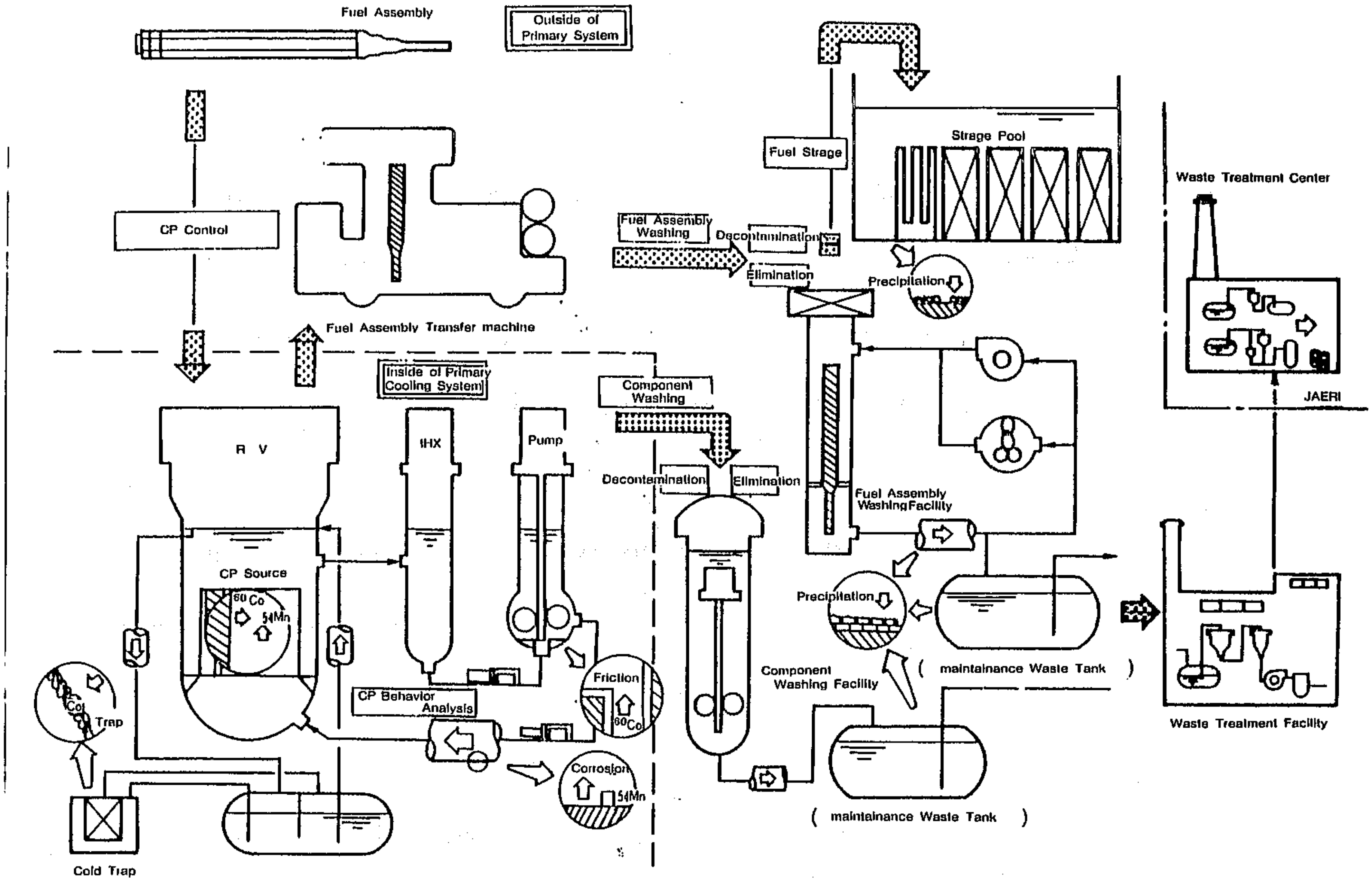
The "ALPHABET" Program

: Test Program for Reduction of Radiation Burden
Caused by Radioactive Corrosion Products.

- Administration of Program

1. CP Behavior : Development of Analysis Technique for CP Behavior 2. CP Control : Development of Technique for CP Control	Inside of the Primary Cooling Syetem
3. CP Decontamination : Development of Technique for CP Decontamination 4. CP Elimination : Development of Technique for CP Elimination	Outside of the Primary Cooling Syetem

A, B, C, D, E ⟵ Alphabet



PROBLEM AND COUNTERMEASURE OF RADIOACTIVE CORROSION PRODUCTS AT "JOYO"

1. Developmrent of Analysis Technique for CP Behavior

◎Clarification of the Mechanism of CP Generation and the CP Transfer and Precipitation Behaviors in the Primary Cooling System

⬇

Systematization of CP Behavior Analysis Code and its Reflection to the Demonstration FBR.

(1) Comprehension of CP Behavior by Out-of-Pile Tests

(2) In-Situ Measurement of CP Distribution and Precipitation Rate at "JOYO"

➡Verification of CP Behavior Analysis Codes : SAFFIRE and PSYCHE

Unification of CP Behavior Analysis Codes

Analysis and Evaluation of "MONJU" Plant System

Reflection of the Results to the Demonstration FBR.

2. Development of Technique for CP Control

◎Reduction of CP Generation Rate

(1) Control of CP Generation Source

- Control of Co Level Included in Structural Materials and Core Materials
- Developnment of Co Free Hard Facing Materials

(2) Control of CP in Sodium

- Development of CP Trap Material (Out of Pile Test)
- CP Trap Test at "JOYO"

3.Development of Technique for CP Decontamination

◎Decontamination of CP Precipitated in the Pipings and Components of Fuel Assembly Washing Facility and its Waste Treatment.

⬇

Development of Synthetic Decontamination Technique for FBR

(1) Morphology Analysis of Precipitated CP

(2) Basic Tests of Chemical Decontamination

(3) Small Scale Test of Chemical Decontamination

(4) Mack-up Test of Chemical Decontamination and Confirmation of Propriety of Obtained Results

(5) Decontamination of Practical Plant ("JOYO")

4. Development of Technique of CP Elimination

◎Elimination of CP Included in Pipings and Components of Liquid Waste Treatment Facility

⬇

Establishment of the System for CP Elimination and its Waste Treatment in FBR

1) Morphology Analysis of Washing Waste Including CP
2) Design, Manufacturing and Operation Evaluation of the System for CP Elimination and its Waste Treatment in the "JOYO" Liquid Waste Treatment Facility
3) Synthetic Evaluation of CP Eliminetion Performance and Establishment of Waste Treatment System

5) Safety

Safety

0. Overall

Till criticality of Monju in early 1991, Safety Engineering Division at OEC and Safety Engineering Block at HQ, are much devoted to the resolution of homeworks posed by the regulatory and to the confirmation of CP-related work. However, some of the Monju-related work, especially experimental facilities built or planned can be used for the future demo plants. Details of the past safety work can be learned from the Annual Report of Safety Engineering Division (PNC SN 943 84-06), September, 1984, whose copies have been sent to DOE in October, 1984.

1. Reactor Engineering Section

The section is in charge of thermohydraulics of the plant, components, subassemblies both in single and two phase regimes. The main emphasis is in the area of low heat flux and natural circulation/convection phenomena.

—EVST natural circulation tests

- An acrylic vessel of 1/3–scale, 1/6–sector with 36 simulating S/A heaters.
- COMMIX–PNC used for pre–analysis.

—DRACS natural circulation tests

- An acrylic model of 1/3–scale with simulating heat source is being built.
- COMMIX–PNC for pre–analysis.

—PLANDTL loop

- A new versatile Na loop will be built by early 1987. The loop is initially utilzed for low heat flux behavior for resolution of decay heat boiling under Monju LOPI with a bypath line. The loop can be modified later for natual circulation tests.
- Domestic codes as well as SSC–L and COMMIX–PNC are being used for predicting loop behavior.

— Analytical tools

- Much advance has been achieved in the past few years with domestic codes in single and two-phase regimes.

2. Reactor Safety Section

The section has been and will be for a few more years to the study of severe accidents in response to the regulatory homeworks.

—VECTOR vessel

- Na vapor expansion/condensation study aimed for HCDA bubble collapse.
- SIMMER is used for validation and model improvement.

—JET vessel

- aims at basic studies of molten core materials and the lower-lying structure, with simulant material first. The study also aims at vapor blanketing effect of the jet drop and accompanying MFCI. Induction heating is employed for creating the melt.

—Cabri and Scarabee

- PNC has been a junior partner of Cabri since its beginning, and will join a partial test matrix of Scarabee as a junior partner in early summer, 1985.

—Consequence Analysis

- By the mandate of regulatory, a Monju PRA is requested to PNC. Phenomenological scenario development is in progress in similar fashion to the KfK-IA Conditional Risk Study for SNR-300. The energetics path has been proved safe with extensive scale-models and analysis. The difficulty is rather in the mild energetics path which involves recriticality, materials relocation discharge and debris cooling. In addition to the US codes (SAS, SIMMER), a code, APPLOHS was developed to analyze the protected loss of heat sink scenario.

3. Plant Safety Section

The section consists of two major parts : containment studies and sodium-water reactions.

—Sodium fire and aerosols behavior

- After a series of scoping studies (SOFT–series), a new facility, SAPFIRE will start operation in May, 1985. The facility consists of SOLFA–1 which aims at a Na fire effect in the operating floor of the SG room, Na drainage to the smothering tank below and the effectiveness of the smothering equipment. The second rig ($80m^3$) called SOLFA–2 is a closed vessel with capability of creating fires inside or feeding controlled aerosol density from an outside chamber. Endurance of scaled equipments or the behavior of aerosols can be studied.
- Domestic codes modified from US codes have been satisfactory in the subject area. Especially, ABC–INTG for aerosols has proved quite capable.

—Containment and source term

- A third rig FRAT in the above facility will serve in the future to the study of fission products chemistry in Na.

- Need is felt to mechanistically follow accident sequences in conjunction with the consequence analysis of Monju PRA and CONTAIN is selected as the tool. Individual module tests have been conducted and compared with separate codes.

—Na–Water reactions

- Large leak experiments with 2–1/4Cr 1Mo steel are completed and only failure propagation and micro–leak tests continue at reduced rate. For future demo plants, plans are being discussed for immersed–type rupture disks, reactions in high Cr steel tubes, and double tube SGs.
- The SWACS code series has been used for Monju licensing and PNC is quite satisfied with the series.

4. PRA Studies

A small task force team conducts the systems analysis of Monju at HQ, and the consequence analysis of identified paths is performed by the three sections at OEC.

—Systems analysis

- The work which examines the final design of Monju has already yielded possible changes in control and operating procedures in Monju.
- Though not performed yet, seismic PRA must be implemented as the most important external event. Fires (ordinary and Na-caused) are already in the work plan.

—Database development

- In January, 1985, an agreement has been reached between DOE and PNC to interchange the US and the PNC data on Na components. PNC is constructing the FREEDOM DBMS which will feed PNC data in compatible format to CREDO.

5. Recent WG Activities

- 84/11 : SM on sodium fires and aerosols
- 85/01 : Conclusion of SMA on CREDO/OEC
- 85/01 : Observer attendance to the US–UK Safety Design Principles Meeting
- 85/10 : SM on thermohydraulics including boiling

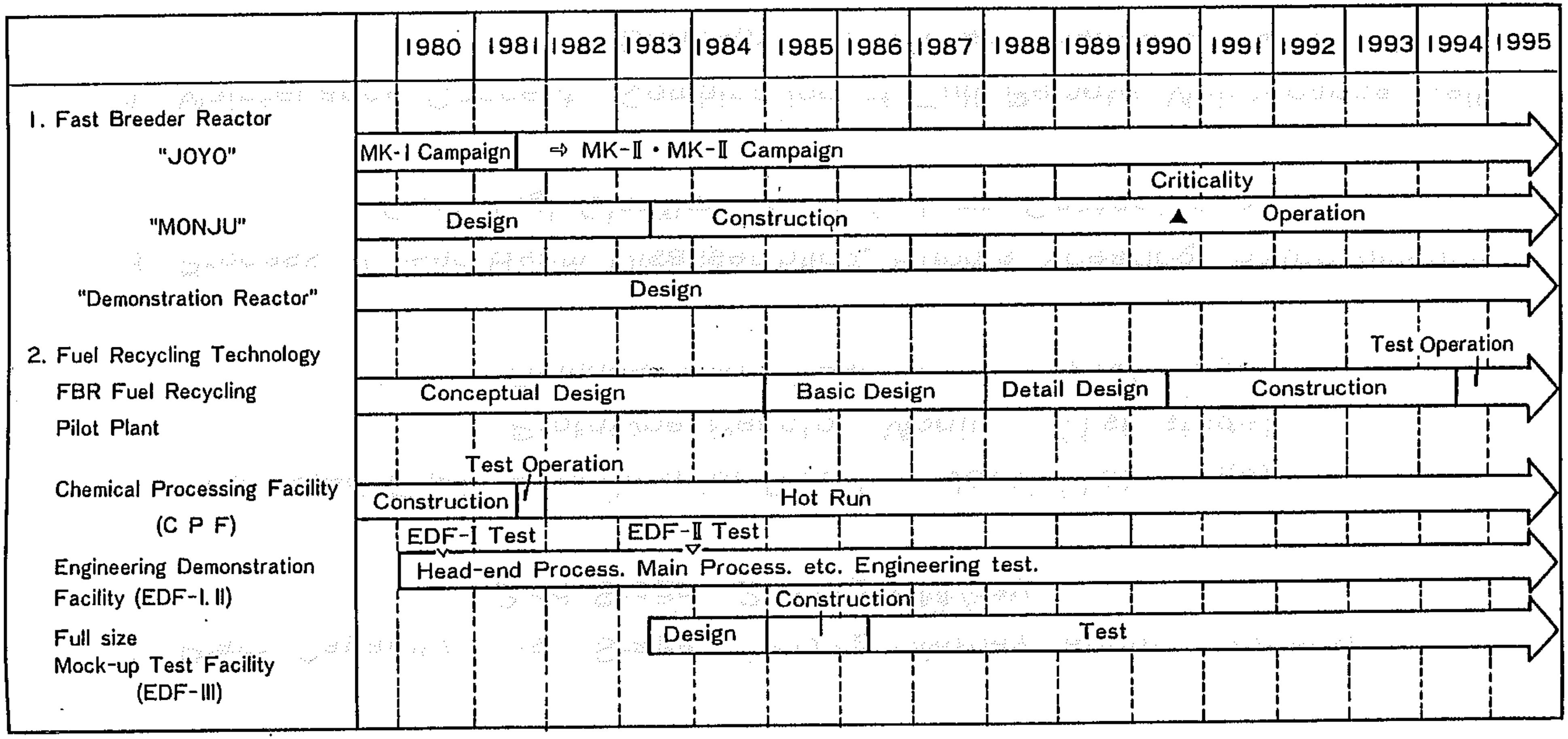
6. Conclusions

- The US has been an excellent mentor in the past and still retains deeper expertise. FBR/Safety should pursue research where data is still in sufficient and enchange the new data with the extant US data. PNC should encourage US involvement in conducting experiments and developing/validating computer codes.

6) FBR Fuel Reprocessing

Development of FBR Fuel Recycling Technology

Mar. 1985



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FBR Fuel Recycling Pilot Plant

1. Plant Capacity : 1st Stage－120kg HM/day (about 10years)
   2nd Stage－240kg HM/day

2. Processed Fuel : Experimental Reactor 'Joyo' (1st stage)
   Prototype Reactor 'Monju' (1st stage)
   Demonstration Reactor (2nd stage)

3. Process : Laser-Beam Disassembling, Bundle Shearing, Batch dissolver
   Centrifugl Clarifier, Pulse-Column Contactor, etc.

4. Maintenance Concept : Combination of Full Remote Maintenance Cell
   Concept and Contact Maintenance Cell Concept

5. Operation Start : 1st Stage－around 1995
   2nd Stage－around 2005

FBR Fuel Cycle Working Group Activities

- Initial Areas of Exchange Relate to:
  - (1) PNC-DOE Criticality Data Exchange Program
  - (2) Remote System Technology Development

- Future Exchange/Cooperation Items:
  - (1) Process Technology Development
    - Collaboration and exchange of data about centrifugal contactor, centrifugal Clarifier, etc.
    - Exchange of hot test data
  - (2) Exchange of data and cooperative study about material of reprocessing process equipments
  - (3) Exchange of data and cooperative study about technology of analysis and measurement

7) Waste

Activity of PNC/DOE Waste Management

AREAS OF ACTIVITY:

- High Level Waste
- Transuranic (TRU) Waste
- Waste Isolation

SUMMARY OF MEETINGS:

- Oct. 1980	USA	Establishment of Joint Working Group
- Oct. 1981	Japan	First DOE/PNC Joint Working Group Meeting
- Sep. 1982	Japan	Specialist Meeting on Ceramic Melter Design
- Oct. 1982	USA	Second DOE/PNC Joint Working Group Meeting
- Jun. 1983	Japan	Specialist Meeting on TRU Waste Management
- Jun. 1983	Japan	Specialist Meeting on Sodium Waste Management
- Dec. 1983	USA	Short Visit on TRU Waste Management
- Feb. 1984	Japan	Short Visit on Immobilization and Melting of TRU Waste
- May 1984	Japan	Short Visit on Basic Studies of TRU Waste Management
- Jul. 1984	Japan	Third DOE/PNC Joint Working Group Meeting

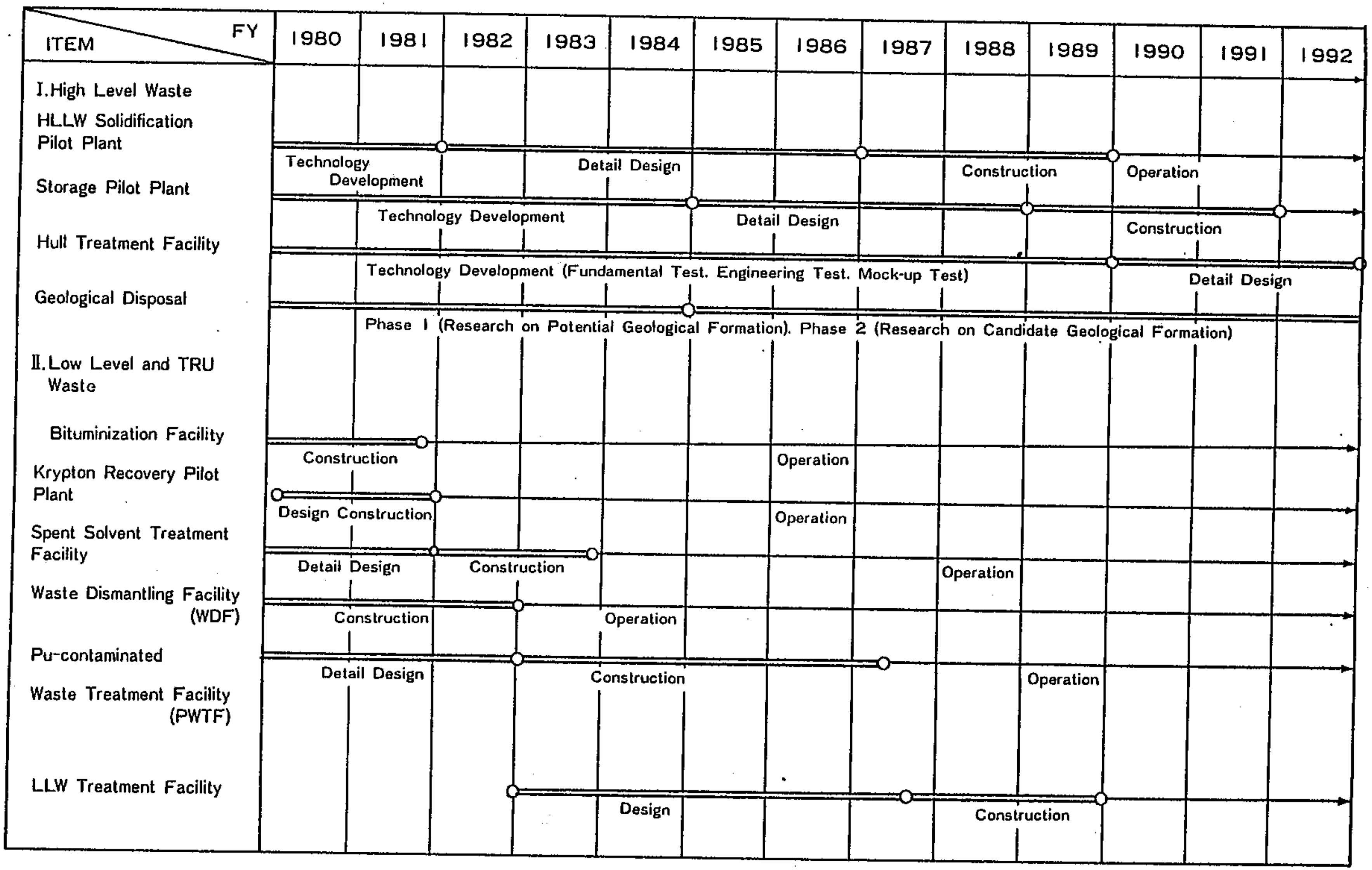
CONTRACT COMPLETED:

Mar. 1983 - Nov. 1983 Review and Evaluation of PNC Ceramic Melter System Design

PRESENT ACTIVITIES

- DOE and PNC presently working on development of an Agreement for DOE/PNC Collaborative Testing of the Radioactive Liquid-Fed Ceramic Melter. Objective is to cooperate in the demonstration of the ability of the melter to vitrify civilian high level waste through an experimental program, and to provide each organization an opportunity to observe and participate in future civilian applications of the technology developed in the experimental program.

- Specialists Meeting on High Level Waste Vitrification to be held in US in Summer 1985.


ITEM
FY
1980
1981
1982
1983
1984
1985
1986
1987
1988
1989
1990
1991
1992
I. High Level Waste
HLLW Solidification Pilot Plant
Technology Development
Detail Design
Construction
Operation
Storage Pilot Plant
Technology Development
Detail Design
Construction
Hull Treatment Facility
Technology Development (Fundamental Test. Engineering Test. Mock-up Test)
Detail Design
Geological Disposal
Phase I (Research on Potential Geological Formation). Phase 2 (Research on Candidate Geological Formation)
II. Low Level and TRU Waste
Bituminization Facility
Construction
Operation
Krypton Recovery Pilot Plant
Design Construction
Operation
Spent Solvent Treatment Facility
Detail Design
Construction
Operation
Waste Dismantling Facility (WDF)
Construction
Operation
Pu-contaminated Waste Treatment Facility (PWTF)
Detail Design
Construction
Operation
LLW Treatment Facility
Design
Construction

Development of the Vitrification Process

(1) Cold Engineering Tests

Engineering Test Facility (ETF)

to develop the vitrification process and equipment.

(2) Cold Mock-up Tests

Mock-up Test Facility (MTF)

to demonstrate the vitrification process and to develop the remote operation/handling technology.

(3) Hot Laboratory Tests

Chemical Processing Facility (CPF)

to confirm the denitration and Joule-heated melting procedures of full-active liquid waste.

8) Profile of Major Test Facilities

Partial List : Test Capabilities of Oarai Engineering Center, PNC

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Sodium Supply & Purification System	Sodium Supply and Purification for SPTL and TTS Loop.	3	500	650	102	72/3	
Large Scale Sodium Pump Test Loop (including 42" Bellows Test Articles) (SPTL)	Performance and Endurance Test on "MONJU" Pump, and In-Sodium Test of Piping Bellows	12	450	100	19	72/3	
Preheating Temperature Controling Test Apparatus	Test on Preheating System with PTC Elements	3	500	5	–	84/11	
Self-Orientation Test Apparatus	Test on Self-Orientation Mechanisms for "MONJU" Fuel Subassembly	–	R.T	–	–	84/7	
Sodium Level Meter Calibration Test Loop (under Planning)	Calibration Test of "JOYO" & "MONJU" Level Measurment Sensors.	1	600	20	0.5	(85/5)	
Shield Plug Thermal Insulation Test Apparatus (SPINTA)	Test on Thermal Insuration Performance of "MONJU" Shield Plug Mock-Up	4	540	614	71	75/4	
Ex-Vessel Transfer Machine Test Loop	Mock-up Test of "MONJU" Ex-Vessel Fuel Transfer Machine	1	285	19		75/3	
Control Rod Drive Mechanisms Test Loop	Mock-up Test on "MONJU" Control Rod Drive Mechanisms	2	600	67.5	2.2	73/4	
Nak Flow and Heat Transfer Test Loop	No Operation Preliminary Test on Heat Transfer between Na/Nak for DRACS. (Future Plan)	1	550	330	0.7 Nak	76/9	

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Flow Cap (TON)	Compl	Note
1MW Steam Generator Test Facility	Basic and Detailed Tests of Steam Generators.	2	540	12,000 Electric	20 t/h Na	71/4	
		1/4	513	90	2 t/h Water		Steam Press. 173atg.
50MW Steam Generator Test Facility	(1) Performance Tests of Steam Generators.	12	540	50,000 LPG		74/6	
	(2) Performance Tests of Control System and Plant Dynamics Characteristics.						
	(3) Performance Tests of Plant Operation Procedures, Operation Monitoring System and Accident Simulated Behavior.						
	(4) Performance Tests pf Sodium-water Reaction Detection System.						
	(5) Performance Tests of Auxiliary Cooling System.						
	(6) Endurance Tests of Explosive Plugging for Steam Generator Tubes.						
	(7) Endurance Tests of Steam Generators.						
	(8) In-cover gas Tests of Rupture Discs for Steam Generators						

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Sodium Thermal Shock Test Loop	Thermal Striping Test of Upper Core Structures of "MONJU".	6	560	290	16	73/2	
Small Thermal Shock Test Loop	Thermal Fatigue and Thermal Ratcheting Test of Simple Structures.	3	600	55	0.1	79/8	
Large Capacity Electromagnetic Pump Test Loop	No Operation	8	560	38	2.4	77/3	
Structural Integrity Test Rig (SITR)	Thermal Behavior Test near Sodium Free Surface of "MONJU".	3	600	300	15	81/12	
Thermal Transient Test Facility for Structures (TTS)	Strength Test of Various Structures under Combined Thermal and Mechanical Loadings at High Temperatures.	10	650	1180	38	84/6	
Sodium Piping Thermal Transient Test Loop	Thermal Fatigue and Thermal Ratcheting Test of Simple Structures.	2	550	360	4.2	72/2	
Structural Component Buckling Test Rig (under construction) (SBT)	Buckling Test of Piping Bellows and Other Structural Components.	Electric Motor	650	130	Air		
Acostic Emission Measuring & Analyzing System		-	-	-	-	74/4	
Large Bellows Creep-Fatigue Test Rig for Piping Bellows (BCFT)	Creep-Fatigue Test of Piping Bellows.	Oli Press 150t	650	130	Air	85/4	
Air-cooling Thermal Transient Test Facility (ATTF)	Thermal Transient Test of Tubesheet Structures of Steam Generator	8	650	30	Air	83/9	
Creep-Buckling Tesst Rig for Piping Component	Creep-Buckling Test of Piping Components.	-	650	60	Air	81/9	
Multi-Loading Test Rig	Creep-Behavior Test of Beams under Combined Primary and Secondary Stresses.	Oil Jack 20tx4	650	146	Air	80/8	

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Bi-Axial Creep Testing Machine	Bi-axial Creep Behavior Test of Tubes.	Oil Press 10t	600	5	N2 gas	80/8	
Creep Test Loop-1	In-Sodium Creep Test on High Velocity Sodium Effect for Structural Materials.	3/4	600	105	0.52	79/6	
Fatigue Test Loop-1	In-Sodium Creep-Fatigue Test for Structural Materials.	1/2	700	30	1.3	79/1	
Fatigue Test Loop-2	In-Sodium Fatigue Test on Sodium Velocity Effect for Structural Materials.	3/4	600	105	0.4	79/6	
Thermal Fatigue Test Loop	Development of Evaluation Method on Creep Fatigue Strength of Weld joint.	3/4	650	500	0.5	-	under planning
Sodium Instrument Test Loop	Development and In-Sodium Test on Core Exit Flow-meter	3	600	220	2.9	74/12	
Thermocouple Responce Test Loop	Development and In-Sodium Test on Thermo-meter.	3/4	600	23	0.08	78/1	
Material Test Sodium Loop-1	Corrosion, Mass-transfer and Creep Tests for Structural and Core Materials.	3/4	700	500	1.0	70/4	
Material Test Sodium Loop-2	Do.	3/4	700	600	1.8	72/3	
Carbon Transfer Test Loop	Carbon Transfer Test between Austenitic Stainless Steel and 2 1/4 Cr-Mo Steel.	1/2	600	42	0.3	75/11	
Sodium Exposure Test Loop-1	To provide High Velocity Sodium Environmental Effect to Structural Materials. (After Sodium Exposure Test, Change in Mechanical Properties of Specimens are tested.)	3/4	700	105	0.4	79/6	
Sodium Exposure Test Loop-2	Do.	3/4	550	105	0.4	79/6	
Sodium Exposure Test Pot	To provide Sodium Environmental Effect to Core Material Specimens.	3/4	730	25	0.4	81/7	

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Structural Materials Sodium Eposure Test Pot	To Provide Sodium Environmental Effect to Structural Material Specimens.	3/4	750	4	0.35	82/9	
Self-Welding and Wearing Test Loop	In-Sodium Tribology Tests of Sliding Materials.	3/4	700	138	0.5	72/3	
Activated Material Test Loop-2	In-Sodium Tests of CP Behavior and CP Trap.	3/4	650	20	0.03	73/11	
Fission Product Test Loop	In-Sodium Tests of FP Behavior and FP Trap.	3/4	650	25	0.04	75/2	
Sodium Impurity Measurment Test Loop	In-Sodium Tests of Impurity Behavior and Online Monitors.	3/4	760	300	2.0	70/4	

Name of Facility	Purpose	Equipment	Note
In-air Test Machines for Structural Materials.	In-air Structural Materials Tests and Data Analyses.	Data Processing System Creep Testing Machine (x100) Fatigue & Creep Fatigue Testing Machine (x 9) Tensile Testing Machine(x 3) Impact Testing Machine(x 1) Relaxation Testing Machine (x 2) Multi Creep Testing Machine (x 2)	
In-air Testing Machine for Specimens after Sodium Exposure	In-air Structural Materials Tests of Specimens after Sodium Exposure.	Relaxation Testing Machine (x 2) Creep Testing Machine (x 5) Fatigue & Creep Fatigue Testing Machine (x 2)	
Equipments for Metallurgical Examination	Metallurgical Examination of Specimens after Sodium Exposure.	Transmission Electron Microscope (x 1) Scanning Electron Microscope (x 2) Ion Micro analyzer (x 1) X-ray microprobe Analyzer (x 1)	

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Sodium Mist Heat Transfer Test Loop	To study the rate of sodium vaporization, the behavior of sodium mist in the cover-gas space, etc.	0.5	600	11	0.7	73/1	
Thermal Emissivity Measuring Apparatus	To measure thermal emissivity of liquid sodium surface and of structures covered by sodium film.	0.5	600	-	0.2	80/10	
Hydrodynamics Test Loop	Prototypic in-vessel flow characteristics tests and flow distribution tests using a 1/2 scale model of MONJU reactor vessel with internals.	26	50	1550 Main Pump	300 Water	72/10	
Fuel Assembly Hydraulic Simulation Test Loop	Tests on hydraulic simulation of the prototype fuel assembly of local blockage in fuel rod bundle and the basic flow characteristics of grid-type rod bundle.	6	70	100	7	75/3	
Ex-Vessel Storage Tank Test Rig	To test decay heat removal capability by natural convection in an EVST using a 1/3 scale, 1/6 sector model of MONJU.	1	60	18	1	85/2	
Water Cavitation Test Loop	Water cavitation test for pressure reducing device of reactor components, e.g., entrance nozzle, flow regulating orifice, etc.	6	120	100	20	78/7	
Plant Dynamics Test Loop (PLANDTL)	Various thermal and flow transient tests to establish more rational design rules for safety related equipments.	4	625 (950)	1720	10	87/3	The first test is a demonstration of the decay heat removal capability under the LOPI simulation of MONJU.
Sodium Mixing Test Loop	To study sodium mixing effects on the heat transfer within a fuel assembly. Also to conduct heat transfer tests on a tube bundle and interassemblies, particularly for low flow regime.	4	600	200	2.2	72/8	

Name of Facility	Purpose	Max Pipe Dia (in)	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Decay Heat Boiling Test Loop	Sodium boiling tests at decay power levels to examine (1) steady dryout conditions under wide parameter regions includig reversal flow and (2) boiling supression and/or excursion produced by the interacting buoyancy and two-phase pressure drop effects.	2	650 (950)	500	3	84/10	
Fuel-Sodium Interaction Facility (FSI)	Effects of fuel-sodium interactions under the hypothetical accident sequence of LMFBR	2	600	150	2.5	79/3	Tests completed 1984

Name of Facility	Purpose	Vessel size	Max Temp (C)	Heating Cap (KW)	Fluid Cap (TON)	Compl	Note
Large Leak Sodium-Water Reaction Test Rig (SWAT-1)	To obtain fundamental data on large leak and intermediate leak wastage	0.4 m ID x 2.5 m H	500	170	0.18	70/10	
Small Leak Sodium-Water Reaction Test Loop (SWAT-2)	To ovtain data on small leak and micro-leak phenomena and to develop a leak detection system	0.4 m ID x 2.5 m H	500	240	1.0	72/3	
Steam Generator Safety Test Facility (SWAT-3)	To validate the overall integrity of the secondary system against large scale spdoim-water leaks	1.3 m ID x 6.5 m H	550	770	12.0	75/3	
Micro-Leak Sodium-Water Reaction Test Rig (SWAT-4)	To clarify the self-enlargement of nozzles in micro-leaks	0.15 m ID x 0.7 m H x 3	550	33 x 3	0.0135	81/3	
Acoustic Leak Detection Test Rig (SWAT-5)	To develop an acoustic leak detection system	1.35 m ID x 6.25 m H	R.T.	-	5.0	82/8	

Name of Facility	Purpose	Geometry	Structural Materials	Volume (m³)	Maximum Over Pressure (kg/cm² g)	Maximau Temperature (℃)	Sodium Supply Capacity (TON)
Sodium Fire Test Rig (SOFT-1)	Basic Tests on Partial Model of Sodium Fire Mitigation Systems Aerosel Proof Tests on Post-Accident Monitors and Decay Heat Removal Systems	Floor:3mx3m Height: 3m	Steel	27	atmospheric	150	0.18
Sodium Leak Fires, and Aerosols Test Rig (SOLFA-1)	Demonstration of Sodium Fire Systems Tests on Realistic Sodium Fire Accidents	Floor:5mx5m Height: 7m 2-Story high*	Reinforced concrete Steel	175	+0.03	Concrete 150 Steel 650	15
(SOLFA-2)	Large Scale Tests on Sodium Fires and Aerosol Behavior Tests for Code Validation	Diameter:3.6m Height: 9m	Stainless Steel	80	+ 2	450	
Fission Product and Radioactive Aerosol Release Test Rig (FRAT-1)	Basic Tests on Sodium Fires and Aerosol Behavior Tests on Reaction between Sodium and Fereign Materials Tests on Source Term	Diameter:1.3m Height: 2m	Stainless Steel	3	+ 3	550	0.18

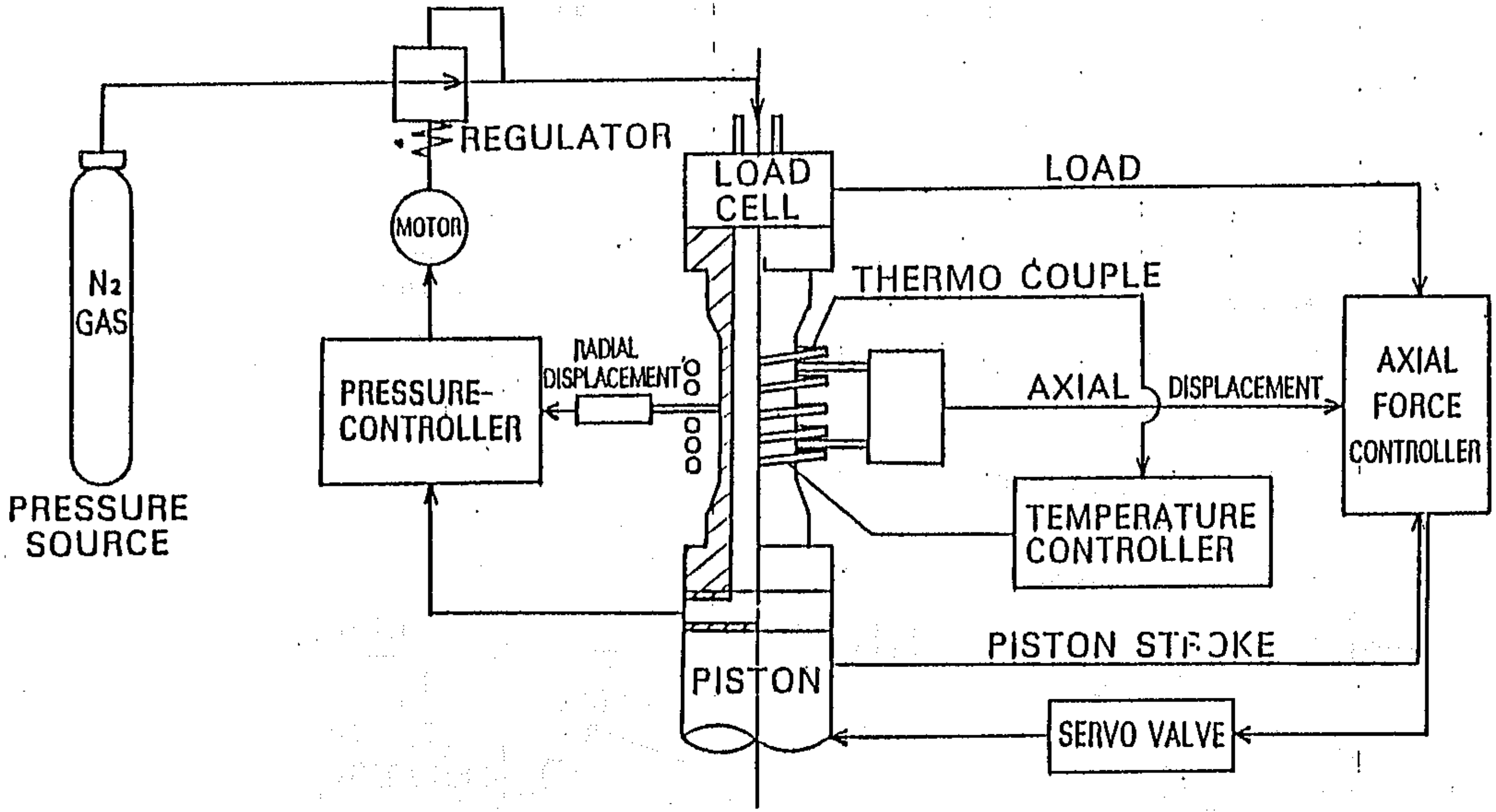
* Note : Partial 1/3 Scale Model of the Monju Secondary Building

TEST FACILITY FOR STRUCTURAL ELEMENT AND COMPONENT

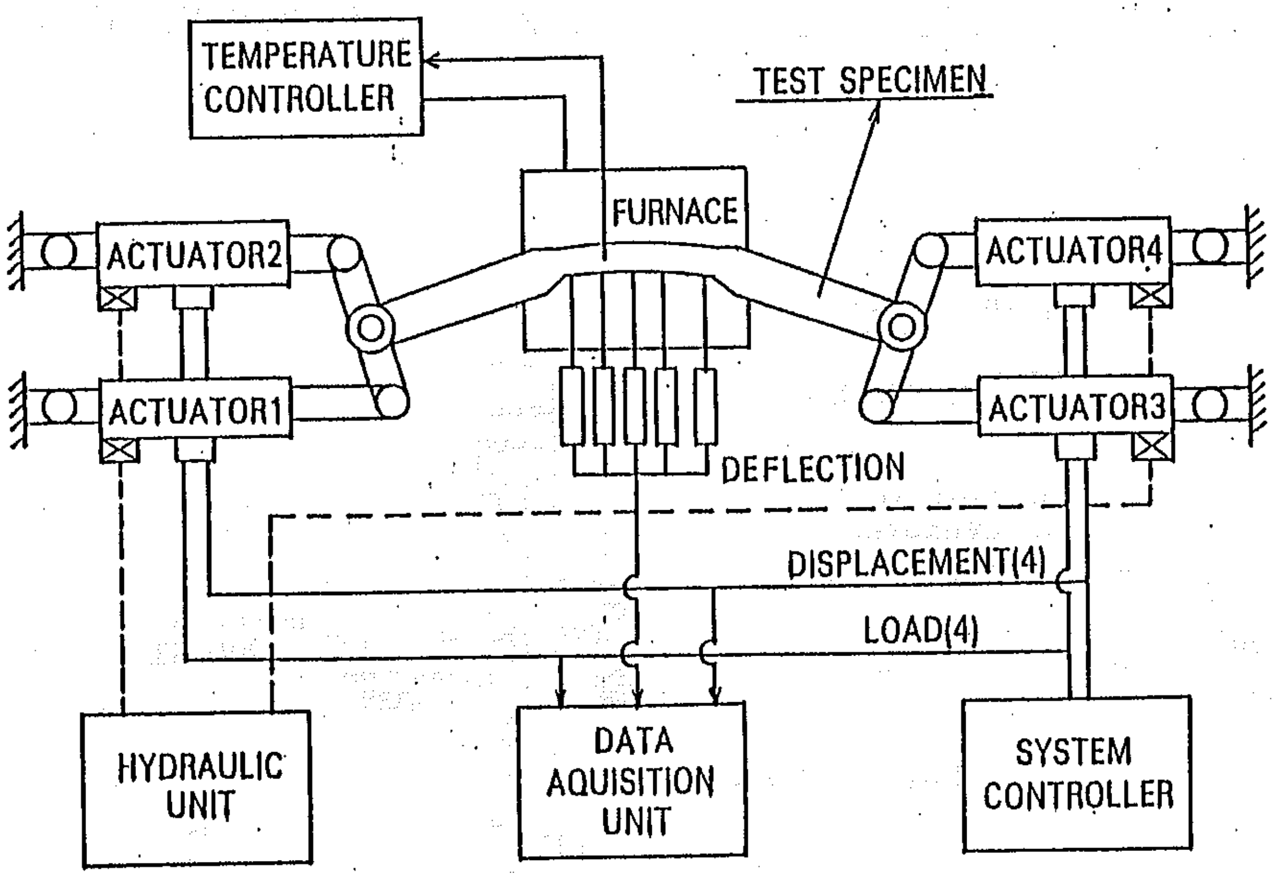
8) – 11

Power Reactor and Nuclear Fuel Development Corporation

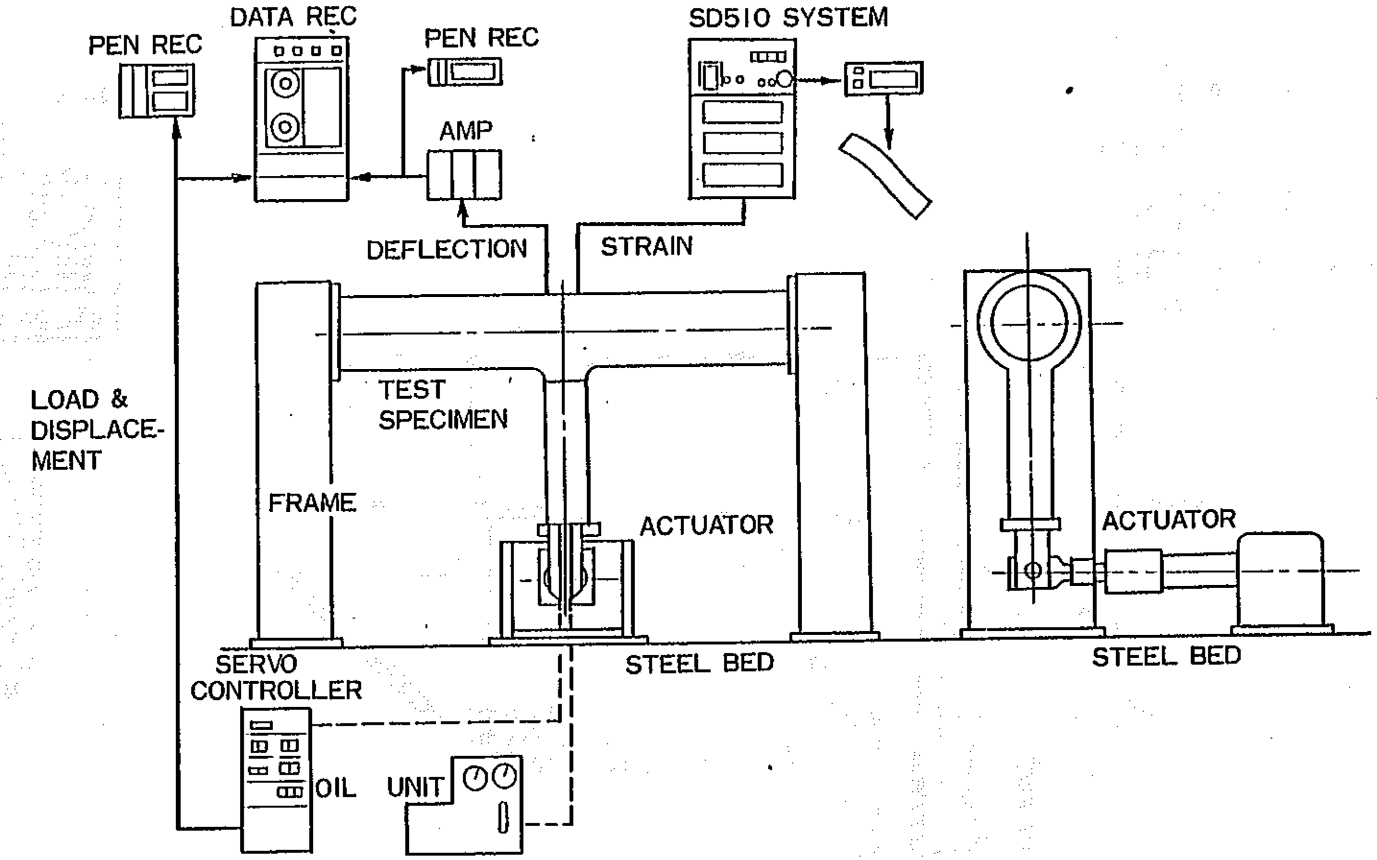
8) — 12



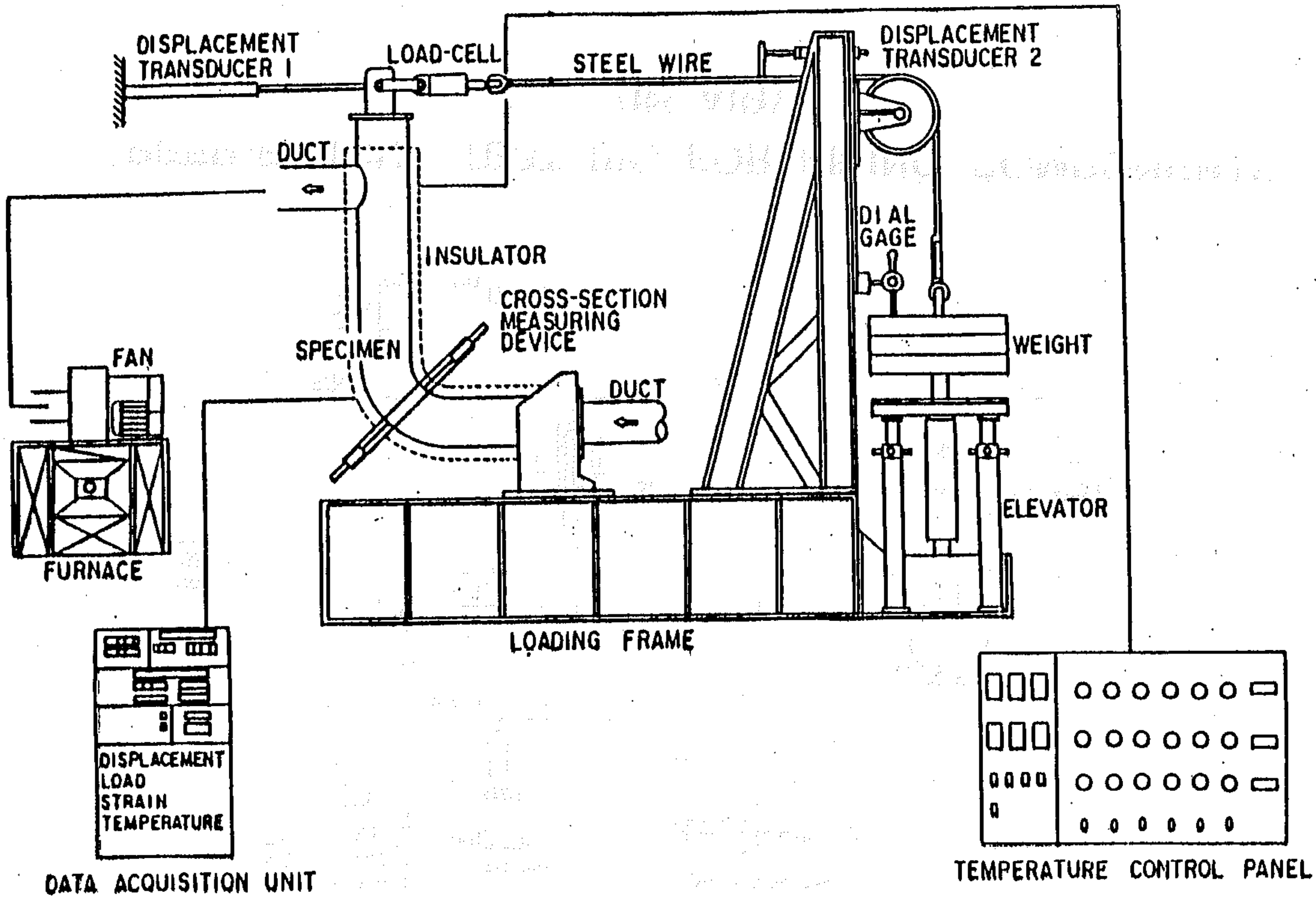
FLOW DIAGRAM OF BI-AXIAL CYCLIC CREEP TESTING MACHINE



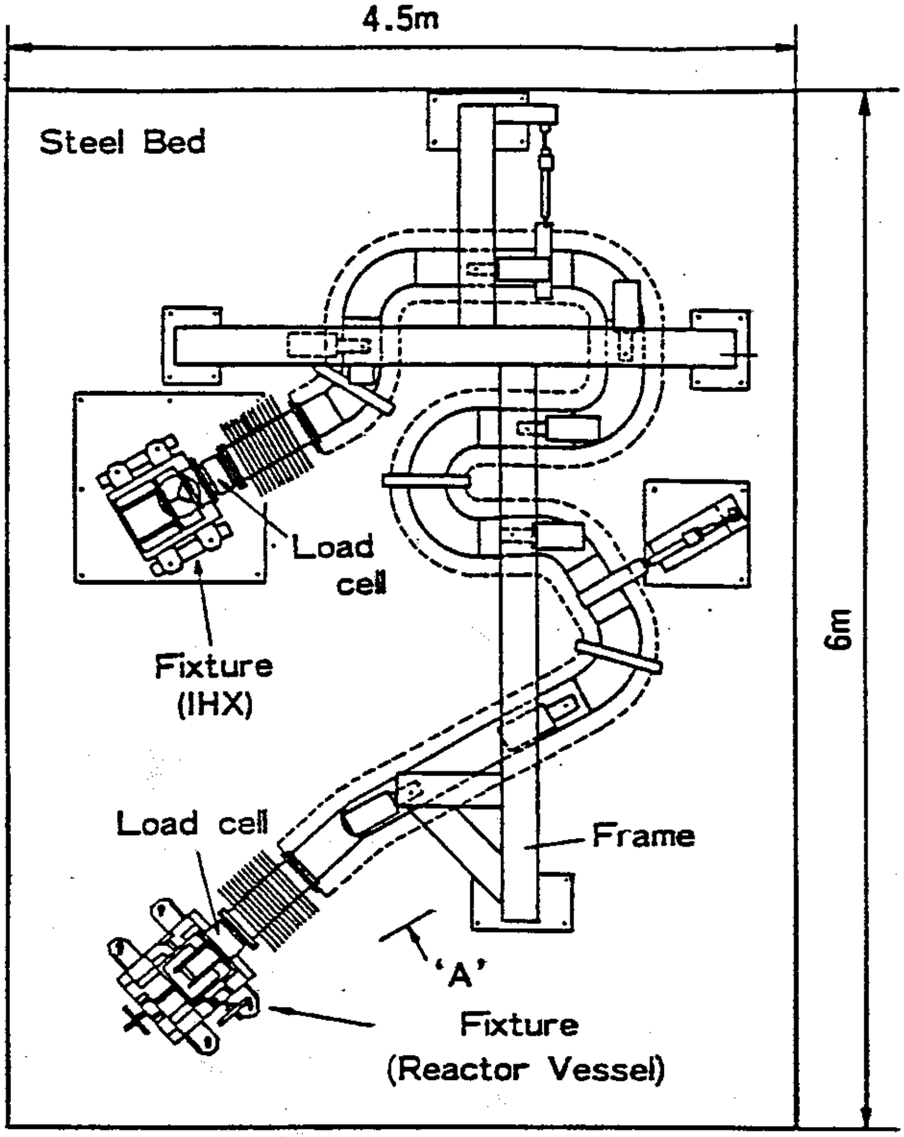
FLOW DIAGRAM OF MULTI-LOADING TEST RIG

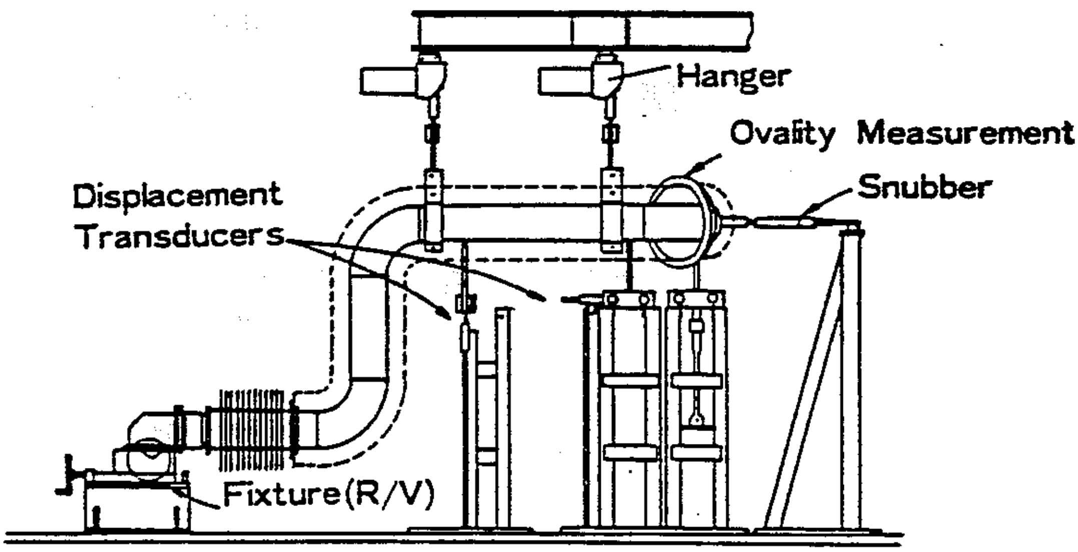


CREEP-FATIGUE TEST RIG FOR PIPING COMPONENTS (IN AIR)



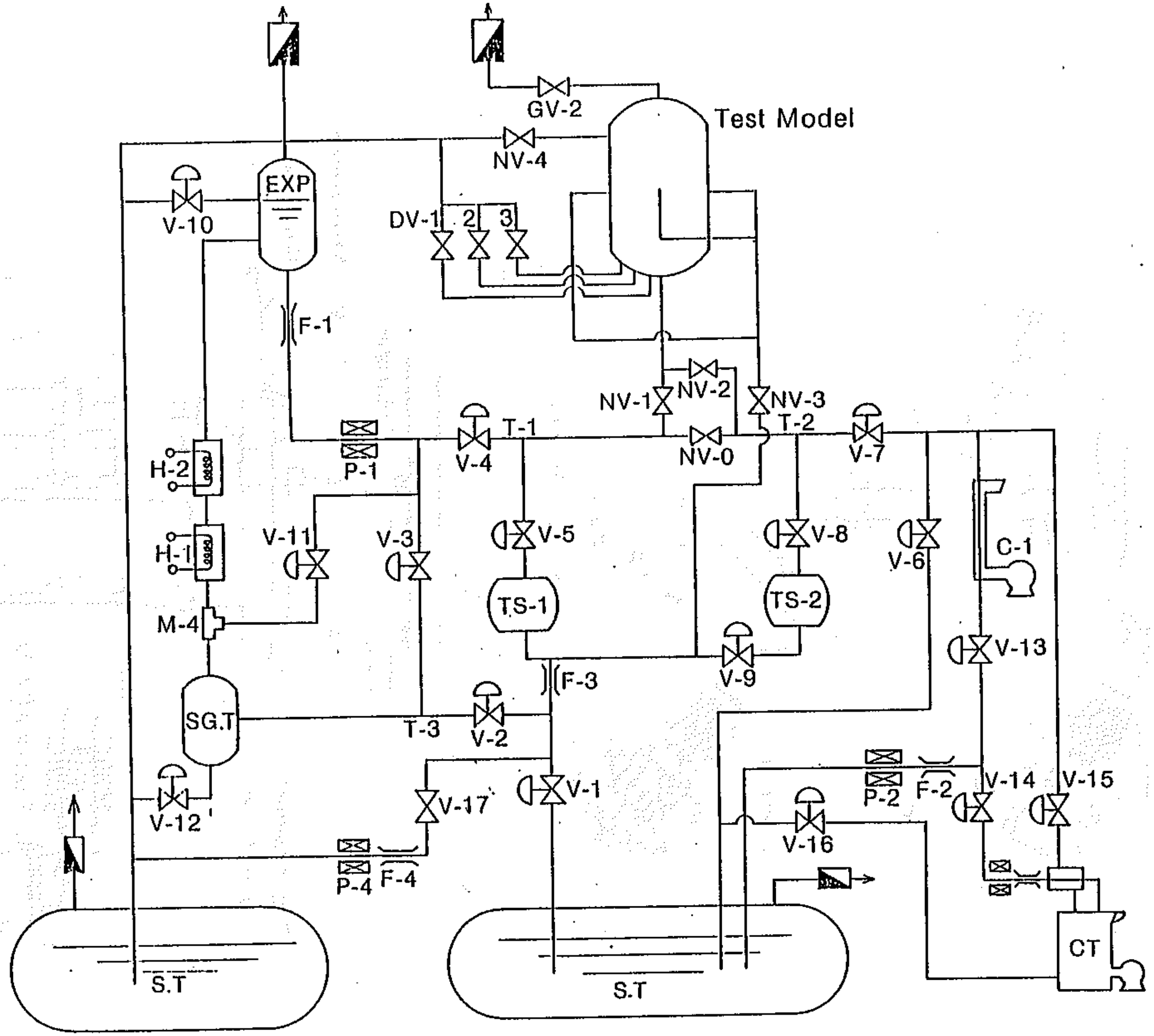
SET-UP OF CREEP-BUCKLING TEST RIG FOR PIPING COMPONENTS





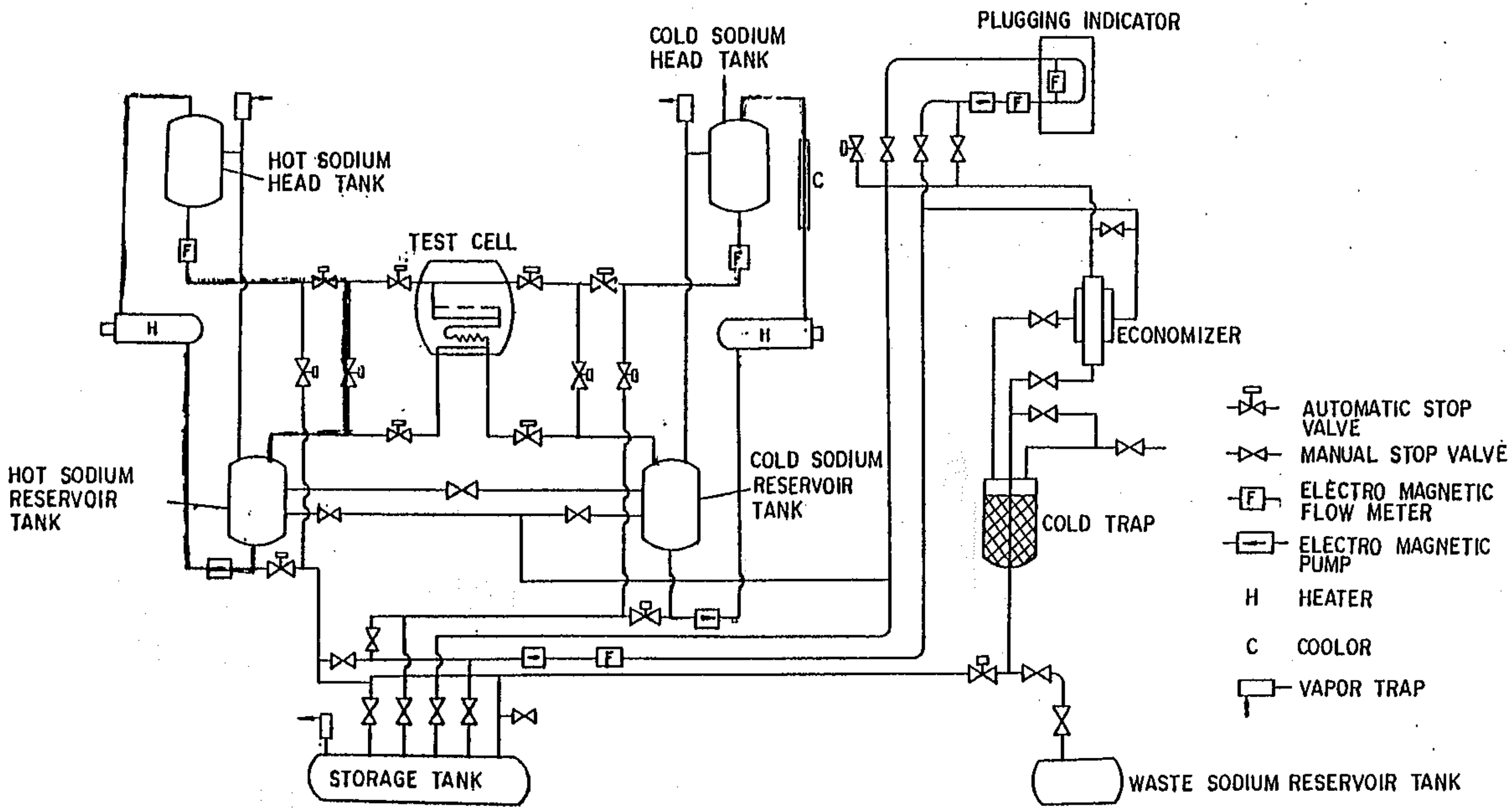
View from 'A'

Model of Hot Leg Piping of 'MONJU'

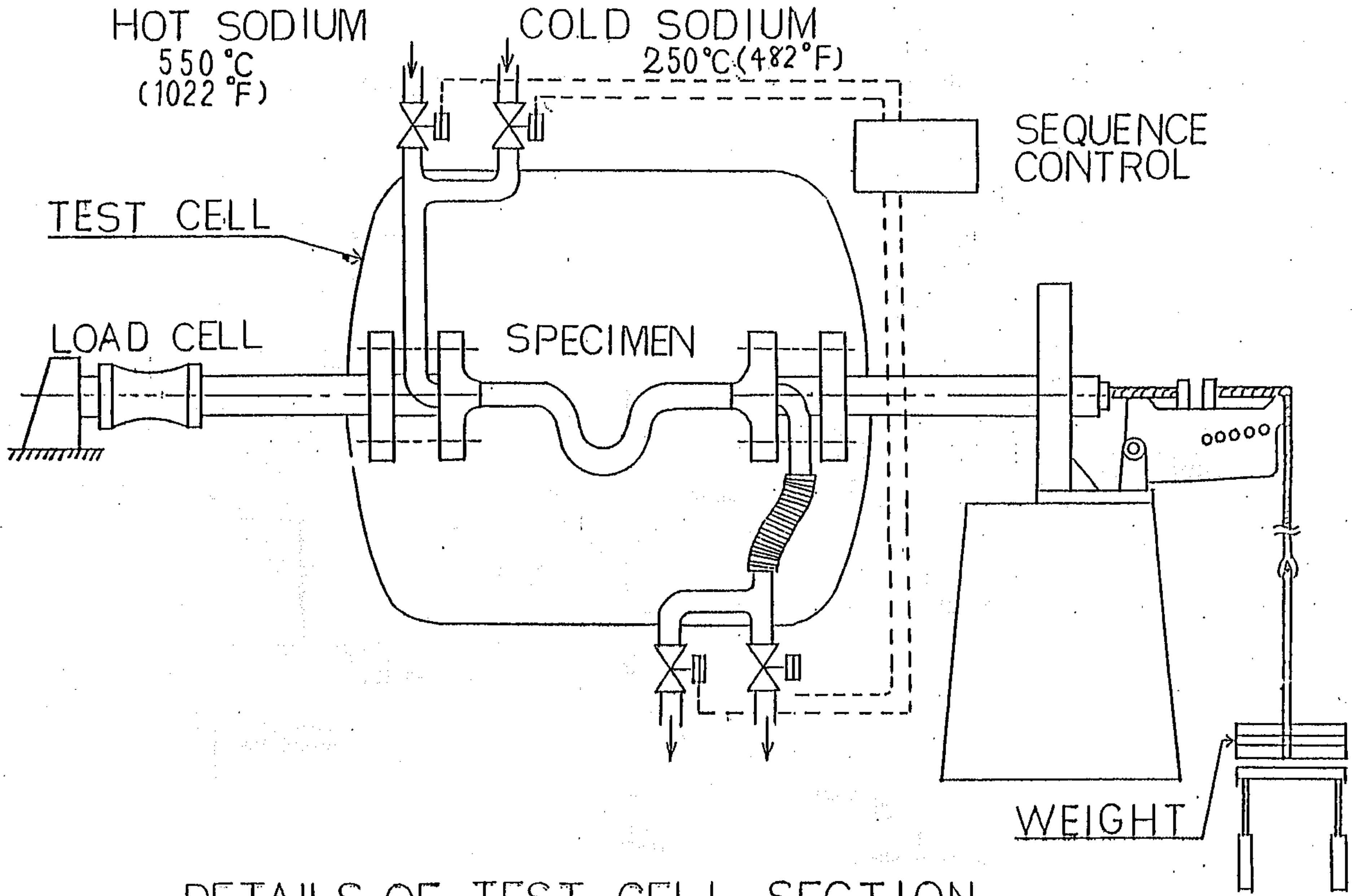


Small Thermal Shock Test Loop

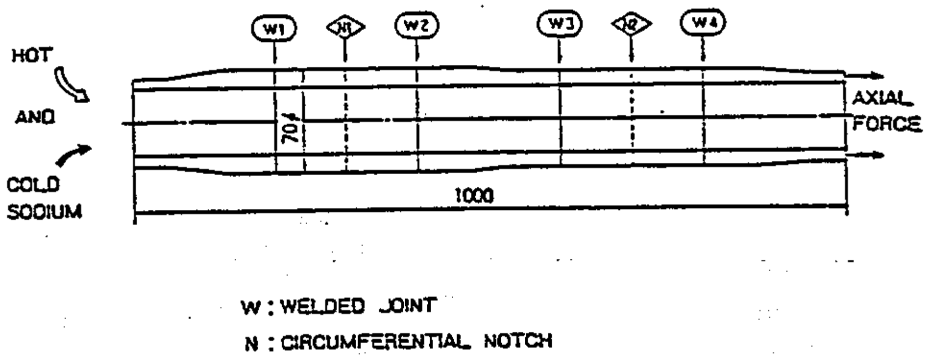
8)-18



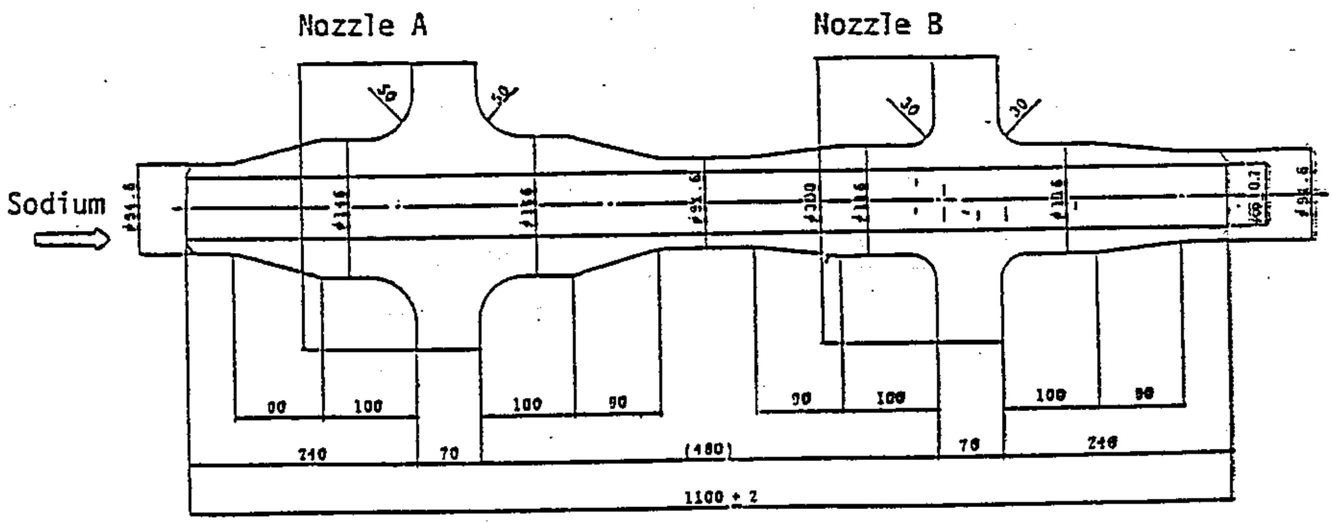
FLOW DIAGRAM OF SODIUM PIPING THERMAL TRANSIENT TEST LOOP



DETAILS OF TEST CELL SECTION

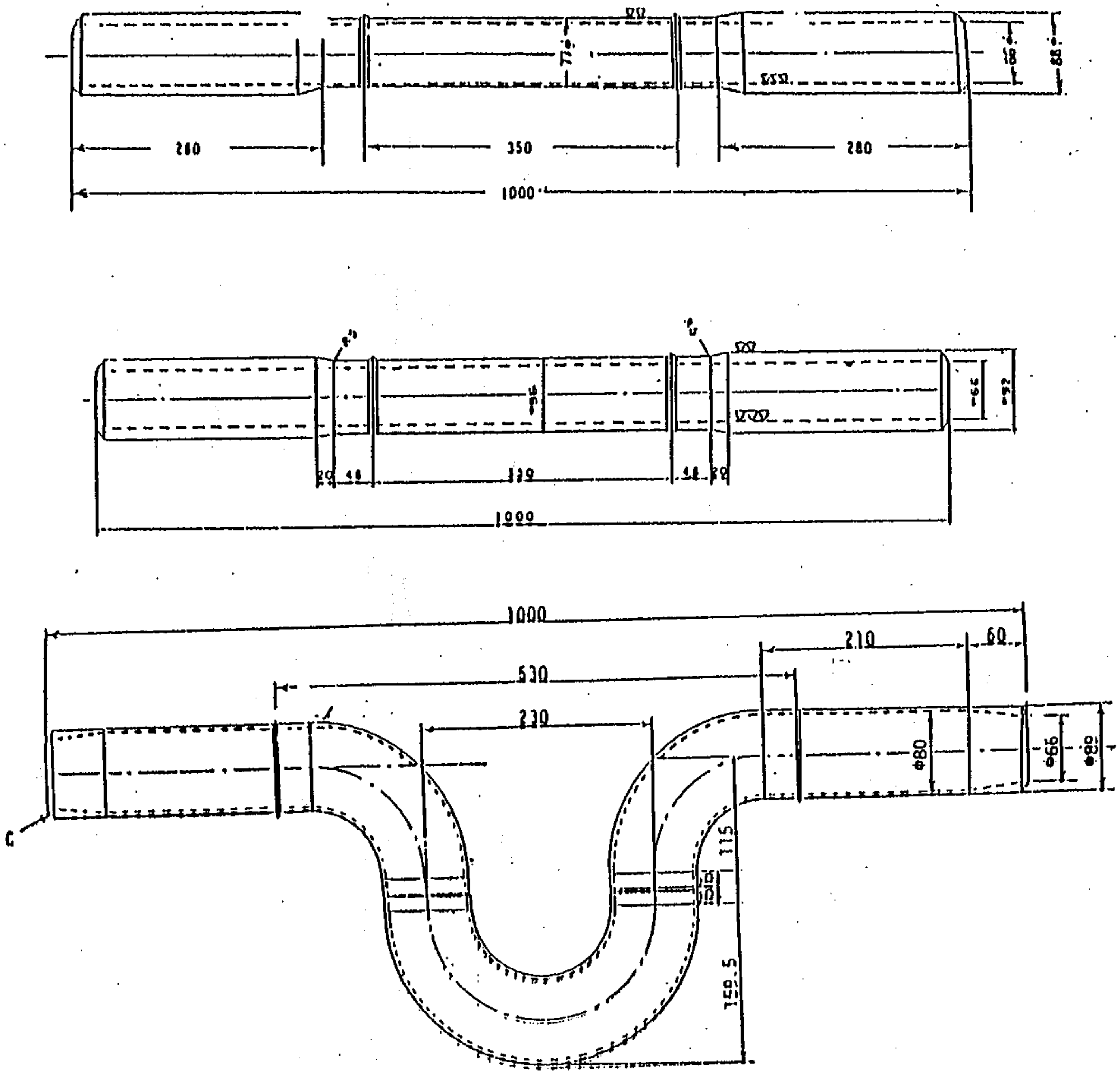


(A) Welded Joint Model

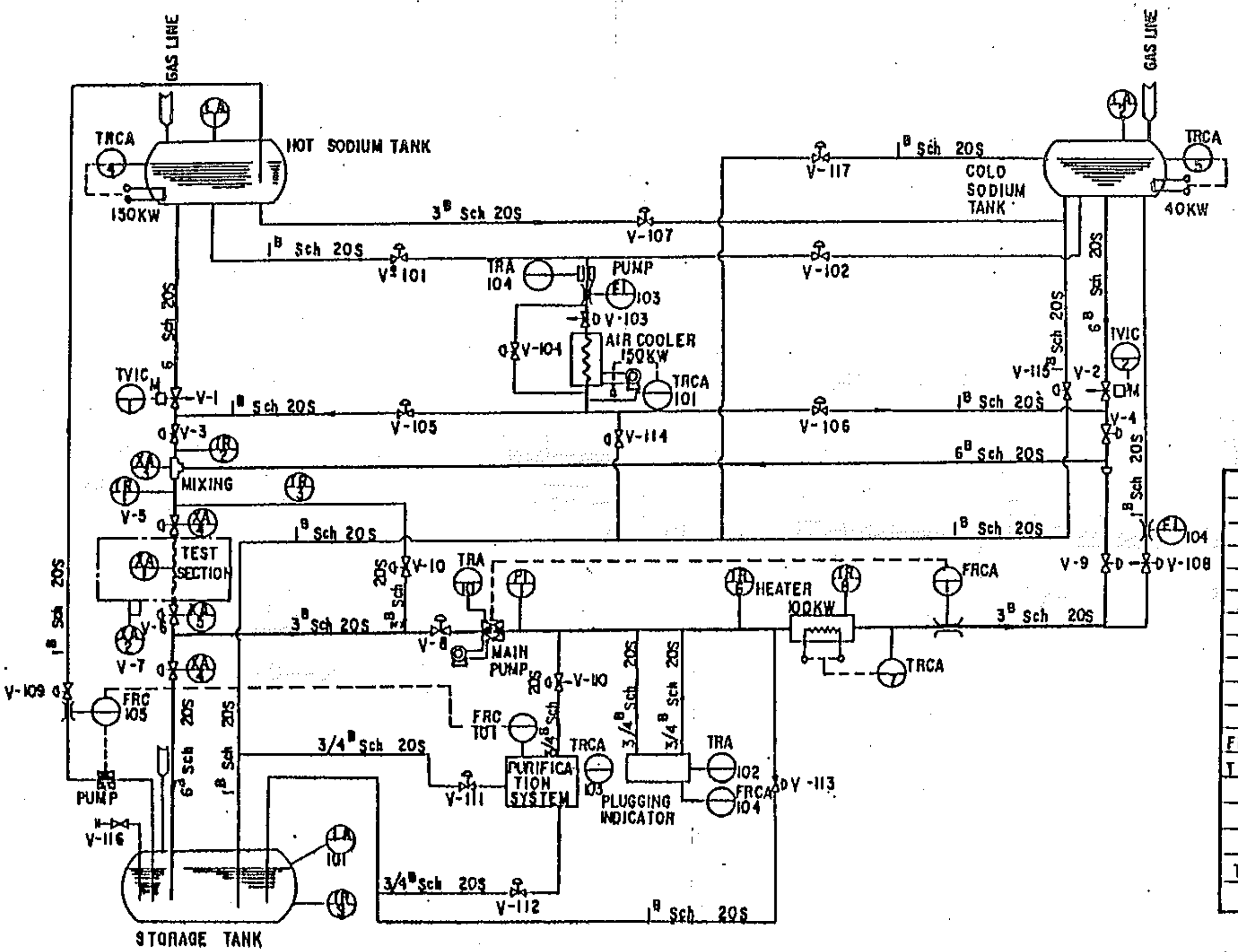


(B) Nozzle Model

Models of Thermal Fatigue Test

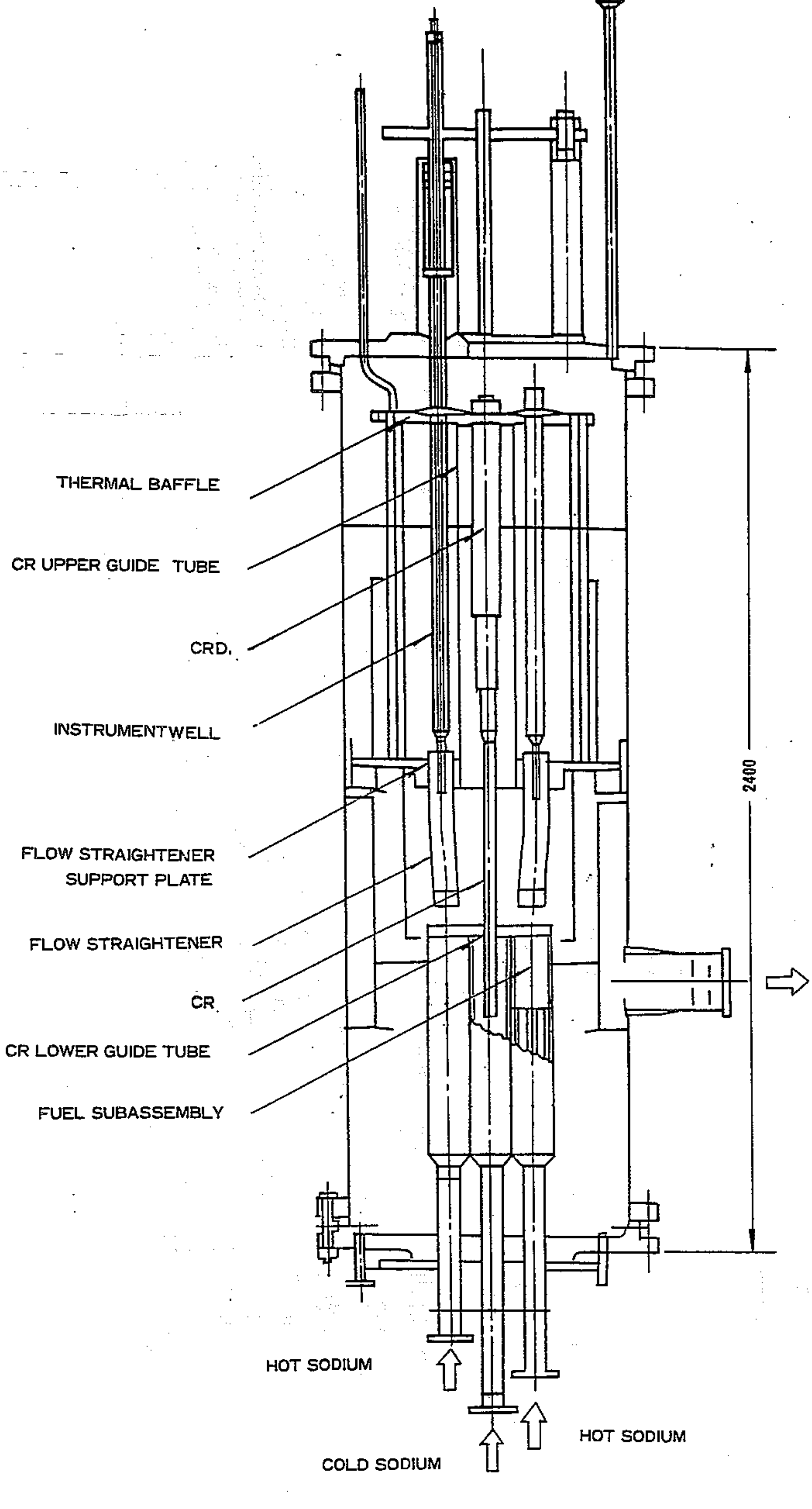


Models of Racheting Test



———	MAIN SODIUM LINE
———	AUX. SODIUM LINE
- - - -	CONTROL LINE
	AIR OPERATE VALVE
	MOTOR OPERATE VALVE
	MANUAL OPERATE VALVE
	DUMPER
	E.M. DUMP
	E.M. FLOW METER
	BLOWER
	METER
FRCA	FLOW RECORD, CONTROL ANNOUNCIATOR
TRCA	TEMP RECORD, CONTROL ANNOUNCIATOR
LA	LEVEL ANNOUNCIATOR
PI	PRESSURE INDICATOR
XA	LEAK DETECTOR
TVIC	VALVE POSITION CONTROLER
	FLOW CONTROL VALVE

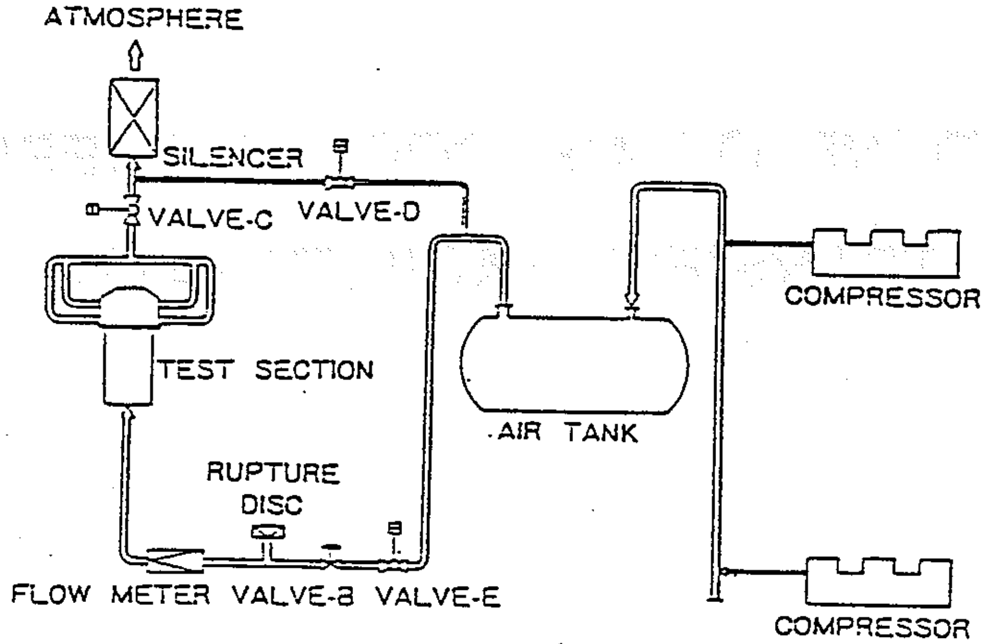
FLOW DIAGRAM OF THERMAL SHOCK TEST LOOP



7-Assembly Sodium Test Model(Thermal Striping Test)

OUTLINE OF AIR COOLED THERMAL TRANSIENT FACILITY (ATTF)

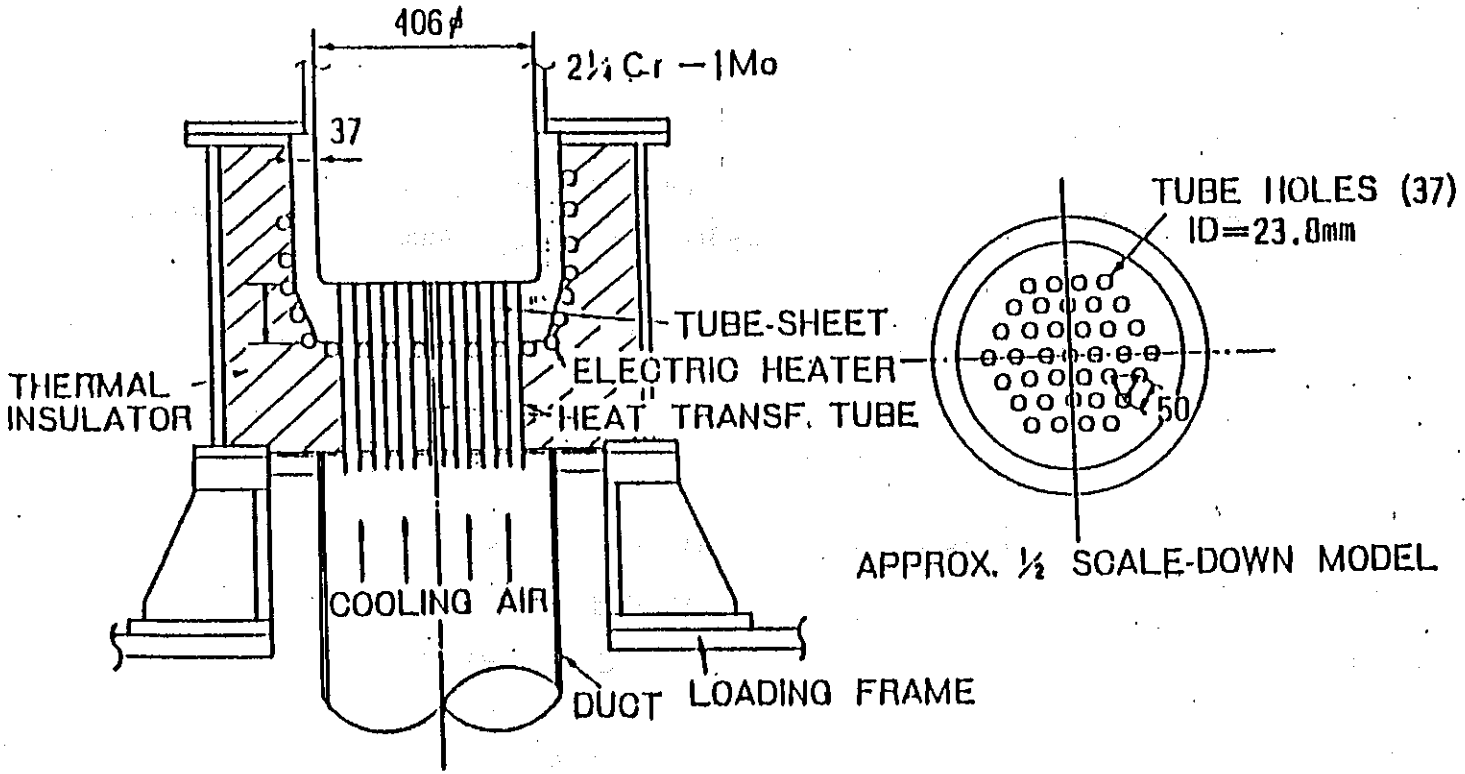
Power Reactor and Nuclear Fuel Development Corporation



FACILITY SPECIFICATIONS

TEST TEMPERATURE	550℃ MAX
TEST PRESSURE	8 kg / cm^2 MAX
MAIN PIPING SIZE	8 -IN
COMPRESSED AIR SUPPLY	35kg / cm^2G
THERMAL TRANSIENT	550℃ → 150℃
(DOWN RAMP)	IN ~ 4 MIN
MECHANICAL LOAD	TO BE CONSIDERED

AIR-COOLED THERMAL TRANSIENT FACILITY (ATTF)

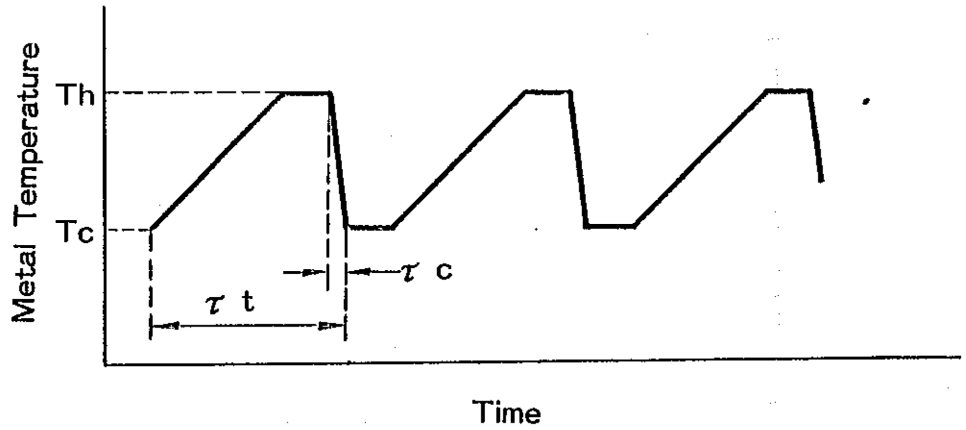


EVAPORATOR TUBE-SHEET STRUCTURE
TO BE TESTED AT ATTF

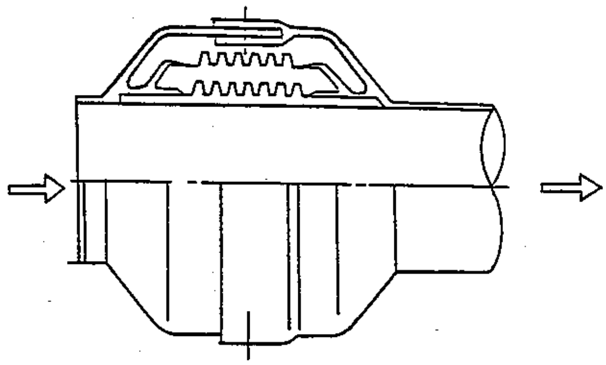
SENSORS

	Behavior Test				Failure Test		
	Bond Type				Weld Type		
	Test	Gage Length	Gage Base Length	Max. Temperature	Gage Length	Gage Base Length	Max. Temperature
Strain	B-1	1 mm	4 mm	250 ℃	10 mm	14 mm	600 ℃
	B-2	2 mm	8 mm	350 ℃			
Temperature	• 0.5 mmϕ Thermocouple • 1 mmϕ Thermocouple				0.5 mmϕ Thermocouple		

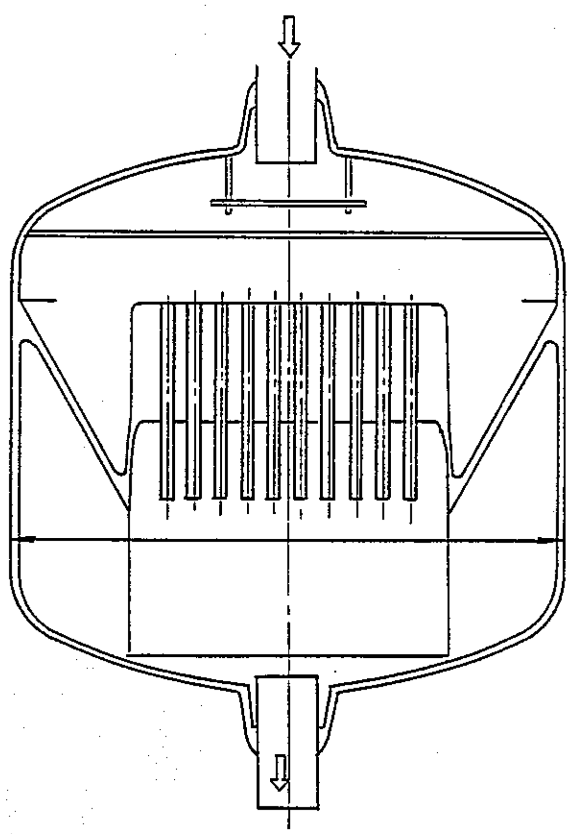
TEST CONDITION



Test	Th(℃)	Tc(℃)	τ t(H)	τ c(MIN)	N (cycles)
Behavior Test (B-1)	260	0	12	4	10cyc.× 4 levels
Behavior Test (B-2)	350	50	12	4	〃
Failure Test	550	100	4	4	1000～2000



CANDIDATE TEST ARTICLE 2 (EXPANSION JOINT)



CANDIDATE TEST ARTICLE 3 (TUBESHEET STRUCTURE)

DESCRIPTION OF 'THERMAL TRANSIENT TEST FACILITY FOR STRUCTURES' (T T S)

8)-30

MAY, 1984

STRUCTURAL ENGINEERING SECTION
O-ARAI ENGINEERING CENTER
P N C

OBJECTIVE OF TESTING

(A) DEMONSTRATE STRUCTURAL INTEGRITY OF CRITICAL STRUCTURES UNDER MORE SEVERE LOADINGS THAN ACTUAL ONES ENCOUNTERED BY LMFBR PLANT COMPONENTS.

(B) VALIDATE CURRENT STRUCTURAL DESIGN CRITERIA BY CONFIRMING INHERENT DESIGN MARGIN TO FAILURE INCORPORATED IN DESIGN CRITERIA

(C) DEVELOP MORE IMPROVED DESIGN CRITERIA FOR FUTURE LMFBR PLANTS

(D) CONFIRM EFFECTIVENESS AND STRUCTURAL INTEGRITY OF STRUCTURES FOR THERMAL MITIGATION

(TTS IS CAPABLE OF MEETING ANY ONE OR COMBINED REQUIREMENTS ABOVE)

CAPABILITIES

- IMPOSE VERY SEVERE THERMAL LOADINGS
  (HOT AND COLD SHOCK)

- IMPOSE MECHANICAL LOADINGS
  (COMBINED WITH THERMAL LOADINGS)

- ACCOMMODATE MEDIUM SIZE MODEL
  (~2 METERS IN DIAMETER)

- ACCOMPLISH TEST UNTIL FAILURE OCCURS
  (PROTECTION AGAINST SODIUM LEAKAGE,
  REPETITIVE APPLICATION OF SEVERE LOADINGS)

LOADING

- THERMAL TRANSIENTS

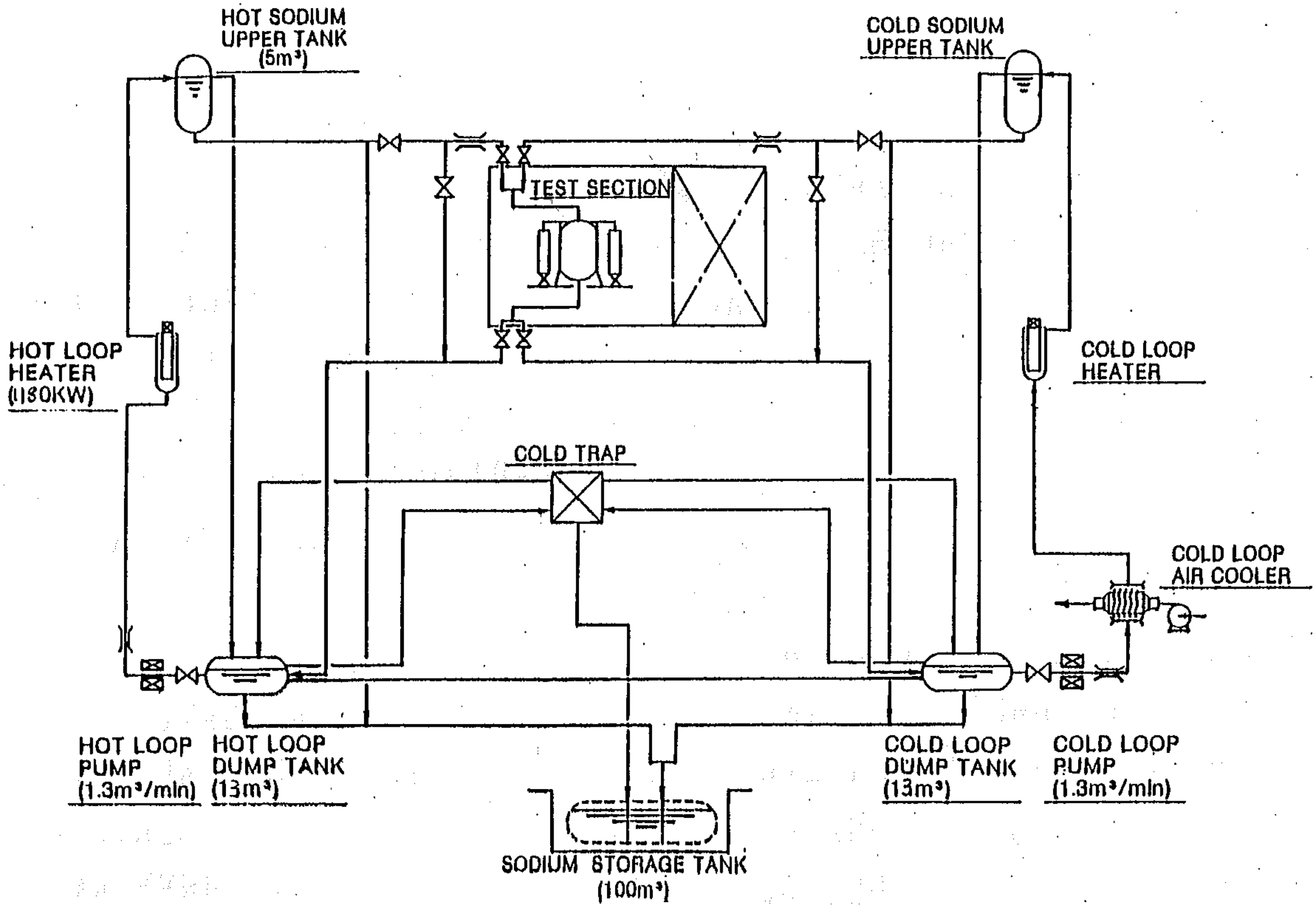
DOWN RAMPS	650 °C TO 250 °C IN 10 SEC
UP RAMPS	250 °C TO 650 °C IN 10 SEC
TIME FOR ONE CYCLE OF TRANSIENTS	APPROXIMATELY 60MIN/CYCLE FOR A THICK WALL (~50MM)-MEDIUM SIZE(~1M IN DIA.) VESSEL MODEL

- MECHANICAL FORCE

ELECTRO-HYDRAULIC ACTUATOR

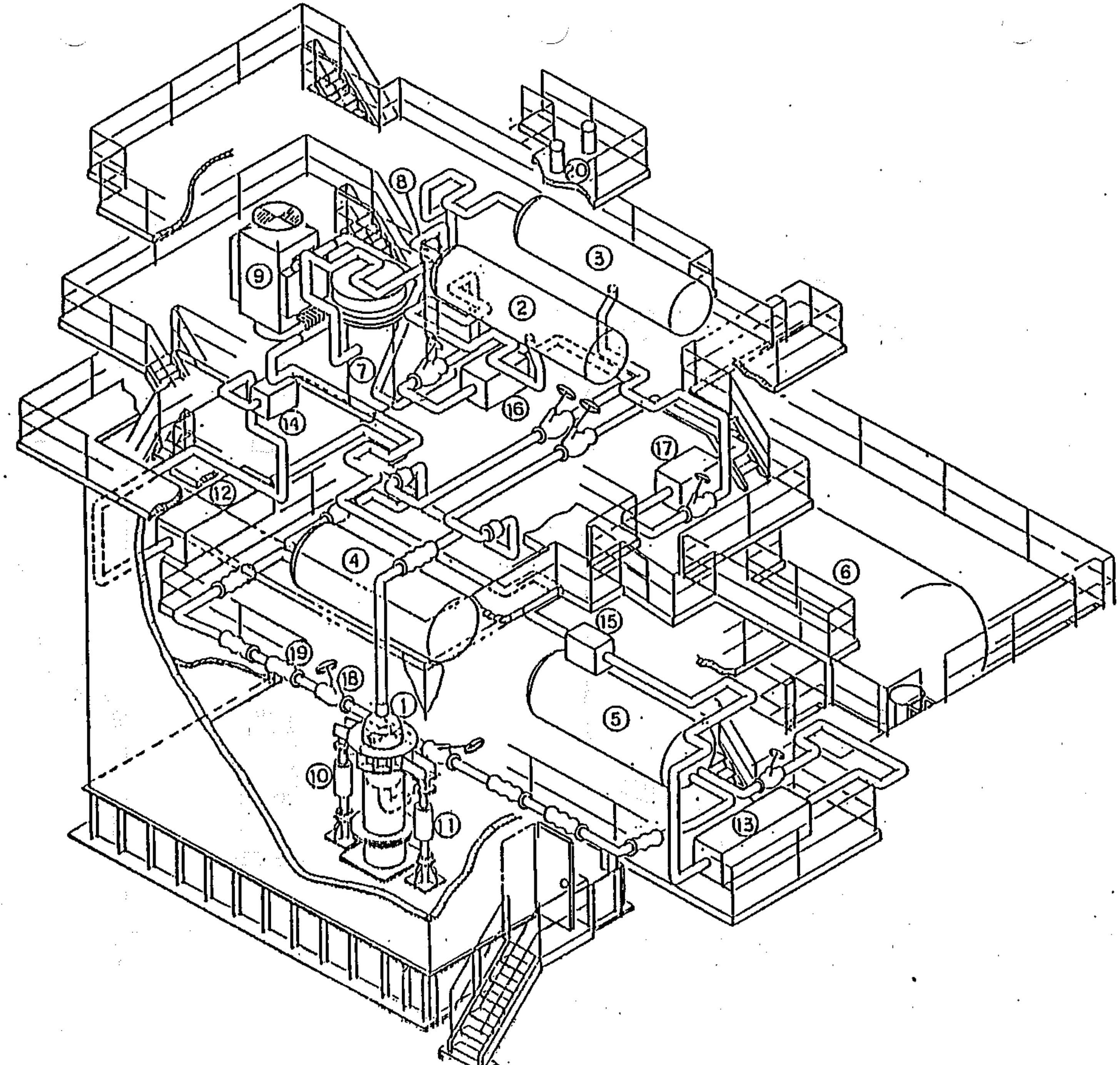
2 UNITS : EACH UNIT CONTROLLABLE INDEPENDENTLY

MAX. EXCITATION FORCE	±20TONS/UNIT
MAX. STROKE	±100MM
MAX. CYCLIC SPEED	0.4HZ/±100MM WHEN ONE UNIT WORKING 0.2HZ/±100MM WHEN TWO UNITS WORKING

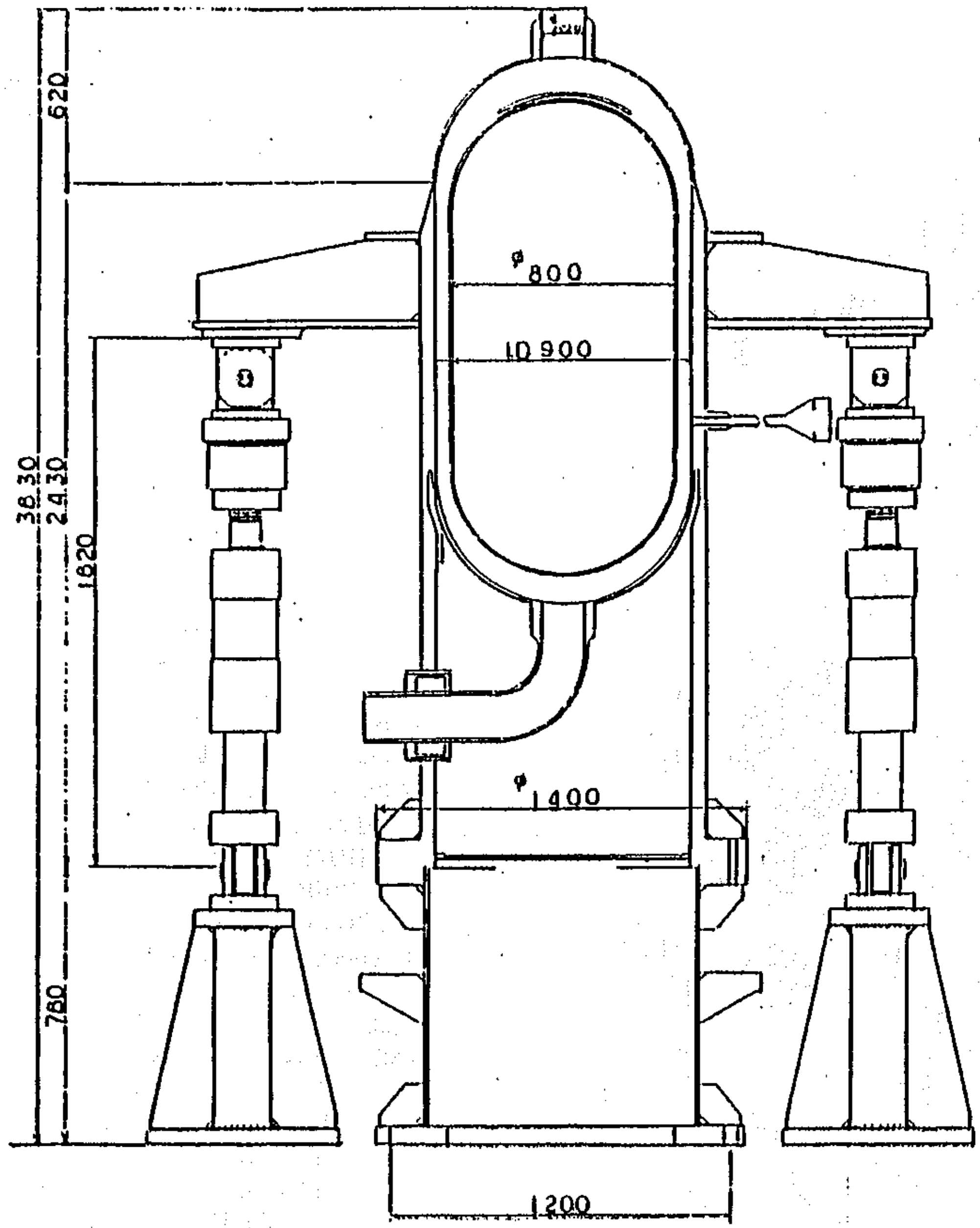


FLOW DIAGRAM

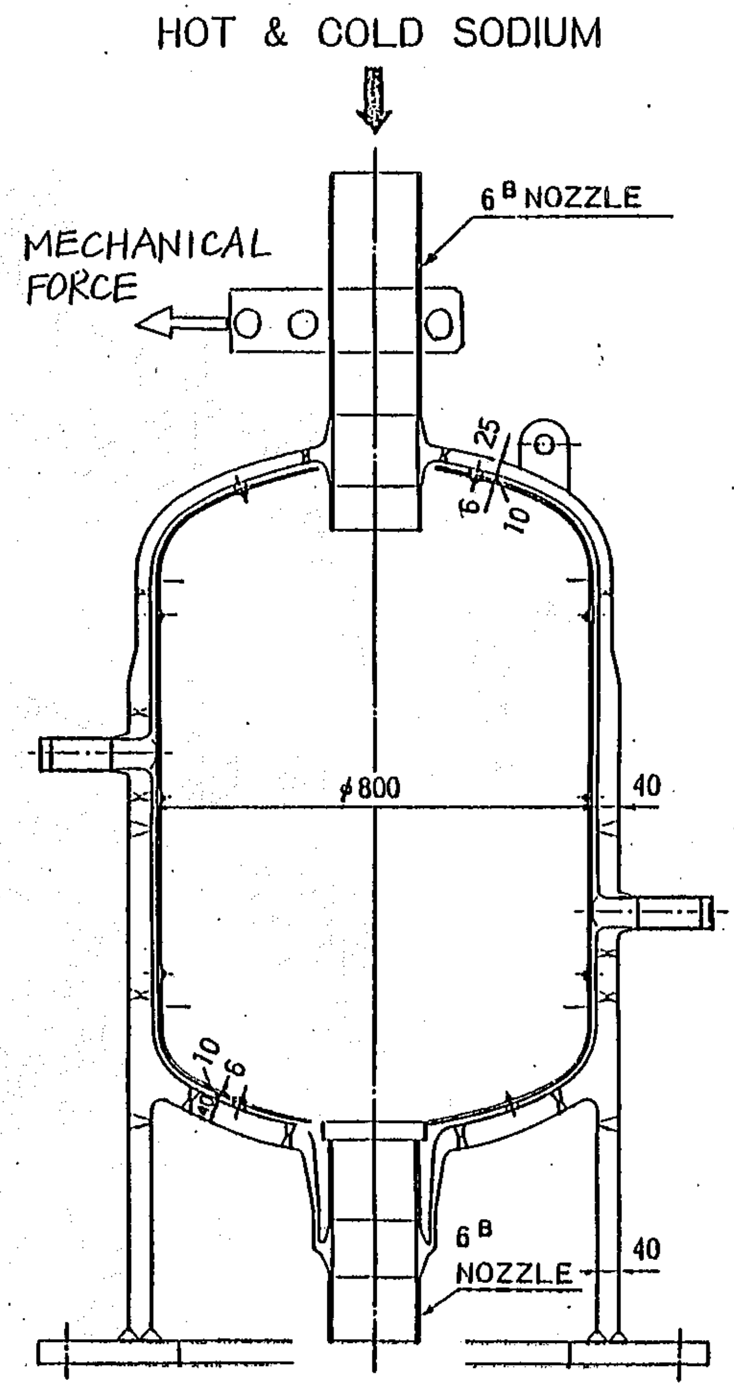
1. TEST MODEL
2. HEAD TANK ($5m^3$)
3. HEAD TANK ($5m^3$)
4. DUMP TANK ($13m^3$)
5. DUMP TANK ($13m^3$)
6. STORAGE TANK ($100m^3$)
7. ELEC. HEATER (1180kw)
8. ELEC. HEATER (60kw)
9. AIR COOLER (800000kcal/H)
10. ACTUATOR (±20ton, ±100mm)
11. ACTUATOR (±20ton, ±100mm)
12. ELEC. MAG. PUMP ($1.3m^3$/min)
13. ELEC. MAG. PUMP ($1.3m^3$/min)
14. FLOW METER ($1.5m^3$/min)
15. FLOW METER ($1.5m^3$/min)
16. FLOW METER ($5.0m^3$/min)
17. FLOW METER ($5.0m^3$/min)
18. VALVE
19. BELLOWS
20. VAPOR TRAP



THERMAL TRANSIENT TEST RIG FOR STRUCTURES (TTS)



TEST ARTICLE FOR TRIAL TEST



VESSEL MODEL FOR 'MONJU'

TEST FACILITY FOR STRUCTURAL MATERIAL TESTS

Power Reactor and Nuclear Fuel Development Corporation

The following table lists the in-air material test machines of PNC, including those lent to outside institutions.

		Facility-1	Facility-2	Facility-3	outside institutions	total
Tensile test machines				3		3
Charpy impact test machines				1		1
Relaxation test machines			2	2	4	8
Creep test machines	uniaxial		5	100	80	185
	biaxial	15				15
Fatigue test machines	uniaxial		2	9	14	25
	biaxial	1			1	2

The others : optical microscope; transmission electron microscope; scanning electron microscope; X-ray microanalyzer; ion microanalyzer; penetrometer; surface roughness tester; X-ray diffractometer; carbon, oxygen and hydrogen analyzer, and weight balance, etc..

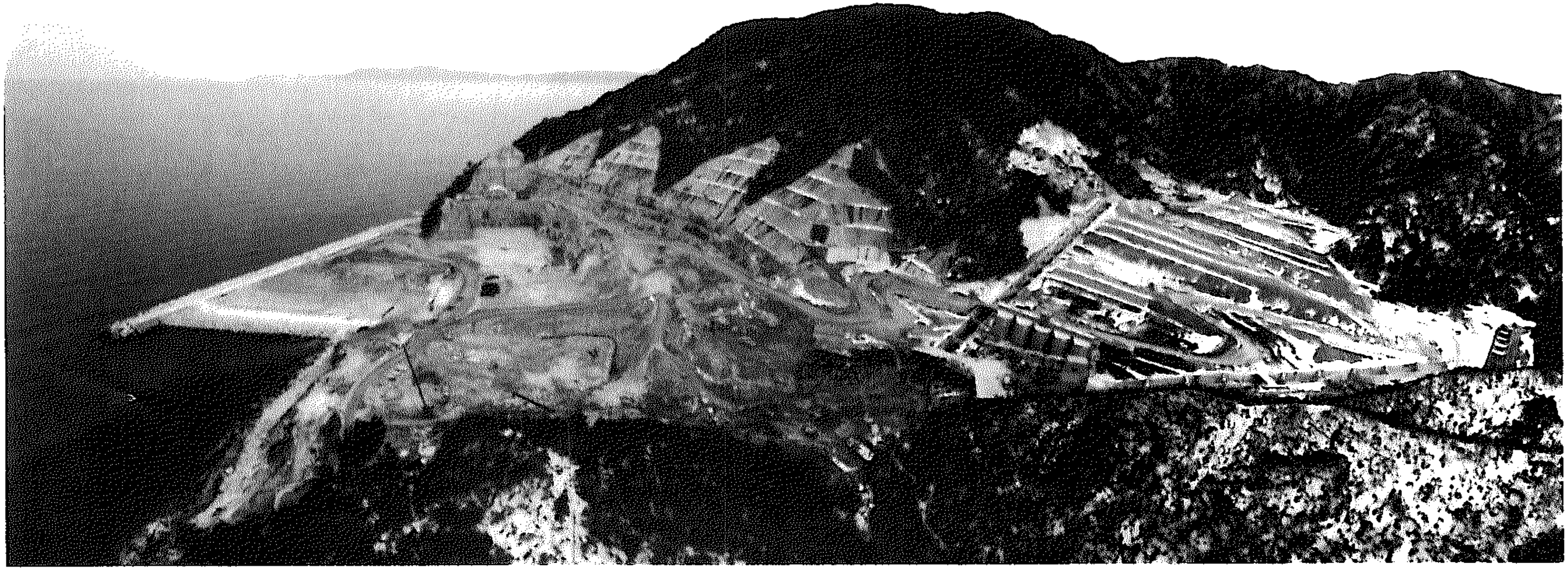
The following table lists the in-sodium material test machines of PNC

		Facility- 1	Facility- 2
Creep test machines	uniaxial	8	7
	biaxial	58	0
Fatigue test machines	uniaxial	2	2

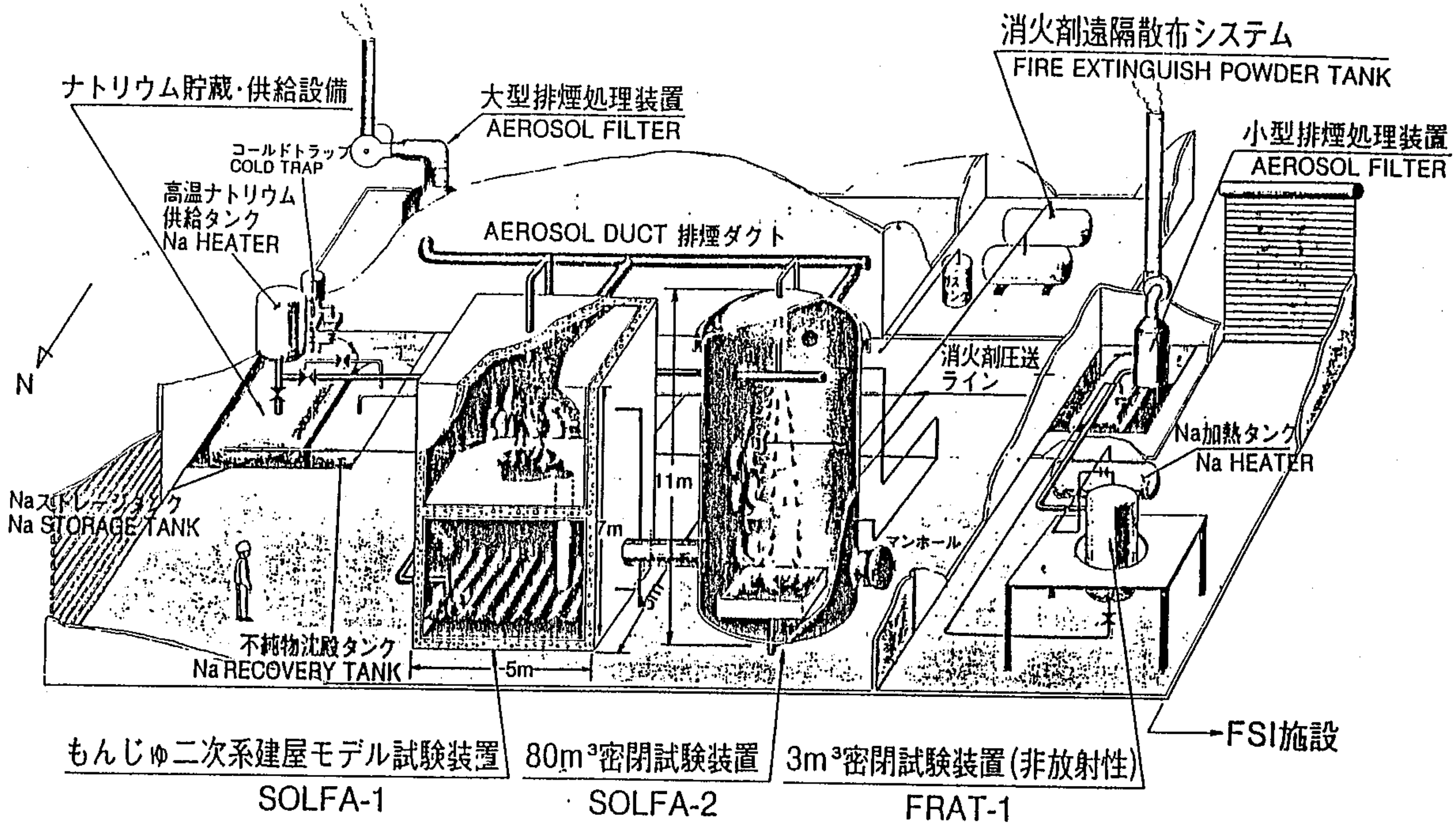
The following table lists the neutron irradiated material test machines of PNC.

Tensile Test Machine	2
Fatigue Test Machine	3
Creep Test Machine	15

The others : periscope, optical microscope, transmission electron microscope, scanning electron microscope, X-ray microanalyzer, ion microanalyzer, penetrometer, X-ray diffractometer, mass spectrometer, autoradiographic device, γ scanning apparatus, pin puncture apparatus, densimeter, melting point apparatus, thermal conductivity apparatus, etc.



"MONJU" SITE TSURUGA FEB, 1985

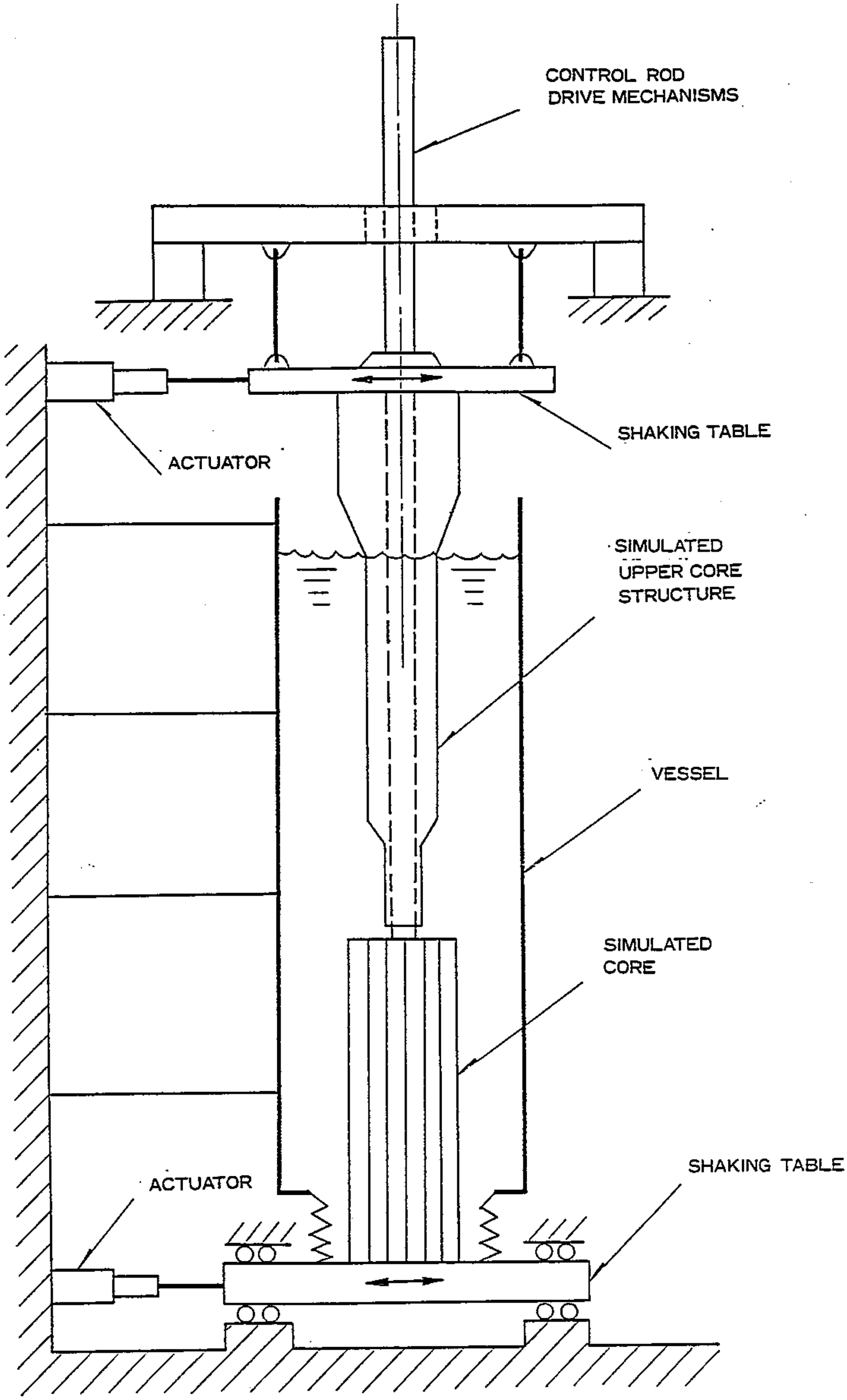


SOLFA (Sodium Leak, Fire, and Aerosol)

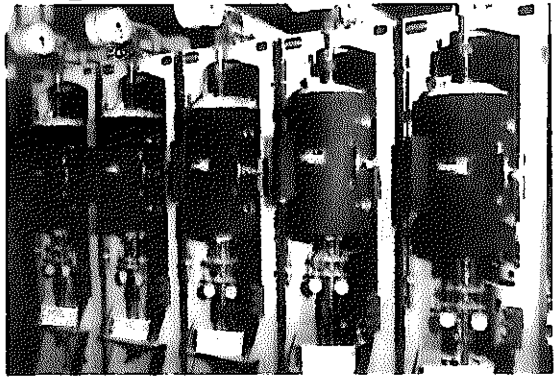
FRAT (Fission Product and Radioactive Aerosol Release Test)

(PSS-SFT-86)

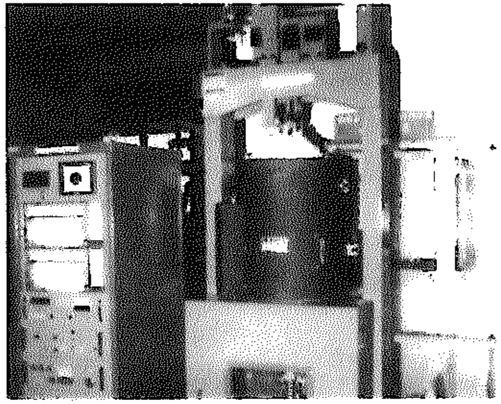
LARGE SCALE SODIUM FIRE TEST FACILITY
(SAPFIRE)
OARAI ENGINEERING CENTER, PNC



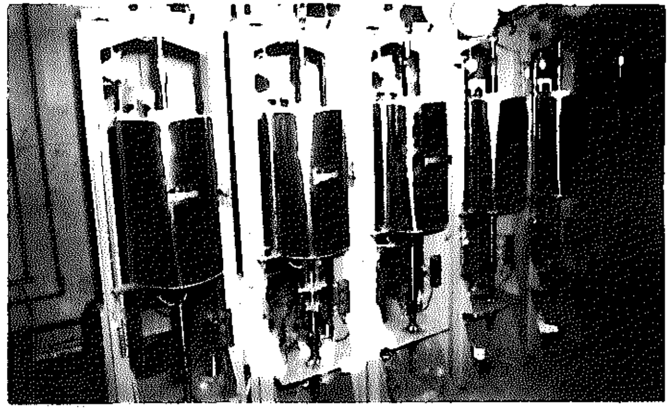
SCRAM RELIABILITY TEST UNDER SEISMIC CONDITION
(PNC OWN FACILITY, INSTALLED IN TAKASAGO, MITSUBISHI HI)



MITUBISHI H I

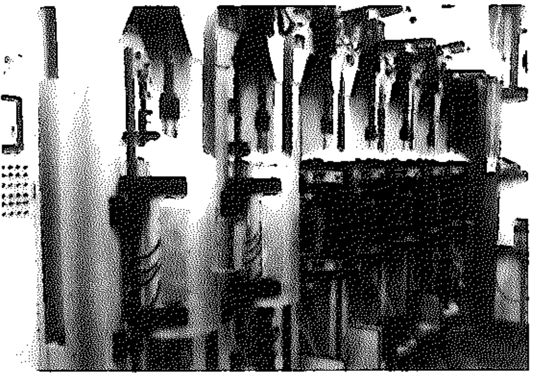


FUJI ELECTRIC

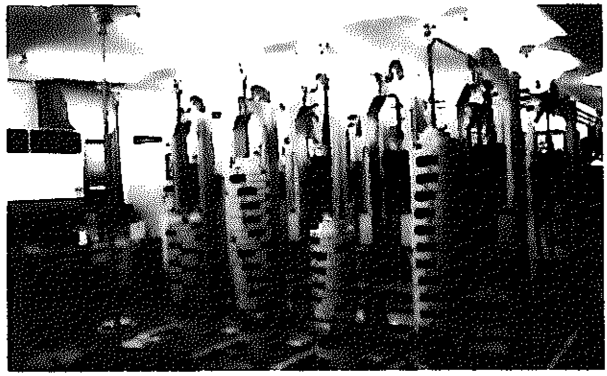


ISHIKAWAJIMA HARIMA H I

KAWASAKI H I



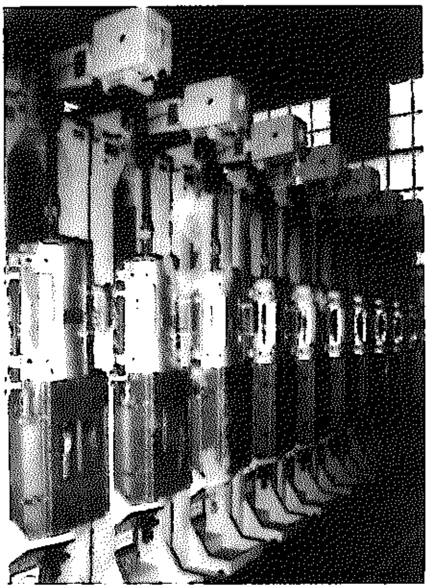
PNC ORGANIZED COLLABORATIVE WORK ON MATERIAL TEST TO ESTABLISH ELEVATED TEMPERATURE STRUCTURAL DESIGN GUIDE



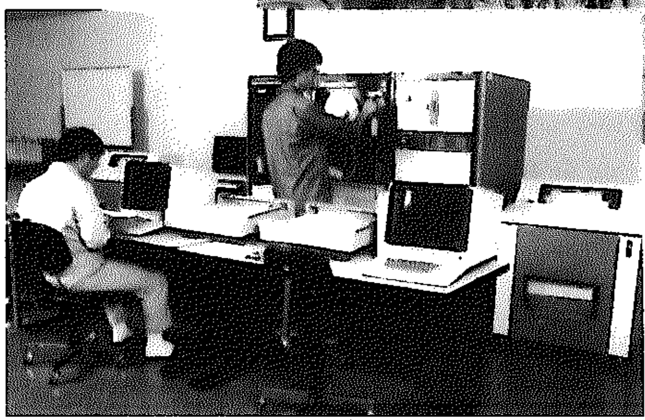
TOSHIBA

UNIAXIAL CREEP TEST MACHINE

HITACHI

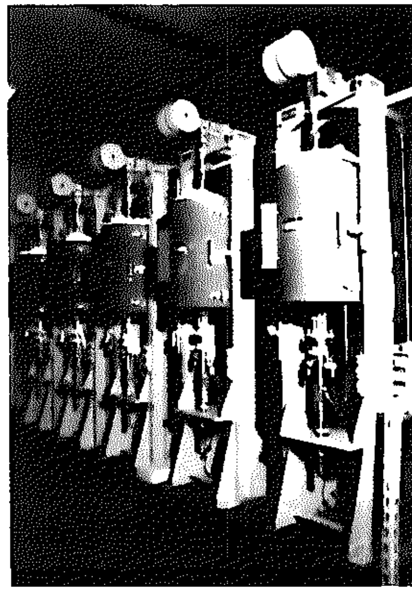


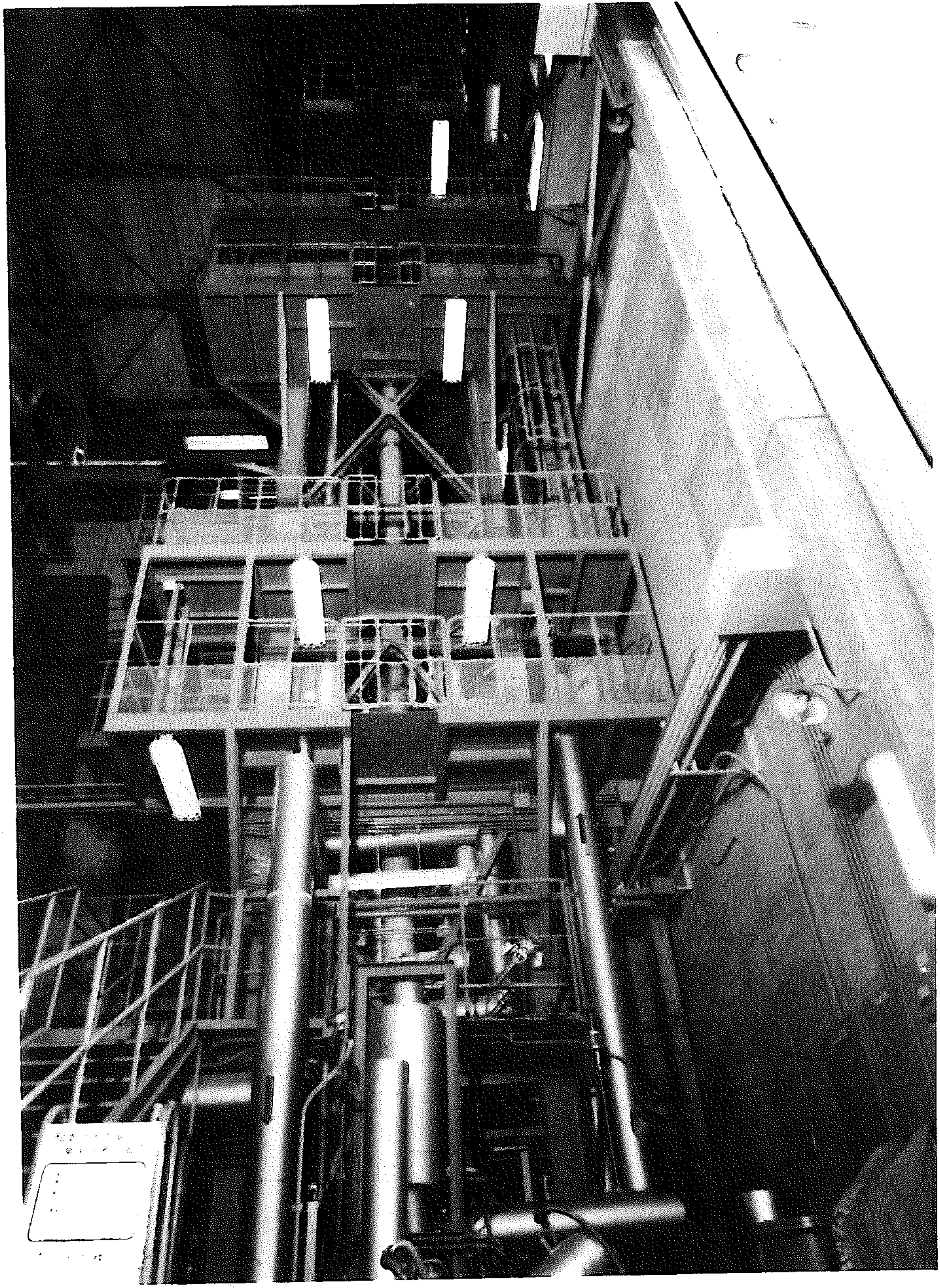
DATA ANALYSES SYSTEM BY COMPUTER SYSTEM



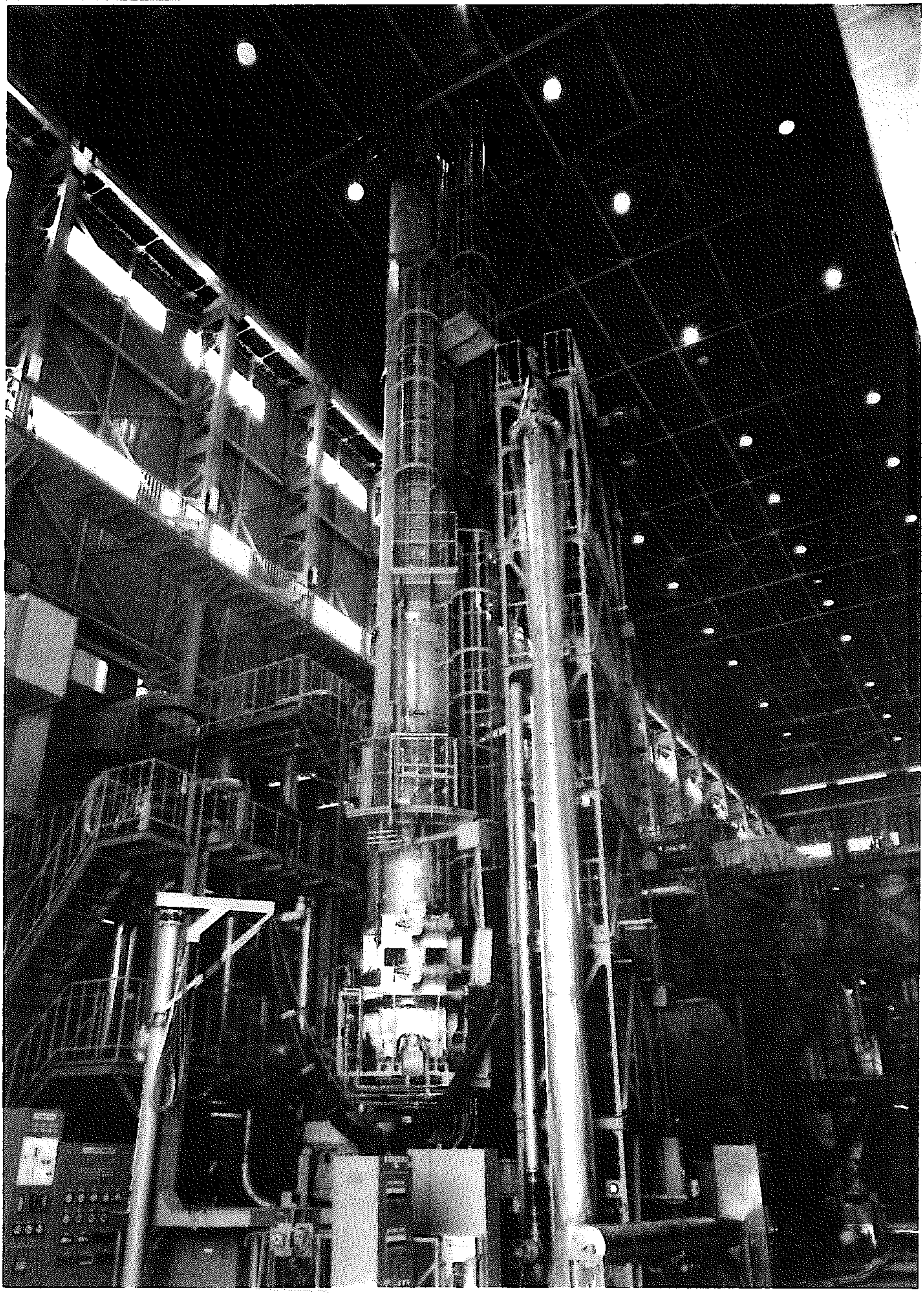
OARAI ENGINEERING CENTER PNC

BABCOCK HITACHI

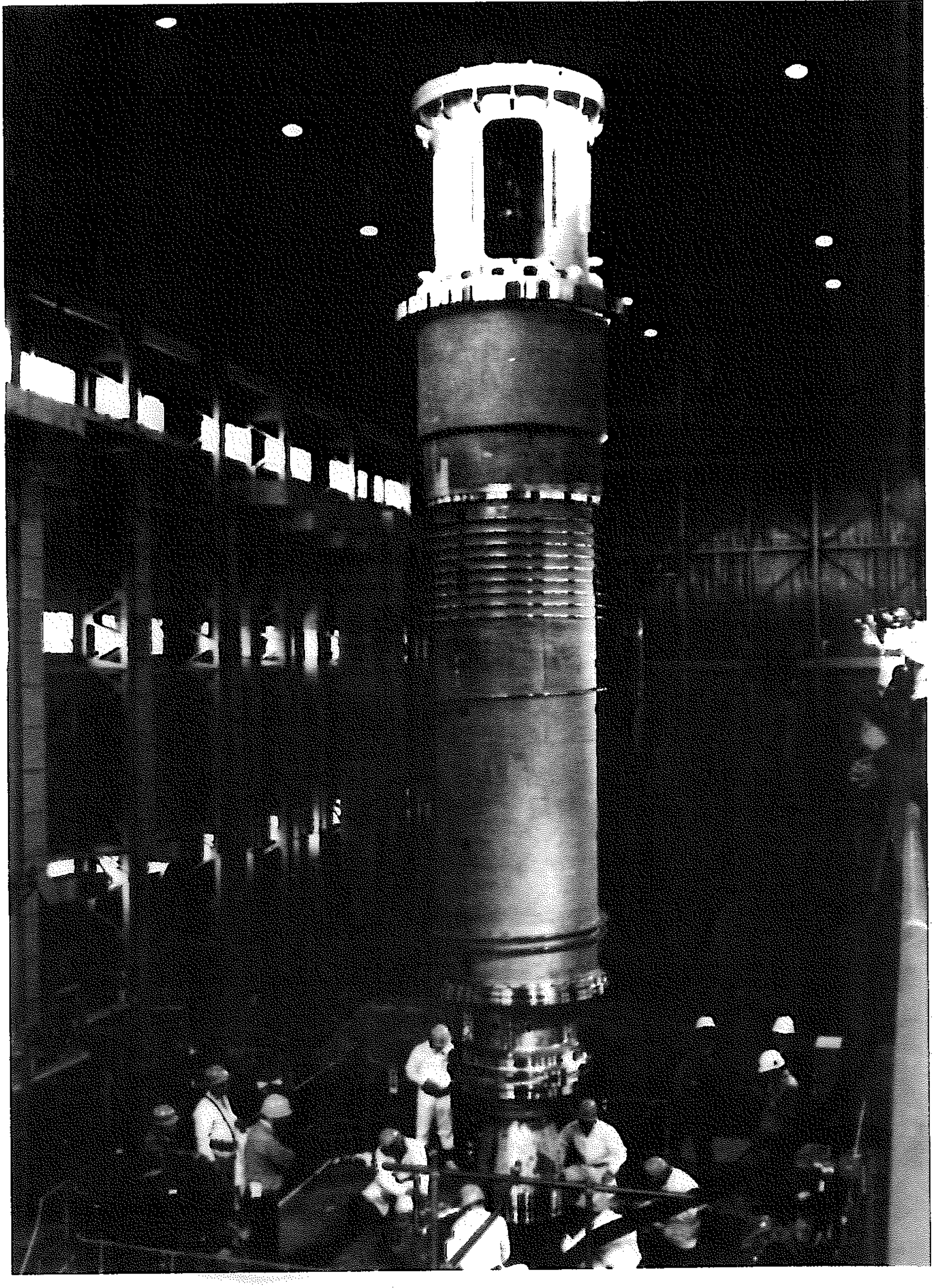




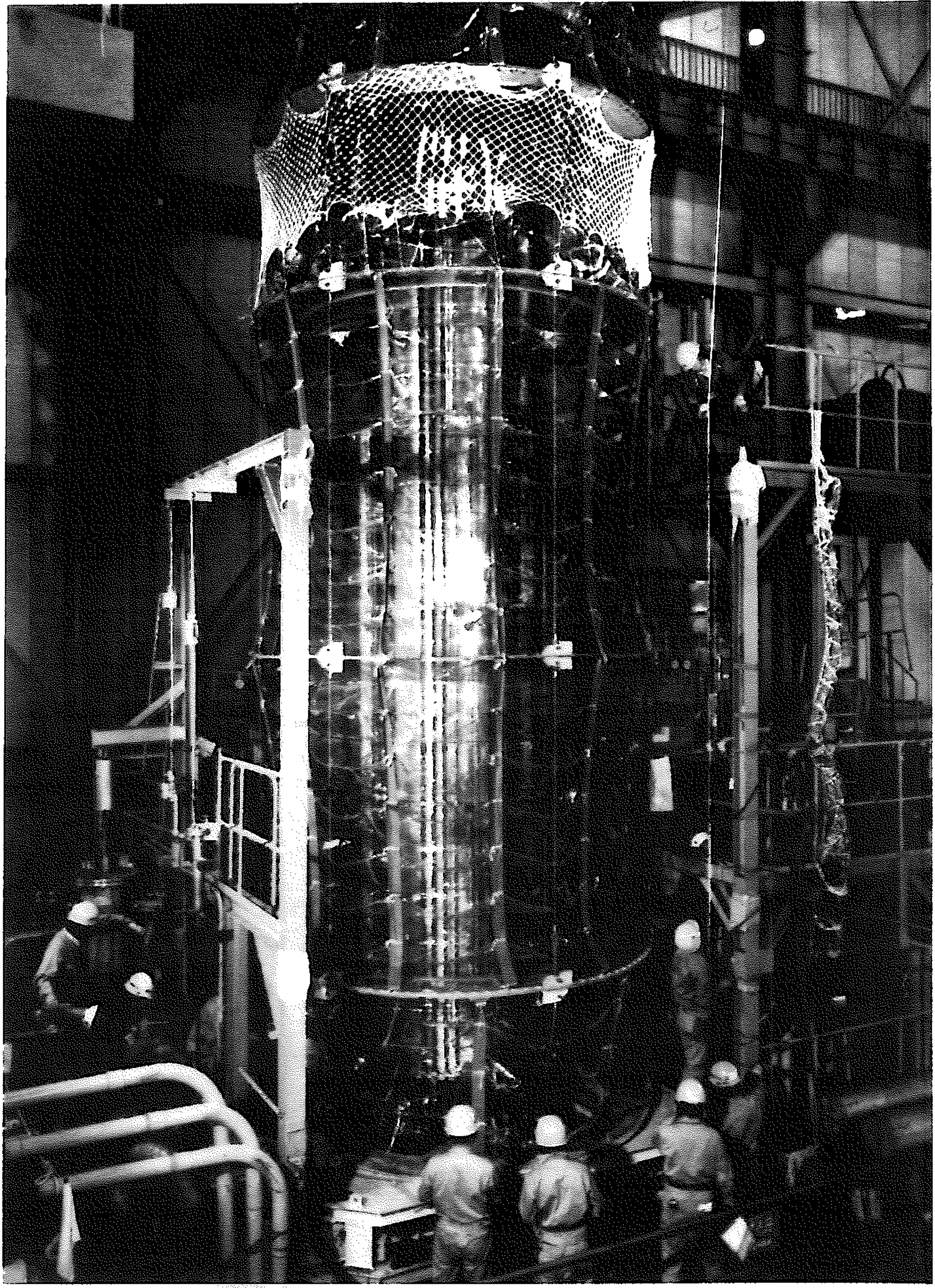
CONTROL ROD DRIVE MECHANISMS TEST FACILITY
OARAI ENGINEERING CENTER, PNC



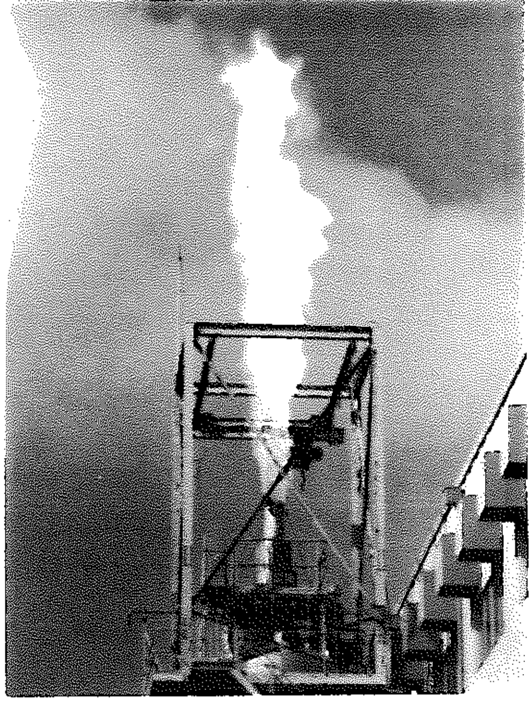
MONJU IN-VESSEL FUEL TRANSFER MACHINE ON
SODIUM TEST RIG
OARAI ENGINEERING CENTER, PNC

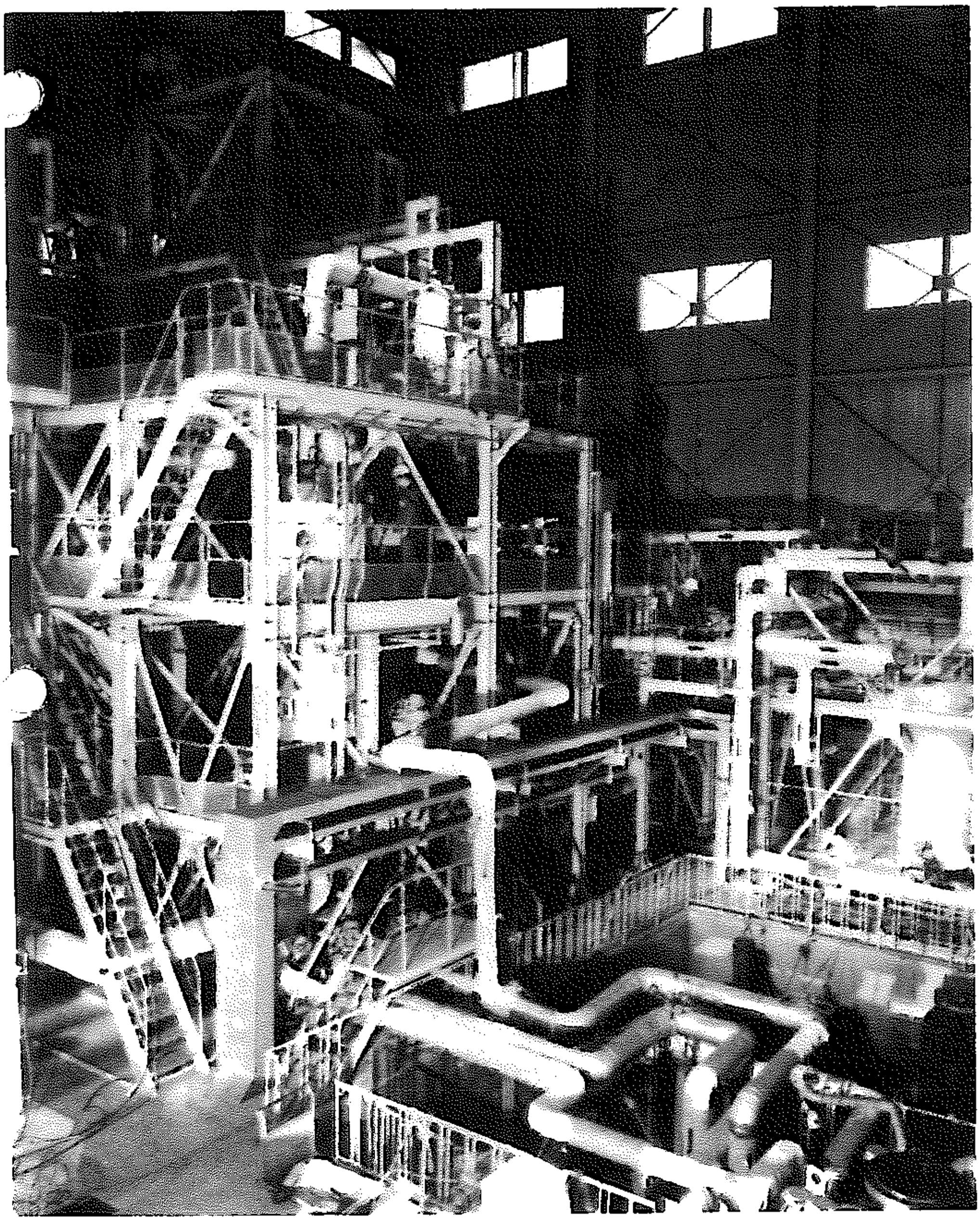


PROTOTYPE PUMP BEING INSTALLED FOR SODIUM
TEST AT
OARAI ENGINEERING CENTER, PNC

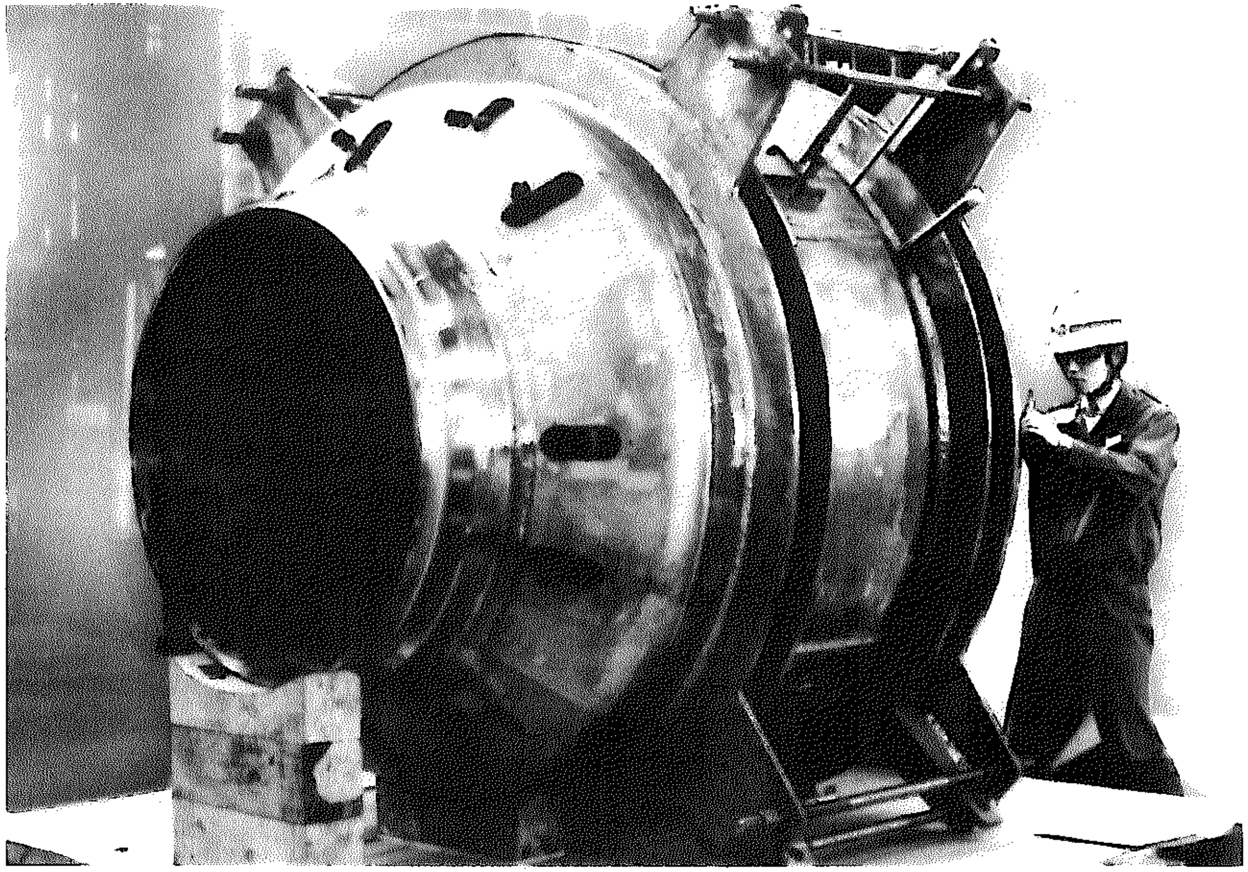


REMOVED TUBE BUNDLE OF EVAPORATOR AT 50MW
STEAM GENERATOR TEST FACILITY
OARAI ENGINEERING CENTER, PNC





STEAM GENERATOR SAFETY TEST FACILITY (SWAT-3)
OARAI ENGINEERING CENTER, PNC



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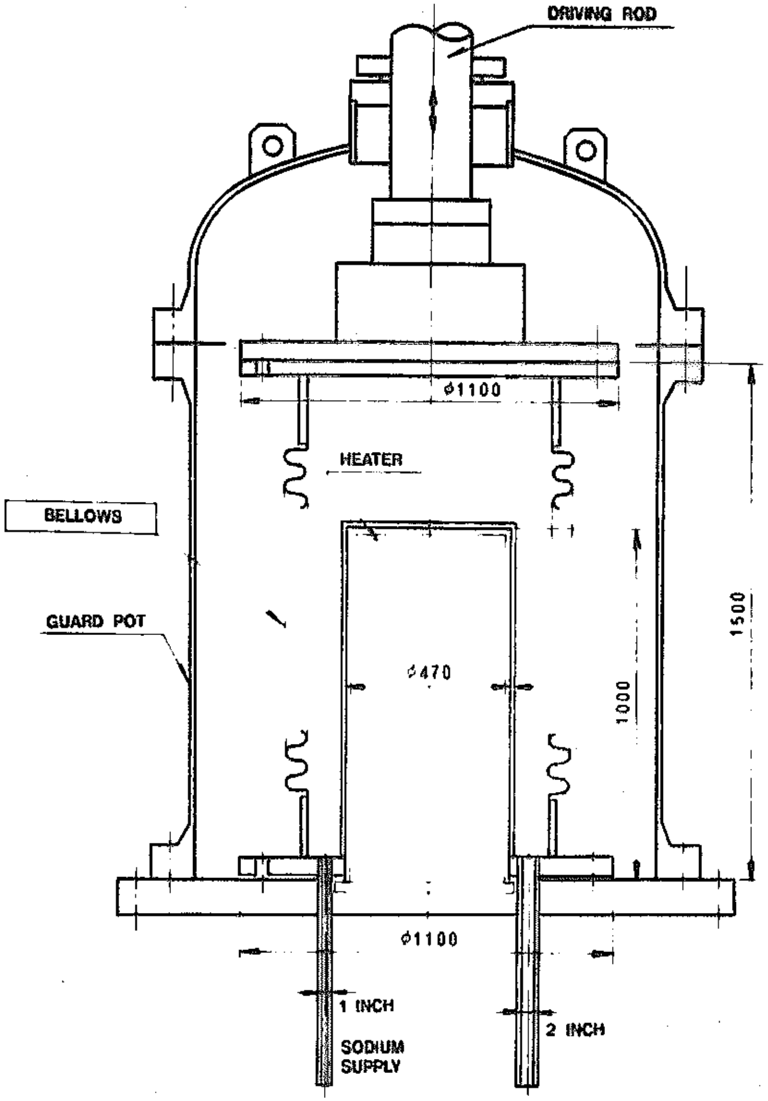
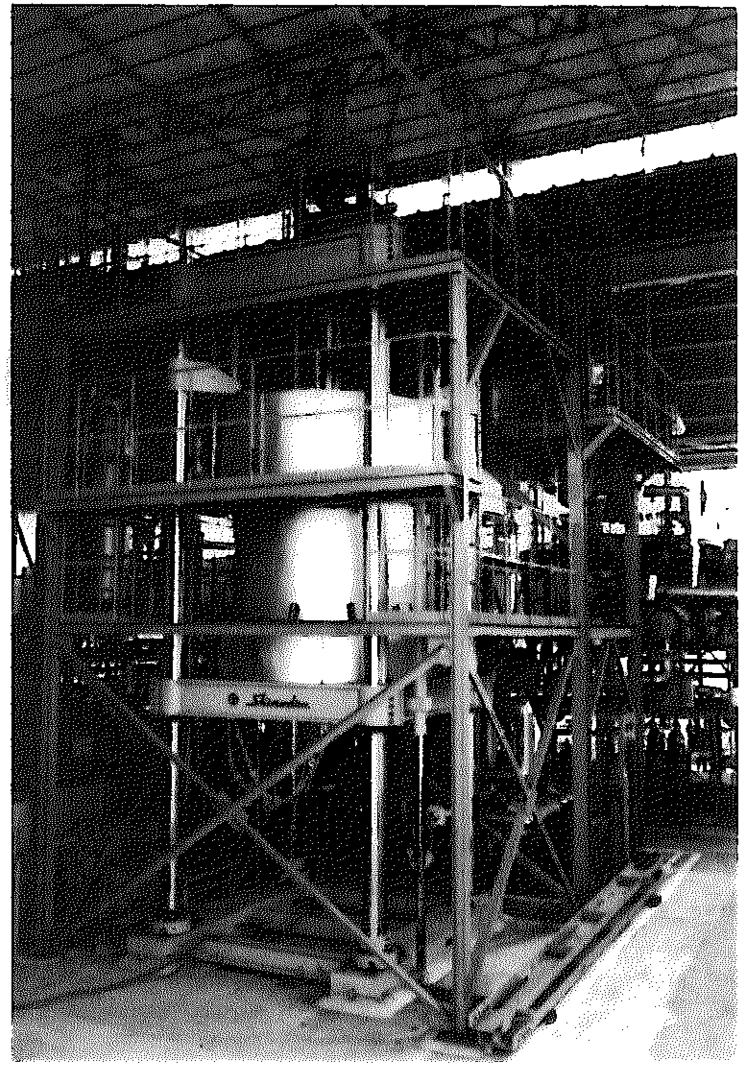
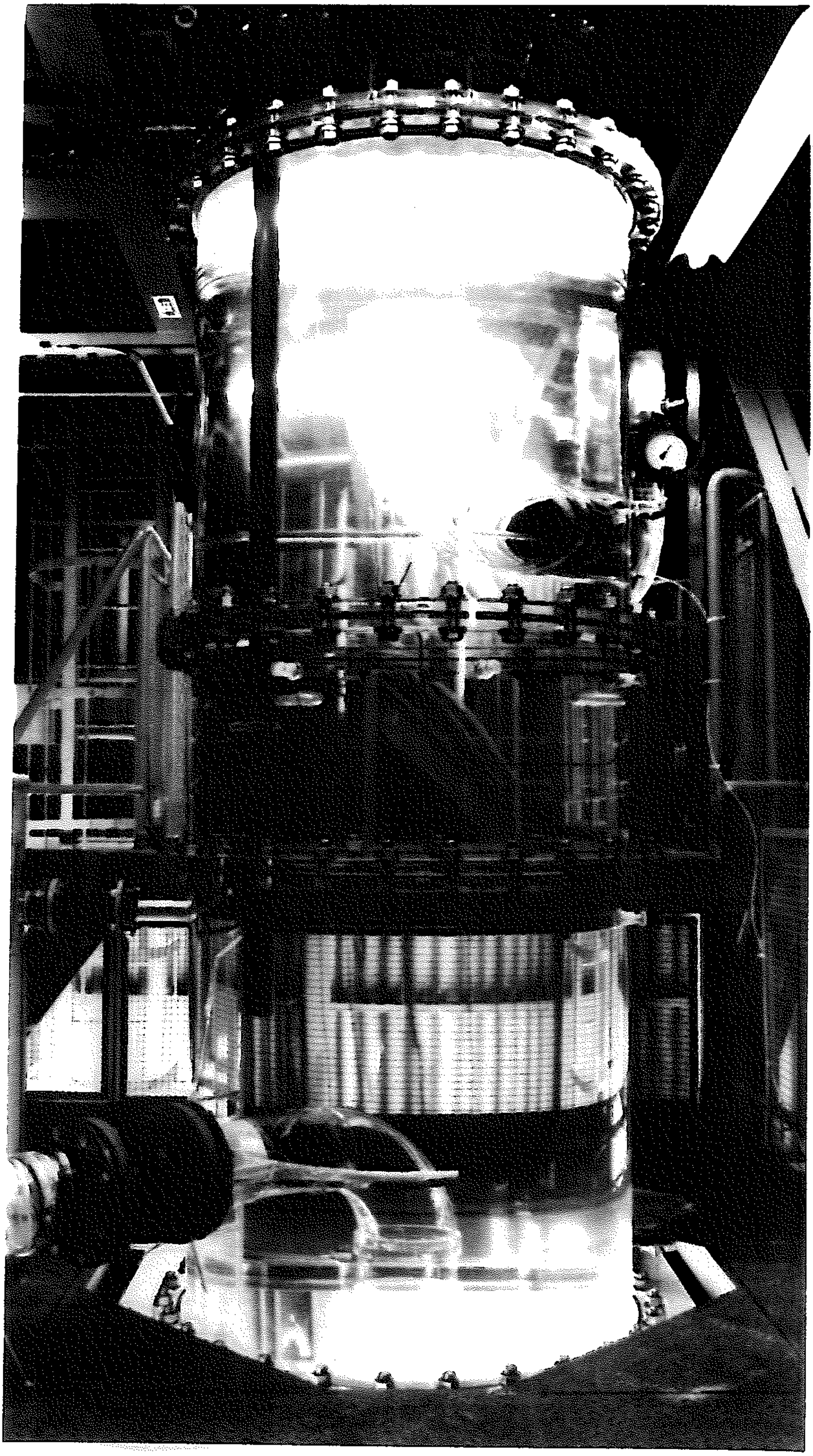
MODEL FLEXIBLE PIPING JOINT, 42 INCH JIMBAL, BELLOWS TYPE, TO BE INSTALLED FOR SODIUM TEST IN LARGE-SCALE PUMP TEST LOOP, OARAI ENGINEERING CENTER, PNC

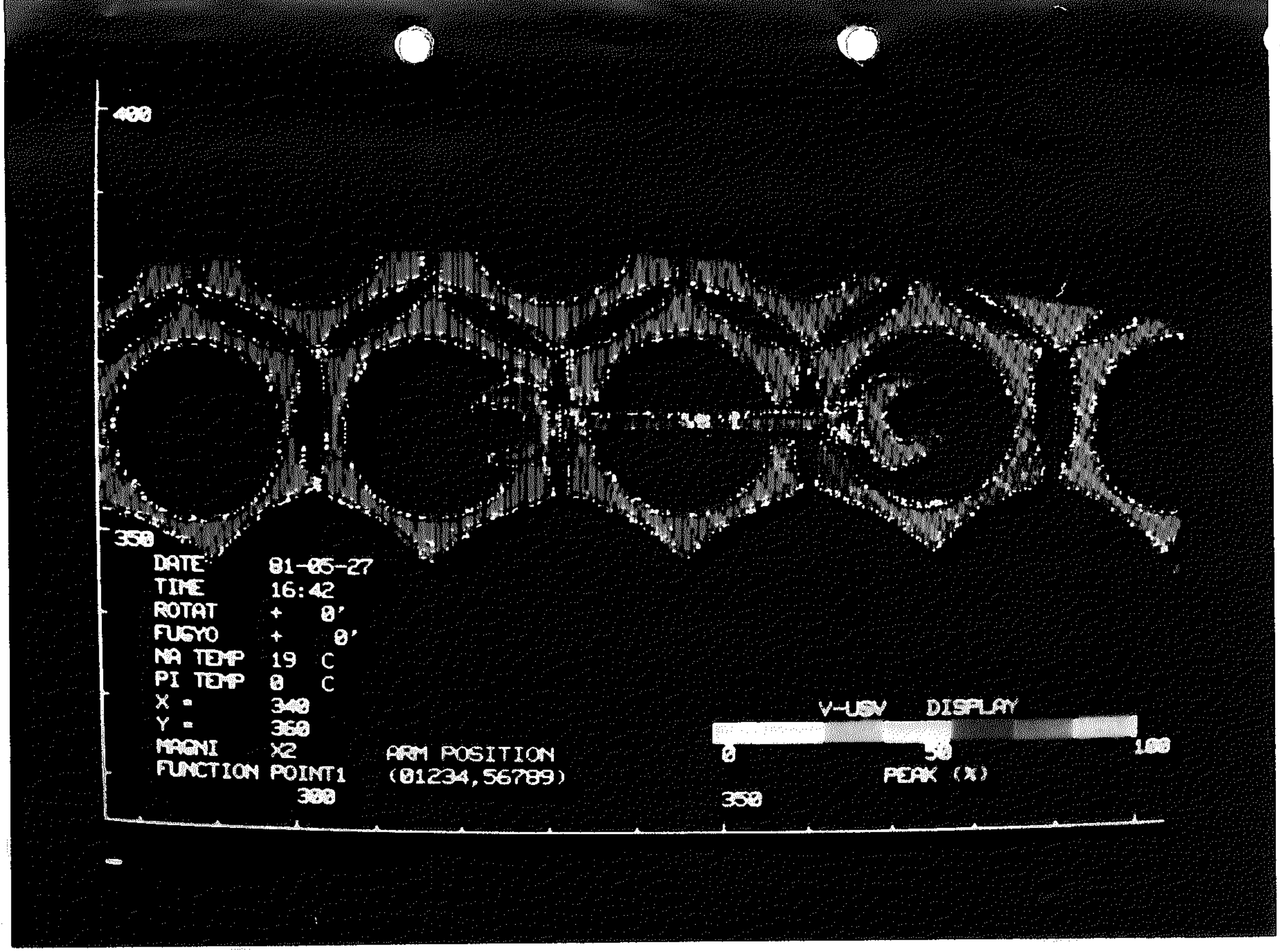




BELLOWS FATIGUE TEST APPARATUS (PNC OWN, INSTALLED IN HITACHI LAB.)



FULL SCALE ACRYLIC MODEL OF PUMP OVER FLOW TANK SUBJECTED TO BUBBLE SEPARATION TEST IN HYDRODYNAMICS TEST FACILITY-2
OARAI ENGINEERING CENTER, PNC



TOP OF CORE ON CRT DISPLAY BY UNDER SODIUM VIEWER,
SODIUM TESTED BY JOINT WORK OF TOSHIBA AND PNC

6. Brochure on PNC Activities