10.1.4 Education and training systems[10-2)]

Since 1992, education and practical training for operators have been systematically carried out. The lessons learned from the Secondary Sodium Leak Accident were reflected in the update of education and training content referring LWRs in the Comprehensive Safety Check.

Content on the basic knowledge including sodium leaks, repetitive education, education for shift supervisors and assistant shift supervisors, etc. was improved. A summary of the content is listed in Table 10-3.

Furthermore, the following training was added after the Niigataken Chuetsu-oki Earthquake in 2007 and the 1F Accident:

- A complex event caused by two or more anticipated events, such as loss of off-site power, and earthquake and tsunami,
- A station blackout which occurs after the occurrence of earthquake, tsunami, and a complex event, and
- A severe accident, such as training of field operations during the station blackout.

10.1.5 Operating procedures[10-2)]

(1) Comprehensive Safety Check

a) Operating procedures under abnormal and faulted conditions

The system of operating procedures under abnormal and faulted conditions was revised by merging similar events and adding the postulated range of abnormal events.

The Monu operating procedures consist of: plant startup/shutdown procedures and equipment operating procedures that are used in normal operation; and operating procedures under abnormal and faulted conditions that are used to cope with accidents/faults. The overall structure of the operating procedures is shown in Fig. 10-5. Operating procedures under abnormal and faulted conditions revised in the Comprehensive Safety Check are shown in Table 10-4.

b) Symptom-based operating procedures

After the Comprehensive Safety Check, it was decided to introduce symptom-based operating procedures (for beyond-design-basis events). A preliminary study had already been launched earlier in 1989 in consideration of the trend in LWRs. Accident sequences were analyzed using the PRA, and the events to be covered were selected and classified. The measures to be taken were then extracted and investigated.

Based on this study, symptom-based operating procedures were implemented in 2004 under the title of “Emergency Operating Procedures II” following naming in LWRs.

Table 10-3 Improvement of education and training for operators

Level and type		Major contents
Apprentice operator	Education	Plant system learning course: Monju systems and equipment Added content: Basic knowledge other than the above
	Training	Beginner course: Normal startup and shutdown using simulator
Novice operator	Education	New content: Safety evaluation (overview of the safety analysis results described in the Application for Reactor Installation Permit)
	Training	Intermediate course: Response to abnormal conditions using simulator Changed content: Response to secondary sodium leaks (simulation to confirm white smoke in local areas, emergency draining operation, etc.)
Middle-class operator	Education	Rules to be complied with for operation
	Training	Advanced course: Response to abnormal conditions using simulator Added content: Emergency Operating Procedures II
Senior operator	Education	New content: Laws and regulations related to operation management
	Training	Assistant shift supervisor course: Response to abnormal conditions in Monju using simulator Added content: Directing response to abnormal conditions
Shift supervisor and assistant shift supervisor	Education	New content: Education for operation manager (matters to be complied with for operation management) New content: Shift supervisor seminar
	Training	New content: Operation supervisor course Added content: Repeated training

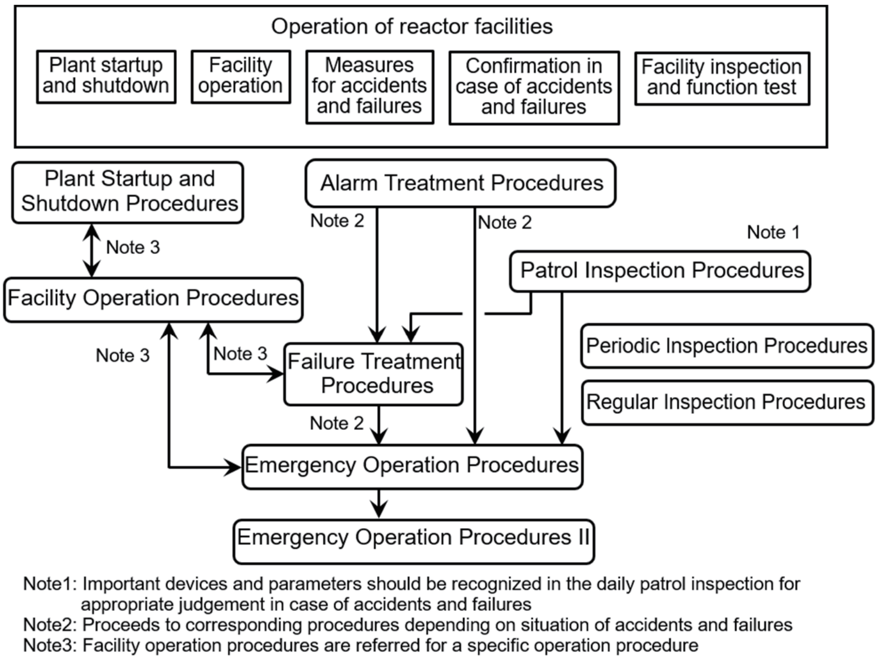


Fig.10-5 Structure of operating procedures

Table 10-4 Structure of operating procedures under abnormal and faulted conditions

[Emergency Operating Procedures] 1. Reactor trip/turbine trip 2. Loss of off-site power 3. Reactivity anomaly 4. Fuel failure 5. PHTS flow anomaly 6. SHTS flow anomaly 7. Primary coolant leakage 8. Secondary coolant leakage 9. IHX heat transfer tube leakage 10. SG tube rupture 11. EVST system sodium leakage 12. Primary argon gas leakage 13. Refueling and fuel handling accidents 14. Gaseous waste disposal system failure [Emergency Operating Procedures II] 1. Reactivity control 2. Core cooling 3. Maintaining reactor coolant level [Failure Treatment Procedures] 1. Primary sodium overflow system failure 2. Evaporator overflow stop valve fail-close 3. Secondary sodium purification system flow rate low 4. Superheater level control system failure	5. Auxiliary cooling system control system failure 6. Steam separator drain valve failure 7. Superheater bypass valve fail-open 8. One main feedwater pump trip 9. Feed-water regulation valve failure 10. Feed-water regulation valve pressure differential control failure 11. Feed-water heating system failure 12. Main steam pressure control system failure 13. Generator load rejection 14. Condenser tube leakage 15. Decrease in condenser vacuum 16. One circulating water pump failure 17. Reactor auxiliary component cooling system failure 18. Loss of control compressed air 19. Spent fuel pool cooling and purification system failure 20. Neutron instrumentation system failure 21. Loss of DC power supply 22. Loss of AC uninterruptible power supply 23. Loss of instrumentation power source 24. Loss of one emergency metal-clad switch gear system 25. Over-contamination of extra-high-tension switching station insulator 26. Fire 27. Remote reactor shutdown 28. Earthquake/tsunami 29. Intake anomaly

(2) Accident management guidelines and reflection to operating procedures

Accident management (AM) intended to further improve the safety of Monju was developed and "AM guidelines" were prepared for operation, including implementation system, operating procedures, and education programs. The AM guidelines were submitted in March 2008 to the regulatory authority under the title of "Report on AM development" similar to reports for LWRs in Japan. Review by the regulatory authority was suspended immediately before completion due to the 1F Accident; however, the contents of the AM guidelines were reflected to Emergency Operating Procedures II:

- Previously, response procedures for the loss of decay heat removal function (including the EVST cooling system and spent fuel pool cooling system) due to station blackout were described in core cooling as well as for the failure of all primary and secondary loops and auxiliary cooling system components. From the perspective of usability, response procedures for core cooling and station blackout were separated.
- Based on PRA evaluations, response procedures for the loss of decay heat removal function due to the failure of equipment that cools spent fuel, such as the EVST cooling system and spent fuel cooling system circulation pump, were added.

A general flow chart of Emergency Operating Procedures II is shown in Fig. 10-6.

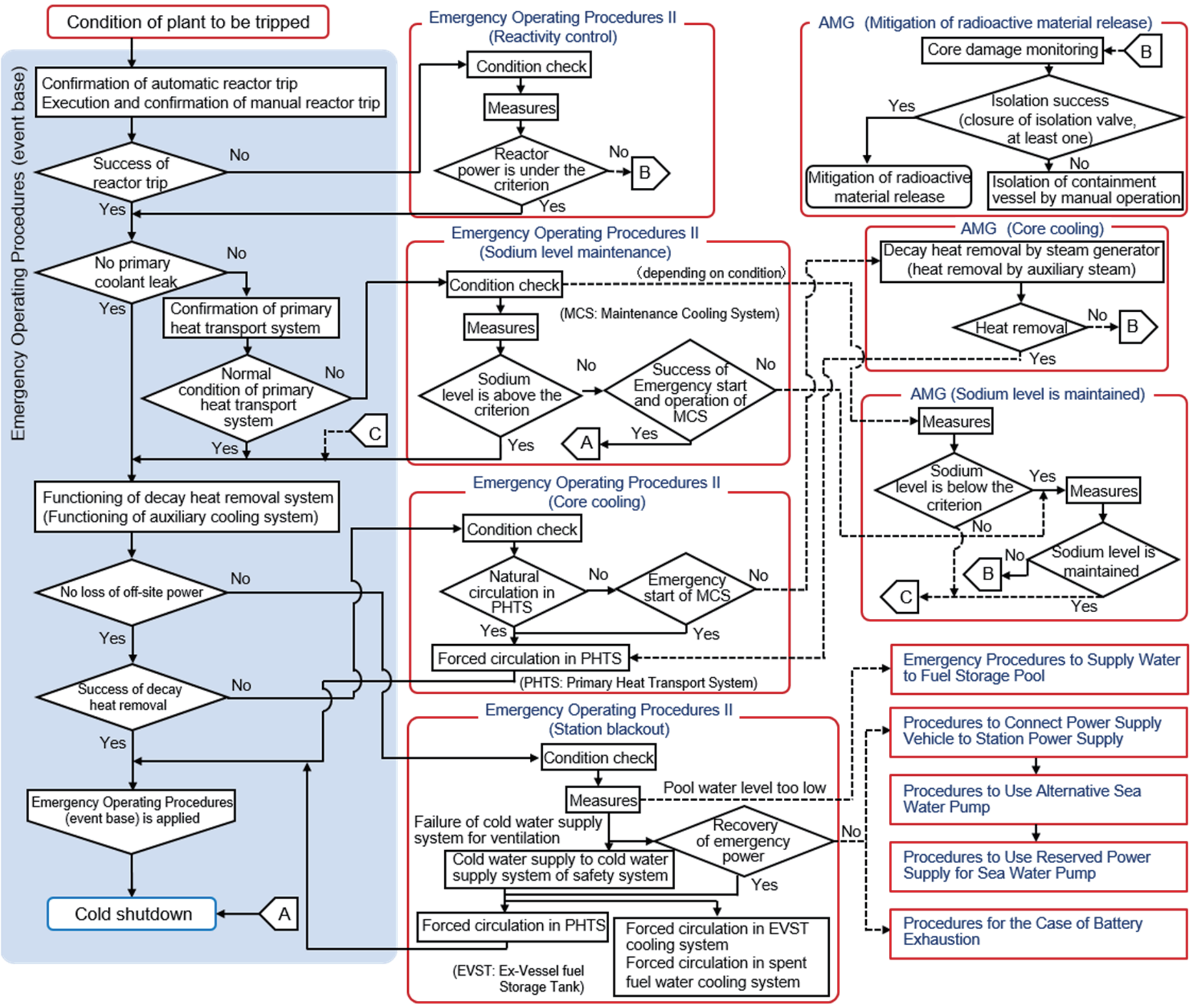


Fig.10-6 A general flow of Emergency Operating Procedures II

10.2 Maintenance

10.2.1 Development and issues of maintenance management technology

(1) Maintenance management before introduction of the maintenance program

Monju maintenance activities started after SKS. Since this period was a stage for preparation of SST, the inspection of required equipment to ensure successful performance of SST was carried out taking into account the test schedule.

Due to the long-term plant shutdown period after the Secondary Sodium Leak Accident, maintenance activities were limited to equipment required to ensure the safety of reactor facilities during shutdown (i.e., systems necessary for decay heat removal, prevention of radioactive material dispersion, etc.). The components that had passed the pre-service inspection and were kept under a long-term shutdown underwent inspection by the regulatory authority to confirm that the component conditions were maintained.

The licensing procedure for the modification work after the Secondary Sodium Leak Accident was completed, and agreement with the local governments was reached in 2005 in advance of site work. The maintenance activities then focused on confirming the integrity of the equipment related to the modification work and plant restart. In this period, malfunctions and defects, such as false alarm from the contact-type sodium leak detectors and corrosion pitting of the annulus ventilation duct were found, and thus the confirmation of system integrity for plant restart needed to be conducted in parallel with the trouble management.

(2) Monju maintenance program

For commercial nuclear power reactors, the regulatory ordinance concerning maintenance management was amended in August 2008, and the inspection system was revised accordingly to improve maintenance activities at each plant and to further strengthen safety practice in maintenance activities. At the same time, the ordinance for the power reactors in the R&D stage was also amended, and thereby an inspection system similar to that for commercial reactors was applied to Monju, which demanded that a maintenance program be put in place.

Preparation for the Monju maintenance program was started in August 2008, and the main part of program was developed in two months from November and put into practice in January 2009. Since there had been no experience in operation and maintenance through an in-service operation cycle at Monju, a program based on preventive maintenance was developed using the available information from Joyo, LWRs, etc. with reference to the Code for Maintenance at Nuclear Power Plants and its Guide (JEAC4209/JEAG4210). Because deterioration of sodium components can be largely ignored in sodium environments (Photo 10-3), function check and visual inspection were basically performed without open inspection or overhaul.

Based on the maintenance program, the pre-service maintenance plan was developed for each of the three pre-service maintenance cycles: the Core Performance Confirmation Tests at zero power, and the PKS at 40% and 100% power.

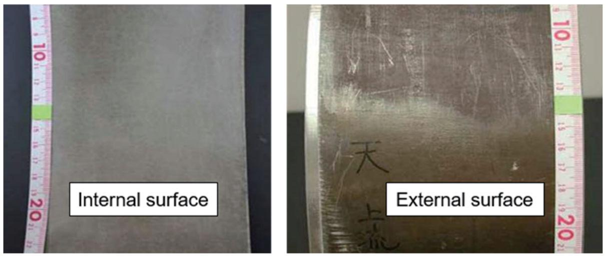


Photo10-3 Appearance of pipe used in the secondary heat transport system (C)

(3) Maintenance management based on the maintenance program

In the first pre-service maintenance cycle (from January 2009 to July 2010), maintenance activities were completed close to on schedule. The activities included system check required for the Core Performance Confirmation Tests and integrity confirmation of the water-steam system equipment.

In May 2009, the regulatory authority issued the “Concept of Safety Confirmation for the Restart of Commissioning” in consideration of the countermeasures for the troubles described in (1) and the progress of quality management, to conclude that the system for ensuring safety was sufficient for the restart of commissioning. Based on this, JAEA resumed SST in May 2010 and started the first stage, the Core Performance Confirmation Tests.

The plan for the second pre-service maintenance cycle (from July 2010) was prepared by reflecting the results of the evaluation of the effectiveness of maintenance for the first pre-service maintenance cycle.

In the second pre-service maintenance cycle, the primary and secondary cooling system equipment were inspected according to the inspection plan to confirm the integrity of equipment required for the PKS at 40% power. The water-steam system equipment, which had been kept in a long-term storage state, also had to be inspected to confirm integrity, but this time according to a special maintenance plan, which is required specifically for equipment under a long-term shutdown state. These inspections and integrity check were confirmed by the regulatory authority staff in the pre-service inspection or On-Site Inspections as needed.

The second maintenance cycle, however, was never completed due to several reasons. First, the overall plant schedule was extended several times due to failures of the equipment important to safety, including the drop of the IVTM in August 2010 and the cracking in a cylinder liner of one of the diesel generators in December 2010. These resulted in a prolonged shutdown period.

Second, the 1F Accident in 2011 significantly affected Monju as well. A higher priority was given to urgent safety measures (diversification of power supply, etc.) similar to LWRs, and the preparatory work for the plant restart was suspended. Consequently, the plant shutdown state was further prolonged and the end date of the second maintenance cycle became uncertain. Under these circumstances, inadequate maintenance management was revealed.

(4) Lessons learned from inadequate maintenance management

When inspection cannot be performed by the time limit defined in the maintenance plan because of a change in plant schedule, nonconformance management is required under the quality management system. However, when such a situation actually happened, the procedure was not appropriately handled and this resulted in many components remaining uninspected at their time limits. The NRA judged this inadequate management to be a violation of the Operational Safety Program and issued an Order for Safety Measures to JAEA, which required the immediate inspection of the uninspected components and revision of the maintenance plan.

In response to this Order, JAEA conducted the analysis of direct and root causes of the inadequate management and compiled the measures for prevention of recurrence. However, even after reporting the results to the NRA, similar violations were repeatedly pointed out at the quarterly Operational Safety Inspections. JAEA attempted to revise the maintenance plan and improve management organization. Despite these efforts, however, the problem of inadequate maintenance management could not be resolved completely.

JAEA attempted to overcome the difficult situation by introducing an all-Japan framework in collaboration with utilities and manufacturers. However, in November 2015, the NRA finally made a recommendation to the MEXT, a government agency supervising JAEA, calling for a complete change in the management body of Monju or revision of the policy on the Monju project.

The Monju maintenance program was first introduced based on regulatory requirements to achieve a level of maintenance management equivalent to that for in-service LWR plants, but the attempt was unsuccessful due to several reasons. First, the maintenance program was introduced too mechanically, conservatively and hastily without sufficient time and effort for preparation. Second, Monju, still in the commissioning stage, had never gone through a cycle of plant operation and periodic maintenance, with staff clearly lacking experience in implementing and reviewing the maintenance program. In addition, an overall management system had not matured in the pre-service stage.

10.2.2 Study on effective maintenance management

Toward the design of future FRs, effective maintenance management was discussed based on the experience in maintenance management described in the previous section (Fig. 10-7).

There is no difference in the basic concept of maintenance between FRs and LWRs, and the same Code and Guide for Maintenance (JEAC4209/JEAG4210) are applied. One of the most important roles of Monju is to acquire operation and maintenance data useful for future FRs. For FR-specific components, a large margin is assumed in the design to compensate for insufficient operation and maintenance experience. Accordingly, the operation and maintenance data from the actual plant are highly valuable for design validation.

Besides, as a sodium-cooled reactor, it is deemed more reasonable from the viewpoint of

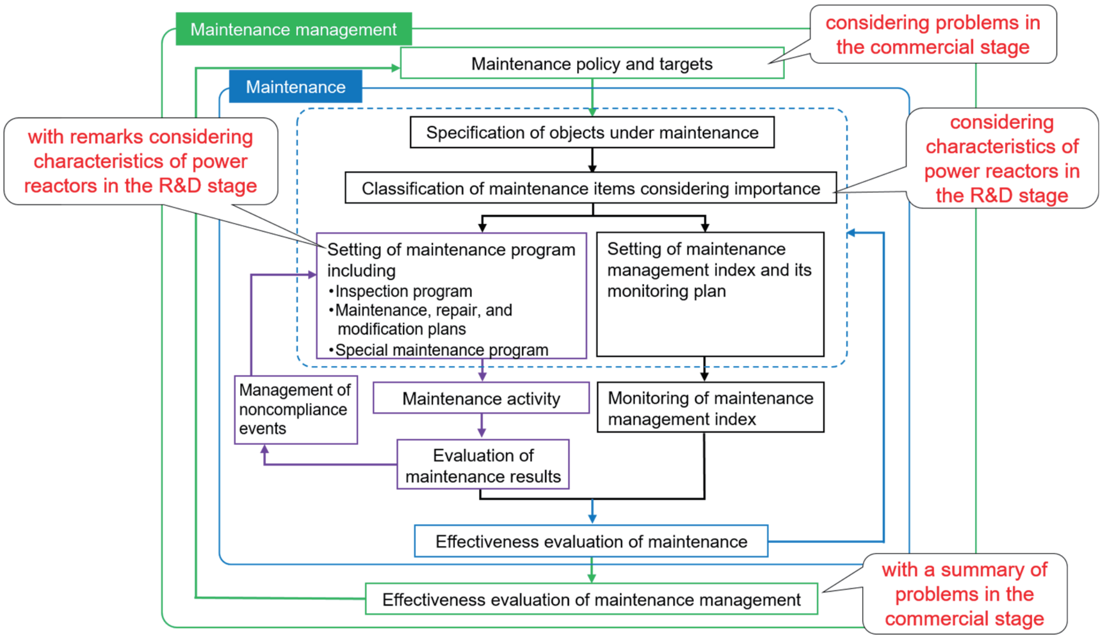


Fig.10-7 Structure of maintenance management

maintenance to rely mainly on continuous sodium leak monitoring of sodium-containing components than it is to conduct overhaul or open inspections.

The discussion on maintenance management in consideration of these features is documented in Ref. 10-5 and can be reflected in maintenance management and activities of future FRs.

To develop a maintenance plan, it is necessary to assume the deterioration mechanisms of the equipment to be inspected and optimize the inspection items. The deterioration mechanisms, which had not been considered in detail in developing the Monju maintenance plan, were examined based on the related R&D, design and construction information, and the findings from earlier reactors, including Joyo. The resultant “summary sheet of deterioration mechanisms” is valuable in the development of a maintenance program in future FRs.

10.2.3 Shortening periodic inspection schedule

In the Monju design, the inspection period was set based on the following pattern: 4 (5) month operation, 1 month refueling, 4 (5) month operation, 1 month refueling, and 2 month inspection for the low-burnup core. (The figures in parentheses are for the high-burnup core.) Namely, the target period of the refueling and periodic inspection processes was set at 3 months.

Monju was kept in a cold shutdown state for an extended period of time. During this time, a periodic inspection took about 6 months without refueling or the containment vessel leak rate test (CV-LRT). This means a longer inspection period is necessary in a normal operation stage. The environments specific to FRs are another factor requiring an additional inspection period. Namely, the sodium temperature needs to be decreased for sodium component inspection and increased for function tests after inspection. In the PHTS rooms, in addition to the sodium draining, the nitrogen atmosphere needs to be replaced with air. It is also necessary to transfer the sodium in the loop to be inspected to a tank of other loops and solidify the remaining sodium, which would increase the time required to prepare for inspection.

Future FRs will require significant rationalization of the periodic inspection process, and major possible solutions are as follows:

- Moderation or elimination of constraints, such as sodium solidification and temperature setting required for inspection,
- Extension of effective life of dew-point meter used in CV-LRT,
- Procurement of spare parts for components with a long inspection interval, and
- Shortening of time for preheating and decreasing temperature of large components.

10.2.4 Maintenance and repair experiences

A large number of events with respect to maintenance and repair have been experienced since the completion of component installation in May 1991.

Table 10-5 lists information on the major faulty events important or useful for future FR design selected from the data accumulated as of the end of March 2017, excluding the accidents and failures described in Chapter 11. Figure 10-8 shows the numbers of maintenance records collected and their breakdown by equipment.

The representative and important cases, not included in Table 10-5 are outlined below:

(1) Load increase of fine CRDM

During SKS and SST, an alarm indicating an abnormal load to fine CRDM unit 2 was set off three times. A disassembling investigation revealed that sodium compounds had become stuck in the gap between shield and drive shaft in the upper guide tube (Fig. 10-9). A possible cause of the increased load was thought to be the dirt (a film of impurities floating on the surface of the sodium) left during the initial sodium charge in combination with the configuration in which sodium becomes stagnant. As a countermeasure, flow holes were added to the upper guide tube at the surface wetted by sodium to facilitate sodium flowing. This event might occur not only during the initial sodium charge but also during sodium charge after work involving the opening of a sodium boundary. To reduce these risks, it is important to minimize the mixing of impurities (air) into sodium systems.

Table 10-5 List of major faults occurred in Monju

No.	Month/ year	Name of event	Overview
1	Feb. 1995	Poor dehumidification in fuel cleaning	[Event] Malfunction of ex-vessel fuel transfer machine gripper in sodium cleaning [Cause] Reaction products between sodium and remaining moisture due to insufficient drying after cleaning prevented the gripper's sliding part from moving. (estimation) [Countermeasure] Heaters were installed along piping surrounding the fuel cleaning tank where moisture is apt to remain, and a vacuum dryer process was added to the cleaning procedure.
2	Feb. 1995	Vibration and noise of flash tank pressure control valve	[Event] Vibration and noise occurred around the flash tank control valve of startup bypass system during plant startup/shutdown process. [Cause] The generation of shock wave during slowdown of a supersonic region that appeared when the secondary-side pressure of the concerned valve was reduced to the critical pressure or less. (estimation) [Countermeasure] The concerned valve was replaced with a low noise valve and piping surrounding the valve was modified (increase in pipe diameter, modification to a parallel system, installation of depressurization mechanism).
3	June 1995	Lack of capacity of steam separator drain valve	[Event] During a test of the water-steam system startup bypass system, the opening of the steam separator drain valve exceeded 90%, and then the test was continued after changing certain procedures. [Cause] The valve vendor's CV value calculation chart was not applicable to water-steam two-phase flow. (estimation) [Countermeasure] The valve stroke was changed from 40 mm to 50 mm.
4	Aug. 1996	Oil entering into the secondary main circulation pump A pony motor	[Event] The casing temperature of the secondary main circulation pump A pony motor was increased during operation. [Cause] Oil entered into the motor due to the screw pumping effect of grinder mark formed during manufacture of the motor shaft. [Countermeasure] Replacement of motor shaft and modification, such as the change in structure of the motor shaft and oil seal holder (reduction in the gap)
5	Since 1998	Condensation in buildings	[Event] Condensation occurred in many rooms from June to September every year. [Cause] Highly humid air entered into at a low atmosphere temperature (component surface temperature). [Countermeasure] Installation of spot coolers and drip-proof covers, and work to prevent air from entering the rooms
6	March 2001	Precipitation of caustic soda from liquid waste disposal facility caustic soda tank	[Event] Precipitation of caustic soda from liquid waste disposal facility caustic soda tank (radiation uncontrolled area) [Cause] Stress corrosion cracking from the outside surface due to chlorides [Countermeasure] Replacement with a tank consisting of an austenitic stainless steel with low carbon content (SUS304→SUS304L). (Welding areas on the outer surface were painted.)
7	Sept. 2007	Clogging of reactor component cooling seawater system strainer due to winter's rough weather	[Event] The reactor component cooling seawater system strainer was often clogged (increase in pressure differential) in winter. [Cause] Intake of seawater accompanied by fishing net and fallen leaves due to rough winter weather, specific to the Japan Sea [Countermeasure] Installation of dust net over the intake

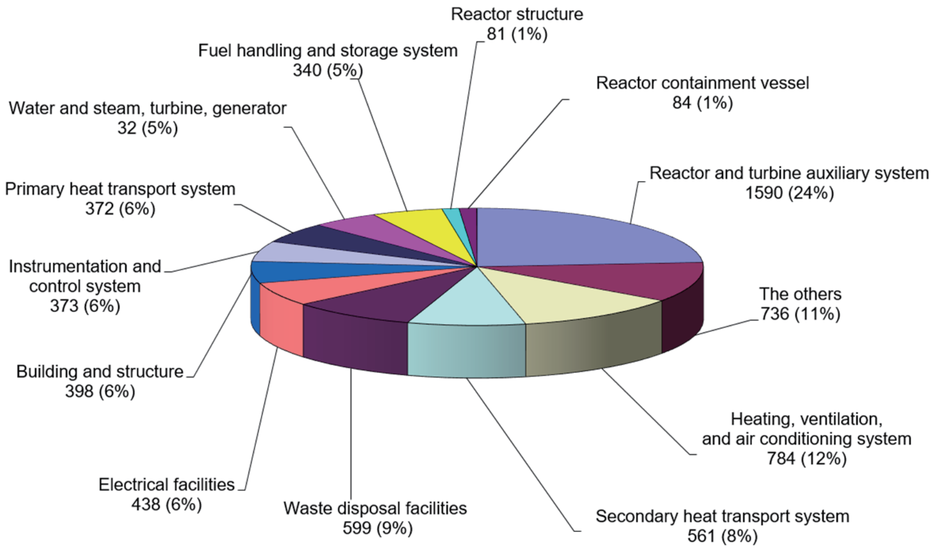


Fig.10-8 Number of maintenance records issued in Monju facilities

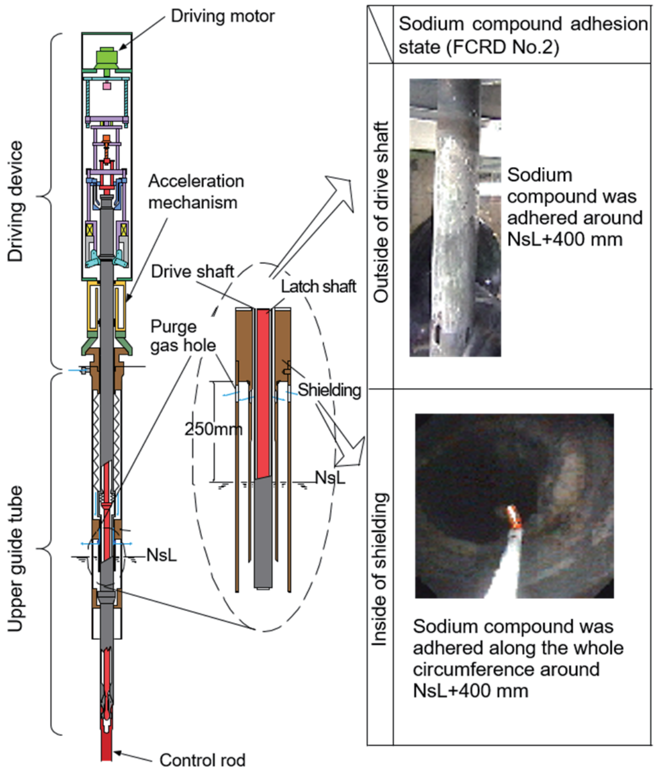


Fig.10-9 Sodium compound adhesion state in FCRD (No.2)

(2) Damage to superheater pressure relief plate (A)

In March 1998, during planned replacement of the pressure relief plate, a small amount of sodium was found adhered to the secondary side (in nitrogen atmosphere) of superheater pressure relief plate (A) (Fig. 10-10). A possible cause was stress corrosion cracking caused by melted caustic soda (sodium hydroxide) that was generated by a reaction of sodium with moisture absorbed in the process of manufacturing the pressure relief plate. A test confirmed that stress corrosion cracking would not occur in an environment with a large amount of sodium vapor where caustic soda changes to sodium oxide by reaction with sodium. This event is regarded as a rare case in which caustic soda did not react with sodium in a narrow space between the vacuum support and the plate where sodium vapor was not supplied. As a countermeasure, the method of polishing the vacuum support was improved.

(3) Increased pressure loss in the primary argon system

Since the reactor startup in October 1995, the pressure differential between RV vapor trap outlet and compressor surge tank increased with increased sodium temperatures. The cause was a blockage formed due to the deposition of sodium vapor to the valves downstream from the vapor trap (Fig. 10-11). As countermeasures, two sintered filter lines and one HEPA filter line were added to the vapor trap outlet. The effectiveness of the countermeasure was not confirmed because the plant has not been restarted.

A two-step process consisting of mist capture by a mist trap and condensation capture using a vapor trap was adopted for the RV vapor trap to effectively remove high-concentration sodium vapor in the cover gas. The performance of the vapor trap was not necessarily satisfactory. The design of vapor traps in future FRs therefore needs to be improved based on the experience of Monju.

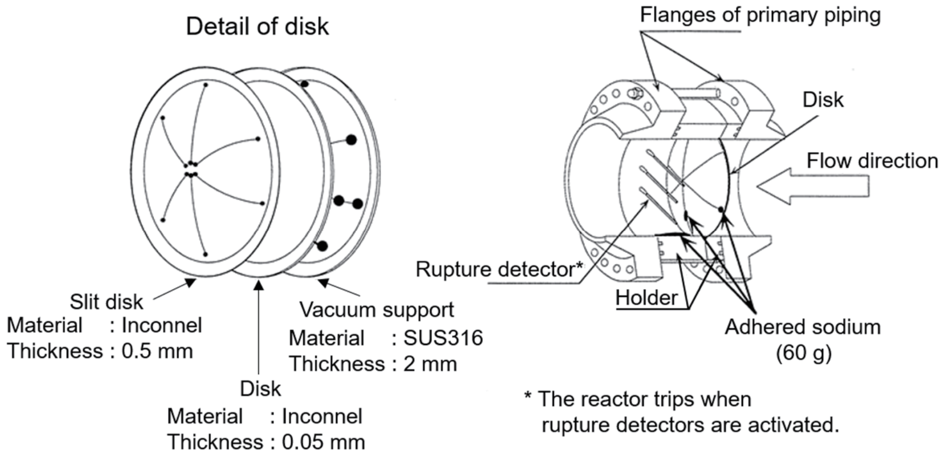


Fig.10-10 Sodium adhesion state around the rupture disk

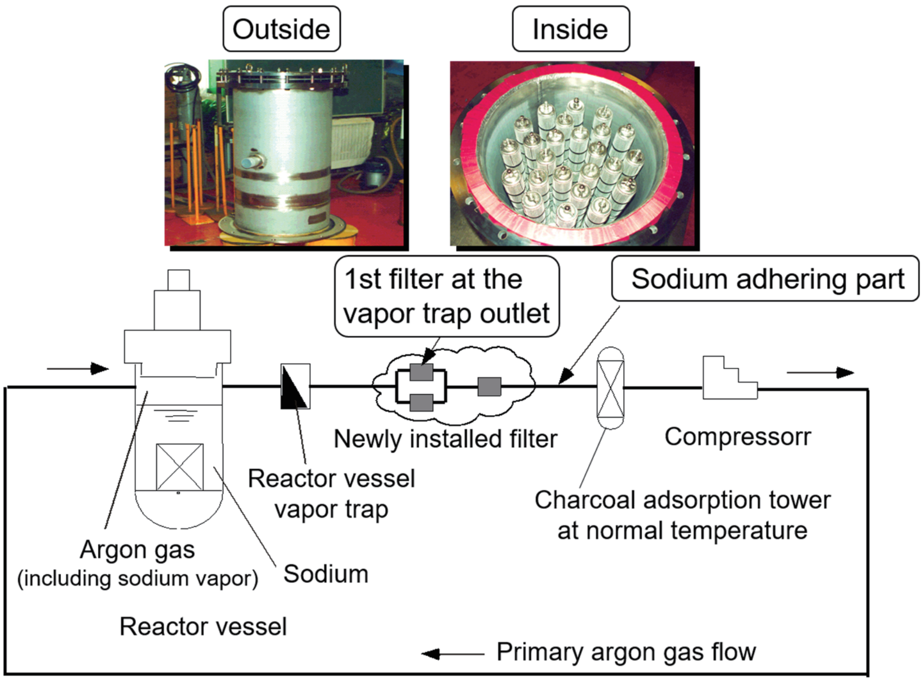


Fig.10-11 Pressure loss increase in primary argon gas system

(4) Behavior of SID during CV-LRT

In August 2008, the CV-LRT was performed during the PKS after the plant modification work following the Secondary Sodium Leak Accident. During the test, the pressure in the CV could not be increased to the specified level. The reason was that the electric current to the sampling pump of the sodium leak detector would reach the thermal relay set value during the pressure increasing process. It turned out that the specific conditions of CV-LRT were not considered in the design of the sodium leak detection system. As a countermeasure, the sampling pump motor was replaced with one with higher efficiency.

(5) Tests of an alternative dew-point meter

A lithium chloride dew-point meter is used for the CV-LRT in Monju. A problem with the meter is that the validity of the lithium chloride solution applied to the humidity sensing section is limited to from 3 to 6 months (based on vendor's recommendation), depending on the service environment. For this reason, in Monju as well as PWR plants, the period of validity is conservatively set at 3 months. In addition, access to the PHTS rooms to re-apply the lithium chloride solution is allowed only when sodium is drained and the nitrogen atmosphere is replaced by air.

A new capacitive dew-point meter was selected as an alternative that did not employ lithium chloride solution, and it was installed inside the Monju CV. Two tests were conducted: verification under the CV-LRT condition and 2-year long-term verification. Test results confirmed the absence of significant difference from the lithium chloride dew-point meter under the CV-LRT condition. The 2-year long-term verification test satisfied the instrument accuracy required by the Code for Reactor Containment Vessel Leak Rate Test (JEAC4203-2008). It was concluded that the capacitive dew-point meter can be used.

10.2.5 Repair technology of sodium components

Much experience with sodium component repair and sodium handling has been accumulated through measures and recovery work for the Secondary Sodium Leak Accident, the drop of the IVTM, etc. The findings in the modification work on the Secondary Sodium Leak Accident and the IVTM are described below:

(1) Modification work after the Secondary Sodium Leak Accident

a) Method for repair of sodium components

Air mixing into sodium systems must be prevented when repairing sodium components. Two methods were examined to determine an appropriate method for preventing air mixing: one used in Phenix using a special rig and another used in Joyo using a plastic bag. In the modification work after the Secondary Sodium Leak Accident, a method employing plastic bags was adopted when opening the sodium boundary, while in cases where the opening could be limited to a small area, a seal method not utilizing a plastic bag was adopted. Schematic drawings of the plastic bag and seal methods are shown in Fig. 10-12.

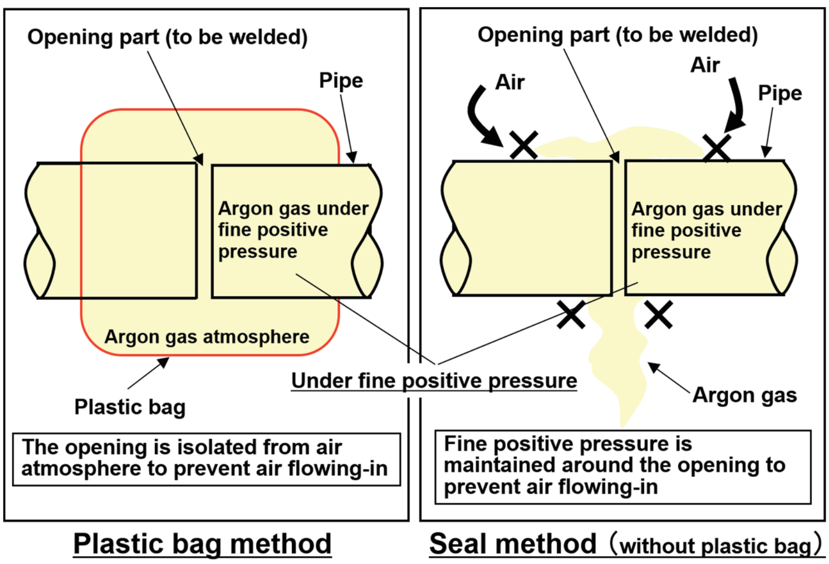


Fig.10-12 Images of plastic bag method and seal method

Before applying these methods, the procedure and effectiveness were confirmed through the validation tests, and the procedure and weldability were confirmed for the weld integrity of the piping in which sodium is deposited. Photo 10-4 shows the appearances of sodium piping modification work.

b) Control during the work

b-1) Oxygen concentration control utilizing a plastic bag

When using a plastic bag, each work was performed after replacing the internal atmosphere with argon gas and reducing the oxygen concentration to less than 2% (actually, less than 0.5%). During the gas replacement, the lids of containers in the plastic bag were opened not to leave stagnant air.

The moisture concentration in the plastic bag was also controlled because generation of caustic soda by a reaction of sodium and moisture mixed into a system leads to corrosion of structural material.

In addition to the above control, the purity (hydrogen concentration, etc.) of the cover gas in a system was monitored by the gas chromatograph.

b-2) Cover gas (argon gas) pressure control

The cover gas pressure in the SHTS is normally controlled at 98 kPa ±10% (gage pressure). However, since work with a plastic bag and welding operation could not be performed at this pressure, a fine positive pressure control unit dedicated to the modification work was employed. During the work period, pipe cutting and welding were performed adjusting the set value of the pressure control unit to keep the pressure at cutting zones to 100 Pa and the pressure at welding zones to 20 Pa in consideration of the specific gravity of argon gas relative to air according to the elevation of each work area.

b-3) Welding operation control

Since a small amount of sodium is deposited to or remains in the existing sodium piping, the remaining sodium may melt due to the heat generated by welding. Accordingly, welding was performed while monitoring the temperature measured by a thermometer temporarily attached near the piping welding zone to ensure that it does not exceed the upper limit (70°C). In addition, since it is necessary to maintain the pressure around a welding zone at a slightly positive level and the pressure is affected by atmospheric pressure variation due to weather changes, the working schedule was also properly managed.

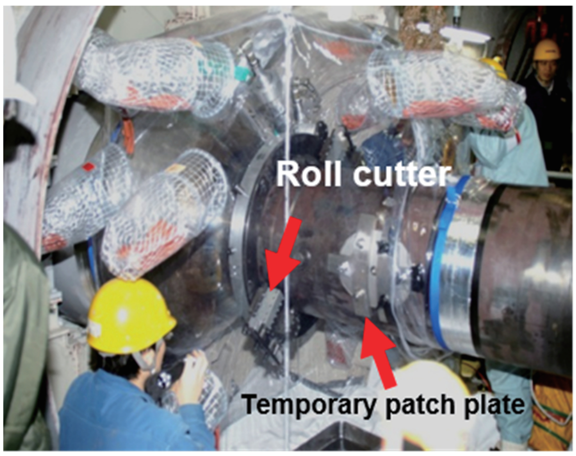
c) Amount of oxygen mixed in systems

The amount of oxygen mixed in the sodium systems was estimated to be a maximum of 1.6 g based on PL meter measurements before and after the work. The main possible oxygen sources are the residual air in plastic bags and impurities adhering to the inner surface of newly installed pipes and valves.

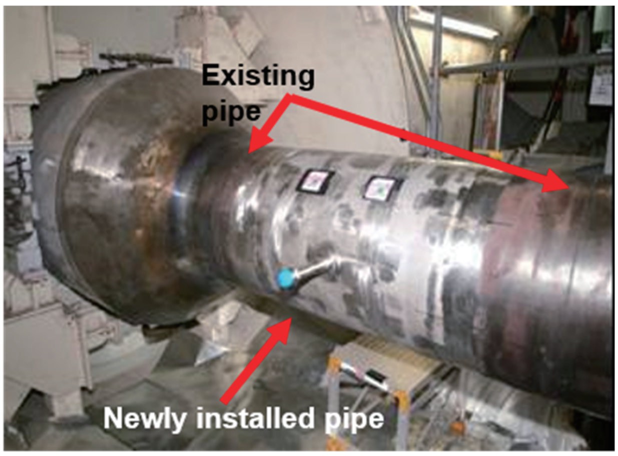
d) Amount of sodium deposited to the removed pipe

Photo 10-5 shows a view of sodium adhering to the SHTS pipe wall.

The sodium remaining in the pipes removed by the modification work was cleaned. Based on the weight difference before and after cleaning the sodium, the total amount of sodium de-



Cutting existing pipe



Installation of improved thermometer

Photo10-4 Modification work in the secondary cooling system posited to piping of the three loops was estimated to be 82 kg. The estimated amount of sodium deposited to loop C piping was larger than that to the other loop piping. The cause is likely a shorter preheating time after sodium draining because the preheater was turned off immediately after draining.

This means that the amount of sodium remaining in systems depends largely on the preheating time after sodium draining, and that the amount of remaining sodium increases with shorter preheating time. In work where the sodium boundary is opened, it is desirable to set a longer preheating time than usual after sodium draining. There are R&D results reporting that the amount of sodium remaining can be reduced by draining sodium at high temperatures, and that the amount at 400°C would be reduced to 1/7 of that at 200°C.

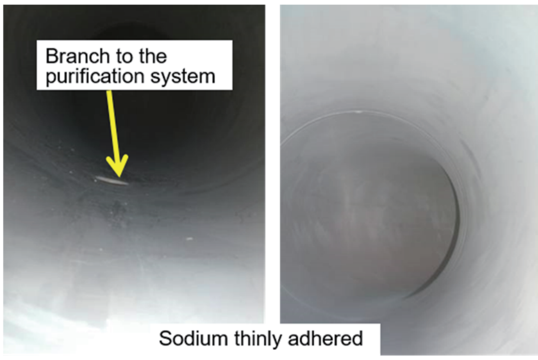


Photo10-5 Sodium adhesion state

(2) IVTM withdrawal and restoration work

This work is characterized by the fact that a large sodium component is handled in an activated environment, though the actual radiation dose is extremely low. Photo 10-6 shows the procedure of the work to withdraw the failed IVTM. The achievements obtained through this work are:

- Establishment of control technology of fine positive pressure and cover gas pressure,
- Streamlining of work procedures (including shortened work hours), and
- Establishment of argon gas replacement technology with large plastic bags.

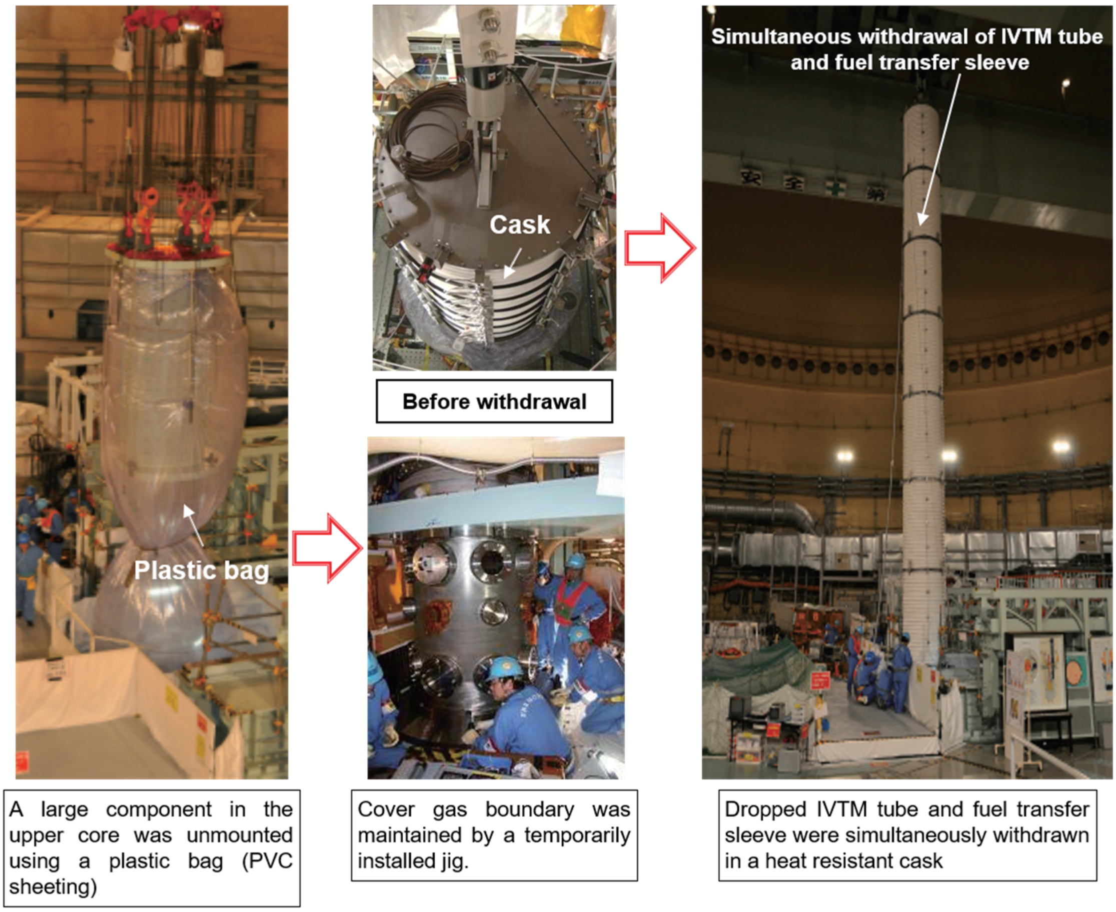


Photo10-6 Recovery work of dropped IVTM

These technologies to repair sodium components are valuable and generally applicable to maintenance, repair, and replacement in future FRs. The experience in Monju was also used in Joyo for withdrawal of the accidentally damaged test rig from the RV and the replacement of the upper core structure.

— References —

10-1) Miyakawa, A. et al., The Prototype Fast Breeder Reactor Monju System Startup Tests Report -- Summary Report of the System Startup Tests <from Criticality Test to Power Up Test (40% Power)>, JNC-TN2410 2005-002, 2005, 278p. (in Japanese).

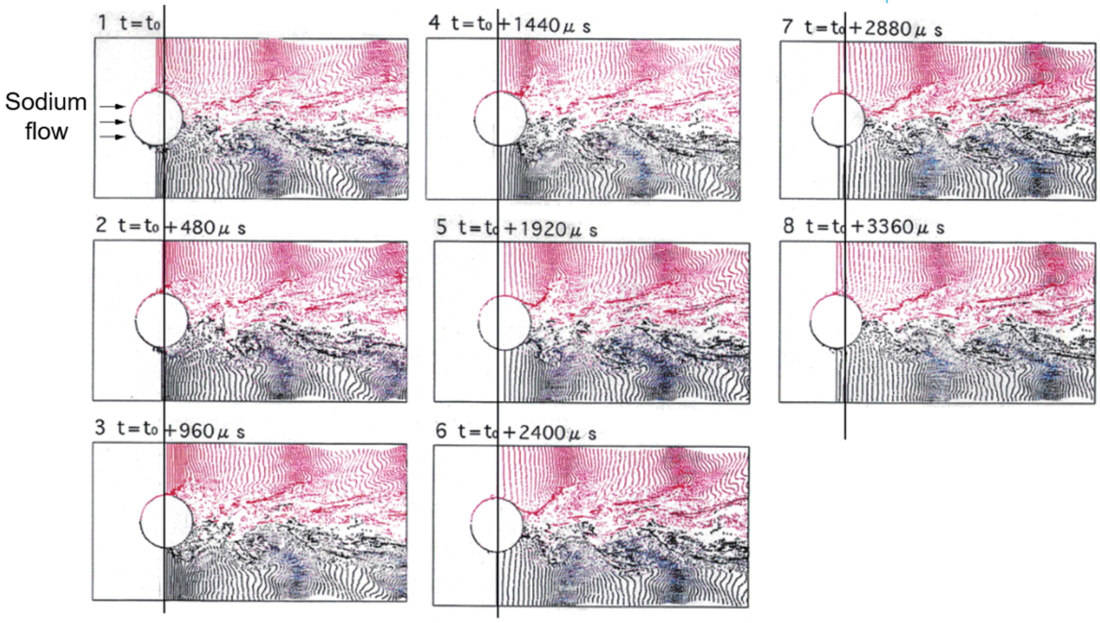
10-2) Japan Atomic Energy Agency, Measures and reports on the Comprehensive Safety Check for the Prototype Fast Breeder Reactor Monju (Amendment of the 5th report), 2010 (in Japanese).

10-3) Ohuchi, K. et al., Design, Manufacturing and Installation of Monju Simulator MARS, PNC Technical Review No. 77, PNC-TN1340 91-001, 1991, pp.88-91 (in Japanese).

10-4) Koyagoshi, N. et al., Transition of Monju Simulator Training owing to Monju Accident and Upgrade of Monju Advanced Reactor Simulator (MARS), JNC-TN4410 2002-001, 2002, 67p. (in Japanese).

10-5) Takaya, S., Maintenance Management of Nuclear Power Reactors at the Stage of Research and Development, JAEA-Research 2016-006, 2016, 66p. (in Japanese).

11. Accidents and Failures



<Hydraulic oscillation analysis of thermocouple sheath>

- **Accidents and failures experienced in Monju, including those specific to sodium-cooled FBRs, contributed to the improvement of plant safety.**
- **The Secondary Sodium Leak Accident in 1995 had a significant impact on the Monju project. Many technological achievements were obtained, especially related to the sodium-induced corrosion mechanisms and hydraulic oscillation of the structure in a flow.**
- **Enhanced public understanding was especially important in promoting the Monju project with respect to its significance and roles as well as safety.**
- **Experience in investigating the causes of the accidents and failures and establishing countermeasures to prevent their recurrence is extremely valuable in design, manufacturing, and construction of future FBRs.**

11.1 Accidents and failures in Monju

Accidents and failures important to safety (hereinafter, "accidents and failures" is used as a general term for accidents, troubles, failures, malfunctions, defects, anomalies, etc.) that occurred in Monju include "accidents and failures under the Ordinances" and "anomalies under the Safety Agreement". The former are specified by the Ordinance on the Installation and Operation of Reactors for the Purpose of Power Generation at the Research and Development Stage applicable to Monju. Events shall be reported to the NRA immediately after occurrence. Subsequently, more detailed situations and countermeasures shall be reported within 10 days after occurrence. The latter are specified in the Agreement for Ensuring Safety in the Surrounding Environment of the Prototype Fast Breeder Reactor Monju (Safety Agreement), and the local governments shall be notified immediately after the occurrence of the relevant event.

In addition, the occurrence of "minor troubles" not specified as accidents and failures important to safety are also reported voluntarily and the information is made available to the public.

(1) Accidents and failures important to safety

Table 11-1 lists the date, name, and overview of the accidents and failures that occurred in Monju from FY 1991 on, when component installation was completed. The 17 events listed in the table were reported under the Safety Agreement (including one anomaly in 1991 before conclusion of the Agreement), and 9 out of the 17 were also reported under the Ordinances.

(2) Overview and causes of accidents and failures

The above 17 events are classified into 14 failures and 3 personal injuries. The causes for the 14 failures are classified into 8 errors in the design phase (hereinafter, "error" is used as a general term for design faults and failures due to errors, lack of knowledge, experience, and consideration) and 6 errors in the operation and maintenance phases. The errors in the design phase consist of 3 evaluation errors due to insufficient experience, 1 insufficient use of new knowledge, 1 insufficient examination of screw rotation prevention, and 3 insufficient consideration of natural phenomena such as snowstorm or lightning. The errors in the operation and maintenance phases consist of 4 work/operation errors and 2 errors due to insufficient recognition. The Secondary Sodium Leak Accident was evaluated as level 1 (anomaly) on the International Nuclear and Radiological Event Scale (INES: 8-level evaluation from 0 (below scale) to 7 (major accident)).

It should be noted, for example, that although thermocouple sheath breakage had occurred also in LWRs, the Accident in Monju raised significant social concern as the first sodium leak in a new-type reactor.

(3) Occurrence frequency of accidents and failures

Figure 11-1 shows the trends in the number of accidents and failures that were reported to NRA from FY 1992 to 2015. The annual number of reports per unit for commercial power reactors is 0.1-0.8/year, while that for Monju is 0.4/year, showing no significant difference in the frequency of occurrence.

Concerning the anomalies reported under the Safety Agreement, according to the Annual Report of Power Plant Operation and Construction[11-1] of Fukui Prefecture, the number of reports since FY 1992 is 16 for Monju, while the average of 15 nuclear power reactors is 27.

From the above statistics, it is evident that the numbers of "accidents and failures" and "anomalies" for Monju are by no means greater than those for LWRs. Nevertheless, the accidents and failures in Monju were taken seriously by mass media because of the strong social concern about Monju, a new-type reactor.

(4) Minor troubles (incidents)

The number of incidents in each FY is shown in Fig. 11-2. The total number of incidents from FY 1996 to 2016 is 111, and the annual average is 5.3/year.

In particular, alarms due to minor failure of sodium leak detector were set off repeatedly, 4 times in FY 2007, 5 times in FY 2008, and 3 times in FY 2009. All of these were false alarms, i.e., no actual sodium leak had occurred. The causes for these false alarms were diverse and included the effects of variation of ambient temperature, volatile materials from paint, electrical fluctuation due to lightning, etc., and human error. Such information has become valuable when using a high-sensitivity sodium leak detector in an actual plant.

Table 11-1 Accidents and failures reported under the Safety Agreement (1/2)

No.	Month /year	Event name	Overview
1	June 1991	Thermal displacement of SHTS piping	[Event] During preliminary temperature increase of the SHTS, the pipe adjacent to the IHX of loop C was abnormally displaced at 120°C. [Cause] Underestimation of the spring constant of the bellows at CV penetration in the design [Countermeasure] Reduction of the obstruction of thermal displacement, including the change from two to one-layer configuration of the bellows (i.e., to softer bellows)
2	Feb. 1995	Pressure drop of water-steam system flash tank	[Event] In the process of increasing the reactor power from about 2% to 10%, the outlet pressure of flash tank located in the water-steam system startup bypass system abnormally dropped. [Cause] Increase in flow resistance due to steam entrainment (underestimation of pressure drop in the design) [Countermeasure] Improvement of structure designed to prevent steam entrainment in the flash tank and drain piping to the deaerator
3	May 1995*	Automatic reactor shutdown due to change in feedwater flow rate	[Event] During control characteristic testing of the feedwater control valve of water-steam system bypass system at 20% power, unexpected switching of the valve control mode from manual to automatic caused an automatic trip of the secondary main circulation pump, and then the reactor tripped. [Cause] Erroneous setting of automatic control constants in the design [Countermeasure] Modification of the control constants based on the results of analysis using measured valve characteristics
4	Dec. 1995*	Secondary sodium leak accident during the 40% power test	[Event] Sodium leaked through thermocouple sheath of the IHX outlet piping into the SHTS piping room during operation at an electric power of 40%. [Cause] Design deficiency of thermocouple sheath and lack of new knowledge on hydraulic oscillation [Countermeasure] Replacement of thermocouples with an improved design. Modification of cooling systems, including the shortening of sodium drain time, and various measures to prevent recurrence based on the Comprehensive Safety Check
5	Jan. 1997	Malfunction of power-receiving equipment (switchgear)	[Event] Power-receiving became unavailable due to malfunction of the protective relay, which was caused by icing on insulators due to blizzard. [Cause] Inconsistent design of interface between transmission line breaker and protective relay unit [Countermeasure] Prevention of icing and salt damage by installation of wind shield wall for extra-high-tension switching station and improvement in the method of cleaning insulators
6	Jan. 1997	Activation of CV isolation caused by false alarm	[Event] The alarm “Containment vessel above-floor atmosphere radioactivity high” was set off upon lightning strike. [Cause] Lightning-induced voltage on receiving switchboard in the monitoring station [Countermeasure] Lightning protection measures, including insulation treatment on the main control room side of monitoring post/station, the use of non-metal material for electric conductors and installation of terminal boxes
7	June 1997	Isolation of primary argon confinement system	[Event] “Primary argon gas system flow high” and other alarms were set off and the system's isolation valve was closed. [Cause] False signal generated by inaccurate communication of work order [Countermeasure] Appropriate documentation of work orders and accurate communication
8	Jan. 1998*	Worker's injury in maintenance and waste disposal building	[Event] During return of the maintenance car to the specified position, a worker slipped and fell, breaking his leg (medical treatment for one month). [Cause] Worker error [Countermeasure] Education on and compliance with Standard Safety Rules
9	Oct. 1998*	Worker's injury during inspection of high-voltage busbar	[Event] During inspection of high-voltage busbar of turbine building, a worker suffered from moderate burns and injury to his face due to electric shock. [Cause] Worker error [Countermeasure] Clear description of live parts in work manuals, use of protective equipment.
10	Apr. 1999*	Worker's injury to right hand in reactor auxiliary building	[Event] Worker's right hand was caught in the door of EVST cooling piping room in the reactor auxiliary building during door opening operation, resulting in injury (cut finger). [Cause] Worker error [Countermeasure] Installation of dedicated jig to improve door opening/closing operation
11	Apr. 2008	Stop of secondary pony motor due to voltage drop by lightning on transmission line	[Event] Secondary main circulation pump pony motors A and B were automatically stopped due to instantaneous voltage drop by lightning, causing a deviation from operational limits. [Cause] Insufficient consideration of natural phenomena in the design [Countermeasure] The startup circuit of secondary pony motors were modified to ensure automatic restart after instantaneous voltage drop.
12	Sept. 2008*	Corrosion pitting in outdoor ventilation duct	[Event] Corrosion pitting occurred in the outdoor ventilation duct installed on the top of the reactor auxiliary building. [Cause] Progress of corrosion due to wet atmosphere for an extended period [Countermeasure] Replacement of the entire outdoor ventilation duct, and planned facility maintenance

* Corresponding also to accidents and failures reported under Ordinances. Item No.9 is a report under the Electricity Business Act and thus, not included in Fig. 11-1.

11. Accidents and Failures

Table 11-1 Accidents and failures reported under the Safety Agreement (2/2)

No.	Month /year	Event name	Overview
13	Dec. 2009	Inoperable air cooler blower during check of automatic load input of emergency diesel generator (DG)	[Event] During check of automatic load input to components important to safety following startup of the emergency DG by deenergizing the emergency bus, the auxiliary cooling system air cooler blower failed to start. [Cause] Malfunction of a breaker supplying power to blower (dust mixture into contacts) [Countermeasure] Thorough operation check of breakers of the plant protection system during startup check
14	Aug. 2010*	Drop of IVTM during clearing after refueling	[Event] During removal of IVTM after refueling, the IVTM was dropped and deformed due to the disconnection of gripper fingers. The screws of gripper finger drive rod turned out be loosened. [Cause] Design and manufacturing errors for prevention of finger drive rod rotation [Countermeasure] Change to a welded integral structure to prevent rotation of finger drive rod, and replacement of the IVTM
15	Dec. 2010*	Cracking of emergency DG unit C cylinder liner	[Event] Cracking occurred on a cylinder liner during load test after inspection of DG. [Cause] Overpressure on cylinder due to work without oil pressure control [Countermeasure] Ensure oil pressure control during removal of the cylinder liner, and clarification of procedures for work involving application of excessive pressure
16	Apr. 2013	Deviation from operational limits during commissioning of DG	[Event] Fire alarm was set off due to exhaust from DG unit C, and the DG was stopped. Since DG unit B was under inspection, “deviation from operational limits” was declared. [Cause] Maloperation of cock by operator and lack of checking by other operators [Countermeasure] Indication of “open/close direction” and “matchmark”, use of special jig for “open/close” operation, and double check
17	July 2015*	Deformation of indicator cock of emergency DG cylinder head	[Event] During overhaul of DG unit B, its cylinder head (450 kg) was dropped while lifting, resulting in the deformation of indicator cock and lubricating oil piping. [Cause] Use of unfamiliar new jig and mistake of crane operation [Countermeasure] The use of an unfamiliar jig was added as an important item in the site rules.

* Corresponding also to accidents and failures reported under Ordinances. Item No.9 is a report under the Electricity Business Act and thus, not included in Fig. 11-1.

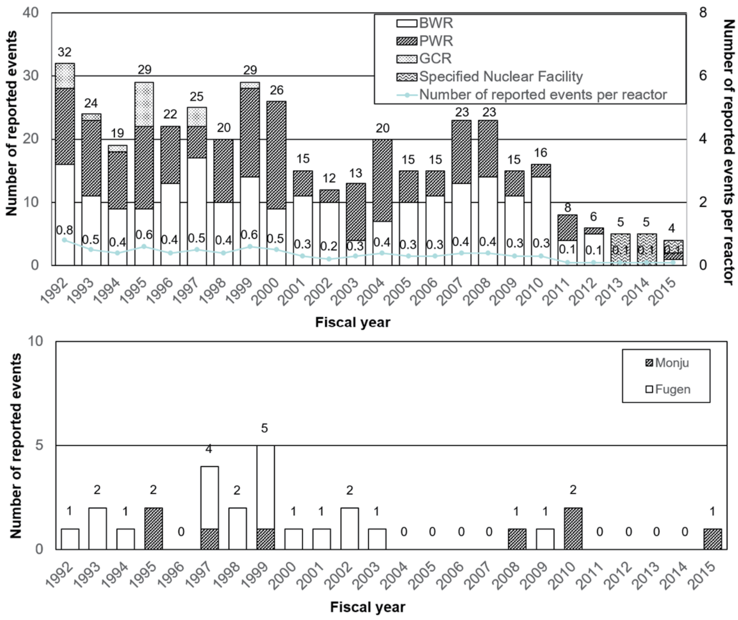


Fig.11-1 Number of accidents and failures of power reactors reported to NRA

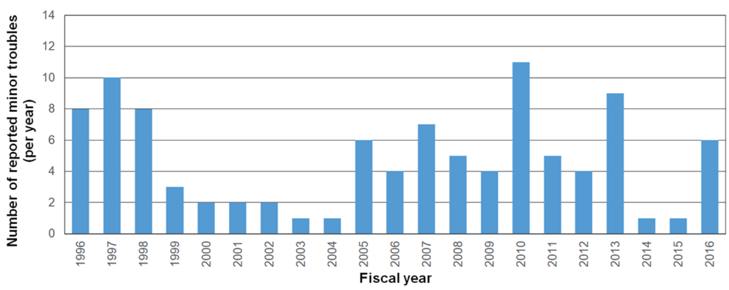


Fig.11-2 Minor troubles in Monju

11.2 Failures specific to a sodium-cooled FBR and lessons learned

Focusing on selected failures specific to a sodium-cooled FBR, overview, lessons learned, and findings useful for future FR design are given below:

(1) Thermal displacement of SHTS piping

In June 1991, a preliminary test to increase the temperature of the SHTS piping was performed during SKS. When the system temperature reached 120°C, the pipe adjacent to the IHX of loop C was displaced toward the RV side (by 36 mm). A part of the pipe was displaced toward the IHX side (by 5.5 mm), which differed from the prediction.

It was determined that the spring constant of the bellows at the CV penetration was three times larger than the design value and that the spring constant of the nitrogen gas seal bellows was somewhat larger than the design value, both of which constrained the thermal displacement of piping.

To cope with the problem, the CV penetration bellows was replaced with softer ones by changing from a two- to one-layer configuration, and the nitrogen gas seal bellows was changed to a reinforced rubber seal. Furthermore, additional pipe supports were installed.

Since the relevant CV penetration bellows had a larger ratio between the crest and pitch compared to the conventional type, the two layers of the bellows interfered mutually (moved as if they were a one-layer bellows), resulting in a spring constant 3 times larger than the design value. This experience is technologically important and can be reflected in the design of bellows that is required to absorb a large thermal displacement.

(2) Pressure drop of water-steam system flash tank

In February 1995, during the reactor startup test, in the process of increasing the reactor power from 2% to 10%, the outlet pressure of flash tank in the startup bypass line of the water-steam system dropped temporarily.

This pressure drop was caused by the following mechanism: flow resistance was increased by steam entrainment due to a circulating flow of hot water in the flash tank, and then the heat to be recovered in the deaerator was decreased. The increased amount of steam consumed then caused the pressure drop.

As countermeasures, the flash tank structure was modified to prevent steam entrainment, and the drain pipe to the deaerator was replaced with a larger one. Subsequently, normal heat recovery was confirmed by testing in the actual plant (Fig. 11-3).

The startup bypass line of the water-steam system was introduced as an advanced system to collectively recover the waste heat produced in the evaporators of the three loops to ensure effective use for heating feedwater before aeration into the superheaters. However, the steam entrainment could not be predicted since the complex configuration of the three loops merged to a single tank from different angles was difficult to investigate in the design stage. The possibility of generating a circulating flow in the tank was not studied by experiment

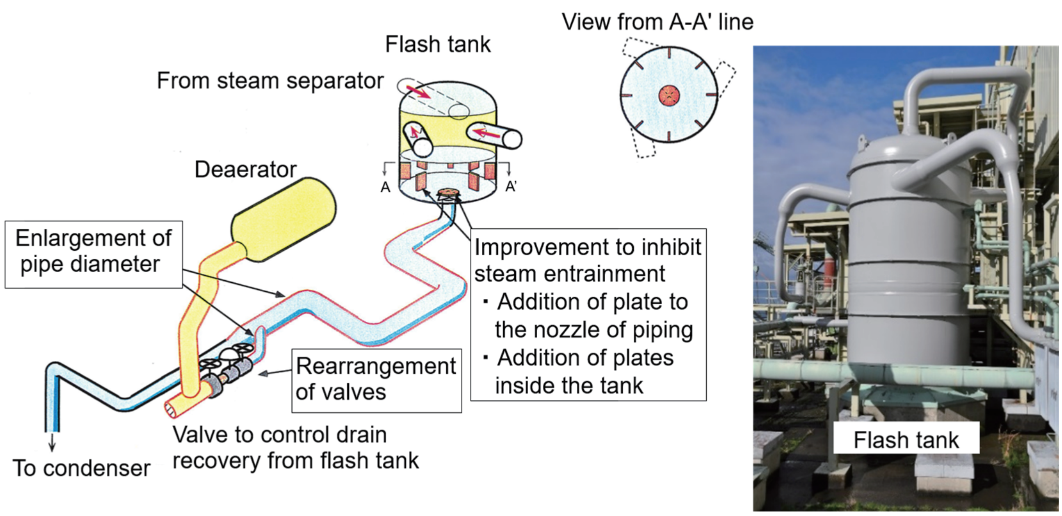


Fig.11-3 Flash tank and drain piping of water and steam system (improved)

or analysis. This experience highlighted the importance of design verification work when an advanced system design aiming at streamlining/improving efficiency is adopted, even if it is an extension of a general-purpose technology proven in water-steam systems.

(3) Secondary Sodium Leak Accident[11-2), 11-3)]

In December 1995, during the 40% power test, at a reactor thermal power of 43% (electric power of 40%), the "IHX C SHTS outlet sodium temperature high" alarm and fire alarms were simultaneously set off, followed by the "SHTS sodium leak" alarm. One of the operators went immediately to the SHTS loop C piping room and observed white smoke. The reactor was manually shut down after reducing the reactor power. A detailed investigation confirmed that the sheath of thermocouple installed in the IHX outlet piping was broken and about 640 kg of sodium leaked to the room atmosphere from the bottom end of the connector through the opening created by the break (Fig. 11-4).

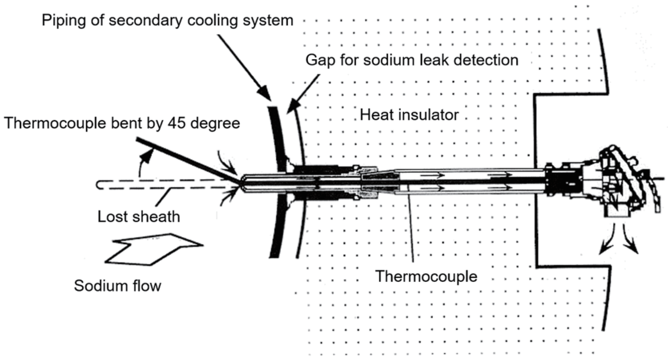


Fig.11-4 Breakage of the thermocouple and sodium leak path

As shown in Fig. 11-5, holes were formed on the air duct and grating floor located under the leaked piping area, and the leaked sodium was piled up on the floor in an oval shape with a diameter of about 3 m and a volume of about 1 m^3. In addition, the sodium compound (sodium aerosol) generated during the burning of sodium in air diffused to the other rooms of SHTS loop C and outside through the HVAC system.

The cause of the thermocouple sheath failure was investigated by various tests and analyses. It was clarified that the narrow part of the thermocouple sheath of a stepped structure was oscillated by the hydrodynamic force of the symmetric vortexes generated in flowing sodium. The resultant high-cycle fatigue concentrated at the stepped part of the sheath caused the breakage of sheath. The result of the numerical simulation of a coupled structural and hydrodynamic analysis is shown in Fig. 11-6. It

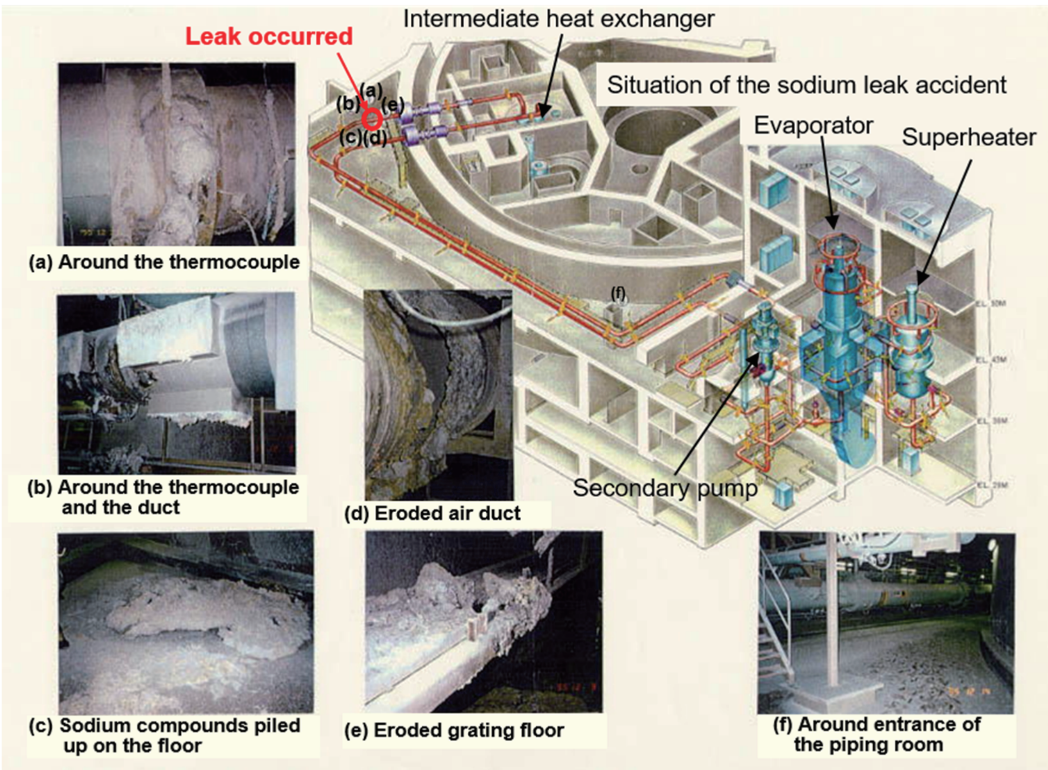


Fig.11-5 Scene of the Secondary Sodium Leak Accident

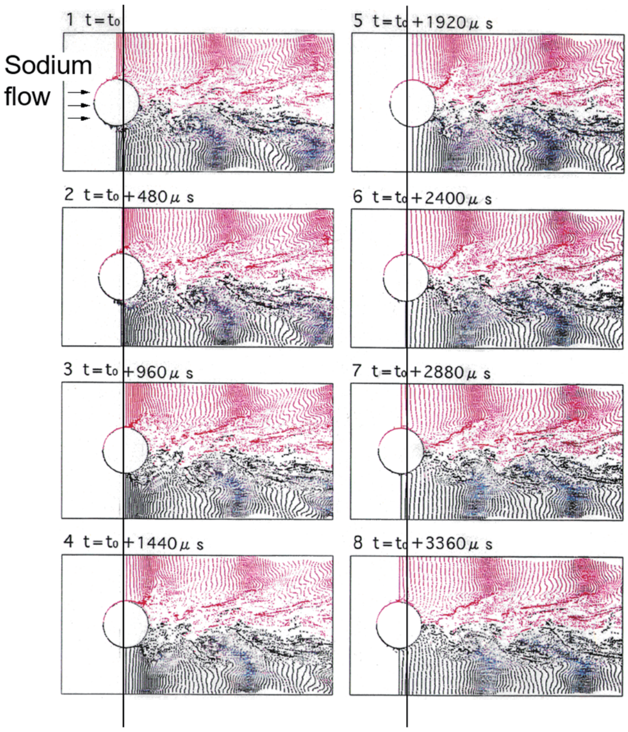
successfully reproduced the generation of symmetric vortexes and thermocouple oscillation (i.e., side-to-side displacement of the thermocouple centerline). When the thermocouple is stationary, the alternating vortexes are generated, while when assuming the coupled oscillation with the flow field, the symmetric vortexes are generated as they are in this case.

Concerning the influence of the burning of leaked sodium, an integral out-of-pile experiment was conducted to simulate the phenomena that actually took place under the accident conditions at the Monju site. In the experiment, a sequence of events were reproduced from leak initiation, the burning of spilled sodium, interactions with the structures, etc. The reproduction experiment was analyzed by the computer code that is used in the safety analysis of Monju.

Concerning the corrosion behavior of steel structures, two types of steel corrosion mechanisms, Na-Fe complex oxidation type corrosion and molten salt type corrosion, were identified based on various simulation tests. The latter type was not previously known, and the model used to predict the corrosion thinning rate of steel floor liner was revised (Fig. 11-7). Na-Fe complex oxidation type corrosion is induced by a mechanism in which complex oxides are produced by the reaction between sodium oxide and iron. This corrosion actually occurred on the floor liner during the Secondary Sodium Leak Accident. On the other hand, the molten salt type corrosion, which occurred in the reproduction experiments, is induced by a mechanism through which peroxide ions melted in sodium hydroxide under the specific condition of the experiments. The corrosion rate of the molten salt type is about five times larger than that of the Na-Fe complex oxidation type.

Based on the above results of cause determination, the following measures against sodium leak were taken to prevent recurrence:

Photo11-1 Integrated sodium leak monitoring system



Note) Solid line shows the neutral position of the well.
The 8 figures represent one cycle of vibration.

Fig.11-6 Symmetrical vortex generated at the back side of the Tc well

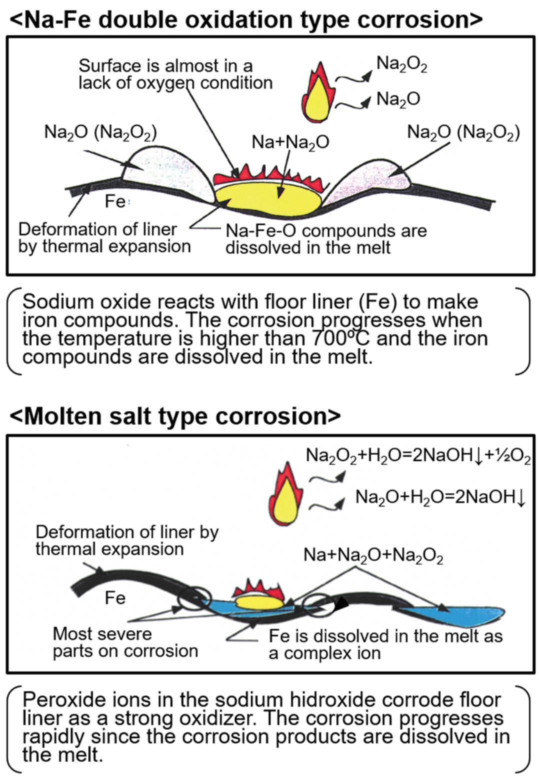


Fig.11-7 Corrosion mechanism of floor liner by leaked sodium

11. Accidents and Failures

- Replacement of thermocouples: The thermocouples were replaced with shorter, tapered ones with no stepped structure in order to prevent the breakage of sheath due to hydraulic oscillation. The number of thermocouples was reduced from 48 to 42 (Fig. 11-8).
- Early detection of sodium leak: A new detection system (cell monitor) was installed. The system consists of a smoke sensor to detect a small leak, and a heat sensor to detect room temperature rise, effective for an intermediate leak. The system activates an alarm in the main control room and automatically shuts down the HVAC system (Fig. 11-9). An integrated sodium leak monitoring system which displays all information related to sodium leak in the main control room, as shown in Photo 11-1, was newly introduced.
- Reduction of the amount of sodium leak (shortening of drain time): To rapidly drain sodium, modifications were made to add sodium drain piping, increase the diameter of the existing drain piping, multiplex and use of motor-operated drain valves, etc. (Fig. 11-9).

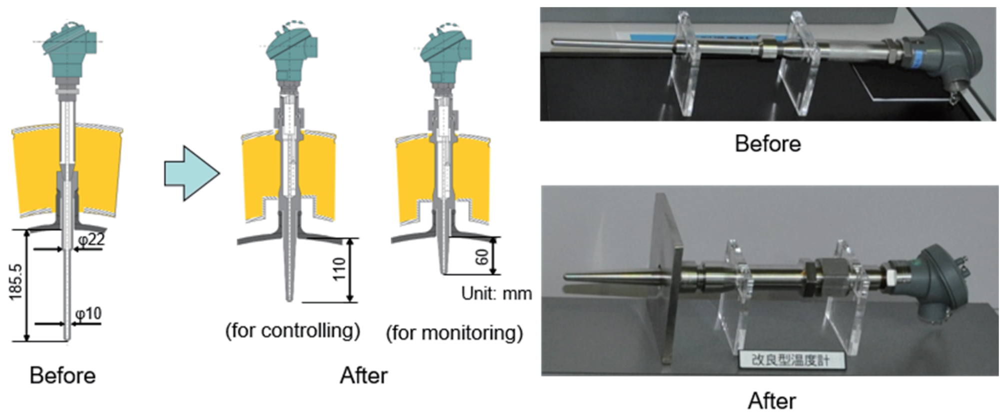


Fig.11-8 Improvement of thermocouple

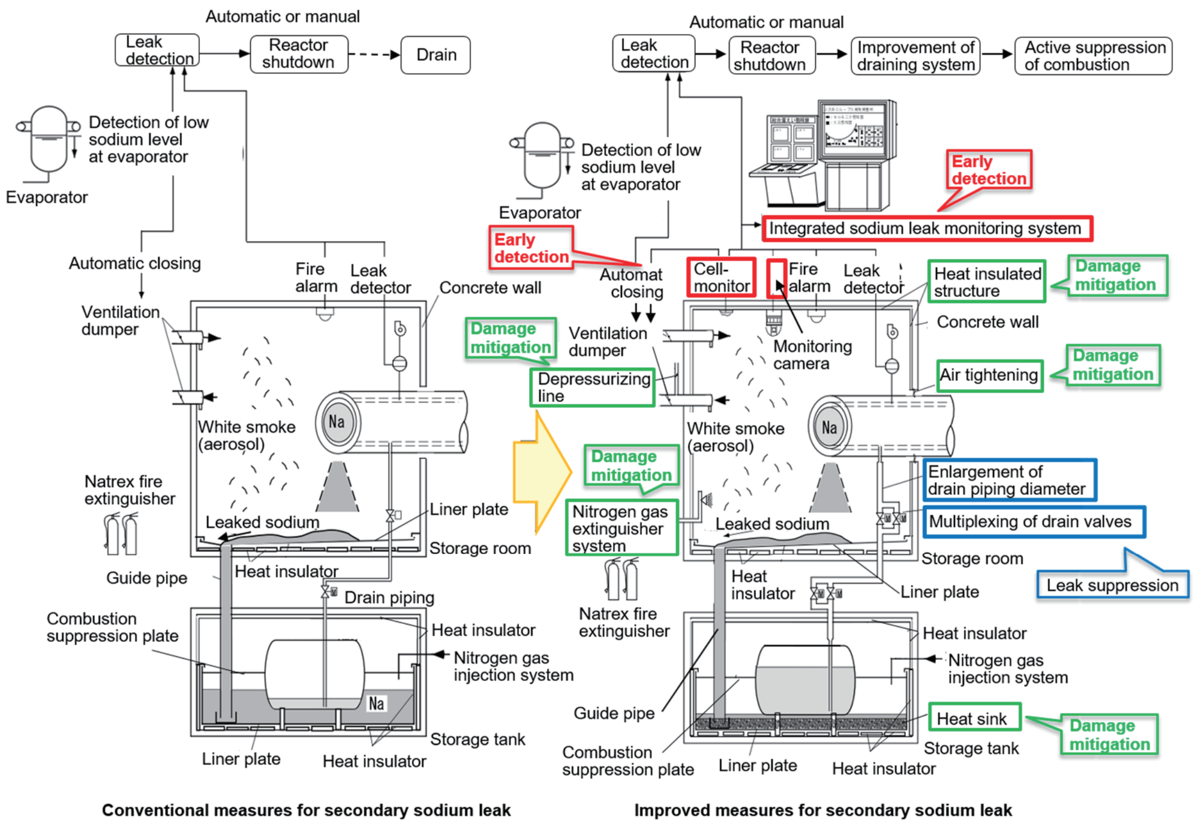


Fig.11-9 New safety measures for sodium leak in the secondary cooling system

- Suppression of sodium fire: A nitrogen gas injection system was installed into the SHTS rooms. In addition, the SHTS rooms were compartmentalized by loop (Fig. 11-9).

In addition to the above system modifications, operating procedures during sodium leak accidents were revised to perform emergency reactor shutdown and sodium drain, and immediately stop the operation of the ventilation system for early accident termination and the prevention of sodium aerosol dispersion.

The Secondary Sodium Leak Accident did not affect the core cooling function and had no influence on building integrity or external environment; however, the accident is taken seriously because of its relation to the sodium technology, one of the key FR technologies. Besides, the handling of information on the accident was criticized as inappropriate. This served as the impetus for JAEA to work toward thorough information disclosure.

The findings related to the flow-induced oscillation were generalized to develop the Evaluation Guides for Hydraulic Oscillation of Cylindrical Structure inside Piping (Japan Society of Mechanical Engineers standards: JSME S 012-1998).

(4) IVTM dropping[11-4)]

In August 2010, the IVTM was dropped inside the RV from 2 m above the installed position when being removed after refueling. It was being lifted using the auxiliary handling machine (AHM) for removal from the RV, (Fig. 11-10).

The drop was due to the unstable holding of the IVTM with AHM gripper fingers, which were not fully opened to hold the IVTM. The AHM gripper finger drive rod is plane-shaped, and must be carefully designed, manufactured, and maintained in order to prevent rotation. However, the sufficient care to prevent rotation was not taken in replacing the power cylinder of AHM, resulting in the screws gradually becoming loose to induce the finger drive rod to rotate. As a countermeasure, the finger drive rod was changed to one with a welded structure that does not rotate, and the function was enhanced to detect an un-lifted condition.

In the recovery work, an in-vessel observation device was newly developed to observe the IVTM that dropped inside the RV. The device was tested in a full-scale mockup of the actual conditions. Functions were confirmed on a heater installed at the observation mirror, illumination, and camera settings. The device was then inserted into the RV and the damaged state of the IVTM was observed (Fig. 11-11). It was shown that the IVTM inner guide tube was deformed and interfering with the inner surface of the fuel transfer hole sleeve. Since the damaged IVTM could not be withdrawn by the normal method, it was removed together with the sleeve. Finally, the new IVTM was manufactured and installed.

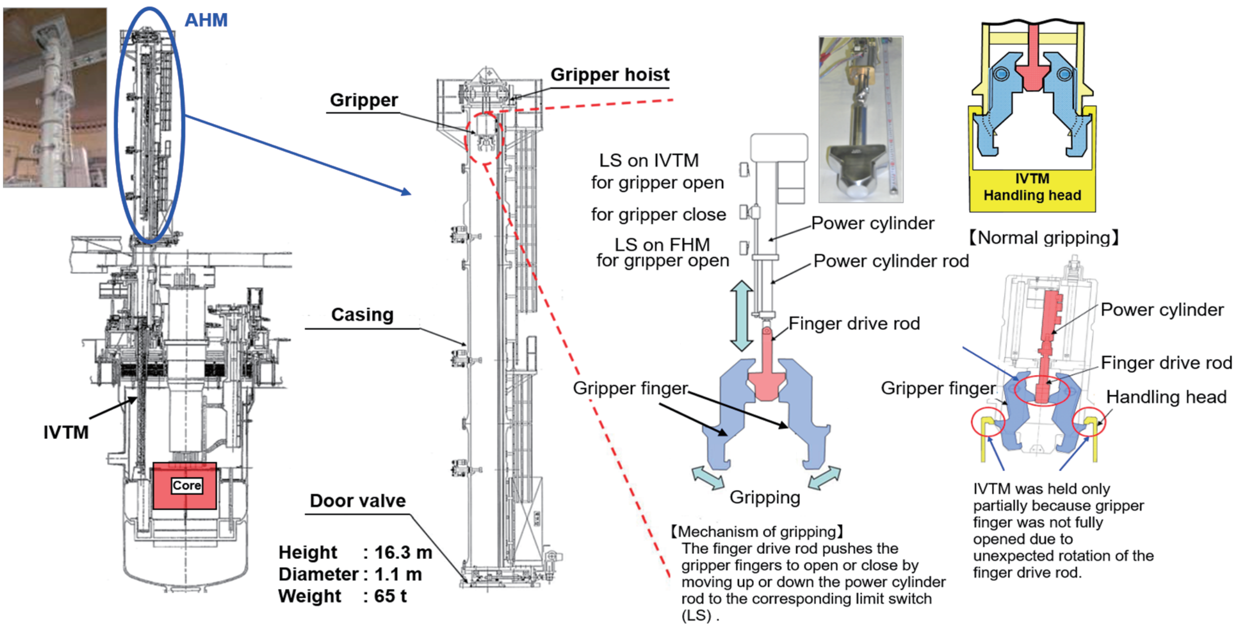


Fig.11-10 In-vessel transfer machine (IVTM) and Auxiliary handling machine (AHM)

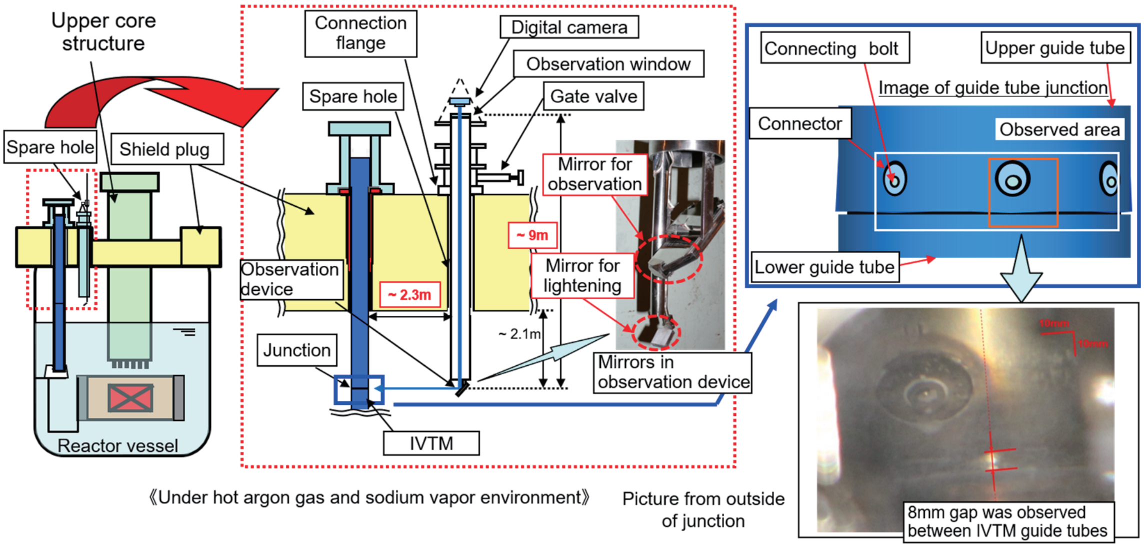


Fig.11-11 Remote observation of IVTM

The in-vessel observation technique is useful for future sodium handling in a high-temperature argon gas and sodium vapor atmosphere.

(5) False alarm events for sodium leak detectors

Two false alarm events in sodium leak detectors occurred as described below.

In the “false alarm event for contact-type sodium leak detector (CLD) in the primary maintenance cooling system”, a sodium leak alarm was set off in March 2008, from the CLD attached to the RV inlet stop valve of the primary maintenance cooling system. The cause was the over-deep insertion of CLD when attaching to the valve because a fitting (sealant) designed to insert and fix the tip of the detector at a specified position was poorly fixed. As a countermeasure, the sealant-type CLDs were replaced with Swagelok-type CLDs.

In the “false alarm event for sodium leak from SHTS loop C”, a sodium leak alarm was set off in January 2009, from the radiative ionization detector (RID), a gas sampling type detector in SHTS loop C. It was presumed that the RID reading was falsely increased by the absorption of volatile gas generated from painting work in the turbine building. As a countermeasure, monitoring of the RID reading is strengthened during the painting work, and the painting in the RID monitored area is performed while the sodium is drained.

Findings concerning manufacturing management, causes for performance deterioration, and knowledge for system improvement have been accumulated from the defects and failures of sodium leak detectors in Monju. In terms of operation, effective measures to prevent the activation of false alarms were taken according to detector type and location by thoroughly investigating and extracting the causes of the false alarms (Table 11-2).

In addition to technical concerns, since there is significant public concern about sodium leaks, even a false alarm can cause strong public anxiety about reactor safety. When installing sodium leak detectors in future FRs, it is appropriate to take into account the required safety functions and detector reliability.

11.3 Notification of accidents and failures

When an occurrence of anomaly to be reported is confirmed, it must be reported promptly to the regulatory body and local governments as “notification information”.

An event that may lead to “anomaly” is reported immediately as “immediate communication information” in Monju.

When judged to be a minor incident below the above levels, the information is reported as “minor incident information” in order to maintain good relationships with the local communities.

For the mass media, “notification information” is directly explained at the press centers immediately after notification, and the other information is reported on a weekly basis.

The “weekly report” was introduced as a part of thorough information disclosure to recover trust in Monju that was lost after the Secondary Sodium Leak Accident.

11.4 Risk communication using collection of accident and event cases

For the operation of Monju, it is important that the information with respect to accidents and failures is shared in advance by the stakeholders, especially the local communities including local governments, fire and police departments, and mass media. The information would include what is postulated, what preventive measures are taken, and how they are coped with. To facilitate this information sharing, a booklet with a collection of representative cases of accidents and failures being postulated in Monju was developed (Photo 11-2). Each case document consists of an overview, consequences, response, recovery period, information classification, preventive measures, INES scale, and illustrations of the area where the accident occurs. A total of more than 138 out of about 900 cases of the accidents and failures that occurred or were postulated in Monju, Joyo, foreign FRs, and domestic LWRs were selected for inclusion in the booklet (Table 11-3).

Photo11-2 Collection of accident and event cases

Table 11-2 Summary of false sodium leak alarms

No.	Cause of false alarm	Affected detector					Experience (Yes/No)	Remarks
		SID	DPD	RID	CLD	Air atmosphere cell monitor		
1	Change in outside air temperature, and change in atmosphere temperature due to startup/stop of HVAC system			○			Yes	Change in signal processing for alarm monitoring from 24-hour to 1-hour deviation
2	Volatile components of heat insulator			○			Yes	The increase rate of system temperature is limited to 5°C/h or less.
3	Volatile components of paint			○			Yes	Ensured control of paint work and monitoring of RID reading
4	Dust		○	○		○	Yes	Multiplicity of air atmosphere cell monitors
5	Frequency variation of power supply system			○			Yes	—
6	Electrical noise due to lightning, power switching, etc.	○	○	○			Yes	—
7	Startup of sampling pump/blower and adjustment of sampling flow rate	○	○	○			Yes	Full attention is required to prevent activation of alarm during operation.
8	Detachment and aeration of RID filter (including training), replacement of DPD filter		○	○			Yes	Full attention is required to prevent activation of alarm during work.
9	Short circuit, opening and breaking of signal cable due to work, etc.				○		Yes	
10	Metal powder and chips				○		No	Cleanliness control during replacement and inspection
11	Smoke in welding, etc.			○		○	No	Work should be performed after sodium drain in the work area
12	Change in atmosphere pressure due to startup/stop of HVAC system			○			No	Full attention is required to prevent activation of alarm during operation.
13	Change in atmosphere pressure due to increased pressure during the CV-LRT	○					No	Full attention is required to prevent activation of alarm during work.

(Note) SID: Sodium ionization detector, DPD: Differential pressure type detector, RID: Radiative ionization detector, CLD: Contact-type sodium leak detector

Table 11-3 Number of responses to accidents and failures

Classification	Number of events
Sodium leak (primary/secondary sodium, sodium in EVST)	16
Sodium sticking/blockage, mixing of foreign material into sodium	14
Failure of instrumentation and control system	14
Failure/malfunction of component	14
Leak of water/steam (steam, seawater, water, chemicals, oil)	9
Deformation / destruction of structure (vessel, piping, fuel subassembly, heat transfer tube)	8
Radioactivity release (gas, liquid, solid)	8
During the Modified Parts Confirmation Tests (MKS), PKS, SST	4
Fire (electric, turbine oil, controlled area, welding)	4
Fault in electrical system (diesel, power generation equipment, motor)	4
Sodium-water reaction (small leak, intermediate/large leak)	2
Fuel failure	2
Others (personal injury, external event, human error)	23
Subtotal (as of February 2008)	122
Events assumed to occur during Core Performance Confirmation Tests	4
Troubles that occurred at domestic plants after February 2008	5
Events assumed to occur during refueling	1
Troubles that occurred in Monju after February 2008	6
Subtotal (those added as of October 2009)	16

— References —

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11-4) JAEA, Deformation of In-Vessel Transfer Machine due to a Drop, 2012, (in Japanese), Retrieved March 1, 2019, from https://www.jaea.go.jp/04/turuga/jturuga/press/posirase/1203/o120312.pdf (accessed March 1, 2019).

Message for the Future

The development of Monju has extended over half a century. The hope and passion devoted to the dawn of FBR development can be vividly remembered even today from a book “A Ten-year History of PNC”. The development of Monju started as a national project to secure stable domestic energy supply. Upon commencement of the project, Susumu Kiyonari, the then PNC president, stated his strong determination that “We shoud not proceed on the *easy path* of technology introduced from abroad, which has been practiced since the Meiji era and always entails technological backwardness. Instead, our project should be based on domestic technologies. This is the *hard path* because of possible trial-and-error and design changes associated with technological uncertainties, implanted distrust in domestic technologies, and the continuous temptation to return to the introduction of technology from abroad.”

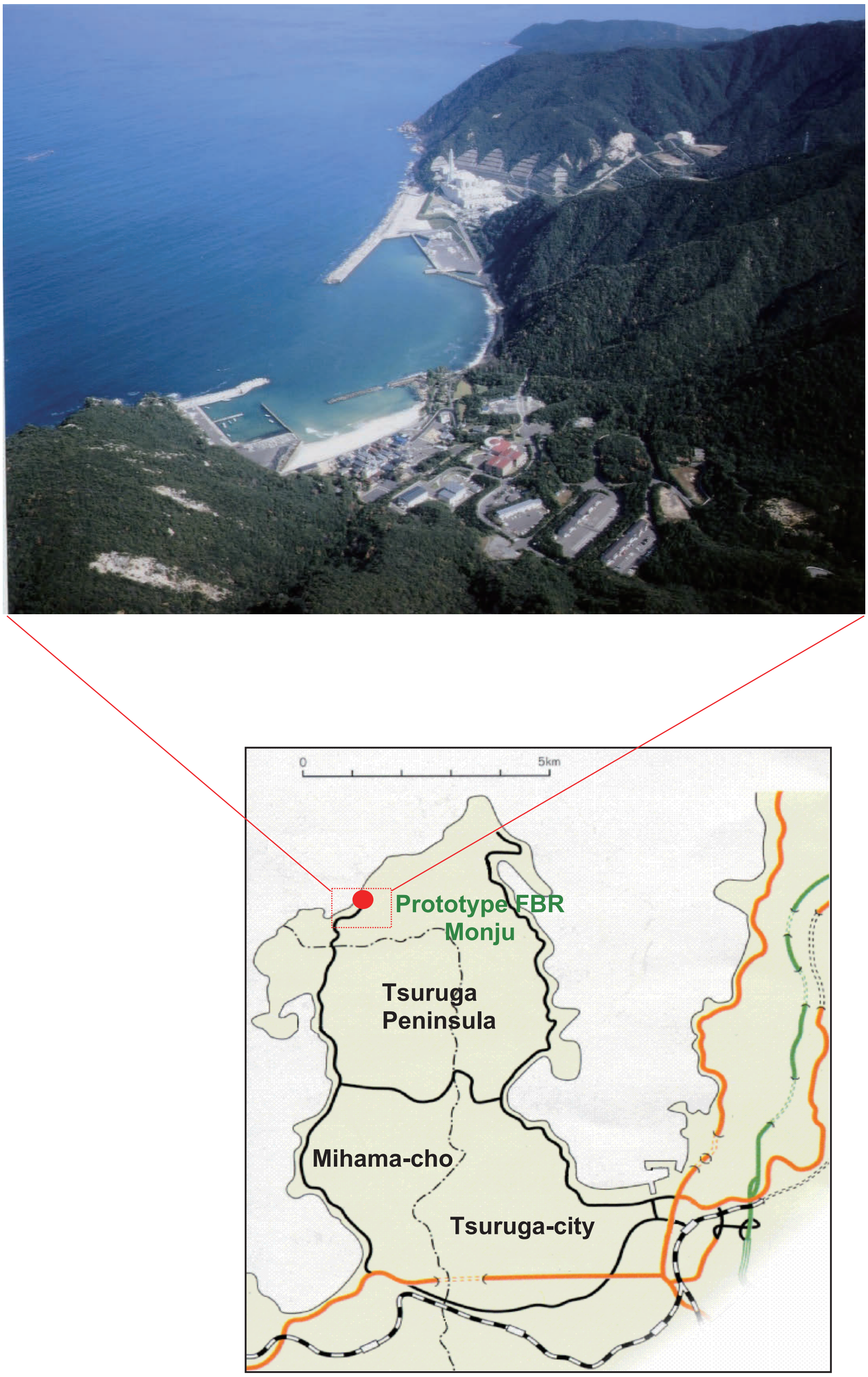
The history and achievements of Monju development are described in detail in this report. Through creation of the R&D infrastructure, such as the Experimental Fast Reactor Joyo and the 50-MW SG test facility at the OEC, and effective use of international cooperation, Monju was developed, designed and constructed, and then achieved commissioning operation at 40% rated power. These achievements confirmed the validity of design and R&D activities and demonstrated that the level of Japan's FBR technology is comparable to that of advanced countries. Furthermore, successful completion of manufacturing and construction as scheduled is a remarkable achievement, knowing Monju is the first of a kind in Japan.

The long-term shutdown was a painful experience for all concerned. Maintaining ambition to resume operation and contribute to the R&D toward fast reactor commercialization has helped to keep motivation high. The enthusiasm of all staff and the emotion of local media when resuming operation of Monju were unforgettable.

Future FR development is likely to be prolonged depending on changes in energy policy in Japan. As the technological environment in society changes significantly, such as the spread of AI and robotics, the way to proceed R&D will certainly change. Even in, and because of, such changes, the technologies and experience acquired in Monju are valuable and it is important to use them. Only those experts with an enterprising spirit may extract new knowledge and lessons out of these. A pioneer engineer expressed the difficulty and importance of construcing an actual plant, saying “Building an actual plant exceeds prototyping one hundred times”.

Major components of Monju were custom made based on thorough R&D. The accidents and failures experienced occurred mostly in the modified general-purpose products after being streamlined or improved for higher efficiency, rather than in major FR components. To construct and operate FR plants in the future, the technologies for every plant detail should be appropriately considered according to their importance, in addition to the desk works and study in R&D facilities, as nuclear engineering is called an integrated technology.

We hope the technologies and experience acquired in Monju will long be utilized as valuable assets that Japan has developed independently.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5km
Prototype FBR
Monju
Tsuruga
Peninsula
Mihama-cho
Tsuruga-city

Appendix

1. Major Specifications

2. Key Events in Monju Project

1. Major Specifications

The rationale behind the selection of Monju's major specifications was that experience with a prototype reactor would lead to the development of technologies for commercial reactors that were based mainly on Japanese concepts, while taking into account possible progress in existing technologies and the trend of preceding prototype FRs developed abroad (Table A1-1).

Table A1-1 Major specifications of Monju

Reactor type	Sodium cooled loop type
Thermal power	714 MWth
Electric power	~280 MWe
Fuel	Plutonium-uranium mixed oxide
Burnup of fuel	80 GWd/t (average of discharged fuel)
Fuel cladding	SUS316-equivalent steel
Cladding temperature	Max 675°C (mid-wall)
Breeding ratio	1.2
Number of cooling loops	3
Location of circulation pump	Hot leg installation
Reactor temperature	397°C / 529°C (inlet / outlet)
Secondary sodium temperature	505°C / 325°C (hot leg / cold leg)
Main steam conditions	127 kg/cm^{2}G, 483°C
Steam generator (SG) type	Helically-coiled, separate type
SG layout	Concentrated arrangement
Refueling scheme	Single rotating plug with a fixed arm
Refueling interval	~6 months
Decay heat removal	Installed in the secondary cooling system
Method to ensure coolant level	Pipe routing at high elevation with guard vessels

(1) Coolant: Sodium

In FRs, neutrons emitted in a fission reaction do not need to be slowed down, thus liquid metals such as sodium, NaK (alloy of sodium and potassium), lead bismuth and mercury, and gases such as helium, carbon dioxide, and steam have been widely investigated as coolant candidates in FR developing countries.

Mercury, NaK, and lead bismuth were first used in small experimental reactors. However, sodium became the standard coolant choice worldwide based on its overall performance in terms of heat removal capability, high boiling point, compatibility with structural materials, chemical activities, behavior during leakage, etc. As a result, sodium was adopted in Monju as well.

(2) Reactor type: Loop type

By the time the Prototype Reactor Phase 1 Design was initiated, construction of prototype FRs had started in the U.K.'s PFR and French Phenix, where a tank-type (or pool-type) reactor design was adopted, whereas plant design study was ongoing in the U.S. CRBR and German SNR300, where a loop-type reactor design was adopted. The advantages and disadvantages of the two types were examined in reference to these designs.

As for the tank type, a serious technical difficulty was foreseen in the attempt to confirm the reliability of the tank and components in a short time. For example, substantial R&D activities would have become necessary including data accumulation on thermal hydraulics and temperature distribution in the tank, thermal strain, etc. In addition, the tank would be much larger in a commercial plant, and this would pose a critical problem in securing the seismic integrity, especially in earthquake-prone countries like Japan. The resultant cost of R&D would also be considerably higher than that for the loop type.

The accessibility to components is another important factor for the reactors in the R&D stage, where components are subject to continuous improvement or replacement with newly developed ones. It is also necessary to reflect R&D results carried out in parallel to design. Based on these considerations, the loop-type reactor was judged preferable and selected.

(3) Power: 714 MWth, 280 MWe

Considering both technological extrapolation from Joyo (five times larger) and extrapolation to a future large reactor (three to five times larger), 300 MWe is the optimum electric power for the prototype reactor.

Performance related to the reactor power includes steam conditions of 127 kg/cm^{2}G and 483°C, burnup of 80 GWd/t, and a refueling interval of 6 months. The reactor power to satisfy such performance should be at least 300 MWe. That is, for example, in a 200 MWe class, a larger excess reactivity is required to compensate for the burnup reactivity loss, and thus more control rods should be provided, leading to higher local power peaking. As a result, plant performance such as the steam conditions would be lowered and technological extrapolation to a future large reactor becomes difficult.

In those days, foreign prototype reactors were generally around 300 MWe: French Phenix and the U.K.'s PFR were 250 MWe and BN350 of Russia (then the Soviet Union) was 350 MWe. German SNR-300, which was under construction, was 300 MWe, and the U.S. CRBR, just before construction, was 380 MWe. With this in mind, 300 MWe was judged to be appropriate for Monju, and the thermal power was set at 714 MW based on the target thermal efficiency of a sodium cooled FR, 42%. Concerning the steam conditions in the SG, the decision was made to change from a reheat cycle to a non-reheat cycle later in 1977 for simpler components and easier operation. The resultant thermal efficiency was reduced to 40%, and the reactor power was changed to 280 MWe accordingly.

(4) Fuel: Plutonium and uranium mixed oxide fuel

Metal fuel was also studied for its high density of heavy elements to pursue high breeding performance; however, it was not suitable for high burnup since significant deformation due to swelling would be caused by irradiation. Meanwhile, oxide fuel used in LWRs has been successfully fabricated and irradiated, and is known to be suitable for high burnup. Thus, a plutonium and uranium mixed oxide fuel, which was mainstream globally, was employed.

(5) Average burnup of discharged fuel: 80 GWd/t

The target burnup of a large FR in the commercial stage, generally, was set at a core-averaged burnup of 100 GWd/t or a maximum burnup of 150 GWd/t; however, because of a concern for the swelling of fuel and cladding, the averaged burnup was set at 80 GWd/t. In addition, it was tentatively limited to 55 GWd/t until irradiation data for the cladding material could be obtained up to a target fast neutron fluence.

(6) Cladding material: 316-equivalent steel

Austenitic stainless steel has been widely used as cladding in foreign FRs because it has a good compatibility with sodium and fuel, high resistance to irradiation damage by fast neutrons, high temperature structural strength, etc. Austenitic stainless steel was also evaluated to have good performance in Joyo, and thus SUS316 stainless steel was selected for Monju. Through repeated trial manufacturing with domestic materials and in-pile and out-of-pile tests, SUS316-equivalent stainless steel was developed by adjusting the content of additive elements within the range of the standard composition of SUS316 in order to suppress swelling by irradiation.

(7) Maximum cladding temperature: 675°C at mid-wall

Reduction of the creep strength at high temperatures was a concern regarding stainless steel used above 650°C. Expecting technological progress, the maximum cladding temperature was initially set at 700°C (inner surface). However, it was later reduced to 675°C at the mid-wall because it was predicted in a detailed design study that the reduction of creep strength might become significant before the target burnup is reached.

(8) Number of cooling loops: 3

In the Prototype Reactor Phase 1 Design, a 3-loop concept was selected from possible 2- to 4-loop options. A 2-loop concept exhibits better economy, but redundancy for safety could not be secured in case of the shutdown of one loop. Thus, the 3-loop concept was adopted.

(9) Location of the main circulation pump: Cold leg installation

Installation of the circulation pump in the cold leg was adopted in Joyo. Disadvantages of this include a longer pump shaft length, a larger hot-leg pipe diameter, and a larger IHX due to pressure loss restrictions. On the other hand, a serious disadvantage of hot-leg installation is thermal shock associated upon plant startup or shutdown, for which large-scale and long-term development would be required to establish countermeasures. Thus, the cold-leg installation was selected. The secondary main circulation pump was also installed in the cold leg for direct reflection of the development results of the primary pump.

(10) Sodium temperature: Reactor inlet/outlet 397/529°C SG inlet/outlet 505/325°C

The reactor outlet coolant temperature was aimed at 550°C to 580°C in the Prototype Reactor Phase 1 Design in 1969. A parameter survey conducted with a reference value of 510°C taken from the steam condition of the then thermal power plants showed feasibility up to 550°C. In the Monju Phase 1 Design in 1971, the reactor outlet temperature was set at 540°C in consideration of the detailed thermal hydraulic design of the core and the temperature limits of structural materials used in the SG.

Subsequently, in the Adjustment Design from 1974, the maximum cladding temperature was lowered in order to achieve the target burnup of 80 GWd/t. As a result of adjustment including the increase in the coolant flow rate, the reactor inlet/outlet coolant temperatures were changed from 390/540°C to 397/529°C.

The SG inlet sodium temperature was changed from the initial value, 510°C, to 505°C in response to the change to non-reheat cycle. The steam generator outlet temperature was determined to be 325°C from heat balance.

(11) Main steam conditions: 127 kg/cm^{2}G and 483°C

In the Prototype Reactor Phase 1 Design, the main steam conditions were set at a pressure of 169 kg/cm^{2}G and a temperature of 510°C. This is based on the assumptions that the temperature and pressure should be as high as possible taking advantage of sodium cooling, the steam turbine of an existing new thermal power plant could be used as it is or slightly modified, ferritic steel could be used for superheated portions as in the thermal power plant, and the maximum fuel cladding temperature would be set at 700°C.

Subsequently, in the Monju Phase 1 Design, the main steam conditions were changed to 127 kg/cm^{2}G and 483°C considering changes in the reactor outlet coolant temperature, etc. as described in item (10).

(12) SG type: Helically-coiled, separate type

A helically-coiled type was selected because of good heat transfer performance, easier absorption of thermal expansion, compactness, and easier size increase, although this complicates the configuration.

Concerning the arrangement of evaporator and superheater, an integrated type is advantageous from the perspective of volume reduction. However, a separated type was adopted in consideration of the following factors: there were concerns regarding flow instability on the water side; material that can satisfy required high-temperature strength on the superheater side and anti-stress corrosion crack performance on the evaporator side was under development during design phases; and a separated type was adopted in earlier foreign plants.

(13) SG layout: Concentrated arrangement

In a loop-type reactor, it is desirable that the loops of the primary cooling system be arranged symmetrically around the reactor core in the same shape. On the other hand, only one turbine generator is installed, and a steam system, condensate/water supply system, piping, etc. are collectively arranged in a rectangular turbine building in a thermal power plant (concentrated arrangement).

Monju also adopted the concentrated arrangement and arranged the 3 SGs adjacent to the turbine building. In this case, the length of the SHTS piping differs among loops. A pressure loss adjustment mechanism was then provided to equalize system pressure loss.

(14) Refueling scheme: Single rotating plug with a fixed arm

Two methods, a hot cell plug removal method and under plug operation method, were investigated.

In the hot cell plug removal method, a shield plug is pulled up in a hot cell under an inert gas atmosphere, and fuel subassemblies are exchanged by visual observation from a window. In this method, the deposition of sodium vapor is inevitable, and there are many technological difficulties such as working in an inert atmosphere, and large-scale and long-term R&D was required.

The under plug operation method could be developed through relatively small-scale R&D utilizing experience in Joyo. The double-rotating plug method used in Joyo or single-rotating plug/fixed (variable) arm method could be candidates. Among these, a single-rotating plug method was adopted for ease of scale-up to a commercial FR.

(15) Refueling interval: 6 months

The target refueling interval for a commercial FR is around 1 year based on experience obtained in LWRs. In Monju, it was set at about 6 months for design flexibility on the excess reactivity for burnup compensation, the number of control rods, the change in power distribution, the burnup of discharged fuel, the number of refueling batches, etc.

(16) Decay heat removal method: IRACS in the SHTS

Initially, the PRACS (Primary Reactor Auxiliary Cooling System) method, in which the heat removal coil was incorporated into the IHX located inside the reactor containment vessel (CV), was examined as a decay heat removal method during reactor shutdown. This was because the number of related components located along the route to the ultimate heat sink would be reduced and higher reliability would be achieved when heat is removed from a place closer to the core. However, the PRACS method was later abandoned due to problems with the simplification and manufacturability of the IHX and the complicated local thermal hydraulics of the primary coolant. In 1978, during the Manufacturing Preparation Design, the IRACS (Intermediate Reactor Auxiliary Cooling System) method, in which an air cooler (auxiliary cooling system) is branched from the SHTS to be installed in parallel to the SG, was adopted instead.

(17) Method of ensuring coolant level in case of sodium leak: Pipe routing at high elevation with guard vessels

In the early conceptual design, the double pipe method employed in Joyo was adopted in Monju for piping in the PHTS to ensure the coolant level necessary for reactor core cooling in case of a sodium leak accident. However, the double pipe system has a complicated structure and is difficult to manufacture and inspect. In the Monju Phase 3 Design in 1972, complicated pipe routing design became practical based on progress in the structural design policy, and single-wall pipe routing at high elevation and use of guard vessels were adopted in Monju. That is, the reactor vessel, other components, and connecting pipes installed at low elevations are provided with outer vessels (i.e., guard vessels), and other pipes are routed at high elevations.

2. Key Events in Monju Project

Table A2-1 Key events in Monju project (1/6)

Year	Month	Monju, JAEA	Government, local government, etc.
1956	1		Establishment of Japan Atomic Energy Commission (JAEC)
	5		Establishment of Science and Technology Agency (STA, present MEXT)
1962	8		Establishment of Special Committee on Power Reactor (to discuss basic policy of FBR development)
1964	8		3rd International Conference on the Peaceful Uses of Atomic Energy (Geneva Conference)
1966	5		JAEC formulates "Basic Policy for Power Reactor Development" to develop FBRs as a national project.
1967	10	Foundation of Power Reactor and Nuclear Fuel Development Corporation (PNC)	
1968	3		The Government decides the PNC's Basic Policy for Power Reactor Development (to construct a commercial FBR before 1990, half funded by the private sector).
	4		Government decides 1st Basic Plan for Power Reactor Development (to develop a prototype reactor: sodium-cooled MOX fueled FBR of 200–300 MW expecting start operation before 1980).
	9	Orders placed for Preliminary Design of prototype FBR with 5 manufactures.	
1970	3	Construction of the experimental FR begins.	
	4	"Monju" and "Joyo" named.	
		PNC selects Siraki, Tsuruga City as a possible site of Monju construction.	
1971	7	PNC signs an agreement jointly with U.K. for full-scale core mockup experiments for Monju.	
		PNC signs a construction contract for the 50-MW SG facility.	
1972	6		Long-term Plan for Research, Development and Utilization of Nuclear Energy formulated (FBRs would be the mainstream of nuclear power generation).
1975	7		Tsuruga City council adopts a petition for accelerated construction of Monju from the Shiraki region.
	8		Special Committee on Advanced Power Reactor Development starts Check & Review (C&R) of Monju project.
1976	5	Application for Preliminary Survey Approval to Fukui Pref.	
	6		Fukui Pref.: Approval of Preliminary Survey of the site
	8		Special Committee on Advanced Power Reactor Development submits the C&R report to JAEC (concluding that the plan for the prototype FBR is appropriate).
1977	4	Joyo: Initial criticality	
1978	8	Environmental Impact Statement submitted to the Government and Fukui Pref.	
	10		Establishment of Nuclear Safety Commission (NSC)
	11	Meetings with local residents to explain Environmental Impact Statement held.	
1979	2	Review under Natural Parks Act begins (Natural Environment Investigation Report submitted to Fukui Pref.).	

Table A2-1 Key events in Monju project (2/6)

Year	Month	Monju, JAEA	Government, local government, etc.
1980	2		Cooperation agreement on construction of Monju concluded with 9 electric utilities, Electric Power Development Co., and Japan Atomic Power Company.
	4		Establishment of Fast Breeder Reactor Engineering Co. (FBEC)
			Ministry of International Trade and Industry essentially completes Environmental Impact Review.
	11		NSC formulates “Safety Evaluation Policy of Liquid-Metal Fast Breeder Reactors”.
	12		Fukui Pref. approves the start of the Safety Review.
		Submission of Application for Reactor Installation Permit of Monju	
1982	2		Meetings with local residents to explain the safety of Monju (on the results of Safety Review Round 1)
	3		STA requests construction of Monju to local governments (approved by the Governor of Fukui in May).
	5		Siting and construction of Monju approved at Cabinet meeting.
	7		NSC holds the second public hearing.
		Construction of Plutonium Fuel Production Facility begins.	
1983	1	Construction of access road to the site begins.	
	2	Agreement on Fisheries Compensation concluded between PNC and Tsuruga City Fisheries Cooperative.	
	3		Construction of Shiraki tunnel begins (ends in March 1985).
	4		NSC reports the result of the Safety Review Round 2 to the Prime Minister.
	5		The Prime Minister issues the Reactor Installation Permit.
		Agreement for Ensuring Safety in Surrounding Environment Relating to the Construction Work, etc. with Fukui Pref. and Tsuruga City concluded.	
	8	Construction of Monju tunnel begins (ends in March 1985).	
1984	1	First Contracts on component manufacturing with 4 manufacturers	
	11	Breakwater and revetment completed.	
	12	Application for Approval of Design and Construction Methods No.1 to STA	
1985	9	Application for Construction Work Permit under the Natural Parks Act	
			Actions for the Declaration of Nullity of the Reactor Installation Permit and Injunctive Order of Reactor Construction filed.
	10	Application for Building Confirmation submitted to Fukui Pref.	
		Reactor plant construction work begins.	
		Ceremony for the commencement of construction of Monju. The site renamed the “FBR Monju Construction Office”.	
		Excavation of foundation begins (ends in April 1986).	
	11	Construction contract (for reactor and auxiliary buildings) concluded.	
1986	7	Erection of the CV starts (ends in April 1987).	
1987	10	Construction of unloading wharf completed.	

Table A2-1 Key events in Monju project (3/6)

Year	Month	Monju, JAEA	Government, local government, etc.
1989	10	Monju Operation Preparation Office organized.	
		Fabrication of fuel subassemblies started at PNC's Tokai Works.	
1990	4	275 kV power-receiving equipment starts operation.	
	7	The number of site workers peaks at 3,761.	
	10	Construction of Administration Building completed.	
1991	3	Sodium reception begins on site.	
		Monju Advanced Reactor Simulator installed.	
		Component installation completed.	
	5	Comprehensive System Function Tests (SKS) begin.	
	7	Sodium charged to the secondary loops.	
	8	Sodium charged to the primary loops.	
1992	7	Arrival of 1st batch of core fuel subassemblies on site	
	12	CV-LRT test finished successfully.	
		Preliminary plant performance tests begin.	
1993	6	Preliminary plant performance tests end.	
	10	Criticality tests begin.	
		Loading of inner core fuel subassemblies begins, Reactor Operational Safety Program introduced.	
1994	1	Loading of outer core fuel subassemblies begins.	
	4	Initial criticality achieved.	
	5	Initial core configuration completed. Criticality tests completed.	
		Reactor physics tests begin (continues until November 15).	
1995	2	Reactor starts for nuclear heating tests.	
	5	Reactor power reaches 10% of the rated power.	
	6	Reactor power reaches 40% of the rated power.	
	8	Initial connection to power grid, power increase tests begin.	
	12	Performance evaluation test during plant trip	
		Secondary Sodium Lead Accident occurs (December 8, 1995).	
		First report on the Accident submitted to STA.	
		First report on Abnormal Occurrence submitted to Fukui Pref., Tsuruga City, etc.	
1996	1		Request from 3 prefectural governors (Fukui, Fukushima, and Niigata) submitted to the Government.
	2	Thermocouple removal work ends.	
	3	Construction of ISI device calibration building completed.	
	4	Broken thermocouple sheath, once missing, recovered.	
	5		STA releases an interim report on cause investigation.
	9	Accident location (piping room) opened to the public.	
	10		STA organizes “Monju Comprehensive Safety Check Team”.
	12	Comprehensive Safety Check begins.	

Table A2-1 Key events in Monju project (4/6)

Year	Month	Monju, JAEA	Government, local government, etc.
1997	1	JAEA officially participates in WANO.	
	2		STA releases a report on the cause investigation results.
			Council on Fast Breeder Reactors begins (until November).
	3		STA rates the Secondary Sodium Leak Accident at INES level 1.
	7		Government issues an order to stop Monju operation for one year.
	9	Activities of Comprehensive Safety Check compiled.	
	11		Council on Fast Breeder Reactors submits the report “Desirable Direction of FBR R&D” to JAEC.
1998	3		STA issues the results of Comprehensive Safety Check.
	5	Construction of International Technology Center Research Building begins in Shiraki.	
	10	Establishment of Japan Nuclear Cycle Development Institute (JNC reorganized from PNC)	
			NSC forms “Monju Safety Confirmation Working Group”.
1999	5	First Tsuruga International Energy Forum held.	
	10	Information Building (MC Square) of the International Technology Center opens in Shiraki.	
2000	3		Fukui District Court rejects appeal by plaintiff for both administrative and civil actions (then, the plaintiff appeals to High Court).
	8		Monju Safety Confirmation Working Group releases a draft report of the cause of the accident and countermeasures.
	12	Request for Prior Consent of Application for Amendment on Reactor Installation Permit for modification of measures against sodium leak submitted to Fukui Pref. and Tsuruga City.	
2001	1		Ministry of Education, Culture, Sports, Science and Technology (MEXT) established, Nuclear and Industry Safety Agency (NISA, present NRA) established in Ministry of Economy, Trade and Industry (METI).
	6	Aquatom facility opens in Tsuruga City.	Fukui Pref. and Tsuruga City accept Application for Amendment on Reactor Installation Permit on the modification work.
		Application for Amendment on Reactor Installation Permit (the 4th amendment: the modification work for the measures against sodium leak, etc.)	
		Plan and status of the Comprehensive Safety Check reported to NISA.	
	7		Fukui Pref. forms “Expert Committee on Monju Safety Investigation”.
2002	12		METI approves the Reactor Installation Permit (4th amendment).
		Application for Approval of Design and Construction Methods on the modification work submitted to METI.	
2003	1		Kanazawa Branch, Nagoya High Court passes judgment to invalidate the Reactor Installation Permit through court procedure of administrative action.
			Government files the Final Appeal to the Supreme Court.
	11		Report from the Expert Committee on Monju Safety Investigation submitted to the Governor of Fukui Pref.
			Governor of Fukui Pref. requests for ensuring safety of Monju.

Table A2-1 Key events in Monju project (5/6)

Year	Month	Monju, JAEA	Government, local government, etc.
2004	1		METI approves the Amendment of the Design and Construction Methods on the modification work.
	5		First meeting of Committee on Energy Research and Development Centralization Program of Fukui Pref. held.
	11	Visitors to Monju numbers 80,000.	
	12		Supreme Court accepts the Final Appeal by the Government.
2005	2	"Response to ensuring the safety of Monju" submitted to the Governor of Fukui Pref.	
			Fukui Pref. and Tsuruga City consent to the Request for Prior Consent of Construction Plans on the modification work for measures against sodium leak.
	3	Preparation of the modification work begins.	
	5		Supreme Court quashes the judgment of High Court on the administrative action (Government won the suit).
	9	The modification work begins.	
	10	Establishment of JAEA by integration of JNC and JAERI	
	11		NISA: "Monju Safety Confirmation Examination Meeting" begins (until the 28th meeting in February 2011).
2006	10	Application for Amendment on Reactor Installation Permit (5th amendment: change in fuel compositions, etc.)	
	12	Modified Parts Confirmation Tests (MKS) begin (end in August 2007).	
2007	6	Sodium charged into SHTS loop C (damaged loop in the Accident).	
	7		The Niigataken Chuetsu-oki Earthquake occurs in 2007.
	8	Plant System Confirmation Tests (PKS) begin.	
2008	2		METI approves the Application for Amendment on Reactor Installation Permit (5th amendment).
2009	7	84 fuel subassemblies refueled.	
	8	Preparation of commissioning begins (until January 2010).	
2010	2		NSC accepts the evaluation results by NISA of Comprehensive Safety Check.
		Proposal made to Fukui Pref. and Tsuruga City to consult on restart of the commissioning under the Safety Agreement.	
	3		"Council on Monju" held among Ministers of MEXT and METI, and the Governor of Fukui Pref.
	4		Fukui Pref. and Tsuruga City: Acceptance of restart of the commissioning (JAEA President receives)
	5	Commissioning restarts (Core Performance Confirmation Tests start, till July).	
	8	IVTM drops down.	
	11	Deformation of IVTM observed with a special in-vessel viewing device.	
2011	2	Water-Steam System Confirmation Tests (MKS) begin.	
	3		The Great East Japan Earthquake and the Accident at Fukushima Daiichi Nuclear Power Station occur.
	6	IVTM withdrawal work	
	10	The MKS interrupted before supplying water to evaporators, and the water-steam system brought back to a storage state.	

Table A2-1 Key events in Monju project (6/6)

Year	Month	Monju, JAEA	Government, local government, etc.
2012	8	IVTM restoration work completed.	
	9		Establishment of Nuclear Regulation Authority (NRA)
	11	Inadequate maintenance management reported to NRA.	
2013	1	Report in response to the Order to Implement Safety Measures submitted to NRA.	
	10	Monju Reform Initiative begins (until May 2015).	
2014	3	Comprehensive Report on Additional Geological Survey of On-site Fracture Zone submitted to NRA.	
	9	Report on Reform of JAEA submitted to MEXT.	
	12	Report on the measures against inadequate maintenance management submitted to NRA. Application for Approval of Amendment on Operational Safety Program also submitted.	
2015	10	Report on Importance Classification of Safety Functions submitted to NRA.	
	11		NRA makes recommendations to MEXT Minister.
	12		MEXT holds Special Committee on the Management of Monju (until May 2016).
2016	12		The 6th meeting of the Council of Ministers for Nuclear Energy issues the fast reactors development policy and Government's policy on dealing with Monju to move to decommissioning.

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国際単位系（SI）

表１．SI 基本単位

基本量	SI 基本単位 名称	記号
長さ	メートル	m
質量	キログラム	kg
時間	秒	s
電流	アンペア	A
熱力学温度	ケルビン	K
物質量	モル	mol
光度	カンデラ	cd

表２．基本単位を用いて表されるSI組立単位の例

組立量	SI 組立単位 名称	記号
面積	平方メートル	m^2
体積	立方メートル	m^3
速さ，速度	メートル毎秒	m/s
加速度	メートル毎秒毎秒	m/s^2
波数	毎メートル	m^{-1}
密度，質量密度	キログラム毎立方メートル	kg/m^3
面積密度	キログラム毎平方メートル	kg/m^2
比体積	立方メートル毎キログラム	m^3/kg
電流密度	アンペア毎平方メートル	A/m^2
磁界の強さ	アンペア毎メートル	A/m
量濃度 (a)，濃度	モル毎立方メートル	mol/m^3
質量濃度	キログラム毎立方メートル	kg/m^3
輝度	カンデラ毎平方メートル	cd/m^2
屈折率 (b)	（数字の） 1	1
比透磁率 (b)	（数字の） 1	1

(a) 量濃度（amount concentration）は臨床化学の分野では物質濃度（substance concentration）ともよばれる。
(b) これらは無次元量あるいは次元１をもつ量であるが、そのことを表す単位記号である数字の１は通常は表記しない。

表３．固有の名称と記号で表されるSI組立単位

組立量	SI 組立単位 名称	記号	他のSI単位による表し方	SI基本単位による表し方
平面角	ラジアン(b)	rad	1 (b)	m/m
立体角	ステラジアン(b)	sr(c)	1 (b)	m^2/m^2
周波数	ヘルツ (d)	Hz		s^{-1}
力	ニュートン	N		$m\ kg\ s^{-2}$
圧力，応力	パスカル	Pa	N/m^2	$m^{-1}\ kg\ s^{-2}$
エネルギー，仕事，熱量	ジュール	J	N m	$m^2\ kg\ s^{-2}$
仕事率，工率，放射束	ワット	W	J/s	$m^2\ kg\ s^{-3}$
電荷，電気量	クーロン	C		s A
電位差（電圧），起電力	ボルト	V	W/A	$m^2\ kg\ s^{-3}\ A^{-1}$
静電容量	ファラド	F	C/V	$m^{-2}\ kg^{-1}\ s^4\ A^2$
電気抵抗	オーム	Ω	V/A	$m^2\ kg\ s^{-3}\ A^{-2}$
コンダクタンス	ジーメンス	S	A/V	$m^{-2}\ kg^{-1}\ s^3\ A^2$
磁束	ウエーバ	Wb	Vs	$m^2\ kg\ s^{-2}\ A^{-1}$
磁束密度	テスラ	T	Wb/m^2	$kg\ s^{-2}\ A^{-1}$
インダクタンス	ヘンリー	H	Wb/A	$m^2\ kg\ s^{-2}\ A^{-2}$
セルシウス温度	セルシウス度(e)	℃		K
光束	ルーメン	lm	cd sr(c)	cd
照度	ルクス	lx	lm/m^2	$m^{-2}\ cd$
放射性核種の放射能 (f)	ベクレル (d)	Bq		s^{-1}
吸収線量, 比エネルギー分与, カーマ	グレイ	Gy	J/kg	$m^2\ s^{-2}$
線量当量, 周辺線量当量, 方向性線量当量, 個人線量当量	シーベルト (g)	Sv	J/kg	$m^2\ s^{-2}$
酸素活性	カタール	kat		$s^{-1}\ mol$

(a)SI接頭語は固有の名称と記号を持つ組立単位と組み合わせても使用できる。しかし接頭語を付した単位はもはやコヒーレントではない。
(b)ラジアンとステラジアンは数字の１に対する単位の特別な名称で、量についての情報をつたえるために使われる。実際には、使用する時には記号rad及びsrが用いられるが、習慣として組立単位としての記号である数字の１は明示されない。
(c)測光学ではステラジアンという名称と記号srを単位の表し方の中に、そのまま維持している。
(d)ヘルツは周期現象についてのみ、ベクレルは放射性核種の統計的過程についてのみ使用される。
(e)セルシウス度はケルビンの特別な名称で、セルシウス温度を表すために使用される。セルシウス度とケルビンの単位の大きさは同一である。したがって、温度差や温度間隔を表す数値はどちらの単位で表しても同じである。
(f)放射性核種の放射能（activity referred to a radionuclide）は、しばしば誤った用語で”radioactivity”と記される。
(g)単位シーベルト（PV,2002,70,205）についてはCIPM勧告2（CI-2002）を参照。

表４．単位の中に固有の名称と記号を含むSI組立単位の例

組立量	SI 組立単位 名称	記号	SI 基本単位による表し方
粘度	パスカル秒	Pa s	$m^{-1}\ kg\ s^{-1}$
力のモーメント	ニュートンメートル	N m	$m^2\ kg\ s^{-2}$
表面張力	ニュートン毎メートル	N/m	$kg\ s^{-2}$
角速度	ラジアン毎秒	rad/s	$m\ m^{-1}\ s^{-1}=s^{-1}$
角加速度	ラジアン毎秒毎秒	rad/s^2	$m\ m^{-1}\ s^{-2}=s^{-2}$
熱流密度，放射照度	ワット毎平方メートル	W/m^2	$kg\ s^{-3}$
熱容量，エントロピー	ジュール毎ケルビン	J/K	$m^2\ kg\ s^{-2}\ K^{-1}$
比熱容量，比エントロピー	ジュール毎キログラム毎ケルビン	J/(kg K)	$m^2\ s^{-2}\ K^{-1}$
比エネルギー	ジュール毎キログラム	J/kg	$m^2\ s^{-2}$
熱伝導率	ワット毎メートル毎ケルビン	W/(m K)	$m\ kg\ s^{-3}\ K^{-1}$
体積エネルギー	ジュール毎立方メートル	J/m^3	$m^{-1}\ kg\ s^{-2}$
電界の強さ	ボルト毎メートル	V/m	$m\ kg\ s^{-3}\ A^{-1}$
電荷密度	クーロン毎立方メートル	C/m^3	$m^{-3}\ s\ A$
表面電荷	クーロン毎平方メートル	C/m^2	$m^{-2}\ s\ A$
電束密度，電気変位	クーロン毎平方メートル	C/m^2	$m^{-2}\ s\ A$
誘電率	ファラド毎メートル	F/m	$m^{-3}\ kg^{-1}\ s^4\ A^2$
透磁率	ヘンリー毎メートル	H/m	$m\ kg\ s^{-2}\ A^{-2}$
モルエネルギー	ジュール毎モル	J/mol	$m^2\ kg\ s^{-2}\ mol^{-1}$
モルエントロピー, モル熱容量	ジュール毎モル毎ケルビン	J/(mol K)	$m^2\ kg\ s^{-2}\ K^{-1}\ mol^{-1}$
照射線量（Ｘ線及びγ線）	クーロン毎キログラム	C/kg	$kg^{-1}\ s\ A$
吸収線量率	グレイ毎秒	Gy/s	$m^2\ s^{-3}$
放射強度	ワット毎ステラジアン	W/sr	$m^4\ m^{-2}\ kg\ s^{-3}=m^2\ kg\ s^{-3}$
放射輝度	ワット毎平方メートル毎ステラジアン	$W/(m^2\ sr)$	$m^2\ m^{-2}\ kg\ s^{-3}=kg\ s^{-3}$
酵素活性濃度	カタール毎立方メートル	kat/m^3	$m^{-3}\ s^{-1}\ mol$

表５．SI 接頭語

乗数	名称	記号	乗数	名称	記号
10^{24}	ヨタ	Y	10^{-1}	デシ	d
10^{21}	ゼタ	Z	10^{-2}	センチ	c
10^{18}	エクサ	E	10^{-3}	ミリ	m
10^{15}	ペタ	P	10^{-6}	マイクロ	μ
10^{12}	テラ	T	10^{-9}	ナノ	n
10^{9}	ギガ	G	10^{-12}	ピコ	p
10^{6}	メガ	M	10^{-15}	フェムト	f
10^{3}	キロ	k	10^{-18}	アト	a
10^{2}	ヘクト	h	10^{-21}	ゼプト	z
10^{1}	デカ	da	10^{-24}	ヨクト	y

表６．SIに属さないが、SIと併用される単位

名称	記号	SI 単位による値
分	min	1 min=60 s
時	h	1 h =60 min=3600 s
日	d	1 d=24 h=86 400 s
度	°	$1°=(\pi/180)$ rad
分	’	$1’=(1/60)°=(\pi/10\ 800)$ rad
秒	”	$1”=(1/60)’=(\pi/648\ 000)$ rad
ヘクタール	ha	$1\ ha=1\ hm^2=10^4 m^2$
リットル	L，l	$1\ L=1\ l=1\ dm^3=10^3 cm^3=10^{-3} m^3$
トン	t	$1\ t=10^3$ kg

表７．SIに属さないが、SIと併用される単位で、SI単位で表される数値が実験的に得られるもの

名称	記号	SI 単位で表される数値
電子ボルト	eV	$1\ eV=1.602\ 176\ 53(14)\times10^{-19}$J
ダルトン	Da	$1\ Da=1.660\ 538\ 86(28)\times10^{-27}$kg
統一原子質量単位	u	1 u=1 Da
天文単位	ua	$1\ ua=1.495\ 978\ 706\ 91(6)\times10^{11}$m

表８．SIに属さないが、SIと併用されるその他の単位

名称	記号	SI 単位で表される数値
バール	bar	$1\ bar=0.1MPa=100\ kPa=10^5$Pa
水銀柱ミリメートル	mmHg	1 mmHg≈133.322Pa
オングストローム	Å	$1\ Å=0.1nm=100pm=10^{-10}$m
海里	M	1 M=1852m
バーン	b	$1\ b=100fm^2=(10^{-12}cm)^2=10^{-28}m^2$
ノット	kn	1 kn=(1852/3600)m/s
ネーパ	Np	SI単位との数値的な関係は、対数量の定義に依存。
ベル	B	
デシベル	dB	

表９．固有の名称をもつCGS組立単位

名称	記号	SI 単位で表される数値
エルグ	erg	$1\ erg=10^{-7}$ J
ダイン	dyn	$1\ dyn=10^{-5}$N
ポアズ	P	$1\ P=1\ dyn\ s\ cm^{-2}=0.1Pa\ s$
ストークス	St	$1\ St=1cm^2\ s^{-1}=10^{-4}m^2\ s^{-1}$
スチルブ	sb	$1\ sb=1cd\ cm^{-2}=10^4cd\ m^{-2}$
フォト	ph	$1\ ph=1cd\ sr\ cm^{-2}=10^4$lx
ガル	Gal	$1\ Gal=1cm\ s^{-2}=10^{-2}ms^{-2}$
マクスウエル	Mx	$1\ Mx=1G\ cm^2=10^{-8}$Wb
ガウス	G	$1\ G=1Mx\ cm^{-2}=10^{-4}$T
エルステッド (a)	Oe	$1\ Oe \triangleq (10^3/4\pi)A\ m^{-1}$

(a) ３元系のCGS単位系とSIでは直接比較できないため、等号「 ≙ 」は対応関係を示すものである。

表10．SIに属さないその他の単位の例

名称	記号	SI 単位で表される数値
キュリー	Ci	$1\ Ci=3.7\times10^{10}$Bq
レントゲン	R	$1\ R=2.58\times10^{-4}$C/kg
ラド	rad	$1\ rad=1cGy=10^{-2}$Gy
レム	rem	$1\ rem=1\ cSv=10^{-2}$Sv
ガンマ	γ	$1\ \gamma=1\ nT=10^{-9}$T
フェルミ		1 フェルミ$=1\ fm=10^{-15}$m
メートル系カラット		1 メートル系カラット$= 0.2\ g = 2\times10^{-4}$kg
トル	Torr	1 Torr = (101 325/760) Pa
標準大気圧	atm	1 atm = 101 325 Pa
カロリー	cal	1 cal=4.1858J（「15℃」カロリー），4.1868J（「IT」カロリー），4.184J（「熱化学」カロリー）
ミクロン	μ	$1\ \mu=1\mu m=10^{-6}$m

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