An Introduction of MONJU

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TSURUGA Headquarter Office, JNC
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1. FBR Development Program in Japan

Development of FBR has been promoted since about the middle of 1960’s as a national project centered around PNC (Power Reactor and Nuclear Fuel Development Corporation). The Organization was changed in 1998 to JNC (Japan Nuclear Cycle Development Institute).
1.1 History of FBR Development

Oct. 1967  Establishment of Power Reactor and Nuclear Fuel Development Corporation (predecessor of JNC)
Feb. 1968  Start of Prototype Fast Breeder Reactor (MONJU) Design Study
Apr. 1970  Shiraki district of Turuga City selected as a Candidate Site
Apr. 1977  First Criticality of Experimental Fast Reactor (JOYO)
Oct. 1985  Start of MONJU Site Works (Start of Excavation)
Apr. 1994  First Criticality of MONJU
Aug. 1995  First Connection to the Grid
Dec. 1995  Sodium Leakage Incident at MONJU
Oct. 1998  Establishment of Japan Nuclear Cycle Development Institute
Dec. 2002  Decision of merger of JNC and Japan Atomic Energy Research Institute (JAERI)

出典: サイクル機構ホームページ (英文 Historyより抜粋)
<table>
<thead>
<tr>
<th>Year</th>
<th>Event</th>
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<tr>
<td>1970</td>
<td>Start of Construction</td>
</tr>
<tr>
<td>1980</td>
<td>Design</td>
</tr>
<tr>
<td>1985</td>
<td>Construction</td>
</tr>
<tr>
<td>1990</td>
<td>Pre-operational test</td>
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<tr>
<td>1995</td>
<td>Initial criticality</td>
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<tr>
<td>2000</td>
<td>Initial generator synchronization</td>
</tr>
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- Safety licensing examination
- Approval of detailed design and construction procedure
- Sodium leak
- Construction Shutdown

Out: 進学英文OHP-Current Status and Future Outlook of Monjuより
1.2 Long-Term Program for Research, Development and Utilization of Nuclear Energy

The Atomic Energy Commission of Japan (AEC), which was set up under the Atomic Energy Basic Law, is charged with the important tasks of planning, deliberations, and decision-making necessary to secure peaceful utilization of nuclear energy and to ensure that Japan's research, development and use of nuclear energy are moved ahead in a coherent manner. Since the first Long-Term Program for Research, Development and Utilization of Nuclear Energy (Long-Term Program) in 1956, the Atomic Energy Commission has formulated a total of nine Long-Term Programs, one approximately every five years. From the infancy of the use of nuclear energy, these programs have played an important role in the systematic implementation of research, development and use of nuclear energy. The latest Long-Term Program was issued in 2000.

In the Long-Term Program, FBR Development Program and related MONJU Project are described as follows.
It is, therefore, appropriate to basically reprocess spend fuel and make effective use of plutonium, uranium and other elements, while securing safety and nuclear non-proliferation. Taking economic efficiency into account, Japan should make the reprocessing of spent fuel and the use of recovered plutonium and uranium its basic policy, considering the geographical and resource conditions of the country.

Furthermore, fast breeder reactor and related nuclear fuel cycle technologies (hereinafter referred to as "FBR cycle technologies") can significantly increase the efficiency of uranium utilization, and, when they are commercialized in the future, can make it possible to continue using nuclear power for several hundred years with uranium resources currently known to be both technically and economically available for use, and can reduce the long-term radioactivity of high-level radioactive waste, decreasing the environmental load. In order, therefore, to prepare for an uncertain future and secure promising energy supply options, an important strategy is to devote steady efforts to the development of these FBR cycle technologies.
Plutonium is the most important

Uranium resources are the same as fossil

Plutonium can supply sustainable energy
(Uranium can be used for more than thousand years)

Proven reserves of energy resources in the world

- Petroleum: 1.046 trillion Barrels
- Natural gas: 984.2 billion tons
- Coal
- Uranium

1,046 trillion Barrels
984.2 billion tons

U235: 0.7%
U238: 99.3%

出典: 英語版汛用H151104.ppt (Current Status of Monju)より
1.4 The Significance and Role of the Prototype Fast Breeder Reactor MONJU
(the excerpt from 2000 Long-Term Program)

Because attaining MONJU's specified objective of demonstrating reliability as an operational power plant and establishing sodium-handling techniques will be the basis for evaluation in comparison with other options, it is particularly important to give priority in technological development hereafter to the attainment of that objective.

MONJU is a valuable facility internationally as well, for the sake of FBR development in the future. With this in mind, arrangements will be made to develop "MONJU" and its auxiliary facilities into an international cooperation base open to researchers from Japan and abroad, with the results of their research and development efforts widely shared internationally.

From a long-term point of view, it is important to effectively use "MONJU" as an irradiation bed for generating fast neutrons equivalent to those expected under actual operating conditions. In conjunction with this, fuel production and reprocessing, including the demonstration of elemental technologies and other research and development results, toward the commercialization of FBR technology, will also be pursued.

Another important task of MONJU is to accumulate extensive data on the burnup of minor actinides and the transmutation of long-lived fission products.
2. Outline of MONJU

MONJU is located at the tip of Tsuruga Peninsula near Tsuruga City, Fukui Prefecture, facing the Sea of Japan. It is about 41km SSW of Fukui City, the seat of the prefectural government. Shiraki, the nearest village, is 1.4km SW of the site.

MONJU is a fast breeder reactor (FBR), cooled by sodium. It has been developed to confirm its function as a power-generating FBR plant and technical feasibility for future commercial plant. Thus, the data obtained through operation will be utilized for the development of a future FBR.

In the design of this plant particular attention has been given to safety and to achieving reliable operation. A main cooling system consists of three loops. Although a thermal output is 714 MW, an electric output is 280 MW owing to its high thermal efficiency (40%).
Monju and Fugen

Tsuruga-Bay
Route 27
Monju
Shiraki
Fugen
Mihama P.S (KEPCO)
Tsuruga City
Tsuruga Head Office
Mihama Town
Tsuruga Port
Wakasa-Bay
NEAT
Aquatom
Fugen
Monju
International Cooperation and Technology Development Center

出典 英語版汎用H151104.ppt (Current Status of Monju)より
2.1 Safety Design of FBRs

Safety design of FBRs should use (1) the Safety Design Review Criteria for LWR Power Station, and (2) the Seismic Design Review Criteria for LWR Power Station, and should also consider the following items specific to FBRs in order to minimize the occurrence of faults and anomalies of systems and components, and in case of accidents, to prevent the accidental progression and release of radioactivity to the environment.

1. Reactor Core

The FBR has a fast neutron spectrum for breeding purposes and has a high neutron flux, high power density and operates under high burnup conditions. Hence, in its design it is necessary to consider the effect of high fluence on materials.

Regarding reactivity, the sodium void coefficient of reactivity may be positive in the central region of the core, while the excess reactivity and reactivity change due to fuel burnup may be small. These points should be considered in the design.

2. Fuels

Since fuel elements are used in high temperature sodium and under high burnup conditions, it is necessary to consider effects of creep resulting from high internal pressure and swelling on cladding tubes.

Regarding the neutronic and thermal characteristics of the reactor core, the design should consider possible deformation of the fuel assemblies and should prevent possible blockage of the coolant channels.
3. Sodium

The coolant system can be designed for use under low pressure and highly subcooled conditions with excellent heat transfer characteristics, since the sodium coolant possesses a high boiling temperature. However sodium is chemically active, and therefore design measures should be introduced to inert the cover-gas region above the sodium levels and to provide for sodium fire prevention. Compatibility between sodium and structural materials (corrosion and mass transfer effects), solidification, opacity and activation of sodium should be considered in the design.

4. Sodium Void

To avoid reactivity addition due to sodium voiding, the design should include measures to suppress sodium boiling and cover-gas entrainment.

5. Reactor Shutdown System

The reactor shutdown system consists of control rods. Multiple, independent shutdown systems of high reliability should be required to rapidly shutdown the reactor.
6. Reactor Coolant Boundary and Cover-Gas Boundary, etc.

The reactor coolant boundary should be designed to minimize the occurrence of coolant leakage or failure of the boundary. Against coolant leakage, early and reliable sodium detection mechanisms should be included in the design. Occurrence of leakage or failure of the cover-gas boundaries such as in the reactor cover-gas region should be minimized in the design.

For determining methods for inspection and system design for in-service inspection of these boundaries, the fact that sodium is used as the coolant should be considered.

7. Intermediate Heat Transport System

Design of the Intermediate Heat Transport System (IHTS) should be such as to avoid coolant leakage from the Primary Heat Transport System (PHTS) to the IHTS, and in the event of leakage from the water/steam side to the IHTS, the safety of the plant should be ensured.

8. Decay Heat Removal

The decay heat removal system should be designed such that, in the case of loss of flow or leakage of coolant, the system possesses cooling capability to dissipate the decay heat from the reactor.
9. Containment
The containment vessel should be designed to limit the release of radioactive materials to the environment under postulated accident conditions.

10. Elevated Temperature Design
Creep effects on the structural materials should be considered in the design of components used in high temperature sodium conditions. In the case of austenitic stainless steel, the fact that the coefficient of thermal expansion of austenitic steel is larger than that of ferritic steel, etc., should be considered. The effects on structural materials of the low heat capacity of sodium (such as greater temperature and rate change) should be considered. Thermal stresses of the structural materials under both static and transient design conditions should be considered.

11. Seismic Design
For the seismic design of components, piping, etc., the differences from LWR structures (e.g., low pressure, thin wall, elevated temperature design) should be considered. Seismic design classifications should be made, taking into full consideration the FBR design features.
2.2 MONJU Plant Structure

- Reactor Core
- Reactor Vessel (RV)
- Control Rod Drive Mechanism (CRDM)
- Fuel Handling Machine (FHM)
- Shield Plug (SP)
- Guard Vessel (GV)
- Intermediate Heat Exchanger (IHX)
- Primary Sodium Pump
- Primary Sodium Pipe
- Secondary Sodium Pipe
- Reactor Containment Vessel (CV)
- Ex-Vessel Fuel Transfer Machine (EVTM)
- Secondary Sodium Pump
- Evaporator (EV)
- Super Heater (SH)
- Auxiliary Air Cooler (AC)
- Sodium-water Reaction-Products Storage Tank
- Ex-Vessel Fuel Storage Tank (EVST)
- Water-Steam System Pipe
- Steam Turbine
- Condenser
- Generator
- Transformer
- Polar Crane
- Vent Stack

出典: JNC英文パンフレットPrototype Fast Breeder Reactor Monju
出典:英語版汎用H151104.pptから（Current Status of Monju）より
**Reactor & Primary Loops**

**The Reactor**

The figure shows the major components of the reactor. Included in the figure is the Fuel Handling Machine which is used to change fuel when the reactor is shut down but is removed during operation. At the centre of the vessel is the reactor core where the nuclear reaction takes place. Sodium coolant is pumped into the reactor under the core (see sodium inlet pipe right) at a temperature of 397 °C and flows upwards through the core leaving by the outlet pipe (left) at a temperature of 529 °C. There are three loops which supply and remove sodium; these are called the Primary Loops.

The reactor vessel and the sodium pipes are surrounded by a Guard Vessel; in the unlikely event that the Main Vessel were to leak, the Guard Vessel would ensure that the sodium level does not fall below what is required to cool the reactor core.

**The Above Core Structure**

Above the core is a structure which holds temperature sensors and flow gauges to monitor the condition of each fuel assembly. This structure also contains the Control Rod Drive Mechanisms which are used to withdraw and insert the Control Rods. When the reactor is in the shut down condition, a circular plug set into the above-core structure can be rotated; this holds the Fuel Handling Machine.
The design of primary sodium loops is made more complicated because allowance has to be made for the dimensional changes which take place due to thermal expansion as the reactor is taken from the shut-down "cold" state up to full power. For this reason there are additional pipe bends to accommodate the expansion. The picture is from a computer simulation of the reactor and primary loops.
The Reactor Core

The reactor core is built up of hexagonal assemblies surrounded by a strong support frame. Below the core is a structure called the diagrid (shown red) which is designed to hold the assemblies in place and regulate the correct amount of sodium flow to each assembly. The volume of the core in which the nuclear reaction takes place is very small, about 1.8m in diameter by 93cm in height, or roughly the volume of a Mini car. Yet this small volume produces 714MW of heat.

Although externally they are all identical, the hexagonal assemblies are of three different types: core fuel, blanket fuel, and neutron shield. The figure above shows a top-view map of the MONJU core. The positions at the centre, colored red, are core fuel assemblies containing a mixture of the fuel material Plutonium, and depleted Uranium; the positions colored yellow are also core fuel assemblies but these contain slightly more plutonium than those at the centre. The black positions on the core map are those of control rods. Surrounding the fuel assemblies are the blanket fuel assemblies (colored blue) that contain only depleted Uranium in which plutonium is produced. At the edge of the core are neutron shield assemblies - not shown - which contain no nuclear material, only steel; their function is to reflect escaping neutrons back into the core.
Fuel: The core fuel assemblies consist of a hexagonal steel tube holding 169 fuel pins each about the diameter of a pencil (6.5mm) but 2.8m in length. The cladding tube of fuel pins are also made of steel. At the vertical-axis centre of the pin there is the 93cm fuel section containing a mixture of principally depleted Uranium and Plutonium both in the form of oxide pressed into small cylindrical pellets. Above and below the fuel section is a depleted Uranium only section - often called the axial blanket - which absorbs escaping neutrons to form Plutonium-239. At the top of the pin is a volume which is left empty in order to accommodate the gases which are released from the fuel during the nuclear reaction. The fuel pins are sealed; it is preferable that the radioactive materials should not leak into the sodium coolant as this would contaminate the sodium making maintenance of all the Primary Loop components, pumps etc., more difficult. As a precaution the reactor is equipped with a system which can detect and locate a single leaking fuel pin in the core. The fuel assembly would then be removed.

The radial blanket rods are similar to the fuel rods in design except that the pins contain only depleted Uranium are a little 'fatter'. The neutron shield assemblies contain solid steel rods.

Control Rods: There is a total of 19 control rods; these are of three distinct types: Fine Control Rods (3), Coarse Control Rods (10), Back-up Control Rods (6). The BCR are used only to start up and shut down the reactor; during reactor operation they are fully withdrawn from the core. The FCR and BCR are used to control the power output of the reactor and to compensate for the gradual loss of reactivity of the core from the beginning to the end of an operation cycle. The Control Rods are held on electromagnets and so if a rapid shut-down of the reactor is required the electric power supply to the electromagnets is cut and the rods drop under gravity into the core with additional force of inert gas or spring coil. Their drive mechanisms are something different in design, because each set of rods at MONJU was designed and built by a different manufacturer.
Primary Sodium Pump

The primary sodium pump is a mechanical pump driven by an electric motor. There is a pump of this type on each of the three primary loops. At its normal full speed of 837rpm it can pump approximately 5100 tonnes of sodium per hour (about 1.4 tonnes per second). The pump is equipped with two motors: a Main Motor which is used during normal operation and a low power back-up Pony Motor which is used only under shut-down conditions.
Intermediate Heat Exchanger

The Intermediate Heat Exchanger (IHX), of which there is one for each of the three primary loops, transfers heat from the primary sodium cooling loop to the secondary sodium cooling loop. Primary sodium flows around the outside of the tubes and secondary sodium flows inside.
Auxiliary Building & Secondary Loops

The Auxiliary Building houses the secondary sodium loops. There are three secondary sodium loops. In each loop, heat is carried by the pumped sodium flow from the Intermediate Heat Exchanger to the two-stage boiler (Evaporator and Superheater). The main reason for having a secondary sodium loop system is to prevent the reactor being affected in the event of a sodium-water reaction in the Evaporator or Superheater.

Secondary Sodium Pump

The Secondary Sodium Pump circulates the sodium around the secondary sodium loop. It is an electrically driven mechanical pump which, at its full speed of 1100 rpm, pumps around 1 ton of sodium per second. It is positioned in the part of the loop where the sodium is coldest (325 °C) as this minimizes the volume of sodium which it must pump. Like the Primary Sodium Pump, the Secondary Sodium Pump is equipped with a "pony motor", a smaller electric motor which is capable of circulating the sodium around the loop at a reduced rate in the event of the deliberate shut-down or failure of the main pump motor. The drawing does not show the motors.
Evaporator

The production of steam to drive the turbine is in two stages. The Evaporator is the first stage. Here water arrives at 240 °C from the feed-water heating system and is converted to steam which leaves at a temperature of 369 °C. The Evaporator consists of a cylindrical vessel with a helical tube bundle inside. Sodium flows through the vessel and water flows through the helical tubes. The sodium reaches the Evaporator after having already flowed through the Superheater; it is at a temperature of 469 °C on entering the Evaporator and 325 °C on leaving it. The sodium then flows back to the Secondary Pump and thence the Intermediate Heat Exchanger.
Superheater

The Superheater performs the second stage of the production of steam for use in the turbine. It receives steam from the Evaporator and heats it to a temperature of 487 °C. The design of the Superheater is broadly similar to that of the Evaporator; a cylindrical vessel containing a bundle of helical tubes. The sodium flows through the vessel outside the tubes and the steam inside the tubes. The Superheater receives hot sodium directly from the Intermediate Heat Exchanger at a temperature of 505 °C; after leaving the Superheater it flows to the Evaporator.
Auxiliary Cooling System

The chain reaction in a nuclear reactor can be stopped in a fraction of a second by dropping the control rods. However, even after the chain reaction has stopped the reactor continues to produce some heat. A part of this is heat stored by the fuel and metal structures in the reactor but most is due to the radioactive decay of the atoms which were produced by fission. The function of the Auxiliary Cooling System is to remove this heat.

The system is essentially very simple. A small branch of the secondary loop bypasses the Evaporator and Superheater and flows through an air-cooled heat exchanger. A blower and dampers are available to control how much heat is lost by the sodium. A small volume of sodium flows through this system all the time. Only one of the three ACS's at MONJU is required to remove all the decay heat from the reactor.
Reaction Tanks

In the Superheater and Evaporator sodium and water are separated only by the wall of the heat exchanger tubes. For this reason these units are equipped with sophisticated monitoring to detect the early stages of a leak through the tube wall. In the event of a large reaction causing a rapid pressure increase special rupture membranes are designed to break allowing the reaction products into the Reaction Tanks. These tanks are located on the roof of the Auxiliary Building above the steam generators.

Overflow and Drain Tanks

At the base of the Auxiliary Building are five rooms which each contain a sodium storage tank. Two of these tanks are Overflow Tanks used to adjust for the volume of sodium in the loop under varying temperature conditions. The other three tanks are Drain Tanks which are used mainly to store the sodium during maintenance on a loop. In the event of a leak of sodium from a secondary loop, the drain valves are opened and the sodium drains into these tanks thereby stopping the leak.

Improvements to the plant are currently being examined that would increase the speed at which the sodium can be drained in order to stop a leak more quickly.
The Maintenance Building is used for the storage, cleaning, maintenance and repair of all major equipment from the sodium circuits and fuel handling system. The track of the Ex-Vessel Fuel Transfer Machine runs across the end of the Maintenance Building. Along its route there are a series of special fuel cleaning and canning facilities set into the floor between its rails.

The Fuel Handling System

Reception of New Fuel → Ex-Vessel Fuel Storage Tank → Ex-Vessel Fuel Transfer Machine → In-Vessel Fuel Transfer Machine → Fuel Handling Machine

Spent Fuel Cleaning Facility → Spent Fuel Canning Facility → Spent Fuel Storage Pond

The MONJU plant must be shut down twice each year in order to refuel the reactor. The refuelling is carried out entirely under sodium as the sodium coolant continues to remove residual heat from the spent fuel assemblies. During a refuelling campaign approximately one fifth of the fuel assemblies in the core will be replaced, one at a time, by new fuel.

This figure shows schematically the route taken by each fuel assembly at MONJU.
New Fuel Storage Cell

Fuel is delivered to the plant in transport casks in which the fuel assembly is carried horizontally. In the New Fuel Storage Cell the fuel is removed from the transport cask and lifted to a vertical orientation. It is then carried to a storage position set into the concrete floor. A steel plug is used to close the storage position. The position is then locked and a security seal is fitted.

Ex-Vessel Fuel Storage Tank

The Ex-Vessel Fuel Storage Tank (EVST) is used for the temporary storage of new fuel before loading to the core and spent fuel from the core. In the EVST fuel is stored in a sodium filled tank. The spent fuel must be stored in this way as the decay of radioactive isotopes causes it to continue to give off heat; typically a spent fuel assembly will spend 18 months in the EVST as it cools.

The EVST can hold 250 fuel assemblies. The fuel assemblies are held in a rotatable rack with 6 concentric rows of fuel positions; the rack is driven by an electric motor above the vessel. There is no distinction between positions for new fuel and spent fuel. The vessel is sealed with a top plug and, above the level of the sodium, is filled with inert argon gas. The main tank is surrounded by a second tank which functions as a leak jacket.
The Ex-Vessel Fuel Transfer Machine (EVTM) is a self-propelled vehicle running on rails and weighing approximately 200 tons. Its function is to carry fuel between the reactor and the new and spent fuel facilities. Like the rest of the fuel handling route it operates entirely under automatic computer control.

The photo shows the EVTM in the Reactor Hall, the top of the reactor is at the bottom right. The track of the EVTM passes through the Containment Vessel door into the Maintenance Building. During reactor operation the track is dismantled and the door is bolted closed.
The In-Reactor Fuel Handling

The in-reactor fuel handling involves the use of two remotely operated machines. These are:

(1) The Fuel Handling Machine

Set in the roof of the reactor vessel there is a rotatable plug to which the Fuel Handling Machine (FHM) is attached with its upper half in the air atmosphere of the Containment Building and its lower half, passing through the argon cover gas, submerged in sodium. By rotating the plug over the core and rotating the FHM on its own axis any position in the core can be accessed. The fuel handling machine consists of a long vertical body with a pantograph arm at its base. The pantograph arm holds an expanding-retracting gripper mechanism which locks into the top of the fuel assembly to be removed and draws it vertically out of the core. The spent fuel is transferred by the machine to a special position at the edge of the core where it is placed in a pot for removal from the reactor. A new fuel assembly is picked up at the same location and the procedure is reversed to load it into the core. The control rods, which are fully inserted into the core and detached from their raising devices during refuelling, are routinely changed in the same way. When the refuelling is complete the FHM is removed from the reactor for cleaning and maintenance.

(2) The In-Vessel Fuel Transfer Machine

This is a much simpler device than the FHM. Its purpose is to exchange the pot containing the spent fuel assembly for an identical pot containing a new fuel assembly.
Fuel Cleaning Facility

When the fuel assembly is removed from sodium small quantities remain. To remove these the fuel assemblies are first washed using a mixture of argon gas and steam and then washed again with water.

Fuel Canning Facility

The cleaned subassemblies are sealed into a steel can before being transferred to the Spent Fuel Pond. The cans are filled with water. Like other aspects of the fuel route this operation is carried under automatic computer control.

Spent Fuel Pond

After having been washed and sealed into a can the final stage of the fuel handling system at MONJU is to store the spent fuel cans under water in the Spent Fuel Pond. From the Fuel Canning facility sealed cans are carried on an under-water railway - the Transfer Car - directly into the pond. The depth of the water tank is 12.5 m, so that all handling operations can be carried out under the water surface. The rack in this pond has a capacity of 1410 fuel cans.
The Water-Steam System

The Turbine Hall is on the top level of this building which also houses the main parts of the water-steam system excluding the actual steam generators.

Adjacent to the turbine are the Deaerator and two High Pressure Feed-Water Heaters, directly below the turbine is the condenser with the three Low Pressure Feed-Water Heaters. The function of the feed-water heaters is to pre-heat the water from the condenser before it reaches the steam generators. Two steam driven pumps are used to circulate the feed-water in normal operation; for starting up the plant when no steam is available there is an electric pump. At the lowest level of the building the blue pipe leading from the right of the building is the sea-water cooling pipe for the condenser.

The Turbine / Alternator

The figure shows a cutaway drawing of the turbine.
MONJU supplies electricity to the grid at a grid frequency of 60 Hz. The turbine is therefore designed to rotate at 3,600 rpm.
2.3 Examples of Safety Measures

(1) Measures against primary sodium leakage
Since primary sodium contains traces of radioactive materials, the release of these materials accompanied by sodium fire must be prevented. Accordingly, the primary coolant system chambers where sodium piping and components are installed are filled with nitrogen gas to prevent sodium from burning if primary sodium leaks. (Fig. 2.3-1)

(2) Decay heat removal measures against large primary sodium leakage.
If the size of the primary sodium leakage were such that the sodium level in the RV comes down below the exit nozzle, then all of the three sodium flow passage would be cut off and the decay heat from the core could not be removed. The MONJU primary piping is installed at high level, higher than the RV exit nozzle. Those piping and components positioned at lower than the exit nozzle are contained in the “guard vessel (GV)” in which the leaked primary sodium is retained. By designing the GV properly, the level of sodium in the RV never comes down below the exit nozzle in the event of a large primary sodium leakage and thus the decay heat removal can be ensured. (Fig. 2.3-2)

(3) Measures against secondary sodium leakage
Since the secondary cooling system chambers are in the air atmosphere, sodium burns if secondary sodium leaks. In order to mitigate the effects of the sodium-fire, it is designed to quickly detect the leak and stop by draining the sodium into the tank. The concrete floor of the secondary cooling system chamber is covered by the slightly inclined steel plate called a “floor liner”. In the event of a large leak, the leaked sodium flows down the floor liner while burning and through the vertical piping by which the leaked sodium is guided into the pit in the basement where the sodium-fire is put out by suffocation. (Fig. 2.3-1)
(4) Measures against sodium-water reaction in the steam generator system

If the heat-transfer tubes of the SG fail, high temperature and pressure water (or steam) leaks out and reacts explosively with sodium generating hydrogen gas. The explosion (in this case the propagation of pressure-wave and sound-wave) is caused by the volume expansion of the hydrogen gas.
The size of the water leak and the resulting effects of sodium-water reaction are described in Fig. 2.3-3.
The chemical equation of the reaction is as follows:

\[
\begin{align*}
\text{Na} + \text{H}_2\text{O} & \rightarrow \text{NaOH} + \frac{1}{2}\text{H}_2 + \text{Heat}(-145\text{J/mol}) \\
2\text{Na} + \text{H}_2\text{O} & \rightarrow \text{Na}_2\text{O} + \text{H}_2 + \text{Heat}(-145\text{J/mol})
\end{align*}
\]

The hydrogen concentration and pressure in the MONJU SG system are constantly monitored. If the water-leak is detected by hydrogen meters and/or pressure gauges, the reactor is shutdown and the sodium-water reaction is to be stopped by blowing water (or steam) out of the system.
In the event of a large reaction causing a rapid pressure increase, special rupture membranes (or rupture disks) installed in the secondary cooling system are designed to break allowing high pressure as well as reaction products into the Reaction Tank (or sodium-water reaction products storage tank). Fig. 2.3-4, Fig. 2.3-5.

The main reason for having a secondary cooling system is to prevent the reactor being affected in the event of a sodium-water reaction in the Evaporator or Superheater.
Fig. 2.3-1 Safety Measures Against Sodium Leakage

Primary Cooling System

Secondary Cooling System
Fig.2.3-2 Preservation of Core Coolability by the GV

Example of sodium, leakage at RV inlet pipe
<table>
<thead>
<tr>
<th>Leak size</th>
<th>Main Effects of Sodium-Water Reaction</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Micro-leak</strong></td>
<td>The leak-jets (i.e., reaction jets or jet flows as a result of a sodium-water reaction) does not reach the adjacent heat exchanger tubes, and thus no effect on tubes.</td>
</tr>
<tr>
<td>&lt; 0.1g/sec</td>
<td></td>
</tr>
<tr>
<td><strong>Small-leak</strong></td>
<td>The leak-jets hit the adjacent tubes at high speed, causing the marked “wastage phenomenon” (i.e., corroding and thinning of the tube wall)</td>
</tr>
<tr>
<td>0.1 〜 10g/sec</td>
<td></td>
</tr>
<tr>
<td><strong>Medium-leak</strong></td>
<td>The “wastage phenomenon” is expected to be milder than the case of small-leak. In the several kg/sec leak, beware of the “high temperature rupture” type failure (i.e., adjacent heat exchanger tubes deteriorate the mechanical strength due to the continued heating by the high temperature leak-jets on the tube wall, resulting in the eventual burst)</td>
</tr>
<tr>
<td>10g 〜 several kg/sec</td>
<td></td>
</tr>
<tr>
<td><strong>Large-leak</strong></td>
<td>A large quantity of hydrogen gas generated makes the secondary system pressure increase rapidly and propagate the initial spike pressure within milliseconds. Then the more gradual pressure increase (quasi-steady pressure) takes over due to the accumulation and the established flow of hydrogen gas.</td>
</tr>
<tr>
<td>&gt; several kg/sec</td>
<td></td>
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</tbody>
</table>

Fig.2.3-3 The size of the water leak and the resulting effects of sodium-water reaction
Fig. 2.3-4 Safety Measures Against Sodium-Water Reaction
Fig.2.3-5 Sodium-Water Reaction-Products Storage Tank
3. Future Program in MONJU

Long-Term R & D Program for Monju

Examination of advanced reactor core
- Development of advanced fuel and core.
  - Longer cycle length, high burn up

R & D for irradiation reactor
- Improvement of irradiation capability

Horizontal View of the Advanced reactor core

- Inner core
- Outer core
- Neutron shield
- Irradiation assembly
- Control Rod

- MA, FP burn up
- Irradiation space 6 ~ 18 assemblies

- Max. linear power 450 ~ 480 W/cm
- Cycle Length 12 months

- Average burn up 150 [GWd/t]

- Low high-burn up Advanced Core ( )
- Advanced Core ( )

出典 英文OHP-Current Status and Future Outlook of Monjuより
Program toward Optimal Commercialized FBR Concept

Feasibility Study on Commercialized FBR Cycle System

- Demonstration of its reliability as a power plant
- Gaining Experience of Sodium-handling Techniques
- Improvement of design methods
- Best operation and maintenance method
- Use as a fast-neutron irradiation test-bed
- Establishment of optimal commercialized FBR Cycle technology

Around 2015

Pre-operational tests of MONJU

Achievement of higher reliability commercialization

Program for optimal commercialization Concept

出典英文OHP-Current Status and Future Outlook of Monjuより
As a centre of research and development activities for FBR cycle technology, MONJU should:

- Perform safety upgrades, and re-start operation!
- Establish FBR plant technology as an operation power plant and sodium-handling techniques
- Become international cooperation base for FBR R&D, by promoting the joint ownership of valuable data generated by Monju, and joint research with overseas countries
- Become an international irradiation facility for fast neutron irradiation
Basic Knowledge of Sodium

Makoto Sawada

September 28, 2004

International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
Presentation Contents

1: General Items of Sodium
2: Favorable Properties as coolant
3: Chemical Properties of Sodium
4: Physical Properties of Sodium
5: Sodium Combustion & Sodium Fire Extinguishment
6: Influence on Human Body & First Aid
1: General Items of Sodium

Handling as a Dangerous Product

- Sodium is regulated as a dangerous product by the fire protection law in Japan.
- Sodium of the quantity over 10kg must be handled in handling facilities permitted by the law.
- Even the case in which handling sodium amount is less than 10kg, the report to fire station is necessary when sodium of the quantity over 2kg is handled.
- In the case of handling sodium at the handling facility, the superintendent who holds the dangerous product handling license must control the work.
- Besides, in the case of carry out handling sodium at the Monju site, the workers must get a certification of completion of the sodium handling license course among our sodium training courses.

Alkali Metal

- Sodium is an element which atomic number is 11 and mass number is 23.
- It is an alkali metal which belongs to 1A group contained such as Li, K, Cs, etc.
- Sodium exists for the sixth among existed elements on earth naturally.
  (Oxygen:49.4%, Silicon:25.8%, Aluminum: 7.6%, Iron:4.7%, Calcium:3.4%, Sodium:2.6%)
- Sodium as metal do not exist independently on earth. It is combined as rock minerals or as salt in the sea (Nacl).
<table>
<thead>
<tr>
<th>Periodic Table of Elements</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sodium (Na)</td>
</tr>
<tr>
<td>23\text{_{11}}Na</td>
</tr>
</tbody>
</table>

【Periodic Table of Elements】
出典：不思議とれるナトリウム、P4、日本原子力文化振興財団
Sodium is a soft silvery metal and is solid at room temperature.

- Sodium can be easily cut with a knife at that temperature. It melts at 97.8 °C and is a fluid as water under inert gas.
- Also, sodium has low density slightly lighter than water even at room temperature.
- Sodium is a very reactive product and reacts with many kinds of other elements.

*Feature of Sodium*

出典：不思議とされるナトリウム、P7-P10、日本原子力文化振興財団
As the manufacturing method of sodium, there are two ways: electrolyze molten caustic soda (NaOH) and molten salt directly. The later method is the general in recently.

Manufacturing of Sodium

【Manufacturing Facility in France】
It is possible that the nuclear fission by the combination of plutonium and fast neutron generates many neutrons compared with the case of nuclear fission of the combination by thermal neutron and uranium.

Sodium does not reduce so much the neutron energy, because its mass is heavier than 23 times compare with the mass of neutron.

Also, sodium has a good property as coolant for FBR, because its neutron capture cross section at the high energy region is small.
Average Neutron Energy at Monju $\approx$ 120keV

Produced Neutron at LWR $\approx 2n$

Produced Neutron at FBR $\approx 3n$

Note: $\eta \geq 2 +$ LOSSES FOR BREEDING

出典: FBR広報素材資料 2版・6-19増殖、科学技術庁、平成2年3月
The absorption cross section for fast neutron is small.
High Performance on Heat Transfer

It is possible that sodium carries out the nuclear reactor operation in which thermal efficiency is high, because sodium has excellent thermal conductivity and heat transfer coefficient.

<table>
<thead>
<tr>
<th>Items</th>
<th>Sodium</th>
<th>Helium</th>
<th>Steam</th>
<th>Water&lt;sup&gt;2)&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature (K)</td>
<td>600</td>
<td>600</td>
<td>600</td>
<td>293</td>
</tr>
<tr>
<td>Pressure (Mpa)</td>
<td>0.1</td>
<td>10</td>
<td>7.5</td>
<td>0.1</td>
</tr>
<tr>
<td>Density (kg/m&lt;sup&gt;3&lt;/sup&gt;)</td>
<td>874</td>
<td>7.86</td>
<td>33.3</td>
<td>998</td>
</tr>
<tr>
<td>Specific Heat (kJ/kg K)</td>
<td>1.34</td>
<td>5.19</td>
<td>3.69</td>
<td>4.2</td>
</tr>
<tr>
<td>Viscosity (µPa s)</td>
<td>327</td>
<td>32.2</td>
<td>21.04</td>
<td>1071</td>
</tr>
<tr>
<td>Thermal Conductivity (W/m K)</td>
<td>75.7</td>
<td>0.254</td>
<td>0.0609</td>
<td>0.60</td>
</tr>
<tr>
<td>Flow velocity&lt;sup&gt;1)&lt;/sup&gt; (m/s)</td>
<td>4.52</td>
<td>47.4</td>
<td>27.1</td>
<td>□</td>
</tr>
<tr>
<td>Heat Coefficient&lt;sup&gt;1)&lt;/sup&gt; (kW/ m&lt;sup&gt;2&lt;/sup&gt; K)</td>
<td>84.7</td>
<td>4.78</td>
<td>5.03</td>
<td>□</td>
</tr>
</tbody>
</table>

1) The case in which it flows of the circular pipe of the 8mm diameter at pressure drop 20kPa per m.
2) Reference Data
Sodium is liquid on a very large range of temperature: from 97.8 ° (melting point) to 881.5 ° (boiling point at atmospheric pressure).

Therefore it is not necessary to pressurize the coolant in the FBR (the light water reactor must be pressurized (PWR ≥ 130 atm., BWR ≥ 60 atm.)). Consequently, in the FBR, the structural design of component becomes easy under the thermal transient shock due to employ thin walled structure of the components. Besides, operation of the FBR is easy owing above mentioned reason.

In the FBR, the natural circulation force is easy to be obtained, because the coolant is liquid phase in the wide temperature range, thus its decay heat removal capability is high.
Another Favorable Properties for Coolant

- Sodium has a good compatibility with many kinds of metallic materials such as uranium and plutonium.

  That is to say, it can restrain produced amounts of radioactive corrosion products (C.P) which is radiation source because that corrosiveness of sodium is low. It means that it is easier to make the countermeasures for the problems of the radioactive waste and the exposure reduction than water.

- Sodium has a good electric and magnetic conductivity.

  Electromagnetic equipments and meters such as electro-magnetic pump and electro-magnetic flow meter can be utilized for the plant operation.

- Density of sodium is light.

  Coolant circulating pump can be miniaturized in comparison with the LWR since its driving force may be small.

- The resources for sodium is rich.

  The unit price of sodium is cheap comparatively.
Temperature Characteristic of Liquid Sodium

Monju Cold Leg Temp. 397 °F = 0.8564

Monju Hot Leg Temp. 529 °F = 0.8252
Sodium easily reacts with oxygen and moisture in air since it is a very reactive product as mentioned. But, its reaction is not violent under the air because sodium is covered with a film layer which made from sodium oxide, sodium hydroxide etc., formed on its surface. Sodium oxide among ingredient of that surface film soon changes to sodium hydroxide due to reaction with the moisture. And finally it changes to sodium carbonate, stable element, trough reaction with carbon dioxide in air. Sodium is stable under the inert gas such as nitrogen or argon gas even it reacts with many elements. Also, sodium does not react with the petroleum group such as paraffin and kerosene and so on.

**Representative Reaction Examples**

- Sodium Oxide ($4\text{Na} + \text{O}_2 \rightarrow 2\text{Na}_2\text{O}$)
- Sodium Peroxide ($2\text{Na}_2\text{O} + \text{O}_2 \rightarrow 2\text{Na}_2\text{O}_2$)
- Sodium Hydroxide ($\text{Na}_2\text{O} + \text{H}_2\text{O} \rightarrow 2\text{NaOH}$)
  \[\text{Na}_2\text{O}_2 + \text{H}_2\text{O} \rightarrow 2\text{NaOH} + \text{1/2O}_2\]
- Sodium Hydrogen Carbonate ($2\text{NaOH} + 2\text{CO}_2 \rightarrow 2\text{NaHCO}_3$)
- Sodium Carbonate ($\text{NaHCO}_3 + \text{NaOH} \rightarrow \text{Na}_2\text{CO}_3 + \text{H}_2\text{O}$)

* This reaction is occurred under the condition over about 400 such as like in case of sodium fire.
The reaction of sodium with water is very violent and producing hydrogen gas which is an explosive gas in air and high reaction heat. Its kinetic is very fast. Main chemical reaction is as follows:

- \[ \text{Na} + \text{H}_2\text{O} = \text{NaOH} + \frac{1}{2}\text{H}_2 + \text{Heat} \ (-145\text{kJ/mol}) \]
- \[ 2\text{Na} + \text{H}_2\text{O} = \text{Na}_2\text{O} + \text{H}_2 + \text{Heat} \ (-128\text{kJ/mol}) \]

The explosion phenomenon by the reaction between sodium and water is caused rapid cubical expansion by high reaction heat.

*出典: Applied Chemistry of the Alkali Metals, P241, Borgstedt & Mathews, 1987*
Sodium reacts with alcohol (ethanol) as shown in the following chemical equation and produces the salt as alcoholic sodium compound and also the hydrogen gas as well as the reaction with water. This salt is called alcoholate or alkoxide.

\[ 2\text{C}_2\text{H}_5\text{OH} \text{ (ethanol) } + 2\text{Na} \rightarrow 2\text{C}_2\text{H}_5\text{ONa} \text{ (alcoholate) } + \text{H}_2 \]

It is well known that the concrete also reacts with sodium because concrete is mainly composed of water and silicon. According to the R&D report, this reaction begins from the sodium temperature over 500°C.

Dominant chemical formula of this reaction is as follows:

\[ \text{Na} + \text{H}_2\text{O} \rightarrow \text{NaOH} + 1/2 \text{H}_2 \]
\[ \text{NaOH} + 1/2\text{SiO}_2 \rightarrow 1/2\text{Na}_2\text{SiO}_3 + 1/2\text{H}_2\text{O} \]

According to some experiments, the reaction between sodium and concrete finishes in about 10 hours, and the erosion data of the concrete by the reaction is reported only 20〜30 cm including the overseas data.

*出典 http://www.tuat.ac.jp/ 毒剤物取扱者試験に必要な専門化学*
Halogens (Cl, Br, I, F) are a strong oxidizing elements and react with sodium immediately. Chemical formula of chlorine with sodium is shown as a sample.

\[ \text{Na} + \frac{1}{2} \text{Cl} = \text{NaCl} \]

As a consequence, all the products containing halogens should be avoid to be in contact with sodium.

Scene of Reaction between Sodium and Chlorine (Cl)
# Chemical Properties of Representative Sodium Compounds

<table>
<thead>
<tr>
<th>Compound</th>
<th>Chemical Formula</th>
<th>Chemical Property</th>
<th>Application (Instance)</th>
</tr>
</thead>
</table>
| Sodium Oxide      | Na₂O             | - Unstable chemical compound  
|                   |                  | - Forming NaOH after dissolving in water  
|                   |                  | - Irritating Odor  
|                   |                  | (Na₂O + H₂O → 2NaOH)                                                            | Bleaching Agent                          |
| Sodium Peroxide   | Na₂O₂            | - Reaction temp. 300-400°C (Yellow)  
|                   |                  | - Forms NaOH after dissolves in water  
|                   |                  | - Deleterious substance  
|                   |                  | - Irritating Odor  
|                   |                  | (2Na+ O₂ → Na₂O₂)  
|                   |                  | (Na₂O₂ + H₂O → 2NaOH + 1/2O₂)                                                    | Analytical Reagent  
|                   |                  |                                                                 | Air Cleaning Medicine                    |
| Sodium Hydroxide  | NaOH             | - Rich deliquescency & strong corrosiveness  
|                   |                  | - Reacts with water and produces much heat  
|                   |                  | - Deleterious substance  
|                   |                  | - White and melting temp. ≈ 318°C  
|                   |                  | - Skin and eyes irritating                                                      | Soap, Paper  
|                   |                  |                                                                 | Manufacturing, Petroleum Refining       |
| Sodium Carbonate  | Na₂CO₃           | - Very stable, never react with sodium either  
|                   |                  | (NaHCO₃ + NaOH → Na₂CO₃ + H₂O)  
|                   |                  | (2NaHCO₃ → Na₂CO₃ + CO₂ + H₂O)                                                   | Na Fire Extinguisher, Glass, Dye, Bathing Medicine |

* Thermal Decomposition
4: Physical Properties of Sodium

In this section, the representative physical properties of sodium are described.

Density
The density of sodium changes by being dependent on its temperature. Its empirical formula of solid sodium and liquid sodium are as follows:
- Equation by J. P. Stone et. Al (1965) see appendix-1 (for Liquid Sodium)
  \[ \rho = 950.1 + t \left(0.22976 + t (-1.46 \times 10^{-5} + 5.638 \times 10^{-9}t) \right) \text{ (kg/m}^3\text{)} \]

Melting Point
The sodium melting point is 97.8 °C at 1 atm.

Boiling Point
The sodium boiling point is 881.5 °C at 1 atm.

Specific Heat
The specific heat is a necessary calorific value for raising of 1 °C, 1g of material.
- \( C_p = 0.34324 + t (1.3868 \times 10^{-4} + 1.1044 \times 10^{-7}t) \) (kcal/kg °C)
  - see appendix-1 (for Liquid Sodium)

Thermal Volume Expansion
2.7% when passing from solid to liquid.
**Viscosity**

The pressure independence of viscosity characteristics can be disregarded.

- Under 500 °C: \( \nu = (0.1235 - 0.0018) \nu^{1/3} \exp \left[ \frac{697}{K} \right] \)
- Over 500 °C: \( \nu = (0.0851 - 0.0013) \nu^{1/3} \exp \left[ \frac{1040}{K} \right] \)

(Where, \( K = t + 273.15 \))

- see appendix-2 (for Liquid Sodium)

**Kinetic Viscosity**

The kinetic viscosity coefficient (\( \nu \)) can be obtained by the absolute viscosity coefficient (\( \nu \)) divides with density.

\[ \nu = \frac{\nu}{\rho} \text{ (m}^2\text{/sec)} \]

- see appendix-3 (for Liquid Sodium)

**Thermal Conductivity**

Equation by G. H. Golden and J.V. Tokar (1967)

\[ \kappa = 86.04 \{0.93 + t \left(1.173 \times 10^{-7} t - 0.581 \times 10^{-3}\right)\} \text{ (kcal/m.h. °C)} \]

- see appendix-3 (for Liquid Sodium)
Sodium usually ignite from 130°C to 300°C approximately. It depends on atmosphere conditions and the way to heat sodium.

When heating the solid sodium, its surface film layer is deformed due to crossing to its melting point and will be finally broken or peeled.

Consequently, sodium vapor comes out from the broken or peeled a film, and it reacts with the oxygen in air. Thus, sodium finally ignites by its reaction heat.
Sodium in air burns with much white smoke called a sodium aerosols. Main reaction formulas are as follows:

\[
\begin{align*}
2\text{Na} + \frac{1}{2}\text{O}_2 &= \text{Na}_2\text{O} \quad (-415\text{kJ/mol}) \\
2\text{Na} + \text{O}_2 &= \text{Na}_2\text{O}_2 \quad (-511\text{kJ/mol}) \\
2\text{Na} + \text{H}_2\text{O} &= \text{Na}_2\text{O} + \text{H}_2 \quad (-173\text{kJ/mol}) \\
\text{Na} + \text{H}_2\text{O} &= \text{NaOH} + \frac{1}{2}\text{H}_2 \quad (-184\text{kJ/mol})
\end{align*}
\]

- The above sodium oxides will further react under the moisture environment and also the hydrogen gas will recombine with the oxygen during sodium combustion.

- The flame of sodium combustion is very low (few mm (gasoline fire: few m)) due to sodium vapor pressure head is low.
- It is suspected that about 20-40% of the combustion products are released with the aerosol.
- Aerosol includes Na$_2$O$_2$ and NaOH which are harmful product to human body and has irritant smell.

- Combustion and flame temperatures are about 650-870 °C and 1350 °C respectively.
- Its flame is orange but no radiant 80cm above the fire the temp. is less than 100 °C.
- When under the condition such as blow 3-4% of oxygen, sodium fire may not occur.
The sodium combustion is divided into the following three types.

- **Pool Type Combustion**
  The leaked sodium spreads on the floor as a pool state and it reacts with the oxygen and moisture on its surface. Though its combustion is calmer than the spray type combustion mentioned below, it may continually continue even after the leakage is stopped. In this type, the designers have to pay attention on the structural strength of the floor which contacted sodium leaked.

- **Spray Combustion Type**
  When sodium leaks from the piping, it becomes the multiple droplets and drops with scattering. At that time, the multiplied sodium droplets react with the oxygen and the moisture on its surface. Main problem of this type is pressurization in sealed atmosphere by the burn up heating.

- **Column Combustion Type**
  In this case, the leaked sodium from the piping falls cylindrically like the waterfall and furthermore it collides to the floor plate and scatters, sodium reacts with the oxygen and moisture. In the actual leakage accident, the sodium combustion type seems to be of this type, and it is more moderate than the spray combustion.
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*出典: 宮原、篠原増殖炉における冷却材ナトリウムの漏れ対策』, 日本燃焼学会誌 第45巻133号 (2003年8月) pp.141-151*
Since extinguishment of sodium combustion is carried out based on the principle of extinguishment by smothering, the most important point is to cover up completely firing area with fire extinguisher.

In the sodium extinguishment, the fire extinguisher must be sufficiently sprinkled over the fire area to prevent occurring re-ignition.

Sprinkling divided into several layers is more effective than a thick single layer in order to prevent spreading the leaked sodium.

”Natrex” (brand name), the fire extinguisher for the metal fire of which main ingredient is sodium carbonate, is developed by PNC in Japan.

In the sodium fire extinguishing, never use water or the general extinguisher.
## Sodium Fire Extinguisher “Natrex”

<table>
<thead>
<tr>
<th>Type</th>
<th>Natrex M20W</th>
<th>Natrex M50</th>
<th>Natrex M200</th>
</tr>
</thead>
<tbody>
<tr>
<td>Extinguisher Amount</td>
<td>4kg</td>
<td>12kg</td>
<td>40kg</td>
</tr>
<tr>
<td>Total Weight</td>
<td>9kg</td>
<td>35kg</td>
<td>224kg</td>
</tr>
<tr>
<td>High</td>
<td>640mm</td>
<td>910mm</td>
<td>1340mm</td>
</tr>
<tr>
<td>Diameter</td>
<td>154mm</td>
<td>227mm</td>
<td>356mm</td>
</tr>
<tr>
<td>Hose Length</td>
<td>385mm</td>
<td>1,750mm</td>
<td>15,000mm</td>
</tr>
<tr>
<td>Emission Tube Length</td>
<td>350mm</td>
<td>350mm</td>
<td>1315mm - 3225mm</td>
</tr>
<tr>
<td>Emission Tube Diameter</td>
<td>73mm</td>
<td>73mm</td>
<td>73mm</td>
</tr>
<tr>
<td>Emission Time</td>
<td>15sec</td>
<td>40sec</td>
<td>65sec</td>
</tr>
<tr>
<td>Extinguishing Area</td>
<td>Max. 1m²</td>
<td>Max. 3m²</td>
<td>Max. 9m²</td>
</tr>
</tbody>
</table>

*出典: 宮原、筒速増殖炉における冷却材ナトリウムの漏えい対策、日本燃焼学会誌 第45巻133号（2003年8月）pp.141-151*
In the sodium fire extinguishing, the worker must wear the equipments for fire fighting shown in the attached picture to protect themselves from fire and smoke included hazardous materials such as sodium peroxide, sodium hydroxide, etc..

In the wearing, the worker have to put on the fire fighting dress without loop and clearance where sodium goes into. And as a principle, sleeve of outer wear and skirt of pants must be worn on long leather glove and half long boots, respectively.

*出典* 宮原、篠原 増殖炉における冷却材ナトリウムの漏えい対策、日本燃焼学会誌 第45巻133号 (2003年8月) pp.141-151
### 6: Influence on Human Body & First Aid

#### Hazard by Touched Body

<table>
<thead>
<tr>
<th>Portion</th>
<th>Symptom</th>
<th>First Aid</th>
</tr>
</thead>
</table>
| Skin    | Alkali Burn| - Remove Remaining Sodium  
|         |            | - Washing by Water  
|         |            | - Showering  
|         |            | - Go to Expert Hospital (as soon as they can)    |

【Alkali Burn】
## Hazard by Inhaling into Body

<table>
<thead>
<tr>
<th>Portion</th>
<th>Symptom</th>
<th>First Aid</th>
</tr>
</thead>
<tbody>
<tr>
<td>Eye</td>
<td>Degrade of eyesight by damage of crystalline lens, (Blindness)</td>
<td>Eye washing (immediately)</td>
</tr>
<tr>
<td>Mouth, Throat</td>
<td>Oral cavity burn</td>
<td>Gargles (immediately)</td>
</tr>
<tr>
<td>Lung</td>
<td>Damage of lung cell (Unable to breathing)</td>
<td>&quot;</td>
</tr>
<tr>
<td>Gullet, Stomach &amp; Intestines</td>
<td>Hole is possible in gullet or in gastric parietal</td>
<td>Compulsory vomiting (immediately)</td>
</tr>
</tbody>
</table>
Appendix

- Appendix-1: for Density and Specific Heat
- Appendix-2: for Viscosity
- Appendix-3: for Kinetic Viscosity and Thermal Conductivity

Attention

The dimension in the appendix’s figures is described with the old type dimension, MKS. So, it needs to pay attention when using it.
Appendix-1
Appendix-2

出展 PNC N 941 75-19、ナトリウム物性値の実用計算式 (1972年までの公表文献に基づく液体と蒸気の物性値) P164、1975年3月
Thermal Conductivity (kcal/mh)

Kinetic Viscosity (m²/sec)

Temperature T(°C)

Appendix-3

出展　PNC N 941 75-19. ナトリウム物性値の実用計算式 (1972年までの公表文献に基づく液体と蒸気の物性値) P165. 1975年3月
Fundamental Knowledge on Sodium Handling Working

Makoto Sawada

September 30, 2004

International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
Presentation Contents

§ 1: Danger of Spontaneous Ignition of Sodium
§ 2: Caution Items on Sodium Cleaning & Disposal Working
§ 3: Disposal of Contaminated Sodium Wastes
§ 4: Cleaning for Contaminated Sodium Components in Monju
§ 5: Caution on Sodium Loop Operation
§ 1: Danger of Spontaneous Ignition of Sodium

Accident caused by Spontaneous Ignition of Sodium with Burnable Materials in Air

◆ The fire may happen, when sodium is left with combustibles under conditions such as reaction heat is insulated.
◆ Even small quantity of sodium, it may ignite spontaneously after several hours under the above mentioned condition.
◆ The fire guessed with this cause have been happened in the JOYO (experimental fast reactor, in Oarai) maintenance building on October first, 2001.

The cause of this fire was supposed that anybody among workers carelessly seemed to discard some sodium pieces with cleaning papers into a carton box.
Demonstration of Spontaneous Ignition of Sodium with Burnable Materials in Air

Test Condition: six pieces of small quantity sodium were left in a carton box of combustibility.

(1) Start Smoking [5 hours passed after started test]

(2) 7 min. after start smoking

(3) 9 min. after start smoking [Ignition]

(4) 9 min. after start smoking [Flame Increasing]

(5) 12 min. after start smoking [Fire Progression]

[Left Sodium Quantity: 0.3g/piece x 6 pieces]

出典：「常陽」メンテナンス建家火災事故調査委員会報告書、添付資料、P4-41-43、2001年11月
The following pictures shows the spontaneous ignition process of metal sodium. (Sodium Quantity: 0.3g)
Oxidization of surface of solid sodium

$\text{Na}_2\text{O}$ change to $\text{NaOH}$ (reaction with moisture)

Deliquescence of $\text{NaOH}$ (increase of $\text{NaOH}$)

Reaction heat ($\text{NaOH} + \text{Na}_2\text{O}$) & $\text{H}_2$ increase

Insulated reaction heat by solution $\text{NaOH}$ and sodium melts

Melted sodium and $\text{H}_2$ float onto solution $\text{NaOH}$

Sodium and $\text{H}_2$ gas Ignite by reaction heat

Air ($\text{O}_2$) → $\text{Na}_2\text{O}$ formed

$\text{Na}_2\text{O}$ → change to NaOH

Moisture (deliquescence) Increase NaOH

$\rightarrow$ reaction heat & $\text{H}_2$ increase

Liquid NaOH (insulate heat)

Melted Na

Ignition Na & $\text{H}_2$ gas

出典：「常陽」メンテナンス建家火災事故調査委員会報告書、添付資料、P4-120-121、2001年11月
### § 2: Caution on Sodium Cleaning & Disposal Working

#### Cleaning and Disposal Method for Sodium Compound

<table>
<thead>
<tr>
<th>Method</th>
<th>Merit</th>
<th>Demerit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Water &amp; Steam Cleaning</td>
<td>◆Easy Operation ◆Perfect Cleaning ◆Easy for Waste Liquid Treatment ◆Cheap</td>
<td>★Risk of Causing Hydrogen Explosion ★Risk of Causing Alkalinity Corrosion During Cleaning ★Need Smoke Control Facility &amp; Plumbing</td>
</tr>
<tr>
<td>Alcohol Cleaning</td>
<td>◆Mild Reaction ◆Low Possibility of Causing Alkalinity Corrosion</td>
<td>★Need Treatment of Alcohol Waste Water ★Risk of Causing Fire by Ignition of Alcohol</td>
</tr>
<tr>
<td>Combustion Treatment</td>
<td>◆Cheap</td>
<td>★Need Smoke Control Facility ★Risk of Causing Component Corrosion</td>
</tr>
<tr>
<td>Carbon Dioxide Cleaning</td>
<td>◆Mild Reaction ◆Safety</td>
<td>★Imperfect Cleaning (Difficult Perfect Cleaning in Narrow Portion)</td>
</tr>
</tbody>
</table>
As well known, the possibility of hydrogen explosion rises when its concentration in the atmosphere exceeds over 4% approximately*.

So, it is necessary to do cleaning operations slowly with low cleaning speed or flushing with inert gas to avoid to reach the explosion region.

Besides it is important that the hydrogen gas would not accumulate inside in piping or vessel during the working.

It should be monitored by the concentration of hydrogen gas in the atmosphere, if it is possible. And that time, the alarm level must be set to conservatively for instance, 1%.

*出典: 石井、水素ガス爆発限界圧力による測定、安全工学、P294-295、1964

In the case of water and steam cleaning, certainly the hydrogen gas is produced based on the following equation mentioned in the former session.

\[ \text{Na} + \text{H}_2\text{O} = \text{NaOH} + \frac{1}{2}\text{H}_2 \uparrow \]

So, it is necessary to do cleaning operations slowly with low cleaning speed or flushing with inert gas to avoid to reach the explosion region.
It is very important to exactly estimate whether the sodium cleaning work is finished completely or not yet, in order to perform it safely.

Especially, man have to pay attention on the cleaning work by means of alcohol for such as the cylindrical tube in which sodium is filled inside.

Since the reaction between sodium and alcohol has terminated soon in appearance due to make film made from water-soluble alcoholate (alkoxide) on the surface of sodium, consequently it is easy to make a misunderstanding that cleaning has been finished completely.

Danger on Alcohol Cleaning

- (1) Alcohol
- (2) Alcohol
- (3) Alcohol

\[ 2C_2H_5OH + 2Na \rightarrow 2C_2H_5ONa + H_2 \]

Sodium spouts from the piping by the reaction between sodium and water.
Actual Accident Example (1) in Japan

《Summary of the Accident》
During carrying out the cleaning operation of a small cylindrical sodium storage bottle (Diameter: 2cm, Length: 30cm) as a final stage cleaning by the water after finished cleaning operation used alcohol, on Oct. 1998, in JOYO, worker B in the below picture was seriously injured to his eyes by the spouted sodium compound from the bottle. At that time, he faced toward to upside in surprising to the big explosion sound. The cause of this accident is that worker misunderstood that almost of sodium had not remained in the bottle. However, a lot of quantity of sodium was remaining covered by the water-soluble sodium compound in the bottle.
Actual Accident Example (2) in France

Summary of the Accident
In Mar. 1994, the accident happened in which one person died and four injured by the explosion during cleaning working of the sodium storage tank for decommissioning of the Rapsodie, experimental sodium reactor, in France. The workers carried out the cleaning operation for changing the sodium remained in the bottom of the tank to the alchoholate by pouring the alcohol into the tank slowly. The cause of the accident is due to a thermal runaway caused by the breaking of the alchoholate molecule to several explosive gases.

In Japan

Prohibition to apply to as the Sodium Cleaning Method
After occurred the above mentioned accident, in Japan, the alcohol cleaning method was prohibited to apply such as for the component cleaning contained sodium its inside.

Usefulness for preventing of Alkalinity Corrosion
Although the alcohol cleaning has been prohibited as the sodium cleaning method, but it is effective to apply for the cleaning operation which needs to avoid alkalinity corrosion occurring of the thin material such as bellows, fuel pin and so on. At any time the quantity of sodium must be limited (less than 1kg, in France) and exactly located. (no hidden area full of sodium)
§ 3: Disposal of Contaminated Sodium Wastes

**Kind of Sodium Nuclide**

During reactor operation, the stable nuclide 23Na in the reactor may change into the various nuclides by nuclear reaction which absorbs the neutron shown in the bottom table.

<table>
<thead>
<tr>
<th>Original Nuclide</th>
<th>Nuclear Reaction</th>
<th>Produced Nuclide</th>
<th>Half-Life</th>
<th>Radiation</th>
<th>Energy</th>
</tr>
</thead>
<tbody>
<tr>
<td>23Na</td>
<td>n, $\gamma$</td>
<td>24Na</td>
<td>15h</td>
<td>$\gamma$</td>
<td>1.37MeV, 2.75MeV</td>
</tr>
<tr>
<td></td>
<td>n, p</td>
<td>23Ne</td>
<td>38s</td>
<td>$\gamma$</td>
<td>0.44MeV, 1.65MeV</td>
</tr>
<tr>
<td></td>
<td>n, 2n</td>
<td>22Na</td>
<td>2.6y</td>
<td>$\gamma$</td>
<td>1.28MeV, 1.63MeV</td>
</tr>
<tr>
<td></td>
<td>n, $\alpha$</td>
<td>20Fe</td>
<td>11s</td>
<td>$\gamma$, $\beta$</td>
<td>1.63MeV, 5.40MeV</td>
</tr>
</tbody>
</table>

In these nuclides, 24Na and 22Na are the most important nuclides.

Though half-life of 24Na is not so long, it becomes the main radiation source during operation because it is very much produced in comparison with 22Na.

While, although 22Na is not so much produced during operation, it becomes the dominant radiation source for during maintenance since it is gradually accumulated due to its longer half-life of 2.6 years.
Case for Non-Radioactive Sodium Wastes

Processing Flow for Contaminated Sodium Wastes in Monju
Case for Radioactive Sodium Wastes

Processing Flow for Contaminated Sodium Wastes in Monju
The handling working of radioactive sodium wastes must be performed inside the clean house in order to prevent the radioactive materials scatter.
§ 4: Cleaning for Contaminated Sodium Components in Monju

Object Components

1. Various Fuel Handling Machines
2. Control Rod Drive Mechanism
3. Primary Circulation Pumps

Operation Process

1) Cooling by Moist Nitrogen Gas
2) Cleaning by Pure Water Flushing
3) Cleaning by Water Bubbling
4) Drying by Hot Nitrogen Gas

Design Cleaning Time (16 Hours/day)

⇒ Control Rod Drive Mechanism Guide Tube: 2 days (for all operation process)
⇒ Circulation Pump: 7 days (for all operation process)

Design Concentration of Hydrogen Gas

◆ Under 4 Vol.% for prevention of hydrogen gas explosion
  (Normal: under 1 vol.%)
Result Examples

【Fuel Handling Machine】

【In-Vessel Fuel Transfer Machine】

【Ex-Vessel Fuel Transfer Machine】
§ 5: Caution on Sodium Loop Operation

Checking of Temp. Distribution

◆ Before to begin charging sodium into the loop, its temp. distribution must be checked. (Monju: 200±30°C)
◆ For the IHX, its setting of preheating temp. before charging sodium must be gradually raising for preventing to occur a large difference temp. between its inside and outside.
◆ Also about the Pump, its bearing temp. is monitored as its inside temp. and the interlocking which locks pump starting under the condition less than 150°C is provided.
Order of Melting & Freezing for Valve

**For Melting**

It must carry out melting from sodium in the piping side. If do reverse operation, the melted sodium will break sheet or metal bellows of the valve by reaction force causing thermal expansion.

**For Freezing**

While, in the case of freezing, its operation must be performed from the valve side.

- Metal Bellows
- Radiation Heat Fin
- Sheet
- : Weak Portion of Valve
Sampling Operation

Caution

◆ It is the most important to surely conduct the tightening of the swage lock on sampling operation.
◆ The sodium leakage accident at the swage lock have experienced some times so far.

Flow Meter

Replacement with N2 Gas

24 h Flushing for Sampling Tube

During Sampling, Monitor Sodium Leak from Swage Lock

N2 Gas Bomb

Swage Lock

To Secondary System

Outlet

Inlet

Exhaust

Monitor O2 Concentration

Preheating Heater

Secondary Na Sampling Box
Sodium Corrosion Mechanism and Sodium Purity Control in Monju

Makoto Sawada

October 5, 2004

International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
Presentation Contents

§ 1: Material’s Corrosion Mechanism by Sodium
§ 2: Sodium Purity Control in Monju
§ 3: Principle and Measuring Method of Plugging Meter
§ 4: Operating Experience of Sodium Purification System in JOYO
§ 1: Material’s Corrosion Mechanism by Sodium

(1) Kinds of Corrosion Mechanism by Sodium

As the materials of FBR’s structure such as vessels, components, piping, etc., the SUS (austenitic stainless steel) is widely used since it has good characteristics against the high temp. strength, radiation hardness and corrosion-resistance. Additionally, the ferritic steel which has a strong resistance characteristic for the stress corrosion cracking (SCC)*, which occurs in the water the environment and steam, is employed as the evaporator’s material. General speaking, a material’s corrosion seldom arises in the FBR whose system temp. is approximately range of 200°C-550°C. However in the case in which the concentration of impurities in sodium is high, corrosion will progress. Moreover, although it’s very slight quantities, corrosion causing dissolution and deposition of structural material’s constituents also must be taken into account. Therefore, the corrosion mechanism of materials by metallic sodium is mainly divided into the following two kinds:

① Corrosion by Impurities in Sodium
② Corrosion by Dissolution and Deposition of Material’s Constituent

*See reference (P16)

出典:FBR情報素材集2版・上、12-2ナトリウム技術「ナトリウムと材料の適合性」、科学技術庁、平成2年3月
出典:PNC PN9520 91-006、高速増殖炉技術読本、第10章第3章材料の液体金属ナトリウム腐食、P10-7.8、動燃大洗工学センター、1991年7月
Corrosion by Impurities in Sodium

This corrosion type is based on interaction between the impurities in sodium and the material’s constituents. It is a chemical reaction between the impurities within the solution and the material’s specific constituents and is strongly linked to the purity control of the sodium.

(a) Oxygen

One must pay attention to oxygen among the impurities in sodium because its solubility in sodium is the highest among them, and has consequential affects on the material’s corrosion. Oxygen in Sodium exists in the form of sodium oxide (Na₂O). In a sodium loop system it produces film, FeO·(Na₂O)₂, by reacting with the iron of the material. This film is dissolved out of the material’s wall at the high-temp. and it’s deposited at the low-temp.. Thus, corrosion of which dissolution and deposition of iron will progress. Such as this, oxygen plays a role of catalyst and corrosion’s quantities is in proportion to oxygen’s quantities in sodium.
Corrosion speed is in proportion to sodium temp. and oxygen concentration.

【Relationship between Corrosion Speed and Oxygen Concentration for SUS】

出典：PNC PN9520 91-006、加速増殖炉技術読本、第10編第3章材料の液体金属ナトリウム腐食、P10-38、動燃大洗理工学センター、1991年7月
Corrosion Speed (cm/year)

Temp. (℃)

Material

Oxygen

<5ppm

<10ppm

~25ppm

Various Materials for SUS and Ferritic Steel (Different Condition on Manufacturing)

【Relationship between Corrosion Speed and Oxygen Concentration for Each Material】

出典：FBR広報素材資料集2版・上、12-2ナトリウム技術「ナトリウムと材料の適合性」、P12-2, 科学技術庁, 平成2年3月
The rate of corrosion of SUS will be less than the rate of ferritic steel under the same conditions in sodium.
As the impurities in sodium besides oxygen, hydrogen and carbon must also be considered. Carbon’s role in corrosion is not that of dissolution or of deposition but as an element that diffuses into the grain boundaries of the material’s structure.

(b) Hydrogen

◆Hydrogen is not so important impurity in sodium from the point of view of material’s corrosion, because it doesn’t exist much in sodium due to its small solubility.

※Hydrogen will be discussed in the next chapter “sodium purity control in Monju”.
Sodium as a coolant has a good compatibility with materials such as stainless and ferritic steels. But, as mentioned it in the former section, the corrosion phenomenon owing dissolution and deposition of the material’s constituents arise though its quantities are very little; the alloy elements of the structural material, Cr, Ni, Mn, Co, etc., dissolve into sodium and deposit to the material’s wall. Besides carbon transportation, carburization and decarburization, are very important phenomena, because even in slight quantities, it will real effect on the material’s structural strength.

(a) Dissolution and Deposition of Alloy Elements

Since sodium has a strong reduction action, the oxide film on the material’s surface is reduced by sodium and consequently material’s alloy elements are directly subjected to the sodium flow. As this result, corrosion will progress through the process in which the alloy elements dissolve at the high-temp. region and deposit at the low-temp. region.

As the dissolution phenomenon is dominant on diffusion speed inside of the material, its speed depends on the environment’s temp. and soaking time.

This phenomenon continues depending on the soaking time in that the dissolve material’s alloy elements at high-temp. will deposit by becoming a super-saturation state at low-temp.. This phenomenon is called “Thermal Gradient Mass Transfer”.

In the primary cooling system, this phenomenon is more serious than the loss of material properties because the alloying elements when activated affect the service-ability of the plant in areas where the alloying elements re-deposit.
Material’s alloy elements, Cr, Mn, Ni, etc., eluate into sodium at the high-temp. region and deposit to the material’s wall at the low-temp. region.

In the primary cooling system, as eluated and deposited alloy elements are becoming radioactive, for instance 54Mn, 60Co, exposure problem is more important comparison with the problem of material’s structural strength.
It's obvious from this figure that each alloy element’s solubility is very low. It’s ppm order when the sodium temp. is several hundreds °C.

【Solubility of Each Alloy Element in Sodium】

出典: PNC PN9520 91-006、高速増殖炉技術読本、第10編第3章材料の液体金属ナトリウム腐食、P10-34、動燃大洗工学センター、1991年7月
Dissolution at High-Temp. Region

Deposition at Low-Temp. Region

【A Example of SUS 316 Steel’s Corrosion Form by Sodium】

出典: PNC PN9520 91-006、高速増殖炉技術読本、第10編第3章材料の液体金属ナトリウム腐食、P10-35、動燃大洗工学センター、1991年7月
The below figure describes the weight variation for SUS 316 steel in the place at a distance from the outlet of the electromagnetic pump (EMP) caused by dissolution and deposition under sodium flows. According to the measurements below, it is clear that the most severe dissolution happens at the place approximately 10m away from the EMP’s outlet.

【Weight Variation for SUS316 Steel under condition of Sodium Flowing】
(b) Phenomenon of Decarburization & Carburization

Carbon is an element which has strong influences on the material’s structural strength even its quantity is relatively small concentration. Since it happens the decarburization phenomenon which carbon contained in the material dissolves out into sodium and the carburization phenomenon which eluated carbon into sodium invades into the material, therefore, it’s necessary to sufficiently pay attention about those phenomenon. Both phenomenon lead to the following degradation of the material’s characteristic.

- **Decarburization** → Deterioration of Tensile Strength and Creep Rupture Strength
- **Carburization** → Deterioration of Ductility

**Free Carbon and Active Carbon**

There are two kinds of carbon, free carbon and active carbon. Although the existing active carbon’s quantity is slight, it has a high chemical interaction characteristic between materials in that its chemical activity is high. Therefore, the active carbon is important regarding material’s strength characteristics.

出典：標準化学用語辞典、日本化学会編集、丸善株、1991年3月
出典：PNC PN9520 91-006、高速増殖炉技術読本、第10編第3章材料の液体金属ナトリウム腐食、P10-13-15、動燃大洗工学センター、1991年7月
Decarburization and Carburization in Secondary Cooling System

◆ The secondary cooling system is a bimetallic system which consists of two kinds of steel, stainless steel and ferritic steel. (shown in the next page)

⇒《Stainless Steel (SUS304) 》: Super Heater, Piping, Components,

⇒《Ferritic Steel (21/4Cr-1Mo) 》: Evaporator

◆ As the ferritic steel alloy contains much more active carbons which has high activity further than the stainless steel, therefore, the active carbons will dissolve out into sodium from the ferritic steel (decarburization phenomenon) and will attack the stainless steel (carburization phenomenon).

◆ However, according to the experiment report*, it’s revealed that the carburization phenomenon for SUS 304 seldom influences its structural strength.


◆ Moreover, the evaluation method using the decarburization speed coefficient obtained from the experiments data (shown in fig., p17) for predicting of the carbon quantities remained in the materials had been established. So, it’s possible to model the structure’s integrity at the end of the plant’s life time.
Secondary Cooling System’s Materials in Monju

Main Heat Transfer System for Monju

- Stainless Steel (SUS304)
- Ferritic Steel (21/4Cr-1Mo) (Evaporator Only)

[Diagram showing the main components of the heat transfer system: Super Heater, Evaporator, etc.]
What is SCC?

◆ In the case of stressed metal in a corrosive environment, “Stress Corrosion Cracking (SCC)”, this material cracks at of below the lower stress level in comparison with the normal cracking level. This is a result of the interaction between the metal and the corrosion environment and mechanical stresses.

◆ The SCC sometimes happens for the SUS under high-temp. and high-pressure water. This phenomenon is caused by the interaction of the following three factors.

1. **Material’s Factor** (Sensitization of Material): Decline of the corrosion-resistance by producing of the chrome lacking layer* on the crystal’s grain boundary causing by weld’s thermal influence

   * Chrome decreases causing by forming carbide due to combining with carbon

2. **Stress’s Factor**: Presence or existence of stress e. g., welding residual stress

3. **Environment’s Factor**: Existing corrosive environment

◆ Therefore, it’s able to prevent occurring the SCC by removing more than one factor among them. It’s listed, for example, using of quite low carbon’s material, or water cooling welding as for the measures of welding residual stress.

Measures for SCC in Monju

◆ In the FBR, the SCC is a concern the SG which contains a lot of water. In this application, Monju employs ferritic steel (21/4Cr-1Mo) the evaporator’s material (both vessel and tube), since it has a strong-resistance to SCC due to its low carbon alloy.

◆ Besides, Monju takes the following measures for the stress’s factor and environment factor.

⇒ To take thorough control of the quality assurance for welding
⇒ To keep the oxygen concentration in the sodium loop to very low level
【Decarburization Speed Coefficient for 21/4Cr-1Mo Steel】
Decarburization and Carburization in Primary Cooling System

◆ The primary system decarburization of core materials such as fuel element cladding on fuel assembly at high temp. is a concern.
◆ As a countermeasure for this problem a new alloy, a modified SUS316, in the reduced carbon concentration via alloying elements Ti, Nb, P, etc., has been developed. This alloy is being used in advanced cladding materials. (See the next page)
◆ While, regarding the structure material except the core materials, the decarburization phenomenon doesn’t occur practically because its service temp. is lower than 550°C.
【Carbon Concentration’s Variation depending on soaking Time in Sodium】

出典: PNC PN9520 91-006、高速増殖炉技術読本、第10章第3章材料の液体金属ナトリウム腐食、P10-42、動燃大洗工学センター、1991年7月
There is little difference between the original material (non-dipped) and the material of dipped in sodium concerning its characteristics of elongation and creep rupture strength.
(1) Impurities in Sodium Loop

(a) Oxygen

Oxygen is the most prevalent impurity in the sodium loop, and it’s carried in causing of mixing of air during the plant construction, fuel handling working, periodical inspection and so on. If the oxygen concentration level is high, it will gives rise to the various undesirable situations such as corrosion of the components, piping, blocking of flow pass, producing of the rust which becomes the radiation resource, etc..

Oxygen’s Solubility in Sodium

◆ Oxygen’s solubility in sodium is dependent on sodium’s temp..
  (See the next page; equation of Eichelberger, empirical equation of PNC)
  ◆ The relationship between above both is that the oxygen’s concentration is 1ppm at 120°C, and is several hundred ppm at range of 300-400°C.
  ◆ The impurities in sodium can be removed by means of the cold trap (CT). Hence, it’s able easily to control the oxygen concentration in sodium. The CT makes use of a characteristic which the impurity solubility has a temp. dependency. That is to say, CT can remove oxygen in sodium as oxide at the mesh portion by keeping sodium’s temp. to low which is equal to its super-saturated temp.
Existing oxygen in sodium is only 1g among sodium of 1 ton, when sodium is cooled to 120°C. At this temp. oxygen is removed an oxide.

Monju Control Limit = 3ppm

(For Primary Cooling System)
The cold trap controls the temp. at the entrance of mesh part.
The impurities in sodium will crystallize and deposit at the mesh part when sodium’s temp. reaches to its super-saturated temp.
Inlet tem. of the C/T is almost corresponding with the plugging temp..

【Relationship between C/T Temp. and Plugging Temp. (Meter Indicator)】

出典：SN2410 86-001、大洗ナトリウム技術専門委員会、プラグ計に関する技術資料、P139、1986年2月
(b) Hydrogen

- The solubility of hydrogen in sodium is lower than oxygen approximately 1 digit*, but it is not be able to be ignored as an impurity for the secondary cooling system.  
  * For example, at 200°C, oxygen is about 12ppm (see p22), while hydrogen is 1ppm (p26).
- Because, produced hydrogen causing by the heat pipe’s corrosion at the steam generator (SG) diffuses tube’s wall and transfers into the secondary loop’s sodium. For that reason, it will have a bad influence on back ground of a hydrogen meter which is a broken detection device of the SG heat pipe and will cause a trouble on reactor operation.
- Hydrogen is able to be remove as a hydride by the CT.

(c) Carbon

- It’s not able to be control carbon concentration in sodium by the CT in that of its very low solubility* in sodium. (see the next page)
Solubility of Hydrogen and Carbon in Sodium

Temp. (°C)

Carbon

Hydrogen

Solubility (ppm)

【Solubility of Hydrogen and Carbon in Sodium】

出典：PNC PN9520 91-006、高速増殖炉技術読本、第10編第3章材料の液体金属ナトリウム腐食、P10-33、原発大洗工学センター、1991年7月
(2) Control Criteria of Impurities in Sodium in Monju

(a) Primary Cooling System

Oxygen Concentration (Design Limit ≤3ppm)

- According to the data of sodium soaking test (390～600°C) for 10,000h, it's verified that it's no problem in terms of decreasing of material's thickness since corrosion speed can be restrained under 10 μm/y if the oxygen concentration is kept less than 10ppm.

- While, oxygen concentration of 3ppm is applied to the fuel's design for evaluating of its structural strength.

- To maintain the plant safety, even 10ppm is no problem as a design value limit for sodium purity control, but 3ppm was employed as the design value limit in order to be more strictly control of sodium purity.

Besides Impurities

Tough not required to control on the design, the concentration of hydrogen and carbon in sodium are set to the following value to get hold of impurity's transfer behavior from cover gas system and behavior of the decarburization and carburization.

- Hydrogen: ≤ 170ppb
- Carbon : ≤ 30ppm
(b) Secondary Cooling System

**Oxygen Concentration (Design Limit ≤10ppm)**

- From the point of view for checking of material's corrosion amounts during in-service inspection, the design value is limited to 10ppm.

**Hydrogen Concentration (Design Limit ≤170ppb)**

- In the secondary cooling system, it’s necessary to watch hydrogen concentration in sodium in terms of the monitoring of SG heat pipe’s leak accident.
- The design value limit concerning hydrogen is set to 170ppb.
- This value is decided based on the background of water leak detection system for monitoring of the SG heat pipe’s leak, if it exceeded the limited value during plant operation, the operator should be stop the reactor operation because of a suspect broken heat pipe.
(3) Features & Specifications of Monju Cold Trap

Features

◆ The mesh part’s side of the Monju CT is a multi-hole plate.
◆ Therefore, as shown in the left picture, it’s possible that sodium flows into the mesh part from not only bottom but also side.
◆ This structure was designed based on the JOYO’s trouble which couldn’t to operate CT because of its pressure loss increased about 3 times (0.12MPa ⇒0.38Mpa) due to sodium flow pass was blocked by sodium oxide’s deposition at mesh part’s bottom area. This matter will be discussed later more detail in § 4.
Specifications

- Type: Forced N2 Gas Cooling Mesh Filling Type
- Number: 2
- Mesh Capacity: 1.5m³ (Effective Area: 0.96m³)
- Impurity Trap Capacity: 170kg O2 Conversion / Unit (Predicted Impurity Contamination Amount: 230kg)
- Inner Diameter 1.33m × Height 4.68m × Cylinder Plate Thickness 16mm
- Rated Flow Rate: 10t/h

出典：FBR広報素材資料集2版・上、「原子炉補助設備、純化系」 P19-8, 科学技術庁, 平成2年3月
Primary Sodium Purification System in Monju

<table>
<thead>
<tr>
<th>No</th>
<th>Temp. (°C)</th>
<th>Flow (t/h)</th>
</tr>
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<tbody>
<tr>
<td>1</td>
<td>529</td>
<td>10</td>
</tr>
<tr>
<td>2</td>
<td>180</td>
<td>10</td>
</tr>
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<td>6</td>
<td>129</td>
<td>10</td>
</tr>
<tr>
<td>7</td>
<td>40</td>
<td>10</td>
</tr>
<tr>
<td>8</td>
<td>35</td>
<td>73</td>
</tr>
<tr>
<td>9</td>
<td>38</td>
<td>73</td>
</tr>
</tbody>
</table>

Economizer

Cold Trap (B)

Cold Trap (A)

Overflow System

Circulation Pump

Auxiliary System (Cooling Water)

Cooler

[Primary Sodium Purification System in Monju]
(1) Principle of Plugging Meter (PL Meter)

By measuring the impurity’s saturation temp. (deposition temp.), it’s possible to get the impurity’s concentration in sodium based on its saturation curve. The PL meter utilizes this principle as well as the CT, it’s composed cooling and heating device, plugging-orifice, thermocouple and flow meter. Its basic system composition is described in the next page.
【Basic System Composition of Monju Plugging Meter】

出典：FBR広報素材資料集2版・上、「運転、計装制御、プラニング計」 P14-11、科学技術庁、平成2年3月
1. When sodium is cooled from the U point which impurity is unsaturated to the S point which is super-saturated, after from its temp. reached to the A point which is a saturation point on the curve, the sodium flow rate's decreasing will begin because the impurity's deposition is starting.

2. Next, when its temp. is risen from the S point toward to the U point, deposition will be continued until the B point due to its temp. is lower less than the saturation temp., but its flow rate will recover after its temp. exceeds the B point which saturated impurity's solution is starting.

3. The above mentioned A point is called the plugging temp. and B point is the unplugging temp..
The correlation between saturation solubility curve and its temp. mentioned in the previous page and CT’s inside distribution is able to be shown like in a left figure.
(2) Measuring Method of Plugging Meter

Measurement is carried out the following procedure.

1) Starting to cool sodium by blower’s operation. (Point X)
2) The sodium flow’s decreasing begins gradually causing the impurity’s deposition starts at the orifice. This point is called a “Plug Temp.”. (Point Y)
   ※ It doesn’t appear the flow’s decreasing phenomenon immediately even sodium temp. reached to its salutation temp. since there is a deposition delay which is caused by the impurity’s deposition crystallization.
3) Stopping blower’s operation when it confirms the flow’s decreasing phenomenon. As a result of this operation, orifice temp. will be risen.
4) Although the sodium flow’s decreasing is maintained for a while due to the crystallized impurity’s solution delay desolation, it will be changed for the increasing subsequently as rising of sodium temp.. This point is called a “Unplug Temp.”. (Point Z)
5) As a practical matter, the temp. at the flow rate is equilibrium state which is balanced deposition and solution of the impurity is employed as a “Plugging Temp.” for getting the impurity’s concentration. (See the next page, Monju performance test data)
According to the empirical knowledge of France:
- Saturation Temp. = Unplug Temp. − 5°C
- Plug Temp. = Plug Temp. + 20°C
Multiple Break Phenomenon

- The case in which several kinds of impurities exist in the sodium loop, the flow locus breaking will appear corresponding with each impurity’s saturation temp. This phenomenon is called “Multiple Break Phenomenon”.

- In early period of the secondary cooling system’s purification operation in the experimental fast reactor, JOYO, the double break phenomenon was observed clearly as shown in the right graph.

- In the right graph, it is supposed that higher side’ break is caused hydrogen and lower temp. side is by oxygen.

- As to the impurity’s deposition time, in turn, hydrogen is the earliest (a couple of few minutes), next is oxygen (20-30min) and other one is few hours.
The results of chemical analysis obtained from operation of the primary and secondary purification systems in JOYO, experimental fast reactor, during 1983 to 1999 for 17 years, are shown on the next page. As the reference, the primary purification system’s outline of JOYO is illustrated in Fig 4.1 along with its heat transfer system.

**[Primary Purification System]**

As mentioned in § 2 (3), the Monju CT in sodium can flow not only from the bottom direction of mesh part but also from its side direction. Therefore Monju CT will be used efficiently. Joyo’s earlier design CT had experienced plugging problems. This was remedied by a re-design which enable radial passage into as well as axial passage into of sodium through the mesh. This modification was done in 1998 and is reflected in Monju’s CT design.

Changing of oxygen concentration data in primary sodium during for before and after CT’s remodeling is described in fig. 4.3. The oxygen concentration in sodium was reflected by the newly designed cold trap. However, every data are maintained approximately from 1ppm to 7ppm, which are less than 10ppm of target control value, even before CT’s remodeling.

**[Secondary Purification System]**

The secondary system’s data are also satisfied control target value as shown in Fig. 4.4. These data are kept within 1ppm to 5ppm which are far less than target control value, 20ppm. Besides, Fig. 4.5 shows hydrogen data in secondary sodium as reference.
※Represented plant data are based on the MK-II core (old core) in Joyo.
Sodium flow was blockaded by sodium oxide’s deposition at bottom area of the JOYO first type CT in 1987.

Sodium oxide’s deposition will form equally along the axis of the mesh part in that sodium flows from its side direction.

【Fig. 4.2 Comparison between Before and After of CT’s Remodeling in Joyo】

出展：PNC ZN9430 92-009 高速実験炉「常陽」1次Na純化系運転経過報告書(4)、新型コールドトラップ低温運転経験と性能評価、1992年6月
Highest value is 6.8ppm (Just before remodeling of CT)

Data for remodeled CT were decreased clearly
(CT Target Control Temp. = 120°C)

【Fig. 4.3(1) Changing of Oxygen Concentration Data in Joyo Primary Sodium】
PL meter temp. in primary sodium system is measured with on line in succession.

【Fig. 4.3(2) Changing of Plugging Meter’s Temp. Data in Joyo Primary Sodium】
【Fig. 4.4(1) Changing of Oxygen Concentration Data in Joyo secondary Sodium 】
PL meter temp. in secondary sodium system is measured with off line manually.
【Fig. 4.5 Changing of Hydrogen Concentration Data in Joyo secondary Sodium】

出展：PNC ZN9410 88-061 高速実験炉「常陽」1次ナトリウム純化系運転経験報告書(3)、コールドトラップの運転・保守経験、1988年11月
Monju Sodium Leakage Accident & Its Safety Measures

Makoto Sawada

October 13, 2004

International Cooperation & Technology Development Center
Tsuruga Headquarter Office, JNC
Presentation Contents

§ 1: Sodium Leak Detection System in Monju
§ 2: Safety Design for Sodium leak in Monju
§ 3: Summary of Monju Sodium Leak Accident
§ 4: Cause of the Monju Accident
§ 5: Review & Lessons Learned from the Monju Accident
§ 6: New Sodium Leak Safety Measures in Monju
§ 7: Corrosion Mechanism of the Liner Plate by Leaked sodium

Appendix
Sodium Leak Accidents in the World’s FBR
## § 1: Sodium Leak Detection System in Monju

### Kinds of Detectors

<table>
<thead>
<tr>
<th>Kinds</th>
<th>Type</th>
<th>Location</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>SID</strong> <em>(Sodium Ionization Detector)</em></td>
<td>Gas Sampling Typed Detector</td>
<td>●R/V*¹</td>
</tr>
<tr>
<td></td>
<td></td>
<td>●Primary Loops</td>
</tr>
<tr>
<td><strong>RID</strong> <em>(Radioactive Ionization Detector)</em></td>
<td>Gas Sampling Typed Detector</td>
<td>●Secondary Loops</td>
</tr>
<tr>
<td><strong>DPD</strong> <em>(Differential Pressure Detector)</em></td>
<td>Gas Sampling Typed Detector</td>
<td>●R/V</td>
</tr>
<tr>
<td></td>
<td></td>
<td>●Primary Loops</td>
</tr>
<tr>
<td></td>
<td></td>
<td>●EVST*²</td>
</tr>
<tr>
<td>Contact Leak Detector</td>
<td>Contact Typed Leak Detector</td>
<td>●Primary Loops</td>
</tr>
<tr>
<td></td>
<td></td>
<td>●Secondary Loops</td>
</tr>
<tr>
<td></td>
<td></td>
<td>●EVST</td>
</tr>
<tr>
<td>Fire Alarm System</td>
<td>Smoke Sensor</td>
<td>●Secondary Loops</td>
</tr>
<tr>
<td></td>
<td></td>
<td>●EVST</td>
</tr>
<tr>
<td>Sodium Level Meter</td>
<td>Electromagnetic Induction Typed Level Meter</td>
<td>●R/V</td>
</tr>
</tbody>
</table>

*1 R/V: Reactor Vessel
*2 EVST: External Fuel Storage Tank

出典：FBR広報素材集第2版、17-4ナトリウム漏えい検出設備、科学技術庁、平成2年年3月
### SID (Sodium Ionization Detector)

<table>
<thead>
<tr>
<th>Method</th>
<th>SID detects a leak by monitoring a change of ion current produced by ionizing sodium aerosol under high temp., in an inert gas atmosphere.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feature</td>
<td>SID is suitable to detect sodium leaks under the inert gas atmosphere because a filament in detector will burn down in the case of using in air.</td>
</tr>
<tr>
<td>Sensitivity</td>
<td>[ \geq 1 \times 10^{-10} \text{g Na/cc} ] (minimum detectable)</td>
</tr>
</tbody>
</table>

Ion current amount is dependent on Na aerosol quantities flowing through the detector. Current will increase in accordance with increasing aerosol concentration.

---

出典：FBR広報素材集第2版、17-4ナトリウム漏えい検出設備、科学技術庁、平成2年年3月
**RID (Radioactive Ionization Detector)**

<table>
<thead>
<tr>
<th>Method</th>
<th>RID detects a leak by monitoring a change of current between two electrodes with gas flow, which depends on sodium aerosol quantities flowed into the ion chamber.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feature</td>
<td>RID is suitable for a detector using in air such as secondary cooling system.</td>
</tr>
<tr>
<td>Sensitivity</td>
<td>$\geq 1 \times 10^{-10}$ g Na/cc</td>
</tr>
</tbody>
</table>

The atmosphere in both chambers (standard and ion chamber) are always ionized by two $^{241}$Am sources. If Na aerosol flows into the ion chamber, current between both electrodes flowing will be reduced because movement speed of ion slows down. Thus it is possible to detect a sodium leak by monitoring a changing of potential difference of between both electrodes.

出典：FBR広報素材集第2版、17-4ナトリウム漏えい検出設備、科学技術庁、平成2年年3月
<table>
<thead>
<tr>
<th>Method</th>
<th>DPD detects a leak by monitoring a changing of difference pressure at the filter.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Feature</td>
<td>DPD is possible to apply both atmosphere conditions under inert gas or air.</td>
</tr>
<tr>
<td>Sensitivity</td>
<td>Increasing of 25% over normal base line</td>
</tr>
</tbody>
</table>

Since particles of sodium aerosol flowed through a detector will be absorbed into a filter, differential pressure will occur. Of course, their quantities depend on the aerosol amounts flowed into the detector.
Contact Leak Detector

<table>
<thead>
<tr>
<th>Method</th>
<th>Contact typed leak detector detects a leak by utilization of sodium property which has a good electrical conductivity.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Features</td>
<td>This detector is fitted to apply to sodium valves, sodium components such as tank, double typed piping, and so on.</td>
</tr>
<tr>
<td>Sensitivity</td>
<td>High Reliability</td>
</tr>
</tbody>
</table>

If sodium contacts both electrodes of the detector, the detector's circuit will be short-circuited or fault grounded. Thus it is possible to detect a sodium leak certainly.
## Sodium Level Meter

<table>
<thead>
<tr>
<th>Method</th>
<th>Sodium level meter measures a sodium level by means of electromagnetic induction property of sodium.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Features</td>
<td>This level meter is suitable to detect a large leak such as occurring a changing of sodium level.</td>
</tr>
<tr>
<td>Sensitivity</td>
<td>High Reliability</td>
</tr>
</tbody>
</table>

When sodium exists on outside of the detector, eddy current will be produced in sodium by current flowing on the primary coil. The eddy current makes the magnetic flux B which flows reverse direction for flux A. Consequently, it is able to measure a sodium level because the magnetic flux of A-B flows on the secondary coil which is in inverse proportion to the sodium level.

出典：FBR広報素材集第2版、14-4運転・計測制御、プロセス計装、科学技術庁、平成2年年3月
## § 2: Safety Design for Sodium Leak in Monju

### Safety Design for Primary System

<table>
<thead>
<tr>
<th>Leak Scale</th>
<th>Detectors</th>
<th>Identification</th>
<th>Reactor Operation</th>
</tr>
</thead>
</table>
| Very Small Scale¹) | ◇SID  
◇DPD                 | ●ANN + Field Check  
●ANN of Both Kinds                                   | Manually Scram                     |
| Middle Scale²)    | ◇R/V Na Level Meter  
◇O/T*¹ Na Level Meter | ●ANN(SID or DPD)  
●Meter’s Changing                                    | "                                   |
| Very Large Scale³) | ◇G/V*² Na Level Meter  
◇Under Floor Temp. in C/V*³ | ●ANN of Na Level Low  
Low for R/V or O/T  
●ANN of G/V Level Meter  
●ANN of High Temp. in C/V | Automatically Scram  
and Isolation of C/V |

1) : The alarm by the leak detectors only activate without changing process data.  
(Speculation Amounts>about 100kg)  
2) : Changing of process data such as sodium level appear. (Speculation Amounts>about 100kg)  
3) : Interlock activates immediately. (Guillotine Rupture)

* 1 O/T : Overflow Tank  
* 2 G/V : Guard Vessel  
* 3 C/V : Containment Vessel
The emergency level is set so that even in case of a sodium leak the primary circuit cooling is not interrupted. The components and piping are installed as high as possible relative to the emergency level ($E_{SL}$). Guard vessels are designed and provided to maintain a level higher than $E_{SL}$ in those portions of the circuit lower than $E_{SL}$. Furthermore, a siphon break function is provided in case a break occurs in the level lower than the $E_{SL}$.
### Safety Design for secondary System

<table>
<thead>
<tr>
<th>Leak Scale</th>
<th>Detectors</th>
<th>Identification</th>
<th>Reactor Operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Very Small Scale</td>
<td>RID</td>
<td>●ANN* + Field Check or ●ANN of Both Kinds</td>
<td>Manually Scram</td>
</tr>
<tr>
<td>Middle Scale</td>
<td>Fire Alarm System</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Very Large Scale</td>
<td>System Interlock</td>
<td>●ANN of Na Level Low Low in S/G) (or) ●ANN of primary Na Outlet Temp. High in IHX</td>
<td>Automatically Scram</td>
</tr>
</tbody>
</table>

* ANN: Indicator Announcement
### Safety Measures from Sodium Leaks

<table>
<thead>
<tr>
<th>Safety Measures for Producing Sodium Aerosol</th>
<th>Primary System</th>
<th>Secondary System</th>
</tr>
</thead>
<tbody>
<tr>
<td>Interruption of Sodium Combustion</td>
<td>Keeping Oxygen Concentration Less Than 2 %</td>
<td>Air (Nitrogen gas will be charge into in partial room when sodium leak occurs)</td>
</tr>
<tr>
<td>Produced Quantities of Sodium Aerosol</td>
<td>Very Few</td>
<td>Many</td>
</tr>
<tr>
<td>Prevention of Contact with Concrete*</td>
<td>Providing of Liner Plate (steel plate and air-tight structure)</td>
<td>Providing of Liner Plate (steel plate and no air-tight structure)</td>
</tr>
<tr>
<td>Safety Measures for Producing Sodium Aerosol</td>
<td>Confining Radioactive Aerosol in Containment Vessel</td>
<td>Preventing Diffusion of Aerosol by Turn Off Concerning Air Conditioning System</td>
</tr>
</tbody>
</table>

*If sodium reacts with concrete directly, the strength of the concrete will be reduced remarkably due to water reduction within the concrete. So, the steel liner (see in the next page) is provided in order to prohibit the reaction by leaked sodium. This problem will be discussed later in § 7 in detail.
【Conceptual Scheme of Liner Plate in the Monju Secondary System】

出典：「もんじゅ」安全性調査検討専門委員会提出資料、2章ナトリウム漏えい対策、図2-2-5、平成15年9月
A sodium leak accident in Monju happened at the secondary outlet piping of IHX in the C loop during the performance test on 40% power on Dec. 8, 1995.
### Progress of the Accident

**December 8, 1995**

<table>
<thead>
<tr>
<th>Time</th>
<th>Event Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pm 7:47</td>
<td>ANN「Secondary Outlet Temp. High of IHX in C Loop」 alerted and a little bit later 「Fire Alarm System (smoke sensors)」 activated.</td>
</tr>
<tr>
<td>Pm 7:48</td>
<td>ANN「Sodium Leak in C Loop」 alerted and also operators confirmed smoke appearing by field checking.</td>
</tr>
<tr>
<td>Pm 8:00</td>
<td>Power down operation was started.</td>
</tr>
<tr>
<td>Pm 8:50</td>
<td>Operators reconfirmed smoke increased by field checking. (approximately)</td>
</tr>
<tr>
<td>Pm 9:20</td>
<td>Reactor was scrammed manually.</td>
</tr>
<tr>
<td>Pm 10:40</td>
<td>Sodium drain operation for the secondary cooling system (C loop) was started. (Am 0:15</td>
</tr>
<tr>
<td>Pm 11:13</td>
<td>Air conditioning system in the sodium leak room was turned off.</td>
</tr>
</tbody>
</table>
Intermediate Heat Exchanger (IHX)
Reactor
Evaporator
Super Heater
Secondary Pump
Air Cooler

【Bird’s-Eye View for Secondary Cooling System】
Scene of Sodium Leak Accident

Broken Thermocouple

【Piled up sodium compounds on the floor with 3m radius and 30cm high】

【Eroded Air Duct by Falling Leaked Sodium】

【Eroded Grating Floor by Falling Leaked Sodium】

《Chemical Analysis of Sediment》
●Sodium Hydroxide (NaOH)
●Sodium Carbonate (Na₂CO₃)
Failure of a Tc Guide Tube

Guide tube (well) was broken by hydraulic vibration.

Sodium Flow Direction

Heat Insulator

Thermocouple

Fractured Surface

Sodium Leaked from Here

Broken Well
Effect to Environment

◆ There was no radioactive contamination of the environment.
   Since this accident occurred at the secondary loop which is non-radioactivity loop, of course, there was no effect of radioactivity to the environment.
   (In Japan, any reactor power plant accident, regardless of any radioactive release, requires an incident report to the local, prefecture and national authorities.)

◆ It was confirming that these were measurable, but insignificant amounts of sodium reaction by-products released to the environment via air conditioning system. In spite of this, there were no human exposures to or injure from the reaction compounds.
   By the result of chemical sampling analysis of aerosol stuck on the exhaust duct filters in the air conditioning system, sodium carbonate and sodium hydrogen carbonate were detected but their quantities were not so important. From this result, it was confirmed that small quantity of sodium compounds were released into environment.

Total Quantity of Leaked Sodium

◆ Total Quantity of Leaked Sodium = 640kg ± 42kg
◆ Leak Rate = 180kg/h (estimated before the beginning of the sodium draining operation)

出典：TN 1340 96-0031996.9 動燃技報No.99、1996.9

出典：「もんじゅ」安全性調査検討専門委員会提出資料、1章もんじゅ事故、1-4、平成15年9月
§ 4: Cause of the Monju Accident

Vibration by Symmetrical Vortex

When a column is put in perpendicular direction against the flow, it may vibrate by vortex generated at its back side as shown in the below figure.

Fatigue Failure by Hydraulic Vibration

Based on investigation results, a broken thermocouple (Tc) well assembly was exposed to severe high cycle fatigue caused by hydraulic vibration, and consequently its major to minor diameter transition portion broke finally.
The hydraulic vibration characteristic of a column put in perpendicular direction against the flow is dependent on the following two factors:

- Relative Relationship between a Frequency of Symmetrical Vortex and a Frequency of Column’s Inherent Vibration
- Column’s Damping Factor

【Remark】

The thermocouple of the broken Tc-well assembly was installed into its well with bending and touched to its inside wall directly.

⇒ Since by design there is no interaction between the well and the Tc in service during vibration, when there is contact the Tc’s damping factor will decline.
§ 5: Review & Lessons Learned from the Monju Accident

① Structural Problem of Tc in Secondary Cooling System
- Shape of the well of which stress concentrates
- Identifying safety measures in anticipation of when a Tc well is broken

② Delay of Correspondence for Sodium Leak Accident
- Delay of shut down corresponding due to misjudging for leak scale (Problems for monitoring performance and risk management ability of operators)
- Enhancing of spreading of aerosol by keeping operation of air conditioning system

③ Continuing of Sodium Leak for Long Time
- Beginning of draining operation required reduced system temperature
- Needing a lot of time for draining operation (Not easy to drain sodium in where horizontal inlet piping of secondary pump)

④ Lack for Measures for corrosion of Liner
- Revealing of possibility for corrosion of liner plate under the sodium burn
§ 6: New Safety Measures for Sodium Leak in Monju

① Improvement of Tc in Secondary Cooling System

Before

After

Cutting Length of Well and Changing to Taper Shape

Metal Seal

Weld Seal

Adding Contact Typed Leak Detector

Toshiba employed same type of Tc in LWR as applying to secondary loop’s Tc.
New typed is almost same type with primary cooling system’s Tc.
Monju was designed as an all-Japanese product by four large heavy industry companies.
Changing to Improved Type Tc in Secondary Cooling System

Notice) A broken Tc had already removed.
② Improvement of Monitoring System for Sodium Leak

Measure Policies for Sodium Leak

Detection of a leak as soon as possible and immediately shut down reactor and drain sodium

(1) Supplement of a Unification Leak Monitoring Panel

◆ Intensive expression of information regarding sodium leak and ITV camera pictures to a new panel located in main control room.

(2) Adding of a Leak Detection System called “Cell Monitor”

◆ More certain and quick detection of sodium leak
(1) Supplement of a Unification Leak Monitoring Panel

Leak Information
- Leak Detectors
- Process Data
  - Na Levels
  - System Temp.
  - Drain, Air conditioning

Fire Information
- Fire Detectors
- Cell Monitors

Visual Information
- ITV Camera

Data Processing
- ANN Logic
- Fire/Leak Data
- Process Data Transition Trend

Application Processing
- Identification of Fire Area

ITV Control Panel

Picture Data

Display Board (in Main Control Room)

Supporting of operator’s operation by intensive display of information related sodium leak to Unification Monitoring Panel

出典：「もんじゅ」安全性調査検討専門委員会資料、第2章ナトリウム漏えい対策、図1-1-10、平成15年9月
To detect a sodium leak quickly and certainly, a leak detection system named cell monitor which consists of smoke sensors and thermal sensors is newly provided.

If a sodium leak happened, immediately the cell monitor sends sodium leak information to a unification monitoring panel and also stops operation of concerned air conditioning system automatically.
③ Improvement of Draining System (Establishing of Quick Drain)

- Shortening Draining Time Operation (50min⇒20min)
- Improvement of Certainty for Drain
Additional Equipment of a Nitrogen Gas Injection System (Aggressive Sodium Fire Extinguishing)

Design Specification
Oxygen Concentration: less than 5%.

N2 Gas Supply System (Existing)

N2 Gas Storage Tank
In the case of large scale of leak accident, leaked sodium will be introduced into the storage room which is located in the lowest area of the building. At this time, it is feared that hydrogen gas concentration may exceed 4%*. That value is the limiting concentration of hydrogen explosion. The hydrogen can be produced by the leaked sodium reacting with humidity which comes out from the wall or ceiling concrete. So, in order to restrain the production of hydrogen gas, the heat insulators and the heat reducing materials will be provided as the Monju remodeling work.

※出典：石井、水素ガス爆発限界の圧力による測定、安全工学、P294-295、1964

※Remark

Hydrogen gas will be produced mainly by the following chemical equations. (1/2 mol of H2 with 1 mol of NaOH)

1) \(2Na + NaOH \rightarrow Na_2O + NaH\) (Sodium Hydride)
\(NaH \rightarrow Na + 1/2H_2\) (Thermal Decomposition)

2) \(Na + NaOH \rightarrow Na_2O + 1/2H_2\)

Like this, NaOH plays a key role in producing of hydrogen gas. Since it is produced by reacting with moisture, that is to say, it is important to restrain producing of moisture to avoid hydrogen explosion. In fact, NaOH plays a key role not only this problem but also in the corrosion of liner plate. (Discussed in the next section in detail.)
Unexpected Occurrence in OEC Sodium Leak Experiment

To study about the Monju leak accident in engineering in detail, a duplicate experiment was carried out in Oarai Engineering Center (OEC) in March, 1996. At that time, some holes penetrated the 6mm thick the liner plate during the experiment. Occurrences such as this were anticipated in the Monju design.

Penetrated Holes  
(Plate Thickness is 6mm)

※Plate thickness is only 1.5mm decreased in Monju Accident
By the result of causing investigation for rising of holes during a duplicating experiment, it was revealed that corrosion type is divided into the following types of two kinds. The environment conditions greatly influence whether corrosion progresses at which type.

- **Na-Fe Double Oxidization Type corrosion**

- **Molten salt Type corrosion**
This type corrodes a liner plate (iron)*1) by mixing leaked sodium and sodium reaction compounds (mainly Na₂O) produced during sodium combustion. In this type, as the sediment is almost covered with burning sodium, its surface parts of almost all become a lack of oxygen condition. This mean shows that sodium peroxide (Na₂O₂) which is a strong oxidizer seldom exists on there. Moreover, Na₂O₂ produced by sodium combustion is probably reduced to Na₂O*2) by remaining not yet reacted sodium. Thus, the corrosion with Na-Fe double oxide does not so progress quickly because Na₂O₂ of oxidizer seldom affects in corrosion mechanism. (See the next page: corrosion speed graph by Na-Double Oxide)

\[ \text{Fe} + 3\text{Na}_2\text{O} = \text{Na}_4\text{FeO}_3 + 2\text{Na}, \quad \text{Fe} + 4\text{Na}_2\text{O} = \text{Na}_5\text{FeO}_4 + 3\text{Na} \]

*1) Fe + 3Na₂O = Na₄FeO₃ + 2Na, Fe + 4Na₂O = Na₅FeO₄ + 3Na; these equations show iron corrosion
*2) Na₂O₂ + 2Na → 2Na₂O

出典：青砥、大気中ナトリウム漏洩下流部における鉄系材料の腐食機構、P37-39、動燃技報No.103、1997年9月
【Temperature Characteristic of Corrosion Speed for Na-Fe Double Oxidization Type】

出典：「もんじゅ」安全性調査検討専門委員会提出資料、2章ナトリウム漏えい対策、参考図1-1-3、平成15年9月
Under the high humidity condition, the corrosion will progress via the following processes.

1. The sediment, sodium oxide compound, produces sodium hydroxide (NaOH) by moisture absorption. The quantity of NaOH increases soon in a few minutes because of its deliquescence. Consequently, it is formed a corrosion environment of a solution on the liner plate.

2. The leaked unreacted sodium, floats on a solution and produces Na₂O and Na₂O₂, etc. during combustion.

3. They will dissolve into a solution and touch to the plate directly. That is to say, the plate is exposed to Na₂O₂ which is a strong oxidizer and will be corroded.

Though sodium plays a key role to reduce Na₂O₂ to Na₂O, but it becomes a source of supply of Na₂O₂ in this type. The corrosion in this type progresses quickly as shown in the next page.

\[
\begin{align*}
\text{Na}_2\text{O}_2 + \text{H}_2\text{O} &\rightarrow 2\text{NaOH} + \frac{1}{2}\text{O}_2 \\
\text{Na}_2\text{O} + \text{H}_2\text{O} &\rightarrow 2\text{NaOH}
\end{align*}
\]

Deformation of Liner by Thermal Expansion

NaOH + Na₂O + Na₂O₂

Fe dissolves out into solution with complex ion

\[
\begin{align*}
\text{Fe} + \frac{3}{2}\text{O}_2^{2-} &\rightarrow \text{Fe}^3\text{O}_4^{3-} \\
\text{Fe}^3\text{O}_4^{3-} + \frac{3}{4}\text{O}_2 &\rightarrow \frac{1}{2}\text{Fe}_2\text{O}_3 + \frac{3}{2}\text{O}_2^{2-}
\end{align*}
\]

these equations show iron corrosion

出典: 青砥、大気中ナトリウム漏洩下流部における鋼系材料の腐食機構、P39-44、動燃技報No.103、1997年9月
【Temperature Characteristic of Corrosion Speed for Molten Salt Type】

出典：「もんじゅ」安全性調査検討専門委員会提出資料、2章ナトリウム漏えい対策、参考図1-1-3、平成15年9月
Why holes penetrated the liner in OEC sodium leak experiment?

The experiment was performed under the following conditions which are very different from those of the Monju accident. At that time, the molten salt type corrosion was not known well. (See a graph in the next page)

1 Sodium Combustion with High Temp.
   To make sure of the TV camera’s sight for recording, forced air was blown into the room. As this result, the leaked sodium burned in a forced wind, consequently the liner plate temp. raised up to high temp. as about 800°C to 850°C in comparison with the Monju accident, less than 700°C.

2 Sodium Combustion in Small Area
   Also, the experiment leaked sodium amount is almost same with Monju accident carried out in small volume (≒170m³) compared with the actual room (≒2,300m³). As a result, much water came out from the concrete wall. From later analysis, it was revealed that the actual room's concrete wall temp. was almost 40°C except near leaked area, while the experiment was reached to high temp. as about 100°C in the almost of all area.

3 Corrosion under Condition of High Humidity and High Temp.
   As the mentioned in the former section, humidity plays a key role in the type of corrosion, i.e., the corrosion environment of solution by sodium hydroxide and sodium peroxide. It is clear that in the case of Monju, this is not a molten salt type corrosion.
According to the analysis, it seemed to penetrate the holes for about two hours in the OEC experiment.
It was verified that the experiment was a completely different type of corrosion then that of the Monju accident.

Comparison between Monju Accident and OEC Experiment

**OEC Experiment** (6mm penetrated)  
- Room Volume: ≡170m³
- Concrete Wall Tem.: All area 100°C
- Discharge Quantity of Water from Concrete Wall & Air Conditioning System: ≡300kg
  - * estimate: (≡220kg)*
  - (≡80kg)*

**Monju Accident** (1.5mm decreased)
- Room Volume: ≡2,200m³
- Concrete Wall Tem.: 40°C (except near leaked area)
- Discharge Quantity of Water from Concrete Wall & Air Conditioning System: ≡170kg~200kg
  - (≡170kg~200kg)*

Moisture Absorption by Aerosol

Contributed Water Amount for Producing of NaOH

- NaOH + Na₂O + Na₂O₂
- It was verified that the experiment was a completely different type of corrosion then that of the Monju accident.

Na + Na₂O

【Molten Salt Type】

【Na-Fe Double Oxidization Type】

出典：「もんじゅ」安全性調査検討専門委員会提出資料、1章もんじゅ事故、図2-2-11、平成15年9月
Although providing of the heat insulators to ceiling and wall in the storage room in order to avoid occurrence of hydrogen gas explosion by leaked sodium had been mentioned in section 6, this measure also will be carried out for other rooms where sodium piping is located as one of Monju remodeling working. This is a very useful way not only the above matter but also to prevent occurring of the molten salt type corrosion on the liner plate by leaked sodium. To be more sure of Monju’s safety, its work will be performed even if it is speculated that the molten salt type corrosion may never actually occur because actual environment conditions are remarkably different compared with molten salt type condition.
What happens if leaked sodium is penetrating into liner plate?

A lot of experiments regarding sodium-concrete chemical reaction were carried out in many countries such as France, United Kingdom, U.S.A as well as Japan. The principle results obtained from these experiments are as followed:

1. The sodium-concrete reaction seldom happens under condition of low sodium temp. such as 300°C to 400°C, it begins at higher temp. i.e. over 500°C.
2. The erosion of concrete stopped at maximum 30 cm as shown in the table below since the produced reaction compounds make a layer on the surface of concrete and it disturbs the reaction.
3. As the hydrogen gas produced by the reaction burned immediately, it is seldom stored in the room.

<table>
<thead>
<tr>
<th>Reference</th>
<th>Erosion Data</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Japan (JNC)</td>
<td>9-18 cm</td>
<td>JNC, Japan Nuclear Society, (90,91)</td>
</tr>
<tr>
<td>USA (HEDL)</td>
<td>3.8-30 cm</td>
<td>Colburm, ENS Int. Mtg. (79)</td>
</tr>
<tr>
<td>French (CEA)</td>
<td>5-14 cm</td>
<td>Casselman, Nucl. Eng. Des. (81)</td>
</tr>
</tbody>
</table>

Reaction Compounds (Na₂SiO₃)

Only few cm on surface eroded (T:30cm, D:18cm)

【Experiment in our Training Facility (Sep., 2003)】
## Other Sodium Leak Accidents in the World FBR

<table>
<thead>
<tr>
<th>Object</th>
<th>Plant Name (Country)</th>
<th>Date</th>
<th>Outline and Cause</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>DFR (U.K.)</td>
<td>May 1967</td>
<td>NaK leakage occurred at the branch pipe weld (T-pipe section) of the reactor inlet piping. It was made clear that the cause was a defect in the welding work and stress concentration (thermal stress).</td>
<td>Appendix-1</td>
</tr>
<tr>
<td></td>
<td>Rapsodie (France)</td>
<td>October 1978</td>
<td>An insignificant amount of radioactive sodium aerosol was detected at the double-pipe section of the primary system, at half power. This phenomenon disappeared when the reactor was running. Although it was assumed that the leakage occurred on the upper part (gas phase) of the reactor vessel, causes were not specified and the leak has never been identified.</td>
<td></td>
</tr>
<tr>
<td>Primary system</td>
<td>BN-350 (Kazakhstan)</td>
<td>January 1982</td>
<td>A drop in the fluid level (approximately 100mm) in the reactor vessel equivalent to the volume of C/T and a drop in the cover gas pressure were observed after preheating the C/T piping of the primary purification system. Immediately after that, leakage was detected at the joint section of the C/T piping because a rise of white smoke was (visually) observed. The leakage amounted to approximately 1kg.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>BN-600 (Russia)</td>
<td>October 1993</td>
<td>A crack occurred in the joint (T-pipe section) between the piping of the primary purification system and the branch pipe. The cause was that the purification system piping was not designed to absorb thermal expansion displacement and a defect in the valve of the branch pipe caused low-temperature sodium to flow, resulting in a disparity in thermal expansion, which generated excessive stress in the joint section.</td>
<td></td>
</tr>
</tbody>
</table>
Appendix-1: Sodium Leakage Accident in the Primary System of DFR (U.K.)

1. Details of the Accident
During full power operation, a leakage of NaK, which was primary coolant, was detected. It was a leakage of 100 or 200 liters per day and it stopped when the reactor was shut down, which made it impossible to specify the leakage spot. Thus, as a result of spending approximately half a year to make investigations using various methods, such as helium gas leakage testing, trace testing using gold dust, and acoustic measurement by bubble sound, it was confirmed that the leakage occurred at a spot a little away from the reactor vessel inlet nozzle in the reactor inlet piping section of one of the 24 loops of the cooling system (see Figures 1 and 2).

2. Cause
Cutting off the piping section in question discovered a crack due to fatigue fracture near the T-weld with the branch pipe of the main piping [the return pipe from the hot trap (NaK impurity removal system)]. The cause was assumed to be a combination of a defect in the welding work (insufficient weld penetration) and the concentration of thermal stress by the T-pipe.

3. Actions and Measures
There were six other T-pipe sections identical to the damaged section. Operation was restarted about a year later after performing difficult work, such as cutting off all the pipes in question, rerouting the piping, and re-welding.

4. Measures Taken for “Monju”
At “Monju,” these problems are prevented by taking every possible measure to ensure the soundness of wells through sufficient quality control. For example, when manufacturing equipment and piping that compose the reactor coolant boundary (boundary in which primary system sodium is contained), scientific, mechanical, and non-destructive tests are conducted with respect to the materials, and non-destructive and pressure-resistant tests are conducted in the manufacturing process, as required under the applicable laws. The T-section is designed to have a structure that causes no temperature differences in the joint section, thereby mitigating the thermal fatigue of the piping. In the event that leakage should occur, it can be detected quickly and certainly.
Figure 1: NaK Leakage Spot in the Primary System of DFR
Figure 2: NaK Leakage Spot in the Primary System of DFR (Details)
<table>
<thead>
<tr>
<th>Object</th>
<th>Plant Name (Country)</th>
<th>Date</th>
<th>Outline and Cause</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Secondary system</td>
<td>Rapsodie (France)</td>
<td>October 1966</td>
<td>There were two insufficiently preheated sections in the piping for injecting sodium into the secondary system. The expansion of the volume of the sodium that stayed behind in the middle resulted in a rupture of the piping. Since injection was continued without being able to detect the rupture, sodium entered through the double pipe up to the outer casing of the intermediate heat exchanger.</td>
<td>Appendix-2</td>
</tr>
<tr>
<td>Secondary system</td>
<td>BN-350 (Kazakhstan)</td>
<td>December 1974</td>
<td>Since the on/off of the preheater associated with the repair of the drain system piping was not recorded and transferred precisely, the piping was heated locally by mistake. The seal weld of the valve was damaged due to the thermal expansion of sodium. Moreover, since the piping connected to the drain tank was preheated, the sodium contained in the drain tank gushed out from the valve. The leakage was detected by the fire alarm. (There were eight instances other than this).</td>
<td></td>
</tr>
<tr>
<td>Secondary system</td>
<td>Monju (Japan)</td>
<td>December 1995</td>
<td>During performance testing (output up testing), the thermometer protection tube (sheath) was fractured, which was installed on the outlet piping section of the main intermediate heat exchanger (IHX) of Loop C of the secondary main cooling system. This lead to sodium leakage. The reactor was shut down by a manual scram performed by the operators. The amount of leakage was approximately 640kg-680kg.</td>
<td></td>
</tr>
</tbody>
</table>
1. Details of the Accident
When the entire secondary system was preheated to re-inject sodium into the secondary system, the injection piping (extending from the drain tank to the intermediate heat exchanger) was ruptured. As a result of continuing sodium injection without noticing this, the piping section and the double-structured section of the intermediate heat exchanger were soaked with sodium.

2. Cause
The injection piping was preheated to inject sodium although the sodium, which is supposed to have been drained previously, remained solidified in the injection piping. Two spots in the piping remained at low temperature because of inadequate preheating control. There was no escape for the sodium left behind between these two low-temperature spots when it was subjected to thermal expansion because of preheating, resulting in a pressure rise, which ruptured the piping (see Figure 1).

The ruptured piping had a double-pipe structure, which made it impossible to detect the rupture. Thus, sodium injection was continued without noticing the rupture. The sodium entered the double-pipe section and reached the gap between the intermediate heat exchanger vessel and its outer casing.

3. Actions and Measures
It took approximately six weeks to remove the sodium from the double-pipe section and the gaps in the intermediate heat exchanger. As recurrence prevention measures, improvements have been made, such as increasing the number of thermocouples to prevent low-temperature sections during preheating.

4. Measure for “Monju”
As to “Monju,” combined with the preheating control system using electric heaters, numerous thermocouples are installed on the sodium drain and injection piping to ensure sodium drainage and prevent the occurrence of low-temperature sections that becomes problematic during preheating. Moreover, sodium leakage sensors have been installed to detect sodium leakage early enough to take appropriate actions even if sodium leakage should occur.
Figure 1: Sodium Leakage at the Double-Pipe Section of the Sodium Injection Piping of the Secondary System of Rapsodie
<table>
<thead>
<tr>
<th>Object</th>
<th>Plant Name (Country)</th>
<th>Date</th>
<th>Outline and Cause</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>BN-350 (Kazakhstan)</td>
<td>May 1973</td>
<td>While heating the sodium acceptance transport tank, sodium leakage from the transport pipe joint section was visually confirmed. The amount of leakage was approximately 200g.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Phenix (France)</td>
<td>July 1976</td>
<td>Due to a thermal expansion disparity between the down tube inside the intermediate heat exchanger and the trunk of the intermediate heat exchanger, a crack occurred in the weld connecting the top lid of the secondary sodium outlet with the internal wall, resulting in sodium leakage in the gap of the double-pipe with some leaked out from the top section.</td>
<td></td>
<td></td>
</tr>
<tr>
<td>S-Phenix (France)</td>
<td>March 1987</td>
<td>Sodium leakage occurred in the external fuel storage tank. The cause of the leakage involved hydrogen (presumed to have been generated as a result of a reaction between the rust generated in the storage tank and sodium, or resulted from residual moisture because of hydraulic testing at the time of manufacture), which was contained in the materials (carbon steel). Residual stress when the storage tank and the cooling coil support plate were welded resulted in a crack, which subsequently developed into a serious one.</td>
<td>Appendix-3</td>
<td></td>
</tr>
</tbody>
</table>

出典：FBR広報素材集第2版、57.FBRのトラブルと事故、科学技術庁、平成2年年3月
Appendix-3: Sodium Leakage Accident at EVST of Super-Phenix (France)

1. Details of the Accident

1) In March 1987, sodium leakage occurred from the external fuel storage tank (storage container for spent fuel, etc.). Volume of sodium: Approximately 700 tons. Diameter: Approximately 9.5m. Height: 13m. Material: Carbon Steel). Initially, the leakage rate was 20 liters per hour, which stopped in the middle of April. It was presumed that it was because the viscosity of sodium increased due to a drop in temperature. It was reported that the total volume of leakage was 20m3.

2) The external fuel storage tank was a double-layer container. Leaked sodium was accumulated in the outer container. The leaked sodium was not radioactive and did not burn because it was in an inert gas atmosphere (normally nitrogen gas is used, but it was replaced with argon gas after the leakage occurred), having no impact on the general public and environment. The external fuel storage tank was separated from the reactor, having no immediate impact on the reactor function.

3) The reactor was shut down in May to examine the leakage spot. First, mock-up fuel assemblies and other were removed from the external fuel storage tank, which was completed in July. Next, the sodium remained in the external fuel storage tank was slowly drained. The argon gas, which was cover gas over the sodium, was replaced with helium gas. A helium gas sensor was installed in the gap between the external fuel storage tank and the protective container. Sensor signals were used to specify the height and direction of the sodium leakage spot. As a result, in September, it was found that the leakage spot was the weld of the plate of the cooling system support rack of the external fuel storage tank, which was approximately 3m high from the bottom of the external fuel storage tank (see Figure 2). Furthermore, several cracks were discovered other than the leakage spot by X-ray examinations, etc.

2. Cause

In January 1988, metal pieces of the section in which the crack existed were collected and subjected to metal analysis by Electricite de France (EDF) and CEA. As a result, it was concluded that the cause of the crack involved hydrogen (presumed to have been generated as a result of a reaction between the rust generated in the external fuel storage tank and sodium, or resulted from residual moisture because of hydraulic testing at the time of manufacture), which penetrated in to the metal, and the crack developed due to residual stress when the fuel storage tank and support plate were welded. The choice of the specific steel Feriritic steel 15D3 is responsible of this incident.
3. Actions and Measures
Since it was difficult to repair the crack, instead of using the external fuel storage tank as a spent fuel storage container, it was converted into a fuel relay transport system used simply as a “path” of spent fuel by filling the tank with argon gas instead of sodium. After conducting the re-inspection of reactor vessel weld shuts, operation was restarted following permission granted in January 1989.

4. Measures for “Monju”
The external fuel storage tank of “Monju” is not made of carbon steel, but of austenitic stainless steel that has high sodium corrosion resistance and high-temperature strength. In addition, sufficient quality control is implemented to prevent similar problems from occurring by employing weld structures creating little residual stress, performing thermal treatment after welding, conducting pressure resistance testing using gas pressure instead of water pressure.
Figure 1: Reactor Structure and External Fuel Storage Tank of Super-Phenix

Appendix-3
Figure 2: Leakage Spot in External Fuel Storage Tank of Super-Phenix

External fuel storage tank
Cooling system support rack
Leakage spot
Protective container

Appendix-3

出典：FBR広報素材集第2版、57-18FBRのトラブルと事故・燃料取扱い系の例、図57-18(2)、科学技术庁、平成2年年3月
World Fast Breeder Reactor (FBR) Information

Hironori OSHITA

October 2004
International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
Current status of FBR in the world

1. United States
2. United Kingdom
3. France
4. Germany
5. Russia
6. Kazakhstan
7. India
8. China
9. South Korea
10. Japan
11. Italy
12. Brazil
13. Europe

Quoted from MapQuest
Introduction

It is not LWR but FBR that first extracted electricity from nuclear energy. Needless to say, the early object of developing FBR was making the most use of uranium by producing plutonium. However recently, surplus plutonium and high cost of constructing FBR have come to cause the reduction and political change on FBR developing. This section mentions the past circumstances and current status of FBR being researched in the world.
1. United States

- FFTF (Richland)
- EBR-I, EBR-II (Idaho Falls)
- CRBR (Clinch River near Oak Ridge)
- SEFOR (Fayetteville)
- Clementine, LAMPRE (Los Alamos)

Quoted from MapQuest
(1) Early research and development for FBR in U.S.

The United States of America embarked on research and development of FBR in early 1940’s. The first FBR named Clementine was constructed at Los Alamos (criticality in 1946). In 1951 the experimental FBR named EBR-I succeeded in generating electricity (earlier than LWR). Liquid fuel was also researched at LAMPRE in 1960s and almost coincidently EBR-I was reconstructed into tank type of FBR named EBR-II. The thermal power was enlarged by the reactor named Enrico Fermi (criticality in 1963) and oxide fuel was researched at SEFOR (criticality in 1972). In 1980 irradiation research reactor FFTF (Fast Flux Test Facility) was constructed and prototype FBR named CRBR was designed.

(2) The political change by President Carter

The president Carter worried about the nuclear test of India and exportation of reprocessing facility from Germany and France to Latin America. He issued a statement of freezing of harnessing plutonium based on nuclear nonproliferation. In accordance with this statement the plan of constructing commercial reprocessing facility and prototype FBR i.e. CRBR was called off.
(3) Conversion of research and development of FBR

The U.S. determined to research and develop a small-sized modular standardized fast reactor with metallic fuel containing minor actinides i.e. Np, Am and Cm etc. combined with dry reprocessing unit, which matches the nuclear non-proliferation principle and cost performance. Consequently PRISM was designed as this type of the reactor, which was referred to by the FBR in Korea and Brazil.

(4) Current status

The Soviet Union collapsed in 1991 and overflow of the plutonium that had been used for nuclear weapon became an outstanding problem. In 1993 the President Clinton determined to freeze harnessing plutonium again and research and development of nuclear cycle including FBR were called off. However the plutonium that had been used for nuclear weapon should be disposed. For these reasons the policy on using plutonium has been changed into burning in light water reactor since 1997.

Quoted from JNC Home Page, ATOMICA
The first fast reactor in the world was designed and built in the State of New Mexico in 1946. It was called Clementine, the reactor fuelled with metallic plutonium and cooled by liquid mercury. Its thermal power was 25 kWt. Clementine served as a source of fast neutrons for research. It was operated for approximately 6 years.

Quoted from JNC Home Page
Although the current main nuclear power plants are light water reactors, the first electric generation extracted from nuclear energy was performed by EBR-I, the fast reactor. It was on December 20th in 1951. Its electric power was 200kWe. An alloy of sodium and potassium was used as the coolant. And its fuel was enriched metallic uranium or an alloy of enriched uranium and zirconium. In November 1955 an accident of melting a part of core occurred. However the core was decontaminated and modified taking two years and restarted. EBR-I had been operated until 1963.

*Special mention
Generated electricity from nuclear energy in the world as of December 31, 1999

- Light Water Reactor: 313,433,000kWe (87.20%)
  - PWR: 232,267,000kWe (64.62%)
  - BWR: 81,166,000kWe (22.58%)
- Heavy Water Reactor: 16,927,000kWe (4.71%)
- Light Water Graphite Reactor: 15,300,000kWe (4.26%)
- Gas Cooed Reactor: 12,915,000kWe (3.59%)
- Fast Reactor: 85000kWe (0.24%)

Quoted from JNC Home Page, ATOMICA
(3) LAMPRE

LAMPRE (Los Alamos Molten Plutonium Reactor Experiment) achieved criticality in 1961. The used fuel was a mixture of molten plutonium and iron. The liquid fuel was held in vented cans around which sodium flowed and received the heat. LAMPRE was performed for about 3 years.

*For reference
At present another type of liquid metallic fueled reactor is being researched. The coolant is helium, which becomes high temperature and high pressure and directly rotates the turbine. The nuclear fuel may burn perfectly and the fission products may be separated by the specific gravity difference, which makes the reprocessing unnecessary. High efficiency of converting nuclear energy to electricity, high burning uranium and high breeding ratio may be expected.
The experimental fast reactor EBR-II achieved criticality in 1961. It generated 20 MWe of electricity while being used for irradiation tests. EBR-II has its core, electricity generating system, fuel manufacturing facility and reprocessing facility on the same site. Every kind of intrinsic safety characteristics was researched and much information was accumulated. EBR-II was operated until 1994.
Enrico Fermi achieved criticality in August 1963. Its aim was to ascertain a large scale of FBR operation and electricity generation and to evaluate its economic ability. In August 1966 a fraction of the lower part of the core obstructed the coolant flow and a part of the core melted down. The core was modified and the operation restarted in July 1970. In November 1972 it was determined to shut down and the decommissioning was finished in December 1975.
The Southwest Experimental Fast Oxide Reactor (SEFOR) achieved criticality in 1969. The main aim of this reactor was to test the dynamic response of oxide fuel to various transients.
FFTFF achieved criticality in February 1980. Its thermal power was 400MWt. The irradiation test started in April. Since then it had been used mainly for irradiation test of fast reactor’s fuel and structural material until April 1992. Although FFTF had accumulated much information, it was determined to shut down due to financial reasons. Since 1993 FFTF has been maintained costing 40 million dollars per year. It is now being actively decommissioned. The cost of decommissioning is more than 40 million dollars.
## Table 1 Fast reactors in United States

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>Clementine</td>
<td>Experimental reactor, Loop type, Hg coolant</td>
<td>Metallic Pu</td>
<td>1946</td>
<td>25kWt</td>
</tr>
<tr>
<td>EBR-I</td>
<td>Experimental reactor, Loop type, NaK coolant</td>
<td>U, U-Zr</td>
<td>1951</td>
<td>1.2MWt 0.2MWe</td>
</tr>
<tr>
<td>LAMPRE</td>
<td>Experimental reactor, Loop type</td>
<td>90%Pu-10%Fe (Liquid fuel)</td>
<td>1961</td>
<td>1MW t</td>
</tr>
<tr>
<td>EBR-II</td>
<td>Experimental reactor, Tank type</td>
<td>Metallic U</td>
<td>1961</td>
<td>62.5MWt 20MWe</td>
</tr>
<tr>
<td>E.Fermi</td>
<td>Experimental reactor, Loop type</td>
<td>U+10%Mo</td>
<td>1963</td>
<td>200MWt 61MWe</td>
</tr>
<tr>
<td>SEFOR</td>
<td>Experimental reactor, Loop type</td>
<td>Oxide fuel</td>
<td>1969</td>
<td>20MWt</td>
</tr>
<tr>
<td>FFTF</td>
<td>Experimental reactor, Loop type</td>
<td>PuO₂ ·UO₂</td>
<td>1980</td>
<td>400MWt</td>
</tr>
<tr>
<td>CRBR</td>
<td>Prototype reactor, Loop type</td>
<td>PuO₂ ·UO₂</td>
<td>-</td>
<td>975MWt 380MWe</td>
</tr>
<tr>
<td>PRISM</td>
<td>Small-sized modular standardized reactor, Tank type</td>
<td>U-Pu-Zr, U-TRU-Zr</td>
<td>-</td>
<td>840MWt 310MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
2. United Kingdom

DFR, PFR (Dounreay)

Quoted from MapQuest
Circumstances of FBR development in U.K.

(1) Past research and development of FBR in U.K.
United Kingdom started operating experimental fast breeder reactor DFR (Dounreay Fast Reactor) in 1959 and prototype fast breeder reactor PFR (Prototype Fast Reactor) in 1974 respectively. However United Kingdom has much petroleum, coal and natural gas. In late 1980s the government estimated the date of the practical use of FBR would be in the middle of 21st century, in early 1990s cut down the budget for FBR and in 1994 determined to abandon the research and development of PFR.

(2) Current status of research and development of FBR
The research and development of FBR was entrusted to private enterprise. The industrial circles in United Kingdom participates in the EFR (European Fast Reactor) project. The conceptual design of EFR was finished in 1993. United Kingdom also participates in the new project in France and cooperates in researching reactor physics, reactor safety and nuclear fuel. The aim of the new project is burning of excess plutonium and transmuting long life radioactive wastes using FBR.
FBR Plant in U.K. (1/2)

(1) DFR (Dounreay Fast Reactor)

The experimental reactor DFR achieved criticality in 1959. The operation lasted until 1977. Although DFR experienced sodium leak accident, it accumulated fuel irradiating and electricity generating data. These data was reflected the prototype reactor PFR. The role of DFR was finished and it was shut down in 1977.

Quoted from JNC Home Page
(2) PFR (Prototype Fast Reactor)

The prototype reactor PFR achieved criticality in 1974. Although it experienced steam generator trouble, the operation lasted until 1994 as a power plant and irradiation facility. PFR succeeded in cooling decay heat by natural convection and in high burning of fuel. At present decontamination and decommission are being performed. Making sodium inactive started. Cesium clearing equipment is connected to sodium disposing plant. Sodium and cesium are to be disposed together.

Quoted from JNC Home Page
Table 2 Fast reactors in United Kingdom

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>DFR</td>
<td>Experimental reactor, Loop type</td>
<td>U +7%Mo</td>
<td>1959</td>
<td>60MWt 15MWe</td>
</tr>
<tr>
<td></td>
<td>NaK coolant</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PFR</td>
<td>Prototype reactor, Tank type</td>
<td>PuO$_2$ +UO$_2$</td>
<td>1974</td>
<td>600MWt 270MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
3. France

- Rapsodie (Cadarache)
- Phenix (Marcoule)
- Super-Phenix (Creys Malville)

Quoted from MapQuest
Circumstances of FBR development in France(1/2)

(1) Past research and development of FBR in France

France started operating the experimental reactor Rapsodie in 1967. The prototype reactor Phenix and demonstration reactor Super Phenix followed. At Phenix the data on operation, maintenance and fuel irradiation has been accumulated for 20 years. The demonstration reactor Super Phenix achieved criticality in 1985 and accomplished nominal electric power 1240MWe in 1986. However also in France the policy for research and development of FBR was converted. It has come to be considered that the urgent problem is not producing plutonium but burning it together with minor actinides. In 1994 the French government decided to use Super Phenix for burning plutonium (CAPRA project), for burning minor actinides and transmuting long life fission products (SPIN project) and for testing the ability of the plant. After some small troubles Super Phenix restarted in 1994 and power raising test was well performed from 1995 to 1996 (16 weeks). However the new government formed in June 1997 (in a coalition with the ecologist party) considered that Super Phenix does not agree with the economy and submitted a report of shutting it down to the parliament. In April 1998 a committee for discussing this problem was set up. In July 1998 the committee submitted a report mentioning the final shutdown of Super Phenix. According to this report the prime minister Jospin permitted decommissioning Super Phenix.

Quoted from JNC Home Page, ATOMICA
(2) Current status of research and development of FBR in France

The CAPRA and SPIN projects were supposed to be performed at Super Phenix together with other research on FBR as an irradiation test facility and a fast neutron source. However in accordance with the decommissioning Super Phenix, these projects were transferred to Phenix and the new project was established aiming at surveying burning plutonium, burning minor actinides in plutonium core (non-uranium core) and in MA burning specialty core and transmuting long life fission products loaded as a target in heterogeneous plutonium core that has no uranium. This new project of burning of minor actinides will be done with several irradiations equipment at Phenix plant until 2008. In this subject, France has settled technical cooperation with Japan and USA to form a trilateral project.

Quoted from JNC Home Page
(1) Rapsodie

(a) Circumstances of the research and development

The experimental reactor Rapsodie had been operated as an irradiation facility since the first criticality in 1967. High burning of fuel has been achieved and various safety tests have been performed for the next prototype reactor e.g. Phenix. Its role as the experimental reactor was over and Rapsodie was shut down in 1983.

(b) Current status

At present the decommissioning of second stage is finished. Third stage of decommissioning is under study.

*Special mention

In 1994 during the sodium removing work by alcohol, an explosion occurred because of the high pressure of generated hydrogen and hydrocarbon originated from thermal decomposition of sodium alcolate.

Quoted from JNC Home Page
The prototype reactor Phenix achieved criticality in 1973. Since then fuel irradiation data had been accumulated although it experienced sodium leakage and abnormal reactivity. The designed plant life 20 years have passed and modification for life extension by 10 years of operation was considered. The official committee determined to combine the former CAPRA project and SPIN project to establish the new project. These projects that had been supposed to be performed at Super Phenix were transferred to Phenix. And also irradiation tests that are related to highly qualified fuel are being planned at Phenix. For these projects Phenix will operated until 2008. In 2004 Phenix celebrated its 30th anniversary.

Quoted from JNC Home Page
The largest demonstration sodium fast reactor in the world Super Phenix achieved criticality in September 1985 and generated electricity in June 1986. In December 1986 its nominal electric power 1240MWe was accomplished. On the other hand Super Phenix experienced some troubles - sodium leakage from extra vessel fuel storage tank, air inlet into cover gas inside the core and partial damage of the turbine building due to a pile of snow. In 1994 the French government determined to convert the aim of the Super Phenix from generating electricity to researching parameters of FBR. However as mentioned before it was determined to decommission Super Phenix due to political agreement with the green party and the planed research was transferred to Phenix.
### Table 3 Fast reactors in France

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rapsodie</td>
<td>Experimental reactor, Loop type</td>
<td>PuO₂ · UO₂</td>
<td>1967</td>
<td>40MWt</td>
</tr>
<tr>
<td>Phenix</td>
<td>Prototype reactor, Tank type</td>
<td>PuO₂ · UO₂</td>
<td>1973</td>
<td>563MWt 250MWe</td>
</tr>
<tr>
<td>Super-Phenix</td>
<td>Demonstration reactor, Tank type</td>
<td>PuO₂ · UO₂</td>
<td>1985</td>
<td>3000MWt 1240MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
4. Germany

SNR-300 (Kalkar)

KNK-II (Karlsruhe)

Quoted from MapQuest
Circumstances of FBR development in Germany

(1) Past research and development of FBR in Germany

The thermal neutron reactor KNK-I was reconstructed as an experimental fast breeder reactor KNK-II. Although the prototype reactor SNR-300 was constructed, there was no prospect of getting permission for loading fuel from the state. Furthermore financial problem made the German government give up developing FBR.

(2) Current status related to FBR

At present Germany participates in the new project in France as a cooperation for researching and developing FBR.

Quoted from JNC Home Page
(1) KNK-II

KNK-II achieved criticality in October 1977 and generated 20MWe electricity in March 1979. The aim of KNK-II was to accumulate experiences of operating FBR and to perform irradiation tests of fuel and structural material. KNK-II accomplished the role and was shut down on August 23 1991 and have been decommissioned since 1994.

*Special mention

Although KNK-I was a thermal neutron reactor, its coolant was sodium. Only the core was reformed to the fast breeder reactor and became KNK-II.

Quoted from JNC Home Page
The construction of SNR-300 was almost accomplished at the end of 1986. However in July 1987 the government of Germany froze the operation of SNR-300 due to the problem of budget and no permission of loading the fuel from the state. In 1991 the government formally determined to abandon the development of SNR-300. The fuel assemblies for SNR-300 are stored in Hanau. 82 of them are already carried to Dounreay U.K. and in succession 123 assemblies are announced to be carried.
# Table 4 Fast reactors in Germany

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>KNK-II</td>
<td>Experimental reactor Loop type</td>
<td>PuO$_2$ +UO$_2$</td>
<td>1977</td>
<td>58MWt 20MWe</td>
</tr>
<tr>
<td>SNR-300</td>
<td>Prototype reactor Loop type</td>
<td>PuO$_2$ +UO$_2$</td>
<td>-</td>
<td>762MWt 327MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
5. Russia

- BOR-60 (Dimitrovgrad)
- BN-600, 800 (Beloyarsk)
- BR-10 (IPPE near Obninsk)
- BN-800 (Celyabinsk)

Quoted from MapQuest
Circumstances of FBR development in Russia

(1) Past research and development of FBR in Russia
Since the period of the former Soviet Union, Russia has been pushed the FBR project. The research started in 1955. The constructed experimental reactors are BR-1 (100Wt), BR-2 (100kWt, Hg coolant), BR-5 (5MWt) and BR-10 (10MWt). The prototype reactor BN-350 is now under the management of Kazakhstan Republic. Russia also constructed prototype reactor BN-600.

(2) Current status
(a) The government of Russia made a statement of four demonstration reactors BN-800. The permission of construction was given for two of them in Celyabinsk and Beloyarsk.
(b) For advancing FBR, conventional sodium coolant reactor and new lead coolant reactor (BREST-300) are being researched for optimizing economy and safety.

Quoted from JNC Home Page
*Special mention (1/2)

< BREST300 >

The former Soviet Union developed a lead-bismuth alloy coolant reactor for nuclear powered submarine more than forty years ago. This type of core is believed safe because it has minus reactivity coefficient. Making use of this technology, a pilot plant of lead coolant reactor BREST-300 of which the electric power is 300MWe was permitted to construct and 1200MWe power plant BREST-1200 was announced to be investigated. ‘BREST’ is the series name of lead coolant reactors in Russia.

< Manufacturing technology of MOX fuel >

Russia announced that new technology for diverting military plutonium to MOX fuel is being developed. It is already succeeded to divert metal plutonium to 50kg of PuO$_2$ at high temperature and this PuO$_2$ is supposed to be used in BOR-60 and BN-600.

Quoted from JNC Home Page
<Dry reprocessing technology>

Russia does not abandon long-term nuclear fuel cycle including FBR. In this statement it is reported that Russia researched new reprocessing technology by molten salt method and tested it for the spent fuel of BOR-60 and BN-800. According to this report the cost reduces to one third of that for conventional purex method. However France indicates that the content of cesium becomes very high by this method.
(1) BR-10

(a) Past circumstances
The experimental reactor BR-10 achieved criticality in March 1973. Carbide fuel and nitride fuel were researched. The accumulated operating time until 1997 is 4,674 hours of which the duration of nominal power is 2,830 hours.

(b) Current status
At present BR-10 is used for making isotopes, irradiating material and radiotherapy.
(2) BOR-60

(a) Past circumstances
The experimental reactor BOR-60 achieved criticality in 1969. Its thermal and electric power was 60 MWt and 12MWe respectively. The availability factor in 1997 was 77.4% along with the thermal power of 48〜55MWt.

(b) Current status
At present irradiation tests of fuel, control rods and structural material are being continued. In December 1998 the irradiation test of MOX fuel made from excess plutonium started.

Quoted from JNC Home Page
(3) BN-600

(a) Past circumstances
The prototype reactor BN-600 has been operated with the electric power of 600MWe since the first criticality in 1980. The availability factor in 1997 was 73.0 %. Although in 1998 the availability factor declined to 48% because of maintenance work for rotating plug, in 1999 37th, 38th and 39th operation went well and the availability factor of 1999 was 74%.

(b) Current status
The designed life of BN-600 is 30yeres (2010 A.D.). The life extension by 10 years for irradiating fuel made from nuclear weapons is being considered.

Quoted from JNC Home Page
(4) BN-800

(a) Beloyarsk
One of the four demonstration reactors BN-800 began to be constructed at Beloyarsk in 1985. However the construction work became suspended due to the accident of the Chernobyl nuclear power plant. The completion of BN-800 in Beloyarsk is set at 2009.

(b) Celyabinsk
The rest of the BN-800 i.e. three BN-800 are supposed to be constructed at Celyabinsk. The plutonium that had been for military use will be used as the fuel of BN-800 at Celyabinsk. It seems that the plan is delayed because of opposition of inhabitants and financial reasons.

Quoted from Bellona News 2002.1.30, ATOMICA
(5) BN-1600

BN-1600 is a demonstration reactor that is enlarged from demonstration reactor BN-800. Although a part of design has been started, Russia wishes that the future design will be done under international cooperation.

Quoted from JNC Home Page
<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>BR-10</td>
<td>Experimental reactor Re-constructed from BR-5</td>
<td>Carbide fuel, Nitride fuel</td>
<td>1973</td>
<td>10MWt</td>
</tr>
<tr>
<td>BOR-60</td>
<td>Experimental reactor Loop type</td>
<td>PuO₂ ·UO₂</td>
<td>1968</td>
<td>60MWt 12MWe</td>
</tr>
<tr>
<td>BN-600</td>
<td>Prototype reactor Tank type</td>
<td>PuO₂ ·UO₂</td>
<td>1980</td>
<td>1470MWt 600MWe</td>
</tr>
<tr>
<td>BN-800</td>
<td>Demonstration reactor Tank type</td>
<td>PuO₂ ·UO₂</td>
<td>-</td>
<td>2100MWt 800MWe</td>
</tr>
<tr>
<td>BN-1600</td>
<td>Demonstration reactor Tank type</td>
<td>PuO₂ ·UO₂</td>
<td>-</td>
<td>4200MWt 1600MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
6. Kazakhstan

BN-350 (Aktau)
The prototype reactor BN-350 was constructed by the former Soviet Union at Aktau (former Shevchenko) on the coast of Caspian Sea. Its thermal power is 1000MWt and electric power is 350MWe (general power supply: 150 MWe, desalinizing: 200MWe). BN-350 was the first prototype FBR in the world. It achieved criticality in November 1972. Electric generation started in November 1973 and the operation lasted until March 1998. BR-350 is able to desalinize 80,000 ton of sea water in a day and supply water to live on and hot water as well as electricity, which has much to do with regional affairs.

Since the collapse of the Soviet Union in 1991, Kazakhstan Republic has managed BN-350. In April 1999 the government of Kazakhstan determined to shut down BN-350 because IAEA-OSART made requirements on safety, operation and managing system, the end of life has come, liquid waste is almost full, and the budget for lengthening the life is tight.

Quoted from JNC Home Page
FBR Plant in Kazakhstan (2/2)

(2) Current status

The preparation for decommissioning of BN-350 is being progressed. The decommission is to be done under the international cooperation. The workshop for decommissioning BN-350 was held under the auspices of IAEA in the last May. The fundamental program for decommissioning is as follows.

(a) First phase
- Preparation period for safe storing of the facility (5 years)
- Removing the fuel and storing it outside the site
- Sodium drain and disposal
- Making detailed plan for decommissioning

(b) Second phase
- Safe storing (approximately 50 years)
- Surveillance and maintenance for the facility

(c) Third phase
- Safe scrapping

Quoted from JNC Home Page
### Table 6 Fast reactors in Russia and Kazakhstan

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>BN-350</td>
<td>Prototype reactor Loop type</td>
<td>PuO₂ +UO₂</td>
<td>1972</td>
<td>1000MWe 150MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
7. India

FBTR (Kalpakkam)

Quoted from MapQuest
(1) FBTR

(a) Past research and development
The experimental reactor FBTR developed under the cooperation with France achieved criticality in 1985 and succeeded in generating electricity in July 1997. FBTR is the first reactor that uses mixed carbide fuel (PuC70%, UC30%).

(b) Current status
The thermal power was changed and the operation is being continued. The largest burnup that has been achieved is 49,000MWD/t. FBTR is also used for irradiation test of Zr-Nb for light water reactors. At present there is a plan of modifying the reactor – the neutron channel and uninterruptible power supply will be replaced with new ones.

Quoted from JNC Home Page, ATOMICA
(2) PFBR

(a) Past research and development

For the prototype reactor PFBR of which the electric power is 600MWe, the design has been evaluated and optimized. MOX (mixed oxide) fuel is supposed to be used. It was announced that the start date, duration and cost of construction are 2001, 8 years and 29.6 billion rupees respectively. And four reactors of the same type will be constructed by 2020. The government of India approved 5.16 billion rupees for developing PFBR during five years from 1997 to 2002. The investment is broken down into 1.75 billion rupees for research and development, 600 million rupees for reprocessing, 1.95 billion rupees for concerned plan and 850 million rupees for affairs related to construction.

(b) Current status

The licensing is in progress. At present preliminary report for safety licensing examination is being made. The site is being evaluated by the ministry of the environment at the point of the effect on the surroundings. Besides the characteristics of the reactor are being researched – 3 dimensional analysis of burnup, fuel and material research, reprocessing technology development, and safety analysis are being performed.
Table 7 Fast reactors in India

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>FBTR</td>
<td>Experimental reactor</td>
<td>PuC-UC</td>
<td>1985</td>
<td>40MWt</td>
</tr>
<tr>
<td></td>
<td>Loop type</td>
<td></td>
<td></td>
<td>13.2MWe</td>
</tr>
<tr>
<td>PFBR</td>
<td>Prototype reactor</td>
<td>PuO₂ · UO₂</td>
<td>-</td>
<td>1250MWt</td>
</tr>
<tr>
<td></td>
<td>Tank type</td>
<td></td>
<td></td>
<td>500MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
8. China

CEFR (near Beijing)

Quoted from MapQuest
(1) Circumstances of nuclear energy development in China

China has much coal, petroleum and water power. The energy consumption of China is also large. However the energy consumption of one person is rather low – one fifth ~ one tenth of Japan and U.S. and half as much world average energy consumption of one person. China much relies on coal for energy, which causes the pollution of the environment and makes the continuous economic growth difficult. As for petroleum and natural gas, increase of domestic production cannot be expected any more. Thus at present import of petroleum from the Middle East and natural gas from Central Asia and Russia are considered. China began to generate nuclear power for commerce in 1994 and two nuclear power plants are in operation. The ratio of nuclear energy in China to other kinds of energy is 1.3%, which is much lower than the world average 17%. However eight commercial nuclear power plants are being constructed aiming at starting commercial operation from 2002 ~2005. Moreover construction of 16 nuclear power plants are suggested later than 2002. China intends to shift the energy policy from light water reactors to fast breeder reactors i.e. plutonium cycle and constructed experimental fast breeder reactor CEFR (China Experimental Fast Reactor).

Quoted from JNC Home Page
(2) Past research and development of FBR in China

CEFR is the first FBR in China of which the thermal power is 65 MWt and electric power is 20MWe. It was designed under the cooperation of France and Russia. The construction was accomplished in August 2002.

(3) The strategy of developing FBR in China

- The operation test of CEFR will be performed.
- The design of prototype reactor (300MWe) will be embarked from 2000 and accomplished around 2009 together with the safety licensing examination.
- The prototype reactor will be constructed during 2009～2015.
- The design of demonstration reactor (1GWe) will be started around 2010.
- The demonstration reactor will be constructed during 2019～2025.
- The design of the commercial reactor (4～6 × 300MWe) will be embarked around 2015.
- The commercial reactor will be constructed around 2023～2030.

Quoted from JNC Home Page
Table 8 Fast reactors in China

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>CEFR</td>
<td>Experimental reactor Tank type</td>
<td>PuO₂ •UO₂</td>
<td>-</td>
<td>65MWt 20MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
9. South Korea

Quoted from MapQuest
(1) Past circumstances
In 1992 South Korea accepted research and development of FBR as one of the national project following the American reactor PRISM that may enhance its intrinsic safety, transmute minor actinides and agree with nuclear non-proliferation principle. Based on this policy, construction of KALIMER (Korea Advanced Liquid Metal (coolant) Reactor – 150MWe) is being considered.

(2) Current status
The plan is being progressed expecting that fundamental design of KALIMER will be finished by 2006 and the construction will be begun in the middle of 2010s.

Quoted from JNC Home Page
### Table 9 Fast reactors in South Korea

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
</table>
| KALIMER             | Advanced liquid metal coolant reactor  
Tank type             | U-Zr | -                 | 392MWt  
150MWe           |

Quoted from JNC Home Page, ATOMICA
10. Japan

Joyo (Oarai-Town, Ibakaki-Pref.)

Monju (Tsuruga-City, Fukui-Pref.)

Quoted from MapQuest
(1) Political basis for FBR development in Japan

Japan had been researching on FBR at JAERI (Japan Atomic Energy Research Institute) since around 1960. AEC (Atomic Energy Commission) began to consider on constructing FBR in earnest and it was formally approved that experimental and prototype FBR with sodium coolant and mixed oxide fuel would be constructed by the domestic technology as the national project in May 1966. The fundamental policies are as follows.

• An experimental reactor with thermal power approximately 100MWt should be constructed in advance for getting technical information and experiences together with for irradiating fuel and structural material as there are many technical problems for constructing FBR.

• After getting sufficient information and experiences at the experimental reactor, a prototype reactor with the electric power approximately 300MWe should be constructed.

• For developing FBR fundamental technical information should be accumulated under the international cooperation as well as self management.

As the organization for performing this project PNC* (Power Reactor and Nuclear Fuel Development Corporation) was established in October 1967.

*PNC was reorganized as JNC (Japan Nuclear Cycle Development Institute) in October 1998. And JNC and JAERI are supposed to be unified next year.

Quoted from JNC Home Page, ATOMICA
(2) At the dawn of FBR development in Japan

Japan has made research and development of FBR based on the policies mentioned in May 1966. An FBR requires a lot of hot liquid sodium. However around the latter half of 1960s Japan had very few technology and experiences for handling sodium. Then sodium handling facilities were constructed at OEC (Oarai Engineering Center) PNC, where basic and applied research concerned with sodium was widely performed – for example, heat transport flow of liquid sodium and its coexistence with structural material (no reaction, no corrosion), sodium purity control, reaction of sodium with water and atmosphere, reactor vessel in the sodium, circulate pump, mockup tests of huge apparatuses (control rod drive mechanism and fuel handling machine). For mixed oxide fuel, fundamental test, fuel manufacturing technique and fuel design had been performed in former AFC (Atomic Fuel Corporation). In 1967 AFC was reorganized as PNC and the mixed oxide fuel for Joyo and Monju was decided to be manufactured at Tokai works of PNC based on the experiences in AFC.

Quoted from JNC Home Page, ATOMICA
(3) Second stage of research and development of FBR in Japan

A post-irradiation examination facility that were composed of irradiated fuel examination area, irradiated material examination area and irradiated fuel assembly examination area was constructed at OEC, where the fuel irradiated by the fast reactors in the United Kingdom, the United States of America, France and Joyo Japan was examined for improving its integrity and reliability. As for the steam generator – one of the most important apparatuses of FBR plant, 1MW steam generating test equipment was set up at OEC in 1971 and 50MW one was set up in 1974 by which thermohydraulic characteristics of sodium, sodium - water reaction, reliability and ability of steam generator were ascertained. Besides reactor physics of FBR, core design, structural material safety assessment were studied in earnest. The research on reactor physics was performed with FCA (Fast Critical Assembly) that had been already constructed at JAERI and establishment of core design method was aimed through collecting basic data for designing and ascertaining the reliability of the method for nuclear calculation.

Quoted from JNC Home Page, ATOMICA
FBR Plant in Japan (1/9)

(1) Experimental reactor Joyo

Quoted from JNC Home Page
(a) The experimental reactor Joyo at OEC is the first FBR plant in Japan that is the compile of the techniques of any field of FBR and composed of the domestic technology from design to construction. The conceptual design of Joyo had been performed at JAERI since 1963 and PNC succeeded it and accomplished the detailed design.

(b) The construction of Joyo started in 1970 and it achieved criticality in April 1977. The thermal power of Joyo achieved 75MWt in 1979, which was called Mark-I core. With Mark-I core the reliability of the design and the characteristics of the plant were ascertained. In 1982 Mark-I was converted into Mark-II core of which the thermal power attained to 100MWt in March 1983.
FBR Plant in Japan (3/9)

(c) Joyo had been operated without serious problem and accumulated technical experiments about design, construction, and operation of FBR plant. And also it obtained much information on verifying the reliability of the design, establishing the operation and maintenance technique, grasping the behavior of the plant and ascertaining the irradiation behavior of fuel and structural material, which played a very important role of developing FBR. The cumulative operating duration until June 1997 is 52,802 hours and cumulative thermal power is 4,306,964 MWh.

(d) In September 1995 Mark-II was permitted to be converted into Mark-III core for improving irradiation ability, which achieved 140 MWt in October 2003.

Quoted from JNC Home Page, ATOMICA
FBR Plant in Japan (4/9)

(2) Prototype reactor Monju

Quoted from JNC Home Page
(a) The conceptual design and main items for constructing the prototype reactor Monju were begun to be investigated in 1968. The design, construction and operation experienced at Joyo, accumulated technical data, evolitional method of design and safety assessment, evaluation of the past research and the status of developing FBR in the world were reflected in the detailed design of Monju. The safety assessment, structural design under high temperature and earthquake-proof became the significant basis for developing FBR.

(b) The nominal electric power of Monju is 280MWe. The construction began in October 1985 at Shiraki Tsuruga-city Fukui prefecture. The installation of apparatuses was completed in April 1991. The function tests were performed until December 1992 and the criticality was achieved in April 1994. The first generating electricity was on August 29th 1995, where the electric power and thermal power were 5% and 40% of the nominal value respectively.

Quoted from JNC Home Page, ATOMICA
(c) On December 8th 1995 during the startup test, sodium leakage occurred at one thermometer on the hot leg of the secondary heat transfer system and the reactor was shutdown. The cause was researched and it turned out that the leakage occurred due to the break of the sheath of the thermocouple caused by the vibration of the sensor in the flowing sodium. In succession for the closer research, safety assessment started in December 1996. The results were announced in March 1998. The nuclear energy committee set up FBR round table conference for discussing what FBR should be in the future and for hearing the public opinion.

(d) Japanese government approved the change of installation in December 2002. However the Nagoya high court Kanazawa branch sentenced that the approval of constructing Monju itself was invalid. At present this problem is pending at the Supreme Court.

Quoted from JNC Home Page, ATOMICA
(3) Demonstration reactor

The aim of constructing the demonstration reactor is to secure the safety, reliability, and operation - maintenance technique same as light water reactors and to put it into practical use with comparable low cost.

In 1981 the former STA (Science and Technology Agency) set up a committee for putting FBR into practical use. The Ministry of International Trade and Industry set up a similar committee. In these committees the policy of developing the demonstration reactor reflecting the results experienced at Monju was discussed. In May 1983 a round table conference for research and development of FBR was set up under AEC and the policy of research and development, assignment of the role among office, university and industry, international cooperation were discussed.

In June 1987 AEC made ‘Long term plan for developing and utilizing nuclear energy’ which required national and private organizations to promote research and development of studying the demonstration reactor according to their assigned role.
The demonstration reactor should be developed based on the experiences at Joyo, Monju and cooperation with foreign countries staring fixedly at the current technical status of FBR development in the world. In 1986 a steering committee for research and development of FBR was organized among Japan Atomic Power Company, PNC, CRIEPI (Central Research Institute of Electric Power Industry), and JAERI all of which have much to do with FBR development and role assignment, international cooperation, items of research and development and the plan for executing with respect to putting FBR into practical use including developing demonstration reactor have been discussed. According to the long term plan made by AEC, a sectional committee for FBR development was set up in May 1987, which issued ‘The way of research and development for FBR’ and made clear the main theme and fundamental problem for putting FBR into practical use. The four organizations mentioned before are cooperating on this project.

Quoted from JNC Home Page, ATOMICA
JAPC (Japan Atomic Power Company) are in charge of constructing the demonstration reactor under the cooperation with PNC, CRIEPI and JAERI aiming at completing it early 2000s.

Taking the change of circumstances on FBR i.e. current status of FBR in the world, uranium market and surplus plutonium into consideration, AEC issued ‘Long term plan for developing and utilizing nuclear energy’ afresh in June 1994, where it is declared that the demonstration reactor will be top entry loop type and generate approximately 660MWe electricity and that safety and efficiency will be improved adopting evolutional techniques that may put the demonstration reactor into practice. The construction of demonstration reactor is supposed to start in early 2000s regarding the progress of developing and operation experiences at Monju. The concerned organizations evolve the required technology and prepare for the construction.

Besides In Japan mainly plutonium and uranium mixed oxide fuel has been researched. However new type of fuel – metal or nitride fuel which may lower the cost and intrinsic safety has also come to be researched in CRIEPI, JNC and JAERI.
### Table 10 Fast reactors in Japan

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>Joyo</td>
<td>Experimental reactor, Loop type</td>
<td>PuO₂ · UO₂</td>
<td>1977</td>
<td>100MWt (140MWt in 2003)</td>
</tr>
<tr>
<td>Monju</td>
<td>Prototype reactor, Loop type</td>
<td>PuO₂ · UO₂</td>
<td>1994</td>
<td>714MWt 280MWe</td>
</tr>
<tr>
<td>DFBR</td>
<td>Demonstration reactor, Top entry loop type</td>
<td>PuO₂ · UO₂</td>
<td>-</td>
<td>1600MWt 660MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMIC A
11. Italy

PEC (Brasimone, near Bologna)

TAIRO (Casaccia, near Rome)

Quoted from MapQuest
Circumstances of FBR development in Italy

(1) TAPIRO
The research and development of FBR in Italy started at TAPIRO a fast research reactor. Its fuel is highly enriched uranium (93.5%). Although its thermal power is low (5kWt), it is able to generate high neutron flux \(4.23 \times 10^{14} \text{n/sec}\). Mainly shielding technique was researched at TAPIRO.

(2) PEC
The construction of an experimental reactor PEC was started in 1972 aiming at starting operation in 1990. However its progress was delayed due to the licensing problem. The Italian parliament excluded the budget for PEC and the it was determined that FBR project would be reduced. Then in August 1986 the construction of PEC was called off.

Quoted from 'Monte Carlo Optimization of a BNCT Facility for treating Brain Gliomas at the TAPIRO Reactor', ATOMICA
### Table 11: Fast reactors in Italy

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>TAPIRO</td>
<td>Research reactor</td>
<td>Highly enriched Uranium</td>
<td>1969</td>
<td>5kWt</td>
</tr>
<tr>
<td>PEC</td>
<td>Experimental reactor Loop type</td>
<td>PuO₂ ·UO₂</td>
<td>-</td>
<td>120MWt</td>
</tr>
</tbody>
</table>

Quoted from ‘Monte Carlo Optimization of a BNCT Facility for treating Brain Gliomas at the TAPIRO Reactor’, ATOMICA
12. Brazil

Quoted from MapQuest
Brazil is promoting FBR development aiming at starting operation approximately in 25 ~ 30 years. Since 1992 the design of an experimental reactor REARA-60 of which the fuel is U-10%Zr has been performed referencing to the American fast reactor PRISM and EBR-II.

Quoted from ATOMICA
Table 12 Fast reactors in Brazil

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>REARA-60</td>
<td>Experimental reactor Tank type</td>
<td>U-10%Zr</td>
<td>-</td>
<td>60MWt 20MWe</td>
</tr>
</tbody>
</table>

Quoted from ATOMICA
13. Europe - EFR

(1) Past research and development of EFR

The conceptual design for EFR (European Fast Reactor) started in May 1988 under the cooperation of European countries. The assessment of the reliability of the conceptual design started in April 1990 and ended in March 1993. Although the preparation program was to proceed afterwards, it was postponed for the reasons that Super Phenix that was to precede EFR became to be decommissioned and that there is no lack of uranium for the time being. Taking these circumstances into consideration the strategy for developing EFR was reconsidered and in 1995 the new strategy was made. The new strategy consists of the affairs below.

- No new FBR will be constructed before 2005 ~ 2010.
- Plutonium and minor actinides burning should be researched for the time being.
- There are many items to be ascertained together with the reference design.
- Innovative technology conception should be considered regardless of the time when the design of a reference core is introduced.
- The development mentioned above should be performed until the construction of EFR is determined.

Quoted from JNC Home Page, ATOMICA
(2) Current status of EFR

The required operating experience and technical information for EFR development cannot be gotten due to the decommission of Super Phenix. According to this situation EFRUG (European Fast Reactor Utility Group) called off the EFR project by December 1998. EFR Associates make the final report of the past technical achievements.

Quoted from JNC Home Page, ATOMICA
### Table 13 Fast reactors in Europe

<table>
<thead>
<tr>
<th>Name of the reactor</th>
<th>Type and features of the reactor</th>
<th>Fuel</th>
<th>First criticality</th>
<th>Power</th>
</tr>
</thead>
<tbody>
<tr>
<td>EFR</td>
<td>Demonstration reactor, Tank type</td>
<td>PuO₂ + UO₂</td>
<td>-</td>
<td>3600MWt 1520MWe</td>
</tr>
</tbody>
</table>

Quoted from JNC Home Page, ATOMICA
Conclusion

As already mentioned, surplus plutonium has come to be the pending problem with respect to the nuclear non-proliferation principle. Moreover the high cost of constructing FBR cannot be bypassed. For these reasons some countries not to say withdraw but have come to reduce research and development of FBR. Therefore recent policy on developing FBR is being converted – not breeding plutonium but burning plutonium produced in LWR together with minor actinides in small scaled FBR, which may lead to the self-consistent nuclear fuel cycle. This concept has come to be considered as the extreme value of existence of FBR. And also new type of fuel, coolant and so on are being researched for enhancing safety and matching the non-nuclear proliferation principle.

Thanks for your attention
Basic Knowledge on Sodium Heat Transfer & Flow

Makoto Sawada

September, 2004

International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
Presentation Contents

§ 1: Basic Principle of Heat Transfer
§ 2: Basic Law of Heat Transfer
§ 3: Heat Transfer in Heat Exchanger
§ 4: Fluid Flow (Calculation of Pressure Loss)
§ 1: Basic Principle of Heat Transfer

Type of Heat Transfer

- **Thermal Conduction**
  Type of which heat moves inside of object

- **Heat Convection**
  Type of which heat moves along with particles

- **Heat Radiation**
  Type of which heat moves without via a vehicle
Heat moves in company with particles of water

**Type of Heat Convection**

**Natural Convection**
Particle’s flow caused naturally in fluid

**Forced Convection**
Particle’s flow forced by such as pump

Natural convection is inefficient because particle’s movement amounts is small.

Forced convection is efficient because particle’s movement amounts is large.
Type of Heat Exchanger’s Heat Transfer

Natural Convection Heat Transfer
Heat movement between fluid and object by natural convection

Forced Convection Heat Transfer
Heat movement between fluid and object by forced convection such as at heat exchanger
Heat Movement in Heat Exchanger

(Type)
- Tube Side: High Temp.
- Shell Side: Low Temp.

Heat flows through forced convection, thermal conduction and forced convection in turn from high temp. side.
Difference of Heat Transfer by Flow’s Direction

**Parallel Flow Type**
- High Temp. Fluid
- Low Temp. Fluid

High Temp. Side Fluid
Low Temp. Side Fluid

Temp. Distribution Parallel Flow

Low side’s fluid temp. never exceed high side.

**Opposite Flow Type**
- High Temp. Fluid
- Low Temp. Fluid

High Temp. Side Fluid
Low Temp. Side Fluid

Temp. Distribution Opposite Flow

It’s possible that low side’s fluid temp. exceeds high side.
Heat Balance in Heat Exchanger

Heat Transfer Quantity $Q = Q_H = Q_C$

$Q_H = W_1 \times C_p_1 \times (T_1 - T_2)$ \text{(kcal/h)} \quad \text{SI Dimension(kJ/s)}$


$Q_C = W_2 \times C_p_2 \times (t_1 - t_2)$


(Remark) Guideline of Tolerance = 10% difference between $Q_H$ and $Q_C$

C) Don’t use volume Velocity

Mass Velocity $=$ Volume Velocity $\times$ Density

※Density $(kg/m^3)$: Use mean temp. between outlet and Inlet temp..
§ 2: Basic Law of Heat Transfer

Heat Transfer Quantity

Q is in proportion to heat transfer area.

Heat Transfer Area

Hot

Cold

Heat Flow Flux

Q = Heat Flow Flux (q) x Heat Transfer Area (A)

Heat Flow Flux (q)

q is heat transfer quantity of which flows to a certain area squarely with per area and per time.

Temp. Gradient

Temp. Difference (ΔT) = Wall Thickness (ℓ)

Heat Transfer Quantity

- Temp. Gradient Small ⇒ Heat Transfer Decrease
- Temp. Gradient large ⇒ Heat Transfer Increase
Flow Condition of Fluid

**Laminar Flow**
- Heat slowly moves vertically by Thermal Conduction.

**Turbulent Flow**
- Heat quickly moves vertically by heat convection.

**Experiment of Reynolds**
Reynolds proved experimentally that flow condition is decided depending on fluid's flow rate.

**Laminar Flow**
Heat slowly moves toward vertically by Thermal Conduction.

**Turbulent Flow**
Heat quickly moves toward vertically by heat convection.
Boundary Layer of Velocity

- Velocity at near wall decreases remarkably by causing friction resistance. (Forming of boundary layer of velocity)
- Velocity at around central in flow pass becomes almost uniform flow (turbulent flow core) due to mixing action.

Flow State inside Heat Pipe

Boundary Film

- Boundary film exists due to since fluid's viscosity.

Velocity Distribution

Turbulent Flow Core

Friction

Wall

Area where flow changes rapidly

Boundary Layer of Velocity

Velocity Distribution

Wall

Boundary Film

Boundary Layer of Velocity
Forced Convection

Boundary Layer of Temp.

Temperature Distribution
Velocity Distribution

Boundary Film
Area of Velocity Changes Suddenly
Area of Temp. Changes Suddenly

Low Velocity
⇒ Thick Boundary Film

High Velocity
⇒ Thin Boundary Film

Hot Water
Wall
Cold Water

Standard Temp.
Rapidly Changing

Temp. Boundary Layer
Velocity Boundary Layer

Boundary Film
Boundary Film
Boundary Film
Boundary Film

Forced Convection

出展・著作者：技術教育ソフトウェア、伝熱の基礎コース。© 日本能率協会マネジメントセンター
Heat Transfer Quantity (HTQ) By Forced Convection

**Newton Cooling Law**

H.T.Q between Fluid and Solid is proportional to both temp. difference.

**Heat Transfer Quantity (H.T.Q)**

\[ H.T.Q = h \times H.T.A \times \Delta T \]

- **Heat Transfer Coe. (Kcal/m²h°C)**
  - An index of heat transfer at boundary film
  - Its value depends on kind and property of fluid and operation condition, etc.

**SI Dimension System**

Heat Transfer Coe. \( h \) (W/m²·K)

K=Kelvin (0°C=273.15K)
Heat Transfer Quantity (HTQ) By Thermal Conduction

Fourier Cooling Law

H.T.Q in Solid Wall is proportional to H.T.A and temp. gradient.

Fourier Law

In case of using proportionality constant $k$

$$Q = k \cdot A \cdot \frac{\Delta T}{\ell}$$

Thermal Conductivity (Kcal/mh°C)

- An index of heat transfer at in object
- Its value depends on kind and property of object

SI Dimension System

Thermal Conductivity: $\kappa$ (W/m·K)
Heat Transmission

Heat Transfer Quantity (H.T.Q)

\[ Q = Q_1 = Q_2 = Q_3 \]

**Newton Cooling Law**

\[ Q_1 = h_1 \cdot A \cdot (T_1 - T_2) \]

**Fourier Cooling Law**

\[ Q_2 = k \cdot A \cdot \left( \frac{T_s - t_a}{\ell} \right) \]

**Newton Cooling Law**

\[ Q_3 = h_2 \cdot A \cdot (t_2 - t_1) \]

Impossible to measure both side temp. of Wall, T2, t2.

It can be obtained H.T.Q by measuring standard temp. of both fluid (T1,t1) even both temp. of both surfaces of wall (T2, t2).

Thus, it can be eliminated T2, t2 as the following.

\[ T_1 - T_2 = \frac{Q}{h_1 \cdot A} \quad \cdots (4) \]

\[ T_2 - t_2 = \frac{Q}{k \cdot A} \quad \cdots (5) \]

\[ t_2 - t_1 = \frac{Q}{h_2 \cdot A} \quad \cdots (6) \]

\[ (T_1 - T_2) + (T_2 - t_2) + (t_2 - t_1) = \frac{Q}{h_1 \cdot A} + \frac{Q}{k \cdot A} + \frac{Q}{h_2 \cdot A} \]

\[ T_1 - t_1 = \frac{Q}{h_1 \cdot A} + \frac{Q}{k \cdot A} + \frac{Q}{h_2 \cdot A} \]
Assuming that this is Proportional Coe. $U$.

Formula of Fourier:

$$Q = U \cdot A \cdot (T_1 - t_1)$$

Heat Transmission Coe. (kcal/m$^2$·h·°C)

$$u = \frac{1}{h_1} + \frac{\partial}{k} + \frac{1}{h_2}$$

$U$ (W/m$^2$·K) (SI Dimension System)
§ 3: Heat Transfer in Heat Exchanger

(1) Calculation of Heat Transfer Area (H.T.A)

Formula of Fourier:
\[ Q = U \cdot A \cdot (T_1 - t_1) \]

A certain portion

This equation is not adequate for calculating whole H.T.Q of the HX because it's only calculates it at a certain portion in HX.

Need optimization of Fourier Formula

H.T.A is obtained by means of outer surface of heat pipe.

Heat Transfer Area (HTA)

\[ Ao = \text{Circumference of Pipe (m)} \times \text{Length of Pipe} \times \text{Numbers of Pipe} \]

Actual H.T.A

Optimized HTA (A)

Experimental Device

Heat Exchanger (HX)

Outlier: 製作者：技術教育ウェア，伝熱の基礎コース，© 日本能率協会マネジメントセンター
The logarithm mean for temp. difference between inlet and outlet of both fluids is supposed to temp. difference of both fluids.

Logarithm Mean temp. Difference (LMTD)

- Hot Side $\Delta T_1 = T_1 - t_2$
- Cold Side $\Delta T_2 = T_2 - t_1$

LMTD = $\frac{\Delta T_1 - \Delta T_2}{\ln \left( \frac{\Delta T_1}{\Delta T_2} \right)} = \frac{\Delta T_1 - \Delta T_2}{2.3 \log \left( \frac{\Delta T_1}{\Delta T_2} \right)}$

Q = $U \cdot A \cdot (T_1 - t_1)$ (Fourier Formula)

The standard temp. of high and low temp. of fluids, T1 and t1, are not able to be measured actually. So, how to get temp. difference between both fluids.
Compensation of LMTD (1/3)

Many Tube Type HX

T1: High Side Inlet Temp.

T2: High Side Outlet Temp.

t1: Low Side Inlet Temp.

Complicated Temp. Distribution

Double Tube Type HX

Hot Fluid

Cold Fluid

Compensation of LMTD

Many Tube Type HX

Compensation Factor (Ft)
Temp. Drop Ratio: \( R = \frac{T_1 - T_2}{t_2 - t_1} \)

Temp. Efficiency: \( P = \frac{t_2 - t_1}{T_1 - t_1} \)

Consider temp. drop at hot fluid

\[ R = \frac{86 - 48}{44 - 30} = 2.71 \]

Consider temp. rise at cold fluid

\[ P = \frac{44 - 30}{86 - 30} = 0.25 \]

0.87 is obtained as Ft from the figure.
Compensation of LMTD (3/3)

Generally, temp. compensation factor $F_t$ is:

$$0.8 < F_t < 1.0$$

Real Temp. Difference $< \text{LMTD}$

$$\Delta T = \text{LMTD} \cdot F_t$$

$$= 28.4 \times 0.87$$

$$= 24.7 \, ^\circ\text{C}$$

Real Temp. Difference

Fundamental Fourier Formula

$$Q = U \cdot A \cdot (T_1 - t_1)$$

Apply Temp. Compensation Factor ($F_t$)

$$Q = U \cdot A_0 \cdot \text{LMTD} \cdot F_t$$
Factor 2: Influence of Density

Density

\[ \rho = \frac{\text{Mass}}{\text{Volume}} \]

Since sodium’s density changes straightly along with its temp., it cannot be regarded as settled value like a water.

<table>
<thead>
<tr>
<th>Temp. (°C)</th>
<th>Density (kg/m³)</th>
</tr>
</thead>
<tbody>
<tr>
<td>200</td>
<td>904</td>
</tr>
<tr>
<td>300</td>
<td>880</td>
</tr>
<tr>
<td>400</td>
<td>856</td>
</tr>
<tr>
<td>500</td>
<td>832</td>
</tr>
<tr>
<td>550</td>
<td>820</td>
</tr>
</tbody>
</table>

出典：PNC PN9520 91-006 高速増殖炉技術読本、（第3編ナトリウムの諸性質）P3-70 1991年7月
Factor 3: Influence of Viscosity

In the actual calculation, Kinetic Viscosity is used as a Value of Fluid's viscosity.

Kinetic Viscosity = \frac{Viscosity}{Density}

<table>
<thead>
<tr>
<th>Temp.(°C)</th>
<th>Kinetic Viscosity Coe. (kg/m\cdot h)</th>
</tr>
</thead>
<tbody>
<tr>
<td>200</td>
<td>162.68</td>
</tr>
<tr>
<td>300</td>
<td>124.20</td>
</tr>
<tr>
<td>400</td>
<td>102.42</td>
</tr>
<tr>
<td>500</td>
<td>88.56</td>
</tr>
<tr>
<td>550</td>
<td>80.82</td>
</tr>
</tbody>
</table>

Viscosity coe. of sodium cannot be ignored because its value changes exponentially like oil.

※1kg/m\cdot h=2.778 \times 10^{-4} Pa\cdot sec
Viscosity $\eta$ : kg·s/m²

出展：PNC N 941 75-19、ナトリウム物理值の実用計算式（1972年までの公表文献に基づく液体と蒸気の物理値）、P164、1975年3月
Factor 1: Influence of Velocity

Velocity is the most effective factor among them.

Volume Velocity is used on calculating Velocity.
Factor 4: Influence of Thermal Conductivity

<table>
<thead>
<tr>
<th>Temp. (°C)</th>
<th>Thermal Conductivity (kcal/m·h·°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>200</td>
<td>70.5</td>
</tr>
<tr>
<td>300</td>
<td>66.3</td>
</tr>
<tr>
<td>400</td>
<td>62.1</td>
</tr>
<tr>
<td>500</td>
<td>57.9</td>
</tr>
<tr>
<td>550</td>
<td>55.8</td>
</tr>
</tbody>
</table>

Thermal Conductivity of Sodium

Effect of sodium’s thermal conductivity is not disregarded due to its value changes depending on its temp..

※1 kcal/m·h·°C = 1.163 W/m·K

出典: 伝熱工学資料改定第3版、液体金属及び溶融塩の物性値、P321、日本機械学会、1975

出展: 製作者: 技術教育ソフトウェア、伝熱の基礎コース、© 日本能率協会マネジメントセンター
Thermal Conductivity $\kappa$: kcal/m\(\cdot\)h\(^{\circ}\)C

Kinetic Viscosity $\nu$: m\(^2\)/sec

出展：PNC N 941 75-19、ナトリウム物性値の実用計算式（1972年までの公表文献に基づく液体と蒸気の物性値）、P165、1975年3月
Factor 5: Influence of Specific Heat

**Specific Heat**

1 kcal/kg°C: Necessary heat to rise 1°C for 1kg object.

<table>
<thead>
<tr>
<th>Temp. (°C)</th>
<th>Specific Heat (kcal/kg·°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>200</td>
<td>0.326</td>
</tr>
<tr>
<td>300</td>
<td>0.321</td>
</tr>
<tr>
<td>400</td>
<td>0.316</td>
</tr>
<tr>
<td>500</td>
<td>0.312</td>
</tr>
<tr>
<td>550</td>
<td>0.311</td>
</tr>
</tbody>
</table>

As changing of sodium's specific heat corresponding its temp. is very small, it is no problem to consider that its value is stable.

※1 kcal/kg°C = 4.187 kJ/kg·K

出典：伝熱工学資料改定第3版、液体金属及び溶解塩の物性値、P321、日本機械学会、1975
## Factor 6: Influence of Flow Pass Diameter

<table>
<thead>
<tr>
<th>V</th>
<th>ρ</th>
<th>μ</th>
<th>κ</th>
<th>Cp</th>
<th>D</th>
</tr>
</thead>
</table>

### Flow Pass Diameter

Diameter of Fluid's Pass Flow

Flow pass diameter is a constant decided by design.

**Tube Side**

- **Inner Diameter of a Tube**

**Shell Side**

- **Corresponding Shell Diameter**

### Arrangements

- **Square Arrangement**

  - De: Corresponding Diameter (De)

- **Triangle Arrangement**

  - De: Corresponding Diameter (De)
(4) Calculating Coe. (U) based on Non-Dimensional Numbers

\[ Q = U \cdot A_o \cdot LMTD \cdot F_t \]


\[ U = \frac{1}{\frac{1}{h_i} \frac{1}{k} + \frac{1}{h_o}} \]

hi: Heat Transfer Coe. for internal surface side
ho: Heat Transfer Coe. for outer surface side

To analyze the existing relationship between physical quantities of each influence factors such as flow, temp., density, etc. is called “Non-Dimensional Analysis”. And, the following relationship is revealed:

Nusselt Number = Reynolds Number \times Prandtl Number

\[ \left( \frac{hD}{k} \right) = \alpha \left( \frac{DV\rho}{\mu} \right)^a \left( \frac{Cp\mu}{k} \right)^b \]

Nusselt Number
Reynolds Number
Plantole Number

Constant Factor \( \alpha \), Exponent \( a, b \) are obtained from experiment.
Nusselt Number (Nu) = Reynolds Number (Re) × Prandtl Number (Pr)

Nusselt Number: Nu = h · D / \kappa

“Nu” shows a ratio of H.T.Q transferred by heat transfer during fluid are flowing to H.T.Q transferred by thermal conductivity when fluid’s flowing stopped. Plainly speaking, “Nu” expresses a index of which whether heat transfer is good or not.

Reynolds Number: Re = D · V · \rho / \mu

“Re” is a ratio of inertia force by fluid’s flowing to viscosity force of fluid. That to say is that “Re” shows flowing state of fluid, and it’s able to judge its state whether laminar flow or turbulent flow.

Laminar Flow = Re < 2,100  Turbulent Flow = Re > 4,000

Transfer Region: 2100 < Re < 4000

Prandtl Number: Pr = Cp · \mu / \kappa

“Pr” is a ratio of boundary’s thickness of velocity and temp. and is an index expressed to what extent growing up of the both layers. (Discuss in detail in the next page)
What mean does Prandtl Number (Pr) have?

Although we learned about the boundary layers of velocity and temp. in page 10 and 11, both layers exist together in one boundary layer as shown in the below figure.

Relationship between Prandtl number and boundary layer’s thickness is expressed as shown in the following relations.

- Pr < 1: Temp. Boundary’s Thickness > Velocity Boundary’s Thickness
- Pr = 1: Temp. Boundary’s Thickness = Velocity Boundary’s Thickness
- Pr > 1: Temp. Boundary’s Thickness < Velocity Boundary’s Thickness

出典: http://www.che.kyutec.ac.jp/chem22/、山村、九州工業大学工学部、応用化学教室、移動理論Ⅲ
In the case of heat transfer by using a low Pr number’s fluid, a region of the temp. boundary layer which is dominant thermal conductivity will increase, i.e. its layer’s thickness will become to be thick. While, inversely, it will become to be thin in high Pr number’s fluid because heat transfer by heat convection is dominant in there. Consequently, as indicated in above figure, heat transfer by the low Pr number’s fluid is low heat flux and inversely high Pr number’s fluid is equal to high heat flux, which since Pr number is proportion in Nusselt number (Nu) and heat transfer coe. (h).
To get High Heat Transfer Coe.

◆ Increasing of Reynolds Number “Re” (Increasing Velocity, etc.)

◆ Employing of High Prandtl Number’s “Pr” Fluid
  - Air, He, H₂, O₂, N₂ : Pr ~ 0.7
  - Steam : Pr ~ 1.0
  - Water : Pr = 13 ~ 5
  - Oil : Pr = 47,100 ~ 276

◆ Increasing of Heat Transfer Area (Employing of Fin Structure)
Heat Transfer Coe. of Sodium

Sodium = Low Pr number’s Fluid ⇒ (Pr ≈ 0.01 ~ 0.004)

(See next page’s figure)

Why Heat Transfer Coe. of Sodium is High even low Pr Number?

(See figure in page 38)

On calculation of heat transfer coe. for HX learned so far, effect of fluid’s thermal conductivity during turbulent flow was ignored because its value is not so high like water. However, it is not able to disregard in the case of using liquid metal which its thermal conductivity is high such as a sodium. Finally, above mentioned effect is expressed to the constant factor (α) especially in the following equation.

\[ \text{Nu} = \alpha \cdot \text{Re}^a \cdot \text{Pr}^b \]

\[ \text{Nu} = 7 + 0.025 \text{Pe}^{0.8} \]  

(Sample: Equation of MARTINELLI-LYON)

Pe: Peclet Number = Re · Pe
Prandtl Number: $Pr = \frac{\eta \cdot Cp}{\kappa} \times g$ (gravity)

Temp. Thermal Conductivity: $a = \frac{\kappa}{\rho \cdot Cp}$ (m²/h)
## Physical Properties of Sodium as a Coolant

<table>
<thead>
<tr>
<th>Items</th>
<th>Sodium</th>
<th>Helium</th>
<th>Steam</th>
<th>Water</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temp. (K)</td>
<td>600</td>
<td>600</td>
<td>600</td>
<td>600</td>
</tr>
<tr>
<td>Pressure (Mpa)</td>
<td>0.1</td>
<td>10</td>
<td>7.5</td>
<td>15</td>
</tr>
<tr>
<td>Density (kg/m³)</td>
<td>874</td>
<td>7.86</td>
<td>33.3</td>
<td>659</td>
</tr>
<tr>
<td>Specific Heat (kJ/kg · K)</td>
<td>1.34</td>
<td>5.19</td>
<td>3.69</td>
<td>6.61</td>
</tr>
<tr>
<td>Viscosity (µPa · s)</td>
<td>327</td>
<td>32.2</td>
<td>21.04</td>
<td>77.2</td>
</tr>
<tr>
<td>Thermal Conductivity (W/m · K)</td>
<td>75.7</td>
<td>0.254</td>
<td>0.0609</td>
<td>0.501</td>
</tr>
<tr>
<td>Flow Velocity (m/s)</td>
<td>4.52</td>
<td>47.4</td>
<td>27.1</td>
<td>6.26</td>
</tr>
<tr>
<td>Heat Transfer Coe.* (W/ m² · K)</td>
<td>84,700</td>
<td>47,800</td>
<td>5,030</td>
<td>42,800</td>
</tr>
<tr>
<td>Required Pump Power (W/m)</td>
<td>4.54</td>
<td>47.7</td>
<td>27.2</td>
<td>6.29</td>
</tr>
</tbody>
</table>

* The case in which it flows of the circular pipe of the 8mm diameter at pressure drop 20kPa per m.

(Calculation Equation For Sodium: \( Nu = 7 + 0.025Pe^{0.8} \), For Water: \( Nu = 0.023Re^{0.8}Pr^{0.4} \))
### Calculation sample of Heat Transfer Coe.

#### Heat Transfer Coe.: \( h = \frac{D}{\kappa} \frac{1}{\text{Nu}} \)

<table>
<thead>
<tr>
<th></th>
<th>Tube Side</th>
<th>Shell Side</th>
</tr>
</thead>
</table>
| Water             | \[
\alpha \left( \frac{DV\rho}{\mu} \right)^{0.8} \left( \frac{3600}{(100D)^{0.2}} \right)
\]
|                   | \( \text{Nu} = 0.36(\text{Re})^{0.55}(\text{Pr})^{0.33} \) | \( \text{Nu} = 4.673 \times 10^{-3}\text{Pe}^{1.661} \) |
| Sodium            | \[\text{Nu} = 0.625\text{Pe}^{0.4} \quad (\text{Pe} \geq 45)\] | \[\text{Nu} = 1.2 \times 10^{-3}\text{Pe}^{1.7} \]
|                   | \[\text{Nu} = 1.5 \sim 2.8 \quad (45 > \text{Pe} \geq 35)\] | \[\text{Nu} = 5 + 0.025\text{Pe}^{0.8}\] |
|                   | \[\text{Nu} = 1.5 \quad (35 > \text{Pe})\] | \[\text{Nu} = 5 + 0.025\text{Pe}^{0.8}\] |

#### Exponent a, b are obtained from experiment.

\( \text{Pe} \): Peclet Number
\[
\text{Pe} = \frac{\rho \cdot C_p \cdot V \cdot D}{\kappa} = \text{Pr} \cdot \text{Re}
\]

**What these equations shown in here are just samples. The best fitted equation will be chosen by parameter analysis.**

出典: 水の式「MAM−CAI、伝熱の基礎コースVer2.22、出光興産㈱、日本能率協会」
ナトリウムの式「PNC SN 941088-049、「常陽」自然循環100MW過渡試験、P99、1988」
As shown in here, thermal resistance is negligible small. 

Influence of Thermal Conductivity at Pipe Wall

Sample Calculation of thermal resistance

Disregarded thermal resistance

{sampling calculation}

Considered thermal resistance

\[ U = \frac{1}{\frac{1}{h_i} + \frac{1}{\frac{k}{\ell/k} + h_o}} \]

Actual Calculating Equation of Heat Transmission Coe. (U)

\[ Q = \frac{U \cdot A \cdot \Delta T \cdot F_t}{U} \]

It's able to ignored thermal conductivity at pipe wall (thermal resistance of tube)

A: Heat Transfer Area
ΔT: Temperature Difference
F_t: Correction Factor

Compensation of Heat Transfer Coe. (hi)

H.T.A (A) = Circumference × Length

Outer Surface (Ao)

Compensation of hi

hio = hi × Ai / Ao
= hi × di / do

(Internal Surface Area)
(outer Surface Area)
(Inner Dia.)
(Outer Dia.)

It’s necessary to compensate hi in that it is based on outer surface area.

Fourier Formula for Whole H.X

Q = U · Ao · LMTD · Ft

U = \frac{1}{\frac{1}{hi} + \frac{1}{k} + \frac{1}{hi o}}


Compensating hi based on outer surface

Q = U · Ao · LMTD · Ft

Out of the authors, technology education software, foundation courses, © IEJ Japan Energy Management Center
Compensation of Contamination Factor

Contamination of inside HX

Heat Transfer at Scale (Thermal Conductivity)

Fourier Formula for whole HX

\[ Q = U \cdot Ao \cdot LMTD \cdot Ft \]

\[ U = \frac{1}{\frac{1}{h_{io}} + r + \frac{1}{h_o}} \]

\[ \frac{1}{h_{io}} : \text{Thermal Resistance at Primary Side} \]
\[ \frac{1}{h_o} : \text{Thermal Resistance at Secondary Side} \]

\( r \): Contamination factor (m²K/W) describes thermal resistance of pipe and is decided experimentally. (MKS Dimension: m²h°C/kcal)

Contamination factor of sodium is smaller one digit comparison with the thermal resistances of primary and secondary. (For instance, 4 × 10⁻⁶ m²h°C/kcal (JOYO IHX))
Administration of Heat Transmission Coe. (U)

【Administration of Heat Transfer’s Decline】
Heat efficiency of HX will decline due to adhesion of rust, trash called scale during operation. Check whether there is any its decline will be done by comparing of calculation results between Heat Balance Formula and Fourier Formula.

Heat Balance Formula
\[ Q = W \cdot C_p \cdot \Delta T \]

Fourier Formula
\[ Q = U \cdot A_o \cdot LMTD \cdot F \cdot t \]

Standard Normal Error Range is less than 10%.

Out: 製作者: 技術教育ソフトウェア、伝熱の基礎コース、© 日本能率協会マネジメントセンター

Caution
On calculation of heat transfer, it must avoid to confuse the dimension system, SI dimension and MKS dimension. Of course, it had better to use the SI dimension.

<table>
<thead>
<tr>
<th></th>
<th>V</th>
<th>( \kappa )</th>
<th>h</th>
<th>Q</th>
<th>( \mu )</th>
<th>Cp</th>
</tr>
</thead>
<tbody>
<tr>
<td>SI Dimension</td>
<td>m³/s</td>
<td>W/m•K</td>
<td>W/m²•K</td>
<td>kJ/s</td>
<td>Pa•s</td>
<td>kJ/kg•K</td>
</tr>
<tr>
<td>MKS Dimension</td>
<td>m³/h</td>
<td>kcal/m•h•°C</td>
<td>kcal/m²•h•°C</td>
<td>kcal/h</td>
<td>Kg/m•h</td>
<td>kcal/kg•°C</td>
</tr>
</tbody>
</table>

引用: 伝熱工学資料 改定第3版、単位換算表、P329-P331、日本機械学会、昭和50年2月
## § 4 : Fluid Flow (Calculation of Pressure Loss)

### Calculation of Pressure Loss at Inside of Pipe (Strait Pipe Portion)

<table>
<thead>
<tr>
<th>Variable</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\Delta P_a$</td>
<td>Friction factor $f$ times average flow rate squared times length divided by twice gravity times density times inner diameter of pipe (Pa)</td>
</tr>
<tr>
<td>$f$</td>
<td>Friction factor</td>
</tr>
<tr>
<td>$\nu$</td>
<td>Average flow rate (m/s)</td>
</tr>
<tr>
<td>$V$</td>
<td>Volume flow rate (m$^3$/s)</td>
</tr>
<tr>
<td>$\rho$</td>
<td>Fluid’s density (kg/m$^3$)</td>
</tr>
<tr>
<td>$D$</td>
<td>Pipe inner diameter (m)</td>
</tr>
<tr>
<td>$L$</td>
<td>Pipe length (m)</td>
</tr>
<tr>
<td>$g$</td>
<td>Gravity acceleration (9.8 m/s$^2$)</td>
</tr>
</tbody>
</table>

**Fundamental Equation**

\[
\Delta P_a = f \frac{\rho \cdot \nu^2}{2g} \cdot \frac{L}{D} \quad (\text{Pa})
\]

- $\nu = \frac{V}{(\pi D^2/4)}$
- $V$: Volume Flow (m$^3$/s)
- $\rho$: Fluid’s Density (kg/m$^3$)
- $D$: Pipe Inner Diameter (m)
- $L$: Pipe Length (m)
- $g$: Gravity Acceleration (9.8 m/s$^2$)

**About Friction Factor ($f$)**

Friction Factor $f$ depends on fluid’s state. That is to say, it changes depending on whether fluid’s state is laminar flow or turbulent flow. Generally, the Moody Curve, shown in next page, is used to get fluid’s friction factor, $f$. [Refer to: http://www.nsknet.or.jp/「管内压力損失」](http://www.nsknet.or.jp/「管内压力損失」)
e: Coarse of Inside Pipe (mm)

⇒ it’s 0 for piping applying to nuclear power plant. Goods on the market is 0.0457

D: Inner Diameter (mm)
Calculation of Pressure Loss at Inside of Pipe (Limited Part)

**Fundamental Equation**

\[ \Delta P_b = \sum \zeta \frac{\rho \cdot \nu^2}{2g} \text{ (Pa)} \]

- \( \zeta \): Resistance Factor at Limited Part
- \( \nu \): Average Flow Rate (m/s)
- \( \nu = \frac{V}{\pi D^2/4} \)
- \( \rho \): Fluid’s Density (kg/m³)
- \( D \): Pipe Inner Diameter (m)
- \( L \): Pipe Length (m)
- \( g \): Gravity Acceleration (9.8 m/s²)

**Resistance Factor: \( \zeta \) (Samples)**

<table>
<thead>
<tr>
<th>Subject</th>
<th>Standard Value</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Expanding by Reducer</td>
<td>0.7</td>
<td>0.4〜1.0</td>
</tr>
<tr>
<td>Reduction by Reducer</td>
<td>0.3</td>
<td>0.15〜0.7</td>
</tr>
<tr>
<td>Bending Part</td>
<td>0.3</td>
<td>Concentrated to 0.3</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Comparatively</td>
</tr>
<tr>
<td>Junction Part</td>
<td>0.7</td>
<td>Almost 0〜1.2</td>
</tr>
<tr>
<td>Butterfly Valve (Open)</td>
<td>0.24</td>
<td>—</td>
</tr>
<tr>
<td>Globe Valve</td>
<td>5.0</td>
<td>—</td>
</tr>
<tr>
<td>Angle Valve</td>
<td>2.6</td>
<td>—</td>
</tr>
</tbody>
</table>

引用: http://www.nsknet.or.jp/「管内圧力損失」
Design of FBR Plant System and its Typical Features

Yoshiaki MATSUNO

September 2004
International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
<table>
<thead>
<tr>
<th>記事</th>
<th>日付</th>
<th>承認</th>
<th>作成担当者</th>
<th>講義担当者</th>
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<td>著・編集・英訳および説明</td>
<td>平成 年 月</td>
<td>松野 義明</td>
<td>松野 義明</td>
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<td>改訂</td>
<td>平成 年 月</td>
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<td>松野 義明</td>
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</tbody>
</table>
### Milestones of Experimental Fast Reactor Joyo

<table>
<thead>
<tr>
<th>Year(s)</th>
<th>Event Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>1970.1</td>
<td>Start of preparation for construction</td>
</tr>
<tr>
<td>1970.3</td>
<td>Construction Start</td>
</tr>
<tr>
<td>1974.12</td>
<td>Construction installation completed</td>
</tr>
<tr>
<td>1975.1-7</td>
<td>Comprehensive function test: In-air test at room temperature</td>
</tr>
<tr>
<td>1975.7~1976.2</td>
<td>Comprehensive function test: In-gas test at high temperature</td>
</tr>
<tr>
<td>1976.2~1977.3</td>
<td>Comprehensive function test: In-sodium test</td>
</tr>
<tr>
<td>1977.3~5</td>
<td>Critical approach (critical 1977.4.24) 64 assemblies</td>
</tr>
<tr>
<td>1977.5~11</td>
<td>Low power test</td>
</tr>
<tr>
<td>1978.4~9</td>
<td>Power increasing test</td>
</tr>
<tr>
<td>1978.10~1979.2</td>
<td>Operation test at 50MWt</td>
</tr>
<tr>
<td>1979.7~8</td>
<td>75MW function test</td>
</tr>
<tr>
<td>1979.8~12</td>
<td>Regular maintenance period</td>
</tr>
<tr>
<td>1980.2~1981.12</td>
<td>Operation test at 75MWt</td>
</tr>
<tr>
<td>1982.1~11</td>
<td>Mark-I → Mark-II</td>
</tr>
<tr>
<td>1986.1</td>
<td>Mark-II critical</td>
</tr>
<tr>
<td>2003.</td>
<td>Mark-III critical</td>
</tr>
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</table>
## Main parameters of Joyo (1)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Mk-Ⅲ (Phase 2)</th>
<th>Mk-Ⅱ (Equilibrium)</th>
<th>Mk-Ⅰ</th>
</tr>
</thead>
<tbody>
<tr>
<td>MW t</td>
<td>75</td>
<td>100</td>
<td>140</td>
</tr>
<tr>
<td>Flow rate of prim. coolant</td>
<td>2200</td>
<td>2200</td>
<td></td>
</tr>
<tr>
<td>Coolant temp. at inlet (°C)</td>
<td>370</td>
<td>370</td>
<td></td>
</tr>
<tr>
<td>Coolant temp. at outlet (°C)</td>
<td>468</td>
<td>500</td>
<td></td>
</tr>
<tr>
<td>Max. power of fuel rod (W/cm)</td>
<td>321</td>
<td>372</td>
<td></td>
</tr>
<tr>
<td>Neutron flux density (av.) ((10^5n/cm^2 \cdot sec))</td>
<td>1.9</td>
<td>2.6</td>
<td></td>
</tr>
<tr>
<td>Neutron flux density (max.) ((10^5n/cm^2 \cdot sec))</td>
<td>3.2</td>
<td>5.1</td>
<td></td>
</tr>
<tr>
<td>No. of cooling circuits (prim./second.)</td>
<td>2/2</td>
<td>2/2</td>
<td>2/2</td>
</tr>
<tr>
<td>No. of prim. main pumps / circuit</td>
<td>1</td>
<td>1</td>
<td>1</td>
</tr>
</tbody>
</table>
Main parameters of Joyo (2)

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Mk- 1</th>
<th>Mk- 2</th>
<th>Mk- 3</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>No. of second. main pumps / circuit</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>No. of IHX / circuit</td>
<td>1</td>
<td>1</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>No. of dump heat exchangers / circuit</td>
<td>2</td>
<td>2</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>Coolant temp. at main IHX</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Prim. Inlet □</td>
<td>468</td>
<td>500</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Prim. outlet □</td>
<td>370</td>
<td>370</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sec. inlet □</td>
<td>350</td>
<td>340</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Sec. outlet □</td>
<td>445</td>
<td>470</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Purpose of the presentation

1. Final and comprehensive safety check of the plant was performed precisely just before the critical test in 1976

2. By engineers and scientists who had not directly involved in the construction work until that time in co-operation with people who had involved in the design and construction work

3. In order to check the plant safety with eyes of the third persons.

4. Establishment of such kind of cross check system seems to be important and effective to find safety problems of the plant.

5. From this activity we found some problems before the critical test.

6. Problems are shown in next slide.
Problems found which required some modifications

1. Modification of hangers and dampers system for primary cooling circuit

2. Deformation of hold-down cylinder for fuel handling machine

3. Modification of support band of the piping for the primary cooling system

4. Modification of hydrostatic bearing of primary sodium pump

5. Change of nitrogen gas bower in the cooling system for concrete shield

6. Measures for floating up of control rods

7. Breaking of blowers of damp heat exchangers
Cutaway of Joyo (常陽)

3 Small rotating plug
4 Large rotating plug
7 Inlet pipe of auxiliary cooling system
8 Inlet pipe of main primary cooling circuit
15 Fuel handling machine
16 Control rod drive mechanism
24 Outlet pipe of main primary cooling circuit
26 Core
30 Reactor vessel
31 Safety vessel
32 Inlet pipe of concrete shield cooling system
Flow sheet of cooling system of Joyo
Temperature and flow rate in equilibrium state (75 MW)
Piping of primary cooling circuit
Thermal transient by reactor scram through neutron flux level high (100 MWt)

1. IHX inlet temp.
2. IHX outlet temp.
3. DHX inlet temp.
4. DHX outlet temp.
5. DHX outlet air temp.

Steep decline by scram

Decline through 1

Air flow rate at DHX

制御系作用による温度上昇
制御系作用による風量低下

1. IHX inlet temp.
2. IHX outlet temp.
3. DHX inlet temp.
4. DHX outlet temp.
5. DHX outlet air temp.
Thermal transient by reactor scram through loss of power (100 MWt)

1. IHX prim. inlet temp.
2. IHX prim. outlet temp.
3. IHX second. inlet temp.
4. IHX second. outlet temp.
5. DHX inlet temp.
6. DHX outlet temp
7. DHX outlet air temp.
P Prim. main pump inlet temp.
S Second. main pump inlet temp.
Bird’s eye view of hot leg of primary main cooling circuit
Modification of support band of the piping for the primary cooling system (1)

1. Piping of the primary main cooling circuit starting from the reactor vessel to the main intermediate heat exchanger (IHX) is called hot leg. Piping starting from the IHX back to reactor vessel flowing through the main circulating pump and electromagnetic flow meter is called cold leg.

2. In general, wall thickness of pipes of FBR cooling circuit is small, because the FBR is operated at low inner pressure and high temperature. Therefore, in order to avoid thermal stress to the piping, redundant length and bendings in 3 dimension are required, being held by hangers and dampers from the walls of reactor building.

3. Joyo adopted for the primary circuit a double-walled pipe for safety sake and therefore the mechanism to support it at hanger and damper was consequently complicated.

4. The pipe is covered by a strengthening band, to which a lug piece is attached for hanger and damper. The band is welded at both sides to the pipe.
5. In case a steep temperature change arises at some reactor transients (reactor scram, loss of power etc) it was revealed by a stress analysis that a very large stress would appear at the peripheral welding line of the inner pipe, which could lead to a serious damage of the pipe, if it is repeated.

6. At the steady state of operation the temperature of the pipe and its attachments is approximately equal to the coolant temperature. However, when the reactor scrams the temperature of the coolant in hot leg declines steeply to cold leg temperature, difference of which reaches more than 100°C when the reactor has been operated at 100 MW.

7. In this transient event large temperature difference would appear between inner pipe and the support band which creates a large stress that exceed the allowable value.

8. To avoid such thermal stress modification was carried out, before the power increasing test.
Modification of support band of the piping for the primary cooling system (3)

9. Also in thermocouple to measure coolant temperature which penetrates both outer and inner pipes and welded to both pipes rigidly, large stress could appear at welded part in case of reactor scram.

10. Modifications were carried out at following locations:

  1) Support band of hot leg pipe of prim. main cooling system (12)
  2) Support band of cold leg pipe of prim. main cooling system (2)
  3) Thermocouple wells attached to hot and cold legs of prim. main cooling system (4)
  4) Thermocouple wells attached to siphon break piping of prim. main cooling system (4)
  5) Thermocouple wells attached to piping for sodium charge and discharge system (16)
Pipe support before modification

Outer tube
Inner tube
Lug
Band
Outer tube
Inner tube
Band
Lug
Band structure after modification
## Stress Analysis and Evaluation for Support Band of Primary Cooling Circuit Pipe

<table>
<thead>
<tr>
<th>Reactor power (Evaluation at the band part)</th>
<th>Primary stress (Inner pressure+Weight+Earthquake) Kg/mm²</th>
<th>Prim.+Second. Stress (Inner pressure+Weight+Earthquake+Thermal expansion+Thermal shock) Kg/mm²</th>
<th>Peak stress (Same to the left) (Max.) Kg/mm²</th>
<th>Fatigue evaluation</th>
<th>Determination</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Stress intensity</td>
<td>Limited value</td>
<td>Amplitude of main stress difference</td>
<td>Limited value</td>
<td>Acc. Damage factor (incl. creep)</td>
</tr>
<tr>
<td>50MWt Initial temp.=435 ºC</td>
<td>1.1</td>
<td>12.5&lt;sup&gt;1)&lt;/sup&gt;</td>
<td>38.8</td>
<td>116.7 (Ke&lt;sup&gt;2&lt;/sup&gt;=1.5)</td>
<td>5.001</td>
</tr>
<tr>
<td>100MWt Initial temp.=500 ºC</td>
<td>1.1</td>
<td>12.5&lt;sup&gt;1)&lt;/sup&gt;</td>
<td>74.9</td>
<td>498.2 (Ke&lt;sup&gt;2&lt;/sup&gt;=3.33)</td>
<td>&gt;1.0</td>
</tr>
</tbody>
</table>

1) Value based on 1.33Sm(t)=0.33S  
2) Ke: Factor to be multiplied, if prim.+second. stress > limited value  
3) OK for 4 years of operation
Results of stress analysis after modification and strength evaluation

<table>
<thead>
<tr>
<th>Reactor power</th>
<th>Primary stress</th>
<th>Primary+Secondary stress</th>
<th></th>
<th>Peek stress (max) (kg/mm²)</th>
<th>Fatigue evaluation</th>
<th>Determination</th>
</tr>
</thead>
<tbody>
<tr>
<td>100MWₜ (500 □)</td>
<td>0.8</td>
<td>12.6</td>
<td>16.6</td>
<td>3Sm(t)= 30.6 (500 □)</td>
<td>0.632</td>
<td>1.0</td>
</tr>
</tbody>
</table>
Types of FBR

Loop type

Pool type or Tank type
## 5.1 Difference between FBR and LWR from viewpoint of structural design

### Comparison of temperatures and pressures of LWR and FBR (Some examples)

<table>
<thead>
<tr>
<th>Items</th>
<th>LWR</th>
<th>FBR</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>LWR</td>
<td>FBR</td>
</tr>
<tr>
<td></td>
<td>PWR (1,100 MWe)</td>
<td>Loop type (Monju) (280 MWe)</td>
</tr>
<tr>
<td></td>
<td>BWR (1,100 MWe)</td>
<td>Tank type (SPX-1) (1,200 MWe)</td>
</tr>
<tr>
<td>Coolant temp. at reactor outlet (Av.)</td>
<td>~325 °C</td>
<td>529 °C</td>
</tr>
<tr>
<td></td>
<td>~286 °C</td>
<td>545 °C</td>
</tr>
<tr>
<td>Temperature difference between reactor vessel out-</td>
<td>~35 °C</td>
<td>132 °C</td>
</tr>
<tr>
<td>and inlet</td>
<td>(325/290)</td>
<td>(529/397)</td>
</tr>
<tr>
<td></td>
<td>~7 °C</td>
<td>150 °C</td>
</tr>
<tr>
<td>(286/279)</td>
<td></td>
<td>(545/395)</td>
</tr>
<tr>
<td>Coolant pressure (Max)</td>
<td>157 kg/cm²G</td>
<td>~8 kg/cm²G</td>
</tr>
<tr>
<td></td>
<td>~70 kg/cm²G</td>
<td>~8 kg/cm²G</td>
</tr>
</tbody>
</table>

- **LWR**
  - Wall: Thick
  - Main load: Pressure

- **FBR**
  - Wall: Thin
  - Main load: Thermal and seismic
5. Outline of structural design of FBR

5.2 Comparison of reactor vessel structures of PWR and FBR (3)

- **LWR**
  - Load controlled stress (1st stress)
  - Thermal stress: small
  - Membrane stress: small
  - Inner pressure: ~150 atm (PWR)
  - Fracture mode: Ductile fracture (Creep fracture at high temp.)
  - Wall thickness: large (PWR)

- **FBR**
  - Displacement controlled stress (2nd stress)
  - Thermal stress: large
  - Seismic load: small
  - Inner pressure: ~10 atm
  - Membrane stress: small
  - Wall thickness: small
  - Fracture mode: Fatigue fracture (creep) and large deformation by repeated stress

---

Outsource: 日本応用講座、構造健全性 - 構造設計
5. Outline of structural design of FBR

5.3 Typical features of structural design for FBR (2)

Assumptions of fracture modes

1) Ductile fracture / Plastic break
2) Large plastic deformation
3) Fatigue fracture
4) Elastic-plastic buckling
5) Creep fracture
6) Large creep deformation
7) Creep fatigue fracture
8) Creep buckling

Same as those of LWR (Low temperature: Non-creep region)

Typical to FBR (High temperature: Creep region)
Fundamental flow of structural design (1)

- Plant operation modes
- Structure of fundamental systems
- Condition of function
- Design condition
- Installation condition

- Thermal transient analysis
- Material
- Basic str. analysis

- Thermal transient condition
- Structure analysis
- Piping rout

- Inner thermal transient analysis
- Performance confirmation

- Temperature distribution analysis
  - Envelop of thermal transients

- Temperature distribution

- Basic structure of equipment
  - Selection of analysis object
  - Analysis model
  - Stress analysis by thermal load
  - Stress analysis by mechanical load

- Piping analysis
  - (Therm.expan., Earthqu., Tare)
  - Pipe stress
  - Forced displacement of nozzles

- Seismic analysis
  - Seismic model
    - Seismic static analysis
    - Seismic dynamic analysis
  - Seismic load (Sharing, Moment)

- Floor response curve

ASSUMED LOAD

STRUCTURAL ANALYSIS

To be continued

出典 ☑️応用講座 Ⅱ、構造健全性 - 構造設計
Evaluation by design standard

- Superposition of stresses
- Selection of evaluation surface
- Classification of stresses
- Calculation of stress strength
- Limitation on primary stress
- Limitation of strain
- Limitation of deterioration by creep fatigue
- End of structural design

Strength evaluation
Conclusion

1. Establishment of plant safety check system introducing different eyes

2. Open minded discussion between the scientists and engineers having been involved and the newly involved

3. If some problems are found to be modified, then the way of modification should be determined, taking the influences of the modification all over the plant into account.

4. Each operator should learn operation of the plant not only from an operating manual but also by complete understanding and deep knowledge about plant structure, dynamics and functions in order to be able to cope with emergency conditions.
Control Ability of FBR core

Akihiro Kitano

October 21, 2004

International Cooperation & Technology Development Center
TSURUGA Headquarter Office, JNC
Contents

1 Introduction of FBR core characteristics
2 Why is it possible to control a nuclear reactor?
3 Reactivity and prompt criticality
4 Reactivity control and self-regulating characteristics
5 Control rod drive concept
# 1 Introduction of FBR core characteristics

## Comparison Features of reactors

<table>
<thead>
<tr>
<th>LW : Light Water</th>
<th>CR : Conversion Ratio</th>
</tr>
</thead>
</table>

<table>
<thead>
<tr>
<th>Neutron</th>
<th>LWLW3-5%</th>
<th>95-97%</th>
<th>U235</th>
<th>U238</th>
<th>Thermal</th>
<th>LWR</th>
<th>Pu</th>
<th>U (depleted)</th>
<th>Fast FBR</th>
</tr>
</thead>
<tbody>
<tr>
<td>CRCoolantModeratorFuelMajor</td>
<td>LWLW3-5%</td>
<td>95-97%</td>
<td>U235</td>
<td>U238</td>
<td>Thermal</td>
<td>LWR</td>
<td>Pu</td>
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<td>LWR</td>
<td>Pu</td>
<td>U (depleted)</td>
<td>Fast FBR</td>
</tr>
</tbody>
</table>
Conversion chain of nuclides (main reaction only)
2 Why is it possible to control a nuclear reactor?

Prompt neutron
Neutrons generated promptly by fission reaction. (235U 99.35%, 239Pu 99.78%)

Delayed neutron
- Some daughter nuclides of FP produce neutrons which are called “delayed neutron”.
- Delayed neutron is divided into 6 groups according to time constant of the precursor.

(Example) 1st group: \(^{87}\text{Br}\), 2nd group: \(^{137}\text{I}\), ...
\(^{87}\text{Br}\) \(\overset{\text{- decay}}{\longrightarrow}^{87}\text{Kr}\) \(\overset{\text{- decay with neutron emission}}{\longrightarrow}^{87}\text{Rb}\) \(\overset{\text{- decay}}{\longrightarrow}^{87}\text{Sr}\)

Delayed neutron fraction
The delayed neutron fraction of i-th group is expressed by \(D_i\). Total delayed neutron fraction is expressed by
\[
\sum_{i=1}^{6} D_i
\]

(Example) \(D\) of U-235 is 0.0065. (Only 1 of 150 neutrons in the reactor is delayed neutron)
Relation between prompt neutron and delayed neutron

Prompt neutron

Fission Product (FP) → Prompt emission → Neutron

2MeV

\[
\begin{array}{c}
\text{235U : 99.35 \%} \\
\text{239Pu : 99.78 \%}
\end{array}
\]

---

Delayed neutron

Fission Product (FP) → Delayed emission → Neutron

500keV

\[
\begin{array}{c}
\text{235U : 0.65 \%} \\
\text{239Pu : 0.22 \%}
\end{array}
\]

\( ^{87}\text{Br}, ^{87}\text{I}, \text{etc.} \) (Delayed neutron precursor)
### Delayed Neutron Fraction of 235U and 239Pu

<table>
<thead>
<tr>
<th>Group</th>
<th>Half Life (sec)</th>
<th>Decay Constant (sec⁻¹)</th>
<th>The Number of Delayed Neutrons per Fission</th>
<th>Delayed Neutron Fraction</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>54.51</td>
<td>0.0127</td>
<td>0.00063</td>
<td>0.000243</td>
</tr>
<tr>
<td>2</td>
<td>21.84</td>
<td>0.0317</td>
<td>0.00351</td>
<td>0.001363</td>
</tr>
<tr>
<td>3</td>
<td>6.00</td>
<td>0.115</td>
<td>0.00310</td>
<td>0.001203</td>
</tr>
<tr>
<td>4</td>
<td>2.23</td>
<td>0.311</td>
<td>0.00672</td>
<td>0.0026055</td>
</tr>
<tr>
<td>5</td>
<td>0.49</td>
<td>1.40</td>
<td>0.00211</td>
<td>0.0008196</td>
</tr>
<tr>
<td>6</td>
<td>0.216</td>
<td>3.21</td>
<td>0.00043</td>
<td>0.000166</td>
</tr>
</tbody>
</table>

Total delayed neutrons / fission: 0.0165
Total delayed neutron fraction: 0.0064

---

### Delayed Neutron Fraction of 235U and 239Pu

<table>
<thead>
<tr>
<th>Group</th>
<th>Half Life (sec)</th>
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</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>53.75</td>
<td>0.0129</td>
<td>0.00024</td>
<td>0.0000760</td>
</tr>
<tr>
<td>2</td>
<td>22.29</td>
<td>0.0311</td>
<td>0.00179</td>
<td>0.00056003</td>
</tr>
<tr>
<td>3</td>
<td>5.19</td>
<td>0.134</td>
<td>0.00138</td>
<td>0.0004320</td>
</tr>
<tr>
<td>4</td>
<td>2.09</td>
<td>0.331</td>
<td>0.00210</td>
<td>0.0006560</td>
</tr>
<tr>
<td>5</td>
<td>0.549</td>
<td>1.26</td>
<td>0.00066</td>
<td>0.0002060</td>
</tr>
<tr>
<td>6</td>
<td>0.216</td>
<td>3.21</td>
<td>0.00022</td>
<td>0.0000700</td>
</tr>
</tbody>
</table>

Total delayed neutrons / fission: 0.0064
Total delayed neutron fraction: 0.0020

---

出典 : FBR広報素材資料 2版, 6-11遲発中性子、平成2年3月
The effective multiplication factor is defined as the number of fission neutrons in one generation divided by the number of neutrons in the preceding generation.

\[ k_{\infty} = \eta \cdot \varepsilon \cdot p \cdot f \]

\[ k_{\text{eff}} = \eta \cdot \varepsilon \cdot p \cdot f \cdot Pf \cdot Pt \]

- \( k_{\text{eff}} < 1 \)
- \( k_{\text{eff}} = 1 \)
- \( k_{\text{eff}} > 1 \)

Neutron

Fission by thermal neutron

Fission by fast neutron

Reproduction factor ( \( \square \) )

Fast fission factor ( \( \square \) )

Total number of fission neutrons ( \( \square \) )

Decrease of neutrons by resonance absorption. (\( p \) : Probability that neutron escape from resonance)

Decrease of fast neutrons by leakage from the reactor (\( Pf \) : Probability that fast neutron does not leak)

The number of neutrons surviving in thermal energy (\( p Pf \))

Decrease of neutrons by absorption in other material than fuel. (\( f \) : Probability that fast neutron is not absorbed)

Decrease of fast neutrons by leakage from the reactor (\( Pt \) : Probability that fast neutron does not leak)
The average neutron energy in the Monju core is 120keV.

Neutrons produced per absorption in LWR: 2

Neutrons produced per absorption in FBR: 3

Energy dependence of...
1. Reactor kinetics in case without delayed neutrons

At the beginning, we discuss the reactor kinetics without delayed neutrons. We assume the infinite system of reactor and derive a simple formula describing the time-dependent behavior of neutron population as a function of the multiplication factor and the neutron lifetime. Let us define:

\[ n(t) = \text{neutron population in reactor at time } t \text{ [neutrons]}, \]
\[ P(t) = \text{neutron production rate at time } t \text{ [neutrons/s]}, \]
\[ L(t) = \text{neutron consumption rate at time } t \text{ [neutrons/s]}. \]
\[ l = \frac{n(t)}{L(t)} = \text{mean neutron lifetime.} \text{[s]} \]

\[ \frac{dn}{dt} = P(t) - L(t) = \left[ \frac{P(t)}{L(t)} - 1 \right] L(t) = (k - 1)L(t) \]

\[ \frac{dn}{dt} = \frac{k - 1}{l} n(t) \]

\[ n(t) = n_0 \exp \left[ \frac{(k - 1)}{l} t \right] \]
Tp : Reactor period

“Tp = 100(s)” means “it takes 100 sec. to increase reactor power to 2.7 times higher”.

\[ P = P_0 e^{t/T_p} \]
Reactor kinetics in case without delayed neutron

**Prompt neutron**

**Delayed neutron**

- Period ($T_p$) = $\frac{l}{(k-1)}$

Neutron life time:
The time of a neutron from birth to death in a reactor

**LWR**

- Neutron life time (prompt neutron) + Neutron life time (delayed neutron) = $10^{-4}$ + $0.06$

**FBR**

- Neutron life time = $10^{-4}$

Reactor Period ($T_p$) = $\frac{1.000}{1.001} - 1$

1 second later → $(2.7183)^{10} \approx 22,000$ times higher!
2. Reactor kinetics in case with delayed neutrons

Delayed neutron fraction is generally expressed by \( \bar{f} \). Prompt neutron fraction is expressed by \( (1 - \bar{f}) \). That is why neutron life time is shown as

\[
\lambda = (1 - \bar{f}) \lambda_p + \sum_{i=1}^{6} \bar{f} \lambda_i
\]

\( \lambda_p \): Prompt neutron lifetime

\( \lambda_i \): Delayed neutron lifetime (\( i \)-th group).

Delayed neutron lifetime (\( \lambda_i \)) almost equal to delayed neutron precursor lifetime (\( \tau_i \)).

\[
\lambda = (1 - \bar{f}) \lambda_p + \sum_{i=1}^{6} \bar{f} \tau_i
\]
Delayed neutron term \((\sum_{i=1}^{6} t_i)\) is estimated to be 0.1~0.085. Total neutron lifetime is

\[
l = l_p + \sum_{i=1}^{6} t_i \times 10^{-4} + 0.1 \times 0.1 \text{sec.}
\]

It is assumed that \(k\) is increased by 0.1\% with \(l = 0.1 (s)\).

\[
T = \frac{l}{(k - 1)} = \frac{0.1}{(1.001 - 1)} = 100 (s)
\]

\[
N_{F(t)} = N_{F(0)}e^{t/T} = N_{F(0)}e^{t\times100}
\]

This implies that it takes approximately 100 (s) to get the reactor power “e” times higher, which means we can sufficiently control the reactor power by control rod operation.
Reactor kinetics in case with delayed neutron

Neutron life time: The time of a neutron from birth to death in a reactor

Neutron life time (prompt neutron) + Neutron life time (delayed neutron)

$LWR$  $FBR$

$\lambda = \lambda_p + \lambda_L = 10^{-4} + 0.085 \ [	ext{s}]$

$\lambda = \lambda_p + \lambda_L = 10^{-7} + 0.06 \ [	ext{s}]$

Reactor Period ($T_p$) = $\frac{l}{(k-1)}$

In case with delayed neutron

$\lambda_p + \lambda_L$

$= 10^{-4} + 0.085 \div 0.1 = 0.1 \ [	ext{s}]$

$\text{Period (}T_p\text{)} = \frac{\lambda_p + \lambda_L}{k_{\text{eff}} - 1}$

$= \frac{0.1}{1.001 - 1}$

$= 100 \ [	ext{s}]$

$1 \text{ second later} \rightarrow (2.7183)^{1/100} \approx 1.2 \times \text{higher}$

$k_{\text{eff}} = 1.000 \times 1.001$

Critical state

Prompt neutron

Delayed neutron

出展：製作者：技術教育ソフトウェア、原子炉物理コース、©(株)日本能率協会マネジメントセンター
Nuclear reactor is designed to keep critical state with delayed neutron. Delayed neutrons enable the reactor to safely achieve critical condition.

Reactor period depends on the neutron lifetime.

Prompt neutron lifetime is so short compared with the delayed neutron lifetime that reactor period depends on the lifetime of delayed neutron precursor.

Reactor period is generally estimated from 60s (LWR) to 100s (FBR). That enables the reactor sufficiently control the reactor power by control rods operation.
Reactivity is often described in unit of "dollar and cent (¢ and $)".

- What is "dollar and cent (¢ and $)"?
- How is relation between “prompt criticality” and “reactivity”?

Reactivity means the deviation of the multiplication factor from critical state. Reactivity is defined as

\[ \Delta = \frac{K_{eff} - 1}{K_{eff}} \] (dollar ($) and cent (¢))

If the multiplication factor (keff) is more than 1.0, reactivity is positive and fission reaction rate increase. Inversely if it is less than 1.0, the reactivity is negative and the fission reaction rate decrease. Consequently reactivity-0-state is “critical” and reactor is operated in “critical” state with constant power.

\[ \text{keff} > 1.0 \quad \Delta > 0 \quad (\text{Positive Supercritical}) \]
\[ \text{keff} = 1.0 \quad \Delta = 0 \quad (\text{Critical}) \]
\[ \text{keff} < 1.0 \quad \Delta < 0 \quad (\text{Negative Subcritical}) \]
Reactivity \( \rho = \frac{(k_{\text{eff}} - 1)}{k_{\text{eff}}} \times 100 \)

Unit: \( \% \) \( \frac{\text{k/k}}{} \)

Subcritical: \( \rho < 0 \)

Critical: \( \rho = 0 \)

Supercritical: \( \rho > 0 \)

Insert of positive reactivity (Withdrawal of CR)

Insert of negative reactivity (Insert of CR)
(1 - $\Delta$)K$_{\text{eff}}$ < 1.0

In case of $(1 - \Delta)K_{\text{eff}} < 1.0$ the critical state can not be achieved only by prompt neutrons. After change of reactivity, neutron flux increases instantly by prompt neutrons which occupied by 99.45% in total neutrons.

After that, neutron flux increase gradually by delayed neutrons to reach the critical state.

$(1 - \Delta)K_{\text{eff}} > 1.0$

In case of $(1 - \Delta)K_{\text{eff}} > 1.0$, the critical state can be achieved only by prompt neutrons. This case is called prompt critical state.

Prompt critical state = $(1 - \Delta)K_{\text{eff}} = 1.0$.

\[ \Delta = \frac{K_{\text{eff}} - 1}{K_{\text{eff}}} \quad K_{\text{eff}} = \frac{1}{1 - \Delta} \quad (1 - \Delta)\frac{1}{1 - \Delta} = 1 \quad \Delta \Delta = \Delta \]

The value of reactivity $\Delta$ is expressed by $\$ (= 100 $\text{C} \text{) as reactivity unit.}$

$^{35}\text{U} : 0.0065 \ (0.65\%)$

$^{239}\text{Pu} : 0.0021 \ (0.21\%)$
**Reaction rate**

- \( n \): Neutron density
- \( v \): Velocity

\[
\text{Reaction rate} = \frac{\#}{\text{sec} \cdot \text{cm}^3} \times \frac{\text{cm}}{\text{sec}} \times \frac{\#}{\text{cm}^3} \times \frac{\text{cm}^2}{\text{cm}^3} \times \frac{\#}{\text{cm}^3}
\]

\[
\phi = \frac{\#}{\text{sec} \cdot \text{cm}^2} \quad \text{Neutron flux}
\]

\[
\Sigma = \sigma N \quad \text{Macroscopic cross section}
\]
Analysis of neutron flux

Neutron density $n(r, E, t)$

Neutron density $\phi(r, E, t) = \nabla n(r, E, t)$

$E = \frac{1}{2} m v^2$

Derivation of diffusion equation

$$\frac{\partial n}{\partial t} = \text{Generation} - \text{Absorption} - \text{Leakage}$$

Production $= v \Sigma_f \phi + S$ \hspace{1cm} \text{Internal neutron source}$

Consumption $= \Sigma_a \phi \quad \Sigma_a = \Sigma_f + \Sigma_c + \cdots$
Leakage = \nabla \cdot \mathbf{J} = -D \nabla^2 \phi

□ Gauss’s law  \int \nabla \cdot \mathbf{J} \, dV = \int \mathbf{J} \cdot d\mathbf{S}

□ Fick’s law  \mathbf{J} = -D \nabla \phi

Diffusion coefficient

\[ D = \frac{1}{3 \Sigma_{tr}} \Sigma_{tr} \text{ transport cross section} \]

\[ \Sigma_{tr} = \Sigma_t - \Sigma_s \]

\[ 1 \frac{\partial \phi}{\partial t} = D \nabla^2 \phi - \Sigma_a \phi + v \Sigma_f \phi + S \]

Diffusion equation
Reactor power changing with time can be estimated by the reactor kinetic equation.

- Effect of delayed neutron
- The flux distribution shape does not change with time.
  (point reactor kinetics model)

The equation of birth and death of delayed neutron precursor

\[
\frac{\partial C_i}{\partial t} = \beta_i \nu \Sigma_f \phi - \lambda_i C_i
\]

Time dependent diffusion equation for 1 energy group is shown as

\[
\frac{1}{\Box} \frac{\partial \phi}{\partial t} - D \nabla^2 \phi + \Sigma_a \phi = (1 - \beta) \nu \Sigma_f \phi + \sum_{i=1}^{6} \lambda_i C_i + S
\]
Variables separation

\[ \phi (\mathbf{r}, t) = \Box h (t) \phi_1 (\mathbf{r}) \]

\[ C_i (\mathbf{r}, t) = C_i (t) \phi_1 (\mathbf{r}) \]

\[ \frac{dn}{dt} + DB^2 \Box n + \Sigma_a \Box n = (1 - \beta) \nu \Sigma_f \Box n + \sum_{i=1}^{6} \lambda_i C_i + S \]

\[ \frac{dn}{dt} = \left[ (1 - \beta) \frac{\nu \Sigma_f}{\Sigma_a + DB^2} - 1 \right] \left( \Sigma_a + DB^2 \right) \Box n + \sum_{i=1}^{6} \lambda_i C_i + S \]

\[ k_{\text{eff}} \]

\[ \therefore \frac{dn}{dt} = \frac{k(1 - \beta) - 1}{l} n(t) + \sum_{i=1}^{6} \lambda_i C_i (t) + S \]

Neutron lifetime \( (l) = \frac{n}{(\Sigma_a + DB^2) \Box n} \)
Pont reactor kinetics equation is expressed by

\[
\frac{dC_i}{dt} = \beta_i \left( \sum_{\nu} n(t) - \lambda_i C_i(t) \right)
\]

Neutron generation time \( \Lambda \) is defined as

\[
\Lambda = \frac{k - 1}{k}
\]

Reactivity \( \rho \) is given by

\[
\rho = \frac{k - 1}{k}
\]

\[
\left\{ \begin{array}{l}
\frac{dn}{dt} = \frac{\rho - \beta}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t) + S \\
\frac{dC_i}{dt} = \frac{\beta_i}{\Lambda} n(t) - \lambda_i C_i(t)
\end{array} \right.
\]
4 Reactivity control and self-regulating characteristics

The reactivity on condition of all CR withdrawal is “excess reactivity ($\bar{\Sigma}_E$)”, which means the reactivity of reactor core itself.

**LWR**

It is necessary to compensate for large reactivity decrease by fuel burnup.

#1 Lower conversion ratio in LWR causes larger reactivity decrease by fuel burnup.

#2 The poison FP such as Xe and Sm work as neutron absorber.

The reactivity decreases by the reactor operation.

Excess reactivity: approx. 25-30%

**FBR**

The burnup reactivity decrease to be compensated is not so large as LWR.

#1 The burnup reactivity decrease is smaller than in LWR because of the higher conversion ratio. (LWR:0.6, FBR:0.8-0.9)

#2 The poison FP (Xe, Sm) does not affect the reactivity of FBR core.

Excess reactivity: approx. 5%
Control Rods have reactivity worth sufficient enough to cover the excess reactivity and further more worth called “Shut-down reactivity margin”.

Reactivity coefficient means the reactivity associated with those of operating temperature and core dimension.

Reactor has self-regulating characteristics and make it possible to control the fission chain reaction automatically. This inherent reactivity effect is called “self-regulating characteristics”.

Nuclear reactor should be designed to make power coefficient as total reactivity effect negative.
The positive reactivity of Reactor power

The negative reactivity of CR

Excess reactivity

Reactivity margin

- CRs are rods inserted. (subcritical state)
- CRs are withdrawn. (critical state in low temperature condition)
- CRs are withdrawn. (critical state in high temperature condition)

Reactivity margin

• \( R_{\text{margin}} = R_{\text{excess}} + R_{\text{positive}} \)
<table>
<thead>
<tr>
<th></th>
<th>CCR</th>
<th>FCR</th>
<th>BCR</th>
<th>CCR</th>
<th>FCR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Power compensation</td>
<td>1.9</td>
<td>1.9</td>
<td>1.7</td>
<td>1.7</td>
<td></td>
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<tr>
<td>Burnup compensation</td>
<td>2.5</td>
<td>-</td>
<td>2.6</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>Margin for operation</td>
<td>0.3</td>
<td>-</td>
<td>0.3</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>Uncertainty in reactivity evaluation</td>
<td>1.0</td>
<td>-</td>
<td>1.0</td>
<td>-</td>
<td></td>
</tr>
<tr>
<td>Reactivity requirement</td>
<td>5.7</td>
<td>1.9</td>
<td>5.6</td>
<td>1.7</td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>7.1*</td>
<td>5.9</td>
<td>7.0*</td>
<td>5.8</td>
<td></td>
</tr>
<tr>
<td>Reactivity worth</td>
<td>1.4</td>
<td>4.0</td>
<td>1.4</td>
<td>4.1</td>
<td></td>
</tr>
</tbody>
</table>

* 設置許可申請書 添八（添付資料）
**Self-regulating characteristics**

Nuclear reactor has inherent safe characteristics in reactivity which controls fission chain reaction automatically.

- Doppler effect
- Negative reactivity
- Decrease of fission chain reaction
- Decrease in temperature (power)
- Stable state

Atoms of $^{238}\text{U}$ vibrate by heat movement. Therefore, the relative velocity of neutron to $^{238}\text{U}$ is changed in some kinetic energy rage. Fuel temperature increase can increase the $^{238}\text{U}$ vibration and broaden the resonance energy range. Consequently, $^{238}\text{U}$ resonance capture increases and works as negative reactivity effect.
### Reactivity Coefficients of Monju

<table>
<thead>
<tr>
<th>Coefficient</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Doppler coefficient *)1</td>
<td>$- (5.7 \sim 7.6) \times 10^{-3} , T , dk/dT$</td>
</tr>
<tr>
<td>Fuel temperature coefficient</td>
<td>$- (3.3 \sim 3.9) \times 10^{-6} , k/k/\circ$</td>
</tr>
<tr>
<td>Structural material temperature coefficient</td>
<td>$+ (6.0 \sim 10.0) \times 10^{-7} , k/k/\circ$</td>
</tr>
<tr>
<td>Coolant temperature coefficient *)2</td>
<td>$+ (1.0 \sim 14.0) \times 10^{-7} , k/k/\circ$</td>
</tr>
<tr>
<td>Core support plate temperature coefficient *)3</td>
<td>$- (10.0 \sim 12.0) \times 10^{-6} , k/k/\circ$</td>
</tr>
<tr>
<td>Power coefficient (Total reactivity reactivity *)4</td>
<td>$- (9.4 \sim 11.0) \times 10^{-6} , k/k/MW$</td>
</tr>
<tr>
<td>Sodium void effect (Max. F/A)</td>
<td>$+ (1.1 \sim 1.5) \times 10^{-4} , k/k$</td>
</tr>
</tbody>
</table>

*)1: High temperature of fuel increase neutron absorption because of broadening resonance energy range.

*)2: Temperature increase reduce the coolant number density, which means more neutron leakage and harder neutron spectrum. The former affects as negative reactivity, the latter as positive reactivity. In case of Monju the former effect is smaller because of the middle size reactor. Totally this phenomenon affects as positive reactivity.

*)3: The core support plate broadening decreases the number density and works as negative reactivity.

*)4: The Doppler effect and the core plate broadening effect are bigger than others.
1. Summary (1)

Control of reactivity of Monju core is performed by control rods of 10 CCRs (Coarse Control Rods) and 3 FCRs (Fine Control Rods). Furthermore, 6 BCRs (Backup Control Rods) are provided in Monju.

CCR and FCR are used in start-up, normal shutdown and power control. In case of off-normal condition with shutdown, all control rods including BCR are rapidly inserted into the core at once by the signal from the plant protection system.
1. Summary (2)

- The main shutdown system and the backup shutdown system are independent from each other. The structural composition such as driving system and rod drop system is diverse form each type.
  (FCR: Hitachi, CCR: Mitsubishi, BCR: Toshiba)

- Upper core structure is equipped with control rods. Core upper structure is cylindrical structure with thin material supported by the rotating plug.

- Control rods are designed to work surely even in case of huge earthquake.
  (The relative change between the top of the reactor core and the bottom of the upper core structure is limited.)
Plan view of Monju core

- Fine Control Rods
- Coarse Control Rods
- Backup Control Rods

Inner Core Region
Outer Core Region
Radial Blanket Region
## 2. Control Rod Drive Mechanism (FCR, CCR)

<table>
<thead>
<tr>
<th></th>
<th>Fine Control Rod</th>
<th>Coarse Control Rod</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number</td>
<td>3</td>
<td>10</td>
</tr>
<tr>
<td>Stroke</td>
<td>1000 mm</td>
<td>1000 mm</td>
</tr>
<tr>
<td>Driving speed</td>
<td>30～300(Max.) mm/min (Variable)</td>
<td>120(Max.) mm/min (Constant)</td>
</tr>
<tr>
<td>Insertion time in shutdown</td>
<td>1.2 sec. or less</td>
<td>1.2 sec. or less</td>
</tr>
<tr>
<td>Drive system</td>
<td>Motor</td>
<td>Motor</td>
</tr>
<tr>
<td>Scram system</td>
<td>- Electromagnetic system - Gas acceleration system - Stroke/latch rod and FCR is inserted together</td>
<td>- Electromagnetic system - Gas acceleration system - Stroke/latch rod and FCR is inserted together</td>
</tr>
</tbody>
</table>
The maximum motor frequency is physically limited by electrical source and load. That is the main reason for the withdrawal speed less than the motor frequency.

FCR is driven with the variable speed (30-300 mm/min) only in the automatic operation mode for more than 40% power range. In case of power range less than 40%, it is driven with 120mm/min constant speed same as CCR.

The electromagnetic force is to be eliminated by the reactor trip signal. Control rod drops, being accelerated by gas pressurization.

It takes less than 1.2 sec. to get the control rod fully inserted by 85% with reactor trip signal.
### 3. Control Rod Drive Mechanism (BCR)

<table>
<thead>
<tr>
<th></th>
<th>BCR</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Number</strong></td>
<td>6</td>
</tr>
<tr>
<td><strong>Stroke</strong></td>
<td>1100 mm</td>
</tr>
<tr>
<td><strong>Driving speed</strong></td>
<td>180 mm/min (Constant)</td>
</tr>
<tr>
<td><strong>Insertion time in shutdown</strong></td>
<td>1.2 sec. or less</td>
</tr>
<tr>
<td><strong>Drive system</strong></td>
<td>Motor</td>
</tr>
<tr>
<td><strong>Scram system</strong></td>
<td>- Electromagnetic system</td>
</tr>
<tr>
<td></td>
<td>- Accelerated by spring</td>
</tr>
<tr>
<td></td>
<td>- Strike/latch rod and FCR is inserted separately.</td>
</tr>
</tbody>
</table>
• All BCRs are kept fully withdrawn during normal operation.

• The maximum motor frequency is physically limited by electrical source and load. That is the main reason for the withdrawal speed less than the motor frequency.

• The electromagnetic force is to be eliminated by the reactor trip signal. Control rod drops and is accelerated by spring force.

• It takes less than 1.2 sec. to get the control rod fully inserted by 85% with reactor trip signal.
<table>
<thead>
<tr>
<th>Item</th>
<th>Going down by driving motor</th>
<th>Activation of electromagnet</th>
<th>Going down by latch motor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Driving position</td>
<td>+50mm</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Latch position</td>
<td></td>
<td></td>
<td>-65mm</td>
</tr>
<tr>
<td>Item</td>
<td>Going down by driving motor</td>
<td>Going up by latch motor</td>
<td>Going down by driving motor</td>
</tr>
<tr>
<td>------</td>
<td>-----------------------------</td>
<td>-------------------------</td>
<td>-----------------------------</td>
</tr>
<tr>
<td>Driving position</td>
<td>-6mm</td>
<td></td>
<td>0mm</td>
</tr>
<tr>
<td>Latch position</td>
<td></td>
<td>0mm</td>
<td></td>
</tr>
<tr>
<td>Item</td>
<td>Initial position</td>
<td>Going down by driving motor</td>
<td>Going down by latch motor</td>
</tr>
<tr>
<td>------------</td>
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<td>-----------------------------</td>
<td>--------------------------</td>
</tr>
<tr>
<td>Driving position</td>
<td>0mm</td>
<td>-6mm</td>
<td>-65mm</td>
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<tr>
<td>Latch position</td>
<td>0mm</td>
<td>0mm</td>
<td>-65mm</td>
</tr>
<tr>
<td>Item</td>
<td>Driving position</td>
<td>Latch position</td>
<td></td>
</tr>
<tr>
<td>----------------------------------------------------------------------</td>
<td>------------------</td>
<td>----------------</td>
<td></td>
</tr>
<tr>
<td>Going up by driving motor</td>
<td>+50mm</td>
<td>+15mm</td>
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<td>Going up by latch motor</td>
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</tr>
<tr>
<td>Inactivation of electromagnet</td>
<td></td>
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</tr>
<tr>
<td>Going up by driving motor</td>
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</tbody>
</table>

**Picture**

- Going up by driving motor
- Going up by latch motor
- Inactivation of electromagnet
- Going up by driving motor
<table>
<thead>
<tr>
<th>Item</th>
<th>□ Going down by driving motor</th>
<th>□ Going up by latch motor</th>
<th>□ Going down by driving motor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Driving position</td>
<td>+50mm</td>
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<td>0mm</td>
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<tr>
<td>Latch position</td>
<td>-19mm</td>
<td>0mm</td>
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</tr>
<tr>
<td>Item</td>
<td>Going down by driving motor</td>
<td>Activation of electromagnet</td>
<td>Going up by driving motor</td>
</tr>
<tr>
<td>---------------------------</td>
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<tr>
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<td><img src="image3.png" alt="Diagram" /></td>
</tr>
<tr>
<td>Driving position</td>
<td>-25mm</td>
<td></td>
<td>0mm</td>
</tr>
<tr>
<td>Latch position</td>
<td></td>
<td></td>
<td></td>
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</tbody>
</table>

- CCR latch operation
- Going down by driving motor
- Activation of electromagnet
- Going up by driving motor
<table>
<thead>
<tr>
<th>Item</th>
<th>Initial position</th>
<th>Inactivation of electromagnet</th>
<th>Going up by driving motor</th>
</tr>
</thead>
<tbody>
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<td><img src="image" alt="Picture" /></td>
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<tr>
<td>Driving position</td>
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<td>Latch position</td>
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<tr>
<td>Item</td>
<td>Going down by C latch motor</td>
<td>Going up by driving motor</td>
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<tr>
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<td><img src="image2.png" alt="Diagram" /></td>
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<td>Driving position</td>
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<td>+1004mm</td>
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<tr>
<td>Latch position</td>
<td>-19mm</td>
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<td>Item</td>
<td>Going down by driving motor</td>
<td>Going down by driving motor</td>
<td>Activation of electromagnet</td>
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<td>-53mm</td>
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<tr>
<td>Latch position</td>
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<td></td>
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<tr>
<td>Item</td>
<td>Going up by driving motor</td>
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<tr>
<td><strong>Picture</strong></td>
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</tr>
<tr>
<td>Item</td>
<td>Initial position</td>
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<td>Latch position</td>
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<tr>
<td>Item</td>
<td>Going up by driving motor</td>
<td>Going up by driving motor</td>
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<td><img src="image.png" alt="Picture" /></td>
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<td>Latch position</td>
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Design of Sodium Component and Its Feature for Cooling System

Mitsuru Kamei

December 2004

International Cooperation & Technology Development Center
Tsuruga Headquarter Office, JNC
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</tbody>
</table>
Comparison of Thermal Expansion for FBR & LWR

Room Temperature

LWR 350°C

FBR 500°C

Ferrite Steel

Stainless Steel

Out: 鋼耐系機器設計と特性（応用講座） 平成 年 月 日 龟井 民
<table>
<thead>
<tr>
<th></th>
<th>Sodium Component</th>
<th>BWR or PWR</th>
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</thead>
<tbody>
<tr>
<td>Fluid</td>
<td>Liquid Metal Sodium Single Phase Fluid Magnetic Fluid</td>
<td>Water &amp; Steam Double Phase Fluid Non Magnetic Fluid</td>
</tr>
<tr>
<td>Density ( g / cm³ )</td>
<td>About 0.832</td>
<td>0.724</td>
</tr>
<tr>
<td>Specific Heat (J / kg °C)</td>
<td>0.86</td>
<td>0.66</td>
</tr>
<tr>
<td>Ratio of Heat Conductive (W / m°C)</td>
<td>66.83</td>
<td>0.5547</td>
</tr>
<tr>
<td>Viscosity coefficient (Pa · s)</td>
<td>2.364 × 10⁻⁴</td>
<td>9.03 × 10⁻⁵</td>
</tr>
<tr>
<td>Max. Temperature</td>
<td>&lt;600 °C</td>
<td>&lt;400 °C</td>
</tr>
<tr>
<td>Difference of inlet &amp; outlet for reactor</td>
<td>132 °C (Monju)</td>
<td>35 °C (PWR) 7 °C (BWR)</td>
</tr>
<tr>
<td>Max. Pressure</td>
<td>10 kgs / cm²</td>
<td>About 160 kgs / cm²</td>
</tr>
<tr>
<td>Chemical Activity</td>
<td>Very Active</td>
<td>Stable</td>
</tr>
<tr>
<td>Efficiency</td>
<td>About 41%</td>
<td>About 33%</td>
</tr>
</tbody>
</table>
# 1. Reactor vessel & Tank

<table>
<thead>
<tr>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Election of Material</strong></td>
<td>Compatibility with Sodium</td>
</tr>
<tr>
<td><strong>Free Space</strong></td>
<td>Volume Expansion of Sodium by $\Delta T$</td>
</tr>
<tr>
<td><strong>Support of Tank</strong></td>
<td>To be loose to escape Thermal Expansion</td>
</tr>
<tr>
<td><strong>Preheating</strong></td>
<td>To be upper side</td>
</tr>
<tr>
<td><strong>Nozzle</strong></td>
<td>- Thermal Shock</td>
</tr>
<tr>
<td></td>
<td>- Cover Gas has not into liquid metal</td>
</tr>
<tr>
<td></td>
<td>- Design Evaluation</td>
</tr>
<tr>
<td></td>
<td>- Position of nozzle</td>
</tr>
<tr>
<td></td>
<td>Gas Space $&gt;10%$</td>
</tr>
<tr>
<td></td>
<td>For Large Tank</td>
</tr>
<tr>
<td></td>
<td>Volume Expansion : exchanging to liquid from solid of Sodium</td>
</tr>
</tbody>
</table>
## Comparison of Loop Type & Tank Type

<table>
<thead>
<tr>
<th></th>
<th>Loop Type(1000MWe)</th>
<th>Tank Type(1000MWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diameter</td>
<td>&lt;10m (Reactor Vessel)</td>
<td>≥ 20m (Main Tank)</td>
</tr>
<tr>
<td>Diameter of Shield Plug</td>
<td>&lt;10m</td>
<td>≥ 20m</td>
</tr>
<tr>
<td>No. of Guard Vessel</td>
<td>3</td>
<td>1</td>
</tr>
<tr>
<td>Connection of Pipe</td>
<td>Core- IHX-Pump-Core</td>
<td>Pump-Core</td>
</tr>
<tr>
<td>Free Surface &amp; Gas System</td>
<td>Necessary of Over Flow System for Adjust of Sodium Level</td>
<td>1 Surface &amp; 1 Gas System</td>
</tr>
<tr>
<td>Impact of Container Vessel</td>
<td>High: Maintenance of IHX Diameter :Arrangement of Components</td>
<td>High: Maintenance of IHX Diameter :Diameter of Main Tank</td>
</tr>
<tr>
<td>Manufacture</td>
<td>Permit to manufacture at Factory for all Components</td>
<td>Necessitate to manufacture for Main Tank</td>
</tr>
<tr>
<td>Maintenance</td>
<td>Even</td>
<td>Even</td>
</tr>
<tr>
<td>Activation of 2nd Sodium</td>
<td>Not necessary neutron shield</td>
<td>Necessary of neutron shield</td>
</tr>
</tbody>
</table>
Loop Type & Tank Type (1)
Loop Type & Tank Type(2)

Tank type

Loop type
Image of welding work reactor vessel

MONJU Reactor vessel

Shell material (Forging roll)

Welding Machine

Rotation table

Large manipulator
Assemble of Reactor Vessel for MONJU

- Parts of 12 are manufactured by Forging roll method
- Welding is done by HOT-TIG method on vertical style
- After welding, manufacture & inspection are done
- Outlook after assemble
Primary Piping System of Tank Type FBR

Primary Piping system is pipe to connect with Core structure & Primary pomp. Bellows is applied for primary Piping System to escape heat expansion. Check valve is set up in Primary Pomp.

出典：高速増殖実証炉 第1巻 表
Internal Pressure Type Expansion Joint for large FBR plant

<table>
<thead>
<tr>
<th>Design Pressure</th>
<th>2 kg/cm²G</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design Temperature</td>
<td>550 °C</td>
</tr>
<tr>
<td>Angle of Displacement</td>
<td>± 1.25°</td>
</tr>
</tbody>
</table>

PNC SN9600 89-002
Protection method of Reactor Vessel on Near Sodium Level for Thermal Stress

Tank Type (super Penix)

Loop Type (MONJU)
2. Sodium Pump

2.1 Mechanical Pump

- Pump of FBR & LWR
- Hot & Cold Leg Pump on Cooling System
- Decision Factor of Sodium Pump Size
- Character of Impeller & Mechanical Seal
- Cover Gas Convection & it’s Provision
- Sodium Bearing & Request from Plant System
- Leak Flow Rate on Sodium Bearing & Change of Time of Leak Flow Rate
- Operation of Pump on Reactor Emergency
Primary Pump of MONJU

Primary Sodium Pump

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of pumps</td>
<td>3 (one per loop)</td>
</tr>
<tr>
<td>Capacity ... Main motor / pony motor</td>
<td>5100 / 600 tonnes per hour</td>
</tr>
<tr>
<td>Head</td>
<td>92 mNa</td>
</tr>
<tr>
<td>Max. operating pressure .... Outlet side / suction side</td>
<td>10 / 2 kg cm⁻² G</td>
</tr>
<tr>
<td>Max. operating temperature</td>
<td>420°C</td>
</tr>
<tr>
<td>Main material</td>
<td>SUS 304</td>
</tr>
<tr>
<td>Main motor ... Power / speed</td>
<td>2000 kW / 837 rpm</td>
</tr>
<tr>
<td>Pony motor ... Power / speed</td>
<td>22 kW / 167 rpm</td>
</tr>
</tbody>
</table>

出典：PNC SN9410 90-031 動力炉の実用化をめざして 109p
Primary & Secondary Pump of JOYO
Outline of MONJU Primary Pump
MONJU Primary Pump

Installation of inner assembly
<table>
<thead>
<tr>
<th>Design Request</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Q-H Curve</strong></td>
<td>Flow &amp; Head at Setting point</td>
</tr>
<tr>
<td><strong>Flow Coast Down</strong></td>
<td>Decay Heat Remove at scram</td>
</tr>
<tr>
<td><strong>Low Level &amp; Low Speed Operation</strong></td>
<td>Decay Heat Remove at Sodium Leak</td>
</tr>
<tr>
<td><strong>Operation of 3 &amp; 1 Loop System</strong></td>
<td>1 Loop operation for other loop failure</td>
</tr>
<tr>
<td><strong>Control of Pump Speed</strong></td>
<td>100% ~ 10%</td>
</tr>
<tr>
<td><strong>Seismic Design</strong></td>
<td>As Class</td>
</tr>
</tbody>
</table>

**Remark**:
- Sodium temp.: 200&390°C
- Relaxation for Thermal shock
- Long Pump Shaft Pony motor of Emergency Power Supply
- Design of Sodium Bearing
- Operation by Pony Motor
- To be continue at earthquake
Flow Diagram of Heat Transport System of Monju
## Consideration of Mechanical Pump

<table>
<thead>
<tr>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Material Election</strong></td>
<td>Hard face material for Sodium Bearing</td>
</tr>
<tr>
<td><strong>Cavitation Character</strong></td>
<td>To be confirm, In-Water Test</td>
</tr>
<tr>
<td><strong>Pump Casing</strong></td>
<td>To be set Break Plate for Cover Gas Convection</td>
</tr>
<tr>
<td><strong>Shaft Power</strong></td>
<td>To be consider, for water testing power</td>
</tr>
<tr>
<td><strong>Leak Flow from Sodium Bearing</strong></td>
<td>Sodium Leak Flow Rate &gt; Water Leak Flow Rate</td>
</tr>
<tr>
<td><strong>Between Shaft &amp; Inner Casing</strong></td>
<td>To be set gas flow for narrow gap</td>
</tr>
<tr>
<td><strong>Maintenance</strong></td>
<td>To be drain sodium of narrow gap</td>
</tr>
</tbody>
</table>

Consideration of Mechanical Pump

Remarks:

- **Material Election**
  - Compatible of Sodium
  - Hard face material for Sodium Bearing

- **Cavitation Character**
  - Nearly Water character
  - To be confirm, In-Water Test

- **Pump Casing**
  - Cover gas Convection in Pump Casing
  - To be set Break Plate for Cover Gas Convection

- **Shaft Power**
  - Operation temperature (400~400°C)
  - To be consider, for water testing power

- **Leak Flow from Sodium Bearing**
  - Leak flow rate may change with time passage
  - Sodium Leak Flow Rate > Water Leak Flow Rate

- **Between Shaft & Inner Casing**
  - Stick by Sodium Vapor
  - To be set gas flow for narrow gap

- **Maintenance**
  - Maintenance cask for pull out of inner casing is necessary
  - To be drain sodium of narrow gap
To keep natural circulation & Level of Reactor Core for sodium leak, Primary piping system is arranged high level position & Guard Vessels are equipped each component and Piping system.

To prevent back flow from non-failure system to Sodium leak system, Check valves are installed to inlet pipe of Reactor Vessel.
# R&D Time Schedule for Mechanical Pump

<table>
<thead>
<tr>
<th>Pump of R&amp;D Test</th>
<th>Design &amp; Manufacture</th>
<th>Stick of Shaft</th>
<th>Water Test</th>
<th>(5m³/min)</th>
<th>(57,000hr)</th>
</tr>
</thead>
<tbody>
<tr>
<td>JOYO 2nd Mock Up Pump</td>
<td>Design &amp; Manufacture</td>
<td>Stick of Shaft</td>
<td>Water Test</td>
<td>Improvement of sodium bearing</td>
<td>(21m³/min)</td>
</tr>
<tr>
<td>JOYO 1st Mock Up Pump</td>
<td>Design &amp; Manufacture</td>
<td>Sodium Test</td>
<td>Water Test</td>
<td>(21m³/min)</td>
<td>(32,000hr)</td>
</tr>
<tr>
<td>MONJU 1st Mock Up Pump</td>
<td>Design &amp; Manufacture</td>
<td>Stick of Bearing</td>
<td>Water Test</td>
<td>(H₂O 80m³/min)</td>
<td>(Na 21m³/min)</td>
</tr>
</tbody>
</table>
# R&D History of Mechanical Pomp

<table>
<thead>
<tr>
<th></th>
<th>No.</th>
<th>Flow rate m³/min</th>
<th>Head m</th>
<th>Speed r/min</th>
<th>Design temp. °C</th>
<th>Motor power kW</th>
<th>History</th>
</tr>
</thead>
<tbody>
<tr>
<td>Trial</td>
<td>1</td>
<td>4.0</td>
<td>2400</td>
<td>450</td>
<td>30</td>
<td>41</td>
<td></td>
</tr>
<tr>
<td>Trial</td>
<td>1</td>
<td>0.9</td>
<td>2950</td>
<td>500</td>
<td>37</td>
<td>44</td>
<td></td>
</tr>
<tr>
<td>Trial</td>
<td>1</td>
<td>1</td>
<td>2400</td>
<td>600</td>
<td>15</td>
<td>45</td>
<td></td>
</tr>
<tr>
<td>Trial</td>
<td>1</td>
<td>1</td>
<td>3000</td>
<td>450</td>
<td>22</td>
<td>45</td>
<td></td>
</tr>
<tr>
<td>Sodium Test Facility</td>
<td>1</td>
<td>5</td>
<td>5 (kg/cm²)</td>
<td>1395</td>
<td>450</td>
<td>105</td>
<td>44</td>
</tr>
<tr>
<td>Joyo 1st mock-up</td>
<td>1</td>
<td>2.1</td>
<td>7.0</td>
<td>930</td>
<td>450</td>
<td>330</td>
<td>47</td>
</tr>
<tr>
<td>Joyo 1st Pomp</td>
<td>2</td>
<td>2.1</td>
<td>7.0</td>
<td>930</td>
<td>450</td>
<td>330</td>
<td>50</td>
</tr>
<tr>
<td>Joyo 2nd mock-up</td>
<td>1</td>
<td>2.1</td>
<td>3.5</td>
<td>985</td>
<td>400</td>
<td>180</td>
<td>48</td>
</tr>
<tr>
<td>Joyo 2nd Pomp</td>
<td>2</td>
<td>2.1</td>
<td>3.5</td>
<td>985</td>
<td>400</td>
<td>180</td>
<td>50</td>
</tr>
<tr>
<td>Monju R&amp;D</td>
<td>1</td>
<td>H₂O 8.7</td>
<td>9.0</td>
<td>850</td>
<td>390</td>
<td>H₂O 1800</td>
<td>52</td>
</tr>
<tr>
<td>Monju 1st pomp</td>
<td>3</td>
<td>Na₂ 2.1</td>
<td>9.4</td>
<td>850</td>
<td>10 kg/cm²</td>
<td>2400</td>
<td>(H8)</td>
</tr>
<tr>
<td>Monju 2nd pomp</td>
<td>3</td>
<td>3.7 10⁶ kg/h</td>
<td>5.5</td>
<td>800</td>
<td>325</td>
<td>800</td>
<td>(H8)</td>
</tr>
</tbody>
</table>
## Comparison of Hot & Cold Leg Pump

<table>
<thead>
<tr>
<th></th>
<th>Hot Leg</th>
<th>Cold Leg</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Temperature of Pump</strong></td>
<td>500</td>
<td>&lt;400</td>
</tr>
<tr>
<td><strong>Head of Pump</strong></td>
<td>IHX + Core + Piping</td>
<td>Core + Piping</td>
</tr>
<tr>
<td><strong>In case of failure IHX</strong></td>
<td>May be leak to 2nd Sodium from 1st Sodium</td>
<td>May be leak to 1st Sodium from 2nd Sodium</td>
</tr>
<tr>
<td><strong>Thermal Shock of pump</strong></td>
<td>Large</td>
<td>Small</td>
</tr>
<tr>
<td><strong>For Length of Pump Shaft</strong></td>
<td>Similar: because of Impeller Level for Sodium Leak</td>
<td>Similar: because of Impeller Level for Sodium Leak</td>
</tr>
<tr>
<td><strong>For Transformation of Pump Casing</strong></td>
<td>Disadvantage: Convection of Cover Gas is Great</td>
<td>Advantage: Convection of Cover Gas is Small</td>
</tr>
<tr>
<td><strong>For Size of IHX</strong></td>
<td>May be permit Compact design for IHX</td>
<td>May be not permit Compact design for IHX</td>
</tr>
<tr>
<td><strong>For Pipe Size</strong></td>
<td>May be permit Compact design of Pipe Size between Pump and IHX</td>
<td>May be not permit Compact design of Pipe Size between IHX and Pump</td>
</tr>
</tbody>
</table>
Cold leg pump & Hot leg pump
### Decision Factor For Sodium Pump Scale

<table>
<thead>
<tr>
<th>Diameter</th>
<th>Decision Factor Shaft Length</th>
</tr>
</thead>
<tbody>
<tr>
<td>(N = N_s \times (H^{3/4}/Q^{1/2}))</td>
<td>Shield for Heat &amp; Radiation</td>
</tr>
<tr>
<td>(N) ; pump speed</td>
<td>Operation Low Revel</td>
</tr>
<tr>
<td>(N_s) ; specific pump speed</td>
<td>(Installation Arrangement)</td>
</tr>
<tr>
<td>(H) ; head/impeller</td>
<td></td>
</tr>
<tr>
<td>(Q) ; flow rate</td>
<td></td>
</tr>
<tr>
<td>To be limited by NPSH for pump speed</td>
<td></td>
</tr>
</tbody>
</table>

**Decision Factor of Impeller Diameter:**

- Head & Flow rate (for Plant request)
- Type of Impeller
- NPSH (Vapor Pressure · installation Arrangement)

**Pump Power**

\[ L = 0.163rHQ/y \]

<table>
<thead>
<tr>
<th>Variable</th>
<th>Unit</th>
</tr>
</thead>
<tbody>
<tr>
<td>(L)</td>
<td>Pump Power (kw)</td>
</tr>
<tr>
<td>(r)</td>
<td>Specific Gravity (kg/l)</td>
</tr>
<tr>
<td>(Q)</td>
<td>Flow Rate (m³/min.)</td>
</tr>
<tr>
<td>(H)</td>
<td>Head (m)</td>
</tr>
<tr>
<td>(y)</td>
<td>Pump Efficiency (%)</td>
</tr>
</tbody>
</table>
Types for Impeller of Mechanical Sodium Pomp

- Double Suction Impeller
- 2 Stage Single Suction Impeller
- Inducer Impeller
### Design & R&D for Mechanical Pump

<table>
<thead>
<tr>
<th></th>
<th>Design Method</th>
<th>R &amp; D</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Bearing</td>
<td>Hydrostatic Bearing by Sodium lubricant</td>
<td>By in sodium testing of small Pump</td>
<td></td>
</tr>
<tr>
<td>Prevent method of Contact with Air</td>
<td>Axial Pump with free surface &amp; Cover Gas</td>
<td>By in sodium testing of Mock Up Pump</td>
<td>Seal Gas for Shaft</td>
</tr>
<tr>
<td>Cavitation (NPSH) Performance</td>
<td>From data of water Pump</td>
<td>Confirm of water &amp; sodium test by Mock Up Pump</td>
<td></td>
</tr>
<tr>
<td>Slow speed operation by Pony Motor by DC current</td>
<td>From experience of water Pump</td>
<td>Confirm of water &amp; sodium test by Mock Up Pump</td>
<td>Long term operation</td>
</tr>
<tr>
<td>Long Pump Shaft</td>
<td>Form Stabilize for Shaft</td>
<td>Confirm of sodium test by Mock Up Pump</td>
<td></td>
</tr>
</tbody>
</table>
Primary & Secondary Pump of JOYO

- suction nozzle
- derive nozzle
- impeller
- sodium bearing
- inner casing
- outer casing
- shaft

- thermal shield plate
- over nozzle
- cover gas nozzle
- level meter
- radiation shield
- mechanical seal
Hydraulic Characteristics Comparison of Sodium & water
(“JOYO” Mock - Up Pump)

Pump Efficiency:
Sodium > Water
Leak Flow Rate for Bearing
Sodium > Water
Sodium Pump Test Loop
Estimation of Coast Down Flow

\[
M = -I \left( \frac{dw}{dt} \right) = -\left( \frac{(GD^2)}{4g} \right) \left( \frac{dw}{dt} \right)
\]

- \( M \): moving torque
- \( I \): mass inertia moment
- \( w \): angler velocity

Then

\[
\frac{\sqrt{2}}{\sqrt{N/60}} M = \left( \frac{GD^2}{4g} \right) \sqrt{\left( \frac{2\sqrt{N/60}}{dN/dt} \right)}
\]

As

\[
m = \frac{M}{M_o}
\]
\[
n = \frac{N}{N_o}
\]
\[
m = -\left( \frac{GD^2}{375} \right) \left( \frac{N_o}{M_o} \right) \left( \frac{dn}{dt} \right)
\]

Then,

\[
K = \left( \frac{375}{GD^2} \right) \left( \frac{M_o}{N_o} \right)
\]
\[
m = -\left( \frac{1}{K} \right) \left( \frac{dn}{dt} \right)
\]

- \( M_o \): inertia at motor switch kg \cdot m
- \( N_o \): rotation speed at motor switch rpm

Solving a differential equation

\[
n \left( rotation \ speed \right) = \frac{1}{\left( Kt + 10 \right)}
\]
Test of flow coast down for JOYU mock up Pump

**Initial Condition**
- Na flow: 21 m³/min
- Na temperature: 370 °C
- Motor speed: 930 rpm
- Motor output: 540 kW

**Aimed Time Constant**
\[ 1/e = 12 \text{ sec} \]

Reactor cooling pump is designed as cooling flow goes down slowly for eliminating remained core heat at an unexpected control rod scrumming.
Cavitation Test of Sodium Mechanical Pump

Pump Suction Head NPSH : Hsn [ mNa ]

\[ Hsv = 10^4 \left( \frac{P_2}{\text{[]}} \right) + \text{hz}_2 - \text{h} \text{ls} \left( \frac{Vs^2}{\text{[g^2]}} \right) - 10^4 \left( \frac{Pvp}{\text{[]}} \right) \]

- \( P_2 \) : suction head (kg/cm\(^2\)Na)
- \( \text{[]} \) : heat density of sodium(kg/m\(^2\))
- \( \text{hz}_2 \) : revel of suction pressure (m)
- \( \text{h} \text{ls} \) : pressure loss from pressure to Pump suction (m)
- \( Vs \) : Pump suction flow speed(m/s)
- \( Pvp \) : boiling pressure of sodium (kg/cm\(^2\))

Cavitation start point :
3% suction head decrease
Keep of Pump operation condition:
Pump speed & flow rate

出典：SN941 76 - 34 ポンプのキャビテーション試験 20p
Sodium Pump Test Loop
Structure of prevention plate

Prevention plate for gas convection:
In cover gas space of mechanical Pump, Natural convection happen to gap between outer casing & inner casing. As the result, Pump casing will be transform.
Temp. Distribution of Pump Casing before & after improvement

Before Improvement
- T > 100

After Improvement
- T < 20
MONJU Primary Pomp
MONJU Secondary Main Pomp

Shaft

Inner Casing

Outer Casing

Impeller
Sodium Bearing from Plant System Request

Pressure Drop on 3 Loop Operation

Pressure Drop on 1 Loop Operation

Pump Head H

Q - H Carve

Flow Rate Q

Pressure Drop on 3 Loop Operation > 1 Loop Operation
Outline of Sodium Bearing

- Bearing housing
- Static bearing
- Shaft
- Outer casing
- Deriver
- Impeller
- Suction
- Diffuser
Fig. 3.22 Hallam Nuclear Power Facility secondary pump after removal.
Change of Over Flow Rate

-JOYO MOCK UP POMP-
### Countermeasure against Lubricant material of sodium bearing

<table>
<thead>
<tr>
<th>Type</th>
<th>To suction</th>
<th>To suction</th>
<th>Over flow column</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Outline of concept</strong></td>
<td><img src="image1" alt="Diagram" /></td>
<td><img src="image2" alt="Diagram" /></td>
<td><img src="image3" alt="Diagram" /></td>
</tr>
<tr>
<td>Example plant</td>
<td>CDS</td>
<td>JOYO Monju</td>
<td></td>
</tr>
<tr>
<td>Regulation of Pomp Level</td>
<td>Possible</td>
<td>Possible to stand bypass side</td>
<td>Possible to keep normal revel</td>
</tr>
<tr>
<td>Characteristic</td>
<td>Very Compact</td>
<td>Compact</td>
<td>Easy operation</td>
</tr>
</tbody>
</table>

Note: SN941 74-68
JOYO Mock Up Pump
Low Speed Operation & Start Up Operation for Pony Motor

Performance of Low Speed Operation
On Power Failure:
Pump will be operated for Low Speed by Pony Motor to remove Core Decay Heat.
This Figure show Pump Character of 10% Flow rate of full Speed flow rate in Water & Sodium.
Pump Speed is 80 rpm

Shift Test of Pony Motor from Main Motor
Pony Pump will start at about 10% flow rate, when Main Pump has trouble or Power loss.

Test result of sodium pump

Test result of pony motor operation

Condition
Sodium temp: 370 °C
Cover gas pressure: 100mmAq
Primary & Secondary Pump of JOYO

- suction nozzle
- derive nozzle
- impeller
- sodium bearing
- inner casing
- outer casing
- shaft

- thermal shield plate
- over nozzle
- cover gas nozzle
- level meter
- radiation shield
- mechanical seal
3. IHX

- Type of IHX
- Outline of MONJU IHX
- Design Consideration of IHX
- R&D for IHX
- Mock Up Test of JOYO IHX
- Heat Transfer Performance of IHX
Comparison of Structure for IHX

Vertical Type & Horizontal Type
(VT) (HT)
For natural circulation; HT > VT
For maintenance; VT > HT

Inside of 1st Sodium & 2nd Sodium
For shock of Sodium Water Reaction; 1st > 2nd

Straight Tube & Helical Coil Tube
(ST) (HCT)
Compact; ST > HCT
T of Temperature on Tube & Shell;
HCT > ST
3. IHX of MONJU

- Upper Plenum
- Support
- Cover, Gas Shell
- Outer Shroud
- 1st Sodium Inlet
- Down Comer
- Upper Tube Sheet
- Heat Transfer Tube
- Bellows
- Lower Tube Sheet
- 2nd Sodium Inlet
- 2nd Sodium Outlet
- Lower Plenum
- Drain
- 1st Sodium Outlet

Baffle Plate
Down Comer

Intermediate Heat Exchanger (IHX)

<table>
<thead>
<tr>
<th>Type</th>
<th>Vertical parallel flow with no sodium surface</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of units</td>
<td>3 (one per loop)</td>
</tr>
<tr>
<td>Capacity</td>
<td>238 MW</td>
</tr>
<tr>
<td>Coolant flow rates</td>
<td>Primary/secondary</td>
</tr>
<tr>
<td></td>
<td>5,100 / 3,700 tonnes per hour</td>
</tr>
<tr>
<td>Operating temperatures</td>
<td></td>
</tr>
<tr>
<td>Primary inlet</td>
<td>522°C</td>
</tr>
<tr>
<td>Primary outlet</td>
<td>537°C</td>
</tr>
<tr>
<td>Secondary inlet</td>
<td>335°C</td>
</tr>
<tr>
<td>Secondary outlet</td>
<td>565°C</td>
</tr>
<tr>
<td>Number of tubes</td>
<td>3,394</td>
</tr>
<tr>
<td>Outer diameter of tube</td>
<td>21.7 mm</td>
</tr>
<tr>
<td>Thickness of tube</td>
<td>1.2 mm</td>
</tr>
<tr>
<td>Outer diameter of cylinder</td>
<td>31.9 mm</td>
</tr>
<tr>
<td>Height</td>
<td>121.1 m</td>
</tr>
<tr>
<td>Main material (Cylinder &amp; tubes)</td>
<td>SUS 304</td>
</tr>
</tbody>
</table>

出典: JNCホームページ もんじゅ建設サイト もんじゅ建設のあゆみ
IHX of MONJU
JOYO IHX Dismantle for MK- 𓀄
IHX of JOYO MK-3

1st Outlet tube

2nd Outlet tube

2nd Inlet tube

2nd drain tube

1st Inlet tube

Inlet window 1st side flow

Heat transfer tube

Outlet Window for 1st Flow

Bypass Seal bellows

IHX of JOYO MK-  ⨰
### Consideration of IHX Design

<table>
<thead>
<tr>
<th></th>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Material</strong></td>
<td>Compatible of Sodium</td>
<td>Mass Transfer of Fuel Pin</td>
</tr>
<tr>
<td><strong>Heat Transfer</strong></td>
<td>Heat Transfer Coefficient of Sodium &amp; Tube</td>
<td>Low Flow Rate: Design for Laminar Flow</td>
</tr>
<tr>
<td><strong>Thermal Shock</strong></td>
<td>Tube &amp; Tube Sheet</td>
<td>At Reactor Scram</td>
</tr>
<tr>
<td><strong>T of Tube &amp; Shell</strong></td>
<td>Expansion Joint for Shell</td>
<td>In Case of Straight Tube Type</td>
</tr>
<tr>
<td><strong>Distribution of Flow</strong></td>
<td>Flat Flow Rate For 1st &amp; 2nd Cooling Material</td>
<td>At low flow rate</td>
</tr>
<tr>
<td><strong>Pressure Drop</strong></td>
<td>Low Pressure Loss</td>
<td>In case of Cold Leg Pump</td>
</tr>
<tr>
<td><strong>Activation of 2nd Sodium</strong></td>
<td>To be prevent activation for 2nd Sodium</td>
<td>In case of nearness for Reactor Core</td>
</tr>
</tbody>
</table>
Efficiency of Heat Exchange

Parallel Flow

Counter Flow

Efficiency: Φ

Parallel Flow

\[ \phi = \frac{1 - \exp\left(-\frac{(1+W)K_F}{W}ight)}{1+W} \]

Counter Flow

\[ \phi = \frac{1 - \exp\left(-\frac{(1-W)K_F}{W'}\right)}{1-W} \]

Apply for JOYO & MONJU
Overall heat exchange
Baffle plate are set on
Outer side for heat transfer tube

symbol
C : °C temperature
W : kcal/h Heat Flux
G : kg/h Flow Rate
C : Kcal/kg Heat Specific
F : m² Surface of Heat Transfer
Q : kcal/h Capacity of Heat Exchange
K : coefficient of overall heat transmission
Φ : efficiency
## Design and R & D of IHX

<table>
<thead>
<tr>
<th></th>
<th>Design Method</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flow Distribution &amp; Pressure Drop</td>
<td>Confirm: water test by partial or sector model</td>
<td>Water flow test</td>
</tr>
<tr>
<td>Thermal Shock for Tube Plate</td>
<td>Confirm: sodium test for thermal shock partial model</td>
<td>Evaluation of Structure Material for Tube &amp; Tube Sheet</td>
</tr>
<tr>
<td>Thermal Expansion Difference between Tube &amp; Shell</td>
<td>Confirm: Fatigue test of Expansion Joint by Thermal Cycle</td>
<td></td>
</tr>
<tr>
<td>Material Corrosion</td>
<td>Confirm: Mass Transfer Test by small material</td>
<td>Simulation of temperature, material &amp; Purification of Sodium</td>
</tr>
</tbody>
</table>
**IHX performance test –1/60 model for JOYO**

Precede test:
construction & operation of JOYO

Test result

Heat capacity: satisfy

Thermal cycle test: no trouble

Material test:
confirm of design life of JOYO
Fig. 2-2  Flow Diagram of the Test Loop
Over all heat transfer coefficient of IHX
JOYO 1/60 model

Design rating point

Test result of K

Shell side heat transfer coefficient

Tube side heat transfer coefficient

Over all heat transfer coefficient

Out典:SN-941 75-80 常陽」ナトリウム冷却系耐久試験装置 25p
Tube-to-tube-sheet attachments

For MONJU IHX
- Expand of tube
- Filet weld
- Tube: to be expand
- Welding: filet weld

For MONJU SG
- Forged or coined spigot
- Tube sheet; forged
- Welding: butt method

Before heat treatment
3. SG & its System

- Type of SG
- SG Type for Point of View From Separation of Steam
- SG of MONJU
- R&D of SG and relative equipment with SG
- Phenomenon on Breakage of Heat Transfer Tube
- Provision on Breakage of Heat Transfer Tube
Flow Diagram of Heat Transport System of Monju
### Consideration of Design for SG

<table>
<thead>
<tr>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
</table>
| **Material** | Water side: SCC  
Sodium Side : compatibility with Sodium | Impurity of Water & Sodium                        |
| **Support**  | Fluid Vibration of Water & Steam               | Wear of tube                                     |
| **Character of heat transfer**  | Water side: Boiling  
Sodium : Na data | Specially : low speed flow rate                   |
| **Stable of Operation**  | on Low load Operation by change from water to steam | On start up operation                             |
| **Failure of Heat Transfer Tube**  | Detector : H₂ gas detector  
Reaction Product system  
Rapture Disk | Sodium-Water reaction  
To be careful for Continue to failure of heat transfer tube |
<table>
<thead>
<tr>
<th>Kinds of SG for Tube Type</th>
<th>Straight tube</th>
<th>Hockey Stick tube</th>
<th>Helical Coil tube</th>
</tr>
</thead>
<tbody>
<tr>
<td>Good Point</td>
<td>Compact</td>
<td>Compact</td>
<td>Escape of Tube Expansion</td>
</tr>
<tr>
<td></td>
<td>Simple structure</td>
<td>Escape of Tube Expansion</td>
<td>Number of tube : little</td>
</tr>
<tr>
<td></td>
<td>Easy for ISI</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Weak Point</td>
<td>Number of tube : many</td>
<td>Number of tube : many</td>
<td>Length of Tube : long</td>
</tr>
<tr>
<td></td>
<td>Thermal shock: tube sheet</td>
<td>Thermal shock: Tube sheet</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Difference of Heat Expansion : between Tube &amp; Shell</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Example of Plant Design</td>
<td>BN-600</td>
<td>CRBR</td>
<td>Monju Super-Phenix 2</td>
</tr>
<tr>
<td>Other: Double Wall Tube (EBR-2), U-type (PFR) etc.,</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

出典：冷却系機器設計と特性（応用講座） 発足 15年 3月 26日 鳥井 潤
Kinds of Steam Generator

- Helical Coil Type
- Straight Tube Type
- Hockey Stick Type
EVAPORATOR

- Feed water inlet plenum
- Feed water piping
- Feed water tube plate
- Na/water reaction products release nozzle
- Downcomer

- Steam outlet piping
- Steam outlet water plenum
- Steam outlet tube plate
- Sodium inlet piping
- Upper plate

- Heat transfer tube of helical coil type

SUPERHEATER

- Steam outlet water plenum
- Steam inlet nozzle
- Steam outlet nozzle
- Steam inlet water plenum
- Na/water reaction products release nozzle
- Upper cylinder
- Shroud
- Sodium inlet nozzle
- Flange
- Skirt
- Lower shell

- Assembling of heat transfer tubes of evaporator

Assembling of heat transfer tubes of superheater
## R & D for SG & Relation System R (1)

<table>
<thead>
<tr>
<th>Material data</th>
<th>Solution</th>
<th>R&amp;D</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Literature: foreign reactor Material test</td>
<td>Material test loop</td>
<td>Water &amp; Sodium</td>
</tr>
<tr>
<td>Difference of Expansion between tube &amp; shell</td>
<td>Helical coil tube type</td>
<td>1MWSG &amp; 50MW SG test facility</td>
<td></td>
</tr>
<tr>
<td>Performance of heat transfer in sodium &amp; water</td>
<td>Literature: foreign reactor Heat transfer test</td>
<td>1MWSG &amp; 50MW SG test facility</td>
<td></td>
</tr>
<tr>
<td>Fluid instability behavior of water side</td>
<td>Literature: foreign reactor Fluid stable test</td>
<td>1MWSG &amp; 50MW SG test facility</td>
<td>Start up of SG</td>
</tr>
<tr>
<td>Sodium leak detector</td>
<td>Development of H₂ gas leak detector in sodium</td>
<td>1MWSG &amp; 50MW SG test facility</td>
<td>PNC type</td>
</tr>
</tbody>
</table>

出典：冷却系機器設計と特性 (応用講座 20), 平成15年10月10日 龜井 満
<table>
<thead>
<tr>
<th>Behavior of Na-H₂O reaction</th>
<th>Solution</th>
<th>R &amp; D</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Confirm: Large, Small leak test by sodium testing</td>
<td>SWAT- 1, SWAT- 2, SWAT- 3</td>
<td>Test condition: Material, temperature, pressure, etc.,</td>
</tr>
<tr>
<td>Behavior of escalation for failure of tube</td>
<td>Confirm: tube model test in sodium &amp; evaluation of analysis</td>
<td>SWAT- 1, SWAT- 3</td>
<td></td>
</tr>
<tr>
<td>Rapture Disk</td>
<td>Confirm: Mock Up Test in sodium test</td>
<td>SWAT- 1, SWAT- 3, 50MW Test Facility</td>
<td>Endurance test</td>
</tr>
<tr>
<td>Treatment System for Reaction Production</td>
<td>Confirm of operation method for hydrogen gas &amp; NaOH</td>
<td>SWAT- 1, SWAT- 3</td>
<td></td>
</tr>
</tbody>
</table>
R&D Time Schedule for MONJU SG

NO.2 Evaporator and Superheater

Operation Time

<table>
<thead>
<tr>
<th>No.</th>
<th>Evaporator</th>
<th>Superheater</th>
<th>End Time</th>
</tr>
</thead>
<tbody>
<tr>
<td>No.1</td>
<td>3,476 hr.</td>
<td>1,076 hr.</td>
<td></td>
</tr>
<tr>
<td>No.2</td>
<td>11,509 hr.</td>
<td>3,755 hr.</td>
<td></td>
</tr>
<tr>
<td>Na Loop</td>
<td>Primary Loop</td>
<td>28,533 hr.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Second Loop</td>
<td>26,242 hr.</td>
<td></td>
</tr>
</tbody>
</table>

(As of end of December, 1981)
Fig. 1 Flow Sheet of 1 MWt Steam Generator Test Loop
50MW Steam Generator Test Facility
-model of MONJU cooling system-
R&D

1959-1972
Static & Dynamic Character Test by Using 50M SG Test Facility

1952-1967
Performance Test & Thermal Shock Test By using 1MW SG Test Facility

Behavior test of water leak on 50MW test facility

Test Facility(1MW)

Test Facility(50MW)

MONJU
## Failure Size of Leak Heat Transfer Tube

<table>
<thead>
<tr>
<th>Size of Hole</th>
<th>Main Behavior</th>
<th>Leak Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Very Small Hole</td>
<td>Self -wastage</td>
<td>0.1g/sec</td>
</tr>
<tr>
<td>Small Hole</td>
<td>Target -wastage</td>
<td>10g/sec</td>
</tr>
<tr>
<td>Middle Hole</td>
<td>Multi -wastage</td>
<td>2kg/sec</td>
</tr>
<tr>
<td>Large Hole</td>
<td>Phenomenon of heat flow</td>
<td></td>
</tr>
</tbody>
</table>

出典：PNC SN0410 - 90 - 031 動力炉の実用化をめざして
Hydrogen Detector in sodium

PNC type hydrogen detector

Permeable Rate & Partial Pressure in Sodium for H₂ gas

Ni Membrane Temp.: 500 °C

- OSH Na (50MW-MK IV)
- ΔE V Na (50MW-MK III N, 3)
- OSH Na (50MW-MK II)
- CT Na (No.1 PNC type)

H₂ Partial Pressure (Torr)

Permeable Rate of H₂
Detection for failure on heat transfer tube of SG

Signal of \(H_2\) Detector

Evaluation on Normal Operation

Outout: JNCホームページ もんじゅ訴訟 電気学会主催 蒸気発生器破損事故
# Detection method for tube failure

<table>
<thead>
<tr>
<th>Method</th>
<th>No. of Detector</th>
<th>Size of detection</th>
<th>Detection Time</th>
</tr>
</thead>
<tbody>
<tr>
<td>H2 Detector for Tube Failure</td>
<td>5 / loop</td>
<td>Very small-small</td>
<td>Minute Order</td>
</tr>
<tr>
<td>Detector for Cover Gas Pressure</td>
<td>2 / loop</td>
<td>Middle -Large</td>
<td>20 sec. – 1 sec.</td>
</tr>
<tr>
<td>Rapture Disk</td>
<td>2 / loop</td>
<td>Middle – large</td>
<td>20 sec. – 1 sec</td>
</tr>
</tbody>
</table>
Treatment in case of Na- H₂O reaction

Leak of water

In case of Large leak

H₂ meter

In sodium

Pressure Gage

In cover Gas

Rapture Disk

In case of pressure increase, By Difference of pressure

NO. of H₂ detector: 5 (each)

Reactor

CRD

Scrum

Cooling system

SG

Valve close

Steam release

Valve open

N₂ gas

To be relieve effect for IHX

To be reduce of Reaction (Na-H₂O)

Other loop

To remove Decay heat

(out)
Treatment equipment for sodium – water reaction on SG

Evaporator

Super Heater

Rupture Disk

Reaction Product Tank

Atmosphere

Ignition

To air open valve

Rupture Disk
Wastage in Steam Generator by Na- H₂O

tube

Water/steam

Pin Hole

sodium

Wastage

Water/steam

sodium

Bubble of Steam

出典：PNC SN0410-90-031 動力炉の実用化をめざして
Rapture Disk

Before

After

Rapture Disk

Structure of rapture disk

FERMI reactor

Reaction solution tank side

Gasket

SG side

Flange

Rapture Detector

Holder

Vacuum support

Disk

Slit Disk

Vacuum Support

Slit disk

disk
Procedure construction of SG

Tube sheet etc.,

Material

Assemble

Prepare for weld

Weld for construction

Bending

Weld for construction

Shell etc.,

Material

Assemble

Prepare for weld

Bending

Assemble

Weld for construction

Nozzle etc.,

Material

Prepare for assemble

Assemble

Bottom plate etc.,

Material

Assemble

Weld for construction

Weld for construction

Construction

To Plant site

Installation
Assemble of Evaporator for MONJU

- **Bending of Helical coil tube**
  - Long of Helical Tube: 86m /tube
  - Welding Number of tube: 4/tube

- **Assemble of tube & tube sheet**

- **Insert to shell**

- **Inspection test on pressure**

- **To site**
Fabrication performance of Steam Generator

- Automatic Helical Coil Bender
- Inner Bore Butt Welding for Tube-to-Tube sheet welding
- Butt structure for tube-to-tube sheet
- Long size heat transfer tube (EV:21.5m, SH:32m) to reduce number of welding
- Micro-focus X-ray inspection for heat transfer tube weld
<table>
<thead>
<tr>
<th>Model size of R &amp; D for component (1)</th>
<th>JOYO</th>
<th>MONJU</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor vessel</td>
<td>1/1</td>
<td>Partial model</td>
<td></td>
</tr>
<tr>
<td>Core Structure</td>
<td>1/1</td>
<td>1/2</td>
<td>In-water test</td>
</tr>
<tr>
<td>Upper Core Structure</td>
<td>1/1</td>
<td>1/6</td>
<td></td>
</tr>
<tr>
<td>Shield Plug</td>
<td>1/1</td>
<td>1/2.5</td>
<td>Thermal Shield test</td>
</tr>
<tr>
<td>Control Rod Drive Mechanism</td>
<td>1/1</td>
<td>1/1</td>
<td></td>
</tr>
<tr>
<td>Fuel Exchange Machine (in-Reactor)</td>
<td>1/1</td>
<td>1/1</td>
<td></td>
</tr>
<tr>
<td>Fuel Handling Machine</td>
<td>1/1</td>
<td>1/1</td>
<td>In-water&amp;Sodium</td>
</tr>
<tr>
<td>Rotating Plug</td>
<td>1/1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Ex-vessel storage tank for spent fuel</td>
<td></td>
<td>Partial model</td>
<td></td>
</tr>
</tbody>
</table>
REACTOR SYSTEMS
The reactor vessel has three inlet nozzles and three outlet nozzles at the lower and upper part of the reactor vessel respectively.
The closure head consists of a stationary plug and a rotating plug. An upper core structure and a fuel handling machine are installed through the rotating plug.
Two shut down systems, main and back up system, are provided to assure reactor safety.
A guard vessel is provided to retain any leaked sodium and assure reactor core coolability in the event of an accident.
Reactor Vessel & Structure of JOYO
Mock-Up Test Model for JOYO Reactor Vessel & Structure System
Core Internal Structure of MOMJU
Shield Plug of MONJU
Behavior test model of Shield Plug for MONJU
CRDM
Consepts of SASS

- Curie Point EM Type
  - Outer Type
  - Inner Type
- Bimetal Type
- Switch Type
  - Thermal Ion Type
  - Reed Switch Type
  - Temp. Sensitive Transformer Type
  - Temp. Sensitive Material + Gravitational Sw
- Hydraulic Pressure Type
  - Tantalum Ball + TSEM
  - Other Float Type
Self-Actuated Shutdown System (Example)
- Curie-Point Electromagnet Type
Fig. 10  Holding force of the proposed TSEM as a function of the temperature.
The upper core structure is a stainless steel-made structure suspended from the rotating plug to hold and guide a control rod drive mechanism and the containment of thermocouples for monitoring the sodium temperature at the outlet of fuel assemblies.

This structure consists of upper housing, upper plate, shield part, connecting cylinder, upper and lower thermal shield plate, flow controller and various kinds of guide tubes.

**SPECIFICATION OF THE UPPER CORE STRUCTURE**

<table>
<thead>
<tr>
<th>Main dimensions</th>
<th>Outer diameter</th>
<th>Height</th>
</tr>
</thead>
<tbody>
<tr>
<td>Shield part</td>
<td>2.6 m</td>
<td>2.7 m</td>
</tr>
<tr>
<td>Connecting cylinder</td>
<td>2.1 m</td>
<td>5.1 m</td>
</tr>
<tr>
<td>Flow controller</td>
<td>Max.diameter</td>
<td>2.1 m</td>
</tr>
<tr>
<td>Main material</td>
<td>SUS 304</td>
<td></td>
</tr>
</tbody>
</table>
### SPECIFICATION OF THE FUEL HANDLING MACHINE

<table>
<thead>
<tr>
<th>Type</th>
<th>Fixed arm type</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number</td>
<td>1</td>
</tr>
<tr>
<td>Capacity</td>
<td>1 (core element)</td>
</tr>
<tr>
<td>Rotating radius</td>
<td>1.67 m</td>
</tr>
<tr>
<td>Main part</td>
<td></td>
</tr>
<tr>
<td>Length</td>
<td>22 m</td>
</tr>
<tr>
<td>Vertical stroke</td>
<td>4.6 m</td>
</tr>
<tr>
<td>Main material</td>
<td>SUS 304</td>
</tr>
</tbody>
</table>

△ Fuel handling machine inside the reactor vessel
Fuel Handling Machine of MONJU

Fuel exchange machine

Car

Cooling equipment

Moving mechanism
For Gripper

Gripper
Coffin
Pot for spent fuel
Block
Door valve

Fuel exchange machine
JOYO Fuel Handling Machine
JOYO Fuel Transfer Machine

Cask car

Fuel Exchange machine
Floor gate valve

Drive mechanism

Upper bearing

Guide tubes

Guide tube plug

Shield plug

Drive shaft

Rotating rack

In-sodium bearing

Fuel transfer pot

Cooling coil

Fuel storage tank

Outer tank

Installation of the rotating rack

Mounting of the drive shaft to the fuel storage tank

動燃パンフレット
Model size of R & D for component (2)  
- JOYO, MONJU -

<table>
<thead>
<tr>
<th></th>
<th>JOYO</th>
<th>MONJU</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary Pump</td>
<td>1/1</td>
<td>1/1*1</td>
<td>*1: flow rate = 1/4</td>
</tr>
<tr>
<td>Secondary Pump</td>
<td>1/1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>SG &amp; system</td>
<td>------</td>
<td>1/5</td>
<td></td>
</tr>
<tr>
<td>IHX</td>
<td>1/5</td>
<td>1/5</td>
<td></td>
</tr>
<tr>
<td>Pipe</td>
<td>1/1</td>
<td>1/1*2</td>
<td>*2: in-air test</td>
</tr>
<tr>
<td>Check valve</td>
<td>1/1</td>
<td>1/5*3</td>
<td>*3: in-water test</td>
</tr>
<tr>
<td>Air cooler</td>
<td>1/50</td>
<td>1/5*4</td>
<td>*4: Natural circulation test</td>
</tr>
<tr>
<td>EMP (auxiliary Pump)</td>
<td>1/1</td>
<td>1/1*5</td>
<td>*5: design change of MONJU</td>
</tr>
</tbody>
</table>

Remark:
- JOYO, MONJU
- R & D: Research & Development
- IHX: Inherent Heat Exchanger
- SG: Steam Generator
- System
- Primary Pump: Main coolant pump
- Secondary Pump: Auxiliary coolant pump
- Check valve: Safety valve
- Air cooler: Air-cooled heat exchanger
- EMP: Emergency Pump
### Comparison of MONJU Primary Pump & Proto Type Pump

<table>
<thead>
<tr>
<th></th>
<th>Monju Pump</th>
<th>Prototype Pump</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Flow Rate</strong></td>
<td>99.6 m³/min</td>
<td>87.6 m³/min</td>
<td>In-Water</td>
</tr>
<tr>
<td></td>
<td></td>
<td>21.0 m³/min</td>
<td>In-Sodium</td>
</tr>
<tr>
<td><strong>Pump Head</strong></td>
<td>94 mNa</td>
<td>90 mNa</td>
<td></td>
</tr>
<tr>
<td><strong>Operating Temp.</strong></td>
<td>397 °C</td>
<td>390 °C</td>
<td></td>
</tr>
<tr>
<td><strong>Pump Speed</strong></td>
<td>837 rpm</td>
<td>850 rpm</td>
<td></td>
</tr>
<tr>
<td><strong>Power</strong></td>
<td>1715 kW</td>
<td>1780 kW</td>
<td>In-Water</td>
</tr>
<tr>
<td></td>
<td></td>
<td>550 kW</td>
<td>In-Sodium</td>
</tr>
<tr>
<td><strong>Shaft Length</strong></td>
<td>~8 mH</td>
<td>~8.5 mH</td>
<td></td>
</tr>
<tr>
<td><strong>Speed Control Method</strong></td>
<td><strong>MFG</strong></td>
<td>Thyristor</td>
<td></td>
</tr>
<tr>
<td><strong>Pipe Size</strong></td>
<td>32 inch/24 inch</td>
<td>12-inch</td>
<td></td>
</tr>
<tr>
<td><strong>Preheating Method</strong></td>
<td>Electric Heater</td>
<td>Hot N₂ Gas</td>
<td></td>
</tr>
<tr>
<td><strong>Overflow Nozzle</strong></td>
<td>8-inch</td>
<td>8-inch</td>
<td></td>
</tr>
</tbody>
</table>
Outlet Dumper
Outlet Duct
Inlet Dumper
Sodium outlet pipe
Sodium inlet tube
Inlet Duct

Sodium to Air Cooler of JOYO MK-
Natural Circulation Test of JOYO

Test Result of Natural Circulation Test of JOYU
This picture shows test result of core temperature & main flow rate, when power supply of main pomp was loosed.
<table>
<thead>
<tr>
<th>Instrument</th>
<th>JOYO</th>
<th>MONJU</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cold Trap</td>
<td>1/1</td>
<td>1/1</td>
<td></td>
</tr>
<tr>
<td>Plugging meter</td>
<td>1/1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Flow meter</td>
<td>1/1</td>
<td>1/1</td>
<td>Calibration Test</td>
</tr>
<tr>
<td>Level meter</td>
<td>1/1</td>
<td>1/1</td>
<td>Calibration Test</td>
</tr>
<tr>
<td>Pressure gage</td>
<td>1/1</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Instrument for reactor core</td>
<td>1/1</td>
<td>1/1</td>
<td>Flow velocity Measurement of temperature</td>
</tr>
<tr>
<td>H₂ detector</td>
<td></td>
<td>1/1</td>
<td></td>
</tr>
<tr>
<td>Sodium leak detector</td>
<td>1/1</td>
<td>1/1:*1</td>
<td>*1:primary &amp; secondary</td>
</tr>
</tbody>
</table>
Development of Cold Trap

- Original cold trap
- JOYO Mock-Up
- MONJU Proto-Type
- Finery cold trap

- Operation technology
- Chemical analysis
- Dismantle technology
- Code development
- Improvement of capture
- Improvement of capture
- MONJU primary Cold Trap

- Air
- Sodium
- N2 gas
- Mesh
- Upper plate
- Sodium Drain nozzle
- Impurity capture
- Body
- Cooling Tube
- N2 gas
Automatic Plugging Meter
Flow Meter for main primary loop on JOYO

Piping system on main primary loop of JOYO is installed by double wall tube
Flow Meter of MONJU Secondary Main Loop

- Sodium Pipe
- Duct
- Magnet
- Insulator
- Electrode Pole
- Cover
Outline of Flowmeter for Monju

Type  Electro-Magnetic Flowmeter (Permanent Magnet)

<table>
<thead>
<tr>
<th></th>
<th>Dimension (mm)</th>
<th>Weight (kg)</th>
<th>Magnet Field (Gauss)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>(A)</td>
<td>(B)</td>
<td>(C)</td>
</tr>
<tr>
<td>(1) Primary (24 in)</td>
<td>2000</td>
<td>1500</td>
<td>2400</td>
</tr>
<tr>
<td>(2) Secondary (22 in)</td>
<td>1600</td>
<td>1400</td>
<td>2200</td>
</tr>
<tr>
<td>(3) ACS (12 in)</td>
<td>1200</td>
<td>1200</td>
<td>1500</td>
</tr>
</tbody>
</table>

Operating Temp.  190°C~400°C
Flow Rate  $\leq 100$ m³/min
Flow Meter Calibration test Facility
图5. BLOCK DIAGRAM OF ULTRASONIC FLOW METER
Sodium Level Meter
(Induction Type)

Outline Structure
- Cable connection
- Support
- Sodium level
- Guide Tube
- 1st & 2nd Coil
- Protection Tube
- Bovine of Coil

Principle Circuit
- Primary Coil
- Secondary Coil
- AC current: 2-3kHz
- Measure of Current & Voltage
- Correct Coil
- To be compensate for Temperature of Sodium and Coil etc.,
Pressure Gauge for Sodium (NaK transducer Type)

- Cell
- Diaphragm
- Thermo Couple
- Test tube
- Capillary tube
- NaK
- Protection Tube
- Diaphragm
- Silicon oil
- Pressure sensor
- Process connection
Small Sodium Leak Detector

Sodium Ion Detector

- Sodium Mist
- Sampling gas
- Filament
- Ion Collector

Radiation Ion Detector

- Standard Ion chamber
- Radiation source
- Insulator
- Electrode
- Ion chamber
- Radiation Source
- Sampling gas
Sodium Small Leak Detection System for MONJU primary system
Probe of temperature measurement for outlet of reactor core

Protection Well for thermo-couple

Thermo-couple:
Flow meter for outlet of reactor core

Outline arrangement at upper core

- Probe
- Thermo-couple
- Guide tube
- Fuel assemble
- Probe tube

Principle of Eddy Current Type flow meter

- Magnet field
- 1st coil
- 2nd coil
- 1st coil
- On static sodium
- On sodium flow

F: 0 m/sec
F: v m/sec
Effected current
Flux
Effected Current
Under Sodium Viewer
-In Reactor Vessel of MONJU-

Sending & receiving of USW

Inner core

Outer core

Blanket & shield region

Upper core structure

Horizontal Move

Reactor Vessel

Inner Cylinder

Reflector plate for Ultra Sonic Wave
The under-sodium viewer of the horizontal type is settled on the fixed plug after reactor shutdown through a hole provided on the fixed plug prior to the fuel handling. The purpose of this device is to confirm that no obstacle exists between the bottom of the upper core structure and the heads of the core elements by a horizontal emission of the ultra-sonic wave and detection of the reflected wave.

**SPECIFICATION OF THE UNDER-SODIUM VIEWER**

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Horizontal stroke:</td>
<td>±37° ~ ±29°</td>
</tr>
<tr>
<td>Angle of elevation:</td>
<td>±3°</td>
</tr>
<tr>
<td>Gap detection limit:</td>
<td>20 mm</td>
</tr>
</tbody>
</table>

| Main material | SUS 304 Carbon steel |

![Reflecting plate in the reactor vessel](image)
2.2 EMP

- Comparison of Mechanical Pump & EMP
- Principle & Type of EMP
- Design Consideration of EMP
- Sodium Test of Mock Up EMP for JOYO
## Comparison of EMP & Mechanical Pump

<table>
<thead>
<tr>
<th></th>
<th>EMP</th>
<th>Mechanical Pump</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pump Efficiency</td>
<td>40〜50% (Max.)</td>
<td>70〜80%</td>
<td></td>
</tr>
<tr>
<td>System Design</td>
<td>No necessary free surface level &amp; cover Gas System</td>
<td>Necessary Free Surface Level &amp; Cover Gas</td>
<td>Layout of Pump Natural Circulation of Cooling System</td>
</tr>
<tr>
<td>Plant Design</td>
<td>Compact Arrangement of Plant</td>
<td>Level Control of Pump Revel</td>
<td>To be consider for sodium leak</td>
</tr>
<tr>
<td>Character of Pump</td>
<td>Static Component</td>
<td>Motor Drive</td>
<td></td>
</tr>
<tr>
<td>Maintenance</td>
<td>Check of Coil &amp; Duct</td>
<td>Check of Motor &amp; Sodium Bearing*</td>
<td>*: to be prepare cask for maintenance</td>
</tr>
</tbody>
</table>

出典：冷却系機器設計と特性 (応用講座) 平成 年 月 日 亀井 滉
Electro Magnetic Pump

Conduction Type

Induction Type

Alternating – current Conduction pump

Conduction Type

Flat linear induction pump

Induction Type
EMP to be required flow coast down
<table>
<thead>
<tr>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Material Election</strong></td>
<td><strong>Compatible with Sodium</strong></td>
</tr>
<tr>
<td><strong>Transformation of Duct</strong></td>
<td><strong>Thermal Expansion of Duct</strong></td>
</tr>
<tr>
<td><strong>Intake of Gas to Duct</strong></td>
<td><strong>Installation &amp; Structure such as Escaping of Gas For EMP Duct</strong></td>
</tr>
<tr>
<td><strong>Cooling of Coil</strong></td>
<td><strong>Coil has to cold on EMP operation</strong></td>
</tr>
<tr>
<td><strong>Preheat of Duct</strong></td>
<td><strong>Cold Point of Duct &amp; Support</strong></td>
</tr>
<tr>
<td><strong>Sodium Drain</strong></td>
<td><strong>Sodium will be drain after stopping of loop operation</strong></td>
</tr>
</tbody>
</table>
Q-H Characteristic of EMP -JOYO mock up -

Flow Diagram of Test Loop

Schematic of the pump Duct

Out:Q/H 941 78 - 76 常陽ナトリウム冷却系耐久試験装置
### Mock-Up Pump for EMP of JOYO 2nd Auxiliary Loop

<table>
<thead>
<tr>
<th>NO</th>
<th>Name</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Flame</td>
</tr>
<tr>
<td>2</td>
<td>Flame</td>
</tr>
<tr>
<td>3</td>
<td>Duct</td>
</tr>
<tr>
<td>4</td>
<td>Linear Motor Core</td>
</tr>
<tr>
<td>5</td>
<td>Liner Motor Coil</td>
</tr>
<tr>
<td>6</td>
<td>Support</td>
</tr>
<tr>
<td>7</td>
<td>Support of Duct</td>
</tr>
<tr>
<td>8</td>
<td>Cooling Fan</td>
</tr>
<tr>
<td>9</td>
<td>Thermal Insulation</td>
</tr>
<tr>
<td>10</td>
<td>Thermal Insulation</td>
</tr>
<tr>
<td>11</td>
<td>Wind Plate</td>
</tr>
</tbody>
</table>

### Flat Linear Induction Pump

1. Duct Form: Flat Structure
2. Trapezoid Pipe for Inlet & Outlet

---

出典：SNR941 73-48 常陽ナトリウム冷却系耐久試験装置
Design Character of Sodium Valve & Pipe

- Type of Sodium Valve
- Consideration of Design & Operation for Sodium Valve
- Consideration of Design & Operation for Piping System
## Consideration of Piping System

<table>
<thead>
<tr>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heat Expansion</td>
<td>To be no effected for component</td>
</tr>
<tr>
<td>Heat Expansion coefficient is large Austenite Stainless Steel</td>
<td>Permitted for component</td>
</tr>
<tr>
<td>Sodium Leak</td>
<td>Guard Vessel &amp; High Level Position for Piping System</td>
</tr>
<tr>
<td>Support of Piping System</td>
<td>T of Pipe &amp; Support Structure</td>
</tr>
<tr>
<td>Support of Piping System</td>
<td>For Thermal Transient</td>
</tr>
<tr>
<td>Nozzle</td>
<td>Thermal Shock</td>
</tr>
<tr>
<td>T-joint Pipe</td>
<td>Thermal Striping</td>
</tr>
<tr>
<td>Sodium Drain</td>
<td>To be Drain at Maintenance</td>
</tr>
<tr>
<td>Slope of Piping system</td>
<td>&gt; 1/20</td>
</tr>
</tbody>
</table>
## Design Consideration of Sodium Valve

<table>
<thead>
<tr>
<th>Consideration</th>
<th>Remark</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Material</strong></td>
<td>Compatible with Sodium Sheet material</td>
</tr>
<tr>
<td><strong>Thermal Shock</strong></td>
<td>To be confirm by test Transformation of Body</td>
</tr>
<tr>
<td><strong>Shaft Seal:</strong> for Bellows seal <strong>for Freeze Seal</strong></td>
<td>Failure by Imperfection Preheat or Vibration of Fluid Sodium Leak by Cooling miss May be apply to small valve May be apply to large valve</td>
</tr>
<tr>
<td><strong>Body Sheet</strong></td>
<td>Failure by Transformation of Body Scar by Hard Dust in Sodium Thermal Shock</td>
</tr>
<tr>
<td><strong>Water Hammer</strong></td>
<td>To be apply Dash Pot for Check Valve</td>
</tr>
<tr>
<td><strong>Cavitation</strong></td>
<td>To be apply Needle Valve for flow adjust</td>
</tr>
<tr>
<td><strong>Closure Time of Shaft</strong></td>
<td>Request from Plant May apply for isolation valve</td>
</tr>
<tr>
<td><strong>Sodium Drain</strong></td>
<td>Gap of Bellows</td>
</tr>
</tbody>
</table>
Piping System of Loop Type FBR (Monju)

To keep natural circulation & Level of Reactor Core for sodium leak, Primary piping system is arranged high level position & Guard Vessels are equipped each component and Piping system.

To prevent back flow from non-failure system to Sodium leak system, Check valves are installed to inlet pipe of Reactor Vessel.
Thermal Striping on T joint - Pipe

In-water testing

High Temp Na

Not good structure

Good structure

Low Temp Na
Check Valve is only one Valve for Primary System. In case of cooling systems has over 2 loop, mission of check valve is to prevent back flow, if main Pump have trouble.
Bellows Seal Valve

Bellows
To be consider Vibration of Valve Shaft by Fluid
To be consider Elasticity & Endurance of Bellows
To be confirm before operating temperature of Valve Body

Bellows Seal valve will be adapted for small valve
Freeze Seal Valve

Gate Type Valve

To be keep below melting point at freeze seal aria.
Freeze seal valve will be adapted for Large Sodium valve

Butterfly Type Valve

Cooling Fin

Freeze Seal

出典：PNC SN0410 - 90 - 031 動力炉の実用化をめざして 112p
Temperature Distribution of 12B Freeze Seal Valve

Room Temp. 9.5°F

Cooling Fin

Na

Na Temp: 340°F

Melting Point of Na

97.8°F
# Key Technical Issues for Commercialization of FBR Power Plants

<table>
<thead>
<tr>
<th>Key Technical Issues</th>
<th>Target of Development</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Long-life high performance fuels</td>
<td>- Reliable fuel with its burn-up above 200,000MWh/t</td>
</tr>
<tr>
<td>2. High performance core for large-scale FBRs</td>
<td>- Optimization of a high performance core for 1,500MWe plant</td>
</tr>
<tr>
<td>3. Plant service at high temperature</td>
<td>- Structural material resisting higher temperature (over 550°C)</td>
</tr>
<tr>
<td>4. Optimization of heat transport and fuel handling systems</td>
<td>- Optimization of systems' layout to reduce the size of reactor building</td>
</tr>
<tr>
<td></td>
<td>- Development of compact and reliable components</td>
</tr>
<tr>
<td>5. Optimization of reactor containment design</td>
<td>- Realistic estimation of rise in pressure at severe accident</td>
</tr>
<tr>
<td>6. Seismic isolation</td>
<td>- Licensable evaluation model and standards of seismic isolation design</td>
</tr>
<tr>
<td>7. Elimination of secondary heat transport system</td>
<td>- Establishment of safety logic and corresponding protection system</td>
</tr>
<tr>
<td></td>
<td>- Development of reliable double-walled tube type steam generator</td>
</tr>
<tr>
<td>8. Reliable decay heat removal system</td>
<td>- Passive decay heat removal system of natural convection</td>
</tr>
<tr>
<td>9. Autonomous plant operation</td>
<td>- Fully automatic plant operation using the artificial intelligence (AI) technology</td>
</tr>
<tr>
<td>10. Optimization of safety logics</td>
<td>- Safety design and evaluation with an adequate margin</td>
</tr>
</tbody>
</table>
Key Technology for Economical LMFBR

Scale Up
- Scale Up Technology: Check Valve, Shield Plug, etc.
- Compact Components: High Speed Pump, Compact Fuel Exchange Machine, New Concept Selector Valve, By-Pass Flow Meter, etc.

Capital Cost
- System Simplification: Expansion Joint, Elimination of Secondary Loop, Cold Trap in Tank, Preheating System etc.
- Manufacturing Cost: Mod. Cr-Mo Steel, Standard Component etc.
- Plant Availability: Fuel Handling Machine by Shoot Method, etc.

Operating Cost
- Maintenance Cost: Robot Technology, ISI Technology, Sodium Cleaning Method for Fuel etc.
- Facility Service: New Insulation Material, Waste Disposral Technique, etc.

Safety
- Safety Logic: DRACS, SASS, Upper Core Instrumentation etc.
- Reliability: Double Wall Type Steam Generator, Emergency Electric Power etc.
Compact Plant Design by Expansion Joint
Secondary Pump & Cold Trap

Steam Generator

IHX & Primary Pump

Reactor Vessel

Image of FBR Large Plant
Outline of Large FBR Plant

- Reactor Vessel
- IHX & Primary Pump
- Primary System
- Secondary system
- Secondary Pump
- Water
- Steam
Combined Component Pump & IHX
Comparison of Customary & Combination System

<table>
<thead>
<tr>
<th></th>
<th>Customary System</th>
<th>Combination System</th>
</tr>
</thead>
<tbody>
<tr>
<td>Weight</td>
<td>361 (ton)</td>
<td>310 (ton)</td>
</tr>
<tr>
<td>Space</td>
<td>100 (%)</td>
<td>90 (%)</td>
</tr>
</tbody>
</table>

**Diagram: Customary System vs. Combination System**

- **Customary System**: Includes a Pump, IHX, and R/V Component.
- **Combination System**: Includes a Pump, IHX, and Combination Component.
## Elimination of Secondary Heat Transport System

<table>
<thead>
<tr>
<th>Target</th>
<th>Rationalization of Heat Transport System by Eliminating Secondary Heat Transport Circuit</th>
</tr>
</thead>
<tbody>
<tr>
<td>Key Technology</td>
<td>Establishement of Safety Logic and Corresponding Protection System</td>
</tr>
<tr>
<td></td>
<td>Development of Double Wall Type Steam Generator and Leak Detector System</td>
</tr>
<tr>
<td>Effect</td>
<td>Construction Cost 15% Reduction</td>
</tr>
<tr>
<td></td>
<td>House Load 30% Reduction</td>
</tr>
<tr>
<td>Development Status</td>
<td>Plant System Design Study and Safety Analysis Study Satisfying Safety Logic</td>
</tr>
<tr>
<td></td>
<td>1MW Double Wall Type Steam Generator Installed for Testing</td>
</tr>
</tbody>
</table>
Conceptual Design of Double Walled Tube Type SG
Concept for Elimination Secondary System (Example)
Outline of Leak Detection System Concept
Compact Type Fuel Handling Machine
Design Effort for Large FBR in Japan

- Compact piping system
- 2 loops
- Combination of IHX & Pump
- Compact Reactor Vessel
- Long Life Fuel
- Relaxation for Earthquake
- Double walled type Steam Generator
- Double walled Type piping system
- Technology of ISI & Repair
- Improvement of Safety SASS DRACS
マイクロフォーカスX線透過試験

図5 放射線透過審査の原理

【出典】(社)日本非破壊検査協会（監修）：非破壊検査シリーズ「非破壊試験概論」1993, p.28
Mechanical seal of Sodium Pump -FERMI-1 Primary pump-