3. Commissioning

- **The commissioning of Monju was carried out to perform final adjustments and confirmation of all the plant systems for full-scale operation, as well as validating development achievements and identifying issues for future reactors.**
- **During the commissioning period, initial criticality and the first connection to the power grid were achieved. It was confirmed that the developed systems and components performed as intended up to the power level of 40%. In addition, the core performance data such as breeding ratio were obtained and successfully used for the development of core analysis technology.**
- **The commissioning tests at the rated power and subsequent full-power operation were not realized due to the Government's policy toward the decommissioning of Monju, in the wake of the 1F Accident.**

3.1 Major commissioning steps

The commissioning (i.e., pre-operational test) was divided into two phases: the Comprehensive System Function Tests (SKS) before loading core fuel and the System Startup Tests (SST) intended to confirm plant performances from the core fuel loading to the start of rated power operation. The major commissioning steps are shown in Fig. 3-1. SST were suspended due to the Secondary Sodium Leak Accident that occurred in December 1995 during the 40% power test, and Monju was kept in a shutdown state for many years since then for plant modification work, etc. To resume SST, the test plans were reviewed and revised, and the Plant System Confirmation Tests (PKS) were performed taking the long-term shutdown into consideration.

3.2 Comprehensive System Function Tests

Examples of SKS are shown in Fig. 3-2. The core with the dummy assemblies was first constructed in May 1991, and then the functional tests on 125 items were carried out for the primary and secondary cooling systems, fuel handling and storage systems, and other systems in in-air tests at room temperature, in-argon gas tests, and in-sodium tests in stages.

In the in-air tests at room temperature, operability and controllability of the fuel handling machine and the IVTM in the RV were confirmed by visual observation. A preheating test performed before sodium charge confirmed that the relevant systems were preheated uniformly as designed.

Sodium was procured in France and shipped to Japan in tank containers. As much as 1,700 tons of sodium was received at the Monju site and charged into the relevant systems through temporarily prepared storage tanks. After sodium charge, flushing operation, sodium circulation, and purification tests in the cooling systems were performed. In an operation test of the CRDM, scram tests were performed in air and in sodium, and it was confirmed that the control rods were successfully and rapidly inserted within the time assumed in the design.

3.3 Original SST

3.3.1 SST planning and implementation teams

Since Monju is a power generation plant constructed while developing major components, systems, and analysis methods from the early conceptual design stage, SST were aimed at performance confirmation similarly to the comercial plants. Further, from the prototype

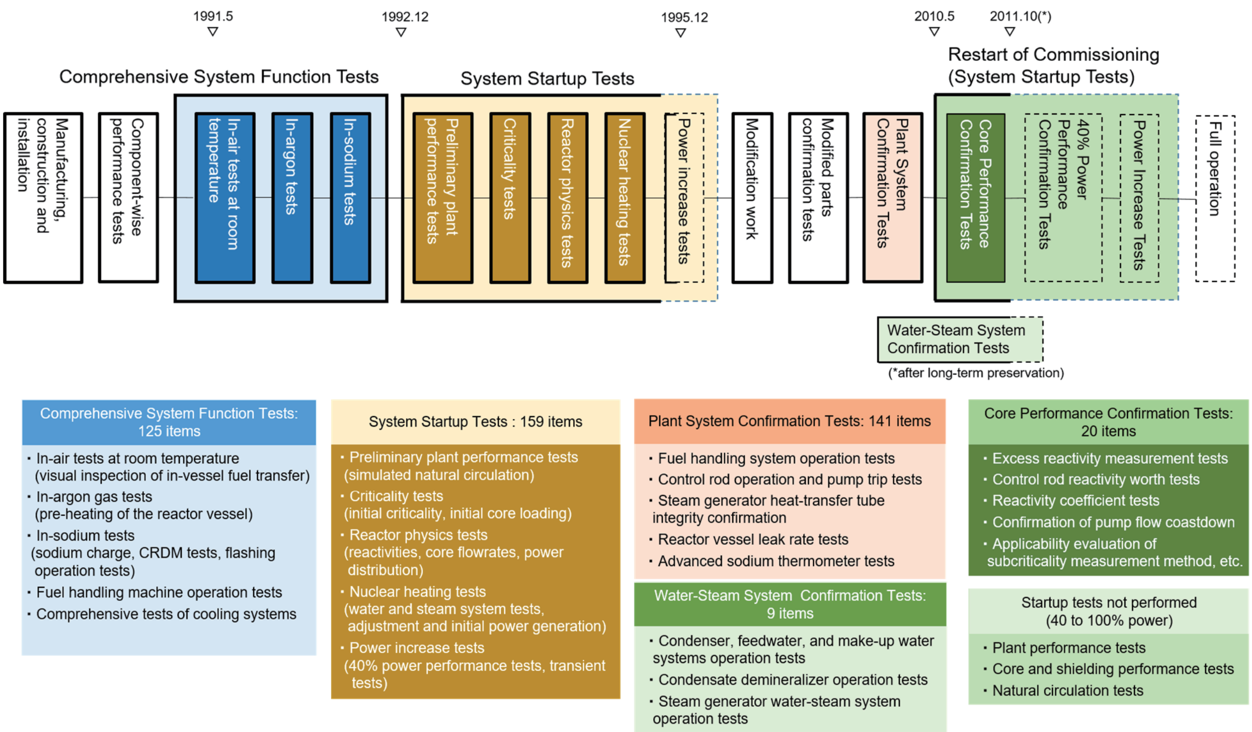
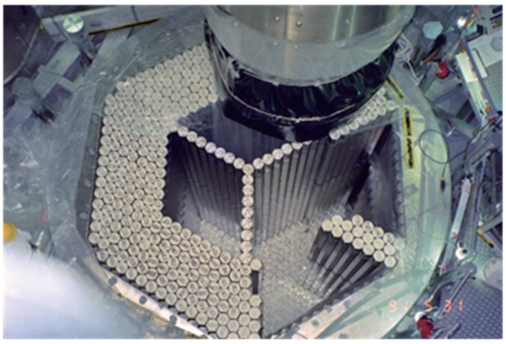
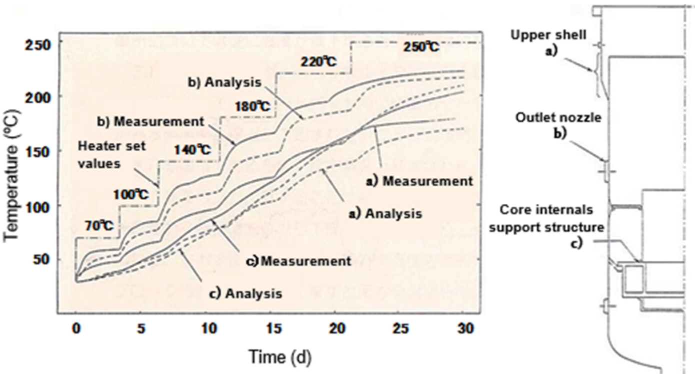


Fig.3-1 Major commissioning steps



Core construction with dummy subassemblies



Reactor preheating performance

Sodium charge

Measurement of primary piping dislocation

Fig.3-2 Examples of Comprehensive System Function Tests

nature of Monju, the achievements of FR development in Japan was evaluated and the technological issues were identified for future FR commercialization.

In planning the test items, many experts in the FR area examined the needs, feasibility, and foreign reactor incidents, while taking account of proposals from future reactor design and R&D teams. As a result, a total of 159 items were identified: 21 on the preliminary plant characteristics, 26 on the core characteristics, 10 on the shielding characteristics, and 102 on the plant characteristics. SST were planned for each of the test steps at zero, partial, and rated powers.

The power level for the first partial power test step was set at an electric power of 40%, at which water is supplied to the SGs and automatic reactor operation is started. It takes less than one year to complete SST for today's LWRs; however, for Monju, as a prototype reactor, the plan for SST was prepared as a two-year program from initial criticality to the start of full-power operation.

Special test devices and facilities were developed in planning the tests. The large ones manufactured include:

- Experimental fuel subassemblies designed to obtain data for the evaluation of power distribution,
- A device for handling activation foils loaded in the experimental fuel subassemblies,
- A device for placing the flowmeter onto the fuel subassemblies to measure the core flow distribution, and
- A device for measuring the temperature distribution in the upper sodium plenum inside the RV.

Additionally, an online network was established to collect and record the data for testing as well as for general plant process parameters at high speed (with an interval of 0.1 s), enabling centralized data storage and evaluation.

The criticality and reactor physics tests were led by the PNC staff, and the nuclear heating and power increase tests were performed by joint teams staffed by engineers from the vendors in charge of the relevant equipment. Many young engineers also participated in these teams to be trained for the future.

3.3.2 Preliminary plant performance tests

The preliminary plant performance tests

were scheduled prior to core fuel loading. During the integrated cooling system test in SKS, personnel were trained to master the skills for plant operations and test management. Through the preliminary plant performance tests, all operating crew and test personnel became accustomed to test operations and management. In addition, rehearsals were held to help the relevant personnel master the complex operating procedures included in the transportation, installation, operation, assembly, and disassembly of the in-vessel flow measurement device and the neutron detection element handling device. These led to successful implementation of the reactor physics test.

Concerning the preliminary evaluation of natural circulation in the primary cooling system, the primary and secondary cooling systems were heated up to 325°C by heat input from pump, and the core flow rate was measured after the auxiliary cooling system was activated by a reactor trip signal and the primary pump pony motor was stopped. As a result, a core flow rate of 80 m^3/h (larger than 1% of rated flow rate) by natural circulation without power supply was measured as predicted (Fig. 3-3).

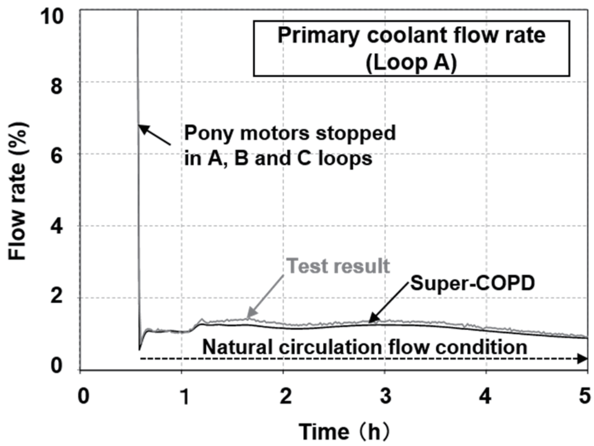


Fig.3-3 Preliminary test of primary system natural circulation (natural circulation flow rate of 1% was observed)

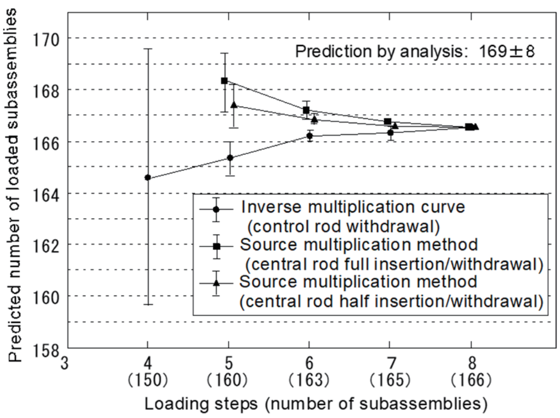


Fig.3-4 Fuel loading in criticality approach

3.3.3 Initial criticality achievement (criticality test)

A total of 108 inner core fuel subassemblies were loaded into the core in two batches from October 13, 1993. Loading of outer core fuel subassemblies (90 in total) into the core started on January 27. When 60 outer core fuel subassemblies were loaded in 7 batches, initial criticality was achieved on April 5, 1994, with the core consisting of 168 core fuel subassemblies (Photo 3-1).

Photo 3-1 Initial criticality (1994/4/5 10:01)

Attention was paid to the following points during the initial criticality approach.

(1) Criticality prediction

Criticality was predicted by both analysis and measurement. Analytical prediction was performed using the FR core analysis system (JFS-3-J2 data, CITATION code, etc.) that had been developed through the analysis of data from the U.S.-Japan joint research for FR critical experiment (JUPITER program), and by use of the correction factors evaluated based on the MOZART experiment, a series of tests to mock up the Monju core.

In actual fuel loading steps, the number of fuel subassemblies to be loaded in the subsequent step was estimated using the inverse multiplication method. The estimated range was narrowed by the source multiplication method with the central control rod inserted or withdrawn to change the level of subcriticality (Fig. 3-4).

(2) Neutron instrumentation system

The criticality approach was monitored using two different neutron instrumentation systems (NIS): two neutron-source-range detectors and two fuel-loading-range detectors that were installed outside and inside the RV, respectively. During the approach, neutron source subassemblies were arranged differently from those in normal operation to improve the efficiency of monitoring neutron multiplication factor. The guide tube of the fuel-loading-range detectors was also used for the shielding measurement in the power increase tests.

(3) Efficient fuel loading

The equipment hatch that covers the top of the RV is open for fuel loading, whereas it is close in reactor operation. In early fuel loading steps, criticality approach operation, that is, control rod operation, was performed with the hatch open. After predicted subcriticality became less than 1%Δk, the criticality approach operation was performed with the hatch closed. Since about one week was required for the opening/closing of the hatch and fuel loading step, new fuel subassemblies were temporarily loaded in the in-vessel rack and the hatch was closed for efficient fuel loading. Since experience in Joyo raised concern about the influence of fuel subassemblies loaded in the in-vessel rack on NIS measurement, fuel subassemblies were loaded only in the 6 rack positions (out of 10) having less influence, and less influenced NIS detectors were used in the criticality approach.

Before the final step of criticality approach, it was predicted that criticality could be reached by loading the 167-th fuel subassembly. Since Monju's initial criticality was becoming a public concern, 2 fuel subassemblies were added in the final loading step to ensure that criticality was reached. Initial criticality was achieved with 168 fuel subassemblies (Photo 3-2).

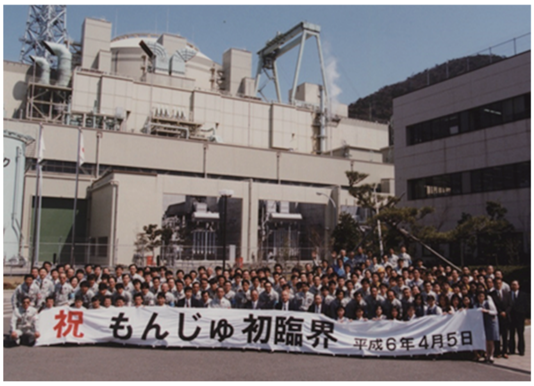


Photo 3-2 Commemorating the initial criticality

3.3.4 Reactor physics tests

While Joyo used core fuel consisting of enriched uranium or plutonium with a higher content of ^{239}Pu fabricated from the reprocessing of gas-cooled reactor spent fuel, Monju used core fuel consisting of degraded plutonium (i.e., plutonium with high contents of ^{240}Pu and heavier isotopes relative to ^{239}Pu) obtained from the reprocessing of LWR spent fuel. Consequently, the Monju core is configured with unique fuel compositions that cannot be simulated by critical assemblies. Thus, a variety of the reactor physics tests were performed for half a year.

After the initial core configuration, the experimental fuel subassemblies were loaded in the core, irradiated, and discharged from the core for the evaluation of power distribution. This procedure was repeated six times. In the intervals of these procedures, various reactivity worths were measured, including for control rod, sodium temperature, coolant, fixed absorber, and fuel subassembly.

(1) Reactivity worths

The reactivity worths were measured and compared with the design values and detailed analyses.

In the measurement of control rod worth, the reference control rod worth was measured at the core center by the asymptotic period method, and then control rod worth at the other positions were measured by the replacement method. The modified neutron source method was tentatively employed to measure reactivity under subcritical conditions; however, it was not successful due to the restriction of neutron detector positions and insufficient count rates. The reactivity meter based on inverse kinetics analysis was effectively used in the test.

The temperature reactivity was measured from 200 to 300°C by changing the core temperature by heat input from pump. The flow reactivity was measured by changing the flow rate from 49 to 100%.

Coolant reactivity was measured by the difference in reactivity with and without sodium around the core center, where void reactivity is positive. Six experimental fuel subassemblies each with sodium or void (helium gas) regions around the axial mid-plane were loaded around the core center for the measurement.

(2) Breeding ratio (power distribution characteristics)

The power distribution characteristics test was the largest in scale. It took considerably longer time for preparation and measurement than any other tests on reactor physics. Experimental fuel subassemblies containing activation foils (Pu, U, Ni, Au, etc.) were loaded and irradiated in the core, and then the reaction rates of the foils were measured. The experimental fuel subassembly is a special subassembly in which a neutron detection element containing the foils is inserted in the central area of a normal fuel subassembly (equivalent to 7 fuel pins). Five core fuel subassemblies, three blanket fuel subassemblies, and four neutron shielding subassemblies were prepared. The effectiveness of an anti-floating mechanism of the detection element was demonstrated in the water hydraulic test loop in designing the experimental fuel subassembly.

Since the neutron detection elements were replaced with new ones for every irradiation of experimental fuel subassemblies, a special device was developed and assembled on site to install and discharge the elements (Fig. 3-5 (a)). Exchanging the elements required as many as several dozen workers.

The neutron detection elements irradiated and activated in the core were connected to a glove box and cut inside the greenhouse, and the foils were discharged (Fig. 3-5(b)). The radioactivity of the activated foils was determined by gamma-ray measurement using a Ge solid-state detector, and then converted to the reaction rates.

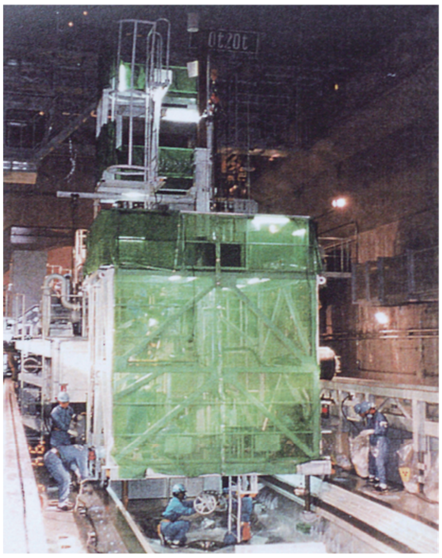
The positions of the experimental fuel subassemblies were selected in a one-twelfth sector (30 degrees) in consideration of the rotation symmetry of core configuration.

The test results showed that the ratios of calculation to measurement (C/E value) of the ^{239}Pu fission reaction rates, which are closely related to the power distribution characteristics, deviated from unity by a maximum of 3% and 5%, respectively, in the core and blanket fuel regions. These values are within the design margins for power distribution: ±5% for the core fuel region and ±10% for the blanket fuel region.

Using the fission reaction data, the C/E values for the maximum linear heat rate in the core region of the initial core was evaluated to be from 1.003 to 1.009. The breeding ratio was also evaluated to be 1.185, which agreed well with the design target of 1.2.

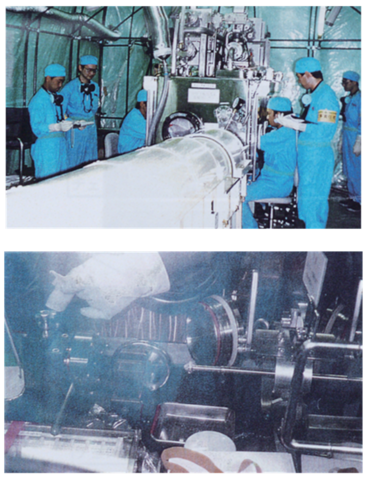
(3) Core flow rate

The core flow rate distribution was measured under reactor shutdown condition by sequentially placing the flow rate measurement device on the top of each core element. The device has a configuration similar to the fuel handling machine that connects the gripper to the top of a fuel subassembly. The device is equipped with an electromagnetic flowmeter in place of the gripper. The handling of the device is similar to that of the fuel handling machine. See 5.2.1 for the measured results.



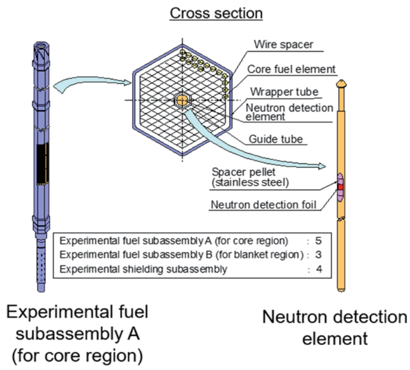
《Equipment to handle neutron detection element》

(a) Replacement of neutron detection elements



《Equipment to handle neutron detection foil》

(b) Take-out of neutron detection foil (using globe box)



Neutron detection foil

Total 2,000 pieces

(1) fission foil

Pu-239, U-235, U-238, Np-237

(2) activation foil

a) (n, p) reaction : High energy neutrons Ni, Ti, Fe (Fe-54)

b) (n, γ) reaction : Low energy neutrons Au, Fe (Fe-58), Na, Sc, Co

Fig.3-5 Tests using experimental fuel subassembly

3.3.5 First connection to power grid and power operation (nuclear heating test and power increase test)

The nuclear heating test started in September 1995 following the startup procedure of the reactor including reactor power increase, preheating and startup of the water-steam system, adjustment of the control systems, and confirmation of system performance.

When the evaporator outlet temperature reaches the saturation temperature (330°C) at the operating pressure (127 kg/cm^{2}G) with the increase in reactor power, water starts to boil and steam is generated in the evaporator. In the subsequent operation of steam admission into the superheater, the control system is adjusted and the operation procedures are revised. These include the adjustment of the switching of the turbine bypass system, while collecting information about the water-steam system characteristics. Various troubles, such as a decrease in flash tank pressure, occurred during the test, and appropriate measures including equipment modification, were taken to resolve the problems.

On August 29, 1995, Monju succeeded in generating electricity and connecting to a power grid as Japan's first FBR, marking its first step as a prototype reactor (Photo 3-3). Subsequently, the reactor power was increased step by step to 40% rated power (see 10.1.1). The total amount of power generated is 102,325 MWh (883 effective hours).

Photo 3-3 Central control room at initial power generation (grid parallel-in)

With the increase in reactor power, the turbine system characteristics were also confirmed, and various adjustments, such as the revision of startup conditions for the steam control valves and turning equipment, were successfully performed. In addition, the performance of the individual systems was confirmed at various reactor power levels.

Data on steam blow characteristics from the SGs were obtained in a plant (turbine) trip test at an electric power of 40% (Fig. 3-6). It was confirmed that the performances of feedwater stop valve and superheater drain valve as well as the depressurization characteristics are appropriate.

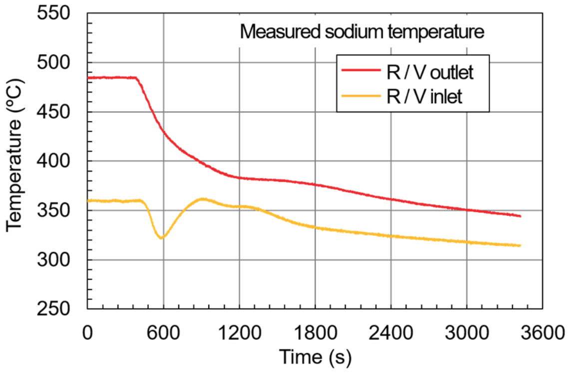


Fig.3-6 Plant trip test at 40% power operation

(1) Temperature distribution in shield plug

The temperature distribution in the shield plug was measured to confirm the function of the nitrogen gas cooling system and to collect information about the temperature rise behavior in case of the loss of the gas cooling. The circumferential temperature distribution of the rotating plug was confirmed to be uniform regardless of elevation. It was planned to adjust the cooling system flow rate during the test at rated power.

(2) Pump flow coastdown

In the primary cooling system characteristics test, the coolant flow rate was increased to 50% under power operation, and various behaviors of the main circulation pumps and systems were investigated. The flow coastdown characteristics of the primary main circulation pumps was confirmed to be as expected and consistent in the three loops as shown in Fig. 3-7 in the plant trip test at 40% rated power. It was also confirmed that the coolant temperature has negligible effect on the pump coastdown characteristics.

(3) Hydrogen concentration in secondary cooling system

The cooling and purification system characteristics were confirmed for the secondary cooling system. The hydrogen concentration in the system is important for monitoring water leak in the SGs and evaluating the removal performance of cold traps. The histories of hydrogen concentrations in sodium and cover gas were measured during SST when reactor power was increased to as high as 40% (Fig. 3-8). Based on these measurements, the hydrogen behavior corresponding to different operating conditions such as coolant temperature was confirmed, and the permeability of hydrogen from the SGs was evaluated. These results were then used to optimize the alarm setting for abnormal hydrogen concentration to improve reliability of a water leak monitoring device.

(4) Sodium vapor behavior

Concerning the primary argon system, an anomaly was found; namely, that the pressure difference between the RV vapor trap outlet and the compressor inlet increased to 5,000 mmAq (49 kPa), much higher than the normal value of 130 mmAq (1.3 kPa). This was caused by the transport and accumulation of sodium vapor toward downstream. Consequently, a filter was additionally installed (see 10.2.4).

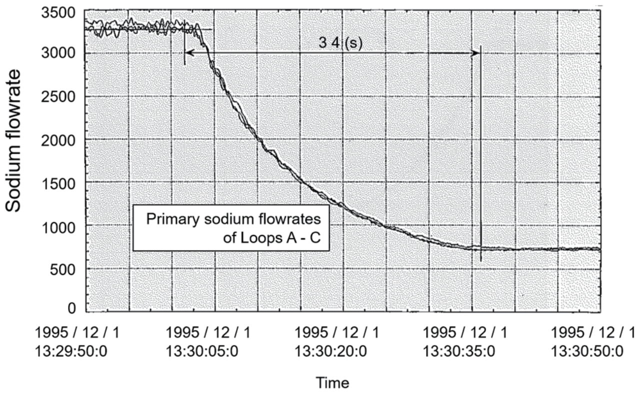


Fig.3-7 Flow coastdown characteristics of primary circulation pump

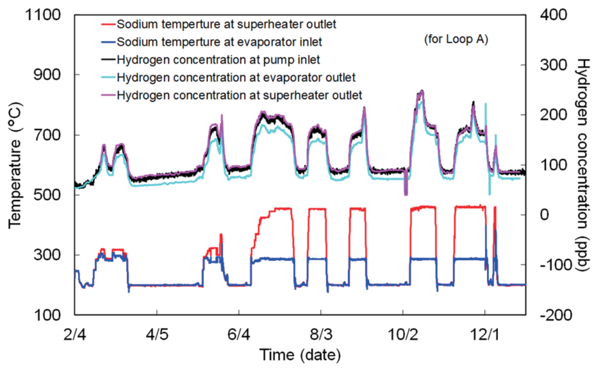


Fig.3-8 Hydrogen concentration in secondary cooling system

(5) Safety margin evaluation

Safety margins in reactor design were evaluated based on the data obtained in the 40% power test and SKS. In the design stage, large safety margins were adopted for the initial and analytical conditions conservatively to obtain more severe results. The results of safety assessment in the design stage were compared with those evaluated using the data measured in Monju. The evaluated accident events included the PHTS circulation pump seizure (the pump stick accident). It was confirmed that there were significant margins in the design, such as an evaluated cladding temperature of 702°C, much lower than the safety assessment result of 796°C (Fig. 3-9).

3.4 Resumed SST

3.4.1 Plan revision

To resume SST after the Secondary Sodium Leak Accident, “Core Performance Confirmation Tests” at zero power and “40% Power Performance Confirmation Tests” were added to the original plan for SST (Fig. 3-1), in consideration of the fact that fuel and plant systems and components were kept in a long-term standby state. The latter Tests were specifically aimed at confirming the startup and operation of the water-steam and turbine systems. In addition, to ensure the core reactivity required to complete SST, new fuel subassemblies were fabricated schedule.

In the development of the revised plan for SST, “Special Committee on the Study of Monju Availability for Various Research and Education” was established in the Atomic Energy Society of Japan to collect and discuss proposals for the testing of neutronic characteristics, thermal hydraulics, plant dynamics, etc.

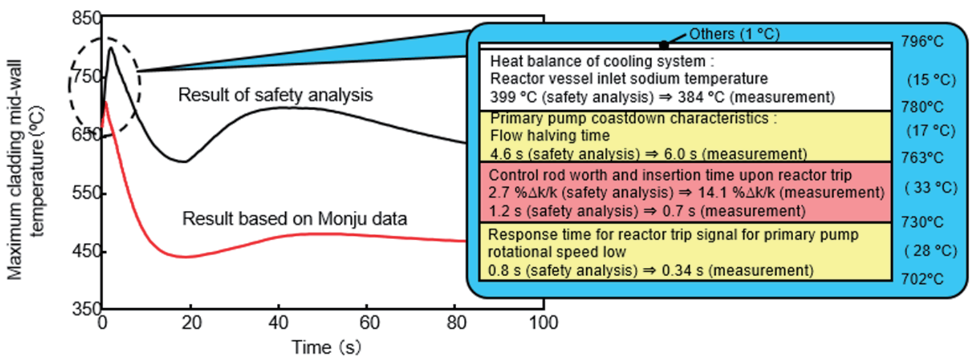


Fig.3-9 Safety margin for design basis accident analysis (primary pump stick accident)

The report of the Committee were reflected in the revised plan. The items newly added in the revised plan included evaluation of the applicability of subcriticality measurement method, evaluation of advanced sodium thermometer performance, a SG tube water leak simulation test, and confirmation of small-diameter pipe vibration.

Before resuming SST, tests of 141 items were performed as "Plant System Confirmation Tests (PKS)" with reference to SKS to confirm that the planned SST could be safely resumed. PKS were performed in the two years, from August 2007 to August 2009.

3.4.2 Americium-containing core characteristics (Core Performance Confirmation Tests)

As the first part of the resumed SST, Core Performance Confirmation Tests were performed for two and a half months from May 2010. Inspectors from the Nuclear and Industrial Safety Agency attended the tests, and officials from MEXT were stationed at the site as well. The requirement for transparency regarding all incidents, including minor troubles, created an atmosphere of anxiety and tension both on-site and in the surrounding communities. Even under such circumstances, the Confirmation Tests were completed successfully as scheduled (Photo 3-4).

Photo 3-4 Core Performance Confirmation Tests

Since about 1.5% of ^{241}Am accumulated in the core fuel due to the decay of ^{241}Pu in the long shutdown period, it was expected that useful data for future study on the burning of minor actinides in FRs would be obtained. Since the refueling was not sufficient to compensate the reactivity loss due to the ^{241}Pu decay, core reactivity was predicted carefully by maximum use of the results of the past tests performed in 1994 and the nuclear data uncertainties. As a result, criticality was achieved within the prediction range. The obtained data have been used for validation of the Japanese Evaluated Nuclear Data Library (JENDL-4.0), specifically the ^{241}Am nuclear data.

Other items performed in the Confirmation Tests included the feedback reactivity confirmation and the advanced sodium thermometer performance evaluation.

The feedback reactivity confirmation test demonstrated that the plant state is stabilized owing to the effect of inherent negative reactivity feedback mechanisms including the Doppler Effect, after addition of a positive reactivity (2 to 6 cents) at the critical condition (Fig. 3-10).

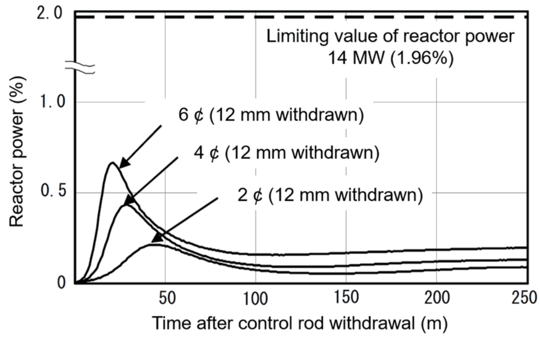


Fig.3-10 Change in reactor power after positive reactivity insertion

The advanced sodium thermometer performance evaluation test was intended to confirm the performance of a newly developed ultrasonic thermometer, which can measure the sodium temperature outside the pipe. An ultrasonic thermometer was installed in the SHTS (loop C) and temperature data were compared with those of conventional thermocouples equipped in the loop. Data processing methods were developed to extract temperature data efficiently from the signal waveforms. It was confirmed that the temperature data by the ultrasonic thermometer agree well with those by the conventional thermocouple (Fig. 3-11).

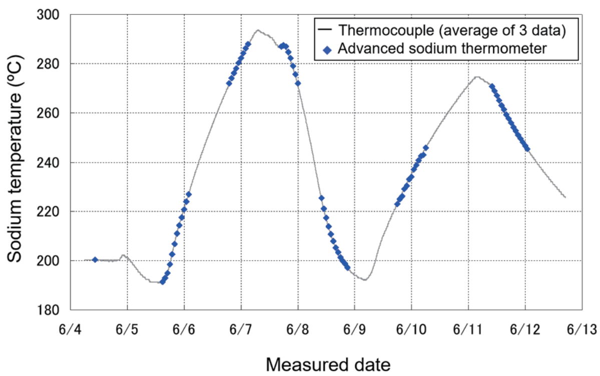


Fig.3-11 Performance of ultrasonic thermometer

3.4.3 Water-Steam System Confirmation Tests

To resume SST, the long-term storage state of the water-steam system was canceled and inspection for resumption was performed from April to December, 2010 (Photo 3-5).

Photo 3-5 Inspection of turbine

Subsequently, in February 2011, the Water-Steam System Confirmation Tests were started. During the tests on nine items, water supply and flushing operation were performed for the condensate and feed-water systems, condensate demineralizer, etc. to confirm the conditions of pumps and the control system. As a result, no anomaly, such as water leak or abnormal vibration, was identified. In addition, operation testing of the steam turbine (gland exhaust fan) and the generator system (gas, cooling, and oil systems) was performed, while water chemistry was carefully controlled to minimize the impurities. Through these tests, it was confirmed that long-term storage was appropriately maintained and the relevant systems were operable without problem.

However, taking into account the social environment in the wake of the 1F Accident that occurred during these test series, it was decided to place priority on the so-called "stress test" against severe accidents and other urgent safety measures to ensure the safety of Monju. Therefore, the Water-Steam System Confirmation Tests were interrupted before supplying water to evaporators; and in October 2011, the water-steam system was brought back to a storage state again.

3.5 Unfinished SST

SST were terminated at 40% rated power without accomplishing the goals of a prototype FBR in the R&D stage: the confirmation of plant performance through SST, verification of design and manufacturing of the components based on the domestic technologies, and the identification of challenges for future improvement.

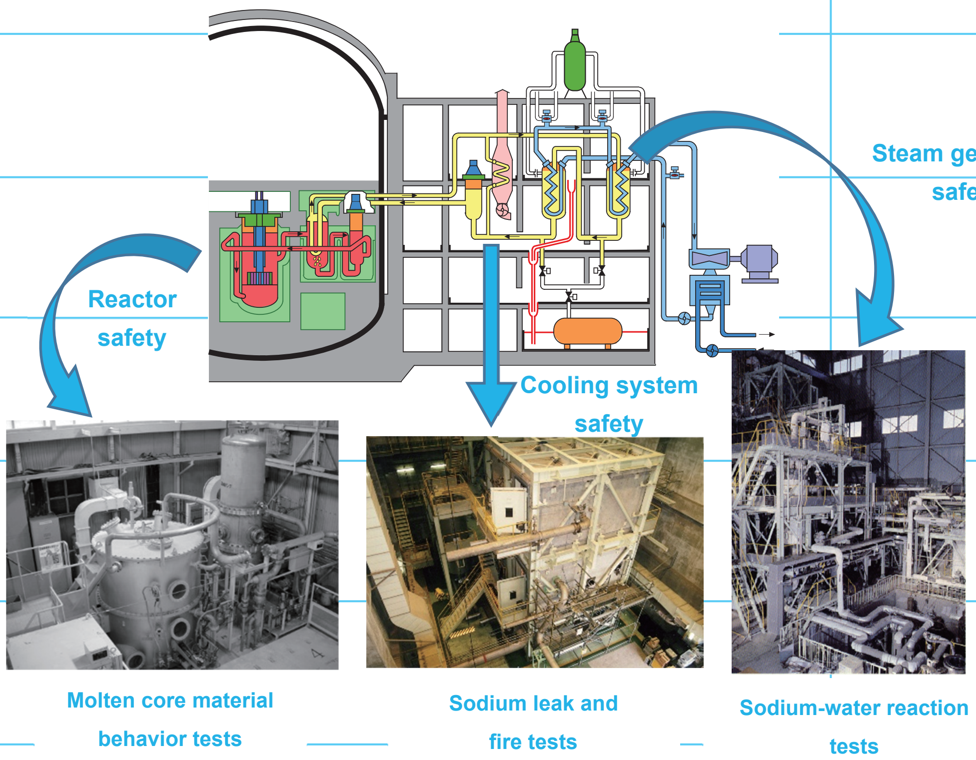
Regarding the core characteristics, valuable data of an actual FBR core fueled by degraded plutonium fuel were obtained from the reactor physics tests and the breeding ratio was indirectly confirmed. On the other hand, it was not possible to obtain data associated with power increase operation, such as the changes in reactivity, power coefficient, feedback and Doppler reactivity coefficients. The data were not obtained either on burnup reactivity, the change in fuel composition, and the burnup behavior of fuel associated with reactor operation at the rated power.

Although sodium-related components and equipment comprising the primary and secondary cooling systems were developed, designed, and manufactured through a number of mockup tests and design studies to be operable at the rated power, data related to performance and behavior at the rated power were not obtained; namely, their performance was only partially verified. Other data that were not obtained were data over the entire plant system at the rated power, such as the cooling capability by natural circulation, the transport behavior of radioactive materials, the combined operational controllability of water-steam and sodium systems, and the transport behavior of hydrogen.

Although these test items remained unfinished, it is believed that the data obtained in Monju are extremely valuable and could be effectively used for future FR development in Japan.

Post stamp commemorating the initial criticality of Monju (issued on May 24, 1994)

4. Safety



- **In order to ensure the safety of Monju at a level equivalent to or higher than that of LWRs, safety design was carried out in full consideration of the characteristics of sodium-cooled FRs.**
- **Through the design, construction and operation of Monju, the safety design policy for sodium-cooled FRs was established. In addition, the safe performance was confirmed through the commissioning of Monju.**
- **Technological basis for safety evaluation of sodium-cooled FRs was established by reflecting the domestic safety research results as well as the safety analysis methods and experimental data obtained through international cooperation.**
- **It was confirmed that the risk level of Monju is extremely low using the methods of probabilistic risk assessment.**
- **Experience of accidents and failures, including the Secondary Sodium Leak Accident, were used to improve safety. In addition, it was confirmed that the safety of Monju could be secured in the event of a station blackout such as occurred in the 1F Accident.**

4.1 Safety characteristics of Monju

To ensure the safety of reactor facilities, it is important to appropriately take into account the design features and inherent characteristics of the reactor. Monju, a sodium-cooled FR with MOX fuel, has the following inherent safety features:

- Sodium coolant with high thermal conductivity and excellent core cooling capability,
- A low-pressure system with a significant margin to the coolant boiling point,
- Capability of operation in stable liquid coolant conditions against pressure variation, and
- Having negative reactivity effects based on the fuel Doppler and expansion effects.

Accordingly, Monju has inherent negative reactivity feedback characteristics and stable operational controllability against disturbances. The addition of excessive positive reactivity due to coolant boiling is unlikely over the entire operation range.

Although the operational performance of Monju during the commissioning tests was limited to short-term partial power operation, stable and safe reactor operational controllability was confirmed. Concerning the operating experience of sodium-cooled FRs, stable and safe reactor operations have been demonstrated in a number of foreign and domestic FR plants, including Joyo.

4.2 Ensuring safety based on FR features

To ensure the safety of reactor facilities, it is essential to provide multiple physical barriers for the confinement of radioactive material. In Monju, similar to LWRs, it was required that the exposure dose during normal reactor operation be reduced following the ALARA (as low as reasonably achievable) principle, and that measures be taken to prevent the occurrence of accidents and mitigate the consequences based on so-called "defense in depth" policy. Namely, the following multilayer safety measures were taken:

a) Prevention of anomalies by improving the quality and reliability of the structure, systems and components comprising the reactor facility and of operator actions,
b) Preventing anomalies from escalating to accidents that may lead to the abnormal release of radioactive material,
c) Mitigation of the consequences (i.e., prevention of significant core damage and abnormal release of radioactive material) in the event of an accident, and
d) Appropriate control of the release of radioactive material in the event of a beyond design basis accident.

The defense-in-depth concept required by the recent international standards consists of five layers, in which the above item d) is explicitly called "measures against severe accidents", and the fifth layer "e) off-site consequence mitigation and nuclear disaster prevention" is added.

To facilitate the Application for Reactor Installation Permit of Monju, the NSC established the Safety Evaluation Policy of Liquid-Metal Fast Breeder Reactors (Evaluation Policy)[4-1], under which the Safety Review was performed. The Evaluation Policy stipulated that the safety review guidelines for LWRs should be used as the baseline. The policy also required countermeasures in case of sodium leak and measures against sodium-water reaction in case of SG tube rupture in order to address the use of chemically reactive sodium, which is specific to FRs. Furthermore, the policy required to mitigate abnormal release of radioactive material associated with the generation of mechanical energy in case of a hypothetical core disruptive accident (CDA), a historical safety concern in FR development in the world.

FRs, being a low pressure system, have significant safety margins against the anticipated operational occurrences and accidents (design basis accidents) that are selected based on the same philosophy of LWR safety evaluation. One feature of FR safety is the absence of design basis accidents that would directly affect the CV integrity, such as the loss of coolant accident (LOCA) postulated in LWRs. Other features of FRs with plutonium fuel include: a) positive sodium void reactivity in the core central region and, b) the possibility of the addition of large positive reactivity upon fuel melting and movement resulting from the fact that the core is not designed in its most reactive configuration. Consequently, there is potential risk of significant energy release due to the occurrence of a recriticality event. This background is the reason CDA is considered from the earliest phase of Monju design by addressing the fourth defense in depth layer (see d) above), which is not taken into consideration in LWRs.

4.3 Establishment of safety design policy

4.3.1 Basic policy for safety design

The basic Monju design principle was to take into consideration the features of FRs according to the Evaluation Policy and, of course, to comply with the safety requirements common to LWRs for power generation. In addition, not only experience in the safety design, evaluation and licensing of Joyo, but also the information and experience on the safety design and licensing of foreign prototype class FRs that were designed earlier than Monju (specifically, CRBR in the U.S. and SNR-300 in Germany) were acquired and effectively used.

4.3.2 Development of safety design policy for Monju

Development of the Safety Design Policy for Liquid-Metal-Cooled Fast Breeder Reactor (Safety Design Policy) was based on the structure of the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities, with careful consideration of the features of FRs, using the safety design standards and practices of earlier FRs, such as the General Design Standards for CRBR.

The structure of the Safety Design Policy is shown in Table 4-1. It was an important achievement for JAEA to have established a basic safety design policy for sodium-cooled FRs, comparable to the regulatory guide for safety review of LWRs, and to build a consensus with regulatory authorities and experts through the Safety Review. The specific contents of and design conformity with the respective design policies are described in the Application for Reactor Installation Permit[4-2].

4.3.3 Safety design for major systems and safety functions

Among the safety design items in the Safety Design Policy, the major systems, important from the perspective of fundamental safety functions: “shutdown”, “cooling”, and “confinement”, are explained below with careful consideration of the features of FR system.

(1) Reactor inherent safety characteristics

Monju has extremely high self-stability against deviations from normal operating conditions thanks to high thermal conductivity of sodium, its good stability against pressure fluctuation, and the fact that reactor operation with single-phase liquid flow is expected. Monju also has inherent safety characteristics, such as the negative fuel Doppler effect with temperature increase and negative reactivity effect due to fuel expansion. This inherent negative reactivity feedback characteristics is effective over the entire operation range.

(2) Plant protection and reactor shutdown systems

The plant protection and reactor shutdown systems are designed to have multiplicity or diversity, and independence, as well as fail-safe features. The reactor shutdown system consists of two systems: the main shutdown sys-

Table 4-1 Classification of Monju Safety Design Policy

General matters for entire reactor facility	Compliance to codes and standards, natural phenomena, human-induced events, environmental conditions, sodium, flying objects, fire, prohibition of sharing, single failure, loss of power supply, testability, evacuation route, communication system
Reactor, and instrumentation and control system	Reactor design, fuel design, reactor inherent characteristics, power oscillation control, instrumentation and control system, electrical system, control room, remote shutdown function
Reactor shutdown system, reactivity control system and plant protection system	Independence, shutdown capacity, maintenance capability during accidents, and shutdown margin of the reactor shutdown systems; maximum control rod reactivity worth, safety functions of the reactivity control system; plant protection system functions during transients, accidents and faults, multiplicity, independence and testability of the plant protection system, independence of the plant protection system from the instrumentation and control system
Reactor cooling system and intermediate cooling system	Function, integrity, leak detection, and damage prevention of the reactor coolant boundary; maintaining reactor coolant, boundary for reactor cover gas, etc., intermediate cooling system, cooling water system, decay heat and other residual heat removal
Reactor containment facility	CV functions, annulus air clean-up system, damage prevention of containment boundary, penetrating piping, isolation valve
Fuel handling and waste disposal systems	Nuclear fuel storage and handling, criticality prevention of nuclear fuel, monitoring of nuclear fuel handling areas; disposal of gaseous, liquid, and solid radioactive wastes, solid waste storage system
Radiation protection and control facilities	Radiation protection, radiation control facilities, radiation monitoring
Others	Consideration to reliability and operator actions

tem having both reactivity control and emergency shutdown functions, and the backup shutdown system having only the emergency shutdown function. In case one of the two reactor shutdown systems is not operable, the remaining system has a shutdown reactivity margin sufficient to emergently shut down the reactor from power operation to the cold shutdown state and maintain subcriticality.

Although the two independent shutdown systems commonly use solid absorber rods, in order to prevent a simultaneous failure due to a common cause, diversity was considered as follows: design and manufacturing were performed by different vendors, different structure was adopted for the delatching part for emergent control rod insertion, and different principles were applied to the control rod acceleration mechanism. Safety considerations for the CRDM, such as diversity, are shown in Fig. 4-1.

(3) Decay heat removal (auxiliary cooling system)

The system to remove decay heat and other residual heat consists of three independent systems, which are comprised of the PHTS, part of the SHTS, and the auxiliary cooling system. The ultimate heat sink of the decay heat removal system is the air, i.e., a system independent of the seawater cooling system, which is much different from LWRs.

In case of the loss of power supply beyond the design basis, since coolant sodium is in a stable liquid state over a wide temperature range and has excellent heat transfer characteristics, natural circulation capability is ensured with a driving force given by the density difference due to temperature difference. This is realized by safety design that provides sufficient elevation difference between the heat source and heat sink, as shown in Fig. 4-2. Natural circulation heat removal provides high reliability as a passive safety function with no need of a power source, and, unlike LWRs, this can significantly reduce reliance on the emergency power supply and feedwater systems.

(4) Decay heat removal (maintenance cooling system)

The maintenance cooling system is designed to remove decay heat from the core and dissipate it to the air through the air coolers during maintenance of the heat transport systems, although it is not used during normal operation. In addition, in case of simultaneous loss of core cooling capability in all the three heat transport systems after an emergent reactor shutdown, core cooling can be still achieved by operating the maintenance cooling system.

Furthermore, even in a serious accident sequence from the primary coolant leak accident with the failure to pump up sodium to the RV from the overflow system and decrease in RV sodium level, core cooling can be achieved by operating the maintenance cooling system.

(5) Ensuring the reactor coolant level

To ensure decay heat removal from the RV even in case of primary coolant leak, the following measures are taken:

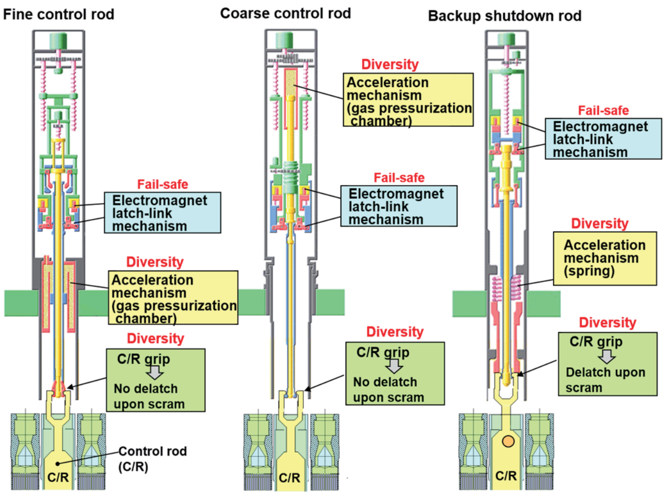


Fig.4-1 Structure of CRDM (fail-safe and diversity)

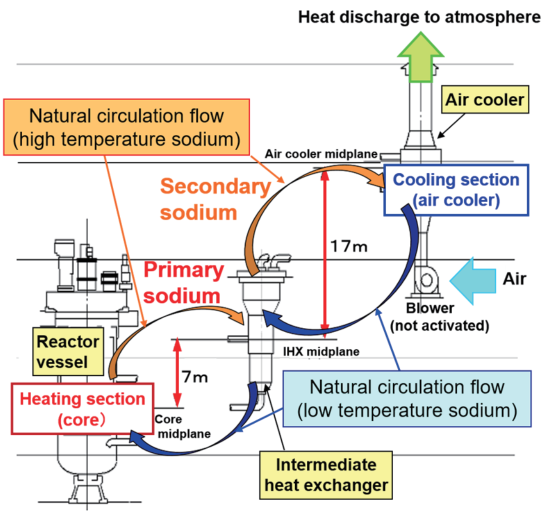


Fig.4-2 Decay heat removal by natural circulation

- High-elevation pipe routing of the primary cooling system and the installation of the guard vessel would limit the amount of sodium leaked following pipe break and would maintain the reactor coolant level required for coolant circulation in the PHTS.
- Restoration of the coolant level by pumping up sodium as needed in case of a coolant leak accident from the overflow system that is operated to maintain a constant RV coolant level during normal operation.

(6) Confinement of radioactive material

The CV is an important engineered safety feature that forms the final barrier of multi-layered physical barriers for the confinement of radioactive materials. As for FRs, which are low pressure systems unlike LWRs, there is no loading mechanisms that significantly affects the integrity of the CV within the design basis accidents; however, considering its importance, the Safety Design Policy set requirements similar to those for LWRs. Namely, the CV is designed and manufactured to withstand the specified pressure and temperature conditions and to maintain the leak rate at a value no greater than the allowable value. Additionally, periodic testing, such as the inspection of the leak rate, is performed to confirm that the functions are maintained well.

In FRs, the reactor primary system boundary can also provide a closed barrier against the dispersion of radioactive material from the reactor, by use of the features such as a low pressure system using sodium as coolant and the existence of an intermediate cooling system. The CDA analysis for Monju, described later, confirmed that the mechanical and thermal consequences of core melt progression could be appropriately accommodated within the reactor vessel.

(7) Safety consideration for the use of sodium

In the safety design related to the chemical reactions of sodium (measures against sodium leak and fire, and measures for sodium-water reaction), it is essential to avoid hampering the basic safety functions of shutdown, cooling, and confinement in case these chemical activity effects become more evident. Therefore, the plant was designed to ensure reactor shutdown or CV isolation by activating the plant protection system through early detection of sodium leak, and thereby to maintain the separation and independence between the cooling systems while mitigating chemical reaction effects to prevent the propagation of the effects to other safety-related systems. Specifically, the following safety considerations were taken in the design:

- The components that contain sodium and that have the liquid surface therein were designed with inert cover gas. The plant was also designed to prevent the loss of safety functions due to sodium freezing.
- Design was performed to mitigate the effects of primary coolant leak that may cause radiation exposure during an accident. For this purpose, rooms with systems and components containing radioactive sodium are equipped with sodium leak detection systems to ensure early leak detection, and the rooms are filled with nitrogen gas with a low oxygen concentration (Fig. 4-3).
- To address sodium leak from the secondary cooling system to the air, in addition to early leak detection, consideration was given to preventing the loss of safety functions due to the effects of sodium fire. Inter-system separation was designed for equipment important to safety to mitigate the effects of sodium leak.
- Since hydrogen is generated when sodium reacts with water in concrete, a steel liner is installed over concrete to prevent direct contact between leaking sodium and concrete during a sodium leak accident.
- In case of a sodium-water reaction resulting from water leak from the SG heat transfer tubes, early detection of the tube rupture and mitigation of the effects of sodium-water reaction are ensured to safely cool the reactor (Fig. 4-4).
- Hydrogen generated from sodium-water reaction is released to the air through the reaction product container for immediate burning without accumulation in the plant.

(8) Ensuring seismic safety

Among the external events to be considered in the design, the occurrence of earthquakes is a major concern in Japan. Seismic design was performed to safely shut down and cool the reactor and to ultimately maintain stable cold shutdown state.

The original seismic design during the initial design and construction stage was based on the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (stipulated by the JAEC in September 1978, and partly revised in July 1981). The facilities are in a rigid structure and based on bedrock. All equipment is classified into S (the former As and A), B and C classes in terms of environmental effects due to radiation possibly caused

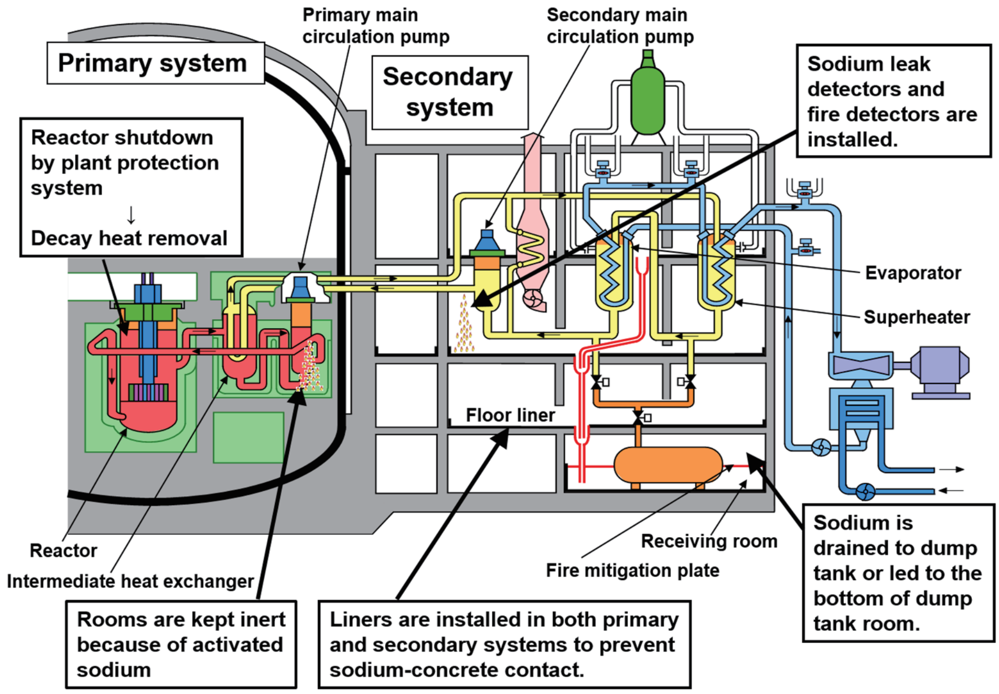


Fig.4-3 Safety assurance on sodium leak

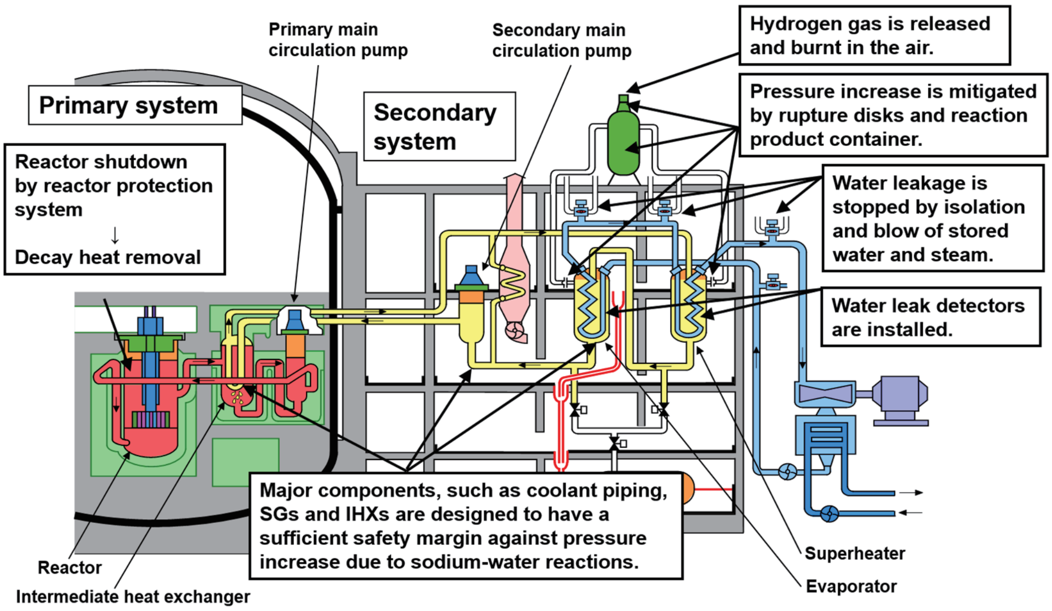


Fig.4-4 Safety assurance on sodium-water reaction

by earthquakes. For the respective classes, the active and passive seismic forces were specified (only passive seismic force for class C) and the structural design of buildings, structure, components, and piping was performed such that the stress generated by the combination of loads is within the allowable limit.

The philosophy of the seismic design for Monju is common to that for LWRs, and the seismic design methods for buildings and structure are the same as LWRs. However, for the equipment and conditions specific to sodium-cooled reactors, the design features of FRs should be considered. For example, since the components and piping systems use low-pressure high-temperature sodium and have a thin-wall large-diameter structure, considerations such as appropriate seismic support without constraining the thermal expansion displacement are required.

After the construction of Monju, the seismic design was reviewed again based on the Regulatory Guide for Reviewing Seismic Design (2006) that was revised considering new findings from the 1995 Southern Hyogo Prefecture Earthquake, etc. and additionally, considering the findings from the Niigataken Chuetsu-oki Earthquake in 2007, etc. (see 4.7.2).

(9) Classification of the importance of safety functions

The classification of the importance of safety functions for Monju was established with reference to the Regulatory Guide for Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities (1990), particularly for LWRs, enacted after the U.S. Three Mile Island accident, and the Design Considerations for Reliability was added to the Safety Design Policy. Namely, structures, systems and components having safety functions were designed so as to ensure sufficiently high reliability and maintain their functions according to the importance thereof. Reliability requirements, such as multiplicity, diversity, and independence, were specified in the individual safety design policies, and approved by the regulatory authorities through the Safety Review. In addition, attention was paid to the relationship (consistency) with the component classification and seismic safety classification related with the structural design during the design approval stage following the Safety Review.

The philosophy of the safety importance classification is the same as that for LWRs. Equipment and components with safety functions are classified into prevention systems and mitigation systems, and they are further classified into levels 1, 2 and 3 according to the importance of their safety functions. The classification of the importance of safety functions for Monju was prepared in consideration of the features of FRs, including that a sodium-cooled reactor is a low-pressure system, with reference to LWRs. Consequently, the safety functions required for high-pressure systems of LWRs are not required for Monju, and the importance of safety functions for mitigation systems against sodium leak and sodium-water reaction described in (7) was adequately classified.

4.4 Safety evaluation for Monju

4.4.1 Purpose of safety evaluation and event selection

The purpose of safety evaluation is to confirm the validity of the basic policy for safety design. The basic procedure for selecting safety evaluation items is common to that for LWRs.

- “Anticipated operational occurrences”

  Single component failure or malfunction, or single misoperation that is expected to occur once or several times during the operating life of the nuclear reactor facility and an event that may occur with similar frequency.

- “Accidents” (design basis accident)

  An event associated with abnormal conditions that is more serious and less frequent than the “anticipated operational occurrences”, but is likely to cause the release of radioactive material from the reactor facility.

The events added by the Evaluation Policy that are not required for LWRs include the “technically inconceivable event”, that is referred to as an “item 5 events” because it was stipulated as the 5-th item in Attachment II of the Evaluation Policy. Item 5 events include CDAs that were evaluated historically in foreign FRs. The purpose of Item 5 event analysis is to confirm the safety margin beyond design basis of the relevant reactor facility, and they were clearly approved as a beyond design basis accident in the Monju Safety Review.

- “Item 5 events”

  A postulated event with lower probabilities and higher consequences than those described as “accidents”. The relevance of event progression to the prevention measures must be sufficiently evaluated and the release of radioactive materials is appropriately suppressed.

Site evaluation accidents (Major and Hypothetical Accidents) are not discussed here, because the regulatory requirements were later eliminated when the NRA established the New Regulatory Standards.

The safety evaluation items were selected through systematic and comprehensive analysis and categorization of various abnormal events anticipated to occur internal and external to the plant, as well as considerations such as selecting a representative event that produces the most severe result. In addition, the safety evaluation items for the preceding foreign FRs were also referred to in selecting the technically inconceivable events. The safety evaluation items selected for Monju are listed in Table 4-2.

4.4.2 Analysis of accidents specific to FRs

(1) Size of pipe opening assumed

Prevention measures for coolant leak were appropriately taken through the specified in-service inspection for coolant boundaries and early detection of coolant leak according to the Safety Design Policy; however, in the safety evaluation, analyses were performed postulating a pipe break. Unlike LWRs, a high-pressure system, austenitic stainless steel with excellent ductility is used in sodium piping for the coolant systems of Monju. Due to the unlikeliness of brittle behavior and low system pressures, there is no risk for a crack before penetrating the pipe wall to rapidly propagate to cause pipe rupture, and accordingly, the so-called “leak before break” is expected.

Regarding the size of the break opening that is important from the perspectives of core cooling during coolant leak and thermal effects due to leaking sodium, it was judged to be sufficiently conservative and appropriate to assume that the break opening is a slit-like opening with a length of D/2, a width of t/2 (D: pipe diameter, t: pipe wall thickness), taking into consideration the fact that a fatigue failure mode due to crack propagation is dominant. Therefore, an opening area of Dt/4 was assumed.

(2) Primary coolant leak accident

In LWRs, a high-pressure system, it is likely that a primary coolant leak would immediately result in the loss of coolant in the reactor pressure vessel, while in FRs, a low-pressure system, coolant leak proceeds only gradually and the core cooling can be stably maintained through the measures for ensuring the reactor coolant level, such as the guard vessel. In addition, fire from leaking sodium and its thermal effect can be mitigated by filling the primary cooling system piping room with nitrogen with a low concentration of oxygen.

The pipe opening size assumed for accident analysis was set at Dt/4, as described in (1). However, the occurrence of guillotine failure of piping, a beyond-design-basis accident, was evaluated as one of the Item 5 events, and it was confirmed that severe core damage can be appropriately prevented by improving the safety margin through design considerations for limiting the coolant leak rate and other measures.

(3) Secondary Sodium Leak Accident

In the safety evaluation of sodium leak accidents, it is necessary, in terms of core cooling, to ensure plant system separation (i.e., no thermal effect of an accident loop on the other intact loops) regardless of the temperature and pressure rises resulting from chemical reactions. For the analysis of sodium fire, analysis codes for spray and pool fires developed in the U.S. were initially introduced and used. Subsequently, those codes were integrated and further improved as the ASSCOPS code that was used for accident analysis of Monju.

Table 4-2 Safety evaluation items for Monju

Event classification	Category	Number of events
Anticipated operational occurrence	Abnormal change in core reactivity or power distribution	3
	Abnormal change in core heat generation or removal	8
	Chemical reaction of sodium	1
Accident (design basis accident)	Accident leading to increased in-core reactivity	3
	Accident leading to decreased core cooling capability	8
	Accident associated with fuel handling	1
	Accident related to waste disposal systems	1
	Chemical reaction of sodium	4
	Accident related to the reactor cover gas system	1
Technically inconceivable event (Item 5 event)	Local fuel failure event	2
	PHTS large pipe break event	1
	Anticipated transient without scram (CDA)	2
Site evaluation	Major accident	2
	Hypothetical accident	1