Safety Design of LMFBRs in Japan

Workshop on Safety Criteria and Options
in Fast Breeder Reactor Design
May 17−19, 1983
Paris

K.Mochizuki, M.Hori, A.Izumi (PNC)
T.Saito, T.Umeoka (FEPC)
Y.Kumaoka (Toshiba)
H.Ogasawara (Hitachi)
I.Aoki, M.Hanawa, M.Ezaki (Mitsubishi)

Contents

Page

* * * * *

1. Present Status and Future Direction of LMFBR Development in Japan

The development of fast breeder reactors in Japan has been pushed forward as a national project by the joint efforts of the government and the private sector, in which the Power Reactor and Nuclear Fuel Development Corporation (PNC) is taking a leading role.

The experimental reactor "JOYO", the first fast breeder reactor in Japan, has been operated extremely satisfactorily since the first criticality in 1977, and has accumulated a great deal of data concerning the function and performance of the plant, by various tests including the natural circulation test under the condition of simulated complete power loss.

In 1982, the modification to the second phase core with the maximum thermal output of 100MW was carried out, and the full power was achieved in March this year. With the rated output operation to be started soon, it is expected that a great deal of technical data will be obtained, including the irradiation data of fuel and materials as well as the experience of operation and maintenance.

Meanwhile, preparatory work for the prototype reactor "MONJU" has been in progress at the site since January 1983, the consents of the government and the local community for the siting being obtained in 1982. As for the safety evaluation, the first examination by administrative authorities was completed in 1981, and the second examination by Nuclear Safety Commission was completed in April this year. Contract negotiations with constructors of the plant are also under way.

In parallel with the design, construction and operation of "JOYO" and "MONJU", extensive research and development activities for the prototype reactor "MONJU" have been carried out including the fuel cycle of the fast breeder reactor, mainly at the Tokai Works and the O-arai Engineering Center of the PNC. From these activities, many technical experiences have been accumulated, as well as various documents and guidelines for design and evaluation, data bank for the material, and the license experience such as the safety evaluation.

With regard to the demonstration reactor, which is to follow the prototype reactor, the schedule and assignment of the role were defined in the "Long-term Plan for the Development and Utilization of Nuclear Energy (1982)" of the Atomic Energy Commission as follows:

The construction of a demonstration reactor is to be started in the early 1990s, and the commercialization will be completed around 2010. It is desirable that utilities should take an active role in the construction and operation of the demonstration reactor with the back-up of the government. Related research and development activities are to be carried out by the PNC as a center, with the increasing participation by the private sector.

In Japan, based on the results of research and development activities carried out in the design activity of "MONJU", the PNC has conducted study on the design of a large-scale reactor since fiscal 1975, and the electric utilities have conducted study on the conceptual design of a demonstration reactor since fiscal 1978 in a coordinated manner, making the best of their own experiences.

Both the PNC and utilities are investigating a number of candidate concepts of a commercial loop-type sodium-cooled reactor with the electrical output of 1,000MW, establishing fundamental plants concerning the optimization in the engineering aspects, improvement of reliability and safety, and increase in availability.

At present, the PNC is placing emphasis on the technological progress which has a great potential of realization by the date of construction, while the utilities attach importance to reliability, and simple operation and maintenance. Although many areas are in common to the designs of the PNC and utilities, there are some differences, for example, in the number of loops and outlet temperature of coolant, and the work is now under way to harmonize the differences. At the same time, in connection with the scale-up of the reactor, both parties are now investigating concepts different from that of "MONJU", including the entering flow of coolant to the reactor vessel, position of pumps, handling method of fuel. Meanwhile, the utilities are also conducting investigation on the design and aseismic tests of a pool-type reactor, with the view to ensuring flexibility concerning the selection of a type of the reactor in future.

Japan intends to make a coordination of various concepts in an early time, including selection of the type of a reactor, and define fundamental specifications of a demonstration reactor which is intended for the future commercialization, while fully utilizing our experiences accumulated in the operation of "JOYO" and "MONJU", and the consideration of the results of technological development in the world.

It is estimated that the cost of construction of fast breeder reactors will increase substantially when it comes to the stage of commercialization, and the development should be carried out efficiently. From this point of view, the international cooperation seems to play a key role. Japan concluded cooperation agreements with almost all the countries which are developing fast breeder reactors, such as France, West Germany, Italy, UK, USA and USSR, and has since cooperated with these countries in such fields as the exchange of information and joint experiments. In the future development of the Japanese demonstration reactors, cooperation with these countries are to be further enlarged, for the purpose of complementing our own development program. In particular, concerning such issues as the reevaluation of the safety design criteria in which the international consensus is important, we would like to cooperate with every nation, who is engaged in fast breeder reactor development, through exchange of opinions and joint research projects.

2. Design and Safety Approach for "MONJU" Plant

2.1 Summary of "MONJU" Plant Design

The prototype LMFBR "MONJU" is a loop type reactor with 280 MWe power output. A site located in the Tsuruga Peninsula in Fukui Prefecture, approximately 400 km West of Tokyo, where several LMRs are in operation, has been decided for constructing "MONJU" by the Power Reactor & Nuclear Fuel Development Corporation (PNC).

The PNC made an application to the Prime Minister for reactor establishment in December 1980. The Science and Technology Agency (STA) examined the safety of the "MONJU" and prepared the draft safety evaluation report (the first examination).

STA submitted this report to the Nuclear Safety Commission (NSC) to hear the opinion with respect to reactor safety in May 1982. The Committee on Examination of Reactor Safety, a essential part of the NSC for safety examination, prepared the review report through the one year examination of the safety of "MONJU". Based on this report NSC submitted the reply note to the Prime Minister in April 1983. (the second examination)

The permission of the Prime Minister for the reactor establishment will be issued near future based on these examinations.

The principal design features and parameters are shown in Table 2.1, and the plant layout and the principal main heat transport system features are shown in Figure 2.1 and Figure 2.2, respectively. Its thermal output is 714 megawatts. It has three loops with both primary and secondary pumps located at cold leg. The core consists of two homogeneous zones of different enrichment and fuel is plutonium-uranium mixed oxide.

Reactor vessel is made of austenitic stainless steel. It is about 18 meters high and 7 meters in diameter. The coolant enters the reactor vessel from the inlet nozzle at the lower part of the vessel at 397°C and exits from the

Table 2.1

Principal Design and Performance Data of "MONJU"

Reactor Type		Sodium cooling loop type
Thermal Power		714 MW
Electrical Power		about 280 MW
Fuel Material		PuO_2-UO_2
Core Dimension	Equivalent diameter	1,790 mm
	Height	930 mm
	Volume	2,335 lit.
Pu Enrichment (Pu fissile %)		Inner core/Outer core
	Initial core	15/20
	Equilibrium core	16/21
Fuel Inventory	Core (U + Pu metal)	5.9 Ton
	Blanket (U metal)	17.5 Ton
Average Burn up of Discharged Fuel		80,000 MWD/T
Cladding Material		SUS316
Cladding Outside Diameter/Thickness		6.5/0.47 mm
Maximum Cladding Temperature (middle of thickness)		675°C
Power Density		283 kw/lit.
Blanket Thickness (axial/radial)		Upper 300 / Lower 350 / 300 mm
Breeding Ratio		1.2
Reactor Sodium Temperature (inlet/outlet)		397/529°C
Secondary Sodium Temperature (IHX outlet/IHX inlet)		505/325°C
Reactor Vessel (height/diameter)		17.8/7.1 m
Number of loops		3
Pump Position (primary and secondary loop)		Cold leg
Type of Steam Generator		Helical coil, once-through unit type
Steam Pressure (turbine inlet)		127 kg/cm^2g
Steam Temperature (turbine inlet)		483°C
Refueling System		Single rotating plug with fixed arm FHM
Refueling Interval		6 months

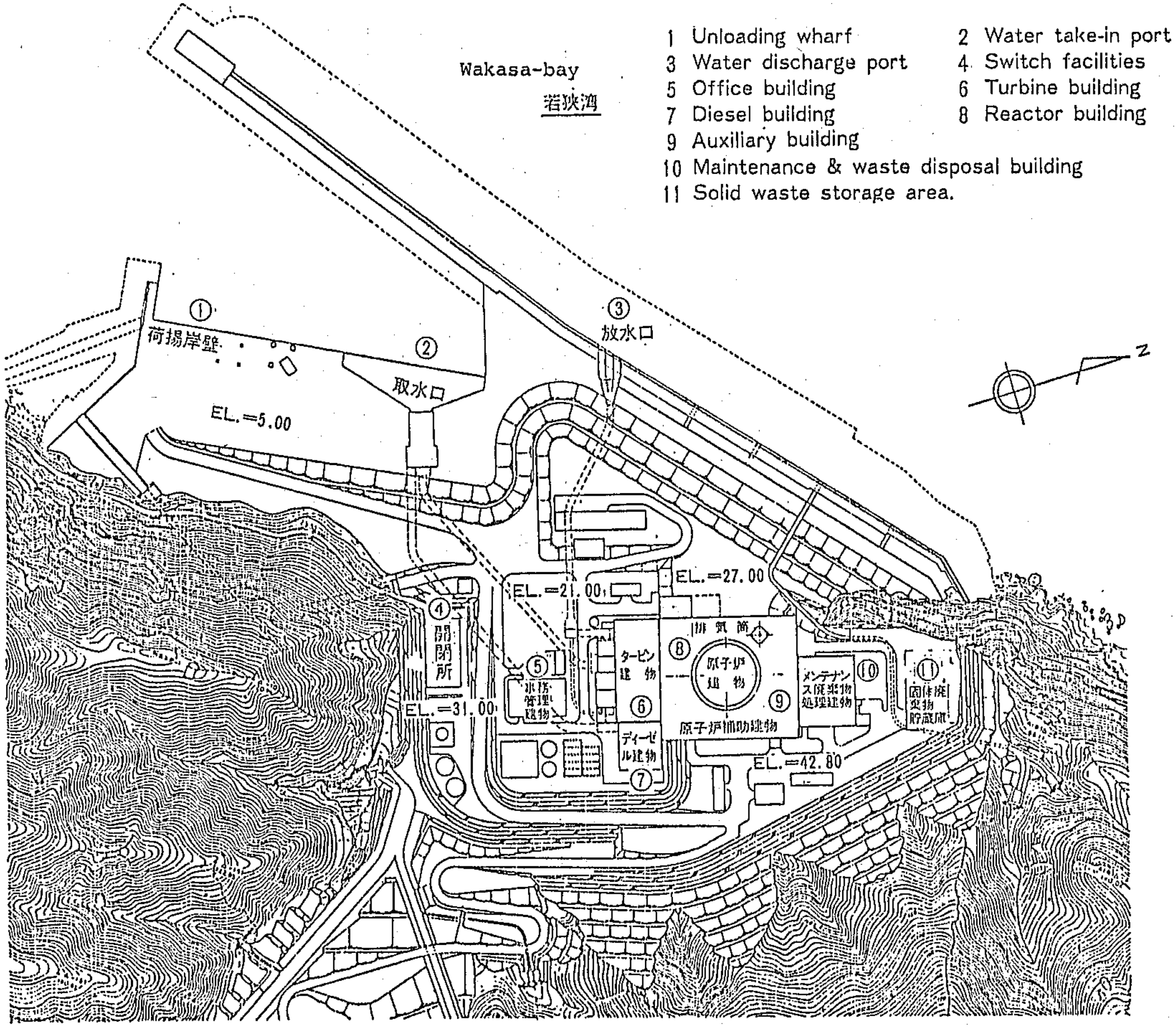


Figure 2.1 Plant Layout of "MONJU"

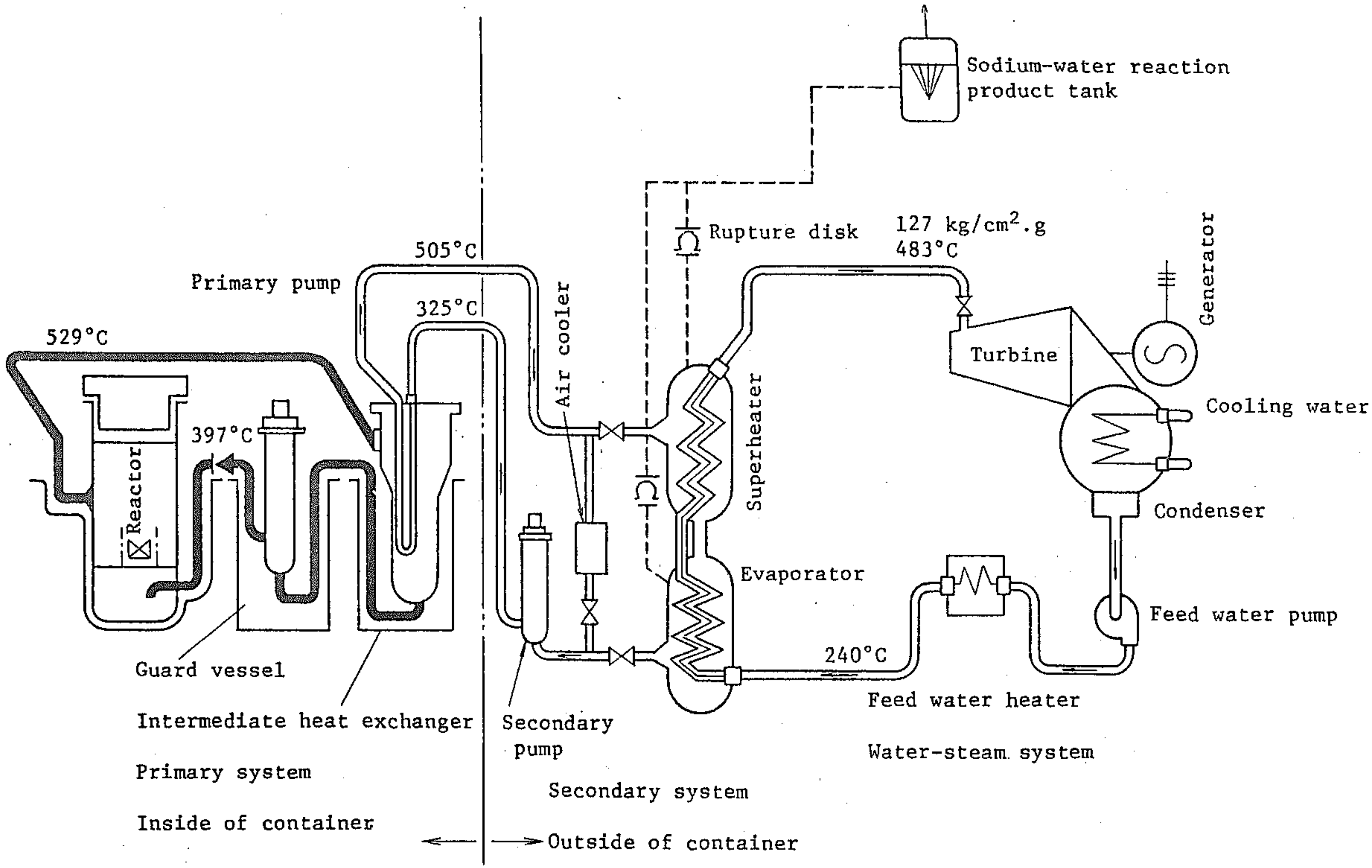


Figure 2.2 Main Heat Transport System Features

outlet nozzle above the core at 529°C. Main primary cool- and pipings are laid at elevated position, while the reactor vessel, pumps and intermediate heat exchangers are surrounded by guard vessel.

Steam generators are of once-through type. There are one evaporator and one super-heater per loop. Heat transfer tubes are of helical coil type. There is a cover gas space above the sodium surface.

Valves are provided at the inlet and outlet side of steam generator system. Auxiliary cooling system branches off from the IHX side of these valves. In the decay heat removal mode of operation, inlet valves of steam generator system are closed and the sodium flow through steam generators is stopped and it flows through auxiliary cooling system, while pumps are shifted to ponymotor operation. Decay heat is released to the air through air blast heat exchanger of the auxiliary cooling system.

2.2 Safety Design and Safety Analysis of "MONJU"

In Japan the basic philosophy on the safety design and safety analysis to assure the safety of LMFBR plants is given in the "Safety Evaluation Policy for LMFBRs" determined by Nuclear Safety Commission in November, 1980. The safety design and safety analysis of "MONJU" are made to meet the philosophy and requirements of the Safety Evaluation Policy.

The philosophy and requirements described in the Safety Evaluation Policy may be summarized as following points.

- to consider fully specific characteristics of LMFBRs in the evaluation of LMFBR safety.
- to consider the experimental data and the methods of analysis with appropriate margins in the safety evaluation.
- to take account of the existing criteria and guidelines which are relevant to the safety evaluation of LMFBRs.

In the Appendix of the Safety Evaluation Policy, it describes about safety design and evaluation of LMFBRs.

In the chapter of this Appendix, titled "Safety design of LMFBRs, it states that safety design of LMFBRs should refer to (1) the Guidelines for the Safety Design of LWR Power Plants, and (2) the Guidelines for the Seismic Design Evaluation of Nuclear Power Plant and also describes the items specific to LMFBRs which should be considered in safety design, such as Reactor core, Sodium, Reactor shut-down system, Reactor coolant boundary, Intermediate heat transport system, Decay heat removal, Elevated temperature design, Seismic design, etc.

The safety design guide of "MONJU" plant has been prepared by PNC in accordance with the philosophy and requirements of this Appendix and it is contained in the additional reports of the application. It is based on the Guidelines for the Safety Design of LWR Power Plants and the criteria which consider the items specific to LMFBR such as sodium, intermediate heat transport system and so on are added to it, and the criteria related to the items specific to LWR power plants such as the requirement for ECCS are eliminated.

The main safety design features of "MONJU" which respond to the requirements connecting with the items mentioned above are shown in following part of this paper.

In another chapter of the Appendix titled "Safety Evaluation of LMFBRs", it states that the Guidelines for the Safety Evaluation of LWR Power Plants should be applicable for the safety evaluation of LMFBRs, taking into consideration of the specific characteristics of LMFBRs. And it shows fundamental principles for the events to be postulated and criteria to be used for judgment. The safety analyses of "MONJU" have been performed in accordance with these requirements.

In Section 4 of this chapter, it describes the items specific to LMFBRs which should be considered in safety analyses such as Neutronic Characteristics, Thermal-hydraulic Characteristics, Mechanical Characteristics and

Chemical Characteristics. It is pointed out as one of the mechanical characteristics that sufficient consideration should be given to determine the mode and size of piping rupture postulated for coolant leak accidents.

Based upon the evaluation of the integrity of the primary heat transport system piping from such view points as the improbability of the non-ductile fractures, anticipated stress variation during the plant life time and the results of the experiments on the creep strength and the fatigue failure strength, the opening of the area of a quarter of the diameter times the thickness of the piping is postulated as the envelop of the magnitudes of the piping failures for the safety analysis of the primary coolant leakage accident for "MONJU".

In Section 5 of this Appendix, a remarkable policy is stated as follows. "Since the operation experience of LMFBRs is limited, safety evaluation should be made for postulated events with lower probability and higher consequences (LP/HC Events) than the accident analysed for the evaluation of safety design features. The event should be studied to the sufficient extent considering the relations among its initiating event and the preventive measures against the progression to the series of subsequent events, to confirm that the associated release of radioactive materials to the environment is appropriately limited."

To respond to the requirement, three events are postulated and safety evaluation of these events are contained in the safety analysis report of "MONJU". These are shown in the following part of this paper.

2.2.1 Aseismic Design

The site where the Japanese prototype LMFBR "MONJU" is going to be constructed is located in the less - seismic zone favorably characterized by rather hard and stable bedrock.

There occur so frequent earthquakes in Japan that we have the special standards for aseismic design, which have been applied to light water reactor nuclear power station.

"MONJU", standing just in the final stage of the safety investigation, is also designed by the use of those standards. Regular investigation on the aseismic design of "MONJU" is to start at the time of application for construction permit of the building and the relating structures.

Basic philosophy of aseismic design in Japan are as follows;

(1) Buildings and structures shall be, in principle, of rigid construction, and the foundation mat shall be directly supported by bedrock.

(2) Two kinds of design earthquakes S_1 (the strongest earthquake) and S_2 (the limit earthquake) shall be used.

(3) The nuclear reactor facilities shall be classified into the three classes A, B and C depending upon the aseismic importance, for all of which aseismic designs are required. The class A facilities and a part of the class B facilities which are thought to be possibly resonant with their support structures shall be checked by dynamic analysis by the use of the design seismic waves S_1 and (1/2) S_2 respectively. Moreover, the whole classes shall be checked by the static method according to the practice of the construction standard of Japan for buildings.

(4) The more important facilities of the class A

Such important facilities as reactor coolant boundaries, control rod drive mechanism, reactor containment, shut-down heat removal system etc. are especially classified as the As class, for which maintenance of their safety function shall be assured by the use of the seismic wave S_2. In this case, inelastic deformation shall be allowed by a certain limit.

(5) An evaluation for uplift shall be required for reactor building

The design seismic wave S_1 for "MONJU" has been determined by the use of more than ten seismic waves which includes the historical records of strong earthquakes in Tsuruga district, and includes the waves analyzed from the active faults near the site. As for the S_1, the maximum velocity amplitude is 13.8 kine, time duration of vibration is 20 sec., and the maximum acceleration is 0.25 g. The design seismic wave S_2 is understood as the limit earthquake which is hypothetically determined from activities of the neighboring active faults. Probability of the S_2 earthquake is thought to be once several ten thousand years. As for the S_2 wave of "MONJU", the maximum velocity amplitude is calculated as 18.2 kine, time duration is 11 sec., and the maximum acceleration is 0.40g. These seismic waves S_1 and S_2 are input at the open surface of the bedrock for dynamic analysis.

Buildings and components are treated as lumped-mass model together with simulation of rigidity of ground as equivalent sway-rocking springs and damping values. As frequency content of ground motion, standard spectrum is used, which is characterized more severe content in the shorter period range than that of U.S. regulatory guide 1.60.

As for damping factors which are important for evaluating responses of components from floor responses, the smaller values, almost a half of those in R.G. 1.61, are used in spite of amplitude of seismic motion either S_1 or S_2.

For reliable seismic evaluations, vibration tests would be performed by the use of shaking tables or actuators, in order to make the vibration characteristics clear, to confirm the safety function under the severe vibration and to observe the damping values. For the case of "MONJU" project, the vibration tests have mainly been performed on (1) reactor vessel for observing the characteristics of oval vibration and for confirming function of the fixture for anti-lateral motion, set under the bottom, (2) fuel assemblies for studying response of multi-assemblies from the view point of dynamic insertion of control rods, (3) control rod drive mechanism for confirming dynamic scrammability of control rods under the S_2 earthquake, (4) primary sodium piping for developing the treating method of the support structures, (5) primary sodium pump for studying seismic features of liquid-static bearing, and (6) steam generators for developing the supports for heat transfer tubes and for studying the total vibration characteristics.

2.2.2 High Temperature Structural Design

It is inevitable for structural design on LMFBR to take a creep characteristics of materials into account because its operating temperature is very high.

The ASME Code Case N-47 of U.S. is one of the standards for the high temperature structural design of nuclear components, which is understood in Japan to include a difficult part for its application to "MONJU" design, because concrete method for treating elastic follow-up of the materials is not definitely clarified. Moreover, it is understood necessary in Japan to review strength of material based on the data of domestic materials.

Therefore, inspite of the fact that the ASME Code Case N-47 is also referred to in Japan, drafts of two kinds of directions[(1),(2)] have been composed based upon the results of several kinds of inelastic analyses, structural component tests, material tests, etc.

These drafts have been precisely checked through the safety investigations for "MONJU" resulting the final standards.[(3),(4)]

The features of these standards include (1) that elastic method is recommended as wide as possible by taking it into consideration that sustained primary stresses are rather low, and (2) that sodium environmental effects and neutron irradiation effects are also involved.

References:

(1) "Structural Design Guide for Class 1 Components of Prototype Fast Breeder Reactor for Elevated Temperature Service (Draft)".

(2) "Structural Design Guide for Components of Prototype Fast Breeder Reactor for Elevated Temperature Service ——— Standards for Strength of Materials ——— (Draft)".

(3) "Elevated Temperature Structural Design Guide for

Class 1 Components of Prototype Fast Breeder Reactor".

(4) "Elevated Temperature Structural Design Guide for Components of Prototype Fast Breeder Reactor —— Material Strength Standards ——".

2.2.3 In-Service Inspection (ISI)

Provisions for periodic inspection are stipulated in Article 29, Periodic Inspection" of the "Law for the Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors", and Articles 47 and 48, "Periodic Inspection" and "Maintenance of Electricity Generating Facilities" of the "Electric Enterprise Law".

The Japan Electric Association issued JEAC4205, 1980 "In-Service Inspection for Light Water Nuclear Power Plant Components" and this code is applied to the current LWR ISI.

In this code, inspection period, accessibility, qualification of inspectors, methods of inspection, evaluation of the results of inspection and corrective measures and repairs are defined. However above mentioned laws do not directly refer to this code. This may be due to the fact that the ISI itself belongs to a relatively new technical area even in the LWR domain so that the industry should accumulate its experience under private code for the time being.

ISI plan for "MONJU" is prepared through safety review process based on the "Safety Evaluation Policy" issued by the Nuclear Safety Comission. This "Safety Evaluation Policy" concludes that use of Sodium as a coolant, experience of "JOYO" and Current LWR codes and rules should be taken into consideration.

Furthermore, "MONJU" has following inherent features:

(1) The "MONJU" reactor coolant boundary can retain the Leak Before Break (LBB) characteristic since there could be no rapidly propagating failure from partially penetrating defects because of the material's sufficient ductility, no possibility of brittle behavior, and low system pressure.

(2) Small amount of coolant leakage is not expected to occur continuously because the "MONJU" reactor coolant boundary is of integrally welded construction. Further-

more, should sodium leakage happen, it can be detected even if the amount is small. Thus, the performance of continuous monitoring for coolant leakage can detect its occurrence at a stage of small leaks.

(3) Necessary sodium levels and continuous cooling of the core can be ensured even in the event of a Primary Coolant Leakage accident.

Under these circumstances, basic ISI plan is established as follows:

The reactor coolant boundary is so designed as to readily detect coolant leakage through such examinations as visual examination with particular reference to the welds on the main equipment and piping, and also through continuous monitoring of sodium leakage throughout the plant service life. Design provisions for accommodating the results of the development activities on inspection equipment are considered. Some structural materials undergo material surveillance tests and are subjected to evaluations of environmental effects as necessary.

The reactor cover gas boundaries are provided with radioactive cover gas monitoring and are subjected to examinations mutatis mutandis to the reactor coolant boundary as necessary.

In-service inspection based on this basic plan is summarized in Table 2.2.

Table 2.2 Summary of Inservice Inspection

Components	Portions Subject to Inspection	Examination Methods
Reactor Vessel	Portions Enclosed by the Guard Vessel	Visual, SLM, MS
	Outside of the Guard Vessel	SLM,RCGM
Shielding Plug	Reactor Cover Gas Boundary	RCGM
Primary Coolant System Pump	Outer Casing	Visual,SLM,RCGM
Intermediate Heat Exchanger	Body	Visual, SLM
	Tubes	Leakage Monitoring
Primary Coolant System Piping	Pipes	Visual,SLM,Volumetric
Primary Coolant System Check Valve	Valve Body	SLM
Core Support Structure	Core Support Plates Core Barrel	MS
Guard Vessel	Body	Visual,(MS)
Primary Auxiriary Sodium System Piping	Pipes	SLM
Secondary Coolant System Pump	Outer Casing	Visual,SLM
Steam Generator	Body	Visual,SLM
	Tubes	Volumetric
Secondary Coolant System Piping	Pipes	Visual,SLM,Volumetric
Auxiliary Cooling System Air Cooler	Tubes	SLM
Ex-Vessel Fuel Storage System	Sodium Boundary	SLM

Definitions of Terms in Table

SLM Sodium Leakage Monitoring

RCGM Radioactive Cover Gas Monitoring

MS Material Surveillance

2.2.4 Loss of Piping Integrity in Primary Heat Transport System and Sodium Fire

In the evaluation of primary coolant leakage accident which is used to evaluate the appropriateness of primary heat transport system design, it is postulated that sodium flows out through the cracked leakage opening of D/2 in length and t/2 in width (D,t: diameter and thickness of the pipe) in primary piping system or a maximum scale leakage opening at a small piping of the connected drain system. A large scale rupture of primary heat transport pipe including double-ended guillotine rupture, therefore, is categorized as beyond design basis accident.

The design policy for the coolant leakage accident in the "MONJU" plant is that the reactor is stopped by an early detection of primary coolant leakage and the coolant level for circulation is secured in the reactor vessel to keep the core being cooled, so that the accident is terminated without causing major damage to the core. The leaked sodium should be handled by adopting a sodium burnup degression provision so as to prevent an excessive temperature rise of the liner and concrete as well as air temperature and pressure rises in the reactor containment vessel, so that the surrounding public is protected from radiation exposure.

(1) Effect of leakage detection signal

Various degree of sodium leakage can be detected by the sodium leakage detectors installed along the component and piping constituting the primary coolant boundary, the sodium level meters in the reactor vessel and overflow tank, the thermometer under the operating floor in the reactor containment vessel, and the leaked sodium level meters in the guard vessel.

(2) Maintenance of core cooling

After shutdown of the reactor, the core is cooled using the slow-speed pump run by the pony motor that is linked to the diesel generator and the air cooler of the auxiliary cooling system. When "single failure criteria" is supposed, fuel cladding may reach a maximum of 740°C under the cooling condition obtained by a single pony motor. The coolant level (emergency sodium level (EsL)) needed for cooling is ensured by installing the piping and component above the EsL and by limiting the space volume between guard vessel and main vessel.

(3) Leaked sodium fire depression

Fire of leaked sodium can be sufficiently depressed by using nitrogen atmosphere (with leak rate of 100%/d at 100 mmAq) of low oxygen concentration ($\leq$3v/o) in the reactor room and the primary heat transport system room. To prevent leaked sodium from directly contacting the concrete, steel liners or storage tank are installed on the floor so that the leaked sodium is kept in the guard vessel, on the floor liner, or in the storage tank for a long time.

2.2.5 Design Condition of Reactor Containment Vessel

The size of reactor containment vessel is determined from the arrangement of main component and piping installed inside, and using the design as strongly-built as practical in the range that can be reasonably attained. It is verified that such a design policy for the reactor containment vessel encloses all postulated accident conditions and ensures its validity. The "loss of reactivity control capability" event is not included in the postulated accidents on the design basis, but the analytical evaluation has revealed that this event will not be greater than the design condition of the reactor containment determined as above.

(1) Integrity of reactor containment vessel at primary coolant leakage accidents

Leaked sodium (evaluated as total of 210 m^3 leakage at 531°C) passes over the face plate of guard vessel support ledge and is collected on the floor liners. The structure temperature is a maximum of 410°C and so below the design temperature of 530°C. The highest temperature of the concrete structure is 120°C and drops to 64°C or below after 30 days from accident, and as a result the concrete is not affected.

The internal pressure of the reactor containment vessel rises to 0.028 kg/cm^2g and the structure temperature rise is several degrees of centigrade. Thus the design condition of the reactor containment vessel (design temperature: 150°C and pressure: 0.5 kg/cm^2g) is satisfactory. The quantity of sodium burnup is 2.7 ton.

(2) Functional evaluation of reactor containment vessel against LP/HC Event

a) At a double-ended guillotine rupture of primary piping system categorized in LP/HC Event, sodium leakage is detected at an early time, so the quantity of leakage is less than 180 m^3 and the burnup quantity of leaked sodium is 2.2 ton. In this case, the highest temperature of floor liner is evaluated as 480°C and the internal pressure of the reactor containment vessel as 0.022 kg/cm^2g, and therefore the integrity of the reactor containment vessel has no problem.

b) Functional evaluation of the reactor containment vessel has revealed that the largest sodium ejection to the operating floor may occur at "LOF without scram event", reaching 290 kg. The integrity of reactor containment vessel has been evaluated against sodium ejection of 400 kg having sufficient allowance to 290 kg. The results revealed that the atmospheric gas in the reactor containment vessel may reach a maximum of 140°C and the internal pressure 0.33 kg/cm^2g, which are under the design condition. In this event, the core debris is held on the core support and retaining plate and as a result, the reactor vessel melt-through cannot occur. According to the above evaluation, the reactor building and containment vessel have been designed as shown in Figure 2.3.

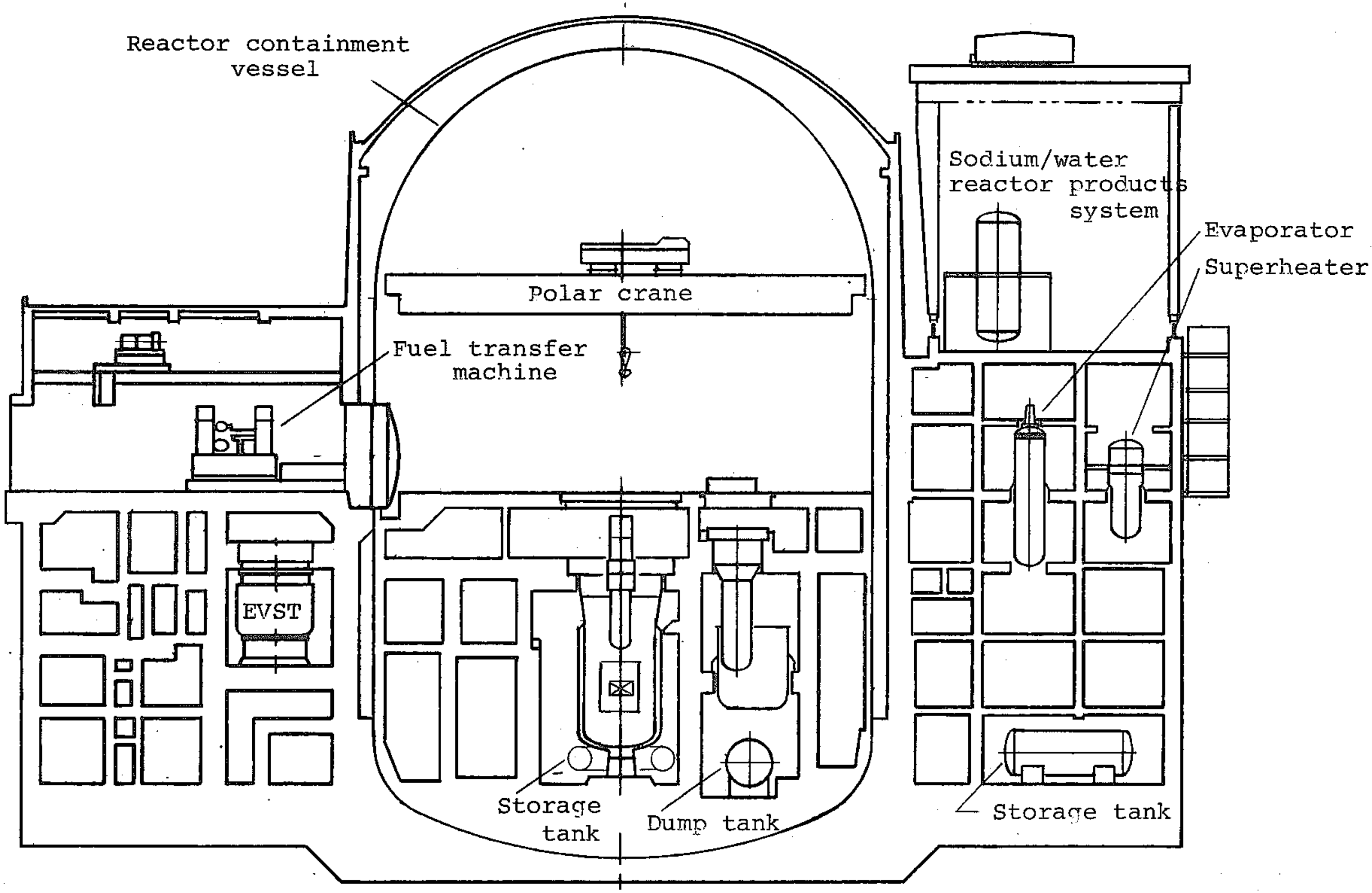


Figure 2.3 "MONJU" LMFBR plant - reactor building -

2.2.6 Sodium Fire in the Secondary Heat Transport System

Secondary sodium circuits do not contain any radioactive materials. Therefore, sodium leaks, even if they occurred, do not bring any radiation hazards to plant operator and the public, and never degrade decay heat removal capabilities indebted to their independent three circuits composition. However, as the "MONJU" auxiliary buildings has the air atomosphere, it is considered that postulated sodium fire might bring sever thermal effect to the building structures.

Design provision against this sodium fire is as follows;

(1) The spilled sodium from main pipings are guided to the storage tank through drain tubes, and that from components and pipings which are located in the lower positions of the plant like the purification system are directly stored on the floor liners of the storage tank room as shown in Figure 2.4. By restraining sodium burning by fire suppression plates, sever thermal effect to the building structures shall be mitigated.

(2) In order to maintain functions of pony motors and air coolers required during Shut-down Heat Removal System operation against aerosols generated from sodium burnings, wall barriers are provided to assure independency of three loops, and also SHRS have been designed for decay heat to be removed by natural circulation.

Finally, how effective above design provision to mitigate thermal effect to the building structures shall be verified by conducting the large scale sodium fire experiments planned in PNC O-arai Engineering Center.

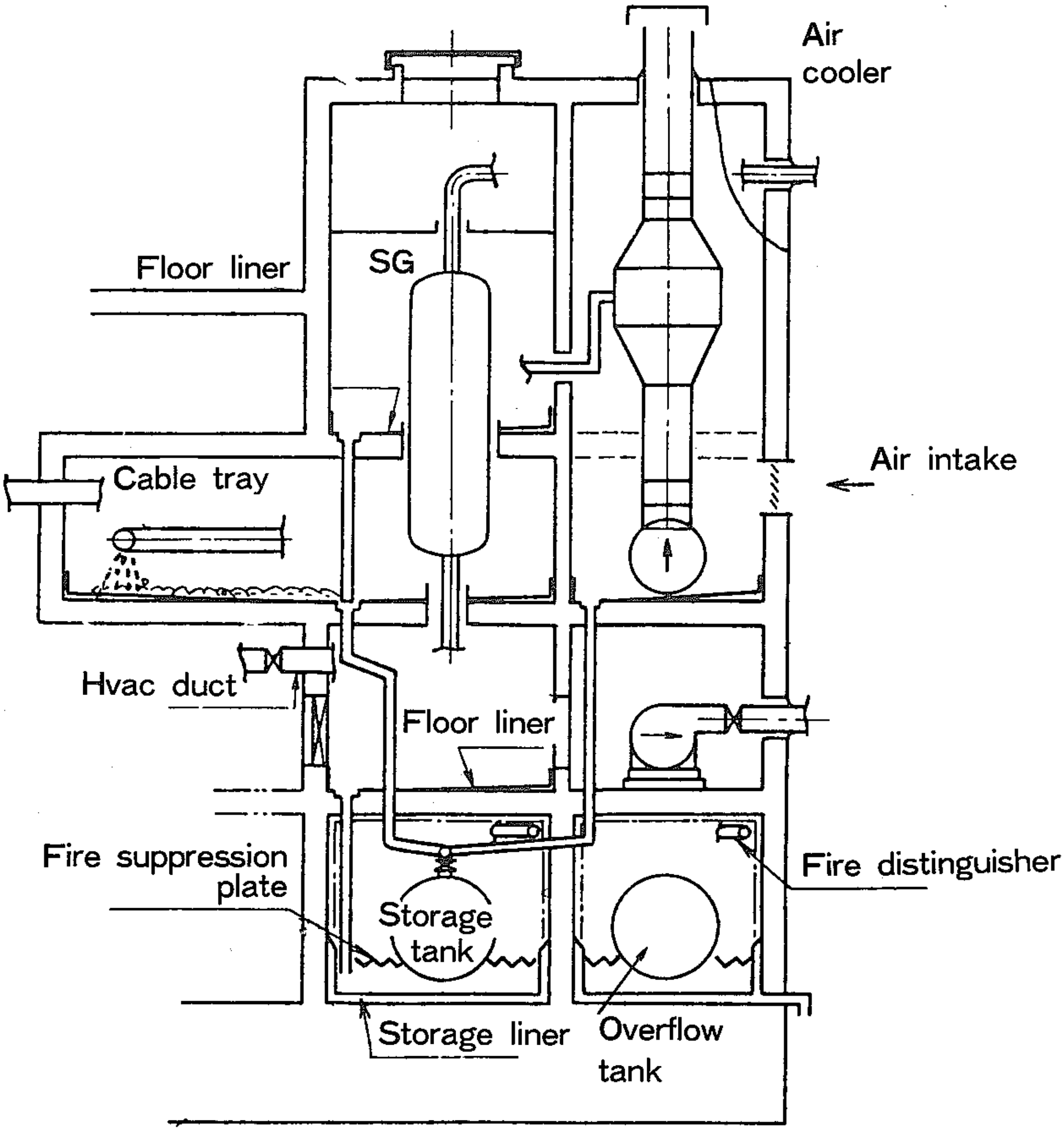


Figure 2·4 Provision against spilled sodium in MONJU secondary coolant circuit

2.2.7 Sodium-Water Reactions

The protection measures for the sodium-water reactions in the steam generator of "MONJU" are as follows. The water-leak detection system which monitors hydrogen concentration in secondary sodium and in cover gas of steam generator is designed to be installed for a small leak. Operators will be able to give the signal for a water-leak after recognizing an alarm of the water-leak detection. For a large leak, the rupture disks, the sodium-water reaction products containment system, and the system which actuates reactor shutdown and isolation of water-steam line automatically are designed to be installed.

(1) The small leak event with leak rate less than 0.1 g/sec is classified as anticipated operational occurrences. Operators can give the water-leak signal based on the alarm of the water-leak detection system with proper time margin. This signal actuates automatic plant shutdown operations such as isolation of water-steam line and blowdown of water in the steam generator, then the sodium-water reaction is terminated. The wastage of the adjacent tubes is estimated to be negligible and the integrity of the tubes is evaluated not to be lost.

(2) Generally, the growth behavior of the sodium-water reaction accident can be described as follows: A leak initiates as a micro-leak due to a faulty weld or another imperfection, and self-enlargement would increase the leak rate to a small or intermediate leak level. Moreover, if no action were taken against the leak, it might grow to a large leak level due to failure propagation effects.

In the practical plant steam generator system, the leak could be detected in the early stage and the relative operations, including an emergency water blowdown, could prevent failure propagation. However, the time required for operations might allow failure propagation to a certain degree; thus the extent of failure propagation should be estimated adequately for the steam generator design.

The design basis leak for the "MONJU" steam generator has been selected as the leak rate equivalent to one plus three double-end-guillotine failures based on the various sodium-water experiments and analyses at PNC.

(3) The analysis of the large water leak was performed for evaporator from a point of view of sodium and water inventories. One tube is postulated to break in a double-ended manner at the lower part of the tube bundle for the evaluation of initial pressure spike. Four tubes are postulated to break in a double-ended manner at the same time for the evaluation of quasi-static pressure, taking account of the effect of failure propagation of tubes.

According to the result of the analysis, the strains of the steam generator, components and piping of the secondary main cooling system, and intermediate heat exchanger do not extend to plastic ones for the initial pressure spike and the quasi-static pressure. The integrity of each installations is evaluated not to be lost.

2.2.8 Reactor Shut-down System (RSS)

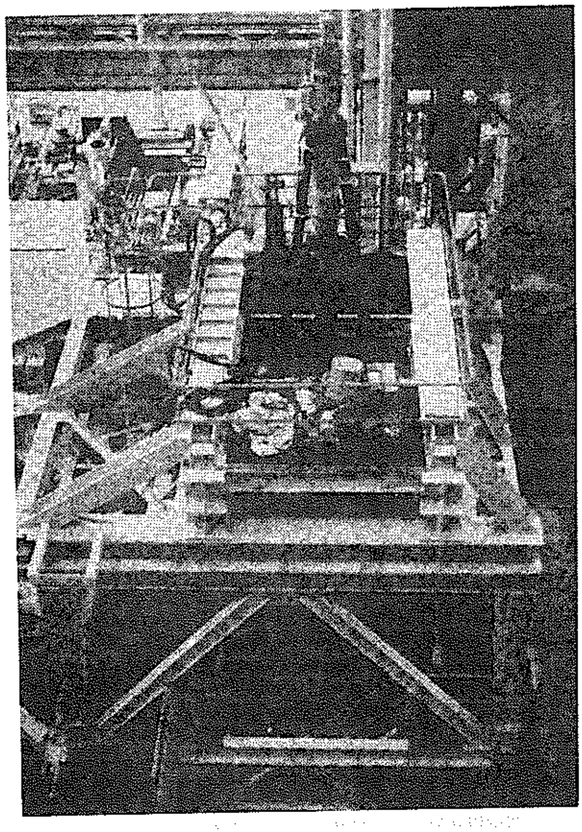
"MONJU" reactor shutdown system is consisted of 13 control rods in primary system and 6 in secondary system. The scram action of a control rod drive mechanism (CRDM) is initiated from actuating its acceleration mechanism by demagnetizing of the holding magnet due to a trip signal, and the control rod cluster (CR) rapidly drops together with the extended stroke and latch motion rod. On the other hand, a secondary control rod drive mechanism is consisted of different system because of diversity requirements on RSS. Its control rod cluster is delatched by demagnetizing of the holding magnet by a trip signal, and only this cluster is accelerated to be inserted by the spring force which is transferred through the acceleration rod.

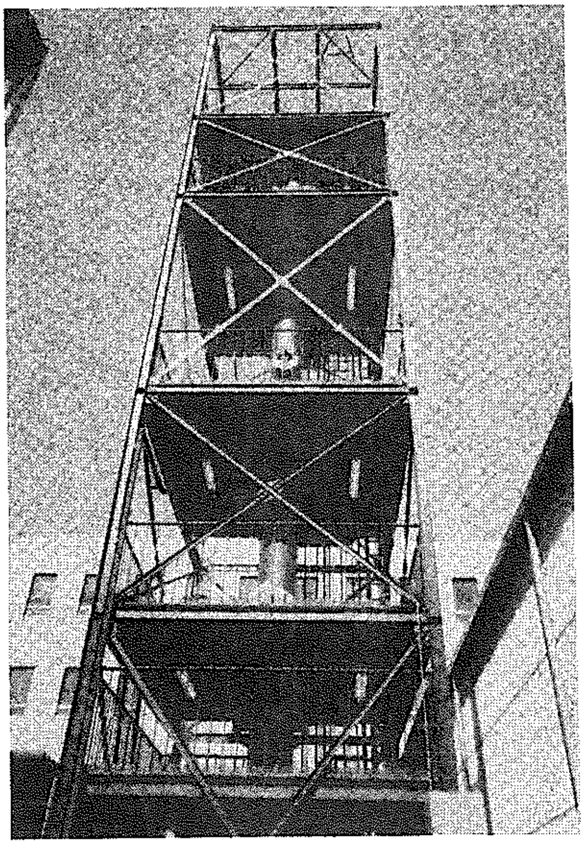
When an earthquake occurs, it will cause the relative displacement between the upper guide to be of CRDM which is contained in the upper core structure and the lower guide tube of CR which is installed into the core, but structural design means between CR and CRDM have been developed for absorbing this displacement to attain the rapid insertion.

The plant protection system has three channels, and is separated into A, B two trains. Both primary and secondary shut-down systems have trip brakers in series respectively in both A, B trains, and when one of each trip braker is actuated, which one of CRs of both shut-down systems can be inserted into the core. Detected variables against abnormal transients have been selected more than two for multiplication required in the shut-down system. Both shut-down systems have enough control reactivity worth to attain the cold shut-down in the condition of one rod stuck having the maximum control worth.

A number of insertion tests by using trially fabricated CRs and CRDMs of both shut-down systems have been carried out in both water and sodium under simulated conditions of displacements and earthquake vibrations as shown in Figure 2.5, and these CRs and CRDMs for tests have never failed to scram during more than 15,000 times of insertion tests. It has been concluded by reliability analyses that this RSS of "MONJU" has enough reliability to regard ATWS as BDBA.

A. Primary control rod tests





B. Secondary control rod tests

Figure 2·5 Overview of MONJU shut-down system insertion tests

2.2.9 Shut-down Heat Removal System (SHRS)

Figure 2.6 shows the system flow diagrams of "MONJU" SHRS. In normal shut-down, decay heat is removed from SG by main motor operations of main sodium coolant circulation pumps, and is released to the sea water by the condenser. But in cold shut-down of the plant, decay heat is removed by auxiliary core cooling systems (ACS) which are consisted of independent three loops with an air cooler for each.

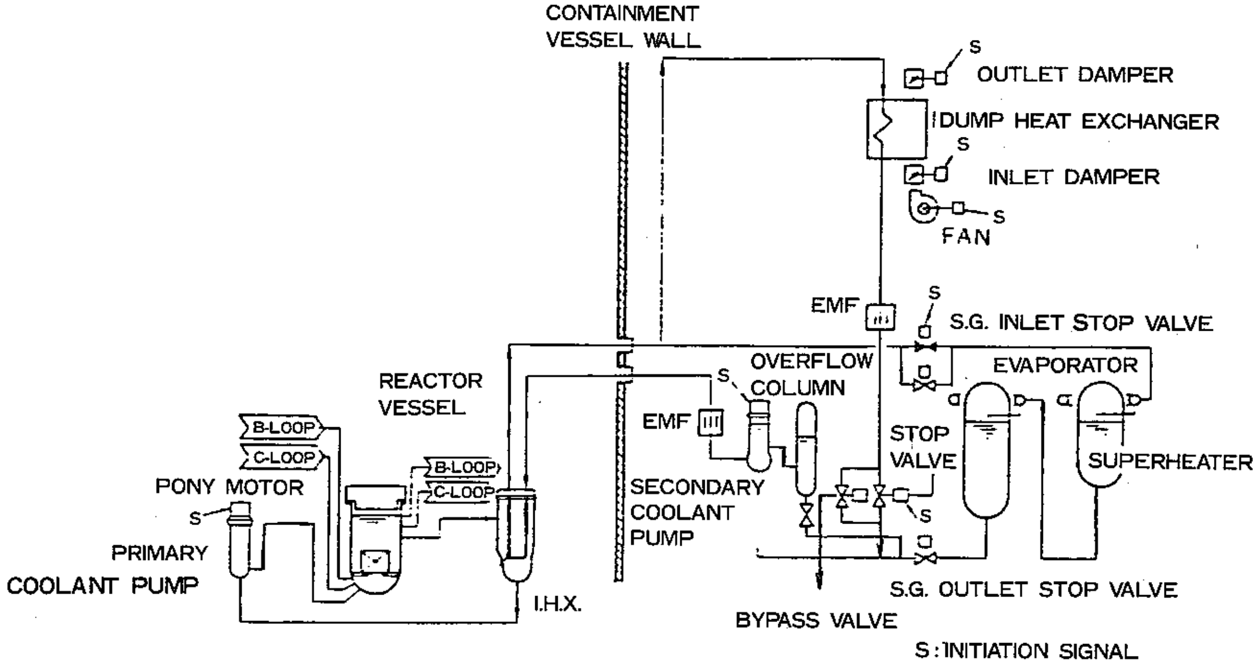
In abnormal transients and accidents, main sodium circulation pumps are tripped immediately after a reactor trip, and ACS are started to be in operation together with pony motor operations. Active components in ACS, pony motors of both primary and secondary circulation pumps are connected to the emergency electrical power source, and if only one loop of ACS is assumed to be operated, it has enough heat removal capacity (15 MW) prevent severe damages of fuels.

In addition to ACS, "MONJU" plant has a maintenance cooling system which remove decay heat during the plant maintenance time. The maximum heat removal capacity of it is 7 MW, and it has the similar flow diagram to that of the main cooling system other than being circulated by EMP.

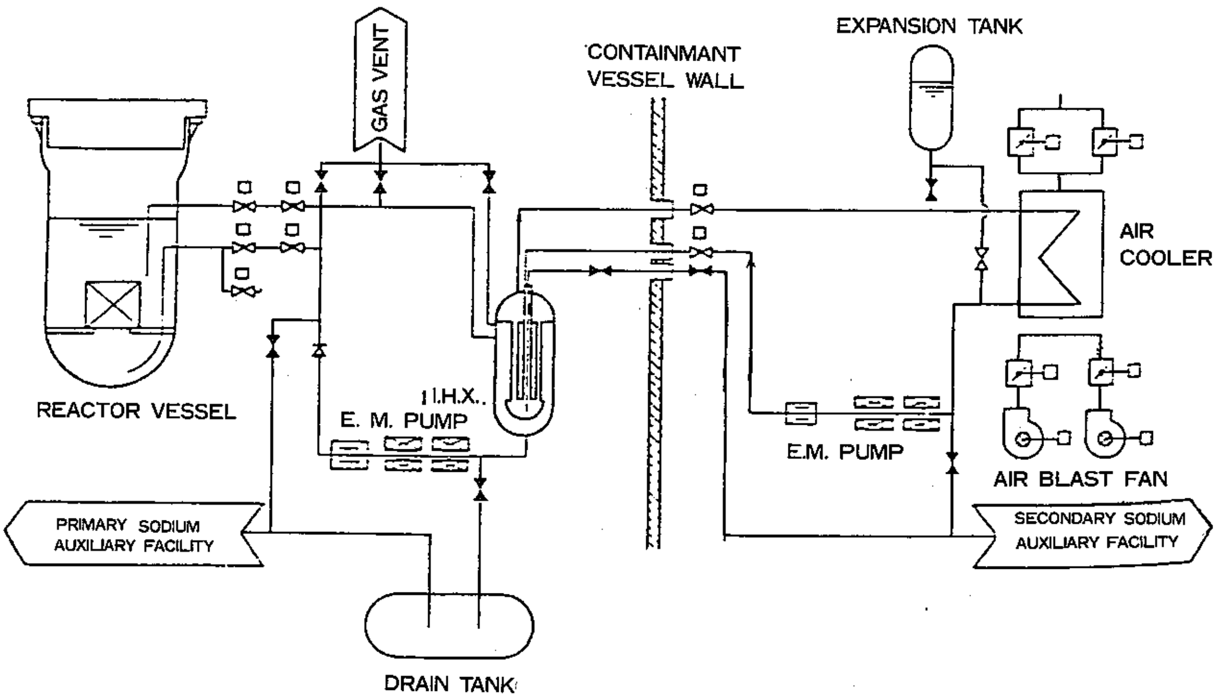
ACS has the heat removal capability by natural circulation by designing suitable vertical level difference among the core, IHX, and air cooler. To confirm natural convection heat removal capabilities, analytical tools of thermal hydraulics have been verified by making use of "JOYO" natural circulation test results.

It has been concluded by reliability analyses on "MONJU" SHRS that unreliability is an order of 10^{-8}/ry when including effects due to the maintenance cooling system, and based on this results, it is considered to be adequate that postulated events to reach core meltdown initiated from loss of heat sinks are concluded to be BDBA.

In order to confirm heat removal capability of ACS, system performance test is being carried out at PNC O-arai Engineering Center as shown in Figure 2.7.

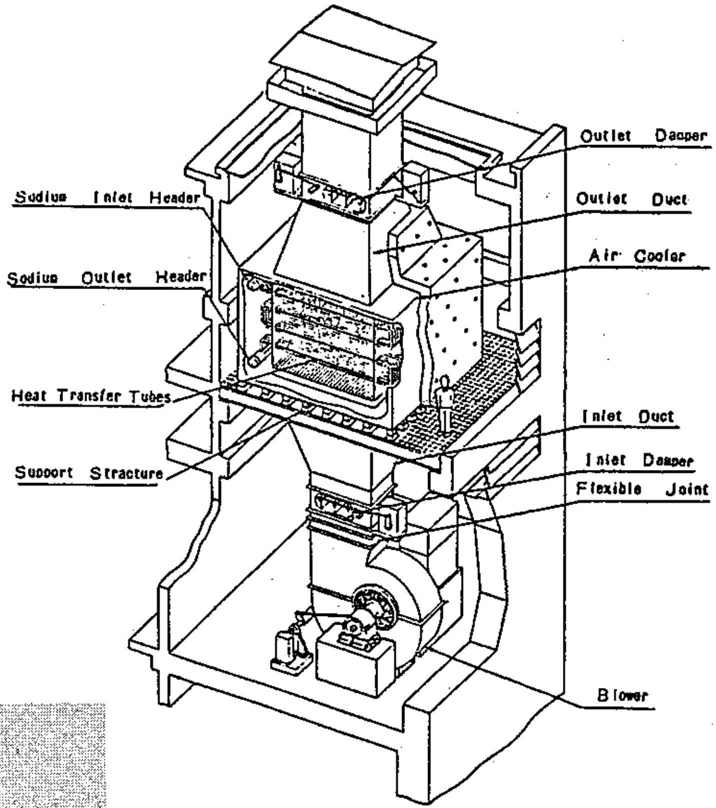


(A) Auxiliary cooling system flowdiagram

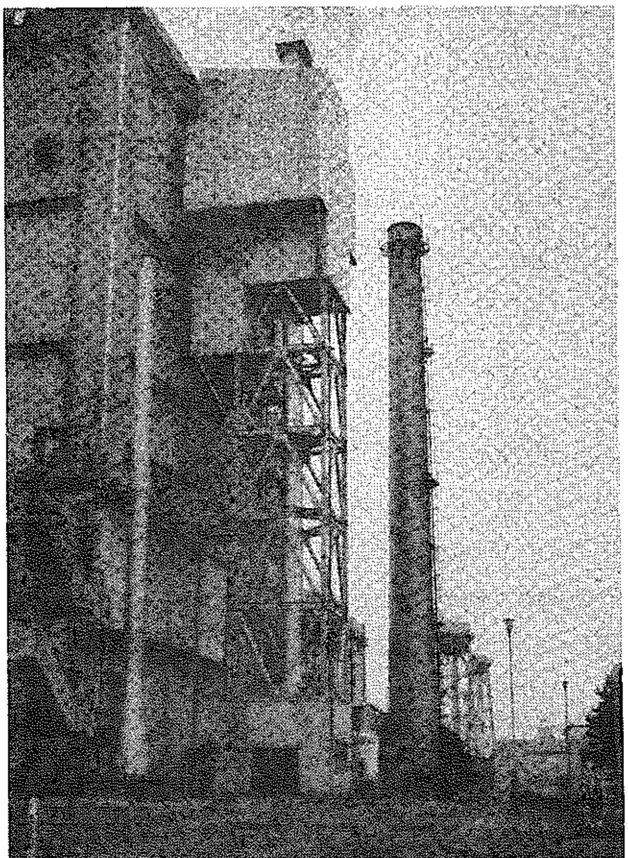


(B) Maintenance cooling system flowdiagram

Figure 2·6 MONJU SHRS flow diagram



Air cooler of MONJU ACS



System tests of MONJU ACS
in PNC Oarai Eng'g Center

Figure 2·7 System performance test of MONJU ACS

2.2.10 Evaluation of Lower Probability Events

According to the "Safety Evaluation Policy for LMFBR's, appendix Section 5" mentioned before, it is necessary to make safety evaluation for the postulated events with lower probability and higher consequences than those accidents analyzed for the evaluation of the safety design features of the plant.

For responding to this requirement, PNC selected following three postulated events:

i) local faults in the core
   - selected from the view point of possibility of progression of local abnormal event in the core.

ii) loss of piping integrity in the primary heat transport system
   - selected from the view point of decrease of core cooling capability due to loss of piping integrity in the primary heat transport system.

iii) loss of reactivity control capability
   - selected from the view point of possibility of large core damage due to loss of reactivity control capability.

Evaluation of these events are as follows:

(1) local faults in the core

It is postulated that the local flow blockage of fuel assembly about two-thirds of the flow area in a fuel pin bundle has occurred or a fuel pin whose linear heat rate is higher than normal by factor of two for the length equivalent to about ten pellets is in advertently loaded to the center of the core.

As the result, rapid propagation of fuel pin failure does not occur and fuel pin failure is limited.

There is no release of radioactive materials to the public.

(2) loss of piping integrity in the primary heat transport system

It is postulated that the loss of piping integrity of primary heat transport system including the case of double ended rupture has occurred.

As the result, the number of fuel pin failure is evaluated about three percent of the fuel in the whole core.

It is, therefore, possible to cool the core without resulting in serious core damage and any radiation hazard to the public.

(3) loss of reactivity control capability

It is postulated that the reactor fails to scram when loss of flow occurred during full power operation by off-site power loss, or when transient overpower occurred during full power operation by abnormal insertion of reactivity due to continuous withdrawal of control rod.

As the result, maximum available work energy is 380 MJ in the former case.

It is secured that the reactor vessel and primary heat transport system have retaining capability of coolant for 500 MJ work energy.

As for post accident heat removal, it is able to cool the debris in reactor vessel by natural circulation.

Temperature rise and pressure increase in containment vessel do not exceed the design condition in the analysis of sodium fire due to ejected sodium to the operation floor space.

Evaluation of these postulated events is performed basically on nominal base not including safety margin considered in the accident analysis to evaluate the safety design features of the plant.

Public radiation dose for thyroid, whole body and bone surface, lung and liver by inhallation of pultonium is under permissible level mentioned in "Guidelines for Reactor Siting" and "Guidelines for the Maximum Permissible Values for Pultonium Dose".

So, it is assured that the release of radioactive materials to the environment is appropriately suppressed.

3. Safety Design of DFBR

3.1 Present Status on the Design Study of DFBR

3.1.1 Introduction

Design studies of DFBR have been carried out by the Power Reactor and Nuclear Fuel Development Corporation (PNC) and the Private Utilities with each other's focusing on different points, in parallel with development of the experimental and prototype reactors. PNC is focusing mainly on identification of R&D programme in accordance with the role of PNC indicated in "the Long Term Programme of Development and Use of Nuclear Energy" issued by the Japan Atomic Energy Commission in 1982. The utilities, from their own standpoints as a user, is focusing mainly on proper incorporation of users' needs into the design; such needs as achievement of reliability and availability goals and improvement of operability and maintainability. The design studies are carried out basically based on the loop-type reactor concept that has been developed in our country upto the prototype reactor. Since some attractive points of the pool-type reactor concept as a large size reactor are recognized, the utilities also study the pool-type concept aiming to maintain flexibility of reactor type selection.

The studies have been promoted under close cooperation of PNC and the utilities including many tasks that extracted the R&D programme and developed technical bases needed for selection of specifications of the basic design planned in the near future.

3.1.2 Design Study of DFBR

PNC carried out the preliminary design for four years since 1975 that was followed by the first conceptual design for the next three years. Single set of overall plant concept with identified items of R&D has been extracted in the studies. PNC is carrying out the second conceptual design now.

The utilities have carried out the conceptual design studies under cooperation of all of the private ten electric power companies since 1978. The study consists of three phases. Regarding the study of the loop-type reactor, key concepts of the design were selected in the phase I and the design was reviewed mainly from the standpoint of operation and maintenance in the phase II. Design specifications are to be established on the basis of the further design of the total system and components in the phase III from FY 1981 through 1983. Regarding the study of the pool-type reactor, a preliminary concept definition was carried out studying the design of the Super-Phenix in the phase I and it was reviewed mainly from the standpoint of seismic characteristics in the phase II. The key subsystems of the pool-type reactor have been designed in the phase III in parallel with model tests studied by the Central Research Institute of Electric Power Industries (CRIEPI).

Some of key specifications on the loop type reactor examined and extracted by PNC and the utilities so far are not the same as those of the prototype reactor "MONJU" as shown in Figure 3.1. The key specifications such as reactor-type (loop vs. pool), the number of loops and reactor outlet temperature will be selected in accordance with national needs for the DFBR as the study and discussion on owner forming proceed. Followings are the figures and table reported in the studies;

Figure 3.2	Schematic Flow Diagram of Loop-Type Plant (PNC)
Figure 3.3	Primary System Arrangement in Containment Vessel (PNC)
Figure 3.4	Schematic Flow Diagram of Loop-Type Plant (Utilities)
Figure 3.5	Reactor Structures of Loop-Type Plant (Utilities)
Figure 3.6	Conceptual Drawing of Pool-Type Reactor Configuration (Utilities)
Table 3.1	Key Parameters of Pool-Type Plant.

The configuration of systems and structures shown above are not fixed. Especially, the structure of the pool-type reactor will be established on the basis of overall concept synthesis through many studies conducted now besides the design study.

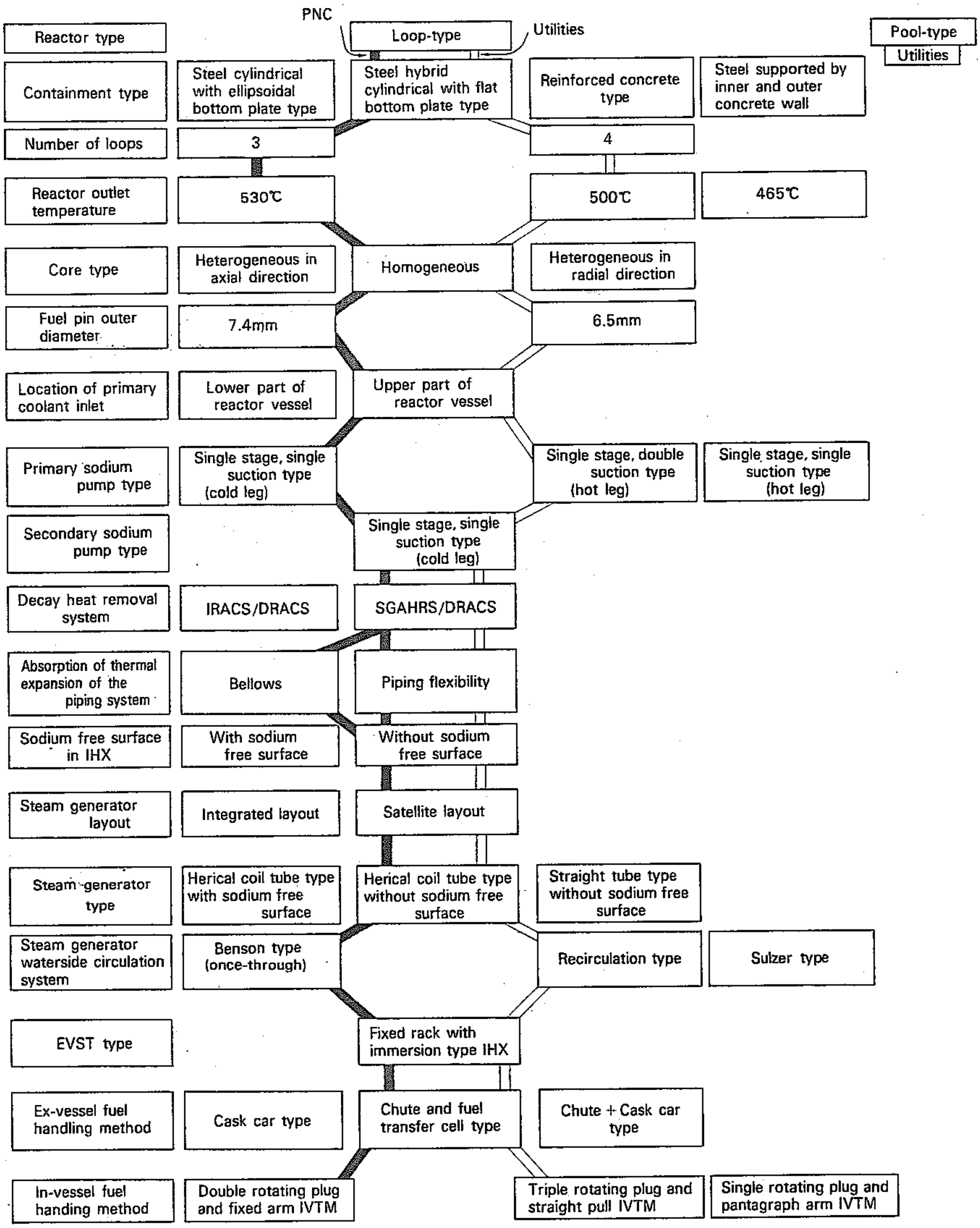


Figure 3.1 Main trade-off study on loop-type plant

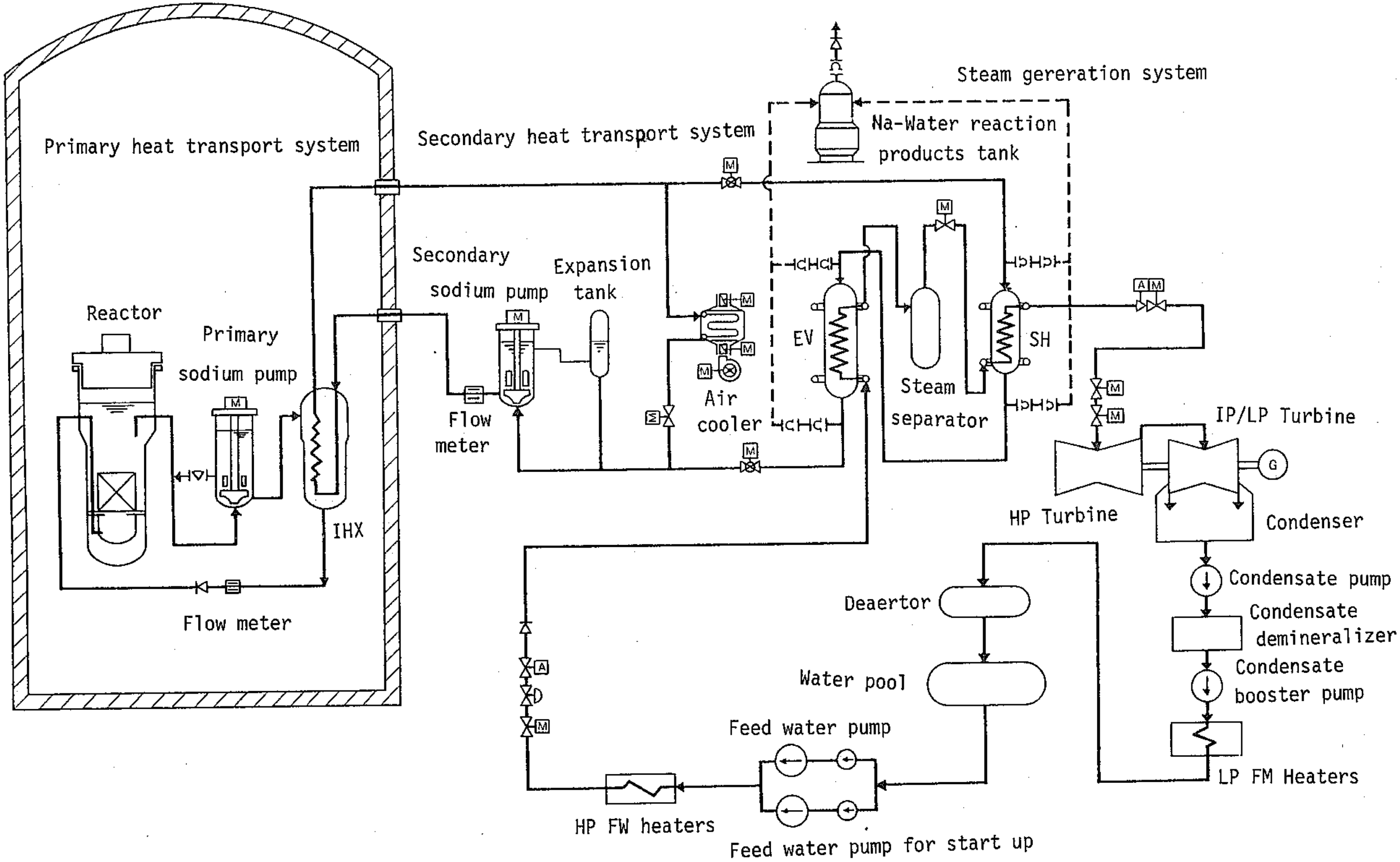


Figure 3.2 Schematic flow diagram of loop type reactor (PNC)

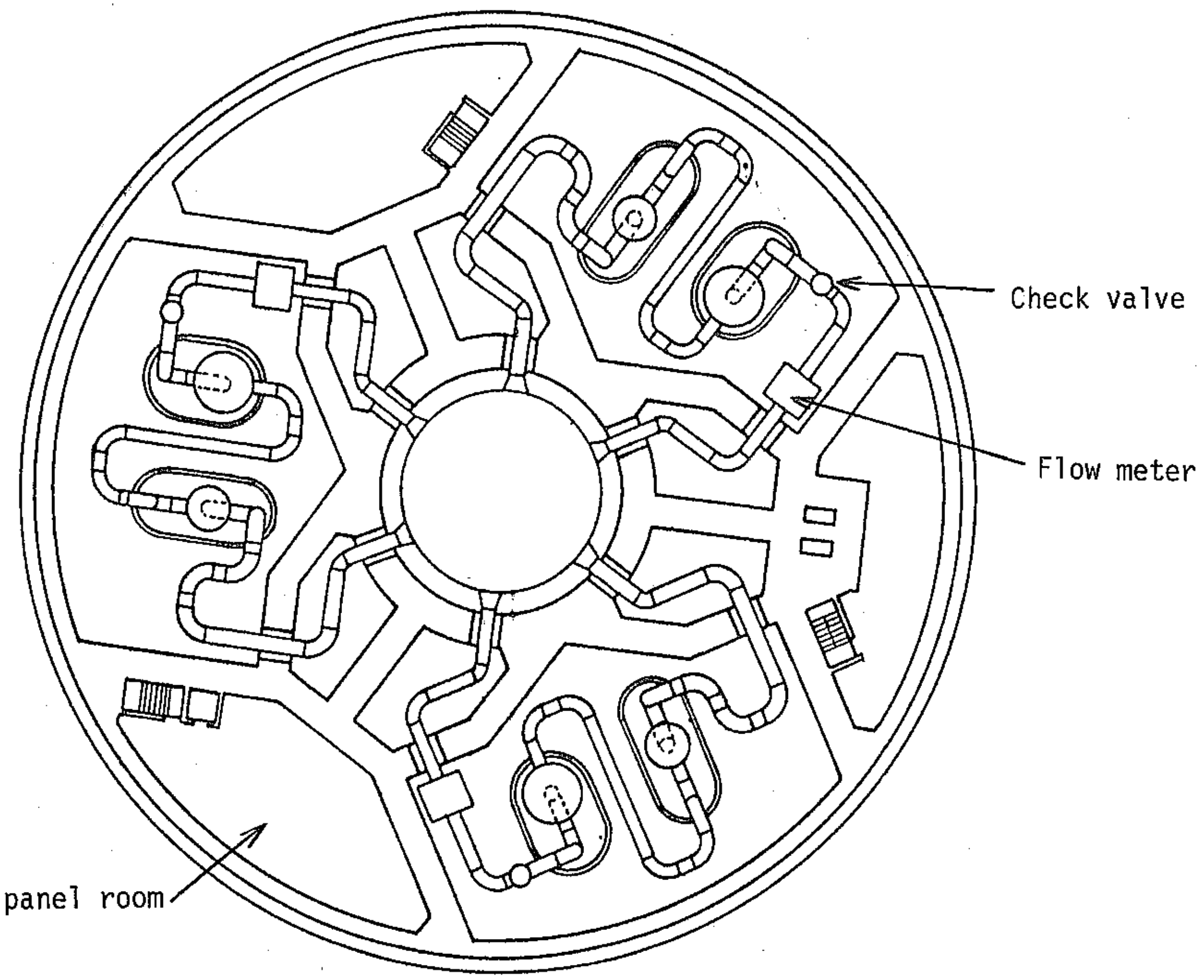
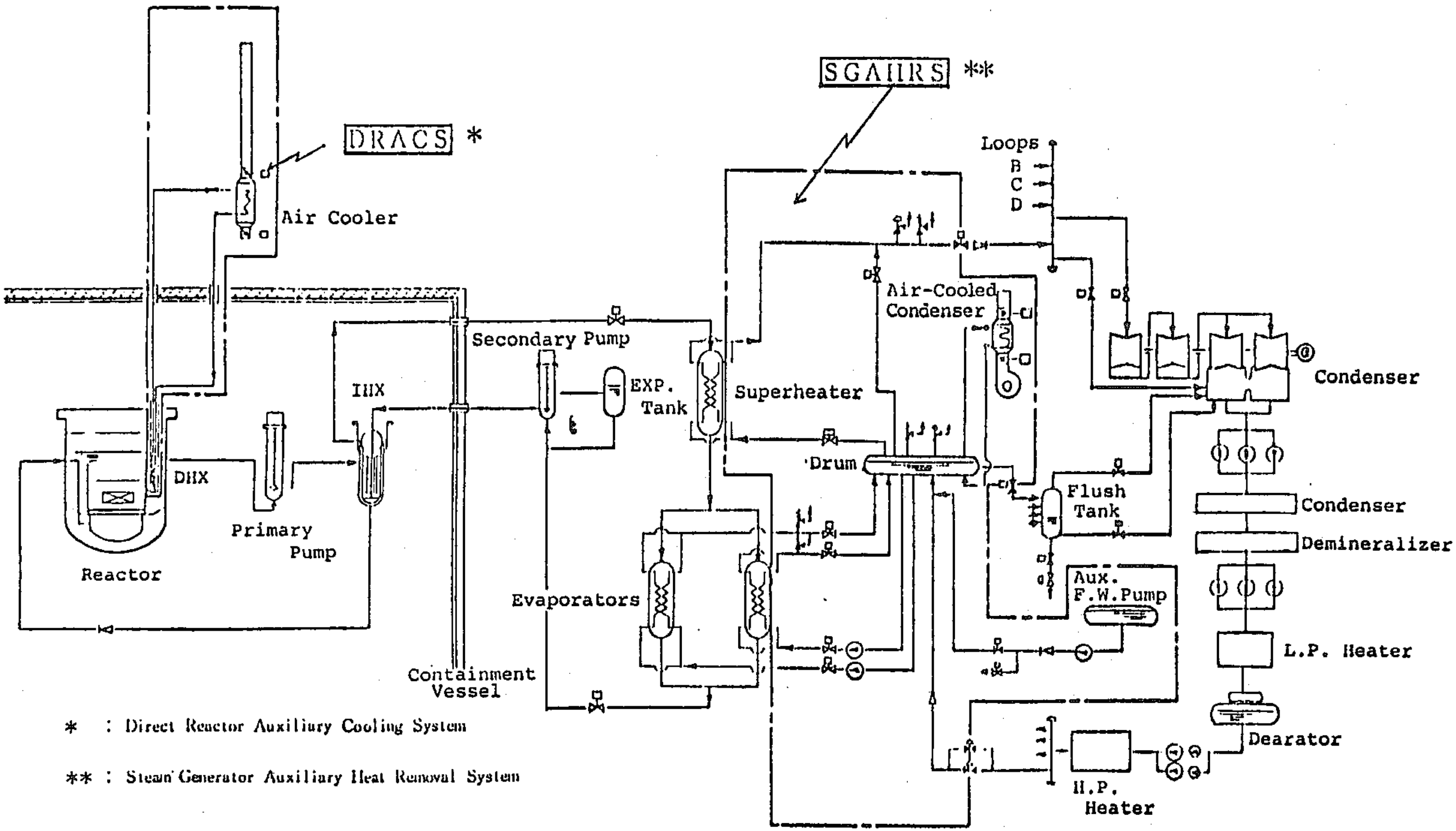


Figure 3.3 Primary system arrangement in containment vessel FL-17.0M (PNC)



* : Direct Reactor Auxiliary Cooling System

** : Steam Generator Auxiliary Heat Removal System

Figure 3.4 Schematic flow diagram of loop-type plant(Utilities)

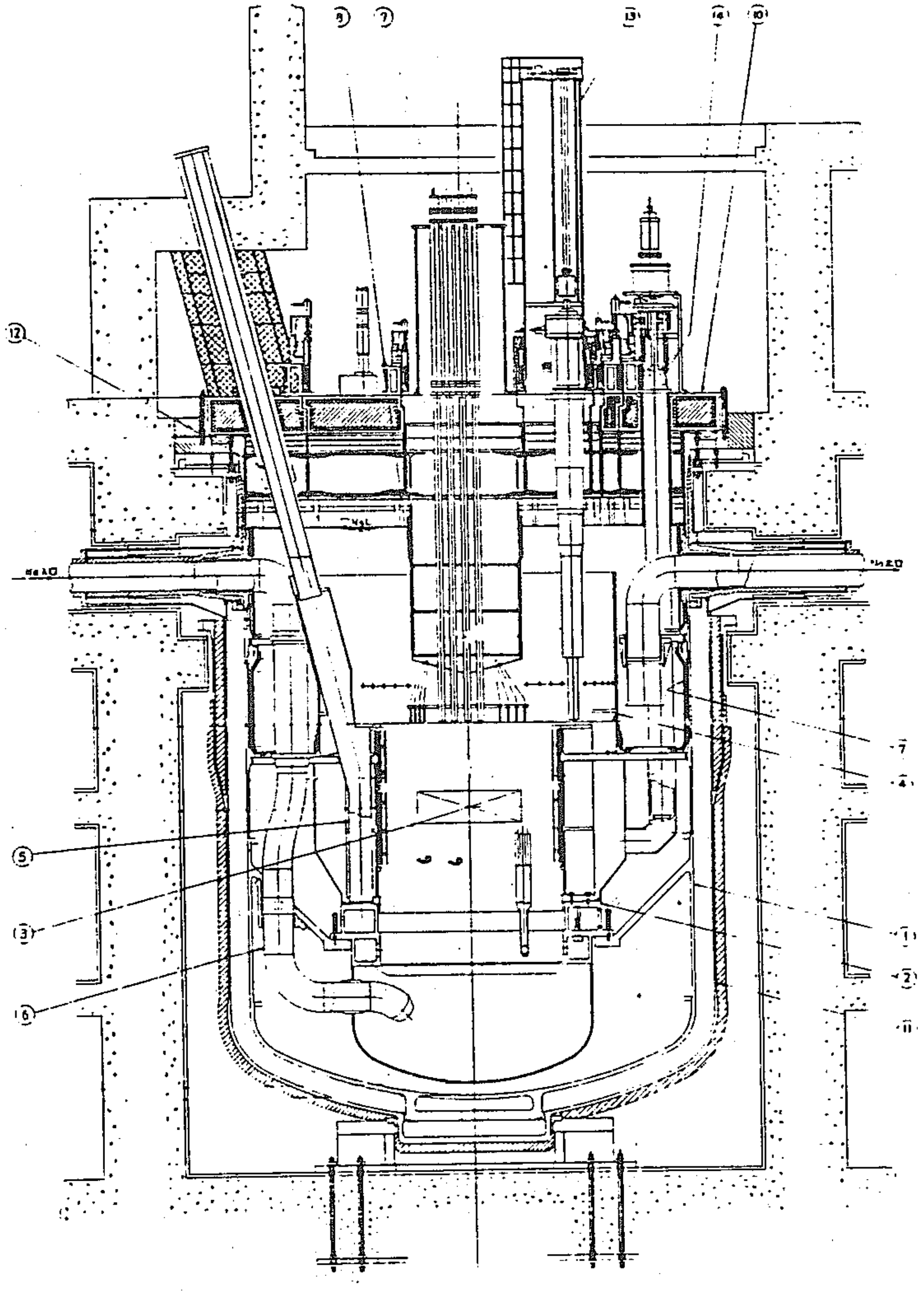


Figure 3.5—(1) Reactor structures(Vertical) of loop-type plant(Utilities)

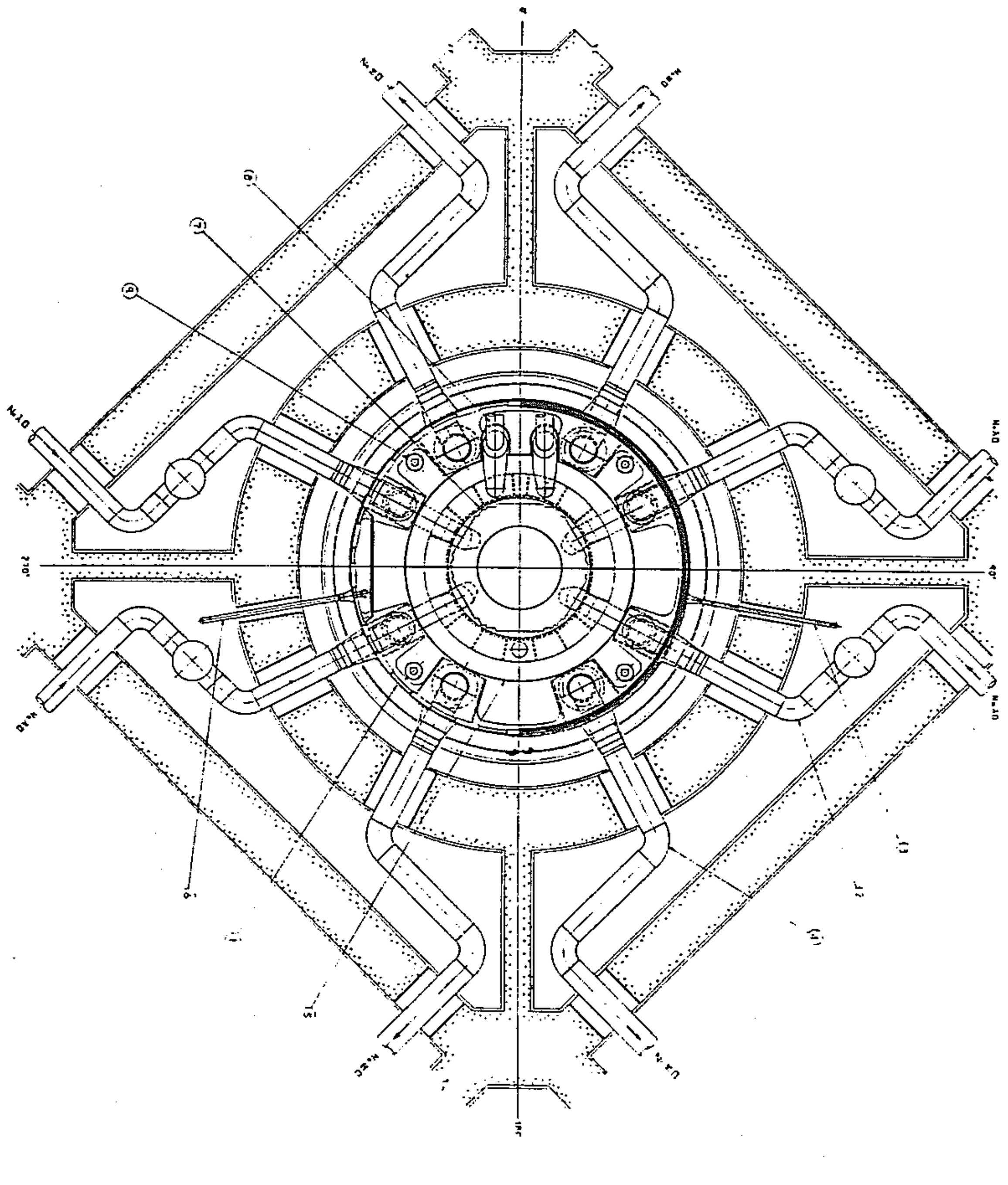


Figure 3.5—(2) Reactor structures(Horizontal) of loop-type plant(Utilities)

No. Item

1. Main vessel
2. Safety vessel
3. Roof slab
4. Rotating plug
5. Core internals
6. Core support structure
7. Thermal shield (against hot-shock)
8. Thermal shield (horizontal)
9. Thermal shield (radial)
10. Neutron shield
11. Upper core structure
12. Primary sodium pump
13. Intermediate heat exchanger(IHX)
14. Ex-vessel fuel transfer machine (EVTM)
15. In-vessel fuel transfer machine (IVTM)
16. Ex-vessel storage tank(EVST)
17. Fuel handling machine
18. Rotating plug
19. Reactor vessel lateral seismic restraint.

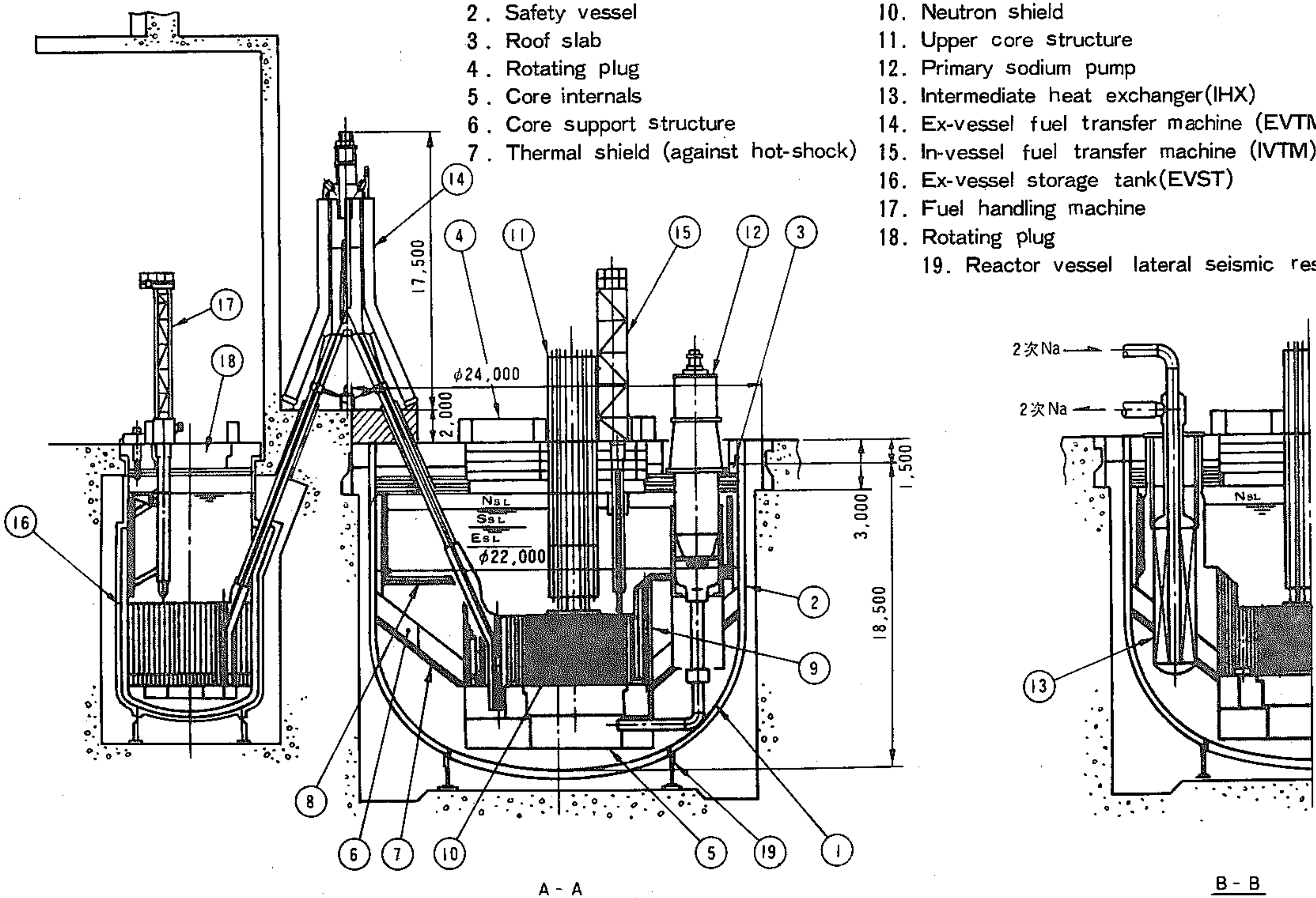


Figure 3·6 Conceptual drawing of pool-type reactor configuration

Table 3·1 Key parameters of pool-type plant

Item	Parameter
Reactor thermal output (MWt)	2,600
Reactor outlet/inlet temperature (°C)	500/365
Number of primary sodium pumps	4
Number of intermediate heat exchangers	6
Inner diameter of main vessel	22
Main vessel wall thickness at shoulder	75
Containment type	Reinforced concrete, cylindrical
Turbine main steam condition Temperature/Pressure (°C/kg/cm²g)	450/102

3.2 Present Status on the Safety Design of DFBR

In executing design studies of DFBR, the assurance of the plant safety occupies the important position of basic design policies. It is aimed as the target to establish licensable designs in the structures, systems and component and to improve the economy of the plant. However, as the safety design criteria forcused on DFBR have not been established currently in Japan, it is necessary that the plant designer's policy on the safety are well prepared through design studies of DFBR, and that technological problems on those are well extracted, and that R&D items to be required against those are clarified. Therefore, we are devoting our effort to make it clear through conceptual design studies what appropriate design criteria are to be applied in executing design studies hereafter, in order to establish the licensability, to improve the capacity factor, the operability and the maintainability of the plant, and to reduce the construction costs. This work, not aiming at establishing the safety design guideline itself of a LMFBR, is mainly forcused on considering what design measures should be taken based on the proper features of a LMFBR when LWR safety design guidelines are applied to DFBR.

Therefore, this work are mainly to describe qualitatively the principal design philosophy explained in the section 3.2.1, plant level safety design policies, and subsystem level safety design policies, and after that to give the quantitative safety design conditions as its realization.

This work is being conducted parallel with executing component designs of the plant, and that to be applied for the loop plant will be completed in 1983. The present safety design policy should be revised by following licensing authorities' direction, and other countries' directions for developments. Moreover, within the safety design policies and conditions, particular requirements which have to be realized in the system and component designs should be revised through future design progresses in taking account of a possibility for the plant realization, reliability, and economical aspects.

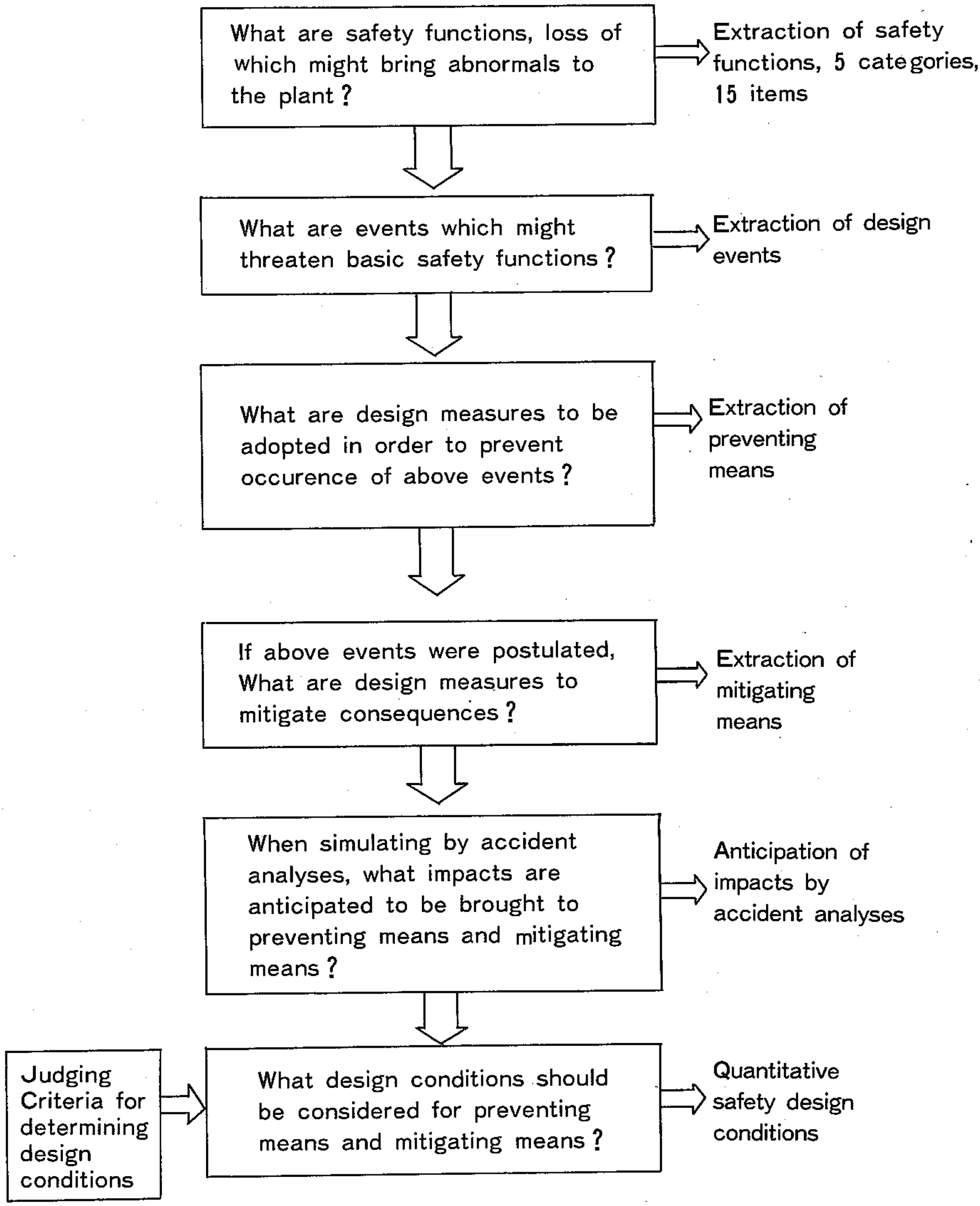
Work for quantifying safety design conditions shall supplement and complete the safety design policy based on quantitative design conditions which are supported by results of safety analyses obtained from safety functions of the plant. A fundamental approach of this work development is illustrated in Figure 3.7, and the safety design conditions have been quantified by starting from extracting the safety functions of the plant, loss of which might bring abnormalies, and by investigating, what are necessary design means to prevent loss of concerned safety functions ?, and what are necessary mitigating means when postulated loss of concerned safety functions ?

Safety functions which should be remarked (shown in Figure 3.8 have been defined by reflecting them to present design philosophy and experience of safety evaluation. That is, at least, design conditions for the system and components important to safety are extracted by based on the engineering judgement that these could be determined by the safety functions. In doing this, the judgement criteria indicating "reference value" have also been determined as its bases.

Many safety conditions have been decided by engineering judgements based on safety analyses conducted by assuming tentative design parameters, and on extrapolating technological experiences obtained from "MONJU", and soon. Hereafter, these should be confirmed by detailed safety analyses and evaluations for DFBR. The position of current safety conditions is shown in Figure 3.9, and an example of safety design conditions which has been obtained so far is simply shown in Figure 3.10.

In the following sections, some topical features on the safety design of DFBR which have been mainly obtained through utilities' sponsored design studies are explained.

Figure3•7 The extraction method of quantitative safety design conditions applied to DFBR



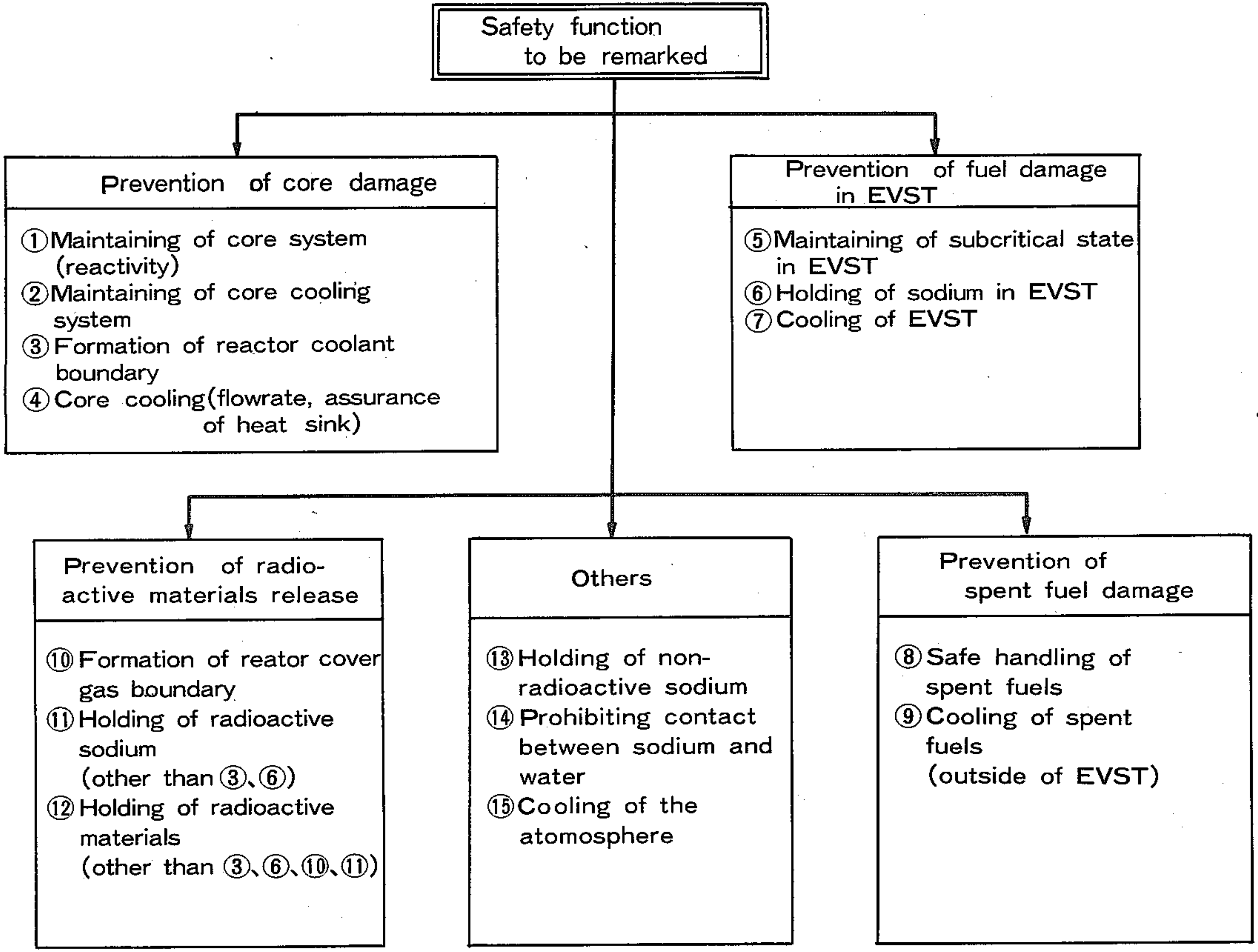


Figure 3·8 15 items of safety functions

Figure 3•9 The position of current safety design conditions

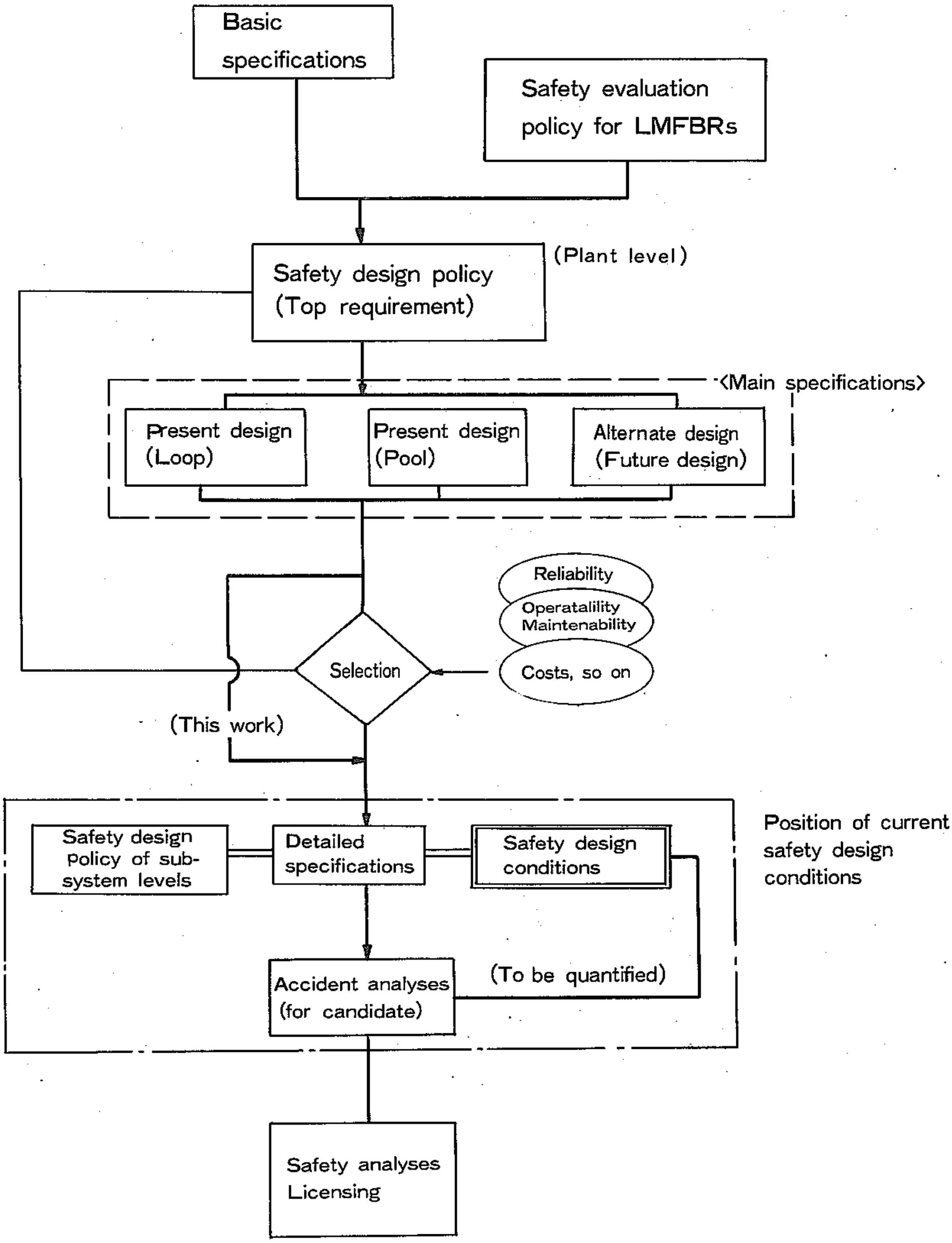


Figure 3·10 An example of safety design conditions

④Maintaining of core cooling — Principal safety functions

Safety functions

Reactor shut-down......
Maintaining heat transport capability

Preventive measures

Mitigating measures

- Core flowrate
- Secondary flowrate
- Heat removal function due to water-steam system
- Long term core cooling

Safety design conditions

- Reactor shut-down system scram performance
- Primary mainpump GD^2
- Primary pony motor lower speed operation function,
- D/G start-up performance; lessthan 12 sec.(see Fig.a)
- Check valve closing characteristics
- Secondary ponymotor lower speed operation
- SGAHRS heat removal function
  DRACS heat removal function (see Fig. b)

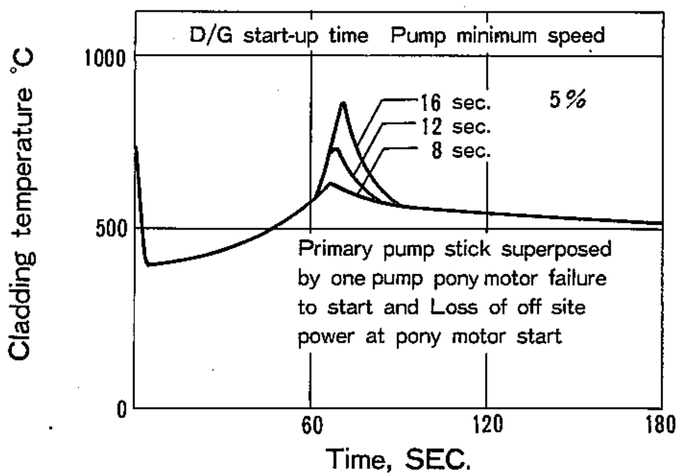


Fig. a D/G start-up performance

D/G start-up time (sec.)	Damage Factor
8	≪0.1
12	< 0.1
16	> 0.1

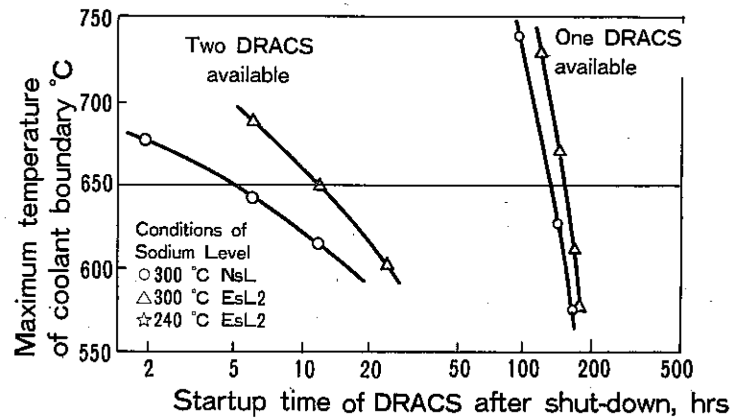


Fig. b Maximum coolant boundary temperature vs. Startup time of DRACS

3.2.1 Principal Safety Design Philosophy

Safety philosophy in the background of the principal safety policies of DFBR are as follows;

(1) Safety design policies are established by fundamentally based on LWR's safety design phylosophy, and by adding special features of LMFBR to them, and by emphasizing the prevention of proper accidents anticipated in LMFBR and the mitigation of their consequences.

(2) Prevention of accident occurrence is required as having the first importance, and in this policy, enhancing the inherent safety nature of a plant and simpler design options are considered to be significant. Mitigating means against consequences of an accident are considered to restraint accident developments in its location and in its initiating phase as much as possible.

(3) If excessive safety margins were provided because of uncertainties of accidental phenomena, it shall not result in damaging plant economics, reliability of systems and component under normal operation, easiness of the plant operation and maintenance. Against uncertain accidental phenomena design basis accidents should be clearly defined and in the initial stage of the accident, both preventing measures and mitigating measures should be thoroughly provided.

(4) Against BDBA, a great deal of effort shall be made to show that the inherent safety nature (self-restraint feature) could well restraint and mitigate their consequences.

(5) The safety level of DFBR shall be targeted to be the same order of magnitude as LWR under currently licensing application.

Based on above safety philosophy, policies on major accidents have been determined and design means have been prepared against them.

3.2.2 Classification of Design Conditions

A nuclear power plant is designed by considering frequent events in occurrence, and also infrequent events which are not considered to be occurred, but have to be postulated from a view of evaluating safety of the plant. In designing a plant while attaining its safety and economic to desired levels, it is reasonable that larger design margins are provided to higher frequent events in order to limit influences due to such events on the plant as smaller as possible. "Classification of design conditions" are introduced based on this basic design policy, and the structural designs for each component of the plant are executed based on the classification of design conditions.

In this classification, anticipated events in the plant are categorized into four groups, and artificially superposing another event like, a single failure to them decides "Design conditions". The levels of safety and integrity of the systems and components to be attained in response to these "Design conditions" shall be decided depending on frequencys of each design condition, taking account of a balance between safety and economics, and the importance of components grade.

Table 3.2 shows an example of the classified design conditions.

Table 3·2 The classification of design conditions

Plant condition \ Superposition	None	Single failure SS	Single failure PS	S 1	S 2	L+SS
I (Normal)	I		II	III	IV	
II (Upset)	II	III		III	IV	
III (Emergency)	III	IV		III	IV	
IV (Faulted)				IV(*)		IV

SS : Single failure of safety systems

PS : Single failure of systems important to safety not included in the above (the process system)

L : not depending on external power source

(*) : more than 0.1 year

3.2.3 Basic Principles of Separation Criteria

Structures, systems and components important to safety shall be separated into more than two groups with complete independence in principle. Those in the utilities' DFBR, as it has 4 loops, are separated into independent four systems as far as technologically attained in accordance with the number of loops of auxiliary core cooling systems. That is, (2+2) separation principle have been adopted.

(1) Including all of related structures and systems, single failure of any active component shall not bring that more than two systems important to safety such as auxiliary core cooling systems can not be operated.

(2) Against postulating sodium leaks in a loop, fire protection walls shall be provided to assure that decay heat removal capabilities of other loops are not lost by fire, and they shall be physically separated and independent in an accident.

(3) Against considering cable fires, and so on, it shall not be occurred that safety important systems of other auxiliary core cooling systems than concerned can not be operated by fire.

In order to realize the plant design based on this principle, design considerations such as deciding necessary heat removal capacities of the system have been executed.

The separation principle of the plant is summarized in Figure 3.11.

Hereafter, judging appropriately separation principles related to the number of loops will be required as a whole by taking account of recent practices in LWR and conducting trade-offs on the plant economy.

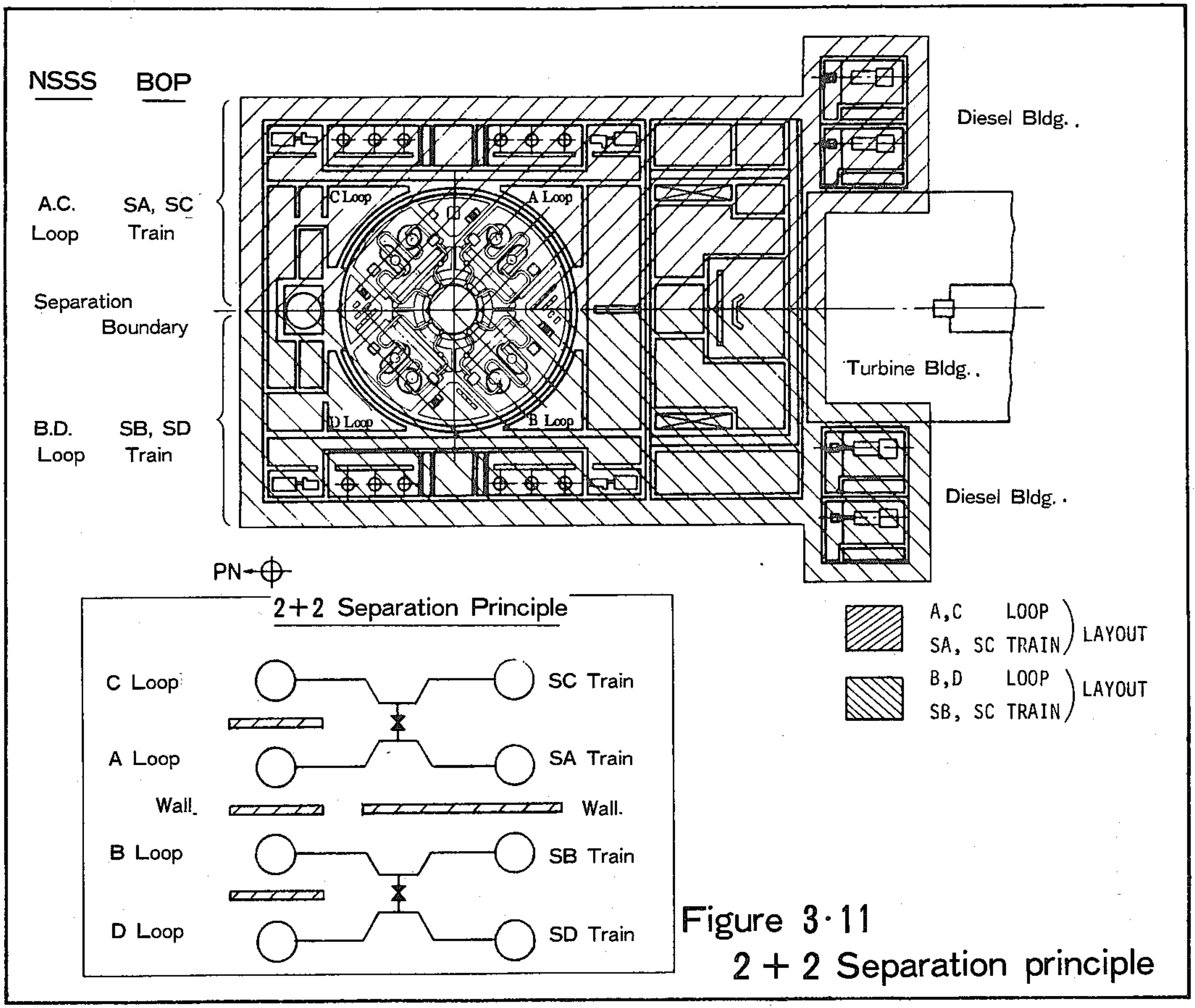


Figure 3·11
2 + 2 Separation principle

3.2.4 Aseismic Design

Basic philosophy of aseismic design for large scale LMFBR in Japan is just the same to that for "MONJU". However the enveloping treatment for the construction sites whose shear wave velocities ranges from 500 to 2,000 m/s is taken into consideration because its site has not yet been announced.

The standard design seismic wave for high seismic zone (for the use of light water reactor) has been temporally used also for the large LMFBR. Resulted floor response is so high that highly rigid structures have inevitably been selected for the components design, resulting the reasonable problems of mismutch against thermal stress inherent in LMFBR.

There is, different from the above conceptual design studies, another project for seismic feasibility study that has been being performed for pool type structure as the joint program between the CRIEPI (the central research institute of electric power industry) and the nuclear power plant vendors. This program includes several basic vibration tests by the use of shaking tables or actuators, to get the data mainly for liquid-elastic vibration properties between liquid sodium and internal structures like partition redan, core support structure and so on, and also includes shaking test of 1/10 scale reactor assembly models for the seismic demonstrations purpose.

The main problems on large scale LMFBR design are understood as follows; to lower the floor response (1) by improving the severe design conditions for ground motion which inevitably imposed to cover the design condition of whole Japan Island, (2) by improving big margins included in the treatment of the ground and (3) by using more reasonable damping values for reactor components, and to select more proper structures to LMFBR by choosing more flexible designs.

3.2.5 In-Service Inspection (ISI)

There is not existed in Japan such a published ISI code system for LMFBR as ASME Sec. XI Div. 3, so in the Utilities study program on DFBR the following approaches has been performed;

(1) To study the history and background of the ASME Sec. XI Div. 3, and

(2) To comparatively evaluate the effects of two existing standards (the ASME code and JEAC-4205: Japanese ISI code for LWR based on volumetric examination) to the reactor pressure vessel and to the primary pipings of DFBR.

It is concluded from the above studies that the ASME Sec. XI Div. 3 taken the characteristics of LMFBR into account, in principle, looks favourable and applicable from the viewpoint of utilities applications. The practical applications of the ASME code is going to be studied upon the DFBR depending on progress of the design.

Problems resulted from the preliminary studies are as follows;

(1) In order to reduce the needs for ISI requirements, large scale forging technology becomes necessary with the big investment for the forging facilities in the future.

(2) There are many difficulties in applying the code strictly because of the poor accessibility and technical matters. Examples are core internals, part of reactor vessel and snubbers for primary piping.

(3) Calculated radiation level around the primary components of DFBR is so high comparing to that of LWR's that rather high radiation exposure will be inevitable for the maintenance.

Resulting from the above preliminary evaluations, it turns to be the problem how much the ASME requirements can be accommodated as the short target for the application, and how much ISI requirements can be reduced by design consideration. In the future, proper philosophy and code for the domestic use of ISI should be developed by taking the characteristics of LMFBR into consideration.

3.2.6 Loss of Piping Integrity in Primary Heat Transport System and Sodium Fire

Similarly to "MONJU" plant, accidents of cracked leakage opening (1/4 Dt) in primary piping system have been postulated for design basis. A large scale rupture including double-ended gillotine rupture is categorized as beyond design basis accident.

The analytical evaluation has revealed that, at a accident of 1/4 Dt rupture, sodium leakage quantity would reach a maximum of 410 ton, and at such a time, the effect of the leakage detector and the core cooling behavior would be same as those in case of "MONJU" plant.

As to a sodium fire, the limit conditions from design are satisfied for the primary cooling loop room concrete and the reactor containment vessel structure as will be described in 3.2.7.

At a large scale rupture, the fuel cladding may possibly dry out as transient phenomenon of the core. With the conservative evaluation applied in the LWR site suitability, the fuel failure fraction would reach 30 to 50% of the whole core. In contract, when this large scale piping rupture is handled with LP/HC Event that uses the actuality evaluation, the fuel fa-lure would not occur.

3.2.7 Design Condition of Reactor Containment Vessel

The design condition setup for the reactor containment vessel is the same as those for "MONJU" plant stated in 2.2.5. The results of analysis and evaluation are briefly stated below:

(1) Integrity of reactor containment vessel at "loss of piping integrity" of design basis scale

The internal pressure of the reactor containment vessel rises to 0.02 kg/cm^{2}g at a sodium leakage (about 243 kg/s for initial 300 seconds, then 90 kg/s followed by gradual drop, totaling to 410 tons), and the highest temperature on the wall of the reactor containment vessel is about 41°C. The building concrete structure does not exceed 90°C. Thus, the design condition for the reactor containment vessel (internal pressure: 0.3 kg/cm^2g, highest wall temperature: 100°C) and the concrete structure temperature limit value (the highest at the surface: 175°C, average: 75°C, and moment-generating type temperature difference: 30°C) are satisfied.

(2) Integrity of reactor containment vessel at HCDA

The events that may cause HCDA have not been studied yet, but the design evaluation has been carried out by assuming the work potential of fuel vapor to be 1000 MJ. In this case, the work done by this fuel vapor is 306 MJ, and at this time, the quantity of ejection sodium to the operating floor is only several hundred kilograms as the bolts are designed not to be deformed plastically in such stress level. Meanfile, the quantity of ejection sodium corresponding to the design condition of reactor containment vessel is 800 kg. In the case of 800 kg sodium being ejected, the atmospheric temperature in the reactor containment vessel is a maximum of 130°C and the wall temperature is a maximum of 50°C. The design condition, therefore, is satisfied enough at HCDA. The reactor building and containment vessel used for the above evaluation is shown in Figure 3.12.

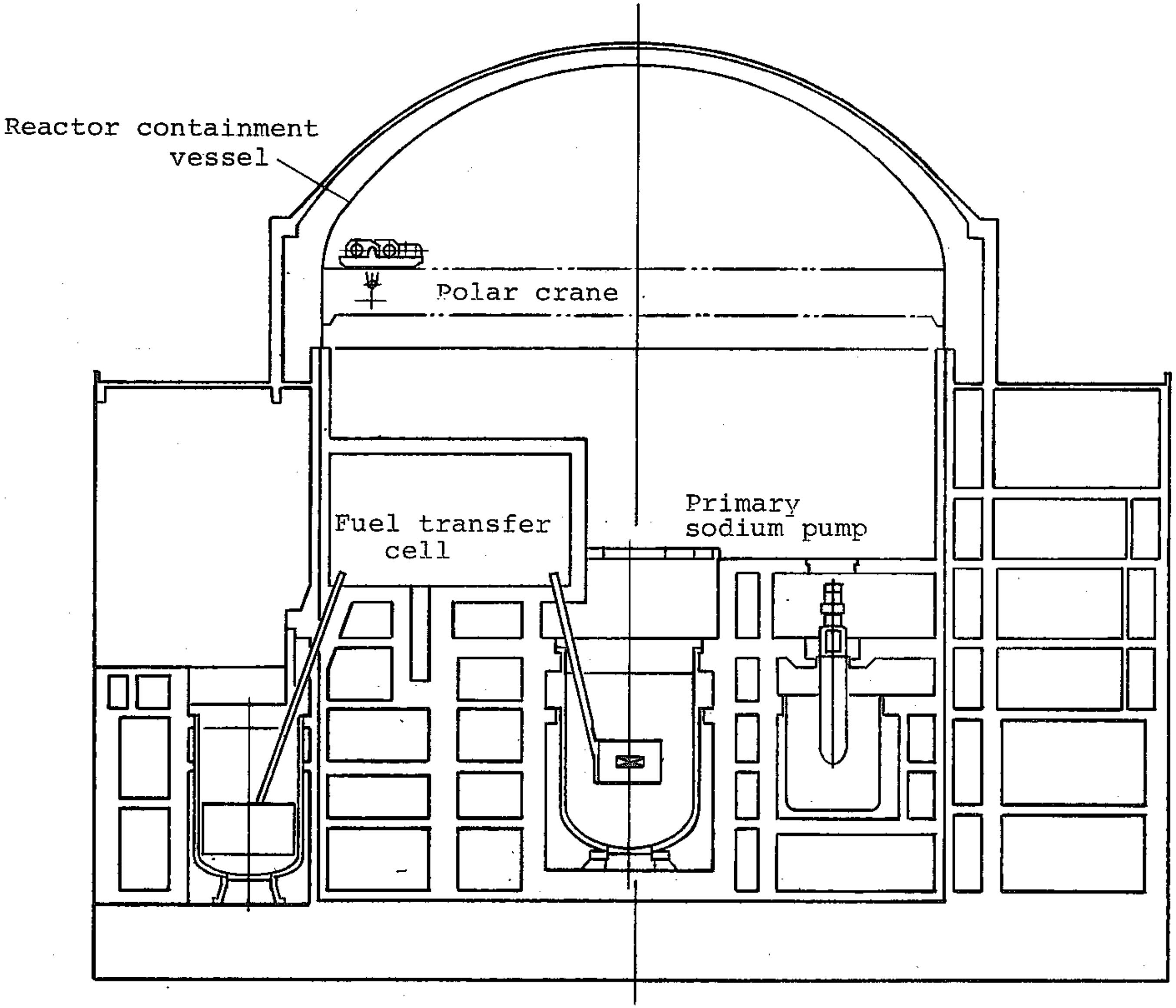


Figure 3.12 Reactor containment vessel concept of the demonstration LMFBR plant

3.2.8 Sodium Fire in the Secondary Sodium Coolant Circuits

Secondary sodium circuits do not contain any radioactive materials. Therefore, sodium leaks, even if they occurred, do not bring any radiation hazards to plant operators and the public, and never degrade decay heat removal capabilities indebted to their independent circuits organization. However, the higher circuit pressure and the larger amount of sodium than "MONJU" will result in larger leak rates of sodium, and may bring severe thermal effect to the building structures.

Design options when postulated 1/4 Dt break as the design condition on a secondary circuit piping leak are as follows;

(1) By providing connected tubes among rooms, leak sodium are collected and drained through these tubes into a leak jacket, or directly into the floor with liner plate which can effectively contain 650 m^3 leaked sodium. Figure 3.13 shows a leak jacket idea.

(2) In order to restraint the concrete temperature increase;

a) Rooms which have possibilities of spray sodium fire are provided with insulated materials like perlite concretes with thickness of more than 100 mm to the whole area of their floors, side walls and ceilings.

b) In case of using a leak jacket, more than 100 KW heat has to be removed from its surrounding. The leak jacket is insulated by 150 mm rocks wool.

c) The leak jacket has a lid whose openings for drained tubes are minimized to be less than 0.5% of the stored sodium surface.

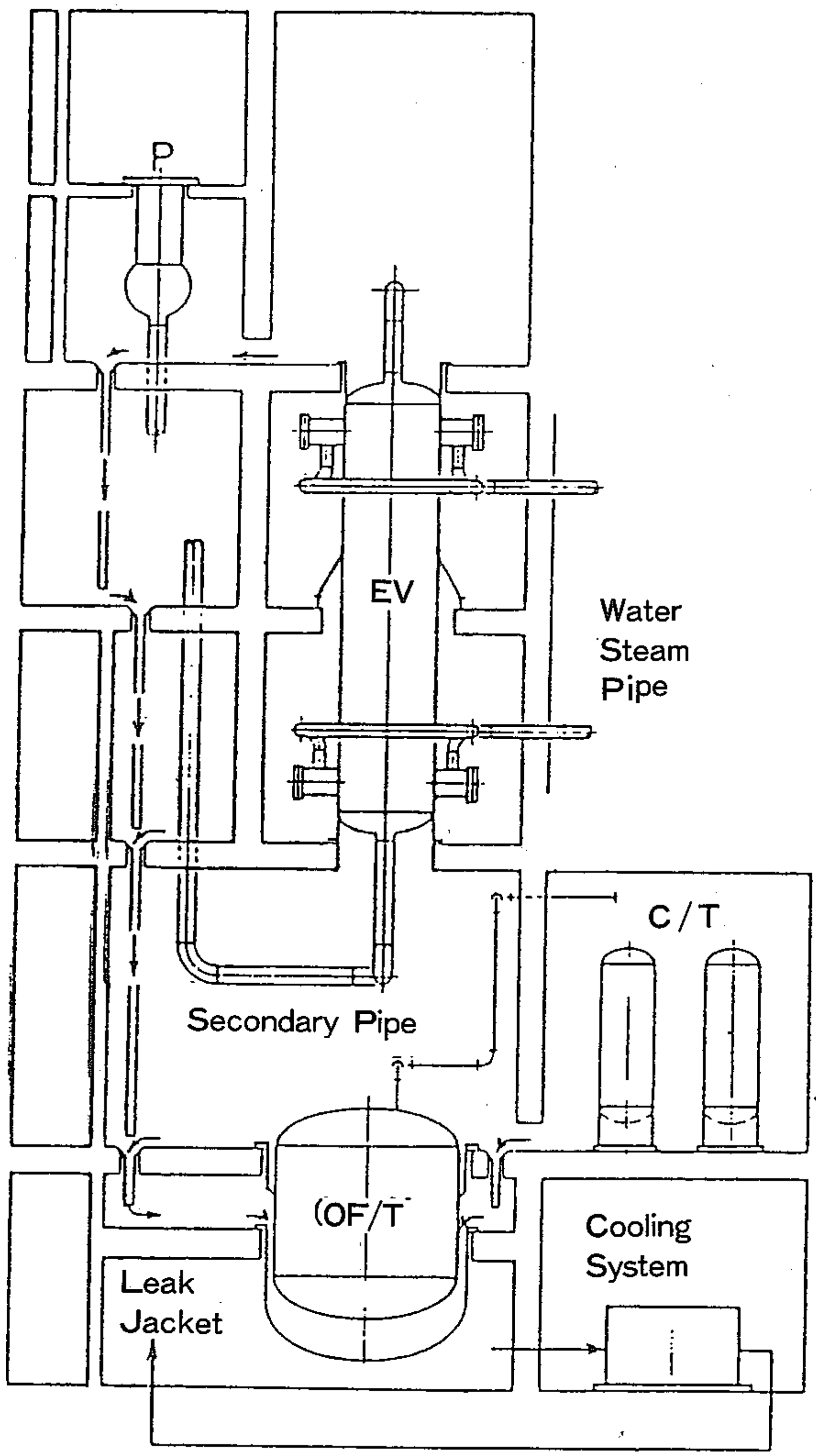


Figure 3・13 Protection against sodium leak in 2ry system

3.2.9 Provision for Sodium-Water Reaction in Steam Generator

A great deal of effort shall be made on quality control in the design and fabrication of steam generators in order to restrict the scale of initial tube failure within microscale of water leak. A leak detection system should be provided against micro leak for limiting water leak rate in small scale. Against an unlikely large water leak, design measures such as a sodium-water reaction product accommodating system are provided to ensure the integrity of intermediate heat transport system.

A maximum water leak postulated for the design measures is set according to the following safety phylosophy:

a) Initial failure: Double-ended guillotine failure of a single tube

b) Failure propagation: Experimental results to date reveals that no failure propagation has been observed in an initial large leak corresponding to double-ended guillotine failure of a single tube, and wastage damage to neighboring tubes is rather significant in small to medium scale of water leak.

Hence, the maximum leak rate in consideration of failure propagation is tentatively set to be less than the total leak rate in the postulated case of guillotine failure of a single tube at initial stage and (one or three) tubes at secondary.

Figure 3.14 presents a concept of design measures mentioned above.

Compared with the measures of the prototype reactor, it is notable that hydrogen detector in the cover gas region is not needed and rupture disks immersed in sodium are needed due to the steam generator type without sodium free surface inside. Though analytical results of water leak accident show the different situation to some extent from that in the prototype, it has been ascertained that there is no problem associated with integrity of intermediate heat transport system. Future work is desired to reduce a design margin included in a sodium-water reaction product

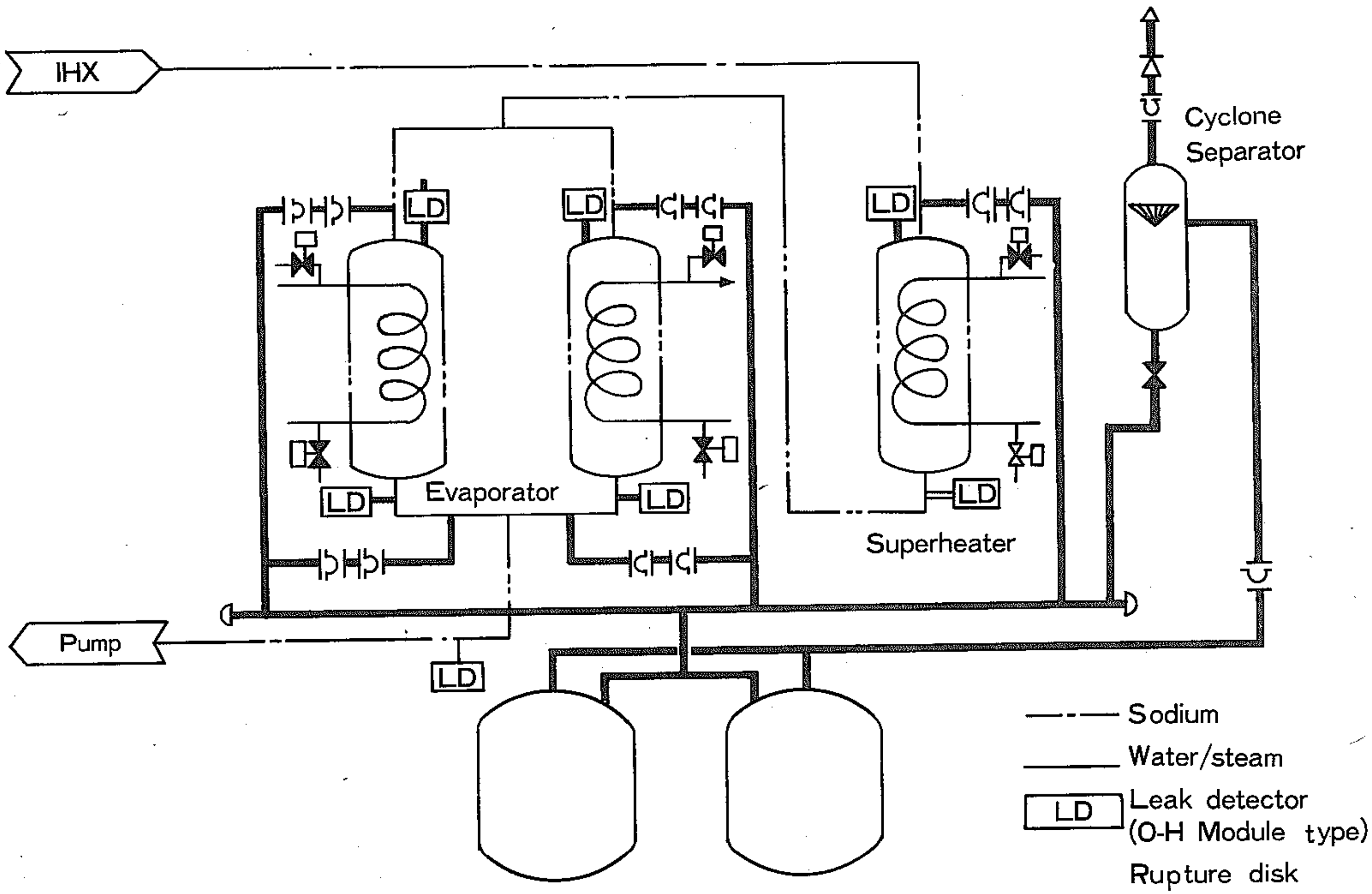


Figure 3·14 Concept of design measures against sodum-water reaction in steam generator accommodating system.

3.2.10 Reactor Shutdown System Reliability

Reactor shut-down systems having enough reliability to exclude HCDA shall be designed in DFBR. In order to enhance the reliability, it is very important to eliminate common cause failures from shut-down systems, and for this purpose, a secondary shut-down system is required, in addition to the primary shut-down system. Each shut-down system is completely and independently separated, and is consisted of components having technologically possible diversities.

In order to judge the design adequacy of shut-down systems, tentative goals of reliability have been decided as follows;

The primary shut-down system:	10^{-7}/demand for random failures (3×10^{-5}/demand for common cause failures)
The secondary shut-down system:	10^{-4}/demand for random failures

The basic concept for current DFBR reactor shutdown system is illustrated in Figure 3.15. Diversities for control rod clusters and control rod drive mechanisms are realized by the combinations of scram methods and acceleration methods of control rod clusters for rapid dropping as shown in Table 3.3.

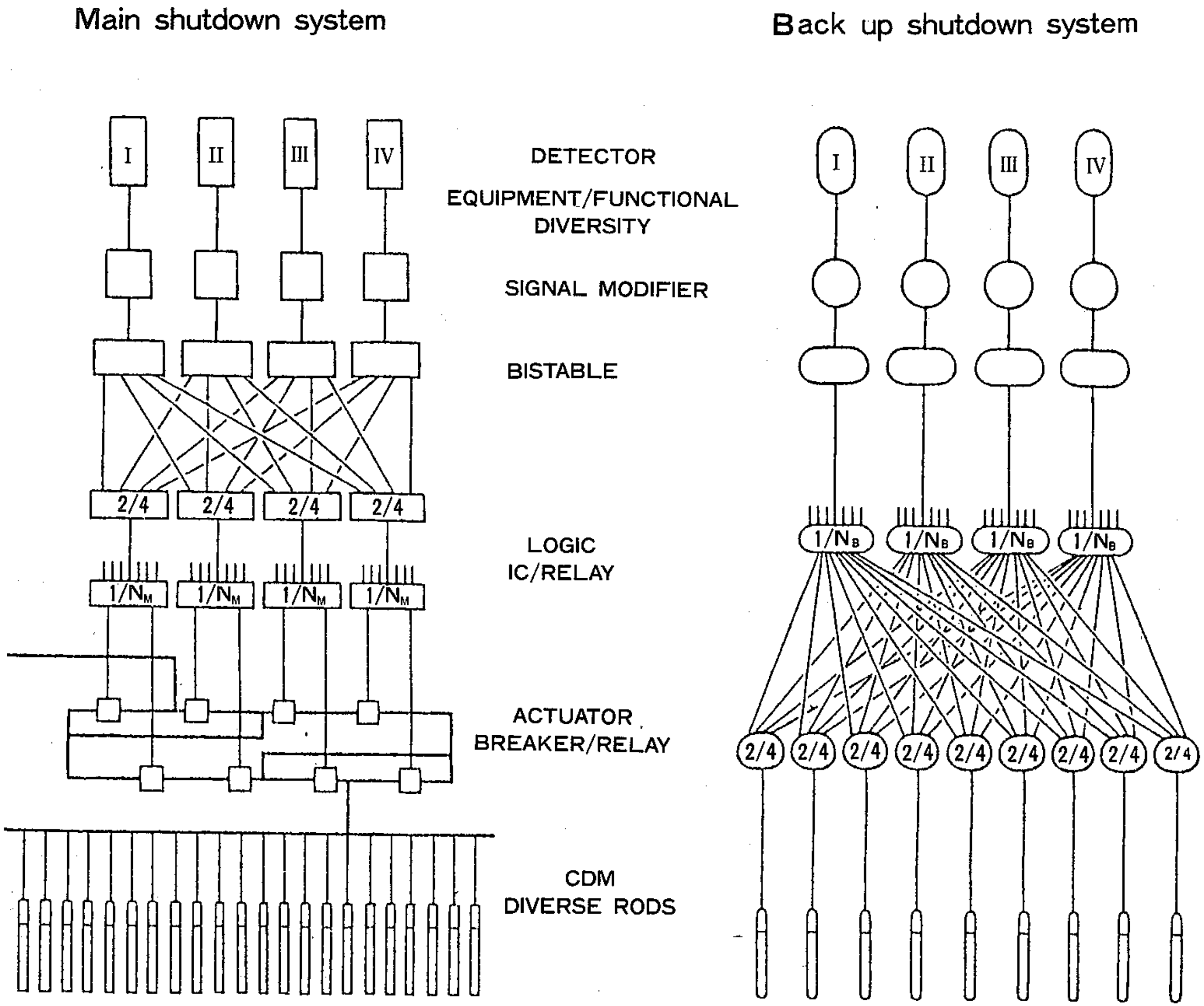


Figure 3·15 Shutdown systems of DFBR

Table 3·3 The composition of reactor shut-down systems

	System		Composition
Reactor Shut-down System	Mechanical (CR, CRD)	Primary	Integral dropping of CR plus CRD, Spring acceleration
		Secondary	Separated dropping of CR, Hydraulic acceleration
	Electrical (Scram logic, Scram breaker)	Primary	2/4 Local coincidence(IC), 2/4 One block actuator
		Secondary	2/4 General coincidence (transistor or relay), 2/4 individual actuator

3.2.11 Shut-down Heat Removal System

As SHRS, four independent systems of SGAHRS (SG Auxiliary Heat Removal System) are installed and the decay heat shall be removed by both PACC (Protected Air Cooled Condenser) and steam blows from the steam blow valves after reactor shut-down. After steam blows are ceased due to the decay heat decrease, it shall be removed by only PACC. Even if the single failure criterion were superposed to any accidental condition, SGAHRS has enough system capabilities (16 MW/one loop) to remove decay heat just after scram by only two loops remained, and one loop heat removal is also possible after two days.

Based on the design policy that SHRS has an important role to maintain its function during a period of reactor shut-down, and any anticipated risks to the public should be eliminated by increasing reliability and diversities of SHRS, DRACS (Direct Reactor Auxiliary Cooling System) has been considered to be installed in addition to SGAHRS against multiple failures postulated during decay heat removal operation mode of the plant, although SGAHRS is enough for SHRS as far as current single failure criterion is applied in licensing.

DRACS is designed to be workable in one day after scram, postulating that forced circulation cooling passes of the primary sodium circuits are disturbed due to the sodium level in the vessel lowering (for example, multiple leaks, and soon), and limiting that the maximum sodium temperature in the upper plenum of the vessel is 650°C.

Current DRACS is consisted of completely independent two systems in its primary and secondary sodium circuit, its air coolers, and its component cooling systems (HVAC). All of its components which include final heat sinks are consisted of passive equipments that any active operation is not required, and all of heat is expected to be removed completely by only natural circulation. In components design, heat removal capacity is assumed to attain 14 MW/two circuits which is equivalent to the decay heat in two days after scram.

Reliability analyses on this SHRS conducted taking account of the heat removal capacity of DRACS, its required time to be started, and its operational mode combined with SGAHRS have been concluded that current SHRS of DFBR have higher reliability to satisfy the initial design policy.

3.2.12 Evaluation of Lower Probability Events

In DFBR, postulated events which should be evaluated as " LP/HC Events " have not clearly defined yet, but reflecting the licensing policy in "MONJU", preventing means against following postulated three events have been designed and safety evaluations have been conducted so that these events could be categorized into BDBA defined as technologically incredible events to be occurred.

(1) Local faults in the core

Local faults in fuel subassemblies shall be prevented by conducting careful quality controls during fabrication and impurity controls of the primary cooling system, and by designing multiple coolant inlets into them. An in-reactor monitoring system is installed for watching operated core conditions, and when its detecting abnormals, the plant shall be manually tripped in order to prevent their growth. As for this, it has been evaluated by analysis that a single fuel pin failure or a few pins failures could not propagate before the reactor shut-down. In spite of above preventive means, local faults of a single fuel subassembly anticipated from a technological viewpoint shall be explained by safety evaluations that those could not be developed into surrounding fuel subassemblies.

Moreover, local accidents in the core which is incredible to be occurred from a technological view point have been evaluated to confirm the safety margin in the condition that DN dectors are available which can generate reactor trip signals when 50 grams of fuel are exposed into the coolant. It is further required that postulated local accident levels and the amount of exposed fuels are moderately estimated in order to reduce the safety margins if they were excessive.

(2) Large pipe break events in the primary sodium coolant circuits

A "Leak Before Break" rationale is applied and the design basis on a failed area of main pipings have been tentatively assumed 1/4 Dt.

Horizontal piping arrangements in the higher location and guard vessels are effective means for keeping the minimum sodium level enough to cool the core against any sodium leaks in the reactor coolant boundaries, and also fire suppression means (linings, coolings by HVAC systems, and so on) are provided.

Against large pipe break events beyond the design basis, it have been concluded by detailed safety analyses that the minimum sodium level required to cool the core could be maintained and the inherent safety nature of the plant could well suppress their consequences. As in DFBR design, cold leg pipings are introduced into the reactor vessel from the above, safety margins against cold leg piping breaks are much increased.

(3) Loss of reactivity control capability events

Occurrence of these postulated events are prevented by increasing the reliability of the shut-down system, by providing two systems which are mutually independent and divers including plant protection circuits as explained in details at the previous section, and these events shall not be included into the design bases.

In case of hypothesizing that above preventing measures did not work, a great deal of effort shall be devoted to explain that consequences could be well mitigated by the inherent safety nature of the plant by developing adequate evaluation methods, but tentatively, options shall be remained that can accommodate mitigating measures defined as beyond design bases (for examples; shield plug design, containment system design, and so on).

3.3 Future Safety Issues

The safety design policy and conditions, having been studied in the historical background described in the previous sections, and methods to have derived them are seemed to be difficult to be applied to our pool-type reactor, not yet fully given a sufficient design study. Therefore, a great deal of work is required on the pool-type reactor in the near future, responding the progress of its design study in Japan. There are also various items to be studied again from the view point of improving plant economics. Reasonable safety design policy with regard to these items are expected to be established through international discussions.

The following future important safety issues in the design of the loop-type DFBR are related mainly to the treatment of LBB and HCDA.

- Diversity grade of reactor shutdown system
- Local blockage scale to be postulated in a fuel subassembly
- Need of diversity and design position of natural circulation regarding a decay heat removal system
- Need and role of in-service inspection (ISI) in safety design
- Piping rupture scale to be postulated in design
- Long-term core cooling of post-accident and others.

How to treat these issues is different among countries with development programme of FBRs. The development is in progress respectively on the basis of the design phylosophy of each country, and it is noted that there are clear differences when comparing design concepts of each country which concretely express a safety level to be achieved in each plant.

In the future development of FBRs, the design to meet required safety level should be reasonably pursued in reducing plant construction cost as much as possible, and the background available for its decision-making when selecting design options is expected to be obtained internationally.

Our country should make an effort also to promote actively research and development and to obtain the information required for the decision-making as earlier as possible.

HCDA is in an important position as safety design, when various studies have to be conducted in order to design DFBR with reasonable construction cost. In our country, safety of the public around site is evaluated on the basis of consequences of specified major accidents and a hypothetical accident both defined in the "Guidelines in reactor site evaluation" issued by the Japanese Nuclear Safety Committee. In this procedure, it is a critical problem how to postulate adequate radiological source term. Occurrence probability of initiating events of HCDA is so small based on the failure probability of reactor shutdown systems and shutdown heat removal systems that these event are incredible from the engineering point of view.

Nevertheless, if it is required to evaluate "the postulated events of which occurrence frequency is further lower than that of unlikely "accidents" but the consequence might be serious", since the operational experience of FBRs is insufficient, as shown in the regulatory document "safety evaluation policy for LMFBRs", it is needed to assure that the release of radioactive materials in the so called BDBA to the environment is appropriately suppressed.

In order to make DFBR acceptable to the public as a plant with enough level of safety, we consider that it is important to prove its safety with focusing efforts on the first and second levels of safety design, preventing occurrence of accidents while reducing construction costs, and accumulating plant operating experiences, as well as with continuing efforts to investigate HCDA phenomena.