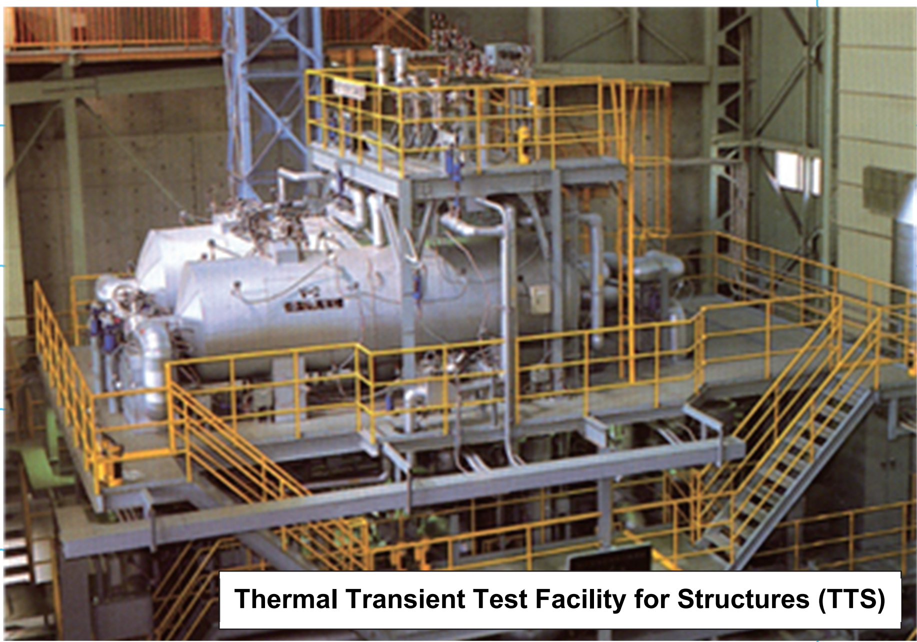
9. Materials and Structures



Thermal Transient Test Facility for Structures (TTS)

- The "Elevated Temperature Structural Design Guide", based on the structural design evaluation methods by elastic analysis, was formulated. This allows more streamlined design compared to the U.S. standards. The guide considers the characteristics of sodium-cooled FRs, the increases in component size and temperatures from Joyo to Monju, and economic improvement.
- The "Material Strength Standards", which also cover the sodium and neutron environmental effects, were formulated through systematic structural material tests for domestic materials. A structural material test database was created.
- A general-purpose non-linear structural analysis code was developed and applied to the design of Monju.
- Structural strength tests were conducted to comprehend the structure failure modes, and to develop and verify the "Elevated Temperature Structural Design Guide". A structural test database was created.
- Seismic tests specific to FR components and structures were performed to establish structural design evaluation methods and confirm seismic resistance and functions.

9.1 Development of Elevated Temperature Structural Design Guide

Compared to an LWR, a sodium-cooled FR has characteristics such as higher operating temperatures, lower operating pressure, and higher transient thermal stress. Structural design of the FR must take these characteristics into consideration. Monju is a prototype FBR with a power generation system, and its technological development was expected to contribute to the improvement of economic efficiency while accommodating an increase in the component size compared to Joyo and a high-temperature environment as well as ensuring safety.

The Monju RV outlet/inlet temperatures are 529°C/397°C. Unlike LWRs, the structural design of Monju must take into account the creep characteristics (the change in strain with time when a constant stress is applied) of the structural materials. That is, as shown in Fig. 9-1, the Monju components are designed to prevent creep rupture, excessive creep deformation, creep fatigue failure, and creep buckling, in addition to ductile rupture, excessive deformation, fatigue failure, and elasto-plastic buckling, which are important in the LWR design.

Since the operating pressure of LWRs (PWRs) is as high as 15.4 MPa, it is internal pressure that causes the main stress in most cases. In Monju, on the other hand, the operating pressure is much lower, less than 1 MPa, the reactor outlet/inlet temperature difference is as large as 130°C, and the heat transfer characteristics of sodium is excellent. The main stress is often represented by steady-state and transient thermal stresses. The modes of thermal stress vary depending on the structural parts, thermal transient conditions, etc. (typical examples are shown in Fig. 9-2).

Monju is a prototype FBR operated at higher operating temperatures with longer design life than Joyo. Therefore, it was essential to develop a technological basis that could be applied to component and structure design and integrity evaluation. Furthermore, a reasonable elevated-temperature structural design method had to be developed to extend the plant design life for future commercialization. To meet these demands, the Elevated Temperature Structural Design Guide for Class 1 Components of the Prototype FBR[9-1] was developed based on the design standards of the American Society of Mechanical Engineers (ASME). In addition, the structural materials of Monju were subjected to systematic material tests, including environmental effects, to develop the Material Strength Standards[9-2].

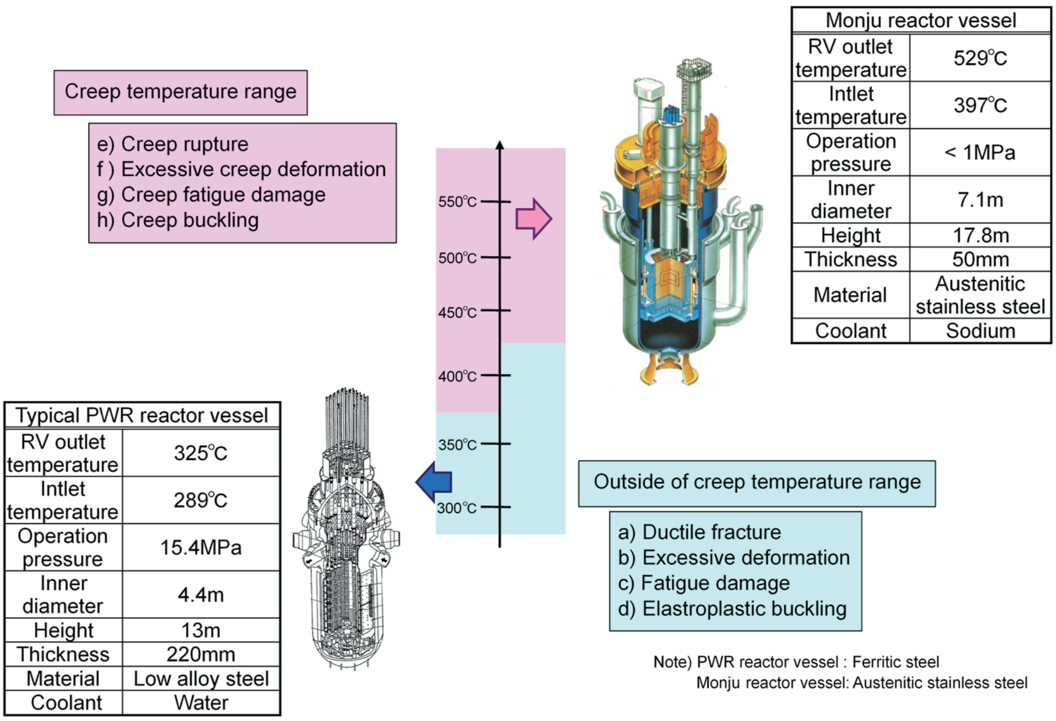


Fig.9-1 Structural design features of FR components

The Monju design was streamlined several times before the final design was submitted. Simplification of the PHTS hot-leg piping layout is a typical example of such efforts. In 1977, the PHTS hot-leg piping was designed to have two downcomers, as shown in Fig. 9-3 (A). This design was employed to keep the thermal expansion stress in the piping at a low level in order to meet the elevated temperature structural design standards of the ASME design standards with elastic analysis. An increase in the pipe length not only affects the weight but also leads to a significant increase in the plant volume, including the thermal insulation and preheating equipment, dead weight and seismic support, equipment for measures against coolant leak, the ventilation and air conditioning system, and buildings. To realize more efficient design for Monju, the Elevated Temperature Structural Design Guide, which enabled a more streamlined evaluation of the load characteristics of FRs mainly consisting of thermal stress, was independently developed. While the Guide followed the basic concept of ASME Code Case N-47, the structural design standards for FRs used worldwide, the evaluation method of the creep damage was further advanced by incorporating such effects as stress relaxation with time.

As a result, the two downcomers were eliminated from the PHTS hot-leg piping to achieve a horizontal pipe routing at high elevation as shown in Fig. 9-3 (B). With the new piping layout, the plant volume can be reduced. At the same time, the reactor coolant level can be effectively maintained in the event of a postulated pipe rupture. A comparison between the Elevated Temperature Structural Design Guide for Monju and ASME Code Case N-47 is shown in Table 9-1. With advice and input from experts from academia, research institutions, and scientific societies, the Guide and the Material Strength Standards were eventually developed as national guidelines ("Structural Design Guide for Class 1 Components of Prototype Fast Breeder Reactor for Elevated Temperature Service" and "Structural Design Guide for Prototype Fast Breeder Reactor for Elevated Temperature Service – Material Strength Standards –"). The guidelines became the basis for the Monju design as well as for the design and evaluation standards for the FR demonstration reactor designed by electric utilities. Today, the standards are published as the "Standards for Nuclear Power Generation Equipment: Design and Construction Standards Edition II – Rules for Fast Reactors –" by the Japan Society of Mechanical Engineers.

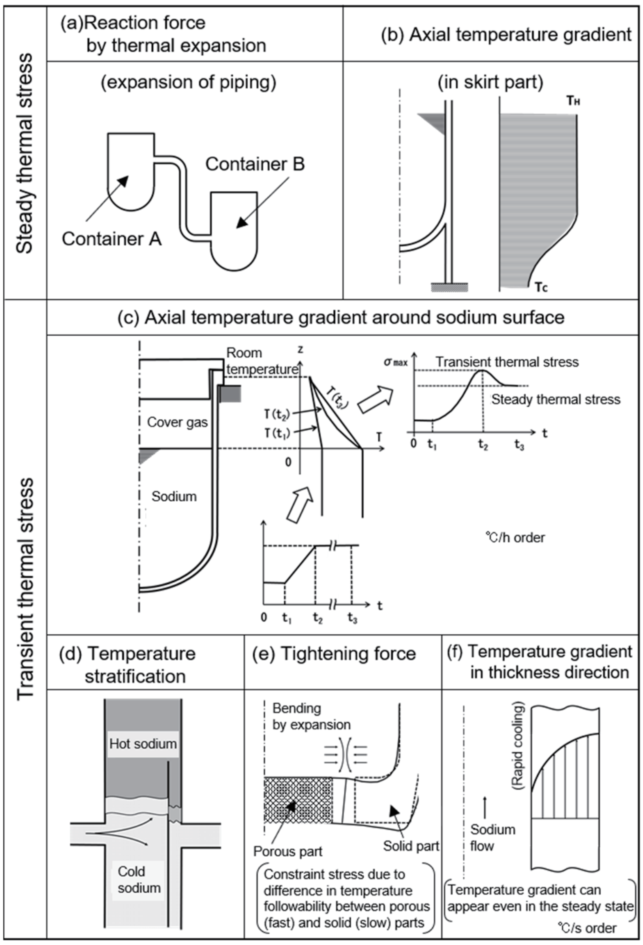


Fig.9-2 Major thermal stress in Monju[9-3)]

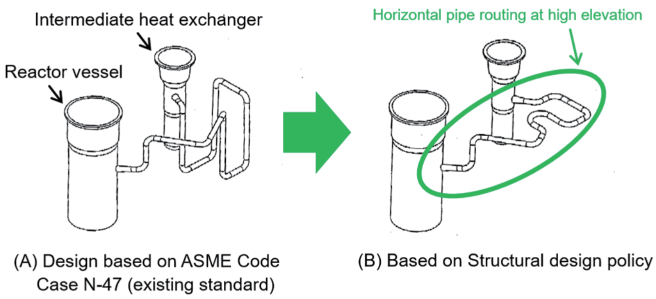


Fig.9-3 Simplification of primary hot leg pipe routing by Elevated Temperature Structural Design Policy

Furthermore, some of the concepts developed in Japan were incorporated in the RCC-MR regulations, the French standards for FRs.

The Elevated Temperature Structural Design Guide was also reflected in the design of the High Temperature Engineering Test Reactor of Japan. In particular, the Material Strength Standards were applied as they were.

Table 9-1 Elevated Temperature Structural Design Guide for Monju and ASME Code Case N-47

	Elevated Temperature Structural Design Guide for Monju	ASME Code Case N-47	Remarks
Scope	Prototype FBR class 1 components	Class 1 components used at temperatures beyond those to which ASME Sec. III NB is applicable	
Materials covered	SUS304, SUS316, SUS321, 2¼Cr-1Mo	Type 304SS, Type316 SS, Alloy 800H, 2¼Cr-1Mo, (Alloy 718*)	* Applicable only to a bolt member
Upper temperature limit	SUS304, SUS316, SUS321 : 650°C 2¼Cr-1Mo : 550°C	304SS, 316SS : 1500°F (816°C) Alloy 800H : 1400°F (760°C) 2¼Cr-1Mo : 1200°F (649°C)	Temperature limits specified by N-47 depend on the allowable values.
Environmental effect	Evaluation methods of the sodium environment and neutron irradiation effects are provided.	Not provided.	
Evaluation method	• Long-term and short-term loads are classified. • A method of judging elastic follow-up is provided. • The strain and creep fatigue damage limits are provided for the following cases: 1) General provisions, 2) Case of low long-term primary stress, and 3) Case of insignificant creep effect. • Design fatigue curve including the strain rate effect (the holding time effect is evaluated as the creep damage.)	• Long-term and short-term loads are partially classified (creep damage evaluation). • No evaluation method for elastic follow-up is provided. • The strain and creep fatigue damage limits are provided for the following cases: 1) General provisions, and 2) Case of insignificant creep effect. (Test No.4) • Design fatigue curve for elastic analysis including the holding time effect (the holding time effect is evaluated as the fatigue damage.)	

9.2 Development of Material Strength Standards

(1) Creep effects

Creep effects are essential in designing FR components. Creep is basically a rate effect, and progresses with time, lowering material strength. Since this time-dependent effect becomes more pronounced at higher temperature, the creep effects need to be evaluated as time- and temperature-dependent effects. This makes the structural design of FRs more complex and difficult compared to that for LWRs, in which such consideration is not necessary.

A wide range of high-temperature material property data is required for FRs, as shown in Table 9-2, to develop the Material Strength Standards of the elevated temperature structural design guide for Monju and to calculate the creep damage coefficients.

Initially, the plan called for formulating the design allowable stress applied to the elevated temperature structural design of Monju by validating the design allowable stress of the ASME Code Case N-47 using the test data of domestic materials and applying the Code Case as it was. However, the reference values for the SUS321, an SG superheater material, and the 2¼Cr-1Mo steel normalized-tempered material, an SG superheater material, were not available in Case N-47. Furthermore, activities for the formulation of design tensile strength (Su), etc. using domestic material data were beginning in the field of LWRs. Thus, with the understanding that Monju was a national project based on domestic technologies, the policy was changed to have the design allowable stress, etc. for the FR formulated in Japan based on the statistical analysis of expanded and organized domestic material data.

The required material data significantly lacking were the data relating to inelastic strain behavior, such as the creep curve and stress relaxation curve; and the data relating to creep fatigue strength evaluation, such as the strain hold effect and strain rate effect. Obtaining such data was one of the major tasks. Intensive R&D on the high-temperature strength of structural materials for Monju was then conducted based on atmospheric structural material tests under an all-Japan framework of experts from the relevant industry, academia, and government organizations in a short period of time.

The activities contributed to the accumulation of new material data, especially creep behavior data in which reliable data were clearly insufficient at that time. Furthermore, following the ASME method, the fatigue life was formulated as a function of the strain range, temperature, and strain rate using the domestic material data. With this, a unique design fatigue curve was developed in Japan. With the obtained formula, the fatigue life (number of cycles to failure) can be predicted as a function of temperature, strain rate, and strain range. The relationship between the predicted and measured values of fatigue life in a temperature range of 450 to 600°C at a strain rate of 0.001/s fits well within a factor of 2 with a 95% confidence interval, as shown in Fig. 9-4.

(2) Sodium environmental effect

In-sodium structural material tests were performed to confirm the corrosion thinning and mass transfer phenomena (elution of nickel, chromium, manganese, etc.) of SUS304, structure material of the primary system, as well as the decarburization in a high-temperature section of SG made of ferritic 2¼Cr-1Mo steel and the carburization of an IHX made of SUS304 in the secondary system.

In a systematic test of domestic SUS304 in flowing sodium, the corrosion and mass transfer phenomena were confirmed to be limited to a surface effect as small as the dimensional tolerance for a component with a thickness exceeding a few mm under the sodium flow conditions of Monju. In addition, change in the strength was investigated through post-immersion tensile tests, in-sodium creep tests, and fatigue and creep fatigue tests. The fatigue test results are shown in Fig. 9-5. The tests confirmed excellent compatibility with sodium of austenitic stainless steel, such as SUS304. The tensile strength and creep strength tend to increase even when carburization occurs and the creep fatigue strength is not influenced by carburization although the long-term creep ductility is lowered.

In a sodium test loop that simulated the secondary system, specimens made of 2¼Cr-1Mo steel were immersed to quantitatively evaluate the depth-direction distribution of decarburization and carburization. The influence of decarburization on the tensile strength was then

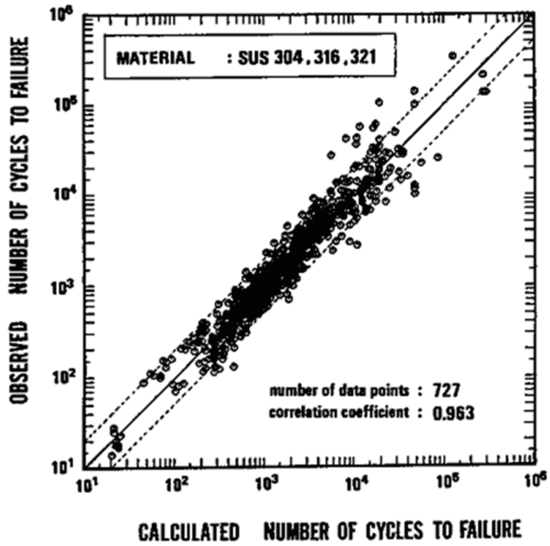


Fig.9-4 Predictability of low cycle fatigue lives for stainless steels[9-4)]

Table 9-2 Relationship between Material Strength Standards and material properties required for development

Type of material strength test	Item of high-temperature material properties	Material strength standard: Maximum allowable stress intensity (So)	Design stress intensity (Sm)	Design stress intensity (St)	Design yield point (Sy)	Design creep rupture stress intensity (SR)	Design tensile strength (Su)	Design relaxation strength (Sr)	Allowable strain range (A), Strain rate 10^{-3} (s^{-1}) (εt)	Allowable strain range (B), Strain rate 10^{-6} (s^{-1}) (εt)	Allowable strain range (C), Strain rate 10^{-8} (s^{-1}) (εt)	Isochronous stress-strain curve	Attachment: Limit of cumulative creep fatigue damage coefficient (D)	Relaxation creep damage coefficient for primary and secondary stress (D*)	Relaxation creep damage coefficient for peak stress (D**)
High-temperature tensile test	0.2% yield strength	O	O		O										
	Tensile strength	O	O				O								
	Tensile stress-strain curve			O		O						O			
Creep test	Creep rupture strength	O		O										O	O
	Steady-state creep rate	O		O											
	Creep strain curve			O				O				O		O	
Fatigue test	Creep fatigue strength								O	O	O				
	Dynamic stress-strain curve														O
Relaxation test	Monotonous stress relaxation curve							△						△	
Creep fatigue test	Creep fatigue strength												O		△
	Cyclic stress relaxation curve														△

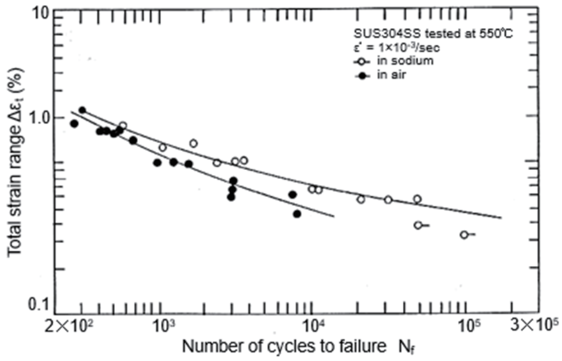


Fig.9-5 Effect of sodium environment on fatigue strength of SUS304[9-5)]

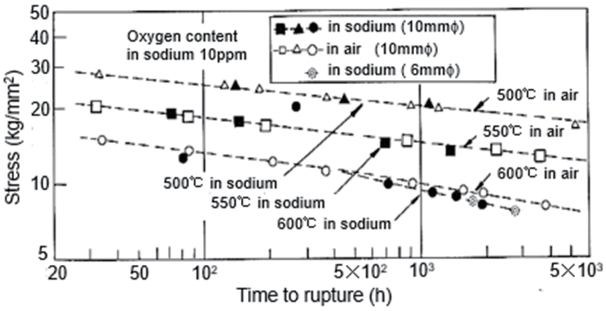


Fig.9-6 Creep strength deterioration due to decarburization of 2¼Cr-1Mo steel

examined. As a result, while the strength was lowered according to the thermal aging effect up to by 550°C, deterioration of the creep strength due to decarburization was observed in a higher-temperature, long-term range at 600°C, as shown in Fig. 9-6. Accordingly, the strength correction coefficient was determined to reflect the decarburization effect in the design.

(3) Neutron irradiation effect

The PIE data from Joyo and Japan Material Testing Reactor as well as from foreign reactors were used to determine the limit values or the strength reduction factors. The upper limit of fast neutron fluence (E > 0.1 MeV) was set below the 95% lower confidence limit (95% LCL) based on the relationship between the fast neutron fluence and breaking elongation shown in Fig. 9-7. The reduction of creep rupture time caused by thermal neutron irradiation (E < 0.4 eV) was set below the 95%LCL based on the relationship between the neutron fluence and the reduction rate of the creep rupture time, as shown in Fig. 9-8.

(4) Database

Various material test data related to the development of the Material Strength Standards were stored and integrated in the database system SMAT.

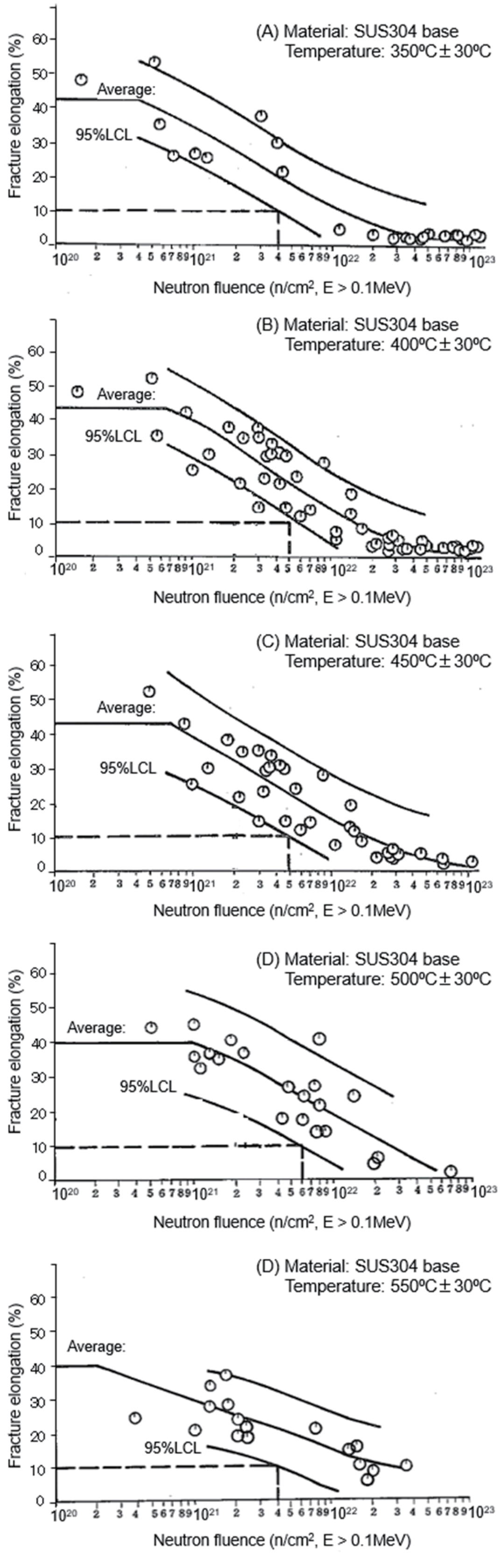


Fig.9-7 Fast neutron fluence dependence of facture elongation

9.3 Development of structural analysis method

The characteristics of FR structural design include high operating temperatures at which creep deformation may occur and plastic deformation generated in some components by high thermal stress, especially high transient thermal stress. Thus, a new analysis method capable of handling both creep and plastic deformation, in other words, an inelastic analysis method, had to be developed.

In around 1975, no general-purpose structural analysis code was available in Japan, and there was no choice but to use commercial codes developed in the U.S. However, such codes are difficult to expand and validate in Japan, and the PNC decided to develop a general-purpose structural analysis code focusing on the inelastic analysis capability.

A cycle of development, validation, improvement, expansion, and application was repeated while gradually adding new capabilities throughout the process to complete the general-purpose nonlinear structural analysis code FINAS, Finite Element Nonlinear Structural Analysis System for practical use[9-6]. The FINAS development phases are shown in Fig. 9-9. The types of analyses handled by FINAS are shown in Table 9-3.

In addition to the structural design of Monju, FINAS was also used to develop Elevated Temperature Structural Design Guide and to analyze and assess structural tests. Furthermore, FINAS was validated against international benchmark problems.

Examples of validation study on the analysis methods unique to FINAS for the typical problems of Monju components are explained below.

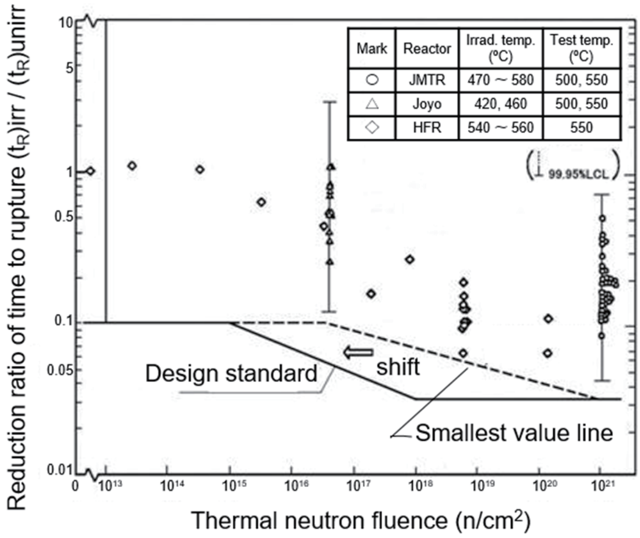


Fig.9-8 Thermal neutron fluence dependence of reduction ratio of time to rupture

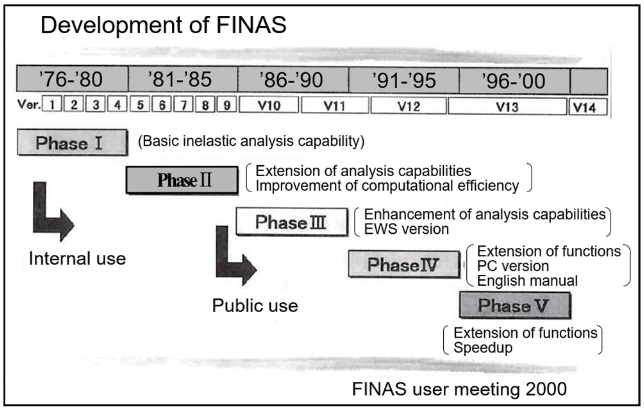


Fig.9-9 Development of FINAS

Table 9-3 Analysis capabilities of FINAS (types of analyses)

Static analysis	・Elastic analysis, thermoelastic analysis, non-axisymmetric load analysis, multi-load parallel processing analysis ・Elasto-plastic analysis, thermal elasto-plastic analysis ・Elasto-creep analysis, elasto-plastic-creep analysis, thermal elasto-plastic-creep analysis ・Swelling analysis ・Large deformation analysis, large deformation inelastic analysis ・Linear buckling load analysis, non-linear buckling load analysis, non-axisymmetric buckling analysis ・Fracture mechanics analysis (hypothetical crack growth method) ・Point-contact problem analysis, plane-contact problem analysis
Dynamic analysis	・Modal response analysis, spectral response analysis, frequency response analysis ・Linear direct integration analysis, non-linear direct integration analysis ・Fluid-structure coupling analysis ・Contact-collision analysis
Temperature analysis	・Stationary heat conduction analysis ・Non-stationary heat conduction analysis ・Radiation analysis

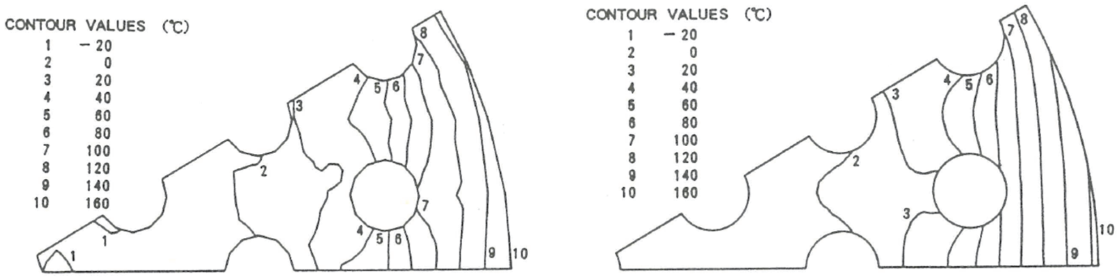
(1) Simplified two-dimensional temperature analysis method for tube plate structures

The tube plate structure used in FR heat exchangers is difficult to design due to the large temperature difference that occurs between a porous region and the peripheral region during thermal transients such as upon a reactor trip. A simplified method for two-dimensional temperature analysis in FINAS was validated using temperature measurement data obtained in a thermal transient test using a half-scale model of Monju SG tube plate conducted at the OEC Air-Cooling Thermal Transient Test Facility. As shown in Fig. 9-10, the analytical values of the tube plate surface temperature in the SG tube plate structural model generally agreed with the experimental values[9-7].

(2) Sloshing analysis in an axisymmetric vessel

The RV of FRs has a relatively thin-walled large-diameter shape and contains a large amount of liquid sodium with a free surface. The vessel has a complex structure to arrange components important to safety such as core structures. Evaluating the seismic safety of such a reactor structure requires a dynamic response analysis method capable of accurately predicting the sloshing (liquid surface fluctuation) response of the free surface, the pressure loads on the vessel wall, and the fluid-structure dynamic interaction. The FINAS analysis method was validated against data obtained from a sloshing test using a cylindrical vessel model that simulated the Monju RV. Figure 9-11 shows a comparison between the analyzed and measured response time history of the liquid surface wave height for a case where three cycles of resonant sine wave were input to a cylindrical vessel model not including the internal components of RV. Comparisons between the analyzed and measured wave height mode and pressure at the time of the maximum wave height are shown in Fig. 9-12. These results indicate that the phenomena can be generally reproduced by FINAS[9-8].

The open access of FINAS encouraged general-purpose use of the code by external parties. The code has been widely used by manufacturers and universities in the fields of nuclear energy, machinery, electric machinery, and automobiles.



Measured temperature distribution

Calculated temperature distribution

Fig.9-10 Surface temperature of tube plate in thermal transient test simulating Monju SG

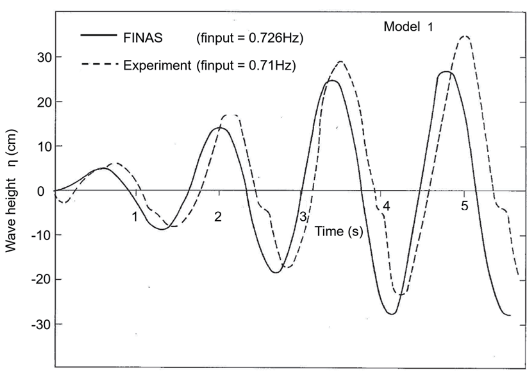


Fig.9-11 Transient response for sine wave excitation

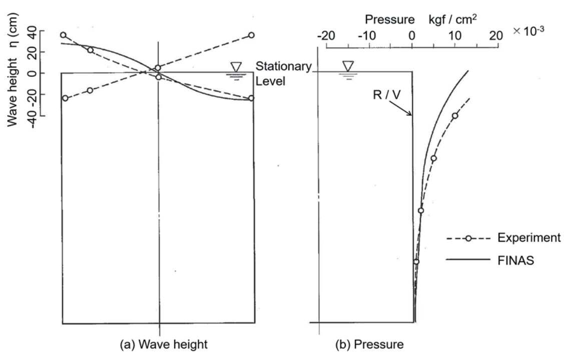


Fig.9-12 Profiles of peak wave and pressure observed in sloshing test

9.4 Structural tests

The structural tests have the following objectives:

- Validation of the structural analysis method related to the structural deformation behavior and reflecting the method in the Elevated Temperature Structural Design Guide,
- Validation of the provisions related to the structural strength in the Elevated Temperature Structural Design Guide, and
- An understanding of structural failure behavior.

Around 1970, virtually no structural tests directly concerning the design standards were conducted in Japan or abroad. Therefore, the early structural tests began with low-cycle fatigue tests and thermal ratchet tests in the area of ASME Section III using simple piping elements. Later, tests using a piping element model, tests of a piping system under a mechanical load that substituted a thermal load in the creep range, and thermal creep ratchet tests were performed with the aim of reflecting the test results in the Elevated Temperature Structural Design Guide. In the process of clarifying the various types of thermal stress to be considered in the design and development of the Elevated Temperature Structural Design Guide, creep fatigue tests were performed for a wide variety of structures and under various thermal stress conditions. When designing the specimens for these tests, the strength evaluation method established in the development of Elevated Temperature Structural Design Guide and the accumulated basic testing experience were proven to be effective. These structural test data and the related analytical evaluations were used to formulate and validate the Elevated Temperature Structural Design Guide. The various strength data were compiled into the STAR structural test database to facilitate search and use.

Three types of structural tests are explained below:

(1) Elastic followup test of PHTS hot-leg piping

In a high-stress section of piping kept at a high temperature, the strain increment associated with the reduction in elastic deformation in the low-stress section is added during a stress relaxation process caused by creep, resulting in increased strain. This phenomenon is called “elastic followup”. The Elevated Temperature Structural Design Guide for Monju stipulates a specific method to calculate the elastic followup strain. This method was developed using simplified inelastic analysis of piping validated by high-temperature tests of many piping elements. Thus, it was desired to confirm the behavior of elastic followup strain in an actual structure. Furthermore, the PHTS hot-leg piping of Monju was designed by the new concept of the horizontal pipe routing at high elevation, and the actual behavior of elastic followup strain in the entire piping system had to be confirmed. To address these issues, an elastic followup test at a high temperature (600°C) was conducted using quarter-scale model of the PHTS hot-leg piping (piping from the RV outlet to the IHX inlet), as shown in Photo 9-1, and the

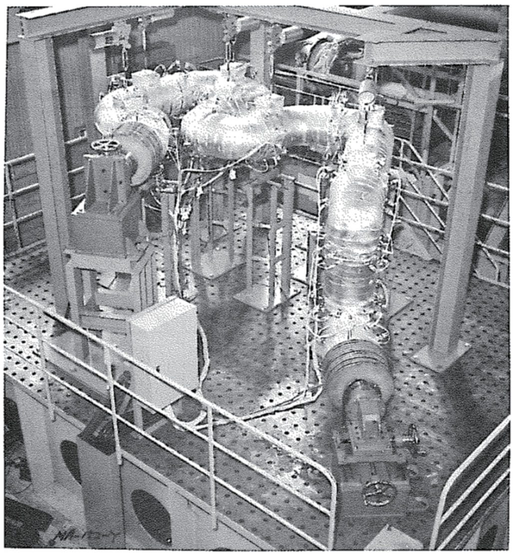


Photo 9-1 Elastic followup test for primary hot leg piping system

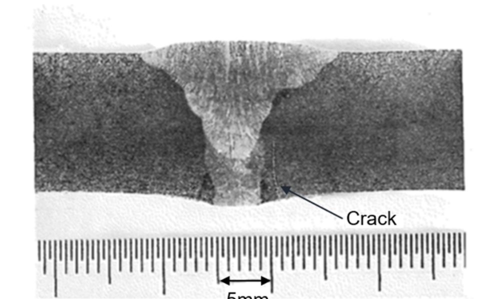
following results were obtained:

- The elastic followup strain caused by the thermal expansion stress is small in the PHTS hot-leg piping, and the thermal expansion stress could be classified as secondary stress.
- The “elastic followup determination method related to the thermal expansion stress” in the Elevated Temperature Structural Design Guide was confirmed to be conservative.
- The developed simplified inelastic (elastic-creep analysis) code was validated.
- The entire piping system was confirmed to exhibit the behavior expected in the design.

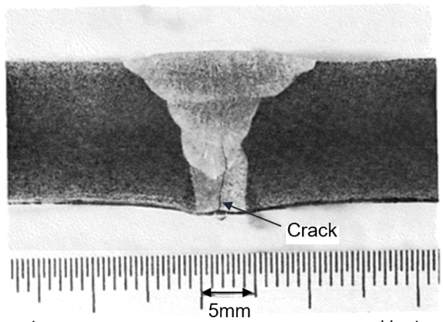
(2) Thermal fatigue test of SUS304 steel pipe weld joint

To evaluate the fatigue or creep fatigue strength of a weld joint, it is essential to consider the discontinuity of a geometrical structure caused by welding deformation and weld crown (buildup due to welding heat) and the material discontinuity caused by differences in materials between the base material, the heat-affected zone, and the weld metal. Moreover, the influence of residual stress should also be considered. Therefore, a thermal fatigue test of a SUS304 steel pipe weld joint was conducted by alternately pouring high- and low-temperature sodium onto a testing device to apply thermal cycles.

Cracks generated in the inner surface of the specimen are shown in Photo 9-2. The photograph shows that the crack in (a) was generated from a gentle crevice formed by a counterboring process and a weld crown on the inner surface of the piping in a heat-affected zone, and that the crack in (b) originated from a crevice generated by heat shrinkage of the weld metal. It was concluded that a safety factor (stress concentration factor) taking into account such shape irregularity should be incorporated in the evaluation of thermal fatigue failure of a weld joint in piping.



(a) W1 weldment - θ = 90°



(b) W1 weldment - θ = 90°

Photo 9-2 Crack in SUS304 welded joint after thermal fatigue test

(3) Integrated structural test

The Elevated Temperature Structural Design Guide should prevent creep fatigue damage when actual components undergo cyclic thermal transients. In order to confirm the overall validity of the damage prevention criteria, a failure test applying a thermal transient load with the magnitude and number of cycles exceeding those assumed in an actual plant, in other words, a test to monitor the generation of a significant crack, was conducted at the Thermal Transient Test Facility for Structures (Photo 9-3).

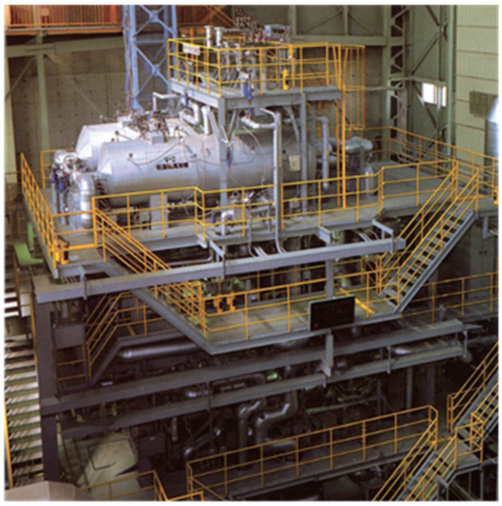


Photo 9-3 Thermal Transient Test Facility for Structures

This facility constituted two (high- and low-temperature) sodium loops, through which high- or low-temperature sodium is alternately supplied to a test model to conduct a creep fatigue test until failure and strength data can be obtained. The test model incorporates typical structural parts to be focused on in terms of the structural strength. The same manufacturing process as those used in the actual plant was applied to the model.

A result of the test is shown in Fig. 9-13[9-9]. The vertical lines in the figure show the measured crack depth. The part without line indicates that no crack occurred. This test confirmed that the creep-fatigue strength method based on the elastic analysis of the Elevated Temperature Structural Design Guide has a large safety margin (a factor of 50 in the gas tungsten arc welded area of the RV outlet nozzle, symbol J1 in the figure, and larger than 50 in other areas). This implies that further reduction in the safety margin is possible through the application of inelastic analysis. Furthermore, it became clear that the thermal shield plates can significantly reduce transient thermal stress. This is because sodium present between multilayered shield plates mitigates the change in temperature of metal surface in contact with sodium.

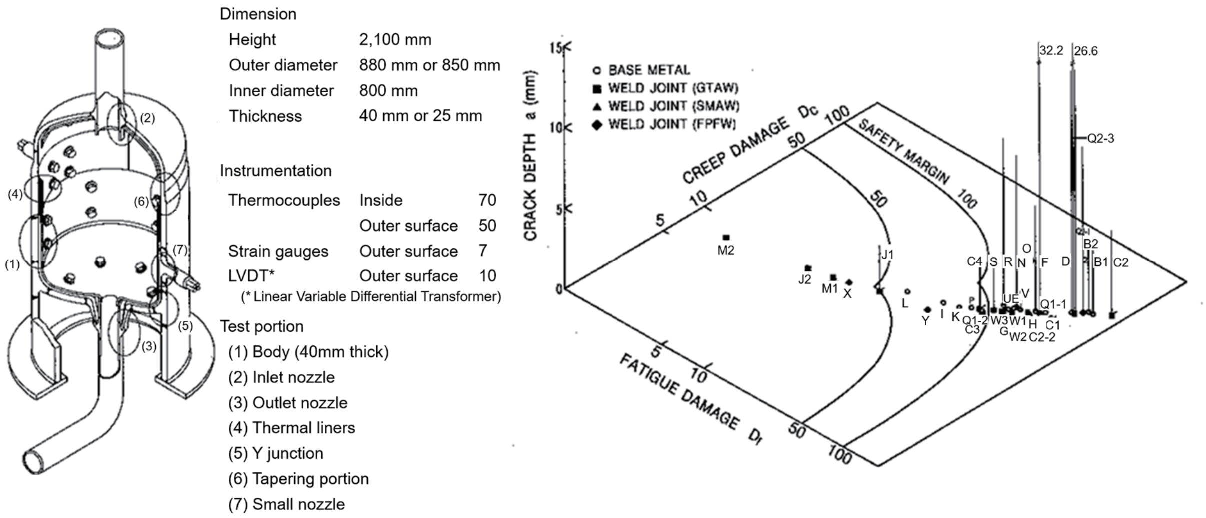


Fig.9-13 Creep-fatigue damage evaluation based on elastic stress analysis
(vertical axis shows crack depth)

9.5 Seismic test

In addition to addressing the problems of elevated temperature structural design, ensuring the seismic resistance of components and structures was an important task in the design and construction of the components. The seismic structural tests were conducted to understand the vibration characteristics and confirm structural strength using test specimens of the major components and structures. The function confirmation tests were conducted to confirm that the functions of the active components could be maintained in the event of earthquake. The test results and the findings obtained are reflected in the assessment of the seismic resistance of the components and structures, the validation of the design, and the validation of the seismic analysis method.

The representative tests conducted for Monju include a group vibration test of the core elements, vibration and sloshing tests of the RV, vibration and buckling tests of the CV, a vibration test of the PHTS piping, and a control rod insertability test.

The vibration test of core element groups is described below as an example. Since the core elements (fuel subassembly, etc.) of Monju are densely loaded on the core support plate with in-between gaps of 0.7-1.0 mm at pad positions, a complex, non-linear vibration involving collision occurs in an earthquake. In order to understand the behavior of the core internals and establish a seismic design method, seismic wave shaking tests were conducted using a single dummy core element unit, four units with a single row, 29 units with a single row, and 37 units with multiple rows. Various data were obtained on the vibration characteristics, deformation, and collision load of the elements and the reaction force of the core support structure. Concurrently, a computer code for analyzing the collision and vibration behavior of the core element group was developed and validated. A test device with 37 core elements is shown in Photo 9-4. The technological basis for the seismic design of the Monju core was established through analysis of the series of tests and code analyses.

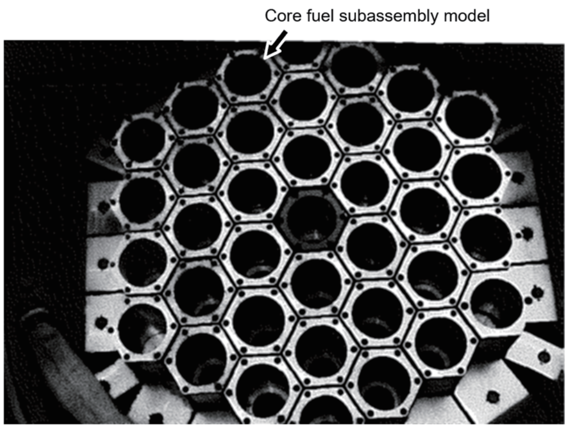


Photo 9-4 Vibration test using 37 elements in water

— References —

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10. Operation and Maintenance

Operation training with simulator

- ▶ Monju plant operation was securely conducted to achieve initial criticality, first connection to the power grid, and 0 to 40% power tests. The operation management system was developed in collaboration with electric utilities.
- ▶ Training and education systems were established, and the operator training simulator was developed. Symptom-based operating procedures were introduced on the basis of operation experience in the commissioning tests.
- ▶ Maintenance and repair experience that can be reflected to the design, operation, and maintenance of future reactors has been accumulated for more than 25 years.
- ▶ Introduction of a maintenance program as an FR plant in the R&D stage identified difficulties in implementing the program. Based on this experience, improvement in maintenance management was discussed.

10.1 Operation

The operation of Monju has the following features compared with that of LWRs.

- Operation is relatively simpler because the reactor is controlled solely by control rod system and sodium systems are controlled mainly by temperature, not by pressure.
- Plant power from 40 to 100% can be controlled by automatic operation.
- Plant startup/shutdown operation takes longer to moderate thermal transient effects on components associated with high-temperature operation.
- Systems for heating sodium and air ventilation are kept in operation even during reactor shutdown to maintain plant integrity.

Photo 10-1 Main control room (first connection to power grid)

Toward the first commissioning of Monju, plant operators were trained and educated through:

- Establishment of operation management system,
- Introduction of an operator training simulator and development of education and training systems, and
- Preparation of operating procedures.

In addition, operators were encouraged to become proficient in plant operation through various opportunities including SKS. As a result, Monju was securely operated to achieve initial criticality, first connection to the power grid, and 0 to 40% power test (Photo 10-1). An outline of the plant startup curve of Monju is shown in Fig. 10-1.

Subsequently, long-term shutdown continued after the Secondary Sodium Leak Accident occurred in 1995. Nevertheless, the operation methods and procedures have been improved, and education and training systems have been updated using the simulator, reflecting experience in the Core Performance Confirmation Tests and lessons learned from the LWR accidents including the 1F Accident.

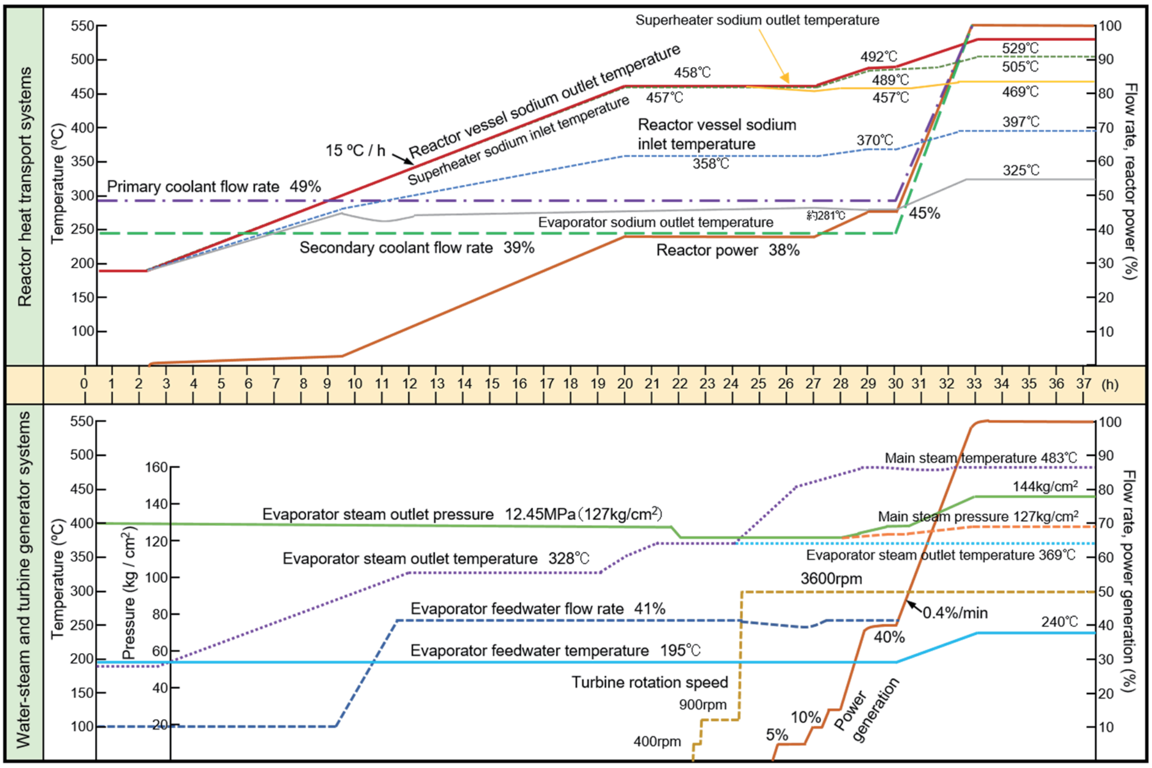


Fig.10-1 Plant characteristic curves in Monju startup (outline)

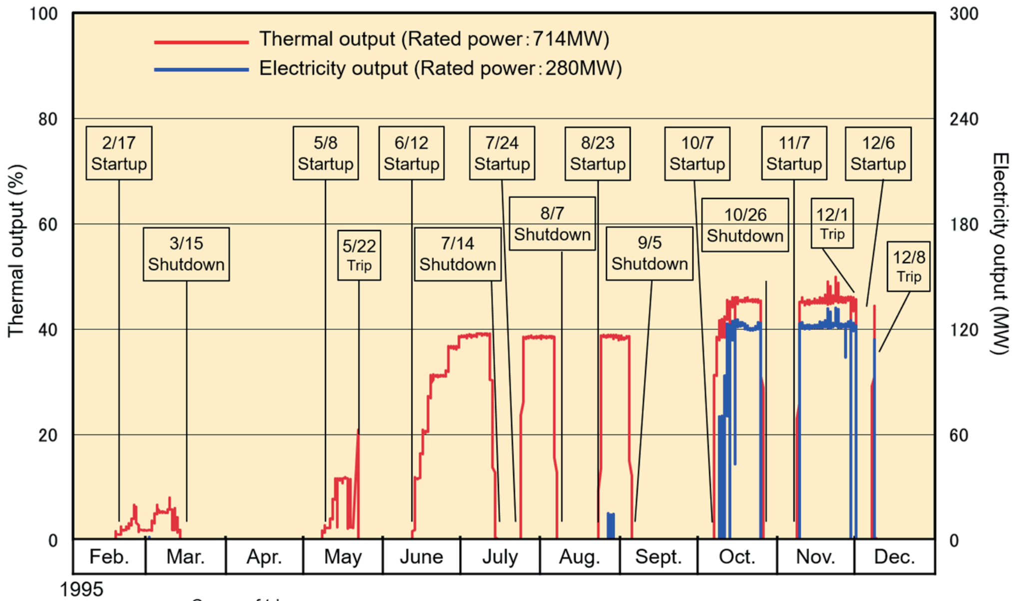
10.1.1 Plant operation records[10-1)]

Major operation achievements are listed in Table 10-1, and operation records during the 40% power test in 1995 are shown in Fig. 10-2. The total reactor operation hours and the power production are as follows:

- Reactor operation hours: 5,300 hours and 45 minutes (from initial criticality to the end of the Core Performance Confirmation Tests)
- Power production: 102,325 MWh (883 hours)

Table 10-1 Operation achievements of Monju

Year	Date	Operation achievement
1993	Oct. 13	Core fuel installation started.
1994	Apr. 5	Initial criticality was achieved.
	May 21	Reactor physics test started.
	Nov. 15	Reactor physics test finished.
1995	Feb. 16	Startup test (40% power test) started.
	May 22	Reactor trip due to feed water pump trip
	July 9	First main turbine ventilation
	Aug. 29	The first connection to the power grid (electric power: 14 MW).
	Dec. 1	Plant trip test at 40% power
	Dec. 8	Manual reactor trip due to sodium leak from the SHTS
2010	May 6	SST resumed. Core Performance Confirmation Tests started.
	May 8	Criticality was achieved.
	July 22	Core Performance Confirmation Tests finished.



Cause of trip
5/22: Due to pump trip in the secondary heat transport system
12/1: Reactor trip test from 40% rated power
12/8: Due to sodium leak in the secondary heat transport system

Fig.10-2 Plant operation records in the 40% power operation

10.1.2 Operation management system[10-2)]

(1) Arrangement of operators

The Monju Operation Preparation Office was established in October 1989 to advance preparations for commissioning. Monju Plant Section 1, which was responsible for operations, was organized from the Operation Preparation Office in May 1991, and employed about 70 personnel in 1993 while performing SKS. It consisted of about 30 PNC employees, including those engaged in the design of Monju and those who worked in Joyo and Fugen, and about 40 temporary staff on loan from the electric utilities.

For the start of core fuel loading, a regular operators' shift system with 5-team 8 hour shifts was established in summer 1993. Each operating shift consisted of 11 to 12 operators, including the shift supervisor, assistant shift supervisor, senior operator, middle-class operators, novice operators, and apprentice operators. Among these, 4 operators were responsible for operation of the fuel handling system. The Operational Safety Program for Reactor Facility was put into effect in October 1993, and the operators were engaged in core fuel installation, reactor physics tests, and power increase tests.

During cold shutdown periods after the Secondary Sodium Leak Accident, the number of operators for each shift was reduced to 6 and operation of the fuel handling system was transferred to Monju Plant Section 3 (currently the Nuclear Fuel and Waste Management Section) according to the change in task assignment.

In April 2009, the operators' shift system was changed to 5-team 12 hour shifts to increase the time and opportunity for education and training of operators. The daytime duty was specified as the "training shift" to clarify that the period was for the education and training. A comparison of the education and training hours before and after the change in operators' shift systems is shown in Fig. 10-3.

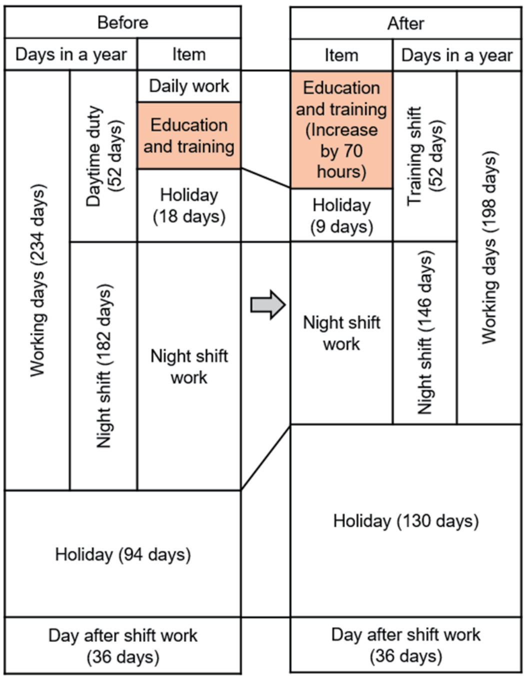
The changes in operators' shift systems from the earliest stage to the Core Performance Confirmation Tests were flexibly arranged as shown in Table 10-2.

(2) Operator competence evaluation

The competence evaluation and qualification examination system for operators was developed for Monju. Manuals were developed pursuant to the Guides on Education and Training for Nuclear Power Plant Operators (JEAG4802-2002), in which operators' qualifications and an examination method for qualification criteria were provided.

In the qualification examination, the Monju Plant Section 1 manager grants operator qualification to personnel who have passed an oral examination after it has been confirmed by the shift supervisor that education and training for each operator level are completed and the required number of years of operating experience are satisfied.

The shift supervisor is qualified based on the "Manual for Qualification for the Operation Supervisor". The plant director grants the qualification to personnel who have passed practical and oral examinations after it has been confirmed by the Monju Plant Section 1 manager



Note: Working hour is unchanged while working day is reduced by introduction of 5-team 12 hours shifts

Fig.10-3 Change in operators' shift systems

Table 10-2 Change of operators' shift systems

Time period	Operation management system and main tasks	Special remarks
From Oct. 1989	Monju Operation Preparation Office was organized to establish an operation management system.	-
From May 1991	Monju Plant Section 1 was organized. Responsible for safety staff shifts with a few operators, and SKS	-
From summer 1993	Regular operators' shift system with 5-team 8 hour shifts was established. 11-12 operators were arranged per shift. Responsible also for operation of the fuel handling system as well as operation for core fuel loading, the reactor physics, and the 40% power tests.	The Operational Safety Program for Reactor Facility was put into effect in October 1993.
From Jan. 1996	Following reactor shutdown, the number of operators per shift was reduced to 6. Operators except shift personnel were in charge of inspection and testing, plant management, and examination of operating procedures.	-
From Dec. 2000	The number of operators (including one shift supervisor) per shift was specified in the Operational Safety Program for Reactor Facility as 7 and 5 or more during reactor operation and shutdown, respectively.	Operation of the fuel handling system was transferred to Monju Plant Section 3 (Nuclear Fuel and Waste Management Section).
From July 2007	Operation system with 5-team 12 hour shifts was introduced on a trial basis. The daytime duty was specified as "training shift". The system was officially applied in April 2009.	Time for education and training was increased by about 70 hours per shift.
From May 2010	8-9 operators were assigned to operation in the Core Performance Confirmation Tests because the water-steam system was not operated.	-