JNC TN3410 99-003(1)

Development of Advanced Thermal Reactor "Fugen"

Japan Nuclear Cycle Development Institute
Fugen Nuclear Power Station

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I. Plant Management

1. History of ATR Development and Role of ATR in Japan

1.1 Decision of ATR Development and Role of ATR

(1) Activity of ad hoc Committee on Power Reactor Development

In October 1964, Japan Atomic Energy Commission (JAEC) organized the "ad hoc Committee on Power Reactor Development" in order to decide the basic policy of power reactor development in Japan.

The ad hoc Committee had discussed, reviewed and evaluated for a year and a half, on the future energy supply in Japan, prospects of power reactor development in the world, etc., and reported the "Basic Program of Power Reactor Development in Japan" to JAEC.

(2) Significance of ATR Development

(a) Decision of ATR Development

Considering the report of the ad hoc Committee, energy situation, power reactor development in developed countries and so on, JAEC decided, in May 1966, to develop ATR (Heavy Water Moderated Boiling Light Water Cooled Reactor) and FBR (Fast Breeder Reactor), with targeting to improve the energy security and to ensure the energy supply in Japan.

(b) Significance of ATR Development

As the technology developed in LWRs (Light Water Reactors) and thermal power stations are applied to ATR, the development of ATR was to be started from the prototype stage, and the significance of ATR development is considered as follows:

① ATR has flexibility of using nuclear fuels, and ATR could be operated with feeding natural U.

② The demands of natural U and enriched U could be reduced, as ATR could efficiently utilize U. At the same time, ATR can supply Pu efficiently to FBR when FBR be commercialized.

③ ATR could cover energy security expected to FBR, though to some extent, when FBR commercialization be delayed.

At that time, OECD evaluated that reasonably assured U resources in the range of 5 to 10 \$/lb U_3O_8 would be about 6×10^5 short-tons ("World U and Th Resources", OECD European Nuclear Energy Agency, 1965) and commercialization of U enrichment plant was considered to be very difficult except USA. Therefore, JAEC give high valuation upon development of the reactor operated with natural U.

(3) Foundation of PNC

In October 1967, Power Reactor and Nuclear Fuel Development

Corporation (PNC) was founded, and the development of ATR and FBR was started in Japan, in large scale.

1.2 Pu Fueled ATR and Its Role [I-1] [I-2] [I-3]

(1) Pu Fueled ATR

The initial core of "Fugen", ATR prototype reactor (165Mwe), was firstly designed to load 1.5 % UO_2 fuel assemblies with 6 MOX fuel assemblies with the consideration of realizing Pu self sustain cycle in which natural U is added with Pu extracted from own spent fuel and is used.

However, 96 MOX fuel assemblies, 124 UO_2 fuel assemblies and 4 special fuel assemblies with pressure tube assembly test pieces were decided to load in the initial core, because of following reasons. And the development of Pu fueled ATR was started as the first Pu fueled thermal reactor in the world.

① Power generating cost with using enriched fuel would be considered more economical than the one with using natural U.

② To use Pu in this type of reactor could make the coolant void reactivity lower, and thereby making the reactor safer and more economical.

③ R&D gave us a good future outlook of MOX fuel fabrication and its utilization in ATR.

④ Commercial cost of Pu lowered incidentally and fortunately.

(2) Role of Pu Fueled ATR

LWRs are mainly used both in Japan and in the world. As ATR has a specific characteristics to use Pu easily, ATR will improve the energy security in Japan, by coupling with LWRs and using Pu extracted from LWRs' spent fuel, as shown in [Fig. I. 1-1]. At the same time, ATR will play as a precursor of following developments in the fields of Pu utilization.

① Development of Pu Utilization in Reactors

② Development of MOX Fuel Fabrication and Inspection

③ Development of Pu Fuel Transportation

④ Development of MOX Fuel Reprocessing

⑤ Development of Non Proliferation and Safeguards

⑥ Public Acceptance of Pu Utilization

1.3 Development History of "Fugen"

An outline of development history of "Fugen" [Fig. I. 1-2] is shown in [Fig. I. 1-3].

(1) Design of "Fugen"

(a) First Conceptual Design

In the fiscal year of 1967, the first conceptual design was made, and five

design concepts proposed by five Japanese nuclear consortiums were unified. Following concepts were fixed, and participation of the design fields and systems were allocated to the five consortiums.

① Plant Arrangement
② Structure of Reactor Proper
③ Refueling and Fuel Transfer System
④ Fuel Assembly
⑤ etc.

(b) Second Conceptual Design [I-4] [I-5]

In the fiscal years of 1968 and 1969, the second conceptual design was made. The plant arrangement, basic specifications and structures of main systems and components were established, with design engineering basis.

(c) Modified Design

In order to reflect R&D results on the design, the second conceptual design was modified. Main modifications are listed in the following.

① 96 MOX fuel assemblies would be loaded in the central region of the core, so that the reactor safety may increase, by lowering the coolant void reactivity.
② Four reactor coolant loops were reduced to two.
③ Heat removal capability was improved by lessening spacer pitch in the central region of the fuel.

(2) Construction of "Fugen"

(a) Approval of "Fugen" development Program

In May 1969, JAEC organized the ad hoc committee on the ATR development program in order to make its check and review. Considering the report of the ad hoc committee, JAEC decided to proceed the ATR development program in October 1969.

(b) Construction of "Fugen"

① Commencement of Construction

In March 1970, PNC made application of construction permit of "Fugen" to the Government, and received the construction permit in November 1970. The construction was started at Turuga in December 1970.

② Commencement of Full Power Operation

In June 1977, all systems and components were installed. And the overall plant performance tests were made. Commissioning was then made, and on March 20, 1979 "Fugen" began to operate at the rated power, 165 MWe.

1.4 "ATR Demonstration Plant" Program [I-2] [I-6]

(1) Check and Review of "ATR Demonstration Plant" Program

In January 1980, based on a good operational record of "Fugen" and progress of design and R&D on "ATR Demonstration Plant" development, JAEC organized the ad hoc committee to make deliberation on significance of ATR commercialization, ATR technology, etc.

Taking the report of the ad hoc committee into consideration, JAEC decided to proceed "ATR Demonstration Plant program", in which Electric Power Development Company (EPDC) would be responsible to construct "ATR demonstration plant" and PNC would support EPDC with making necessary R&D and MOX fuel fabrication. Design works and R&D of "ATR Demonstration Plant" were then in progress.

(2) ATR Development after "ATR Demonstration Plant"

JAEC showed the policy of ATR development in the "Long Term Development and Utilization Program on Nuclear Energy", as follows:

"ATR is a heavy water moderated reactor which has excellent flexibility of using nuclear fuels, to use Pu, slightly enriched U and depleted U effectively and efficiently. Moreover, ATR technology has been developed by ourselves in Japan.

ATR development, therefore, should be made with aiming its commercialization, by improving its technology and economy."

(3) Stoppage of ATR Development

In August 1995, JAEC decided to stop the construction program of "ATR Demonstration Plant" program, considering following reasons.

① Economy of ATR

② Situations were so changed to be able to use MOX fuel in LWRs without any great difficulty

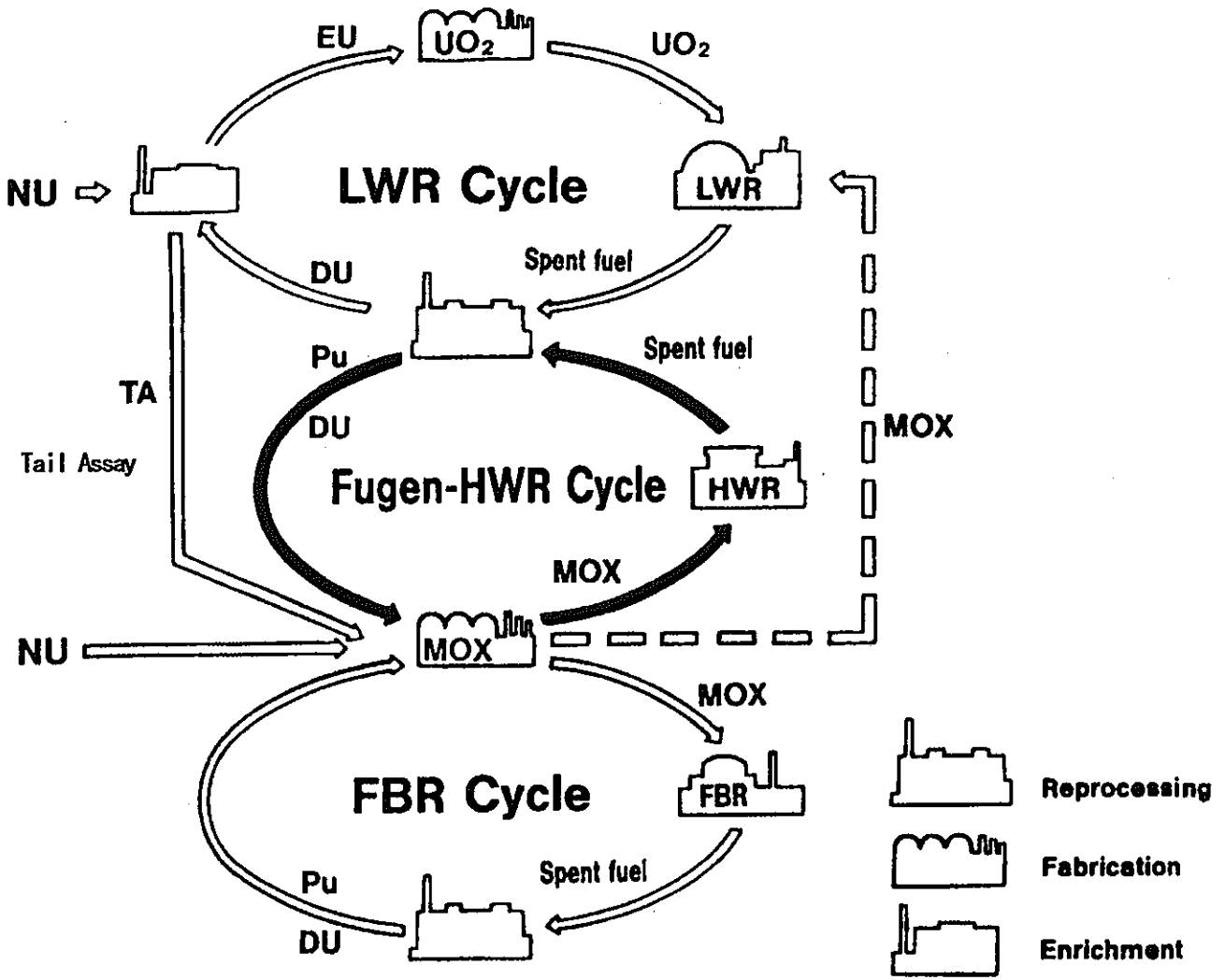


Fig.I.1-1 ATR Fuel Cycle Coupled with LWR and FBR

Fig.I.1-2 General View of "Fugen"

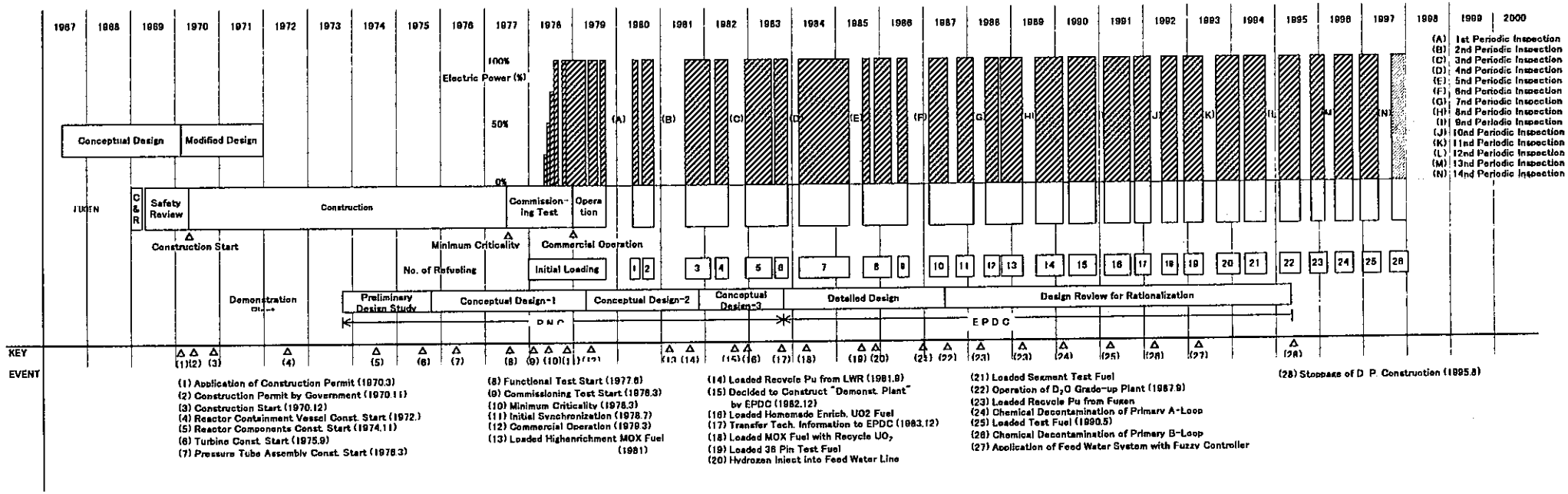


Fig.I.1-3 "Fugen" Development and Operating History

2. R&D, Design and Construction of "Fugen"

2.1 Development Principle of "Fugen" [I-7]

(1) Development Principle of "Fugen"

"Fugen" was developed under the following principles.

① As to important technologies and characteristics associated with ATR, such as core characteristics, safety, etc., full scale testing facilities are to be constructed; thereby clarifying their phenomena and verifying/ validating their characteristics; and at the same time, utilizing to make improvement their technologies.

② As to important components, such as fuel assemblies, pressure tube assemblies, refueling machine and so on, full scale mock-up tests are to be made, so that their performance and manufacturing methods may be cleared, developed and verified/validated, as well as their endurance.

③ Development is to be made under "Design Oriented R&D" basis. Development should be so made that necessary informations for design and/or development may be prepared or transferred to the specified fields or sections by the specified time.

④ Technologies established in thermal power plants and/or LWRs are to be used as possible as we can. They, however, are to be evaluated and verified, if necessary.

(2) Installation of Large Scale Testing Facilities and Testing

At O-arai Engineering Center, PNC, following large scale testing facilities were installed, in order to verify and confirm safety, reactor core characteristics (nuclear, and thermal-hydraulic), endurance of fuel assembly and pressure tube assembly, and so on, as well as to improve their technologies and characteristics in future.

Large scale testing facilities were operated in the fiscal year of 1969, and associated R&D began.

① Critical Assembly (DCA) : [Fig. I. 2-1]
(for verifying and improving nuclear characteristics)

② 14 MW Heat Transfer Testing Loop : [Fig. I. 2-2]
(for verifying and improving thermal-hydraulic characteristics and heat removal limit)

③ Component Testing Loop : [Fig. I. 2-3]
(for verifying and improving endurance and performance of components)

④ Large Scale Safety Testing Facility : [Fig. I. 2-4]
(for verifying reactor safety and improving associated technologies)

The pressure tube monitoring equipment testing facility was installed at O-arai Engineering Center, in order to meet in-service inspection regulation.

[Fig. I. 2-5]

2.2 R&D, Design and Construction of "Fugen" [I-3] [I-8] [I-9] [I-10] [I-11] [I-12] [I-13] [I-14] [I-15] [I-16] [I-17] [I-18]

(1) Plant Configuration and Its Safety

"Fugen" is a 165 MWe prototype of a heavy water moderated boiling light water cooled reactor, and has a specific characteristics, flexibility of using nuclear fuels, to use both MOX and UO_2 efficiently and effectively.

(a) Plant Configuration of "Fugen"

"Fugen" consists of following systems and their configuration is shown in [Fig. I.2-6].

① Reactor Proper
② Pressure Tube Assembly
③ Fuel Assembly
④ Reactor Cooling System
⑤ Heavy Water System, Helium System, Carbon Dioxide System
⑥ Fuel Handling System, Fuel Storage Facility
⑦ Instrumentation and Control System
⑧ Engineered Safety Features
⑨ Reactor Containment
⑩ Turbine Generator System
⑪ Radioactive Waste Disposal Facility
⑫ Heavy Water Upgrading Facility
⑬ Others

(b) Plant Arrangement

Reactor building which contains reactor proper, reactor cooling system, etc., reactor auxiliary building, turbine building and fuel storage building are so arranged as shown in [Fig. I. 2-7].

(c) Main Plant Data of "Fugen"

Main plant data of "Fugen" are shown in [Table I. 2-1].

(d) Safety of "Fugen"

a) Safety Design

"Fugen" was so designed as to ensure safety and to operate with high reliability.

① Reactor Characteristics

Power coefficient is designed negative and coolant void reactivity about 0, in order to keep the reactor safety.

② Reactor Shutdown System

In case of emergency, 49 control rods are dropped into the core rapidly.

The heavy water dump system is also equipped as the back-up reactor shutdown system.

③ Emergency Core Cooling System (ECCS)

Emergency core cooling system consists of 2 independent loops, and works to cool the core safely in case of loss-of-coolant accident. The capability of the emergency core cooling system was tested at the large scale safety testing facility O-arai Engineering Center; based on which the emergency core cooling system was designed.

④ "Fail-Safe" Design

"Fugen" was designed on the basis of "Fail-Safe" which makes components work to keep the reactor safe when trouble happen.

b) Multiple Containment

Multiple containment was so made as shown in the following. The fuel assembly is set in the pressure tube which is in the reactor proper, the reactor proper is surrounded by the iron/water shields. Moreover, these components are contained in the reactor containment. Double containment system was so adopted that the steel made reactor containment is surrounded by the concrete shield wall and air in the reactor containment may not be directly touched to the atmosphere, by keeping the pressure in the annulus negative.

(2) Reactor Proper

(a) Outline of Reactor Proper

The reactor proper consists of calandria, pressure tube assemblies, iron/water shields, as shown in [Fig. I. 2-8]. Calandria tubes are arranged in the square lattice. A calandria tube contains a pressure tube which has a fuel assembly.

A heavy water dump space is set in the calandria so that the reactor may be shutdown in case of emergency, by discharging heavy water in the calandria to the dump space.

Refueling is made from the bottom of the reactor and control rods are driven from the top.

(b) Design of Calandria

The calandria was designed as the Class2 Vessel, defined by "ASME Boiler and Pressure Vessel Code Section III, Nuclear Vessel". However, the calandria forms an unified key structure, ie. the reactor proper, with the iron/water shields and contains heavy water. Therefore, design conditions were specified under strict evaluation, as shown below and strength analysis was made.

Design Conditions of Calandria			
	Max. Internal Pressure (kg/cm^2g)	Max. External Pressure (kg/cm^2g)	Max Temperature (℃)
Normal Operation	0.8	1.0	150
Heavy Water Dump	1.12	1.0	150
Postulated Design Accident (Pressure Tube Rupture)	2.0	1.8	150

(c) Development of Reactor Proper

R&D of the calandria and the iron/water shields are listed in [Table I. 2-2]. As far as R&D on manufacturing and inspection of the calandria and the iron/water shields, mock-up tests were made in the scale of 1/3 in diameter and full scale in height, as the unit reactor structure of pressure tube type reactor would be a pressure tube.

(d) Confirmation of Reactor Proper Installation

Just before sending the calandria and the iron/water shields to the site, confirmation and testing were made at the factory, especially on interface of each works, installing methods, dimensions, in order to avoid human errors and to shorten the installation period.

(3) Pressure Tube Assembly

(a) Outline of Pressure Tube Assembly

The pressure tube assembly, which contains the fuel assembly, consists of pressure tube of Zr-2.5%Nb, upper and lower pressure tube extensions of 304 stainless steel and 403 Mod. stainless steel tube which connects the pressure tube and pressure tube extension, as shown in [Fig. I. 2-9]. As Zr-2.5%Nb and 304 stainless steel could not be welded and their thermal expansion coefficient ratio is 1/3, Zr-2.5%Nb is rolled-jointed to 403 Mod. stainless steel which is welded to 304 stainless steel. [Fig. I. 2-9]

(b) Design of Pressure Tube

a) Pressure Tube Design Code

The pressure tube was designed based on Class 1 Vessel, "ASME Boiler and Pressure Vessel Code Section III", and creep analysis was made following "ASME Code Case Interpretation of nuclear components at Elevated Temperature Service".

b) Main Evaluation Items of Pressure Tube Design

Following are main evaluation items of pressure tube design.

① Allowable Design Stress

The allowable design stress should not be exceeded following strength.

- 1/3 of tensile stress
- 2/3 of yield point
- 4/5 of min. creep rupture strength, or
- 2/3 of average creep rupture strength

② Creep Strain Limit

2.5% was determined as the creep strain limit, considering following.

- Should not be used in the range of tertiary creep region where large creep rate occurs.
- Radial creep of pressure tube should affect little on heat removal of the core.
- Longitudinal creep of pressure tube should not affect the pressure tube assembly and the support associated.

③ Fatigue Analysis

Analysis was made by superposing fatigue and creep, with each other.

④ Fracture Toughness Analysis

The pressure tube is so designed as not to occur unstable fracture.

Fracture toughness of pressure tube was evaluated based on the crack length and hydrogen pick-up at the end of reactor life.

The initial crack length was assumed to be the minimum detectable crack length and propagated at the rate of propagation rate determined by the experiments.

⑤ Design Hydrogen Ingress and Corrosion

- Design hydrogen Ingress : 211 ppm (30years)
- Design corrosion : 0.62 mm
- Corrosion by water : 0.31 mm
- Corrosion by carbon dioxide : 0.23 mm
- Corrosion by fretting : 0.08 mm

(c) Design of Pressure Tube Rolled-Joint

Mock-up tests of the pressure tube rolled-joint were made, based on which the pressure tube rolled-joint was designed.

(d) Development of Pressure Tube Assembly

An outline of pressure tube assembly development is shown in [Table I. 2-2], and described in the following.

a) Development for Prevention of Unstable Pressure Tube Rupture

Confirmation was made that the pressure tube was so designed as not to occur unstable rapture based on the experiments and associated analysis.

Following are typical R&D results of pressure tube assembly.

① Fracture Toughness of Zr-2.5%Nb:[Fig. I. 2-10]

② Crack Propagation of Zr-2.5%Nb:[Fig. I. 2-11]

③ Brittle Characteristics of 403Mod. Stainless Steel:[Fig. I. 2-12]

b) Development of Rolled-Joint

List of rolled-joint development is shown in [Table I. 2-3]

(4) Fuel Assembly

(a) Outline of Fuel Assembly

① Standard Fuel Assembly

As shown in [Fig. I. 2-13], the standard fuel assembly is 28 rod cluster fuel assembly (4 rods in inner circle region, 8 rods in intermediate circle region and 16 rods in outer circle region). The effective fuel length is about 3.7 m, and total length about 4.5 m. Main fuel assembly design specifications are listed in [Table I. 2-1].

The spacer pitch in the central region of the fuel is designed narrow, compared with outer region, so as to obtain higher burnout limit.

② Special Fuel Assembly

The special fuel assembly contains monitoring material test specimens of the pressure tube assembly, as shown in [Fig. I. 2-13A].

(b) Design Basis of Fuel Assembly

a) Design Basis of Fuel Rod

As Pu enrichment of ATR MOX fuel is rather low, ATR MOX fuel was designed based on the design of LWRs' fuel. Main analysis on ATR MOX fuel is shown in the following.

① Analysis on Performance

② Stress Analysis

Stress analysis of fuel cladding was made based on the Class 1 Vessel "ASME Boiler and Pressure Vessel Code, Section III". However, a lower value, whether 2/3 of yield point or 1/2 of tensile strength, was taken as the allowable design stress.

③ Strain Analysis

1 % was taken as the allowable design strain of fuel cladding.

④ Fatigue Analysis

Fatigue analysis was made following "ASME Boiler and Pressure Vessel Code, Section III".

⑤ Backling Analysis

The cladding was so designed not to occur backling against the reactor coolant pressure.

⑥ Analysis on Pellet Cladding Chemical Mechanical Interaction

b) Design Basis of Standard Fuel Assembly

Following are main design basis of the standard fuel assembly.

① The fuel assembly should be safely cooled under the designed transient conditions.

② The fuel assembly should be able to absorb any dimension change caused by irradiation creep and bending of fuel rods.

③ The fuel assembly should withstand against hydraulic vibration.

c) Design Basis of Special Fuel Assembly

The special fuel assembly contains monitoring test pieces of Zr-2.5%Nb and 403 Mod. stainless steel in the center. The special fuel was so designed as to obtain higher neutron flux in the central region than the one in the pressure tube, so that accelerated irradiation for monitoring materials might be achieved.

(c) Development of Fuel Assembly

a) Out-of-Pile Tests

Following tests were done on the fuel assembly, and its performance, characteristics and endurance were confirmed.

① Nuclear Characteristics in DCA

② Thermal-Hydraulic Characteristics and Heat Removal Limit in 14 MW Heat Transfer Testing Loop and Component Testing Loop

③ Endurance in Component Testing Loop

b) Irradiation Tests

Following irradiation tests were done in order to confirm irradiation performance and endurance of ATR MOX fuel.

① Irradiation Tests in JMTR

② Irradiation Tests in HALDEN

③ Irradiation Tests in SGHWR

(5) Reactor Cooling System [I-19]

(a) Outline of Reactor Cooling System

The reactor cooling system consists of two independent loops each of which has two recirculation pumps. The recirculation pumps have a function of changing pump speed at about 40 % of full power.

The coolant supplied into the core forms two phase flow by receiving the heat from the fuel, and led into the steam drums where steam and water are separated. Steam is then sent to the turbine and condensed water is sent back to the steam drums. Water in the steam drum is backed to the re-circulation pumps. Part of the coolant is bred into the purification system to keep the coolant quality at the specified value.

(b) Development of Reactor Coolant System

As the technology of LWR reactor cooling system can be applied to the ATR reactor cooling system, development is made on following specific items of

ATR, and confirmed their performance.

① Thermal Shield between Impeller and Bearing

② Steam Separator in Steam Drum

③ Check Valve at Inlet of Lower Header

④ Main Steam Isolation Valve

(6) Heavy Water System and Helium System

(a) Outline of Heavy Water System and Helium System

a) Heavy Water System

The heavy water system consists of following sub-systems.

① Heavy Water Cooling System

The heavy water cooling system recirculates the moderator, heavy water, and cools the heat generated in and transferred to the heavy water. At the same time, part of heavy water is sent to the upper tube sheets of the calandria and flows down along the outer surface of the calandria tubes; thereby cooling the part of calandria tubes exposed in helium.

② Heavy Water Purification System

The heavy water purification system removes corrosion products to keep the heavy water quality at the specified value, and also controls boron content in the heavy water.

③ Born Supply System

The boron supply system supplies boron to adjust the boron content in heavy water for compensating the reactivity.

b) Helium System

The helium system consists of following sub-systems.

① Helium Recirculation System

Helium covers free surface of heavy water not to degrade the heavy water quality, and the helium recirculation system recirculates helium and sends part of helium to the recombiner, to keep the concentration of deutrium and oxygen in helium under the immflamability limit.

② Helium Purification System

The Helium purification system consists of the recombiner and the purifier. The former recombines deutrium and oxygen decomposed by radiation, and the latter purifies helium by active carbon which removes impurities in helium.

c) Heavy Water Dump System

The free surface of heavy water in the calandria tank is covered with helium which connects the dump space through the communicating valve. During operation, heavy water is kept in the calandria tank by

pressurizing the dump space, while little heavy water is in the dump space. In case of loss-of-coolant accident, the communicating valves open rapidly to equalize the pressure in the calandria and the dump space, thereby heavy water be flushed into the dump space and the reactor is shutdown.

(b) Development of Heavy Water System and Helium System

a) Development of Heavy Water System

Flow test in double structure upper tube sheets and flow test on flowing down along outer surface of calandria tube were done in the full scale. Following were confirmed.

① Heavy water flows inside of double structure upper tube sheets with wetting the inside upper surface.

② Heavy water flows along the outer surface of the calandria tube with wetting its whole surface.

b) Development of Helium System

Heavy water supplied into the calandria tank overflows and down to the heavy water dump tank with involving helium.

This phenomena pressurize helium in the heavy water dump tank and could give driving force to recirculate helium.

The full scale test was then made, and its capability was confirmed. Based on the results the helium recirculation system was designed.

c) Development of Heavy Water Dump System

Heavy water dump test was done with the full scale calandria mock-up, 1/4 sector. Performance of the heavy water dump system was confirmed, to meet the safety design criteria (heavy water level should be lowered by 2m in 2 sec).

(7) Fuel Handling System

(a) Outline of Fuel Handling

The fuel handling system consists of refueling machine, fuel handling machine, fuel transfer chute, fuel receiving machine, etc. As shown in [Fig. I. 2-14], fuel loading and fuel unloading are made in the following way.

① Fuel assemblies, stored in the spent fuel storage pool, are transferred into the fuel pool in the reactor building through the transfer chute

② Fuel assemblies are then handled by the refueling machine and loaded into the core from the bottom of the reactor.

③ Fuel unloading and associated fuel handling are made by the reverse way.

(b) Design Basis of Refueling Machine

Design basis of the refueling machine, the most important machine of the fuel handling system, is as follows:

① On-load refueling can be made.

② To withstand earthquake during refueling.

③ Refueling machine proper is designed following "ASME Boiler and Pressure Vessel Code, Section III, Class I Vessel", as the refueling machine is directly connected to the pressure tube assembly when on-power refueling is made.

(c) Development of Refueling Machine

As to development of refueling machine, important parts were manufactured and tested, and improved, again and again. Finally mock-up test in the full scale, was made and following performance and capability were confirmed.

① Positioning the refueling machine could be made within ±3 mm.

② The refueling machine and the pressure tube assembly could be connected and separated even 10 mm displacement be existed between the snout of the refueling machine and the inlet of the pressure tube assembly.

③ Fuel assembly be not damaged under 10 mm displacement which is evaluated in the Class A design earthquake.

(8) Instrumentation and Control System [I-20]

(a) Outline of Instrumentation and Control System

a) Outline

The instrumentation and control system chart is shown in [Fig.I. 2-15]. The system is controlled and monitored in the main control room.

Important systems and equipment, associated with safety, are so designed as to have the function of defense in depth and fail-safe. as well as testability during operation, as many as possible.

b) Main Control Room

Following instrumentation and control sub-systems are equipped in the main control room.

① Reactor Control

② Reactor Cooling System Control

③ Reactor Auxiliary System Control

④ Engineered Safety Features Control

⑤ Turbine Generator System Control

⑥ Safety Protection System Control

⑦ Nuclear Instrumentation System Control

⑧ Radiation Monitoring System Control

⑨ Process Instrumentation Control

(b) Design Basis of Instrumentation and Control System

a) Installation Criteria of Sensors for Control and Alarm

① The sensors are to be installed in accessible area.

② The sensors of reactor protection system are to install 4 independent ones, and not to be common with other systems.

b) Design Criteria of Process Instrumentation Conduits

① Slope of piping should be over 1/10, so that vapor may not be trapped.

② Helium leak test should be done for instrumentation piping where radioactive gases flow.

c) Safety Protection System

① Function of Safety Protection System

The safety protection system functions to activate alarm, scram, etc. and to make the main steam isolation valve, the emergency core cooling system, etc. work, by abnormal signals of the process instrumentation.

② Independency of Safety Protection System

The circuit of the safety protection system is so designed as to compose of 4 independent circuits and to adopt (one out of two) x two criteria, basically. The 4 independent circuits are physically separated by partition walls, conduits and/or distance, in order to ensure their independency.

③ Reactor Scram

Reactor will be shutdown by signals shown in [Table I. 2-4], to ensure reactor safe.

④ Indication of Alarm

In the main control room, indication of alarm is made in the following.

- Lamp flickers & bell rings : When trouble happens
- Lamp lights without flicker : When alarm stops or releases
- Lamp be put off : When reset

(c) Development of Instrumentation

a) Design Basis of Instrumentation

Instrumentation in LWRs was basically applied to the instrumentation in "Fugen".

b) Nuclear Instrumentation is divided into following 3 power regions, overlapping with each neighbor range.

① Start up Range : SUM (Start Up Monitor)
4 channels
Counting Range ($\sim 10^6$ cps)

② Intermediate Power Range : PUM (Power Up Monitor)

6 channels
Counting range (~12.5% of full power)

③ Power Range : LPM (Local Power Monitor)
16 x 4 = 64 Channels
Counting Range (~125 % of full power)

As neutron flux at LPM (fission chamber) in heavy water would be 3.5 times as much as the one in LWRs. Therefore, ratio of U-235/U234 was taken to 1/3 for fissile material content of LPM fission chamber, thereby making the life of LPM long.

Irradiation test was made on the long life LPM, and its performance and capability were confirmed.

(d) Development of Control Rod

A general view of control rod is shown in [Fig. I.2-16].

Following development was done on controlled.

a) Control Rod Reactivity Worth

Control rod reactivity worth was measured in DCA and analyzed.

Based on the development, decision was so made that the double layers of B_4C balls would be enough.

b) Control Rod Drive System

① Mock-up tests were done on the control rod drive mechanism, based on which the system was designed and manufactured.

② Control Rod Drop Performance

Control rod drop test was made to confirm its performance at scram. The tests showed that its performance satisfied the safety design requirement (80 % insertion within 2 sec).

(9) Engineered Safety Features

(a) Outline of Engineered Safety Features

a) Configuration

The engineered safety features in "Fugen" consists of emergency core cooling system, isolation cooling system, residual heat removal system, containment spray system and steam injection pool cooling system. Each system has two 100 % independent loops. When external power be failed, electric power will be supplied by the diesel generator.

b) Emergency Core Cooling System (ECCS)

When a large pipe rupture accident occurs in the reactor cooling system, the emergency core cooling system (ECCS) works to cool the core without giving much damage. ECCS consists of following systems.

① Accumulator Injection System

In case of a large pipe rupture in the reactor cooling system, water

stored in the accumulator will be injected into the lower header to keep water in the core.

② High Pressure Injection System

In case of small pipe rupture where the steam drum pressure be maintained while its water level be lowered, water will be injected into the steam phase of the steam drum, so that the pressure in the steam drum may decrease, thereby making the low pressure injection system work, as soon as possible.

③ Low Pressure Injection System

The low pressure injection system works at loss-of-coolant accidents to supply water into the lower header to cool the core and to prevent much fuel damage.

c) Isolation Cooling System

When the reactor and the turbine are separated due to the main steam isolation valve closure, the isolation cooling system works to supply water into the steam drums and to keep water in the core.

(b) Development of ECCS

At the large scale safety testing facility, O-arai Engineering Center, full scale tests were done in order to clarify loss-of-coolant accident characteristics, check valve performance, emergency core cooling system capability, etc., based on which the emergency core cooling system was designed.

(c) Core Heat Removal when ECCS does not work

When ECCS does not work at a loss-of-coolant accident, fuel in the pressure tube could be estimated not to melt by radiation heat transfer, if outside of calandria tubes be cooled by water.

This was confirmed by the full scale experiment, results of which is shown in [Fig. I. 2-17].

(10) Core Characteristics

(a) Design Basis of Core [I-16]

The core design was made on nuclear and thermal-hydraulic fields following design basis, and the fuel assembly, reactor cooling system, etc. were so designed as to meet the core design.

a) Nuclear Design Basis

Nuclear design was made by following basis.

① Power Coefficient : Negative

② Coolant Void Reactivity : about 0 or negative

③ Shutdown Margin : more than 0.02

④ Control Rod Worth : Control rod worth and control rod

arrangement should be so made as to keep sub-critical even though control rod with max. reactivity worth not be inserted into the core under the specified liquid poison content in heavy water.

⑤ Refueling Program : Refueling program should be so made in every operating cycle as to satisfy heat removal limit.

b) Thermal-Hydraulic Design Basis

Thermal-hydraulic design was made by following design basis.

① Design Liner Heat Rate : 68.9 kw/m (at 120% full power)
57.4 kw/m (at 100% full power)

② Min. Critical Heat Flux Ratio : 1.5 (at 120% full power)
1.0 (at transient)

③ Hydraulic Stability : Dumping Ratio $\leqq$ 0.25

④ Fuel Temperature : Under melting point

(b) Development of Nuclear and Thermal-Hydraulic Characteristics

a) Development of Nuclear Characteristics

① Critical Experiment

The critical experiments were done on following items to clarify nuclear characteristics of ATR, thereby analytical models, analysis and design codes were developed and the nuclear design was verified.

- Critical Mass & Buckling
- Coolant Void Reactivity and Void Reactivity Coefficient
- Coolant Temperature Coefficient
- Moderator Temperature Coefficient
- Thermal Neutron Life & Delayed Neutron Life
- Power Distribution (Lattice-Wide and Core-Wide)
  [Fig. I. 2-18] and [Fig. I. 2-18A]
- Local Power Peaking Factor
- Lattice Parameter

② Development of Nuclear Analysis Codes

Experimental results and their analysis using nuclear analysis codes were compared and evaluated, based on which nuclear analysis codes were developed by improving models, cross sections, etc.

b) Development of Thermal-Hydraulic Characteristics

① Heat Transfer Tests and Component Tests

In order to clarify the burnout limit, pressure loss, hydraulic stability, and other thermal-hydraulic characteristics in the core, following tests

were done in the 14 MW Heat Transfer Testing Loop and the Component Testing Loop.

- Burnout Test [Fig. I. 2-19]
- Pressure Drop/loss Test
- Natural Convection Test
- Hydraulic Stability Test

② Development of Equations on Characteristics and Analytical Codes

Analysis and evaluation were made on the above tests, and equations of burnout characteristics, pressure loss, etc., as well as analysis codes, such as on flow distribution in the core and so on, were developed.

(c) Core Arrangement and Core Characteristics

a) Core Arrangement

Arrangement of fuel, control rod and nuclear instrumentation are shown in [Fig. I. 2-20].

b) Power Distribution

Power distribution in the core measured and evaluated in commissioning tests is shown in [Fig. I. 2-21].

(11) Plant Dynamics

(a) Outline of Plant Control

An outline of the plant control system is shown in [Fig.I.2-15], and main plant control fields are power control, main steam pressure control and steam drum water level control.

a) Reactor Power Control

Reactor power and power distribution are controlled or regulated by adjusting control rods' positions. Load following is done with 4 regulating rods, each of which is arranged in each quadrant core region. Each control rod is so interlocked that only one control rod may be operated at the same time.

b) Main Steam Pressure Control

Reactor power change is made by changing steam flow with regulating control rod positions. When steam flow be decreased, steam pressure lowers. The steam regulating valves will then be adjusted to make turbine inlet steam pressure constant.

c) Steam Drum Water Level Control

Steam drum water level control is made with 3 element control; by steam drum water level signal, steam flow signal and feed water flow signal.

(b) Design Basis of Plant Control

Plant control is made to make and to operate the plant stable and safe. As stability of plant operation is evaluated by dumping ratio, whose design

values are shown below.

① Channel Hydraulic Stability : ≦ 0.5

② Core Stability (Reactivity) : ≦ 0.25

③ Plant Stability : ≦ 0.25

(c) Commissioning Results of Plant Dynamics

Plant stability was validated in the commissioning. A typical results of plant dynamics are as follows:

① Plant Dynamics of Steam Drum Water Level Set Point Change

[Fig. I. 2-22] and [Table I. 2-5]

② Plant Dynamics of Main Steam Pressure Set Point Change

[Fig. I. 2-23]・[Table I. 2-6].

(12) Safety Analysis [I-12] [I-13] [I-14]

(a) Object of Safety Analysis

a) Safety Principle of "Fugen"

"Fugen" is so designed as to satisfy associated laws, regulations, standards and guides, to have inherent stability and safety with prevention measures. The reactor is designed to ensure its safety by taking "Fail-Safe" design criteria in the safety circuit and mitigating troubles and incidents, even human errors and anomaly of equipment may occur.

"Fugen" is designed, manufactured, construction and operated based on quality assurance program, to ensure reactor safety and to improve its reliability

b) Countermeasures against Accidents

The engineered safety features are prepared to prevent much release of radioactive materials, both outside and inside of the reactor containment, even big accidents happen, such as large pipe rupture in the reactor cooling system.

c) Object of Safety Analysis

Reactor safety is analyzed and evaluated on technically postulated events in order to verify the safety design.

(b) Basis of Safety Analysis

The safety analysis is made by determining the safety judgement criteria, setting analysis conditions for evaluation events and making safety analysis, based on which evaluation and confirmation of safety design are made.

a) Safety Judgement Criteria

The safety judgement criteria are determined as follows:

① Min. Critical Heat Flux Ratio : ≧ 1.0

② Max. Temperature of Cladding : ≦ 1,200℃

③ Fuel Temperature : ≦ Melting Point

b) Evaluation Events

Evaluation events are divided into 3 categories, reactivity accidents, mechanical accidents and other accidents, and are selected as follows:

① Reactivity Accidents

- Start-up Accident
- Control Rod Withdrawal Accident During Power Operation
- Heavy Water Level Change Accident
- Restart Accident after Heavy Water Dump
- Heavy Water Poison Reactivity Accident
- Accident at Refueling
- Control Rod Drop Accident
- Cold Water Injection Accident

② Mechanical Accidents

- Loss-of-Coolant Flow
- Loss-of-Coolant Accident
- Main Steam Pipe Rupture
- Fuel Handling Accident
- Loss of Power Supply

③ Other Accidents

- Main Steam Isolation Valve Closing Failure
- Governor Valve Closing Failure
- Turbine Main Stop Valve Closing Failure
- Loss of Feed water
- Turbine Inlet Pressure Regulating Unit Failure
- Control Rod Drive System Failure

c) Safety Analysis

Safety analysis was made on the above events. The results of typical safety analysis on the accidents of the reactor cooling system are shown in [Fig. I. 2-24], [Fig. I. 2-24A] and [Fig. I. 2-25].

(c) Full Scale Safety Tests and Verification of Safety

Tests were done in the ATR Large Scale Safety Testing Facility, on safety of "Fugen", the engineered safety features, loss of coolant characteristics, the check valve capability and so on. The results were analyzed, based on which the safety design was done.

(13) In-Service Inspection Equipment [I-21]

(a) Outline of In-Service Inspection

a) Object of In-Service Inspection

In-Service inspection is to be made by following "ASME Boiler and Pressure Vessel Code, Section XI". The object is to ensure to be able to operate the reactor safely in the next operating cycle, without any troubles or incidents, by inspecting parts, vessels and piping in important systems and facilities.

b) Scope of In-Service Inspection

In Japan, in-service inspection is to be made following "JEAG-4205", equivalent to "ASME Section XI", which specifies to inspect following items.

① Class-1 Components

② Class-3 Components (Equivallent to Class-2, ASME Codes)

Vessel, piping, pump and valve directly used in engineered safety features

Vessel, piping, pump and valve in reactor shutdown system

Vessel, piping, pump and valve necessary for shutdown reactor

③ Class-4 Components (Equivallent to Class-3, ASME Codes)

Vessel or piping in system attached to engineered safety features

Vessel or piping in spent fuel storage facility and its cooling equipment

c) Methods of In-Service Inspection

Typical inspection methods of in-service inspection are as follows:

① Visual Inspection

② Surface Inspection

Liquid Penetrant Testing

③ Volumetric Inspection

Radiation Penetration Inspection

Ultrasonic Inspection

④ Leak Testing

(b) Development of Pressure Tube Monitoring Equipment [I-22] [I-23]

As ATR is a water cooled reactor, LWR technology of in-service inspection was applied to "Fugen", such as components, piping, etc. However, the pressure tube is a specific structure of pressure tube type reactors, its monitoring equipment (in-service inspection equipment) was developed in the following way.

a) Monitoring Methods

Monitoring pressure tubes is to be made as follows:

① Monitoring Humidity in Carbon Dioxide Passed through Annulus between Pressure Tube and Calandria Tube

② Monitoring by Pressure Tube Monitoring Equipment

Visual inspection and volumetric inspection for pressure tube inner

surface

Pressure tube inner diameter measurement

③ Monitoring Test Pieces of Pressure Tube Assembly

Test pieces of pressure tube assembly loaded in the center of the special fuel assemblies are withdrawn at specified intervals and tested and monitored.

b) Development of Pressure Tube Monitoring Equipment

① Function of Pressure Tube Inspection Head

Function of pressure tube inspection head is specified as follows:

- Should be handled by the refueling machine.
- Should be capable of making visual inspection, ultrasonic inspection and inner diameter measurement.
- Flaw or crack inspection and inner diameter measurement should be made by ultrasonic.

② Development of Pressure Tube Monitoring Equipment

Development facility of pressure tube monitoring equipment was built, and mock-up tests and improvement were done.

The equipment is illustrated in [Fig. I. 2-5].

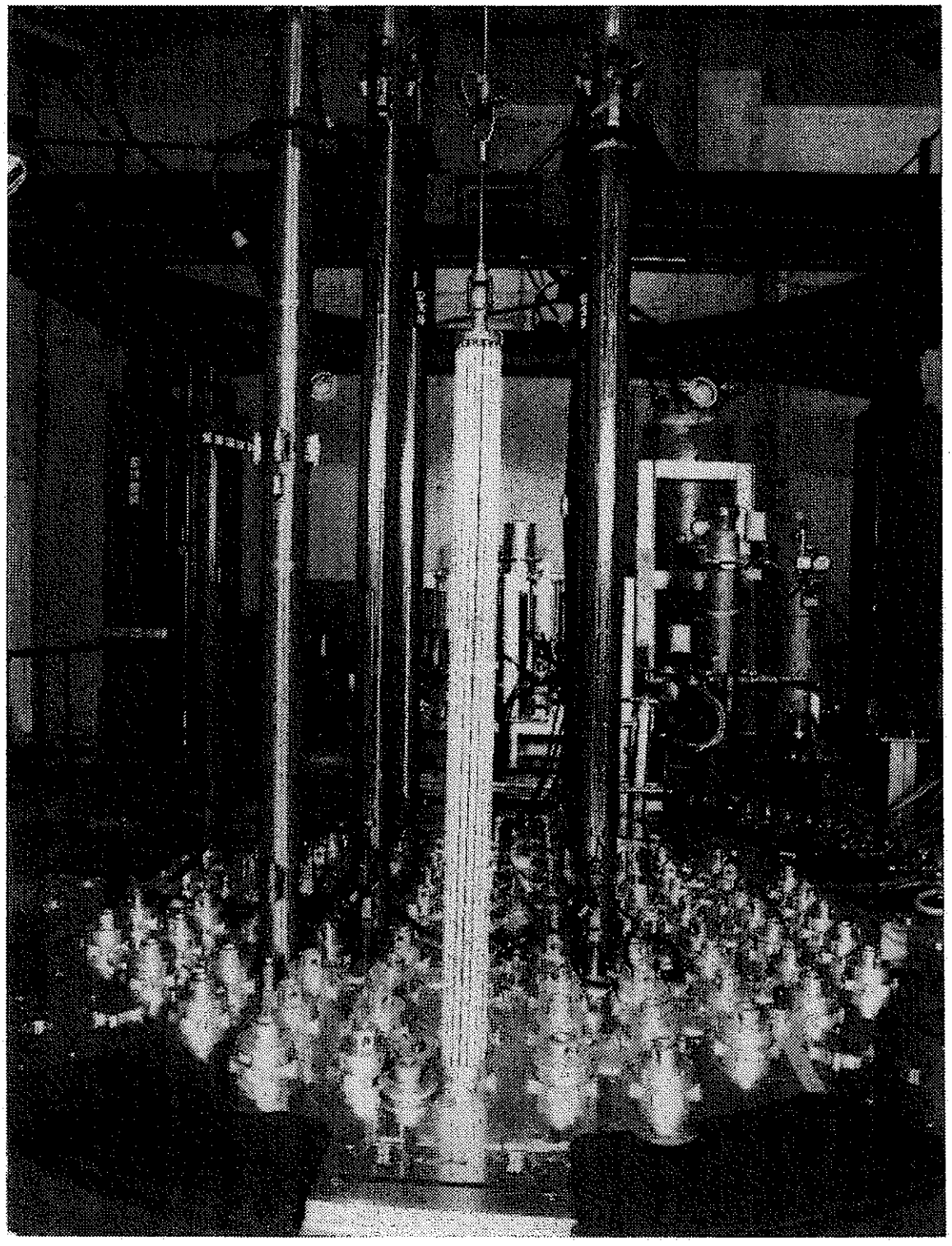


Fig.I.2-1 Deuterium Critical Assembly (DCA)

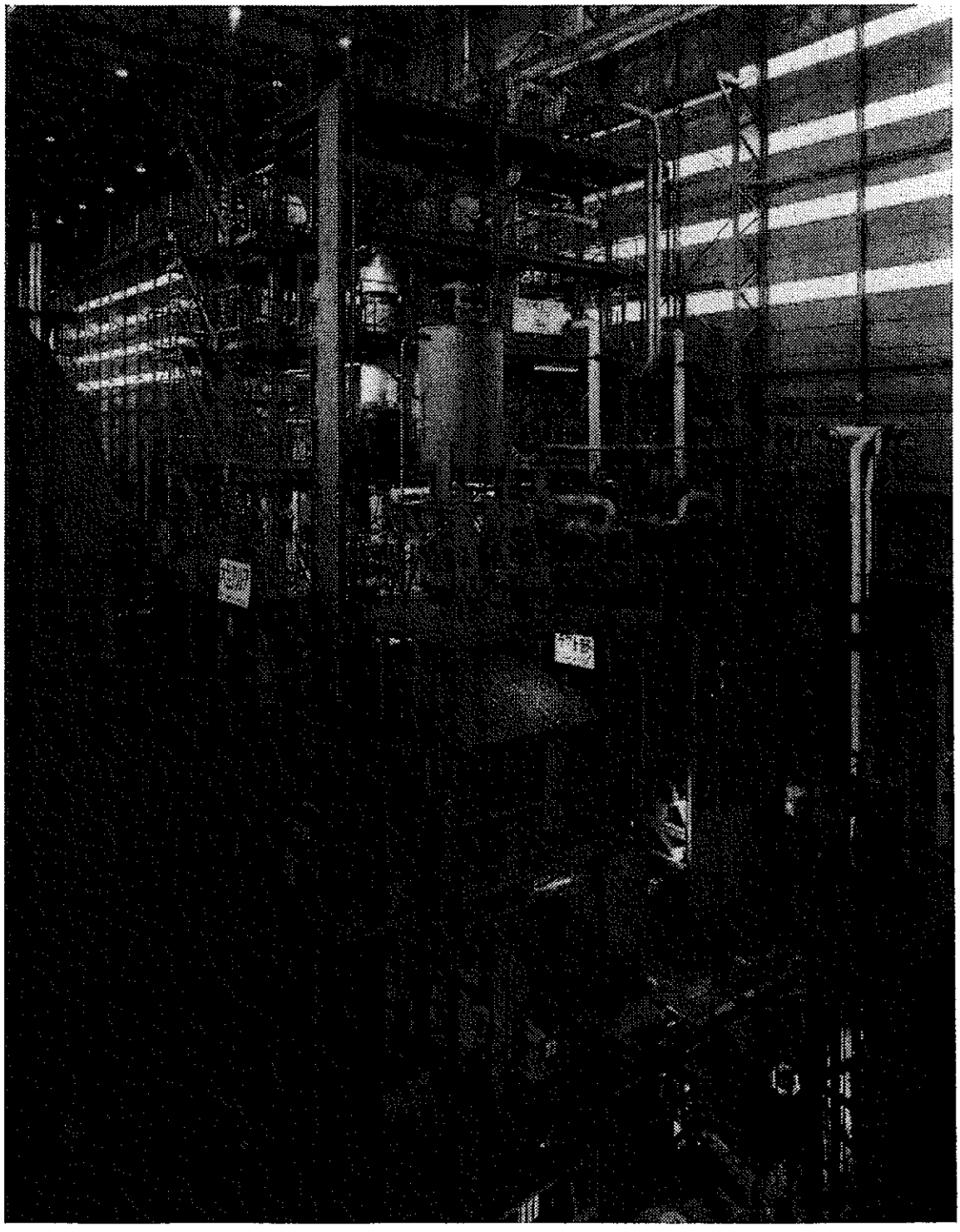


Fig.I.2-2 14MW Heat Transfer Testing Loop (HTL)

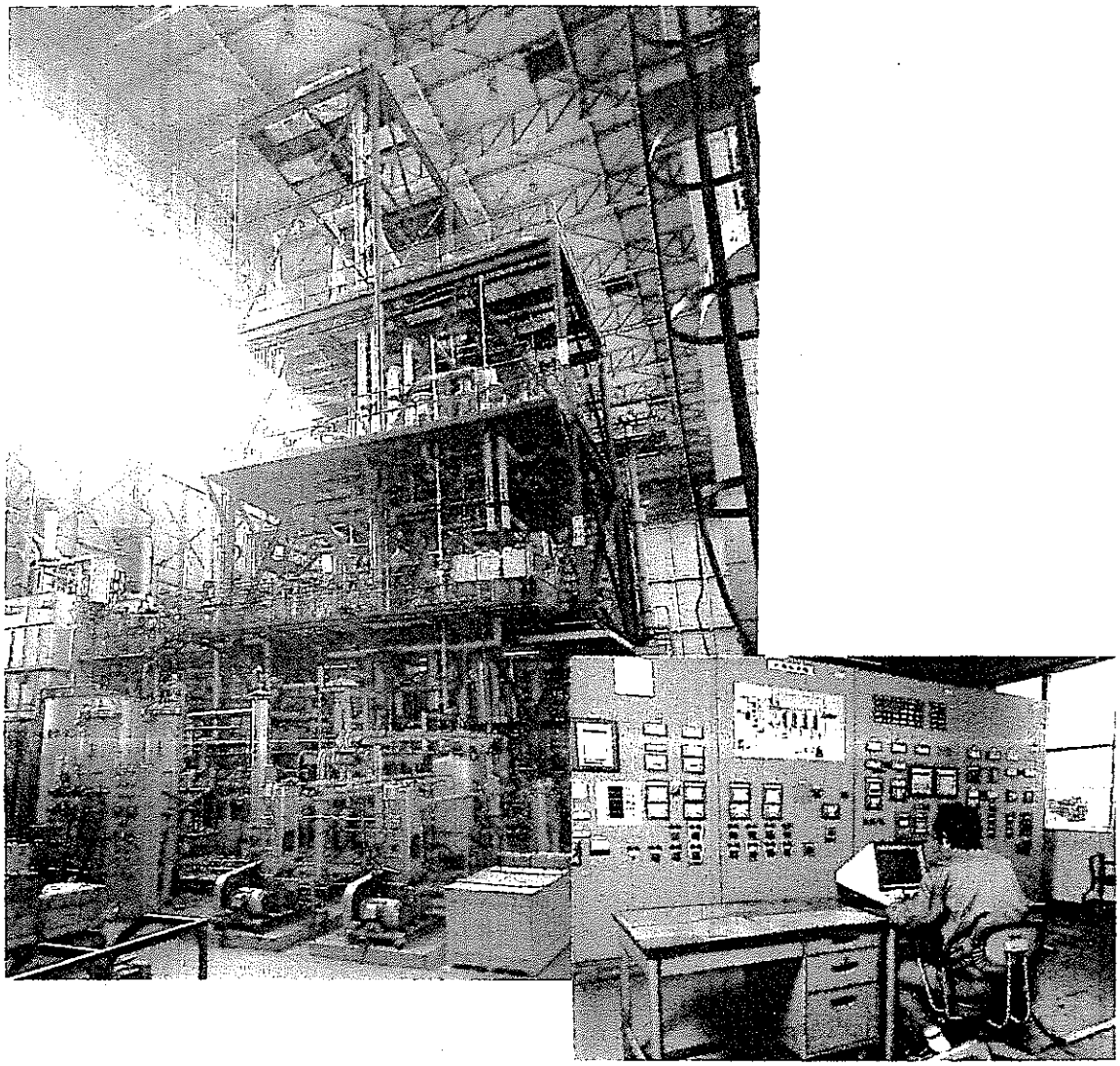


Fig.I.2-3 Component Testing Loop (CTL)

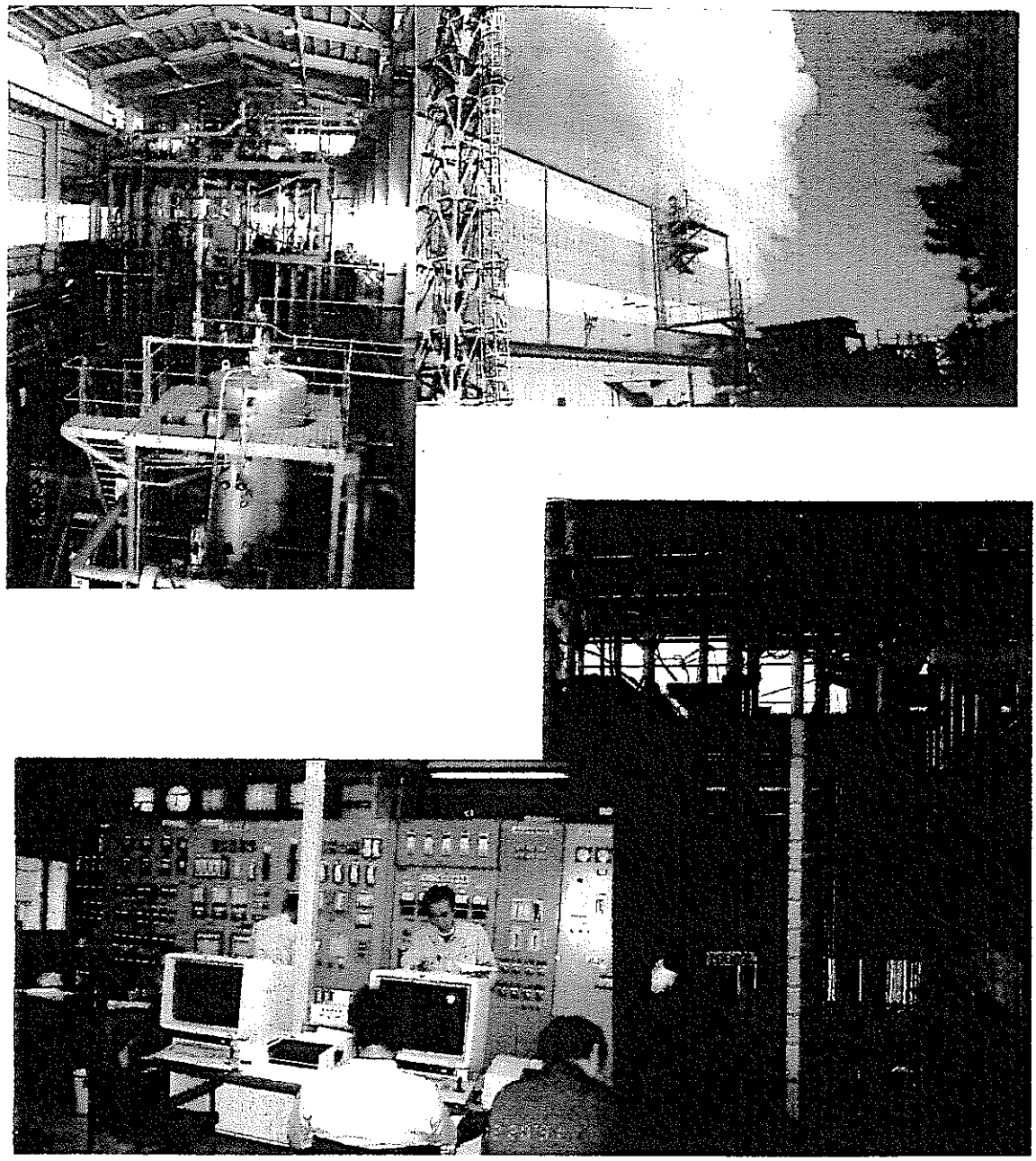


Fig.I.2-4 Large Scale Safety Testing Facility

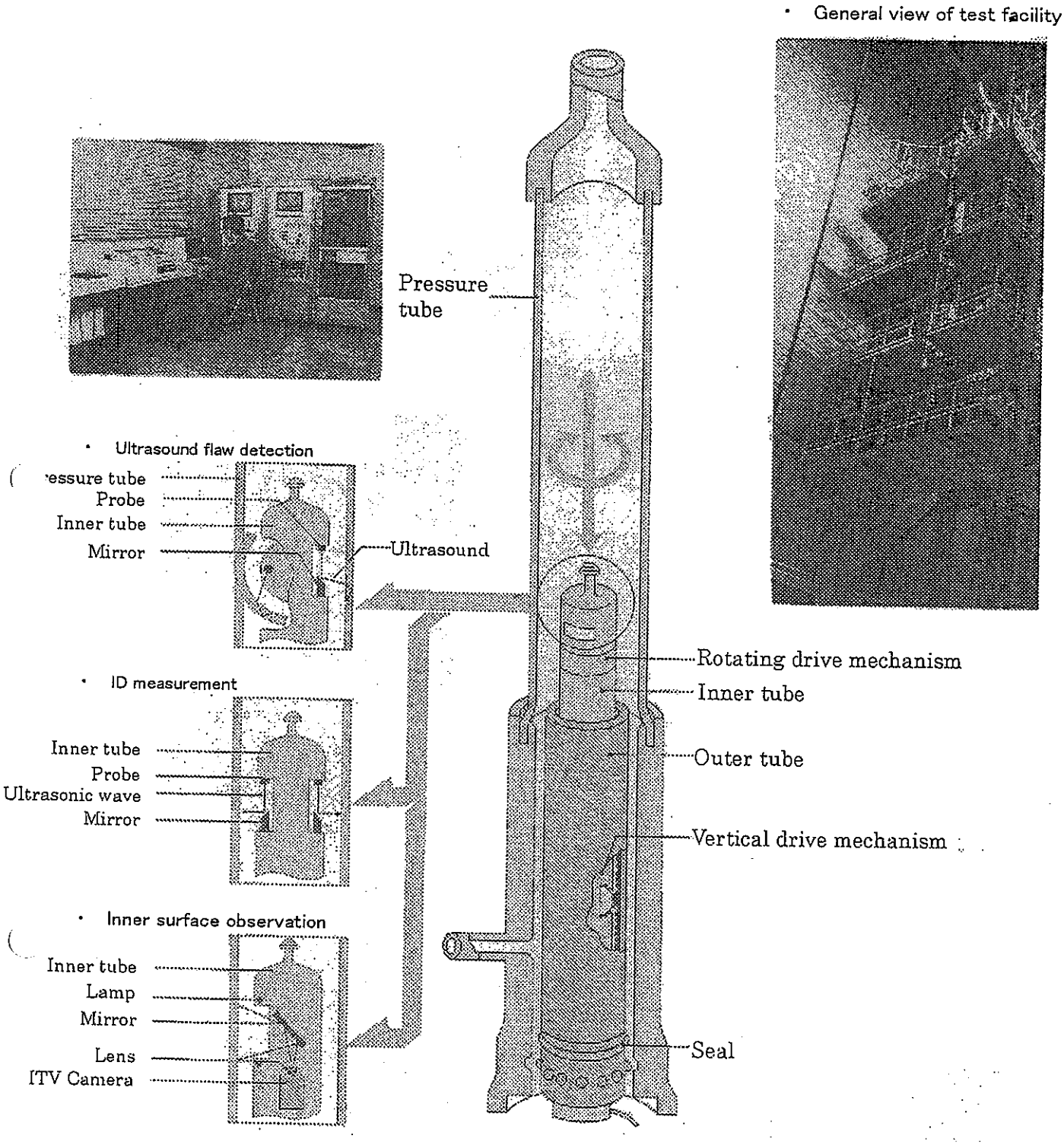


Fig.I.2–5 Pressure Tube Monitoring Equipment

UT speed in water is affected by water Temp. rather much

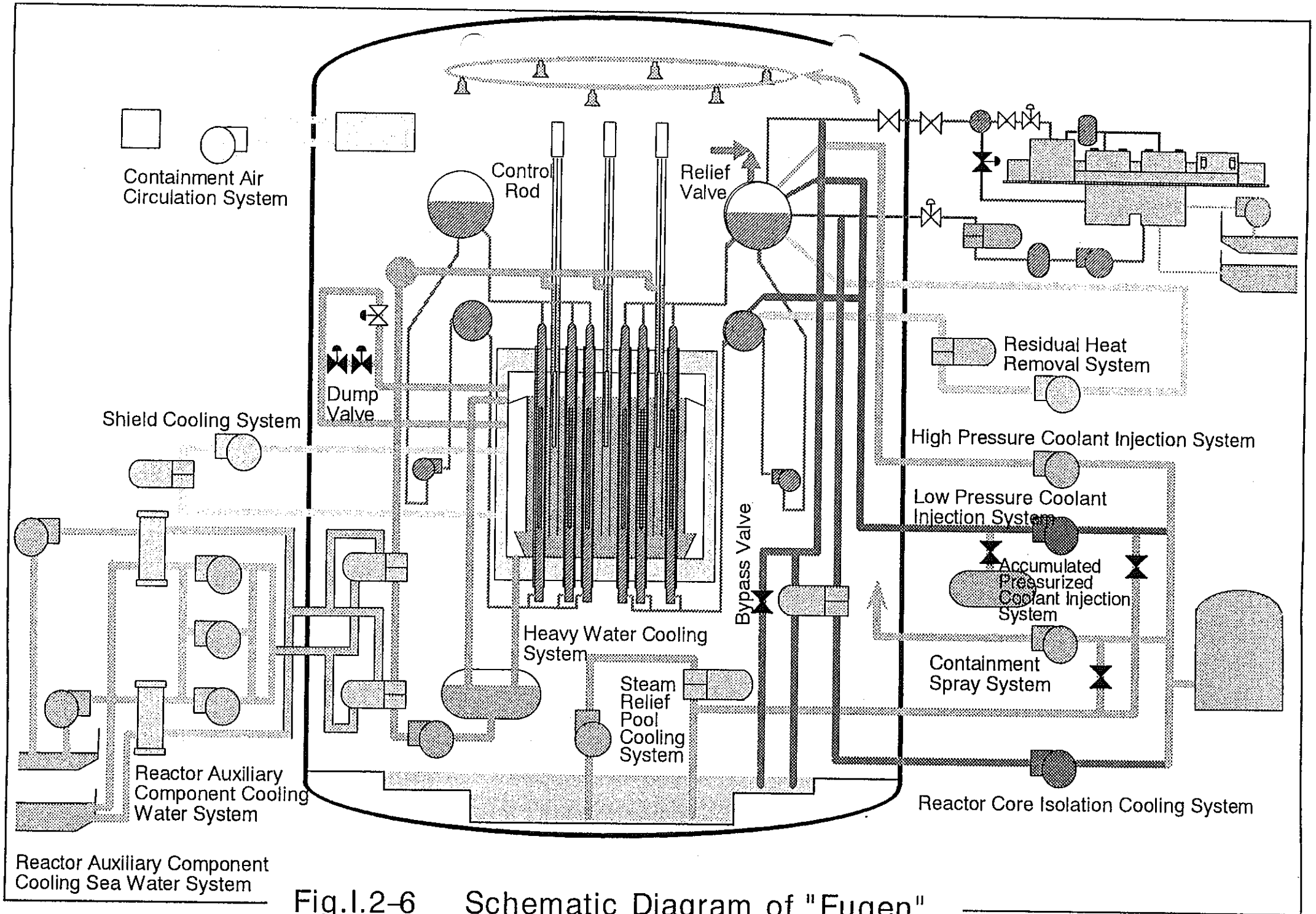


Fig.I.2-6 Schematic Diagram of "Fugen"

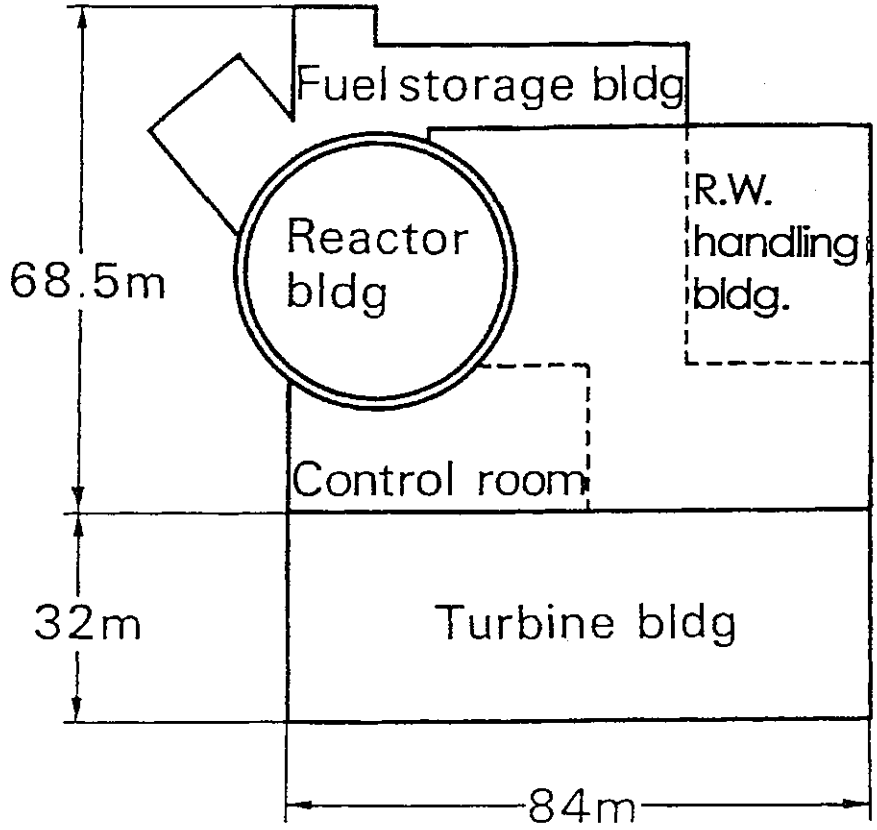


Fig.I.2-7 "Fugen" Plant Arrangement

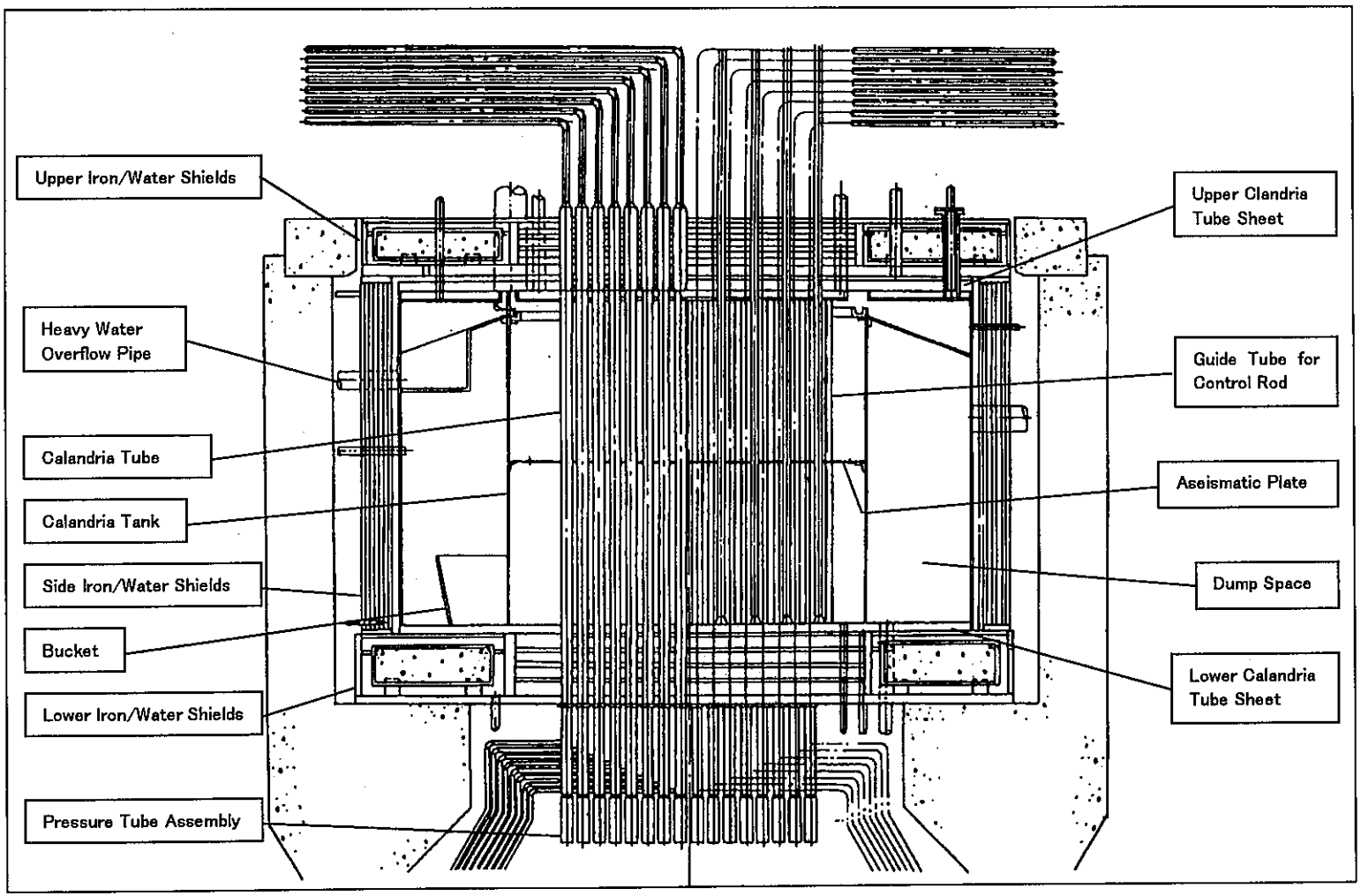


Fig.I.2-8 Cross Section of Reactor Proper

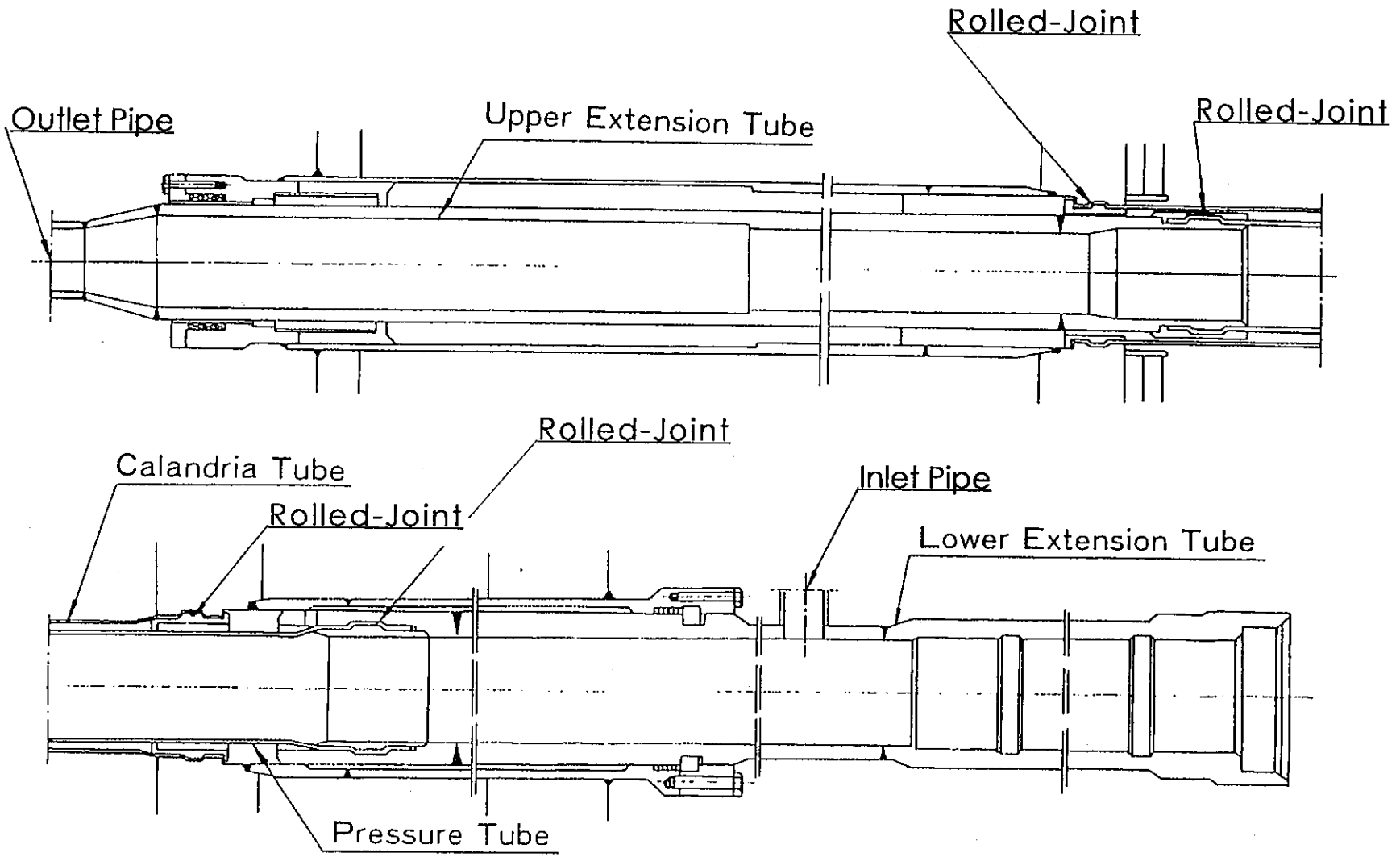


Fig.I.2-9 Pressure Tube Assembly

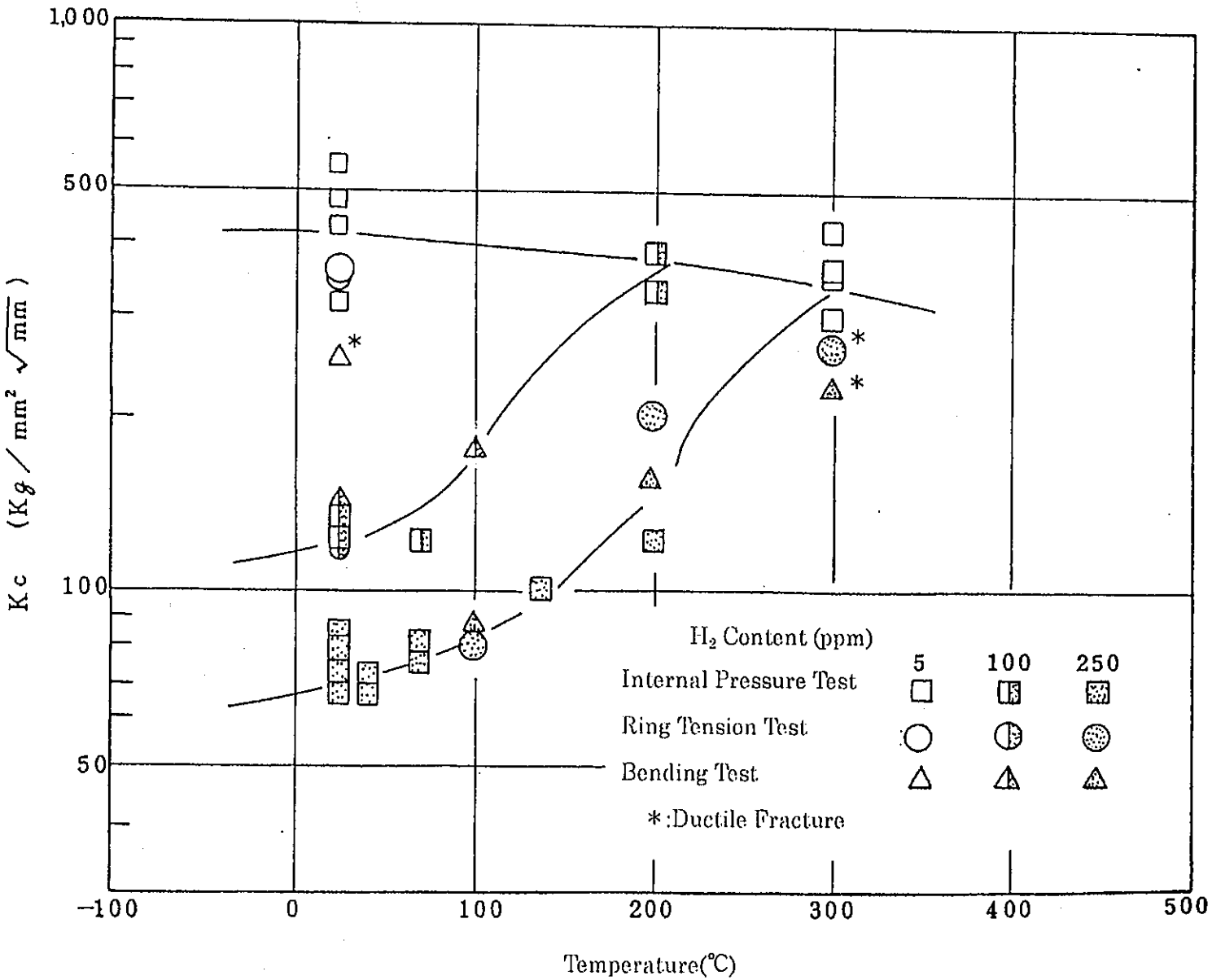


Fig.I.2-10 Fracture Toughness of Zr-2.5%Nb

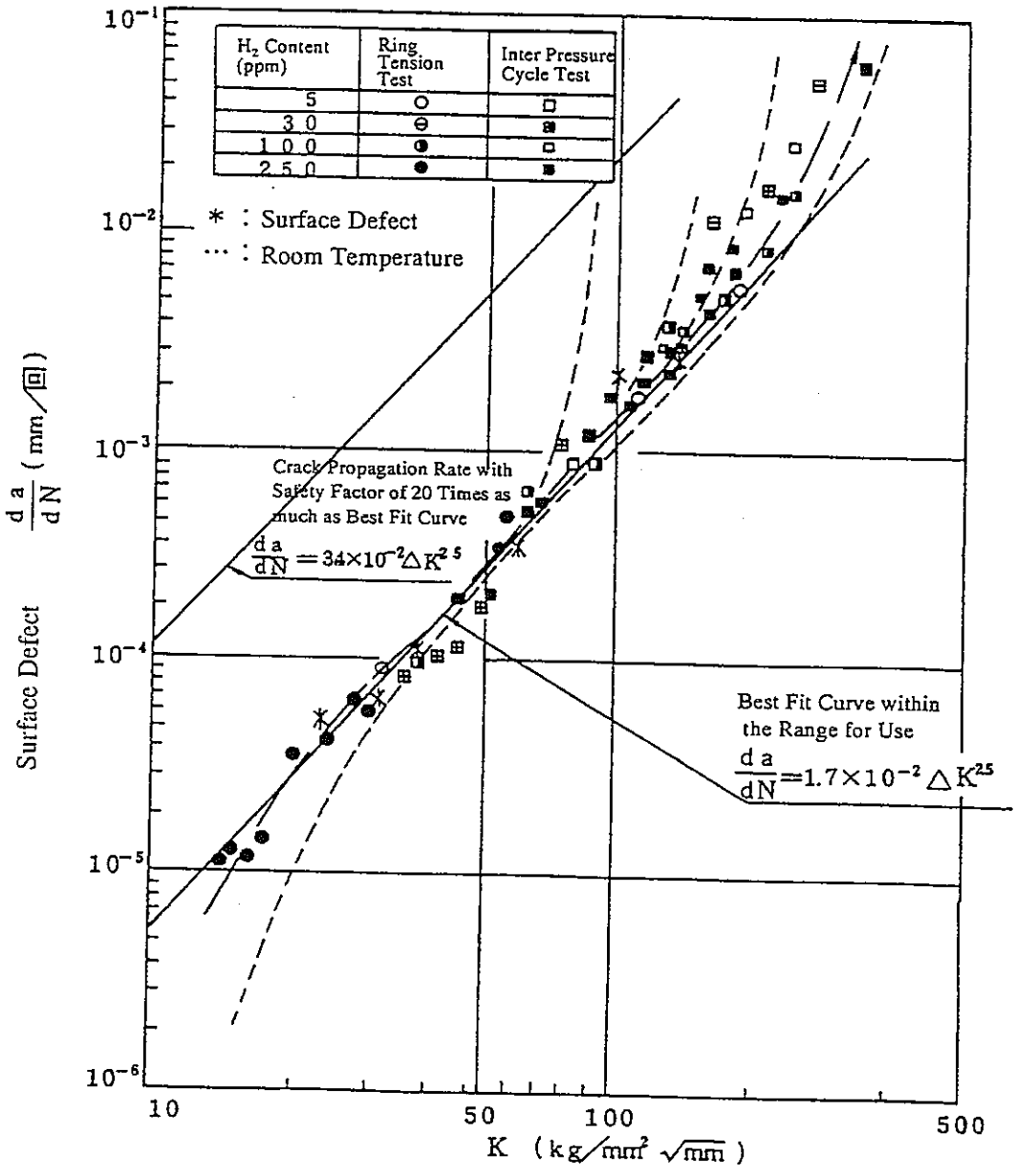


Fig.I.2-11 Fatigue Crack Propagation Rate of Zr-2.5%Nb (300℃)

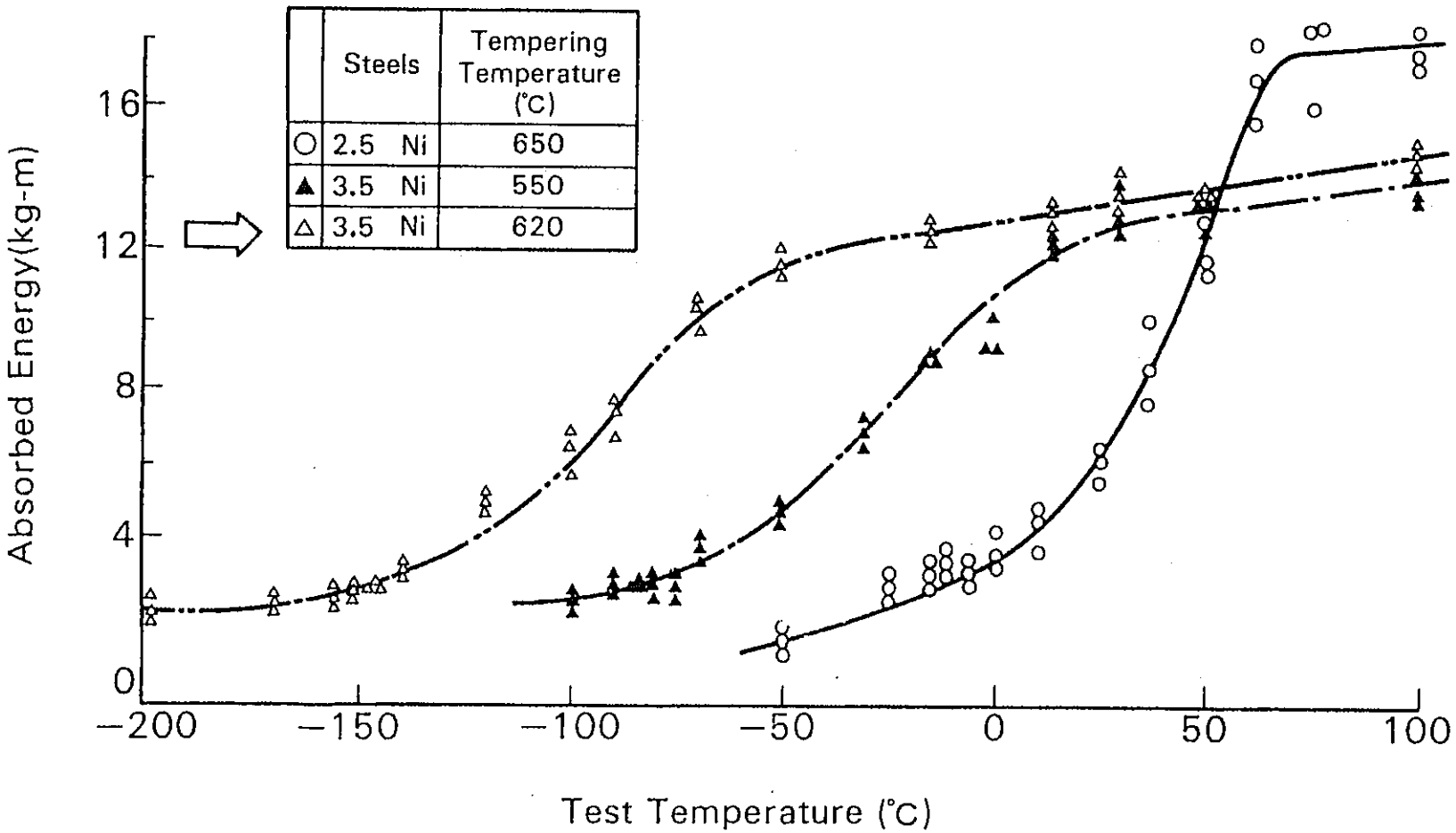


Fig.I.2-12 Transition Curves of 13%Cr Steel with 2.5%Ni and 3.5%Ni

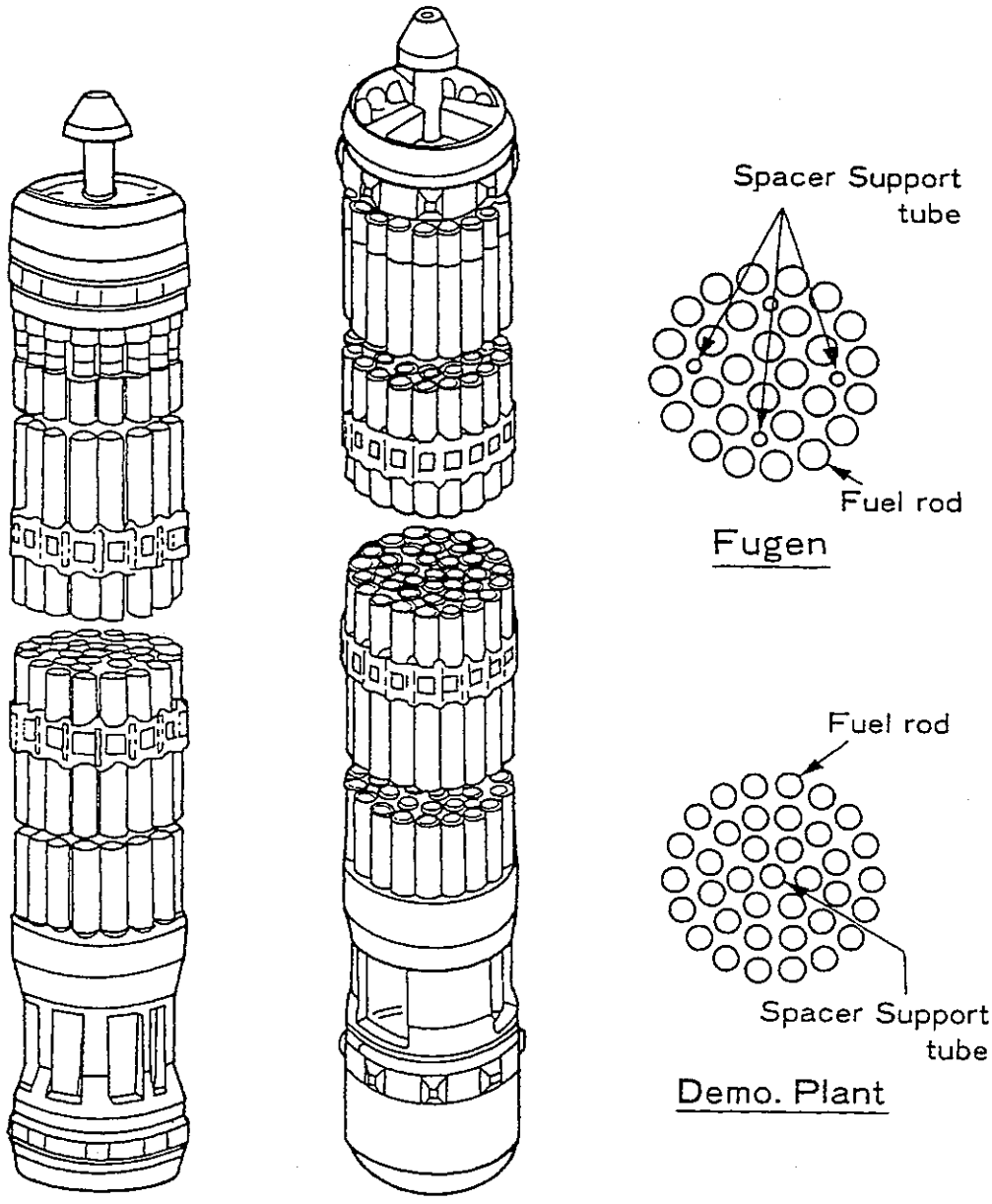


Fig.I.2-13 "Fugen" Standard Fuel Assembly

"Demo. Plant" Standard Fuel Assembly

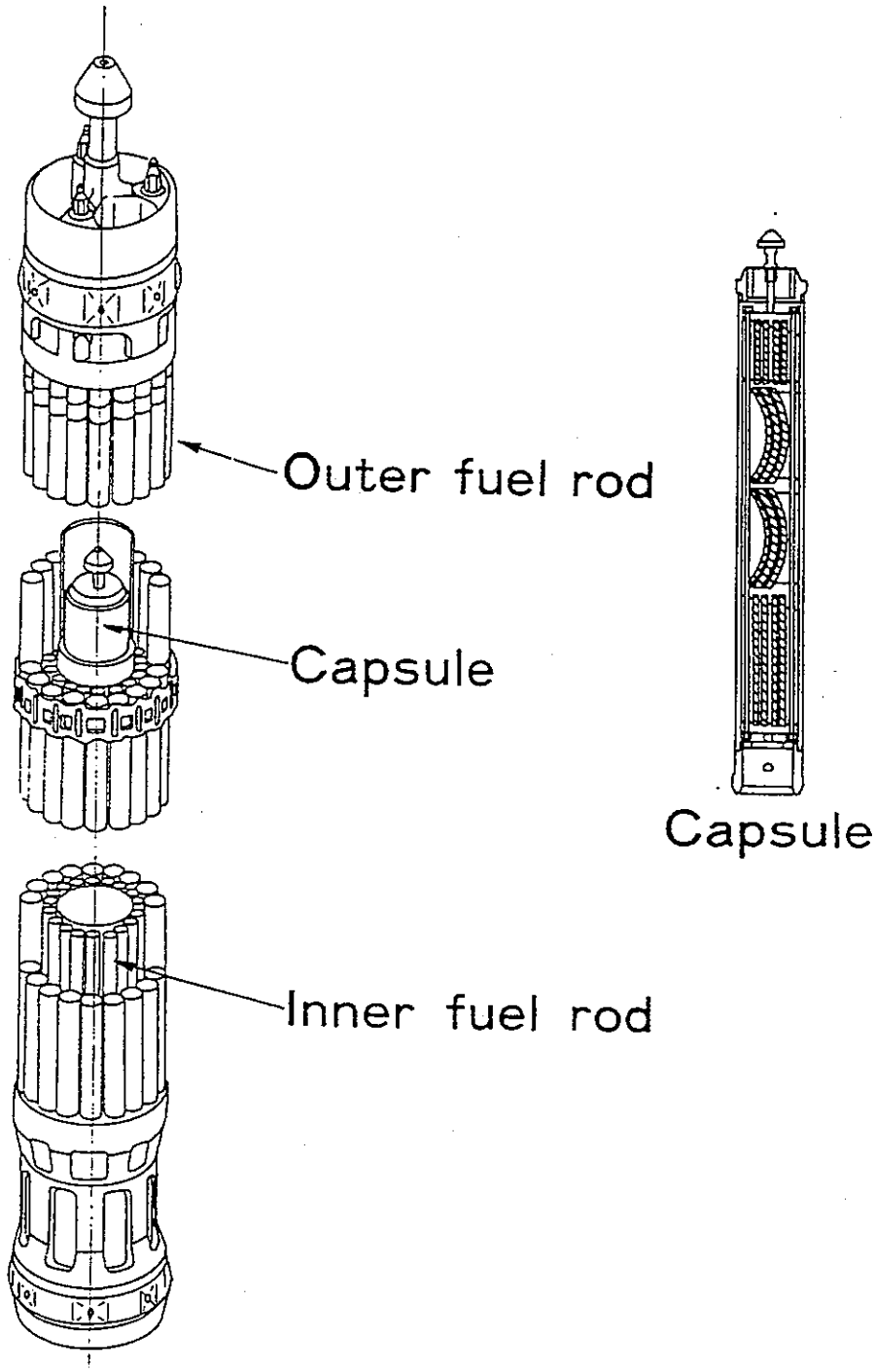


Fig.I.2-13A "Fugen" Special Fuel Assembly

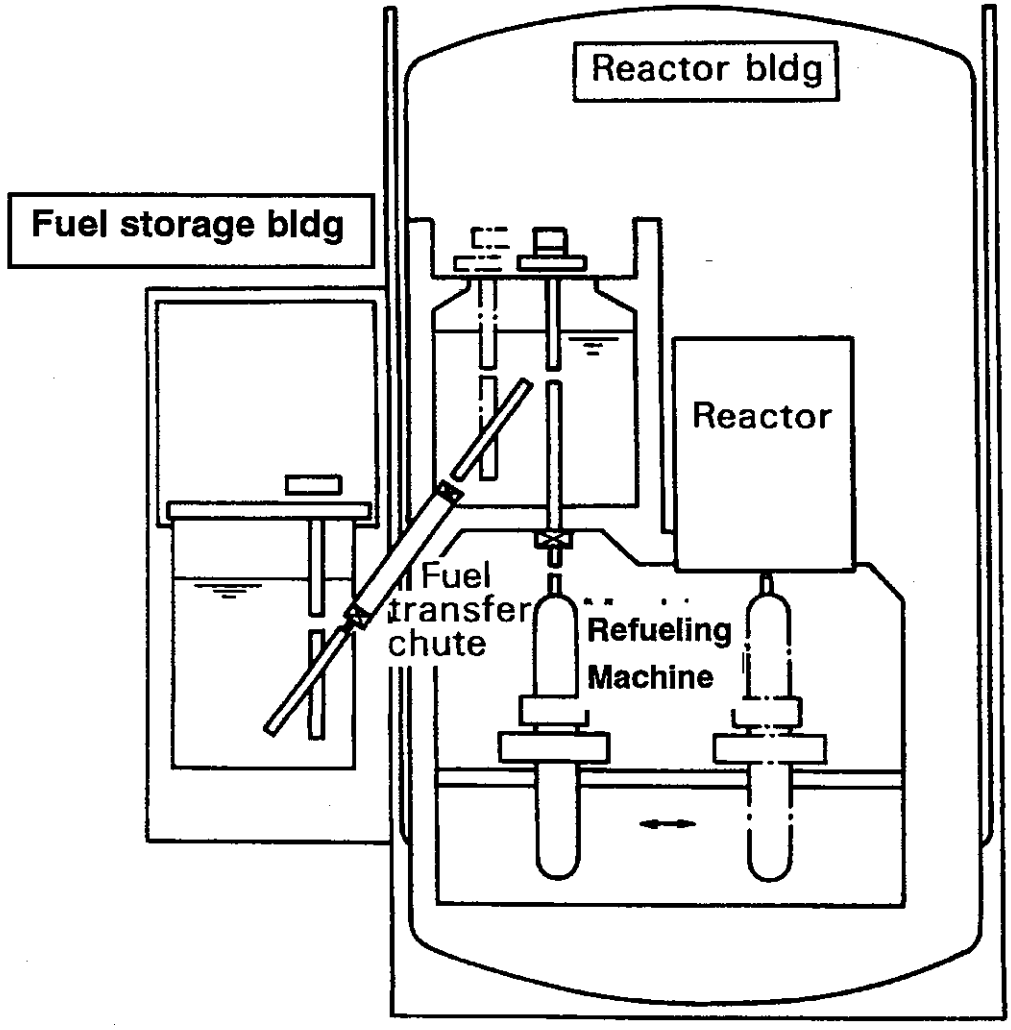


Fig.I.2-14 Fuel Transfer System

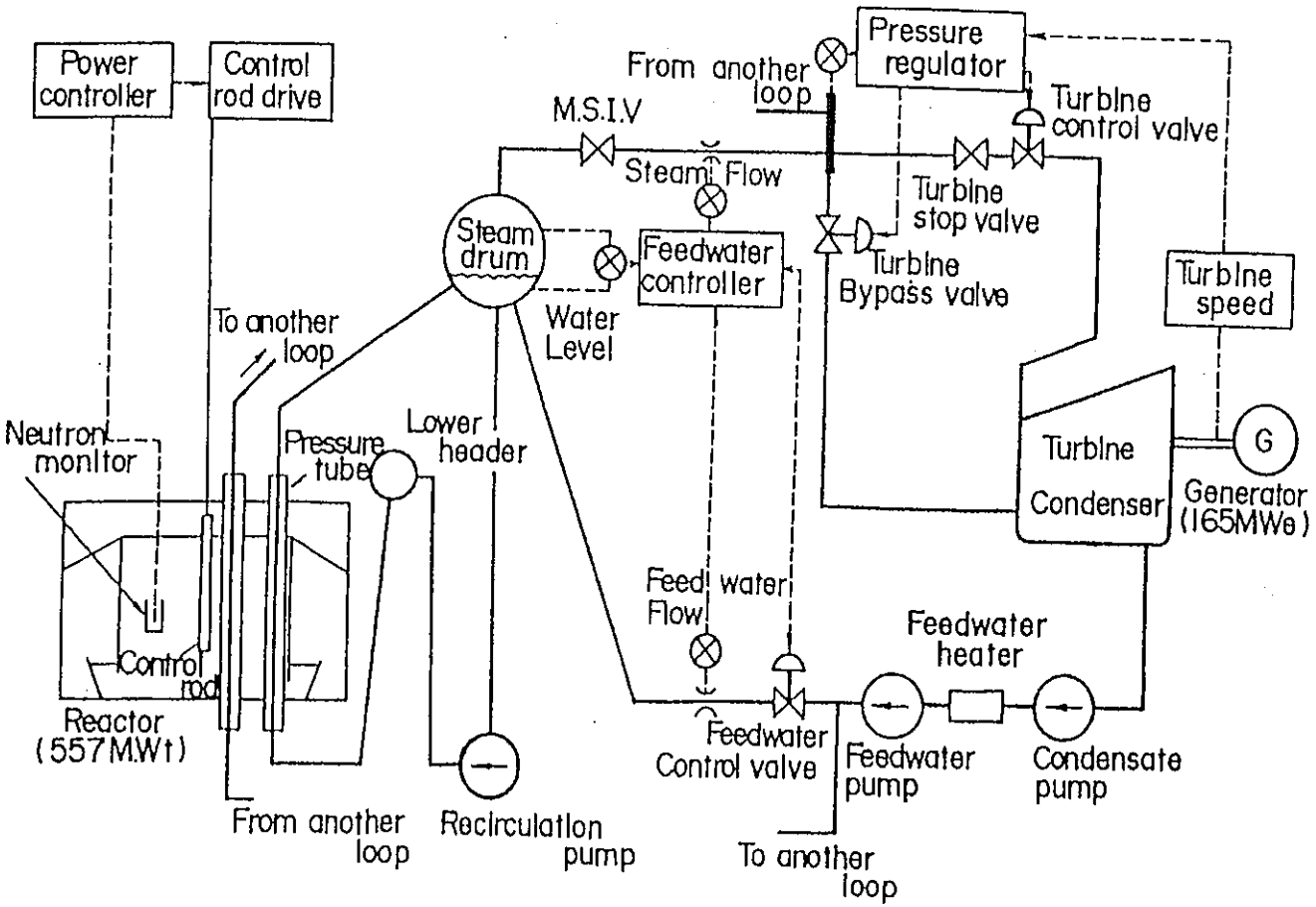
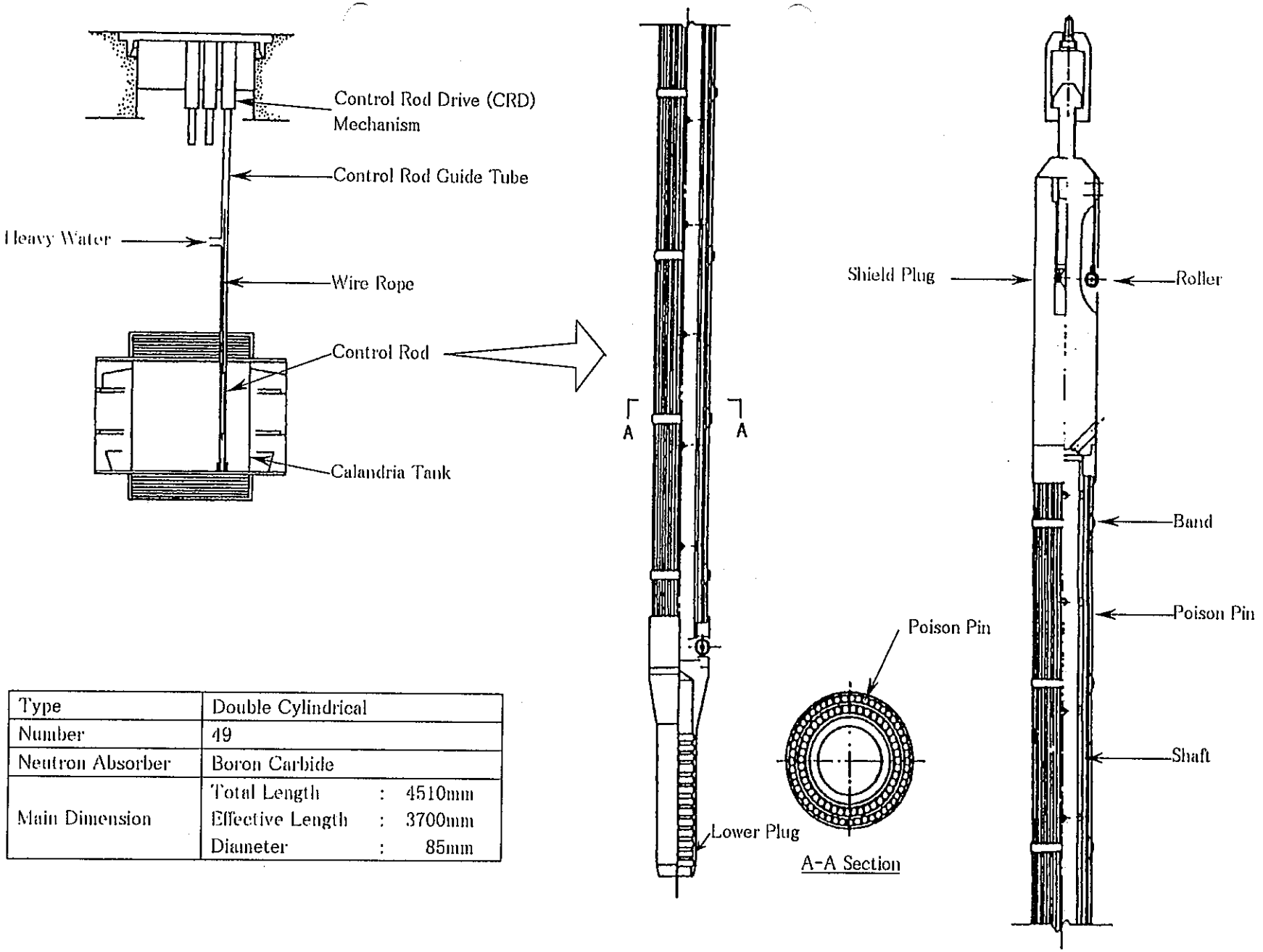


Fig.I.2-15 Plant Control System of "Fugen"



Type	Double Cylindrical	
Number	49	
Neutron Absorber	Boron Carbide	
Main Dimension	Total Length	: 4510mm
	Effective Length	: 3700mm
	Diameter	: 85mm

Fig.I.2-16 Control Rod Assembly

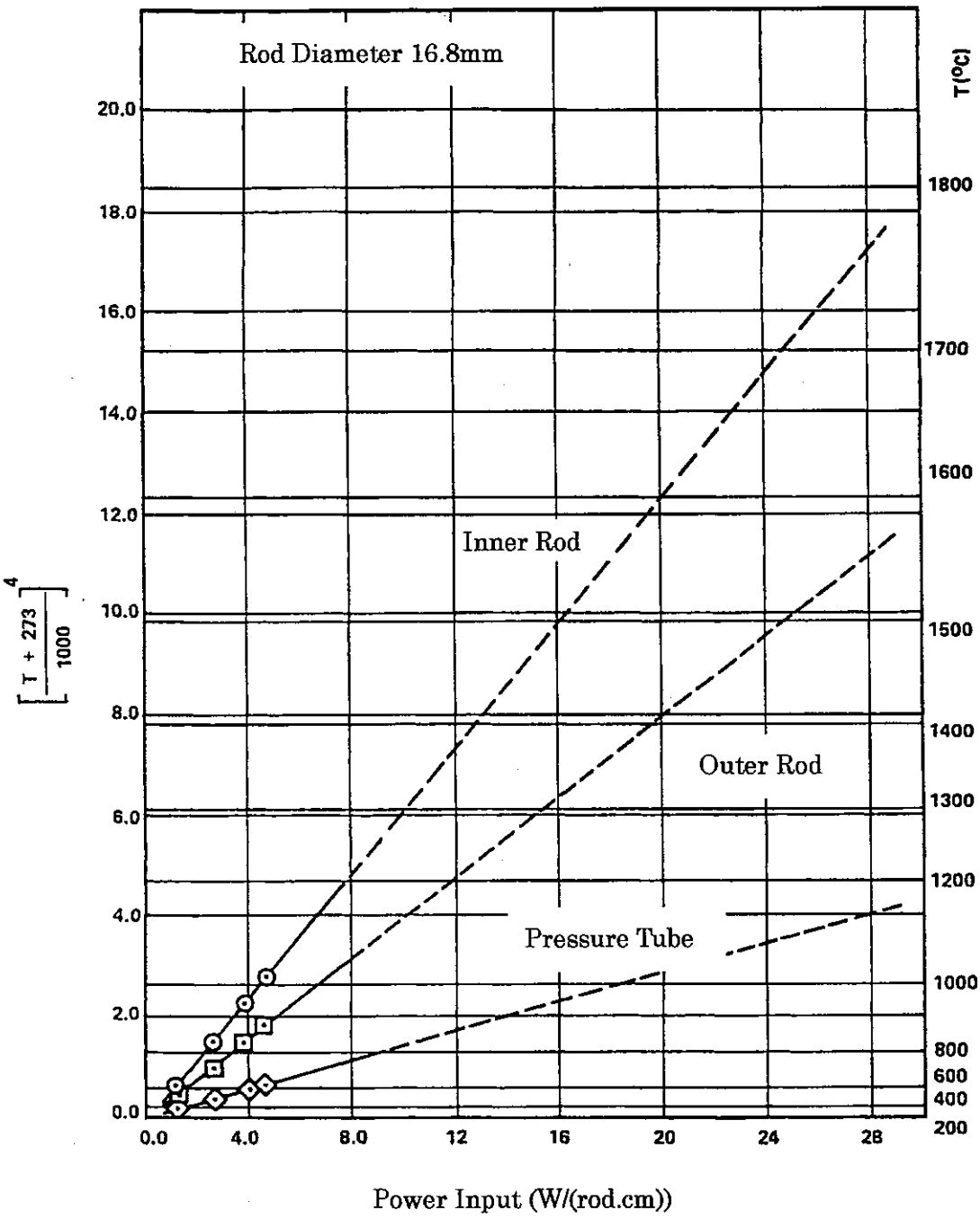


Fig.I.2-17 Decay Heat Removal Characteristics without Coolant in Core

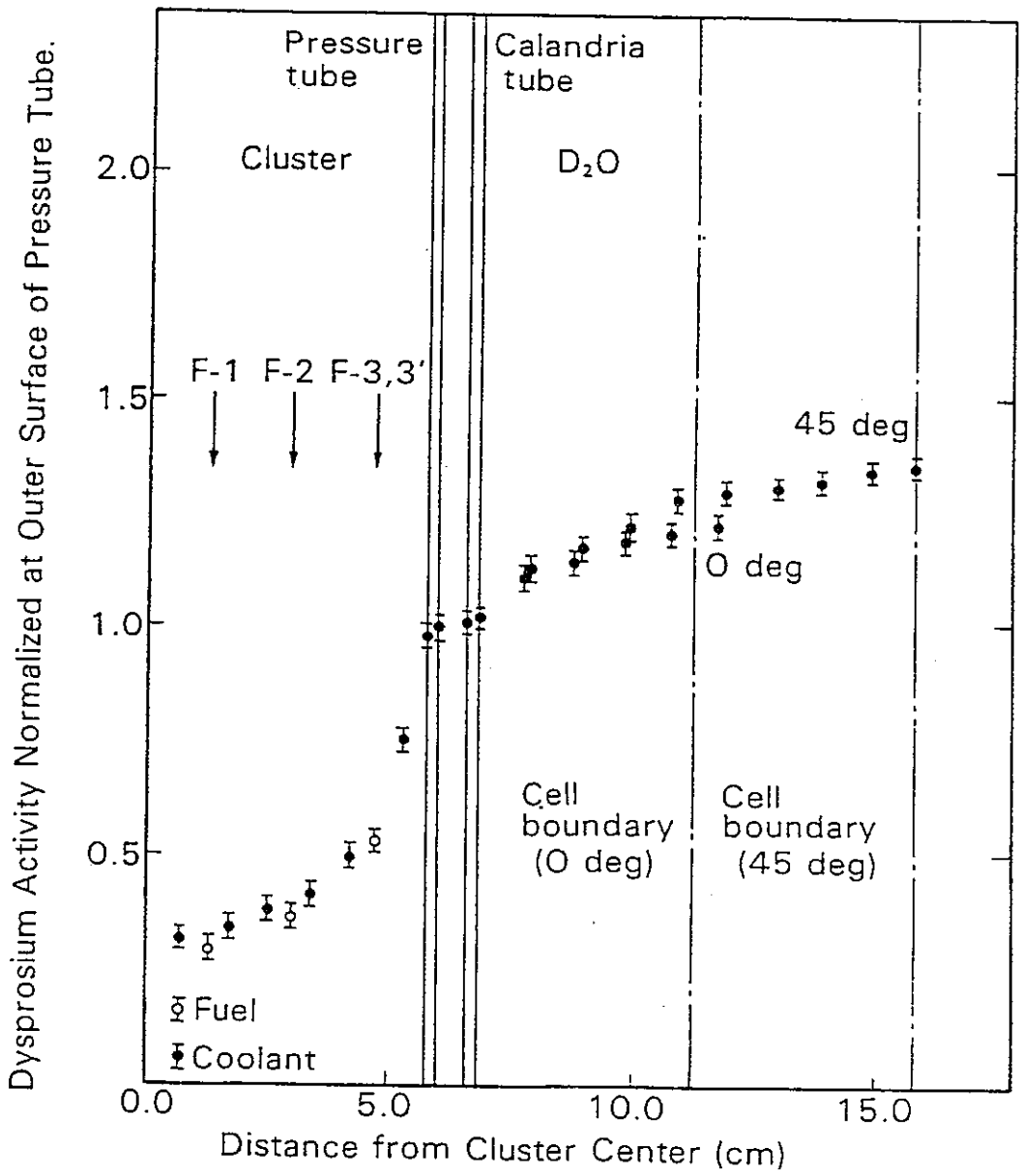


Fig.I.2-18 Dysprosium Reaction Rate in Unit Cell

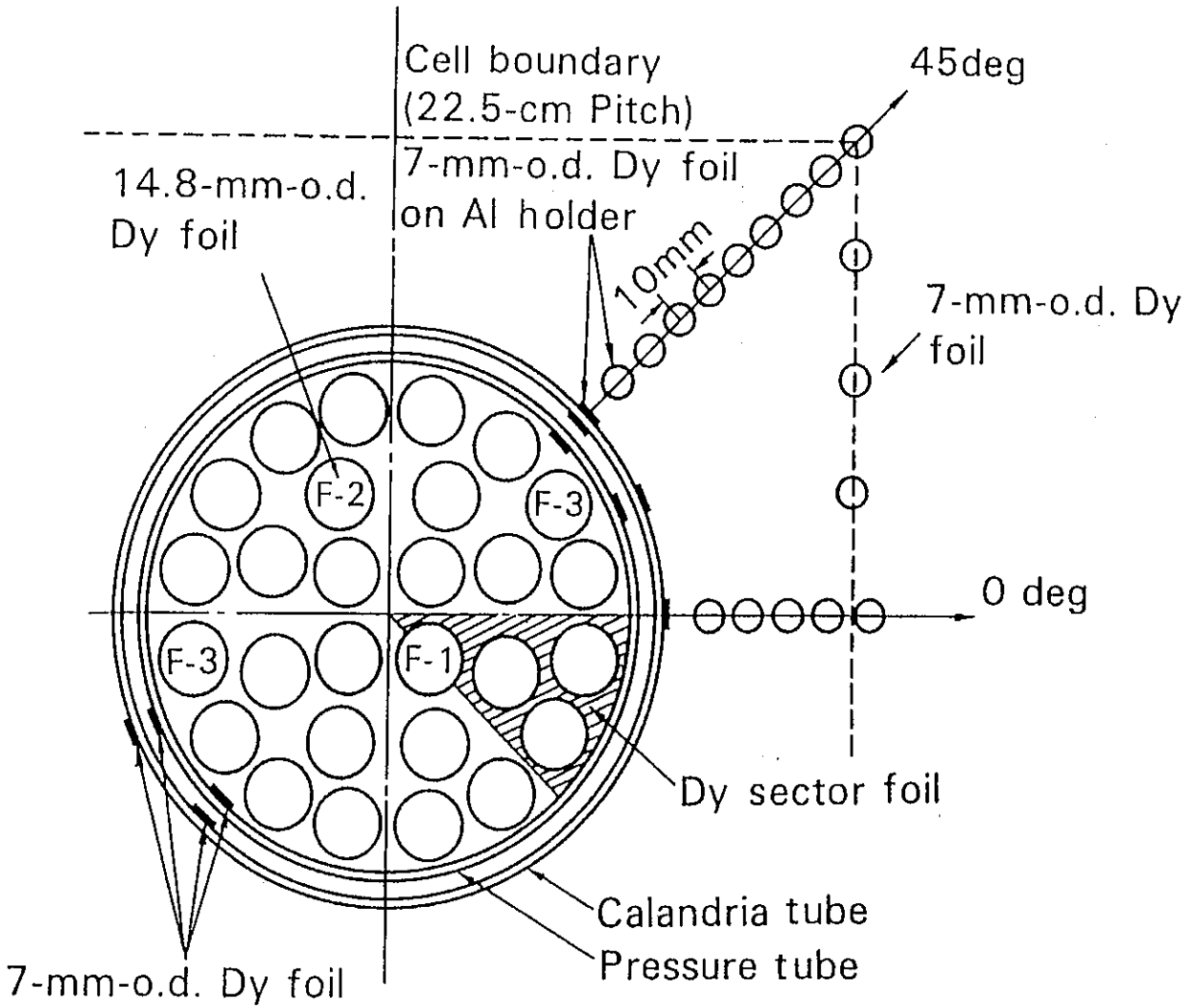


Fig.I.2-18A Arrangement of Dysprosium Foils in Unit Cell

Dy

元素의 일종

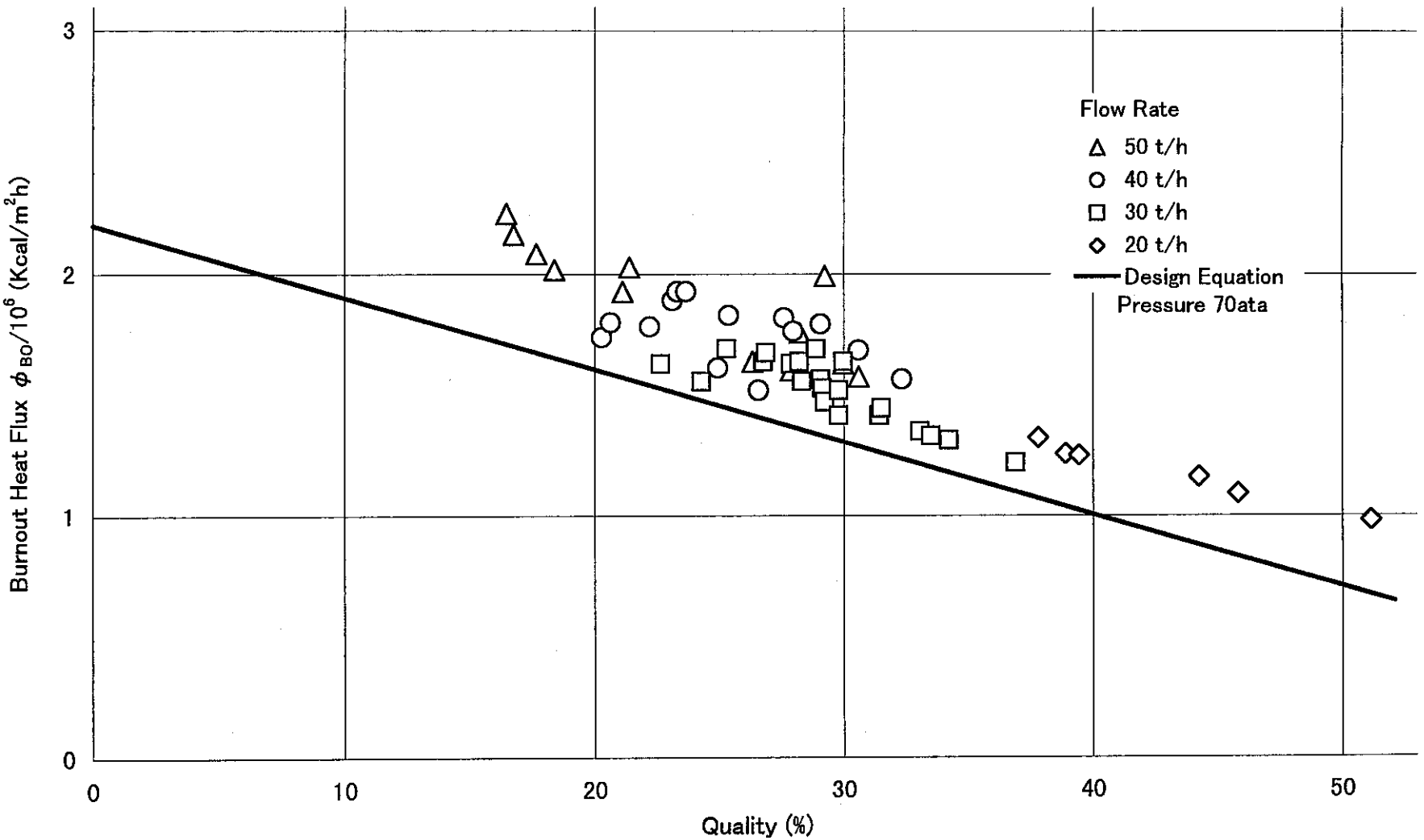
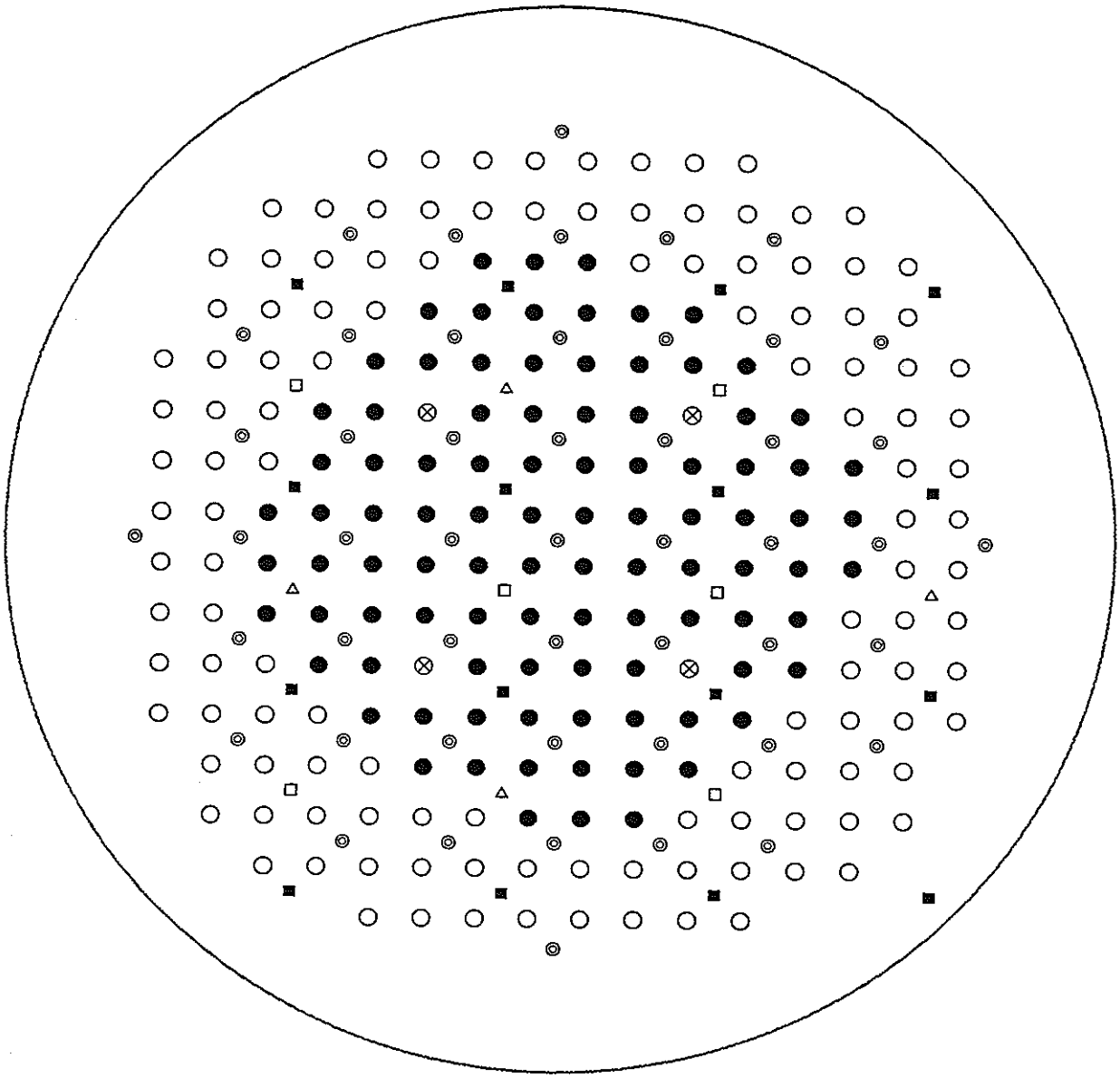


Fig.I.2-19 Burnout Characteristics Obtained in 14MW HTL



Note	Equipment	Number	Note	Equipment	Number
●	MOX Fuel Assenmbly	96	◎	Control Rod	49
○	UO2 Fuel Assembly	124	■	Local Power Monitor	16×4
⊗	Special Fuel Assembly	4	□	Power up Monitor	6
			△	Start up Monitor	4

Fig.I.2-20 Core Configuration (Initial Core)

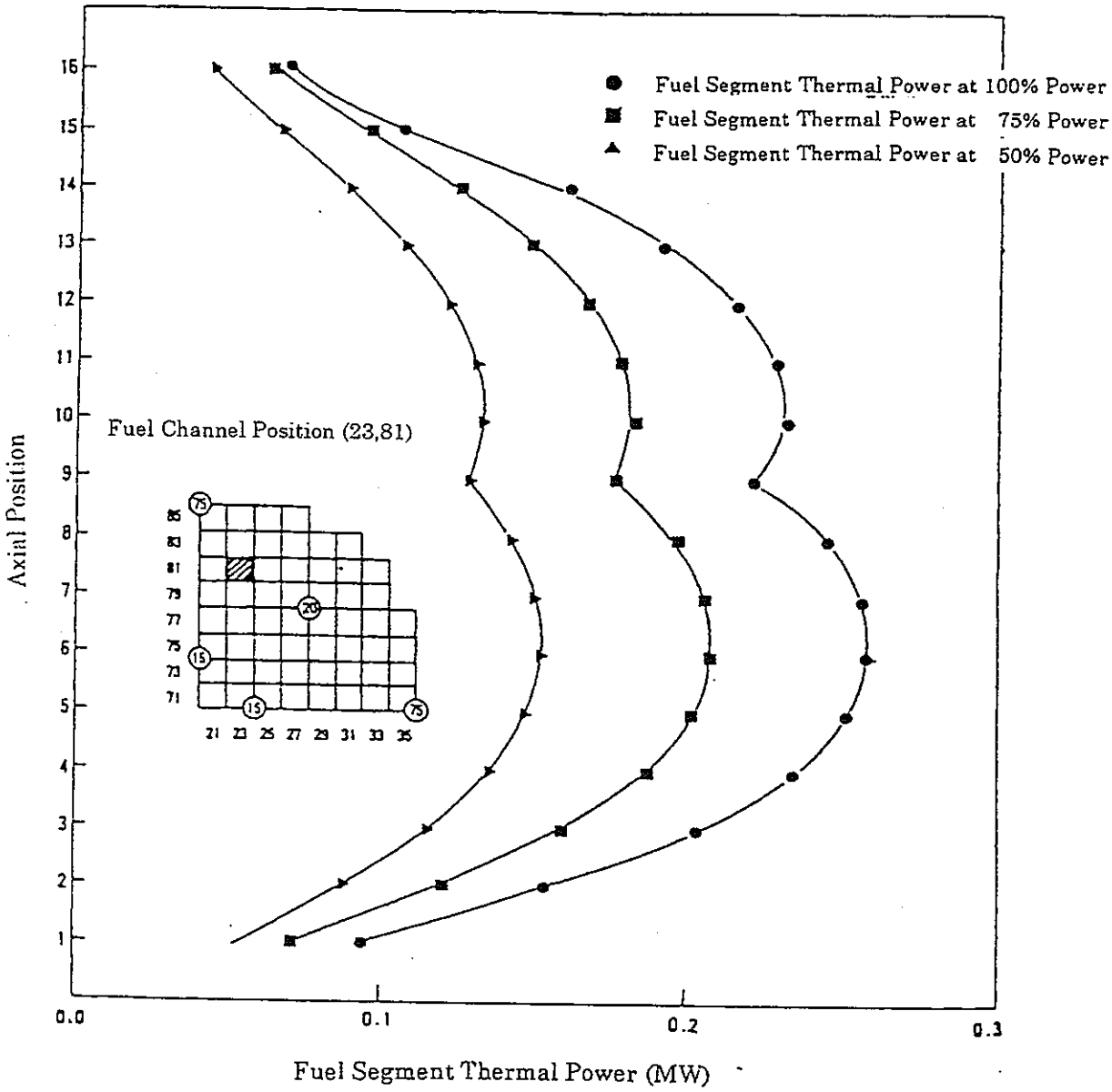


Fig.I.2-21 Fuel Segment Axial Thermal Power Distribution

MARK	UNO	TAG-NAME	SOKUTEI TAISYO	SCALE		(UNIT)
1	021	A-WLSET	(A-WATERLEVEL SET)	-350	- 50	(MM)
2	024	A-SD.WL	(A-STEAM DRUM WL)	-500	- 100	(MM)
3	003	A-FEED.F	(A-FEEDWATER FLOW)	0	- 200	(T/H)
4	009	A-SD.P	(A-STEAMDRUM PRES)	60	- 80	(ATG)

Remarks

S.D. : Steam Drum

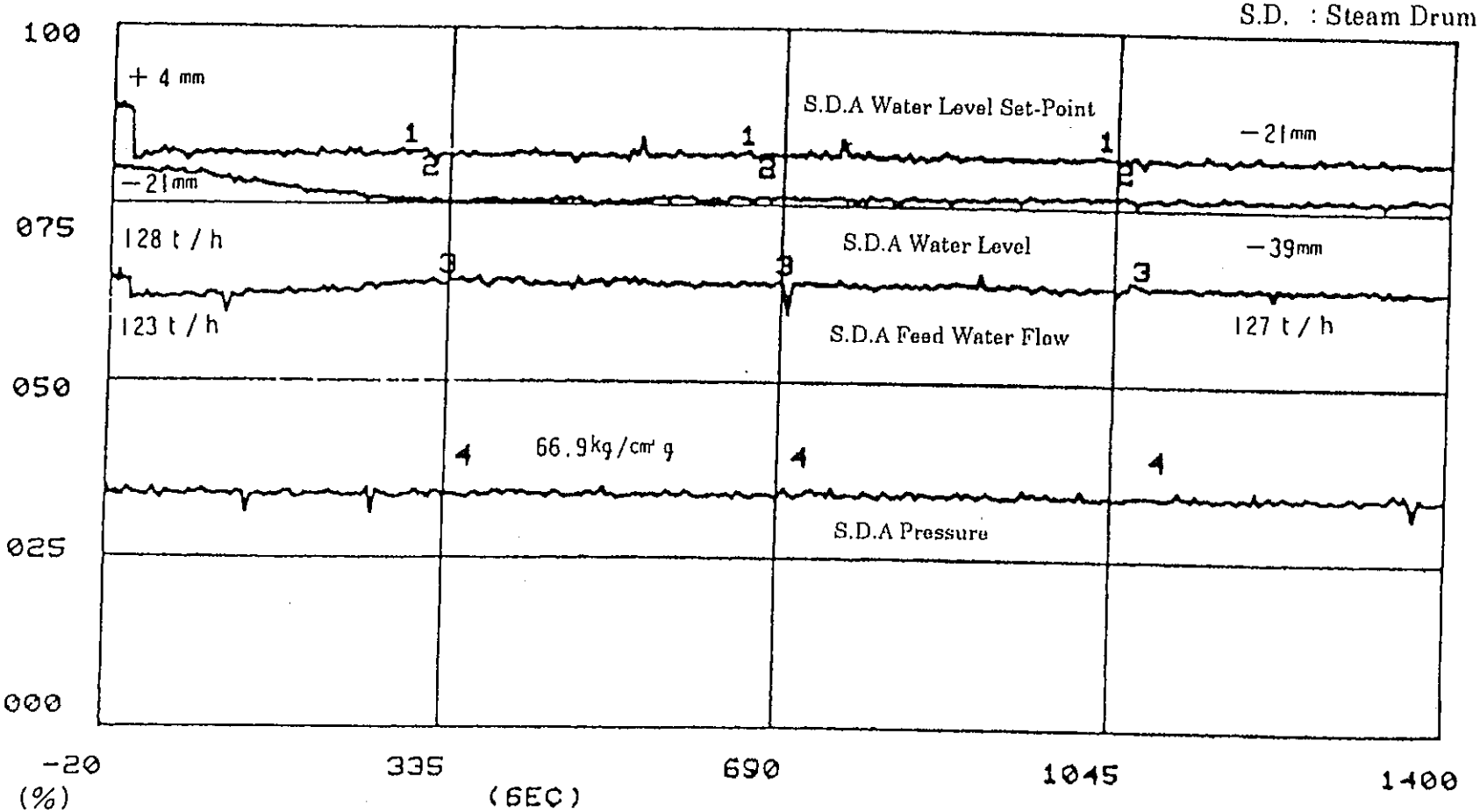
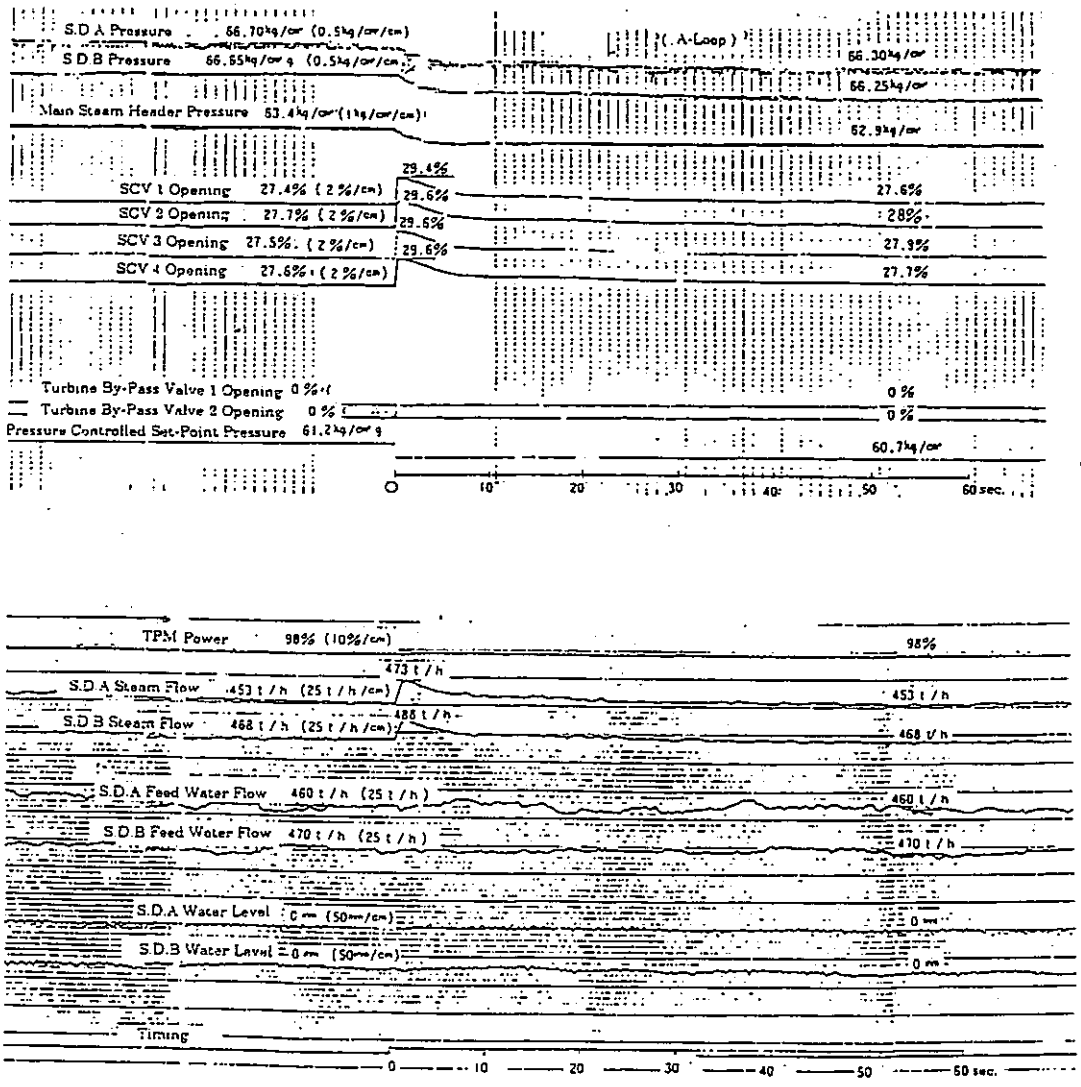


Fig.I.2-22 Steam Drum Water Level Set-Point Change Test

(Test Power : 25% Electric Power, Test Date : 1978/8/8)



Remarks

S.D. : Steam Drum

TPM : Total Power Monitor

SCV : Steam Control Valve

Fig.I.2-23 Steam Drum Pressure Set-Point Change Test (0.5kg/cm^2 Lowering)

(Test Power : 100% Electric Power, Test Date : 1979/1/22)

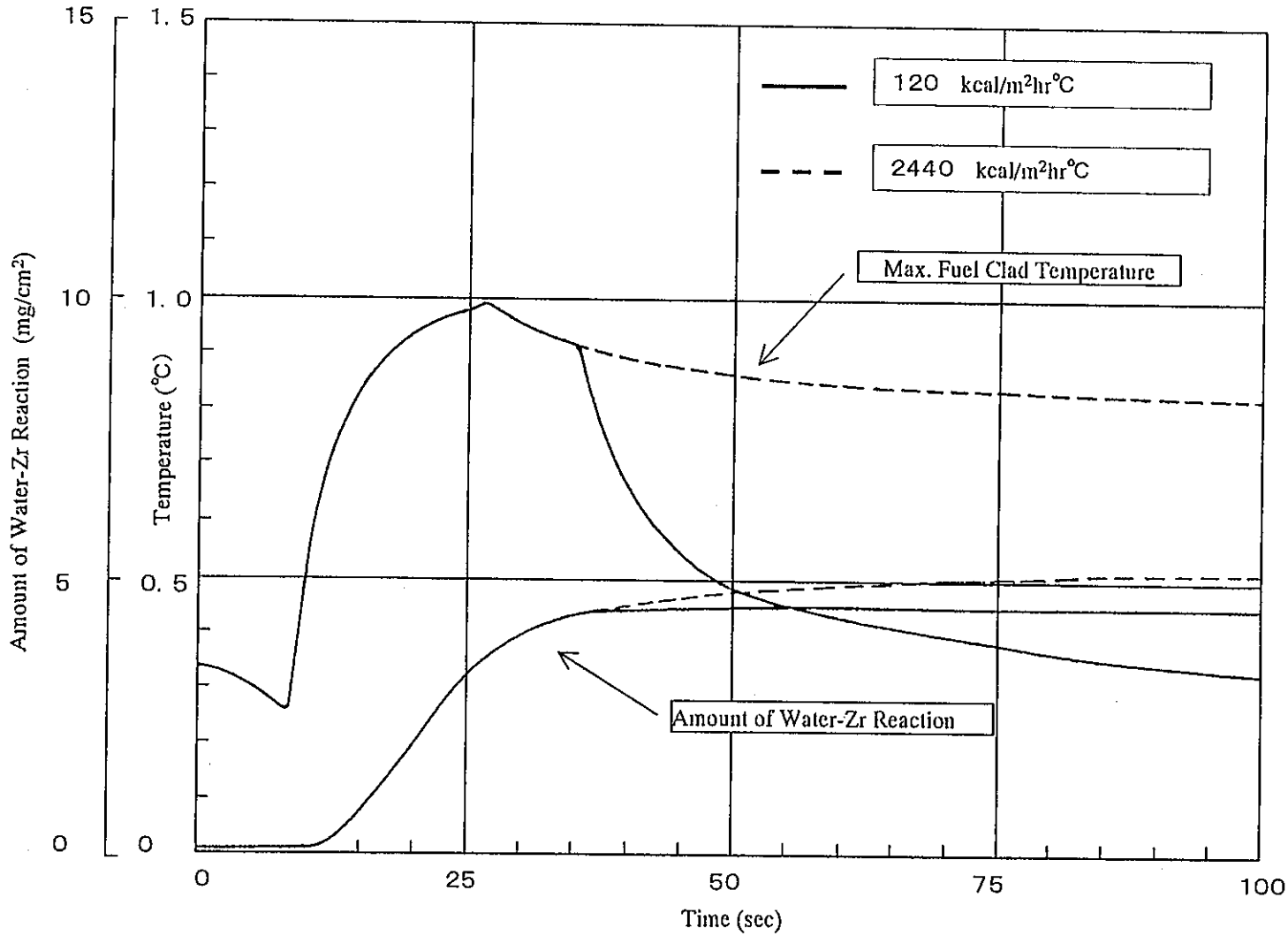


Fig.I.2-24 Safety Analysis on Large Loss of Coolant Accident

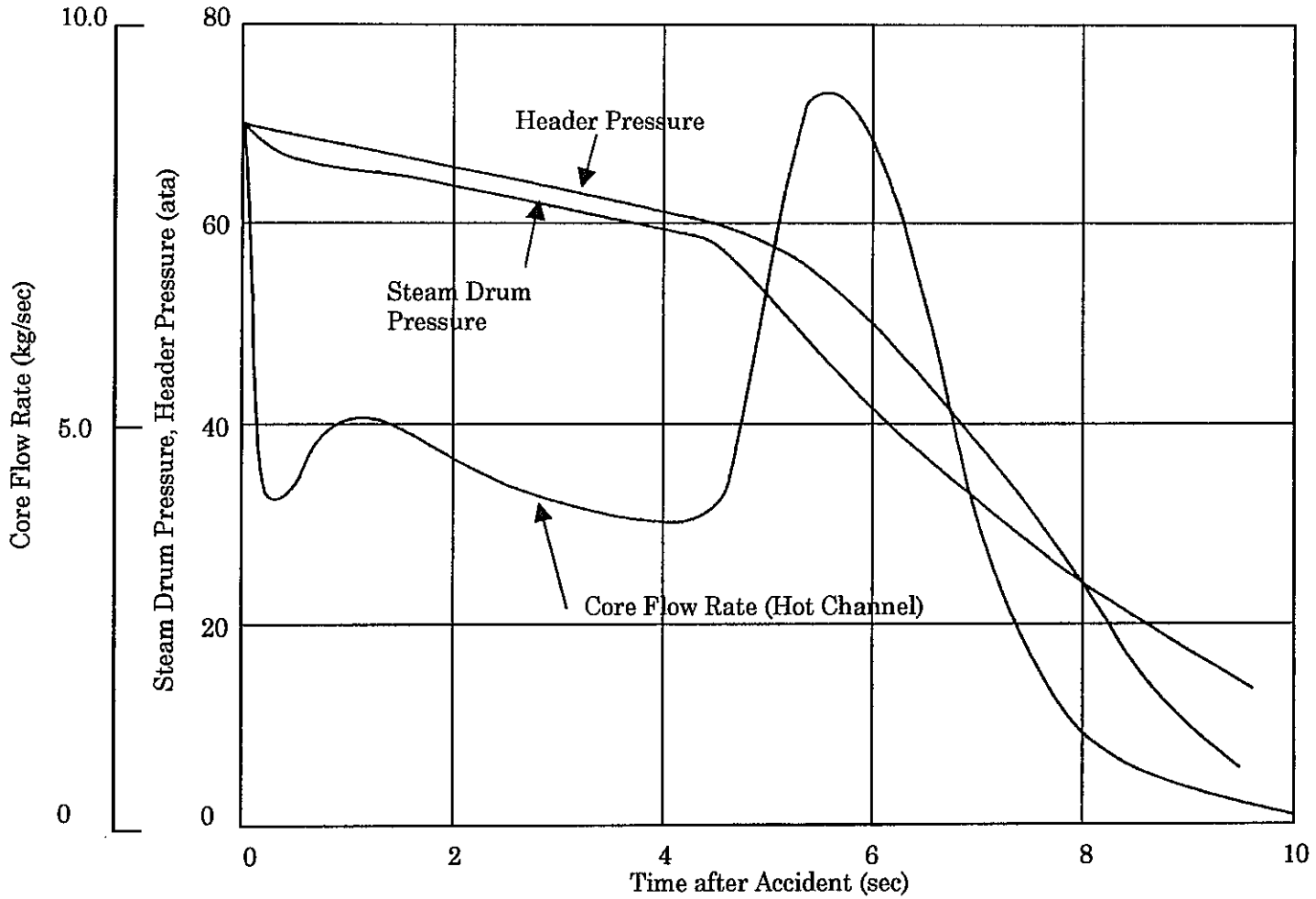


Fig.I.2-24A Safety Analysis on Large Loss of Coolant Accident

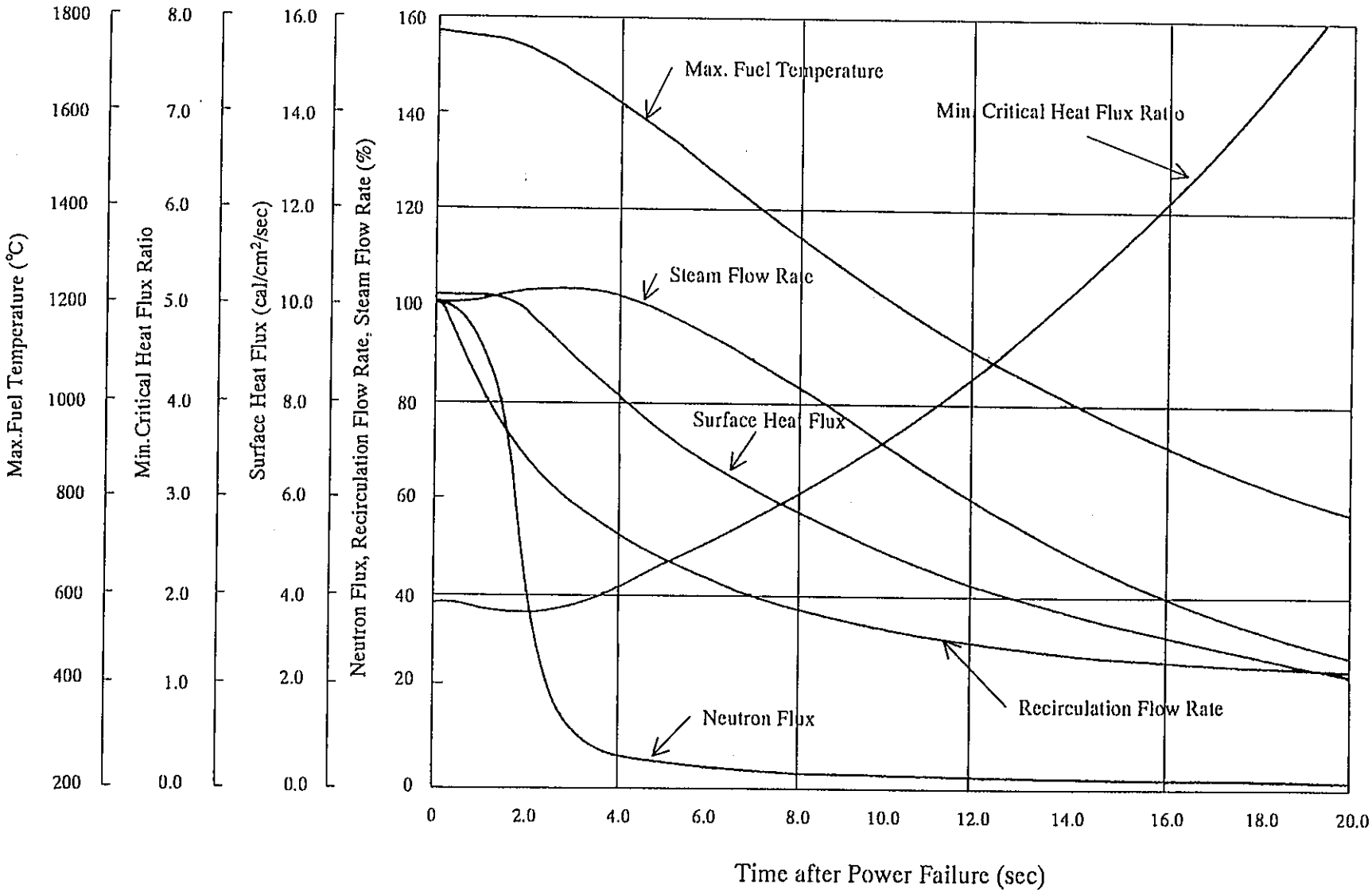


Fig.I.2–25 Safety Analysis on Power Failure for 4 Recirculation Pumps

Table I.2-1 Plant Data of "Fugen"

Location	Tsuruga, Fukui prefecture
Owner and operator	Power Reactor & Nuclear Fuel Development Corporation
Designers	Power Reactor & Nuclear Fuel Development Corporation
Number of Reactors in Plant	One
Rated output per reactor	Gross Thermal power 557MW Gross electrical 165MW Net electrical 148MW Self consumption 17MW
Fuel	Slightly enriched (1.5-1.9%) UO_2 and MOX (0.7-1.3% Puf)
Moderator and reflector	Heavy water (D_2O)
Heat Transport Fluid	Boiling light water
Type	Vertical pressure tube
Construction schedule	Start of construction December, 1970
In-service dates	March, 1979
Buildings and structures	
Reactor building	
Form	Cylindrical wall with elliptical concrete dome
Material	Reinforced concrete
Design pressure	1.35kg/cm^{2}G
Internal diameter	36m
Height of cylindrical portion	67.5m
Wall thickness	1,050mm
Reactor auxiliary building	
Form	Five-storey steel frame
Materials (floors)	Reinforced concrete
Length of building	40m
Turbine building	
Form	Three storey Steel frame
Length	84m
Width	32m
Height	30.4m
Service wing	
Form	Five-storey steel frame
Length	84m
Width	21m
Reactor vessel	
Pressure tubes	
Quantity	224
Material	Zr-2.5%Nb
Inside diameter	117.8mm
Wall thickness	4.3mm
Length	4,770mm

Calandria tank	
Form	Cylindrical vessel
Material	Austenitic Stainless Steel
Height	5,040mm
Outside diameter	
Inner zone	4,910mm
Outer zone	7,950mm
Plate thickness	
Inner zone	25mm
Outer zone	27mm
Radiation shields	
Form	Cylindrical vessel
Material	Carbon steel and H_2O
Height	6,980mm
Outside diameter	9,450mm
Calandria tubes	
Quantity	224
Material	Zircaloy-2 seam welded
Inside diameter	156.4mm
Wall thickness (nominal)	1.9mm
Dump tank	
Form	Cylindrical vessel
Material	Austenitic stainless steel
Inside diameter	2,700mm
Length	6,266mm
Wall thickness	18mm
Reactor physics	
Core data	
No. of fuel channels	224
Cell array	approximately circle 16x16
Lattice pitch	240mm
Core radius	4,050mm
Core length	3,700mm
Average reflector thickness at midpoint	400mm
Fuel load (MOX)	38.3/t
Average thermal flux in core	5x10^{13}n/cm^2・sec
Burnup (thermal energy)	20,000MWd/t (for MOX fuel)
Blanket gas	Helium
Control Rods	
Material	Born carbide in stainless steal tube
Purposes	Power regulating, neutron flux flattening and reactor shutdown
Number	49 (4 for power regulating 0-5 for neutron flux flattening and residual rods for reactor shutdown)
Orientation	Vertical

Liquid poison	
Material	Enriched-boron dissolved in moderator
Purposes	Suppression of excess reactivity for burnup
Concentration	Below 10ppm as ^{10}B
Other shutdown provisions	Dumping of D_2O moderator by gas balance scheme
Moderator	
Type	Heavy water
Total volume in core	$81.5m^3$
Purity	99.75%D_2O by weight
Temperature, maximum	70℃
Heat produced in moderator	33MWth
Flow through Calandria	$1,320m^3/hr$
Moderator temperature rise	13℃
Helium pressure in dump tank	$0.54kg/cm^2g$
Heat transport fluid	
Type	Boiling light water
Quantity	56t
Total mass flow	7,600t/hr
Fluid temperature	
at reactor inlet	277℃
at reactor outlet	284℃
No. of pumps	4 in 2 groups of 2
Steam drum	
Number of boilers	2
Stream pressure at drum	$68kg/cm^2g$
Steam temperature at drum	284℃
Total steam flow	910t/hr
Steam quality at reactor outlet	14%
Feed water temperature	182℃
Fuel	
Type	28 bundles 4100mm long
Diameter of fuel pin	16.5mm including Zircaloy-2 sheath
Pellet diameter	14.4mm
Minimum cladding thickness	0.8mm

Total weight of PuO_2-UO_2 per bundle	174kg
Bundle diameter	111.6mm
Heat transfer area per bundle	$5.36m^2$
Heat flux average	$3.7x10^5kcal/m^2 \cdot hr$
Maximum	$9.5x10^5kcal/m^2 \cdot hr$
Fuel temperature maximum	2,200℃
Maximum cladding temperature	350℃
Turbine Generator	
Turbine	One tandem compound unit per reactor with external moisture separation
Number of high-pressure cylinders	1 single flow
Number of low-pressure cylinders	2 single flow
Speed	3,600rpm
Throttle pressure	$65kg/cm^2g$
Type of condenser	One pass
Cooling water flow	$35,580m^3/hr$
Generator	One per turbine-200,000kVA 3 phase, 60 cycle, 16kV
Transformers	
Main Power	178,000kVA, 3 phase, 60 cycle, 16/275kV
Station service unit	27,500kVA, 3 phase, 60 cycle, 16/6,9kV
System	22,000kVA, 3 phase, 60 cycle, 275/6,9kV
Circuit breakers	
275kV	2-3 cycle, gasblast(SF_6), 2,000amp, 40,000MVA
6,9kV	390MVA
Standby generation and switching	
Diesel Generator	6,000kVA, 6.9kV
Circuit Breakers	7.2kV, 390MVA

Table I.2-2 R&D on Reactor Proper and Pressure Tube Assembly

Components	R&D Items
Calandria and Iron/Water Shields	Establishment of Design Method of Calandria and Iron/Water Sields Structure R&D of Reactor Proper Analysis Codes and Their Confirmation Tests Confirmation Test of Calandria Tube Buckling Strength against External Pressure Vibration Test of Reactor Proper Development of Calandria Tube Rolled-Joint and Its Confirmation Test Development of Control Rod Guide Tube Rolled-Joint and Its Confirmation Test Mock-up Test of Reactor Proper
Pressure Tube Assembly	Establishment of Design Method of Pressure Tube Assembly Development of 403 Mod. Stainless Steel Irradiation Test Out-of-Pile Fracture Toughness Test of Zr-2.5%Nb Development of Pressure Tube Rolled-Joint

Table I.2-3 Results of Pressure Tube Rolled-Joint Development
(Performance Test Results)

Items	Acceptance Criteria	Test Results
Leak Test Air Tightness He Leak Test	$<10^{-3}$Acc/secHe	Upper Rolled-Joint $<1.69\times10^{-4}\sim5.66\times10^{-9}$ Acc/secHe Lower Rolled-Joint $<5.67\times10^{-5}\sim 5.69\times10^{-9}$ Acc/secHe
Pressure Test Water Pressure Test and Then He Leak Test	$<10^{-3}$Acc/secHe	Upper Rolled-Joint $<4.27\times10^{-5}\sim1.01\times10^{-8}$ Acc/secHe Lower Rolled-Joint $<3.42\times10^{-5}\sim 1.38\times10^{-8}$ Acc/secHe
Thermal Cycle Test and Then He Leak Test	$<10^{-3}$Acc/secHe	Upper Rolled-Joint $<1.81\times10^{-8}$ Acc/secHe Lower Rolled-Joint $<4.30\times10^{-7}$ Acc/secHe
Tension Test (after Thermal Cycle Test)	>3.5x(Design Axial Tension)	Pressure Tube Ruptured No Rolled-Joint Ruptured Max. Load :~10x(Design Axial Tension)
Rupture Test		Pressure tube Ruptured No Rolled-Joint Ruptured
Bending Test	>(Design Bending Moment) Upper Rolled-Joint >150kg·m Lower Rolled-Joint >140kg·m	Upper Rolled-Joint >2,170kg·m Lower Rolled-Joint >2,240kg·m
Torsion Test	>(Design Torsion Moment) >110kg·m	Upper Rolled-Joint >697kg·m Lower Rolled-Joint >710kg·m
Pressure Cycle Test and Then He Leak Test	$<10^{-3}$Acc/secHe	Upper Rolled-Joint $<1.2\times10^{-9}$ Acc/secHe Lower Rolled-Joint <Detectable He Leak

Table I.2-4 Reactor Scram List

Scram Condition	Sensor Location	Scram Set-Point	D2O Dump
High Pressure in Containment	Containment	0.2kg/cm^2 g	○
High Pressure in Steam Drum	Steam Drum	72kg/cm^2 g	
Low Water Level in Steam Drum	Steam Drum	Normal-230mm	○
Low Coolant Flow	Outlet of Reactor Recirculation Pump	90% Rated Flow	
High Neutron Flux	PCM Reactor PUM Reactor SUM Reactor	120% Rated Value 95% Full Scale 5x10^5 cps	 ○
High Radioactivity in Main Steam Pipe	Main Steam Pipe outside Containm	6x(Normal Level)	
Main Steam Isolation Valve Closure		90% Opening	
Low Vacuum Pressure in Condenser		584mmHg	
Main Turbine Stop Valve Closure		90% Opening	
High He Pressure	Calandria	0.4kg/cm^2 g	○
Power Failure			
Low D_2O Level	Calandria	Normal-400mm	
Manual Scram with Control Rods	Main Control Room		
Earthquake	Main Control Room Horizontal Vertical	 160 gal 80 gal	○
Manual D_2O Dump	Main Control Room		○
Low D_2O Flow	Outlet of D_2O Cooler	1,100m^3 /hr	
High D_2O Temperature	Outlet of D_2O Cooler	59℃	
Low He Flow	Inlet of Calandria	250Nm^3 /hr	

Table I.2-5 Plant Status Change

(Test Item : S.D.Water Level Set-Point Change Test)
(Test Date : 1978/8/8, Test Power : 25% Electric Power)

25mmDecrease of Water Level in S.D.A
One Element Control by Low Control Valve.
Propotional Gain Gp=5
Reset Time T1=450sec

Items	Unit	Initial	Max	Min	Final	Remarks
TPM Power	%	31.3	31.3	31.3	31.3	
Generator Output	MW	42	42	42	42	
S.D.A Pressure	kg/cm^2g	66.9	66.9	66.9	66.9	
S.D.B Pressure	kg/cm^2g	66.8	66.8	66.8	66.8	
S.D.A Water Level	mm	-21	-21	-47	-39	
S.D.B Water Level	mm	-19	-10	-19	-10	
S.D.A Feed Water Flow	t/h	128	128	123	127	
S.D.B Feed Water Flow	t/h	138	138	138	138	

Remark
S.D. : Steam Drum
TPM : Total Power Monitor

Table I.2-6 Plant Status Change

(Test Item : Pressure Set-Point Change Test (0.5kg/cm^2 Lowering of Pressure Set-Point))
(Test Date : 1979/1/22, Test Power : 100% Electric Power)

Items	Unit	Initial	Max	Min	Final
TPM Power	%	98	98	98	98
Generator Output	MW	165	170	165	165
S.D.A Pressure	kg/cm^{2}g	66.70	66.70	66.30	66.30
S.D.B Pressure	kg/cm^{2}g	66.65	66.65	66.25	66.25
S.D.A Water Level	mm	0	0	0	0
S.D.B Water Level	mm	0	0	0	0
S.D.A Steam Flow	t/h	453	473	453	453
S.D.B Steam Flow	t/h	468	488	468	468
S.D.A Feed Water Flow	t/h	460	460	460	460
S.D.B Feed Water Flow	t/h	470	470	470	470
Main Steam Header Pressur	kg/cm^{2}g	63.4	63.4	62.9	62.9

Remark
S.D. : Steam Drum
TPM : Total Power Monitor

3. Operational Achievements of "Fugen" [I-24] [I-25] [I-26]

3.1 Introduction

Construction of "Fugen" started in December, 1970. After confirming the structures and strengths of equipment and piping and the states of installation, overall performance tests was conducted from July, 1977 to confirm the functions of the system, followed by the startup test from March, 1978. Core and plant characteristics, the performance of various equipment, transient responses in incidents such as load rejection and continuous power operation characteristics were tested for a year to confirm that all of them meet the design performance. The final official inspection and testing (100 hr continuous full power operation) were passed and the full power operation was licencsed on March 20, 1979, and its full power operation began. Since then, "Fugen" has demonstrated the performance and reliability of the ATR system for about 20 years, in addition to the clarification of plutonium utilization characteristics and development of its utilization technology. During this period, moreover, "Fugen" was loaded with a large amount of uranium-plutonium mixed oxide fuels (MOX fuels) for the first time in the world as a thermal power reactor, and their total number up to the end of August, 1997 amounted to 658 assemblies. The average utilization factor of "Fugen" since its start of operation is about 66 %.

The operating cycle of "Fugen" is based on 6 months; accordingly, fuels are replaced twice a year, including fuel replacement at the time of a periodic inspection. In addition, 12-month operating cycle was demonstrated from 1984 to 1985.

Since the start of operation, "Fugen" has met troubles including unscheduled plant shutdown and improvements have been implemented for them, but no trouble originated essentially from the design inherent in the heavy water reactor has ever occurred.

"Fugen" has been operated smoothly except about ten months in 1980 when repair works of the reactor coolant piping were carried out due to stress corrosion cracking (SCC).

The periodic inspection is usually performed for about 3 months annually to make refueling and maintenance of equipment. The 2nd to 6th periodic inspections took 4 to 6 months because of SCC prevention works. In the 7th periodic inspection, the past experience allowed the inspection period (from parallel off to parallel in) to make 73 days which were shortest in the operation of "Fugen".

The operational history of "Fugen" are shown in Fig.I.3-1 and Fig.I.3-2, whereas results of total electric energy generation and utilization factors are

given in Fig.I.3-3.

Of 1239 fuel assemblies loaded into the core since the startup of "Fugen" the number of MOX fuel assemblies accounts for 658, which are the highest plutonium utilization in one reactor in the world as one reactor. The maximum fuel burnup reached about 19,617 MWd/t, and no fuel has ever been failed. In 1988, moreover, MOX fuel assemblies reusing Pu recovered in the Tokai Reprocessing plant from the spent MOX fuel of "Fugen" were loaded to its core, completing the nuclear fuel cycle using plutonium in Japan. The amounts of fissile Pu loaded hitherto, spent MOX fuel assemblies, consumed fissile Pu by burning, and of fissile Pu remaining in the fuels are 809kg, 546 assemblies, about 651 kg, and about 561 kg, respectively.

The cumulative amounts of fissile Pu, total Pu, utilization of MOX and burnup of discharged fuel are given in Fig.I.3-4 to Fig.I.3-7, respectively.

3.2 Facility and Its Features

The plant structure of "Fugen" resembles that of a boiling-water-type light water reactor (BWR) except that it uses heavy water for moderator and its core structure is of pressure-tube type.

Main plant data, main schematic diagram and plant control schematic diagram of "Fugen" are shown in Table I.3-1, Fig.I.3-8 and Fig.I.3-9, respectively.

(1) Reactor

The reactor consists of calandria tanks, pressure tubes, iron/water shields, etc. The calandria is a vertical cylindrical tank provided inside with 224 calandria tubes arranged in square lattice. Outside of calandria tubes is filled with moderator, i.e. heavy water. The pressure tubes are installed in each calandria tube, and fuel assemblies are remotely loaded into the core and withdrawn from the bottoms of the pressure tubes by using the refueling machine.

The cross-section of the reactor is shown in Fig.I.3-10.

(2) Fuel Assemblies

The fuel assembly consists of 28 fuel elements arranged in a 3-layer concentric cluster (4 inner layer, 8 intermediate layer and 16 outer layer), and is an integral-type fuel with the overall length of about 4.5m (about 3.7m of effective fuel length). MOX fuel assemblies containing natural or depleted uranium enriched with 0.8-1.4% fissile Pu are used, and slightly enriched uranium oxide (UO_2) fuel assemblies are sometimes used. The mixing ratio is changed depending on the state of production of fuel assemblies (MOX and UO_2 fuel assemblies are called the standard fuel

assemblies). Besides these fuel assemblies, 4 special fuel assemblies are used. They adopt UO_2 and can accommodate monitoring test pieces of pressure tube materials in their central zones, and are loaded into the central part of the core. The standard fuel assembly and special one are shown in Fig.I.3-11, whereas the arrangements of the initial and 27th operating cycle cores are given in Fig.I.3-12.

(3) Control Rods

The control rod is of double cylindrical layers. Except the peripheral zone of the reactor core, a control rod is provided for each four fuel assemblies, amounting to 49 rods. Of them, 4 rods are set in each quadrant to automatically control the reactor power so as to generate equal power in each quadrant.

The control rod moves up and down within the control rod guide tube installed in the calandria tank filled with heavy water. The control rods are driven by the motor-driven wire drum. In an emergency, the electromagnetic clutch is disconnected, and the rods fall into the core by their own weights. They are also used to control rapid reactivity change and power distribution. Slow reactivity change, caused by fuel burnup, is controlled by adjusting poison (boric acid) concentration of heavy water.

The control rod assembly and its main data are shown in Fig.I.3-13.

(4) Reactor Cooling System

The reactor cooling system consists of 224 pressure tubes which are divided into 2 independent loops in their center. Each loop is equipped with a steam drum, 2 recirculation pumps and a lower header. Four downcomers from the steam drum are combined into 2 before connected to each recirculation pump. In each loop, coolant (light water) is supplied into 112 inlet pipes in the reactor inlet header and flows into the lower part of each pressure tube, where the coolant is heated and boiled to steam water mixture and then flows into the steam drum through 112 outlet pipes. Steam separated in the steam drum goes to the steam turbine, whereas saturated water is mixed in the steam drum with feed water returning from the steam turbine flows back to the recirculation pumps through the downcomers and then to the reactor again as shown in Fig.I.3-9.

(5) Reactor Auxiliary System

One of the features of "Fugen" is the reactor auxiliary system consisting of the heavy water, helium, and carbon dioxide systems.

The heavy water system removes heat generated in the moderator, i.e. heavy water, removes corrosion products and other impurities in it and adjust boric acid concentration. Heavy water overflows from the upper part of the

calandria tank, flows through the dump tank and heavy water recirculation pump to the heavy water heat exchanger, in which it is cooled and then flows back to the calandria tank, mainly through the control rod guide tubes in the core. Part of heavy water flows through the heavy water purification column in which strong acid cation exchange resins and weak base anion exchange resins are filled to remove corrosion products and other impurities. Heavy water contains boric acid (about 90 wt% enriched boron-10) to suppress excess reactivity. Adjustment of boric acid concentration compensates reactivity associated with fuel burnup and controls slow power change. For this purpose, the heavy water system is equipped with the poison feed system and poison removal column in which strong base cation exchange resins are filled.

In the helium system, the surface of heavy water is covered with helium to prevent heavy water from degrading. Deuterium formed by radiolysis of heavy water is combined with oxygen again, thereby preventing the build-up of explosive gas in the calandria tank. Moreover, adjustment of pressure difference of helium between the heavy water overflow space and dump space of the calandria tank allows the heavy water level in the calandria tank to keep constant. Heavy water dump function is also provided; in an emergency, the helium communicating valve can be rapidly made open to eliminate helium pressure difference in the calandria tank to lower the heavy water level rapidly, and thereby leads the reactor to subcritical.

The carbon dioxide system supplies carbon dioxide gas (CO_2) to the gap between the pressure tube and calandria tube. In this system, moreover, moisture content of CO_2 gas on the outlet side is measured by a dew-point hygrometer to detect the leakage of cooling water or heavy water from the pressure or calandria tube.

(6) Engineered Safety Features

The engineered safety features include the emergency core cooling system (ECCS) consisting of the quick, high-pressure and low-pressure water injection systems, the steam discharge pool cooling system, the isolated cooling system, residual heat removal system and the reactor containment. The ECCS, the steam discharge pool cooling system, the isolated cooling system are each composed of two independent systems, each of which has 100% design capacity. Even in case of external power failure, it can be supplied by the emergency diesel-engine generator.

(7) Fuel Replacement Facility

The fuel replacement facility consists of the refueling machine and fuel transfer equipment.

The refueling machine is located under the reactor, moved by the traveling and traversing carriages, and can be connected to all pressure tubes and transferports (through which fresh and spent fuel assemblies pass) by remote control from the central control room.

(8) Other Installations

The cooling water purification, main steam, feed water and condensate systems of "Fugen" are principally the same as those of BWR. The rotational speed of the turbine generator is 3,600 rpm, and the steam turbine is controlled by the electric hydraulic control system (digital-EHC).

3.3 Operation Management [I-27] [I-28]

(1) Operation Management System

The operation management system is shown in Fig.I.3-14

The operation of "Fugen" is managed by three Technical Sections in accordance with their respective roles with the Power Generating Section in charge of operation being the center. The Licensed Reactor Engineer whose nomination is stipulated by law offers his necessary opinions to the plant Director in accordance with the state of operation, and instructs and advises the Power Generating Section Chief, etc. The Reactor Safety Examination Committee is composed of the Deputy Plant Directors, Section Chiefs and Licensed Reactor Engineer, and discusses changes of important operational methods, modifications of main installations and revisions of regulations. The results are reported to the Plant Director and are reflected on plant operation.

(2) Operation System

To carry out 24 hour plant operation, the operators of the Power Generating Section work on 5-group 3-shift basis; namely, one working day is divided into 3 shifts, and each group changes their shifts every 8 days. To make the transfer of works complete, 15-min. overlap time is provided. Besides, the operation records and memos are used between shift leaders and between operators to make the transfer of works more complete.

The shift-works include plant operation, plant control, normal monitoring, surveillance, periodic testing of each equipment, work control, isolation (system isolation) and trial operation.

A group consists of a shift leader, an assistant shift leader, 3 – 5 senior operators, 3 – 5 junior operators and 1– 2 trainees; accordingly, 10 – 13 in total.

The shift leader is a manageal officer, and performs the labor management of his shift operators in addition to the instruction of them. The assistant

shift leaders assist the shift leader, instruct plant operation and control operation.

The senior operators have operation experience of 7 years or more in principal, and mainly in charge operation in the central control room.

The junior operators have operation experience of 1 – 7 years, and mainly in charge of field operation and assist operations in the central control room.

The trainees have operation experience less than a year, and operations by themselves are prohibited.

The operation system of the Power Generating Section is shown in Fig.I.3-15, whereas the classification and qualification of operators are shown in Table I.3-2.

(3) Plant Operation

Although the operation methods of the plant resemble those of BWR, they are different from BWRs in

① Power Control Method,

② Refueling Method,

③ Speed Change-over Method for Recirculation Pumps

④ Operation of Heavy Water/Helium System

and adopt their own operation methods.

Power is controlled by the control rods and adjustment of poison (boric acid) concentration of heavy water. During rated power operation, the automatic power control rods are set at "automatic operation" to respond short and slight changes, whereas slow power changes due to fuel burnup are adjusted by controlling poison concentration.

Unlike the coolant recirculation pumps of BWRs in which flow rate is continuously changed, those of "Fugen" change it stepwise from low speed (450 rpm) to high speed (900 rpm) at about 40-% power; accordingly, the speed change over of the recirculation pumps makes recirculation flow rate roughly twice, and resultant change in void fraction changes the water level of the steam drum, but the recirculation pumps can be safely changed over by setting the water level higher in advance. Zero reactivity coefficient of coolant void leads to little power change due to change in void fraction.

Plant operation follows the safety regulations and operation manuals. But actual operations are carried out in accordance with the detailed operating procedures reflecting the basic policies and restrictions given in the safety regulations and operation manuals. There are 4 kinds of detailed operating procedures as shown in Fig.I.3-16:

① Plant Startup/Shutdown Procedures (which stipulate startup/shutdown of the plant as a whole),

② Operating Procedures in Emergencies (which stipulate the operating procedures in scrum, turbine trip and loss of external power sources),

③ Periodic Test Procedures (which stipulate the procedures at the time of periodic tests)

④ Surveillance and Check Procedures (which stipulate the procedures of surveillance and check of the plant).

In addition, there are system operating procedures stipulating the operating methods of each system and equipment under the plant startup/shutdown procedures.

The plant is operated and controlled under these systems and procedures. The plant startup and shutdown curves are given in Fig.I.3-17 and Fig.I.3-18, respectively.

(4) Education and Training

Education of operations has been carried out by preparing the programs suitable for each class operators to understand necessary knowledge; techniques and attitudes comprehensively. The education and training plans are shown in Table I.3-3 and Table I.3-4.

The concrete educational goals are given to each operator and the plan is so established as to be attained in a given period. In the in-section education, the knowledge, techniques and attitudes necessary for operating "Fugen" are given, whereas in the in-plant education, general labor safety and radiation safety necessary for working nuclear power plants are taught and in in-PNC education the desirable attitudes as PNC personnel and as a full-fledged member of society are taught.

In addition to the above education and training, operational training in the BWR Operation Training Center using the simulator, and the lectures regarding general and wide knowledge of nuclear energy in JAERI and JAPCO are given. Moreover, they have positively applied various qualification tests and joined short courses of study and obtained successful results.

The result of operator training is given in Fig.I.3-19

3.4 Maintenance Management [I-29]

(1) Present State of Maintenance Management

Maintenance management has been systematically performed by establishing the "annual maintenance plan", which include the maintenance plan such as daily and periodic checks of equipment, the modification and replacement plan for equipment necessary for maintenance and improvement, and the technical development and budget plans regarding

testing and inspection.

In daily check and repair, efforts have been paid for early finding and prevention maintenance of various anomalies by trend analysis/evaluation of plant operation states with a plant data processing system, in addition to repair and take preventive maintenance measures for troublesome places found by surveillance of operators or maintenance members. Efforts has also paid for confirming the functions of stand-by equipment by periodic confirmation of the functions of engineered safety facilities.

The periodic inspection has been carried out on the basis of "Long-term Periodic Inspection Plan" and after fully investigating the results of the periodic inspections, operations and maintenance of "Fugen" and the troubles experienced in "Fugen" and other LWRs. Moreover, measures against degrading equipment, and installation modification and replacement works for reliability improvement, labor saving and for exposure reduction have been mainly carried out at the time of the periodic inspection.

As to unscheduled plant shutdown, such as reactor scram, incidents and troubles, they are investigated and evaluated with regard to the structures, materials, state of operation, working environment and design and construction records whenever they occur. Efforts are also made to prevent recurrence by determining countermeasures and scopes through performing necessary testing and inspection including the investigation of resembled cases.

(2) Results of Maintenance

The periodic inspection has been carried out every year on the basis of the "Laws for Regulating Nuclear Raw Materials, Nuclear Fuel Materials and Nuclear Reactors" (hereinafter referred to as "Laws for Regulating Nuclear Reactors"). Since "Fugen" is a nuclear reactor having a power generating facility, inspection based on the "Electric Enterprises Act" has been conducted, but consideration is given to the periodic inspection procedures and implementation methods to avoid double regulations by the "Laws for Regulating Nuclear Reactors" and "Electric Enterprises Act".

The periodic inspection confirms that the facility and equipment meet the technical criteria stipulated by the Safety Examination Guideline and Electric Technique Standards (JEAC) to ensure the safety and integrity of the functions and performance of equipment for continuing safe operation, and that they meet the requirements such as the safety regulations for the reactor facility.

(3) Maintenance Works

Reflecting the results of operation and maintenance and the experience obtained by LWRs, modification and replacement works necessary for the maintenance and

improvement of equipment are performed mainly at the time of periodic inspection. The main purposes of equipment modification works are as follows:

(a) Replacement and Modification Works from Viewpoints of Preventive Maintenance,

(b) Replacement and Modification Works to Improve or Keep Operability and Maintainability,

(c) Works to Improve Operability and to Reduce Radiation Exposure,

(d) Improvement Works Associated with Revision of Technical Standards and Guidelines. (Installation of High-Range Monitors for Emergency Use)

(e) Repair Works Associated with Troubles.

Of these, the measures for SCC (stress corrosion cracking) and corrosion/ erosion are taken from the viewpoint of preventive maintenance, and efforts are taken to prevent troubles from occurring and to ensure integrity and reliability.

Upon implementing these measures, reproducibility and mockup tests are carried out in advance, and the results are reflected on plans. In "Fugen", SCCs were found near the welds of austenitic stainless steel piping of the emergency core cooling system at the time of the scheduled shutdown in November, 1980.

For the piping system, therefore, the applicability of BWR techniques to this trouble was investigated, and the measures were schematically taken in accordance with the "Basic Plan for SCC" which stipulated the ranges of measures, methods and priority. Table I.3-6 shows the state of improvement works in "Fugen".

As a measure for SCC in "Fugen" the piping material was basically changed from 304 stainless steel to SCC-resistant low-carbon 316 stainless steel.

For welds for which replacement of material is difficult, IHSI (Induction Heating Stress Improvement) by which tensile residual stress of the inside surface of a pipe can be reduced by high-frequency heating has been deliberately employed.

To find out SCC in piping welds along the pressure boundary of reactor coolant, integrity has been confirmed by increasing the frequency of inspection (Compared with the usual ISI).

Moreover, for the replacement of materials of inlet and outlet pipes as well as for places to which application of IHSI is difficult, hydrogen injection was employed to reduce the concentration of dissolved oxygen (DO) in reactor coolant, thereby preventing SCC from occurring; this method for improving water quality is called HWC (Hydrogen Water Chemistry). Because the effectiveness of HWC for preventing SCC from occurring was verified by the

demonstration tests of "Fugen", HWC has been employed by injecting hydrogen with target value of DO of reactor coolant equal to or less than 20 ppb since December, 1985.

(4) Incidents and Troubles

The cumulative number of incidents and troubles for 19 years, i.e. from the start of full power operation (March, 1979) to 1997, is 39; the outstanding incidents and troubles are shown in Table I.3-7.

The number of incidents and troubles reported in accordance with the laws and regulations was 30 for 10 years after the start of full power operation; namely, 3 per fiscal year, and about 1.3 per fiscal year after that. The average number of incidents and troubles for 19 years was about 2.2/year.

3.5 Core and Fuel Management [I-30]

The outstanding features of "Fugen" type heavy water reactor are that its core characteristics show little difference whether MOX fuels or UO_2 fuels are used, and that plutonium or recovered uranium (from LWR spent fuels in reprocessing plants) can be effectively and easily utilized.

The fuel assemblies have been exchanged in 90° - rotation symmetrically scattered mode in view of flattening power distribution and easily making fuel management and operation control. A typical fuel loading pattern is given in Fig.I.3-21.

Since the coolant void reactivity coefficient is nearly zero, power coefficient mainly depends on fuel temperature coefficient (Doppler coefficient). Because the void reactivity coefficient of coolant is nearly zero, reactor power little fluctuates and is stable for rapid change in void fraction at the time of changeover of recirculation pump speed at about 40-% rated power.

Fig.I.3-22 shows the change of plant parameters on recirculation pump speed change.

As the average neutron migration area in the core of the heavy water is large, power flattening by control rods is almost unnecessary; accordingly, nearly all the control rods in the core during operation are withdrawn except the automatic power control rods.

Reactivity compensation associated with fuel burnup is performed by adjusting the boron concentration in heavy water, but movement of the control rods is extremely small, and power peaking is gradually become smaller with fuel burnup, resulting in power flattening. The poison concentration after reactor power change and with fuel burnup are given in Fig.I.3-23 and Fig.I.3-24, respectively. Also Fig.I.3-25 shows the change of radial and axial power distributions in a typical operating cycle. The changes with fuel burnup in

maximum linear power heat rating and boron concentration in heavy water are shown in Fig.I.3-26.

3.6 Chemistry Control

Chemistry control means mainly the water quality control of coolant and moderator, i.e. heavy water. The results of water quality control is given in Table I.3-8.

Like in BWRs, reactor cooling water is kept neutral, and its gamma nuclides have been measured twice a week to monitor fuel failure and changes in corrosion products.

To reduce the radiation exposure of operators, measures to reduce the amounts of metallic impurities brought into the reactor have been taken.

Oxygen injection into feed water started to form stable oxygen films on the surfaces of carbon steel piping for reactor feed water system, resulted in the reduction of iron ion concentration in feed water down to 1/10 and thereby suppressing dose rate rise on the surfaces of the piping as shown in Fig.I.3-27.

3.7 Radioactive Waste Management [I-31]

Radioactive wastes have been managed in accordance with the release control goals specified by the safety regulations and with the limit of release concentrations stipulated by the laws. As shown in Table I.3-9, the results of release of radioactive wastes have been satisfactorily below the control goals.

The amounts of radioactive rare gases have been released 10^{10} - 10^{11} Bq/year in the past, and these are due to Ar-41 contained in CO_2 released from the CO_2 system and helium released to suppress its pressure rise stemming from volume expansion of helium at the reactor startup. The amount of released rare gases decreased below the detection limits by taking the following measures; namely, the formation of Ar-41 was suppressed by improving helium purity and decreasing the amount of released CO_2 gas.

Tritium is generated by the activation of moderator, i.e. heavy water, and released when heavy water is handled such as the overhaul of heavy water equipment and sampling for heavy water quality control.

Of liquid wastes, nearly all nuclides contained in radioactive materials except tritium are gamma nuclides, and their origins are almost laundry waste. Therefore, the dry cleaning system was introduced in the fiscal 1990, and the throughway mode was adopted for the laundry in the fiscal 1994; these measures decreased the radioactivity to the detection limits or less.

Like the gaseous wastes, tritium is released by heavy water handling.

3.8 Radiological Safety Management [I-32]

PNC and its subcontractors prepare the radiation protection plan in advance on the basis of the safety regulations, detailed safety regulations and radiological safety management specifications to perform operations systematically and to do thorough operation control.

The results of radiological safety management is given in Fig.I.3-28. The increase in dose equivalent in the modification works stems from the works for SCC measures.

Most of annual dose equivalents stem from those during the periodic inspections. Of them, dose equivalent relating to the normal periodic inspection except the modification works tends to remain in 2 - 4 man·Sv. The reason that the decrease in the rising rate of dose equivalent rate in the 10 th periodic inspection is the chemical decontamination of the system carried out in the 9th periodic inspection.

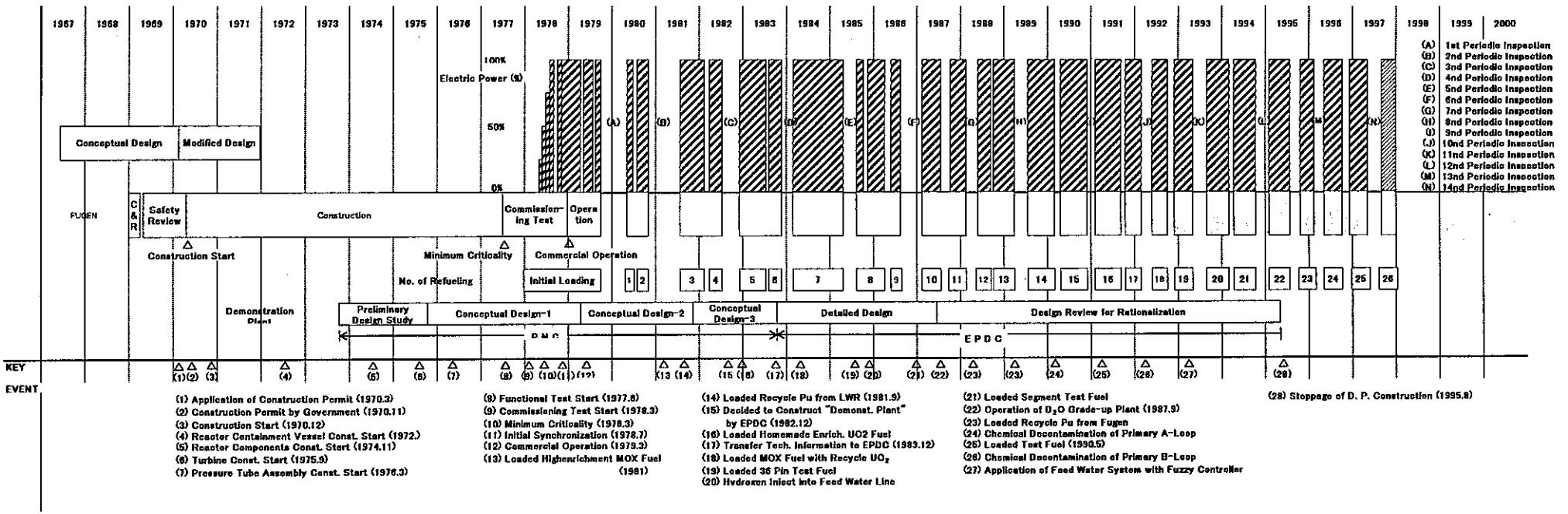


Fig.I.3-1 "Fugen" Development and Operating History

Cumulative generating hour : 109,263hours
electric output : 17,596 million kWH
load factor : 65.9 %
at the end of August '97

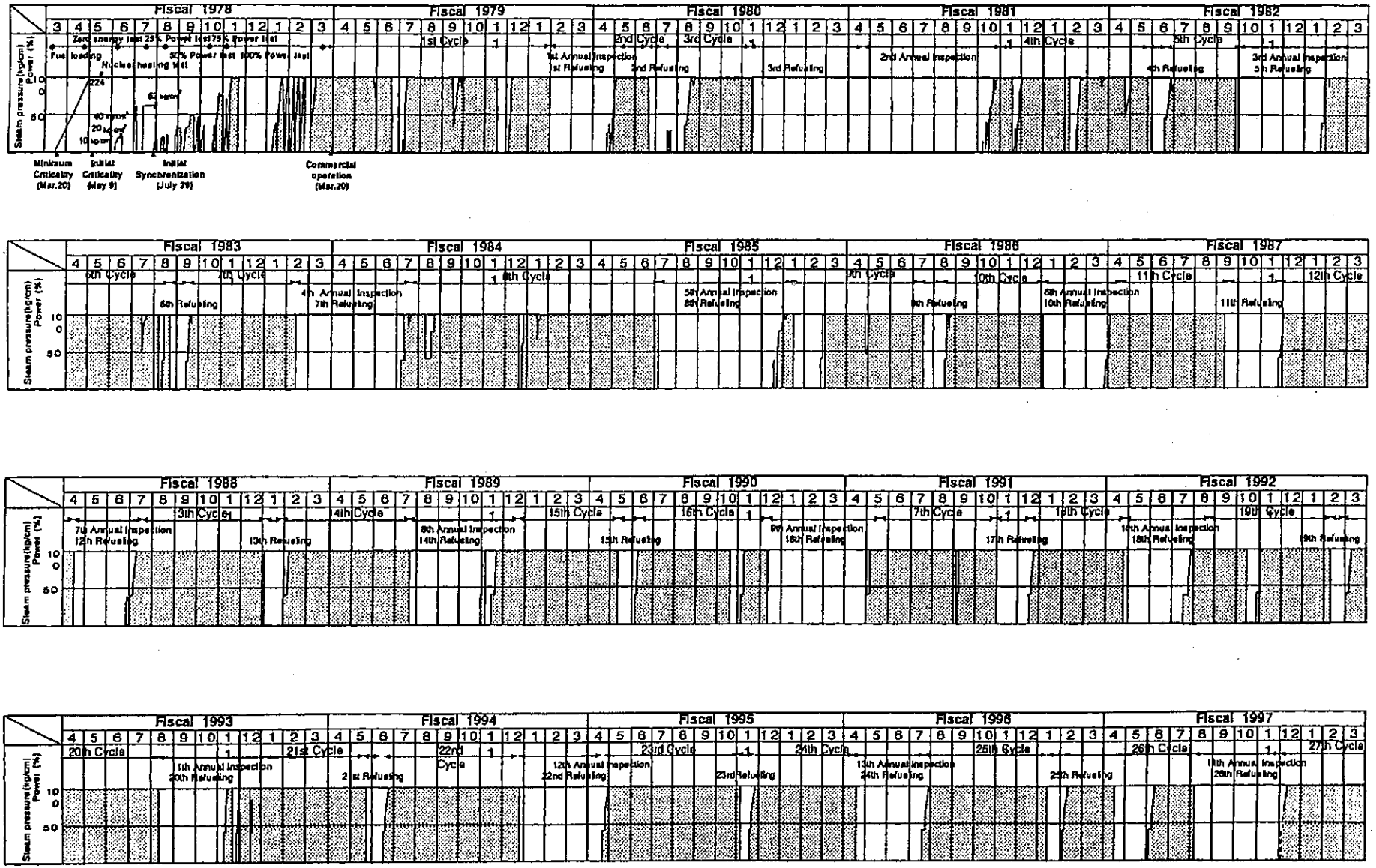


Fig.I.3-2 "Fugen" Operating History

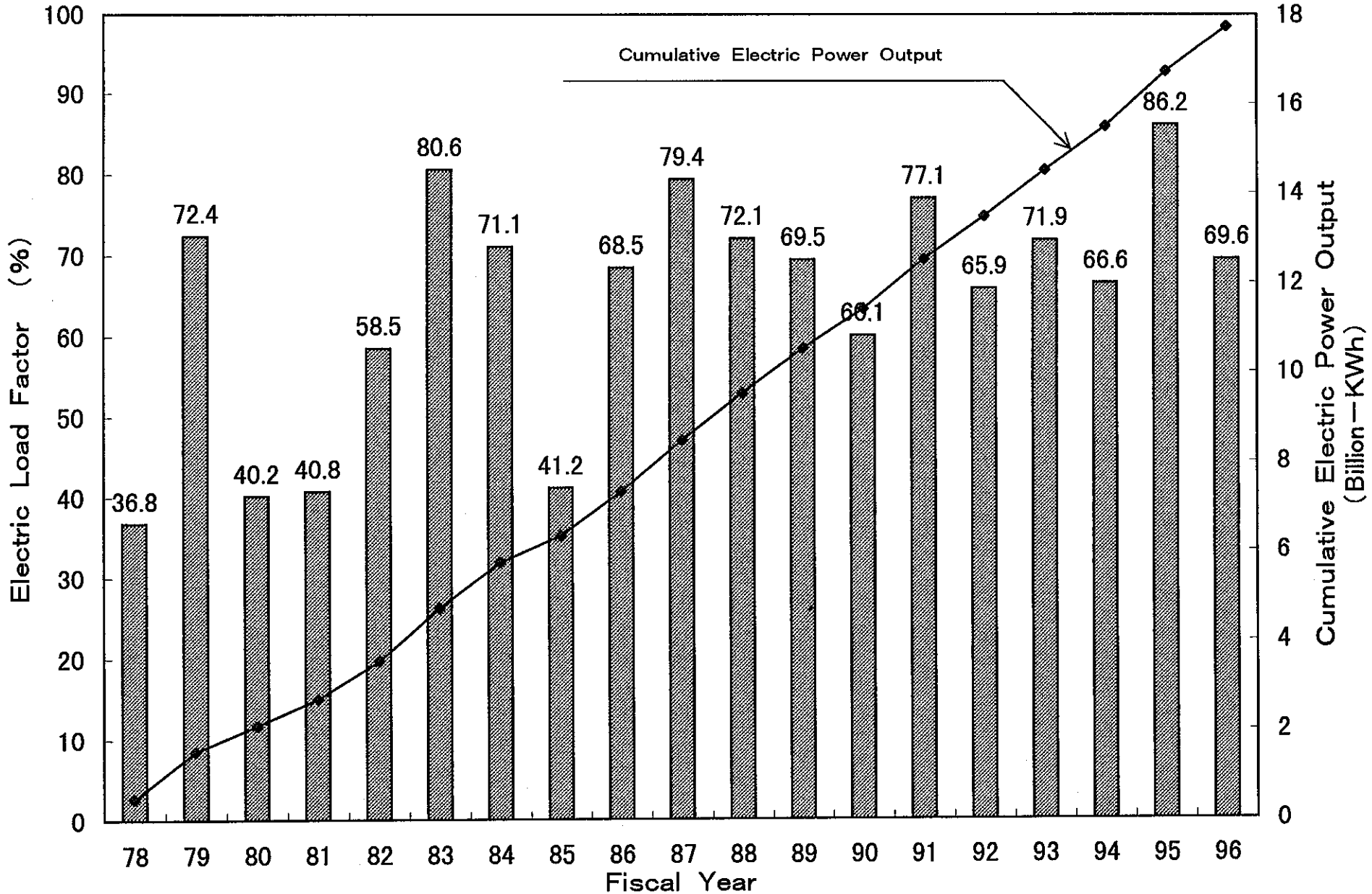


Fig.I.3-3 "Fugen" Operating History
(Load Factor and Cumulative Electric Power)

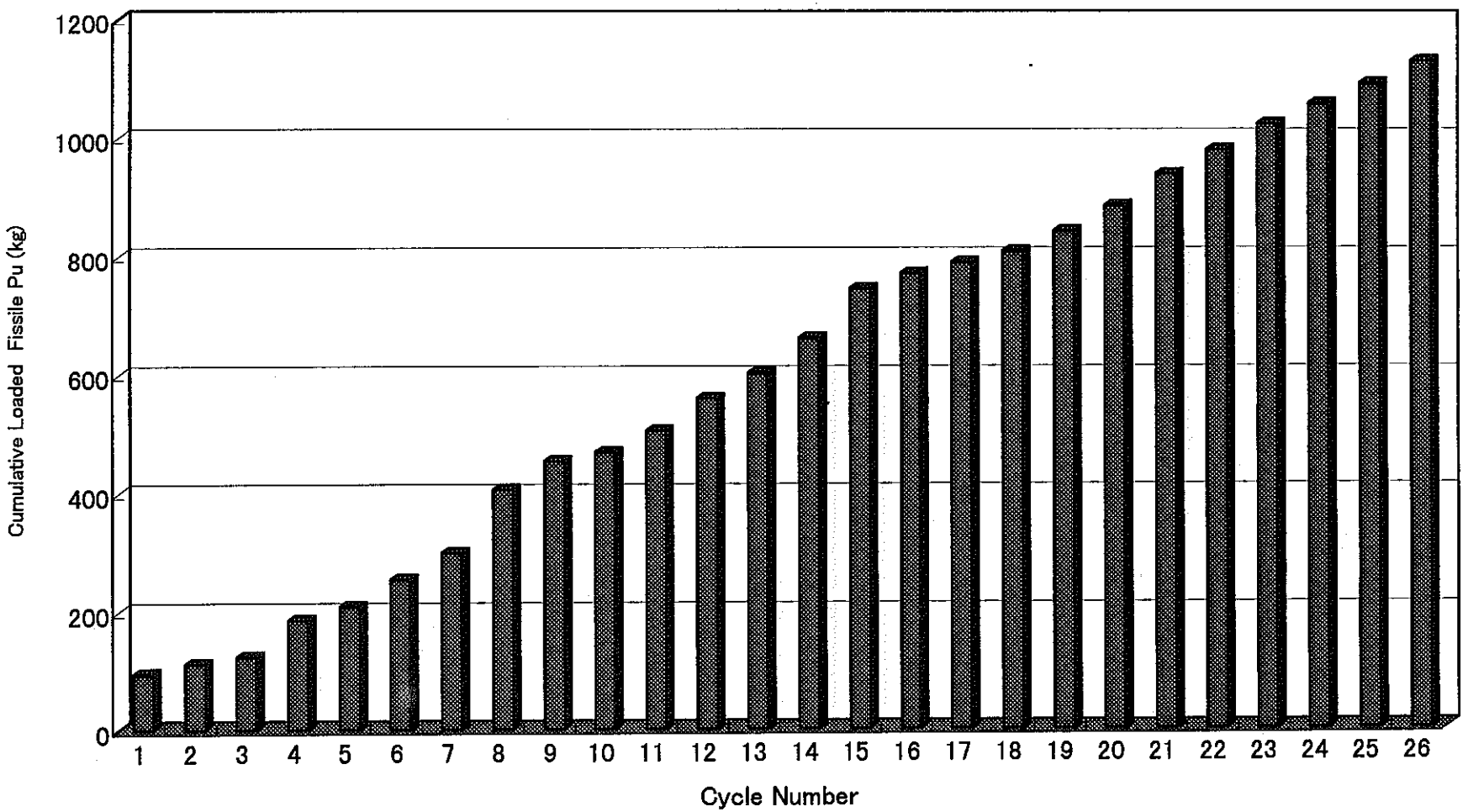


Fig.I.3-4 History of Cumulative Fissile Plutonium used in "Fugen"

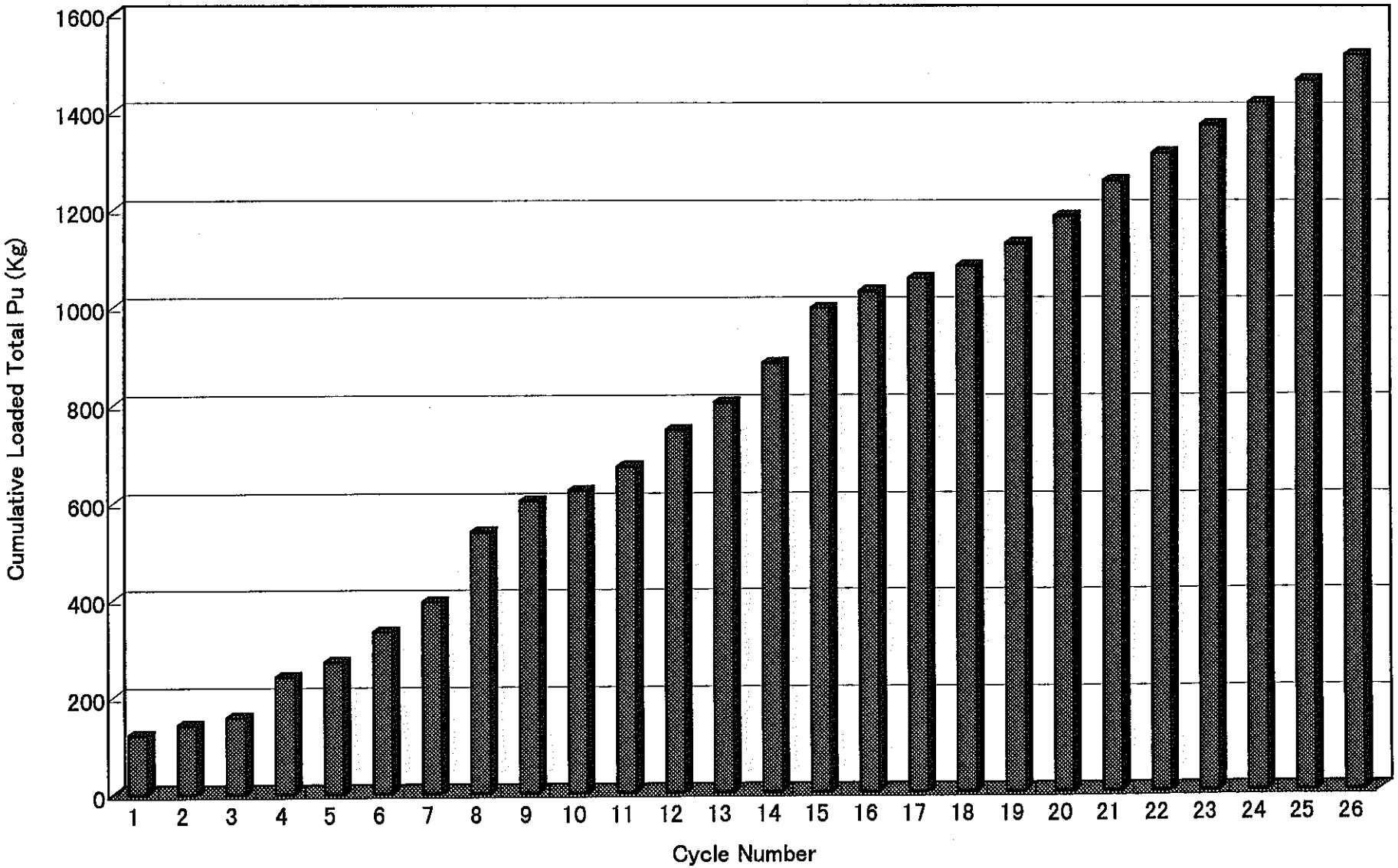


Fig.I.3-5 Cumulative Total Plutonium Used in "Fugen"

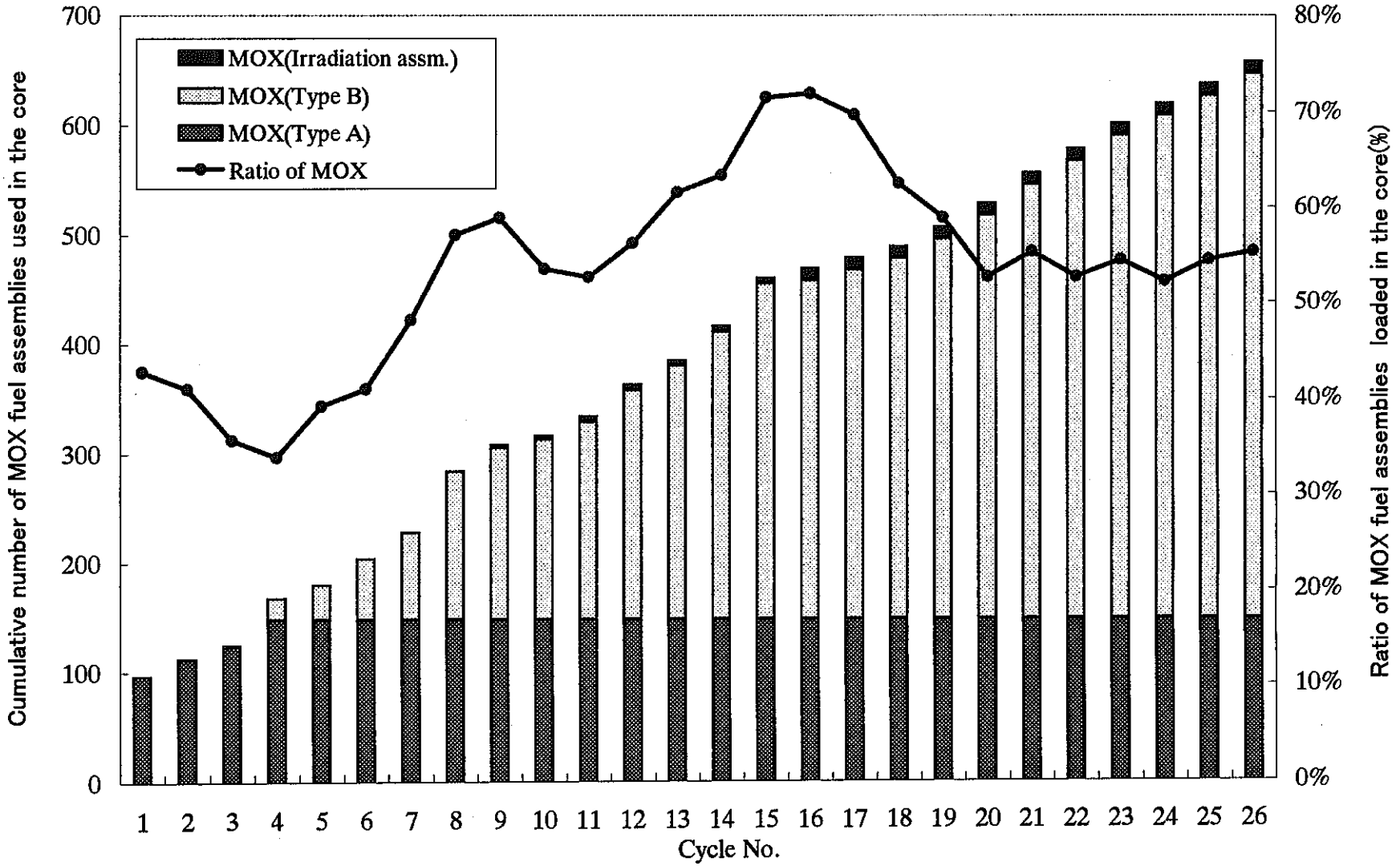


Fig.I.3-6 MOX Fuel Used in "Fugen"

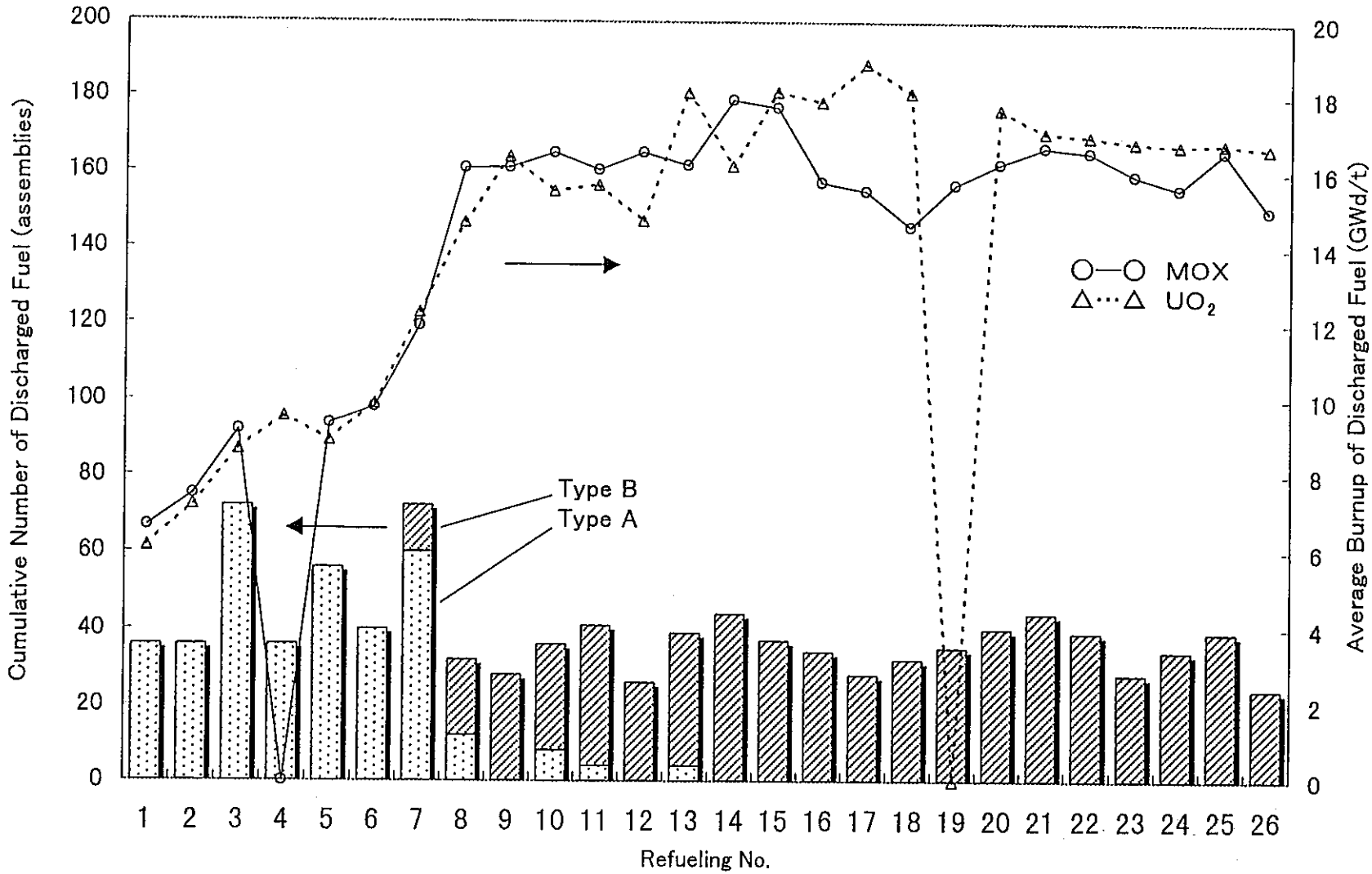
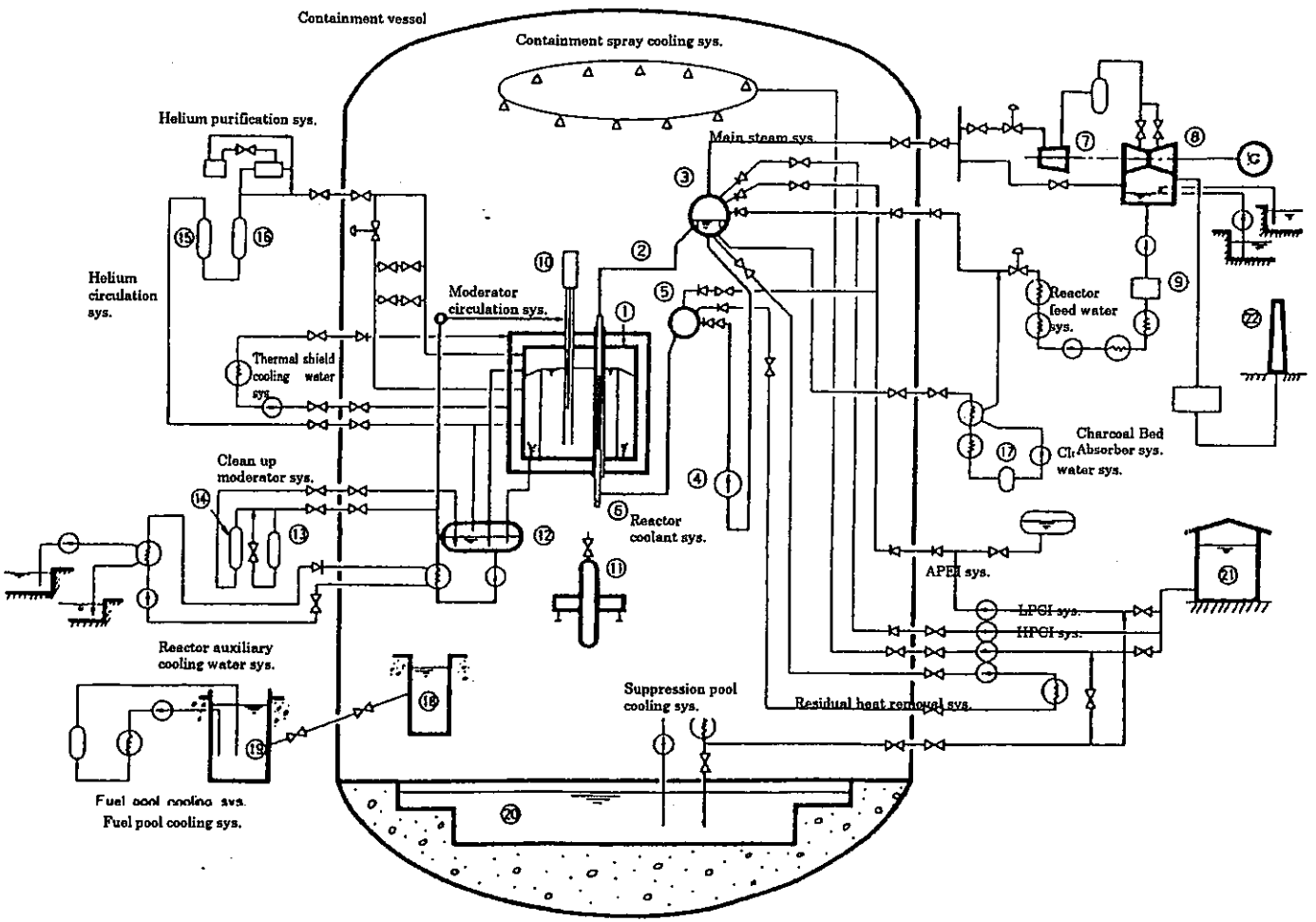


Fig.I.3-7 Cumulative Number and Average Burnup of Discharged Fuel



Key to components

① Calandria tank
② Outlet riser pipes (224)
③ Steam drum (2)
④ Recirculation pump (4)
⑤ Inlet header (2)
⑥ Inlet feeder pipes (224)
⑦ High pressure steam turbine
⑧ Low pressure steam turbine
⑨ Condensate demineralizer
⑩ Control rod drive mechanism
⑪ Refueling machine
⑫ Heavy water dump tank
⑬ Liquid poison removal resin bed
⑭ Heavy water purification resin bed
⑮ Pre-heater
⑯ Recombiner
⑰ Clean up demineralizer
⑱ Fuel exchange pool
⑲ Spent fuel storage pool
⑳ Suppression pool
㉑ Condensated water storage tank
㉒ Main stack

Fig.I.3-8 Schematic Flow Diagram of "Fugen"

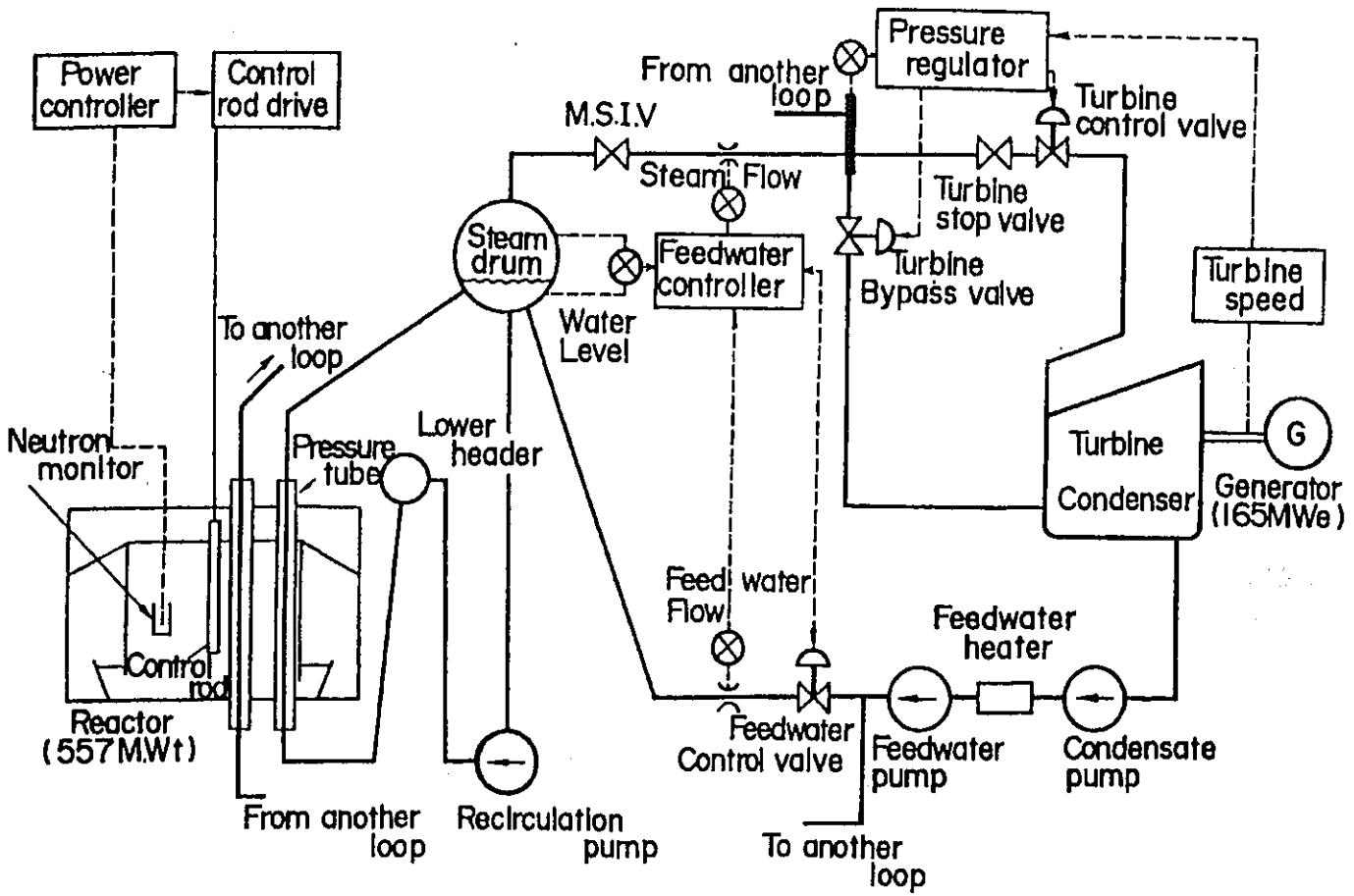


Fig.I.3-9 Plant Control System of "Fugen"

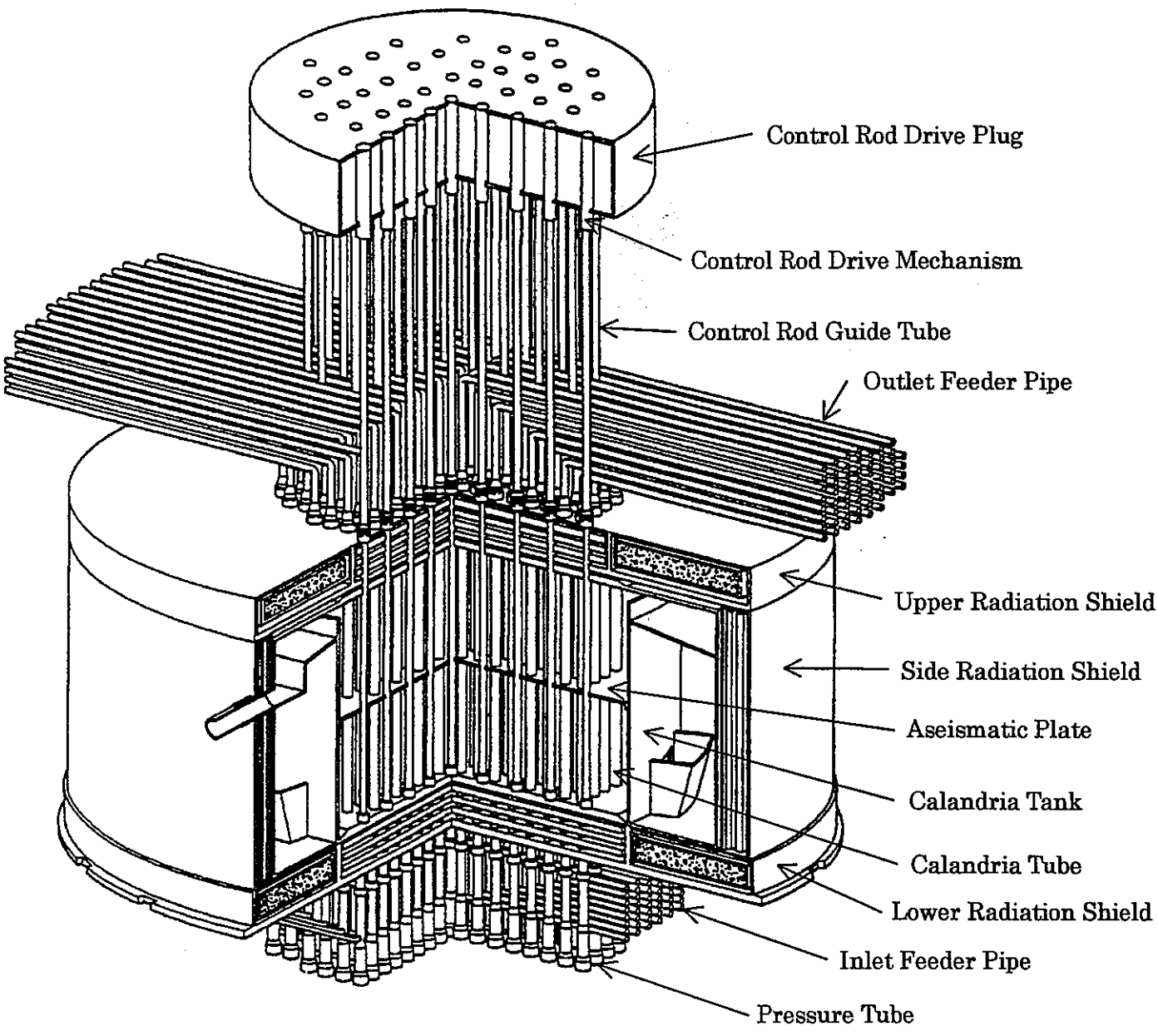
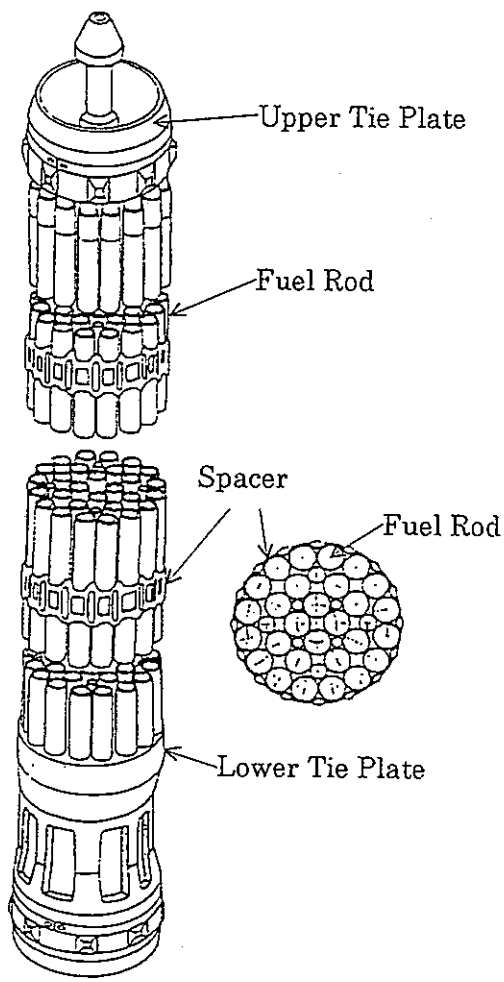
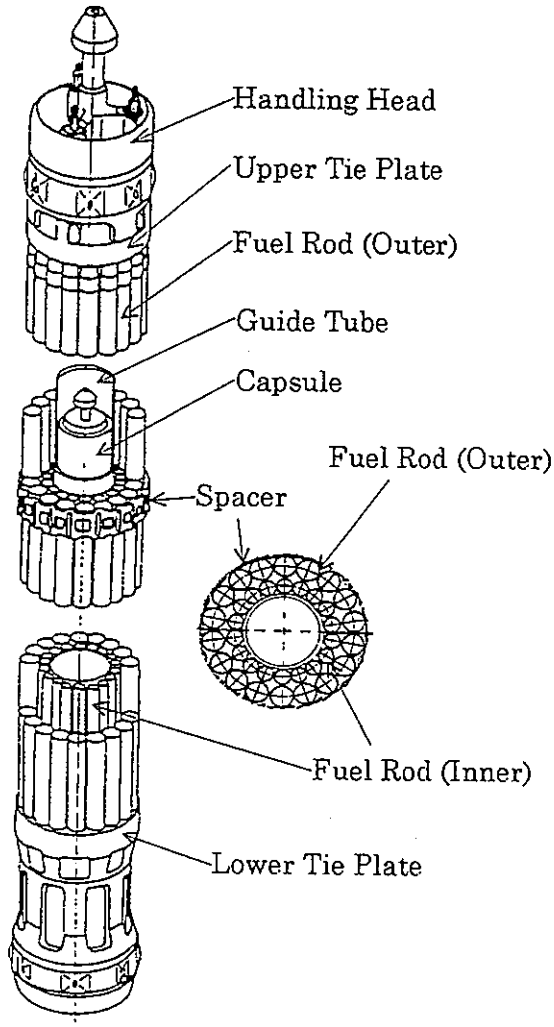


Fig.I.3-10 Vertical Cross Section of Reactor

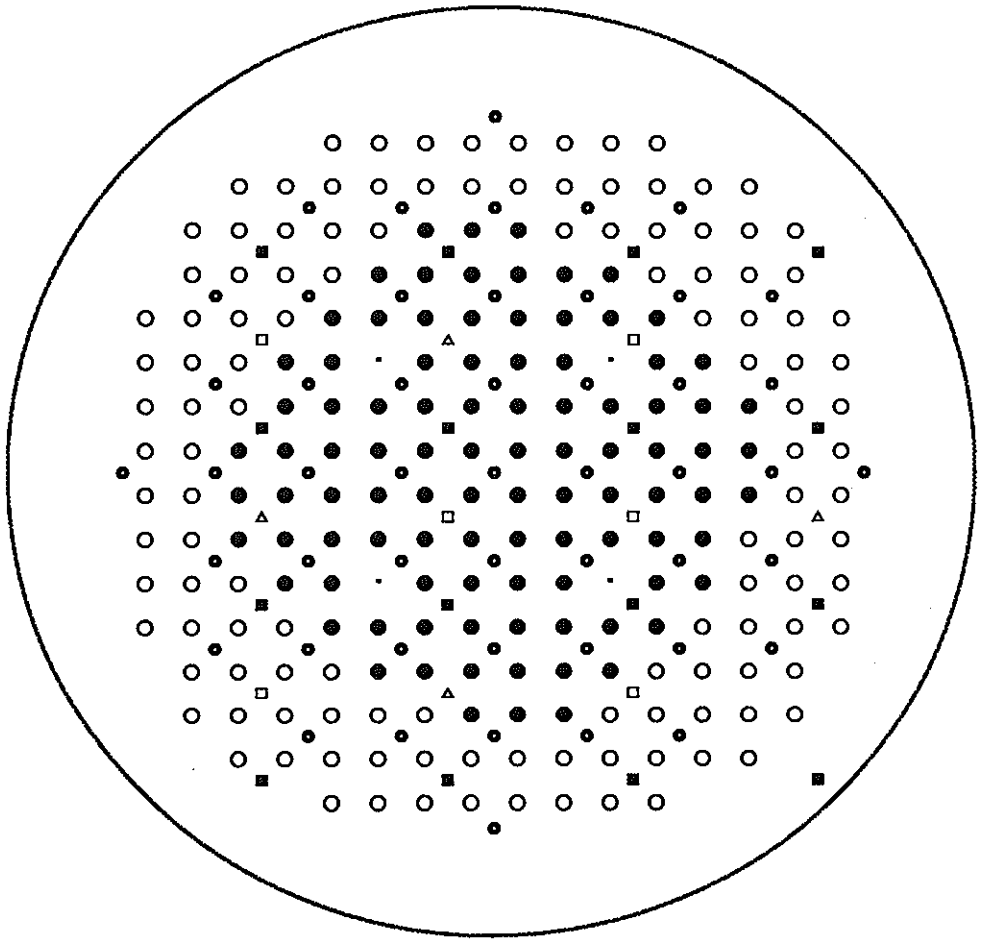


Standard Fuel Assembly



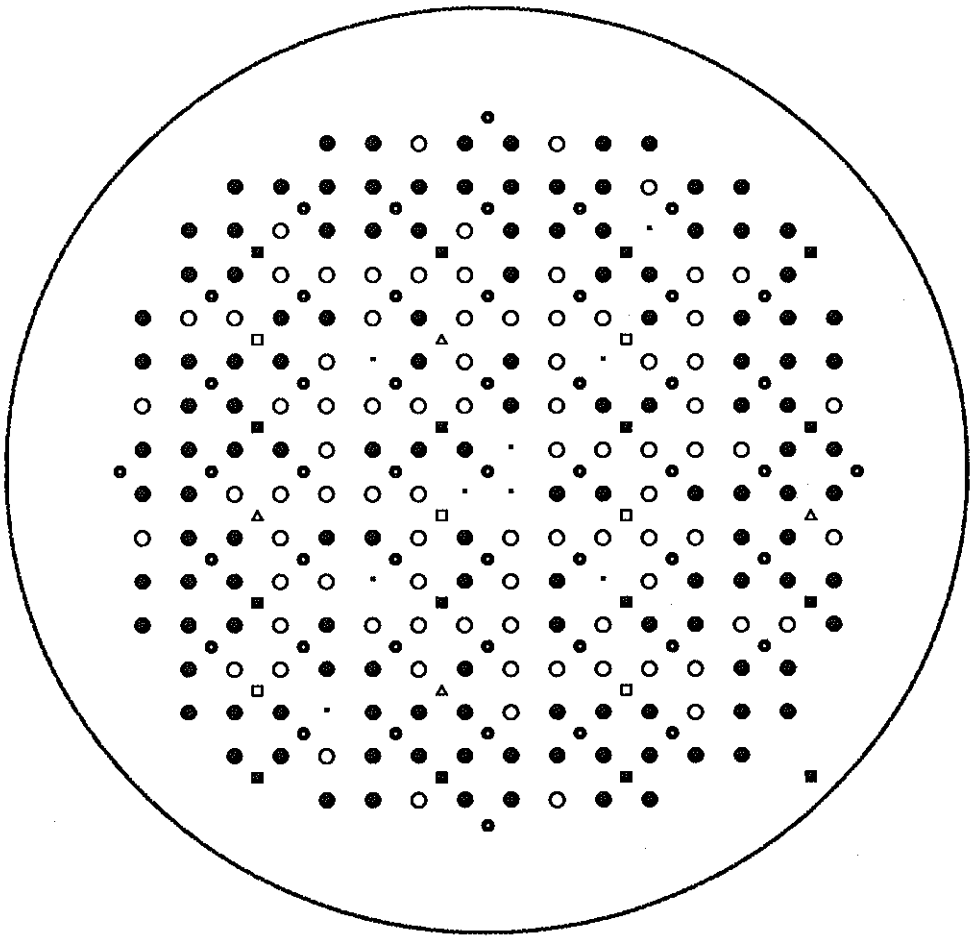
Special Fuel Assembly

Fig.I.3-11 General Schematic View of Fuel Assembly



Note	Equipment	Number	Note	Equipment	Number
●	MOX Fuel Assenmbly	96	◎	Control Rod	49
○	UO_2 Fuel Assembly	124	■	Local Power Monitor	16×4
▪	Special Fuel Assembly	4	□	Power up Monitor	6
			△	Start up Monitor	4

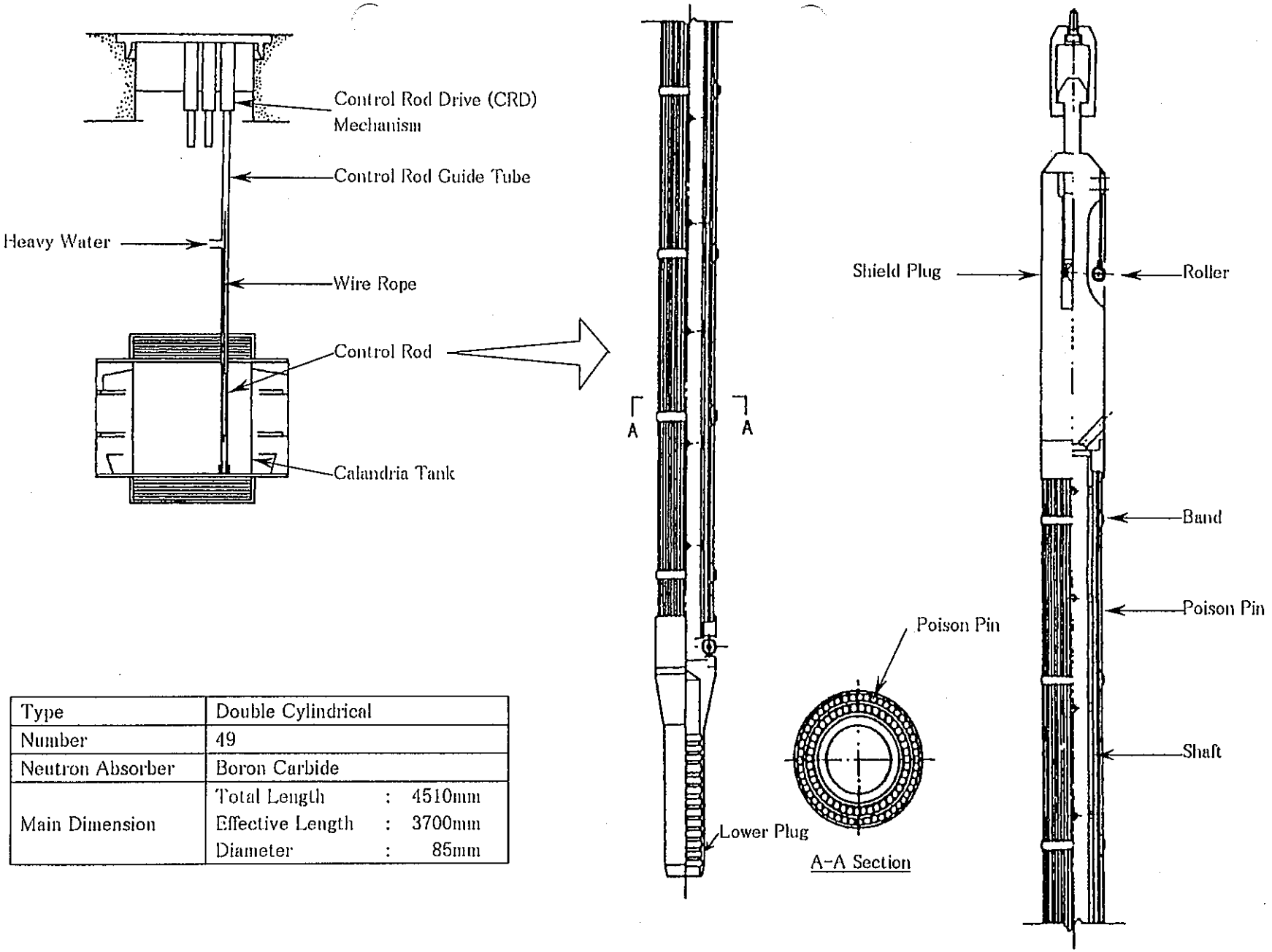
Initial Core



Note	Equipment	Number	Note	Equipment	Number
●	MOX Fuel Assenmbly	133	◎	Control Rod	49
○	UO_2 Fuel Assembly	82	■	Local Power Monitor	16×4
▪	Special Fuel Assembly	4	□	Power up Monitor	6
▪	36 Rod Test Fuel Assembly	3	△	Start up Monitor	4
▪	Segment Test Fuel Assembly	2			

13th Cycle Core

Fig. I .3-12 Core Structure of "Fugen"



Type	Double Cylindrical	
Number	49	
Neutron Absorber	Boron Carbide	
Main Dimension	Total Length	: 4510mm
	Effective Length	: 3700mm
	Diameter	: 85mm

Fig.I.3-13 Control Rod Assembly

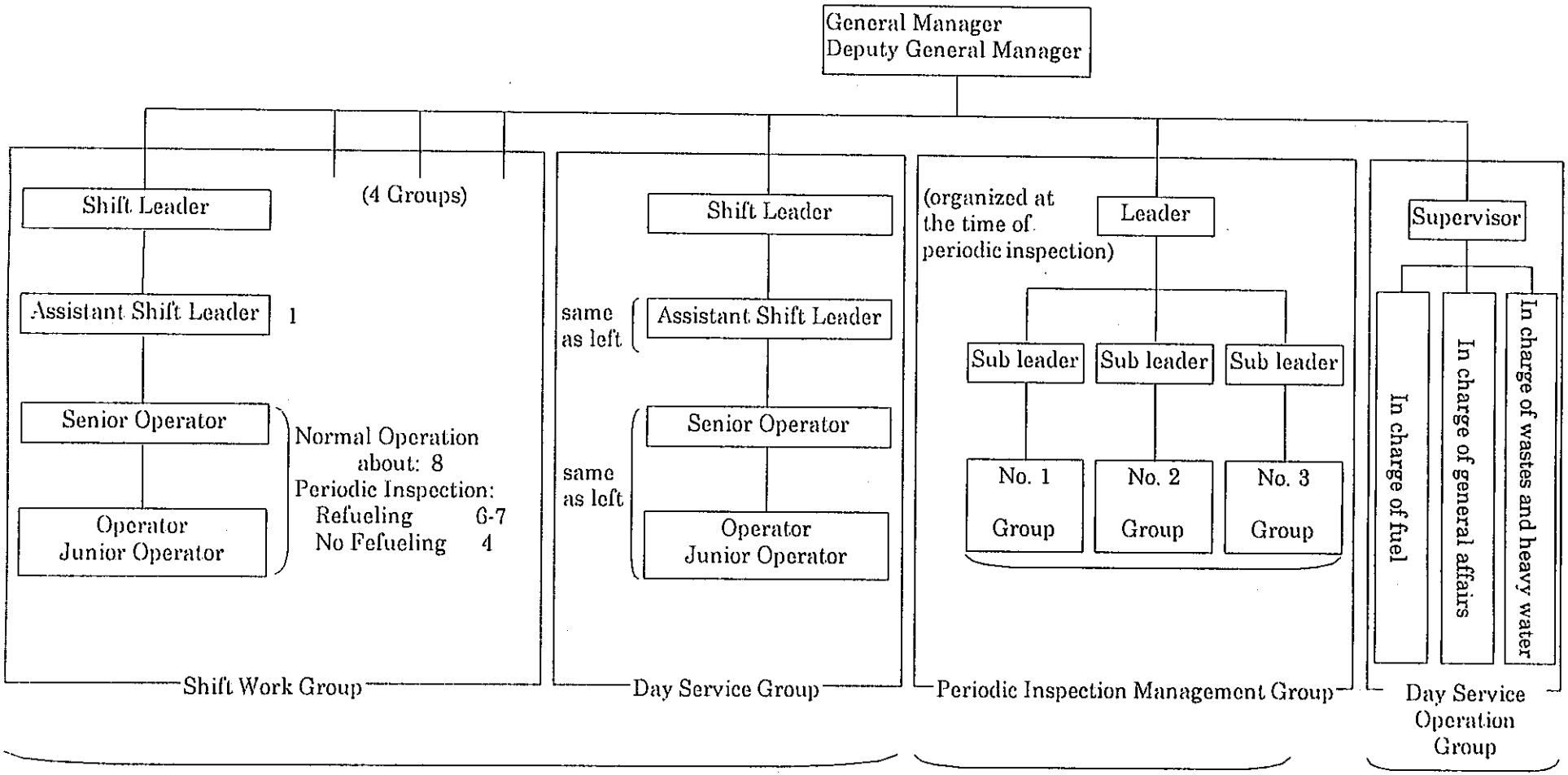


Fig. I.3-14 Operation Management System in "Fugen"

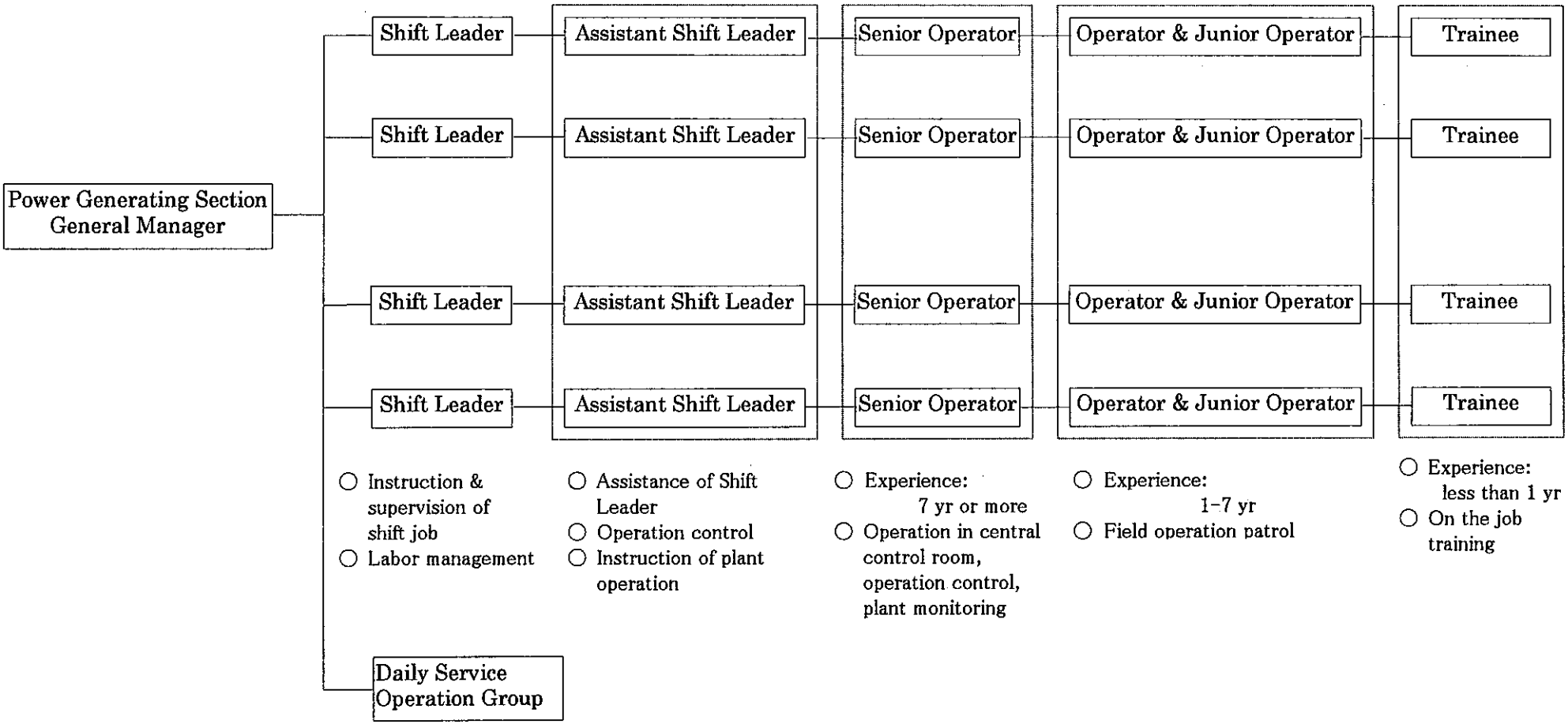


Fig.I.3-15 Operation Management System and Power Generating Jobs

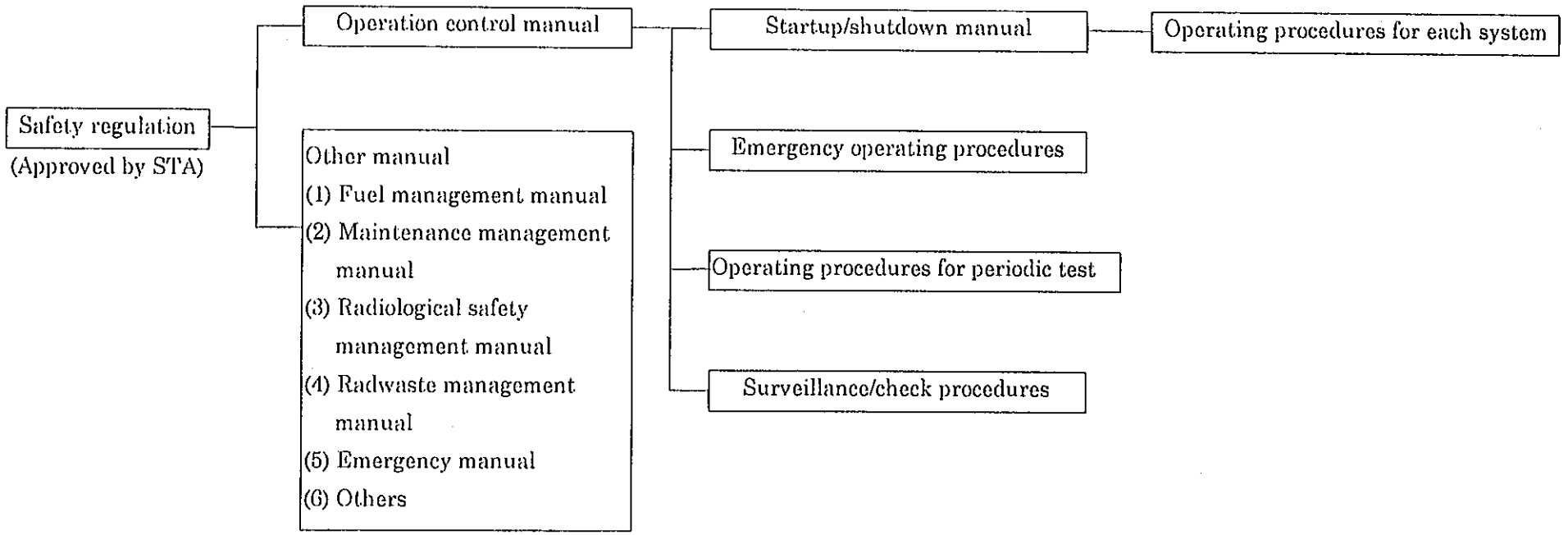


Fig.I.3-16 System of Operating Procedures of "Fugen"

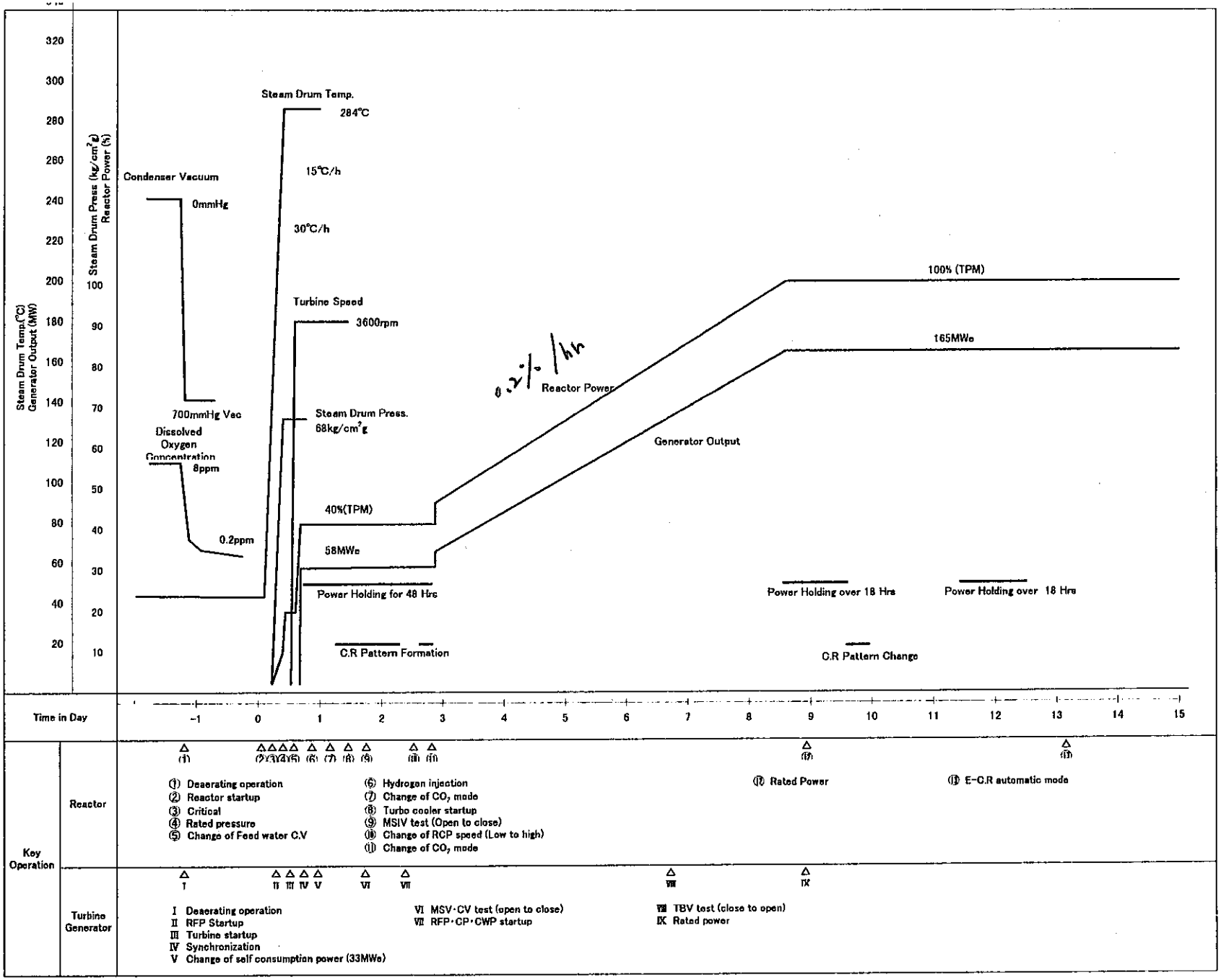


Fig.I.3-17 Plant Startup Procedures

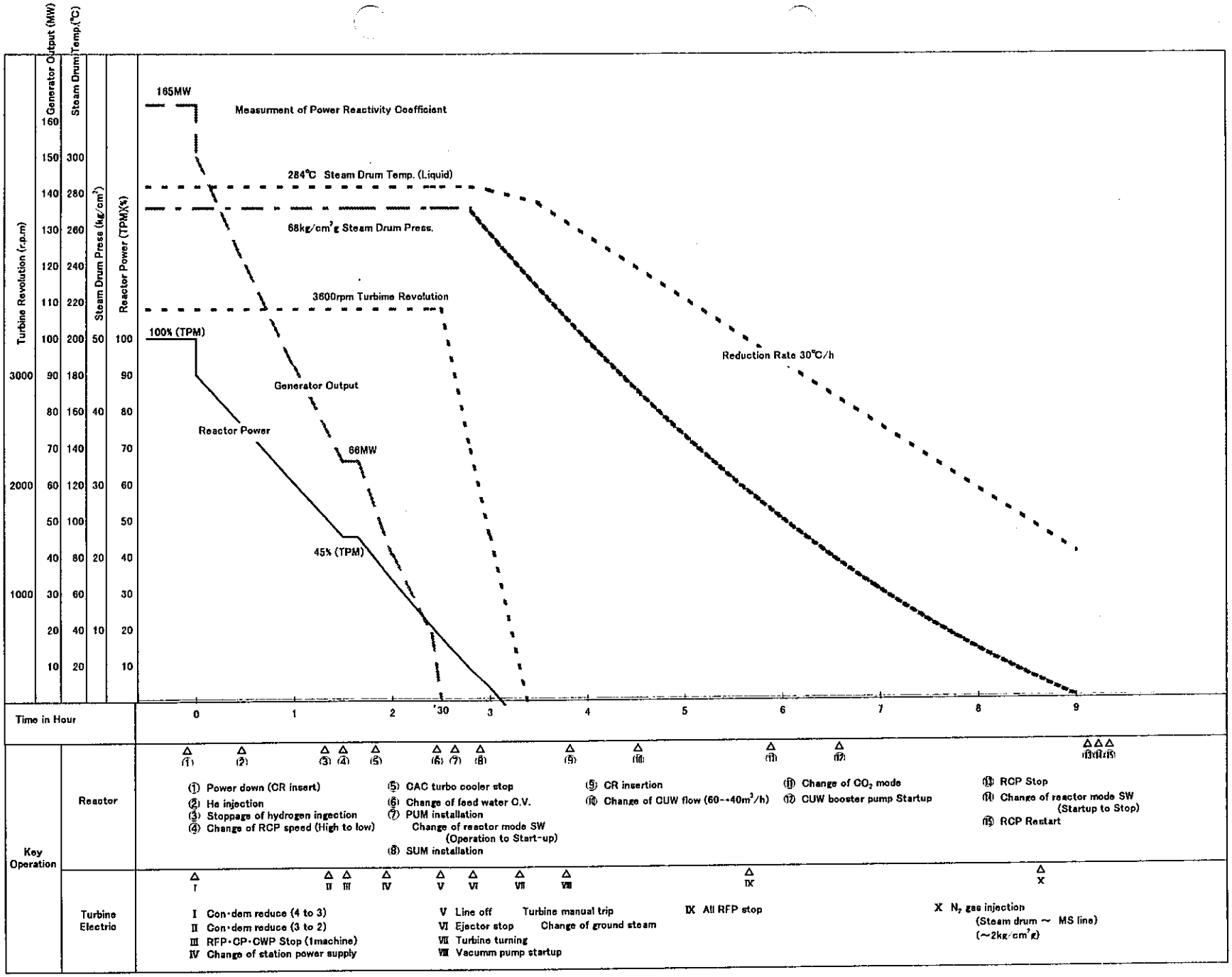


Fig.I.3-18 Plant Shutdown Procedures

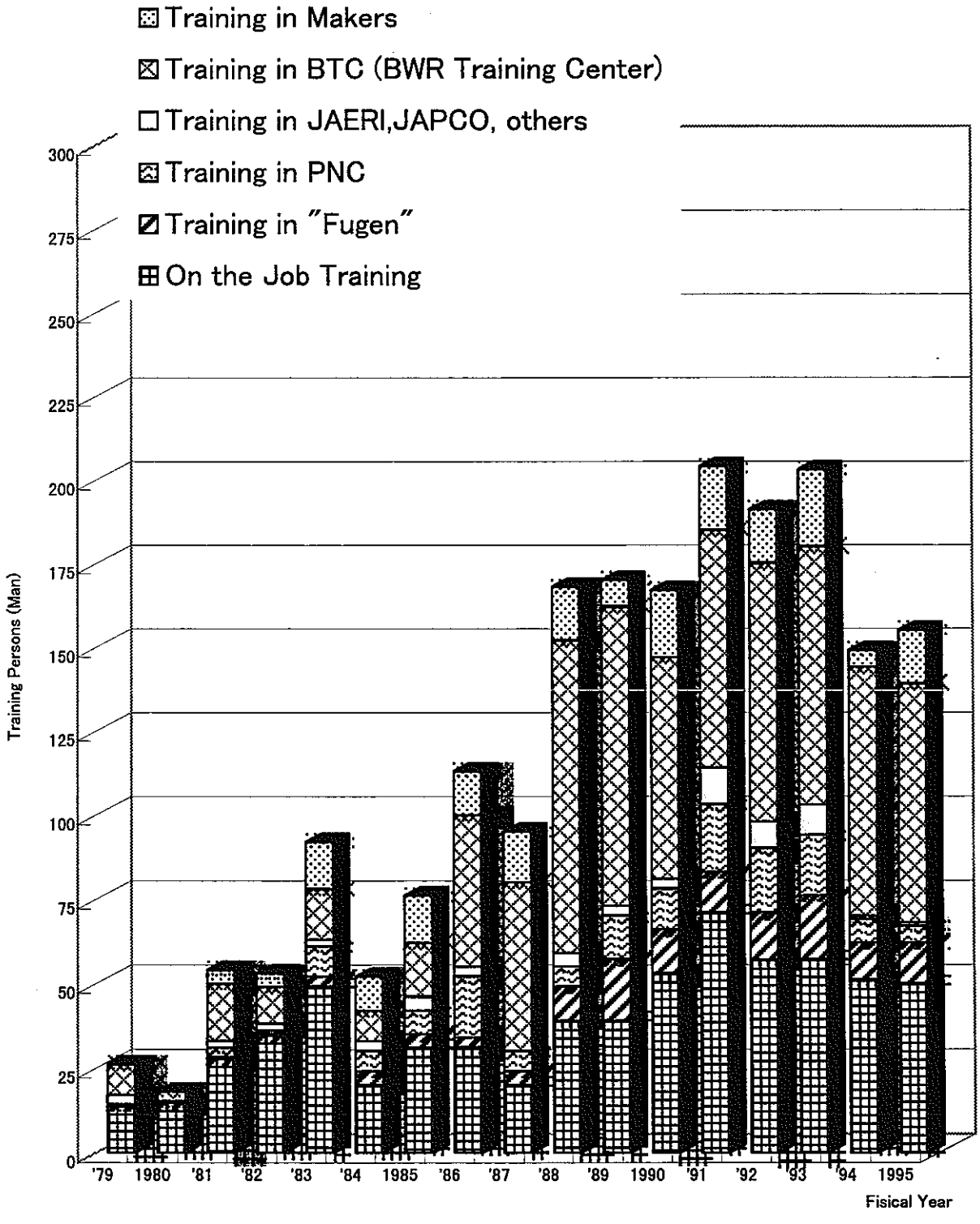


Fig.I.3-19 Result of Operator Training

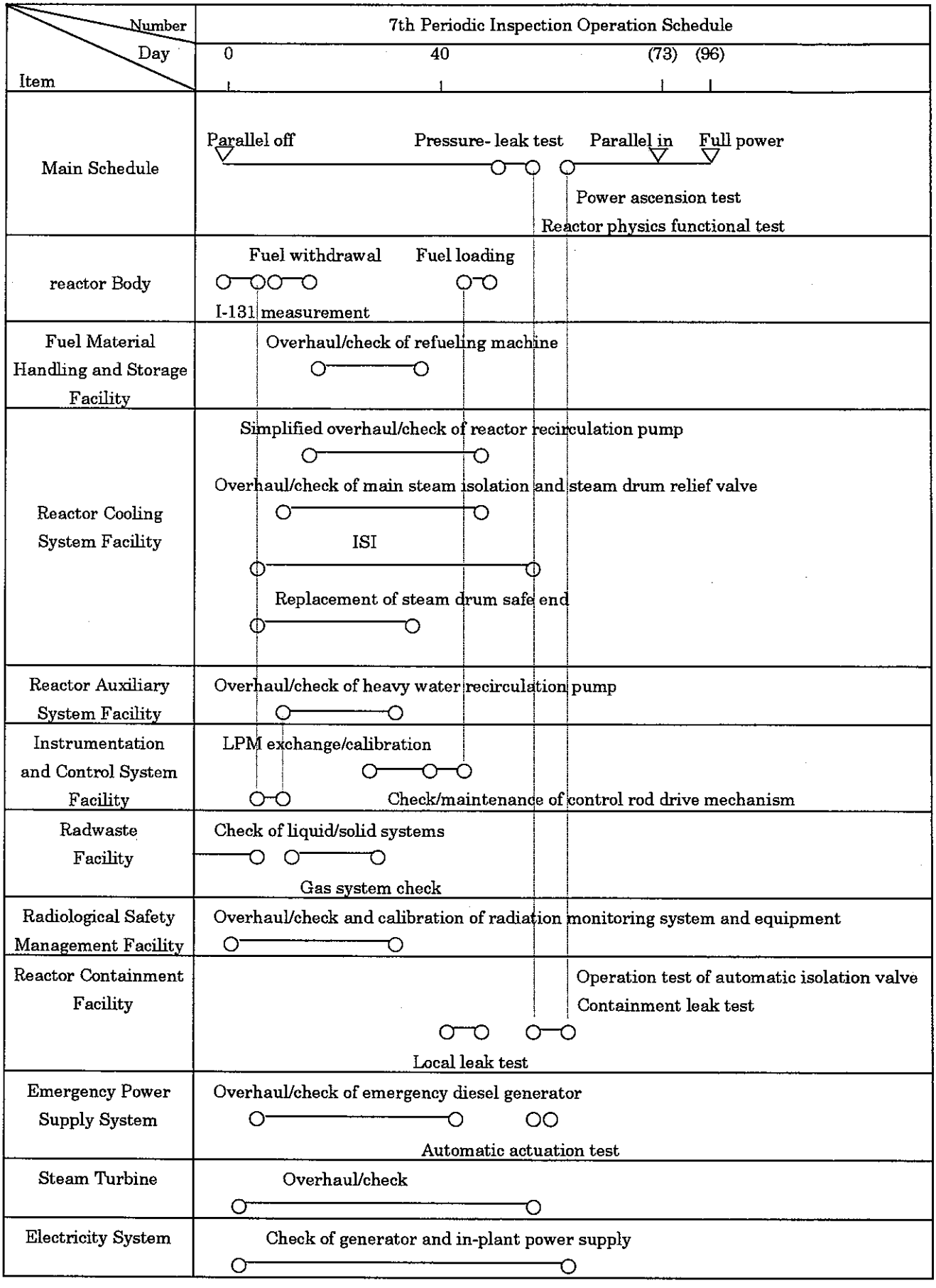
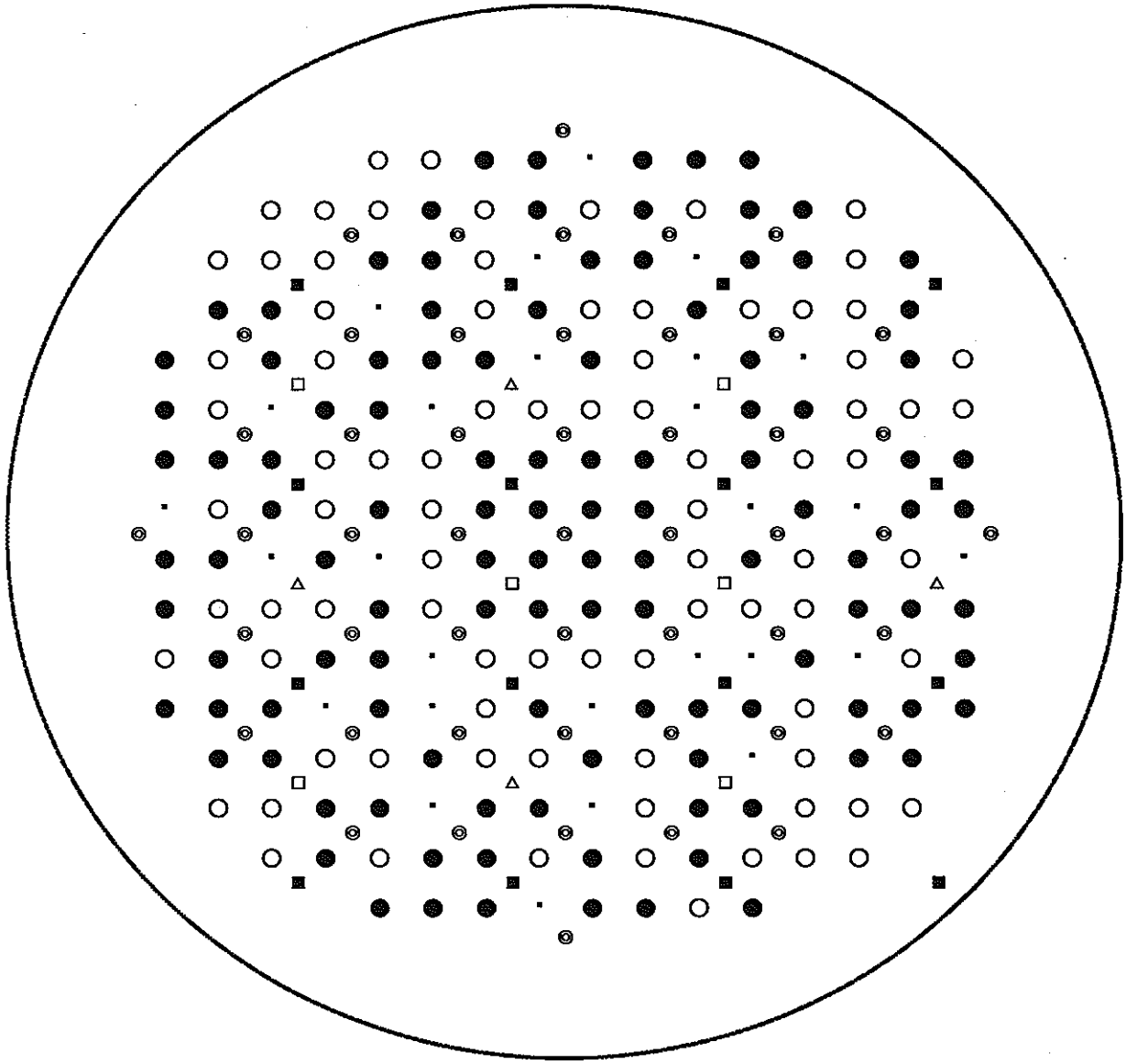


Fig.I.3-20 7th Periodic Inspection Operation Schedule



Note	Equipment	Number	Note	Equipment	Number
○	UO_2 Fuel Assenmbly	84	◎	Control Rod	49
●	MOX Fuel Assembly	112	■	Local Power Monitor	16×4
▪	Special Fuel Assembly	4	□	Power up Monitor	6
▪	UO_2 Fresh Fuel Loaded in This Cycle	6	△	Start up Monitor	4
▪	MOX Fresh Fuel Loaded in This Cycle	18			

Fig.I.3-21 Fuel Loading Pattern
(Core Configuration of 27th Cycle)

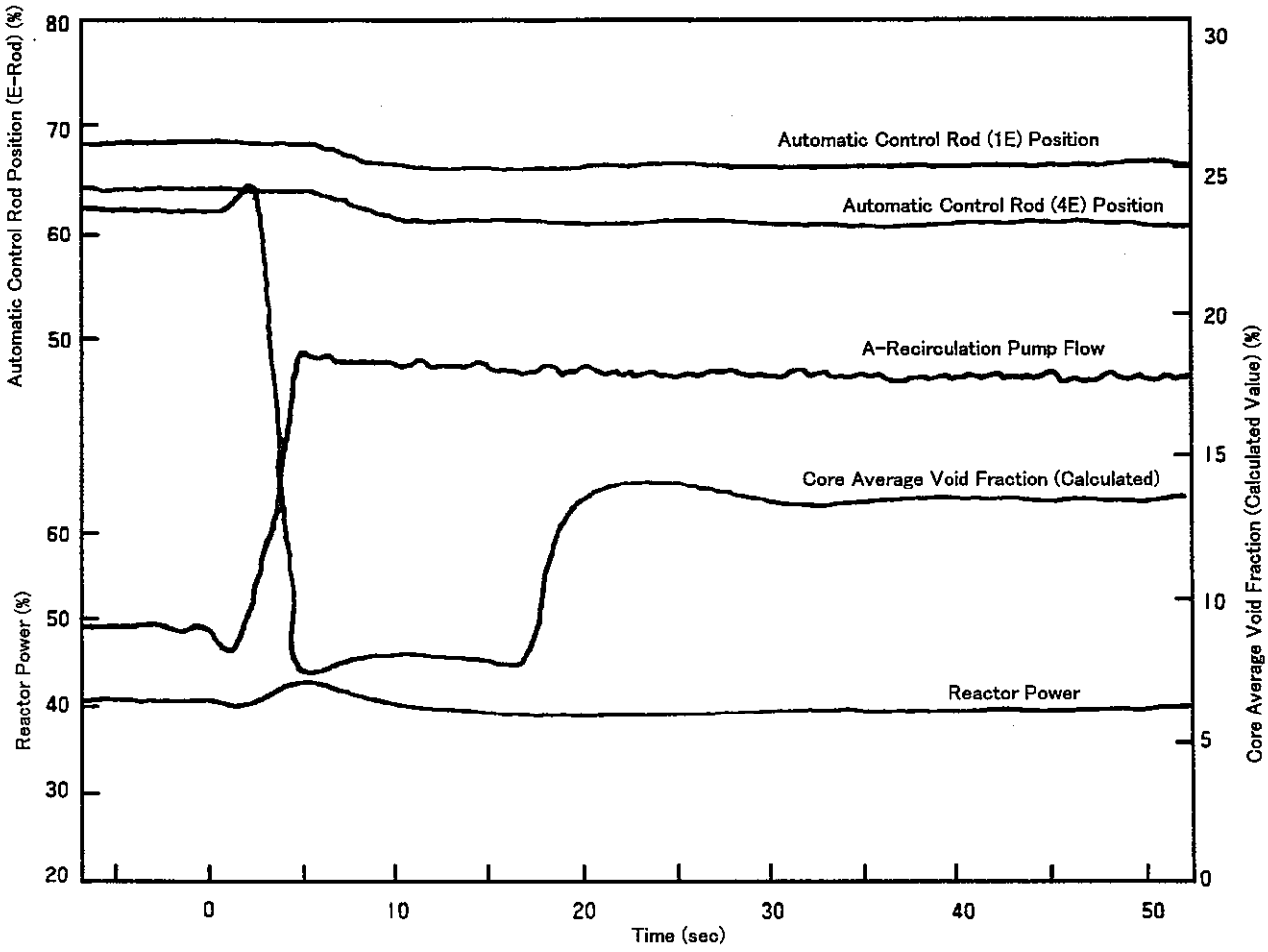


Fig.I.3-22 Plant Transient Characteristics on Recirculation Pump Speed Change

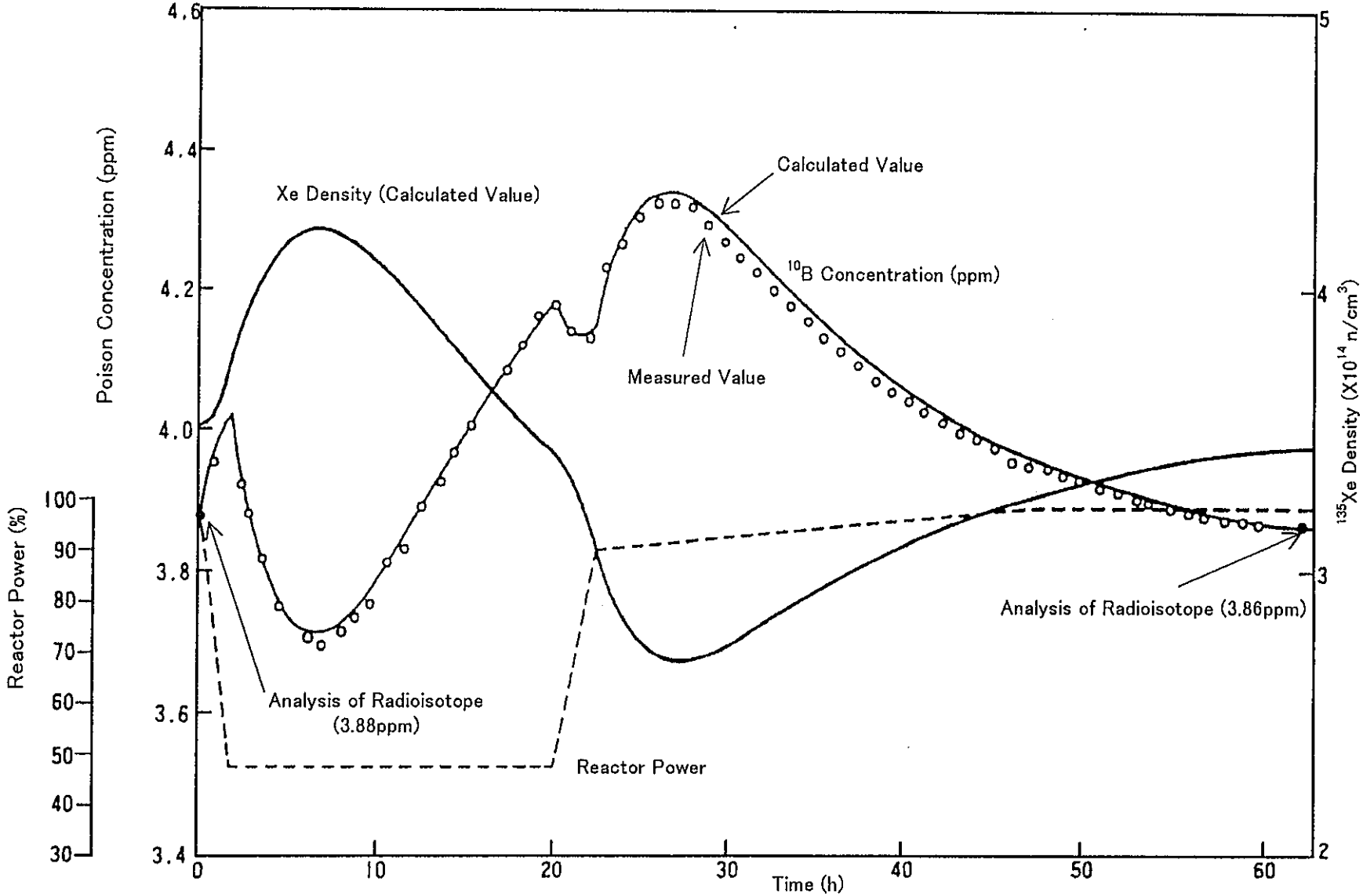


Fig.I.3-23 Poison Concentration after Reactor Power Change

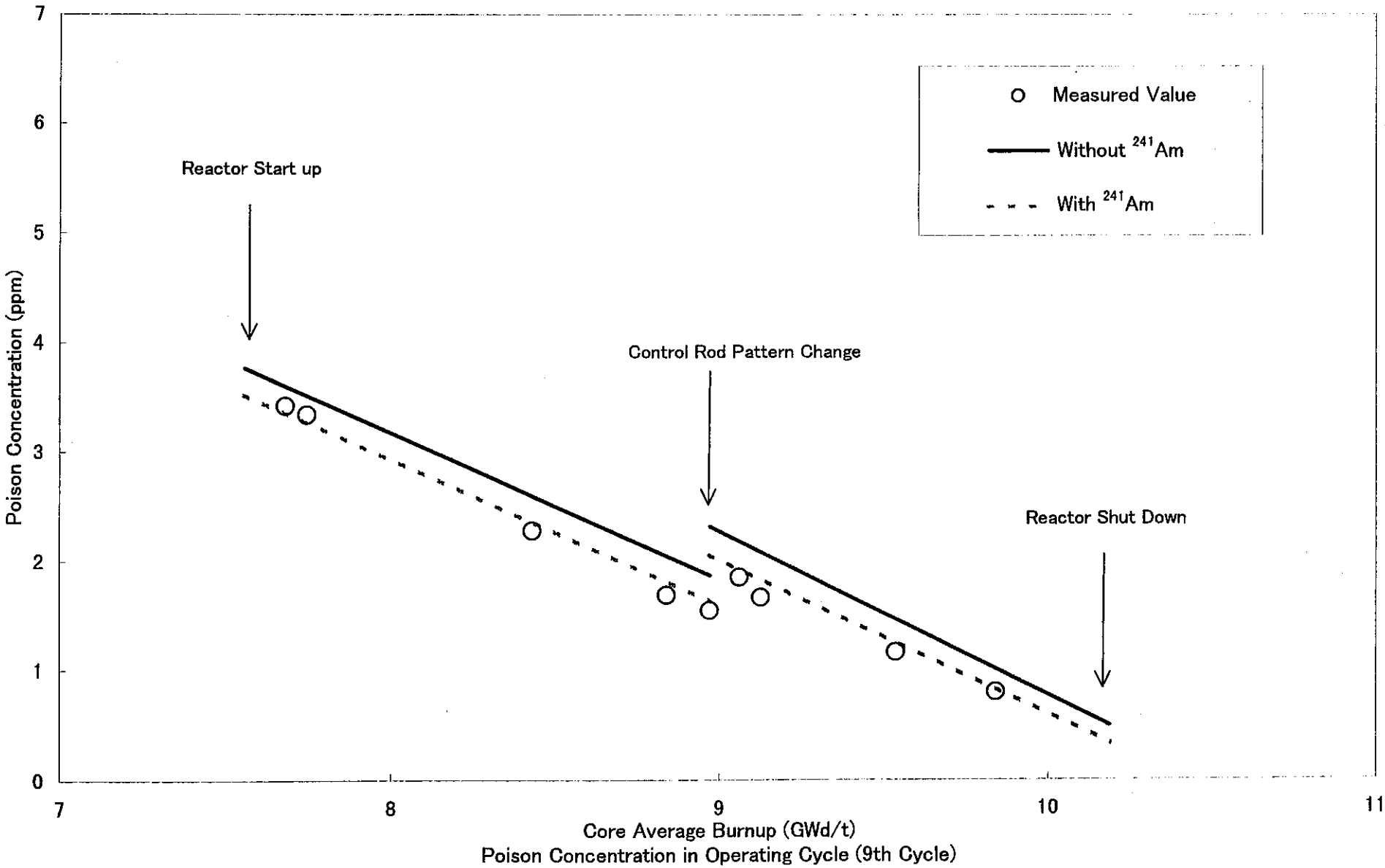


Fig.I.3-24 Change in Poison Concentration with Fuel Burnup

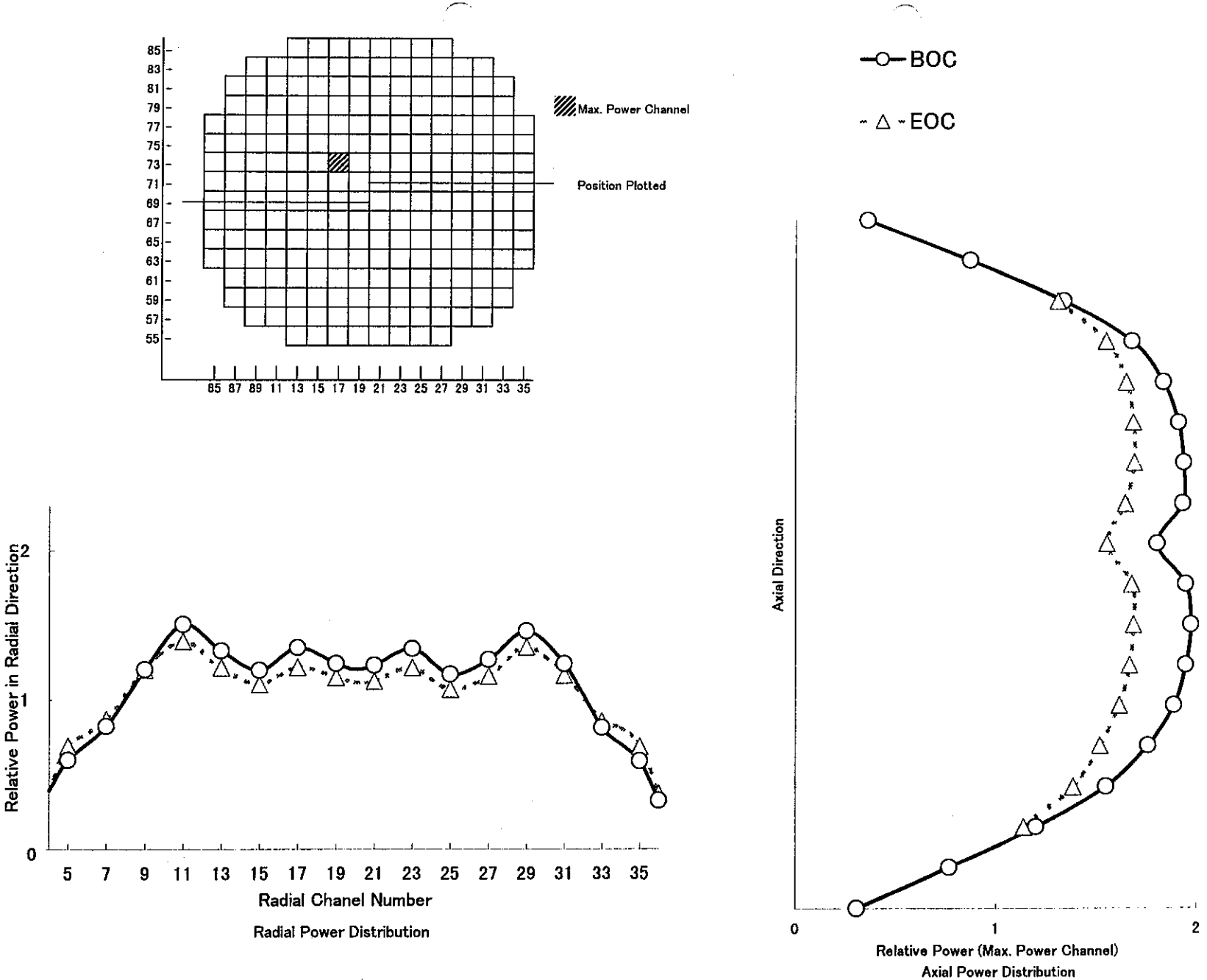


Fig.I.3-25 Radial and Axial Power Distribution Changes in Operating Cycles

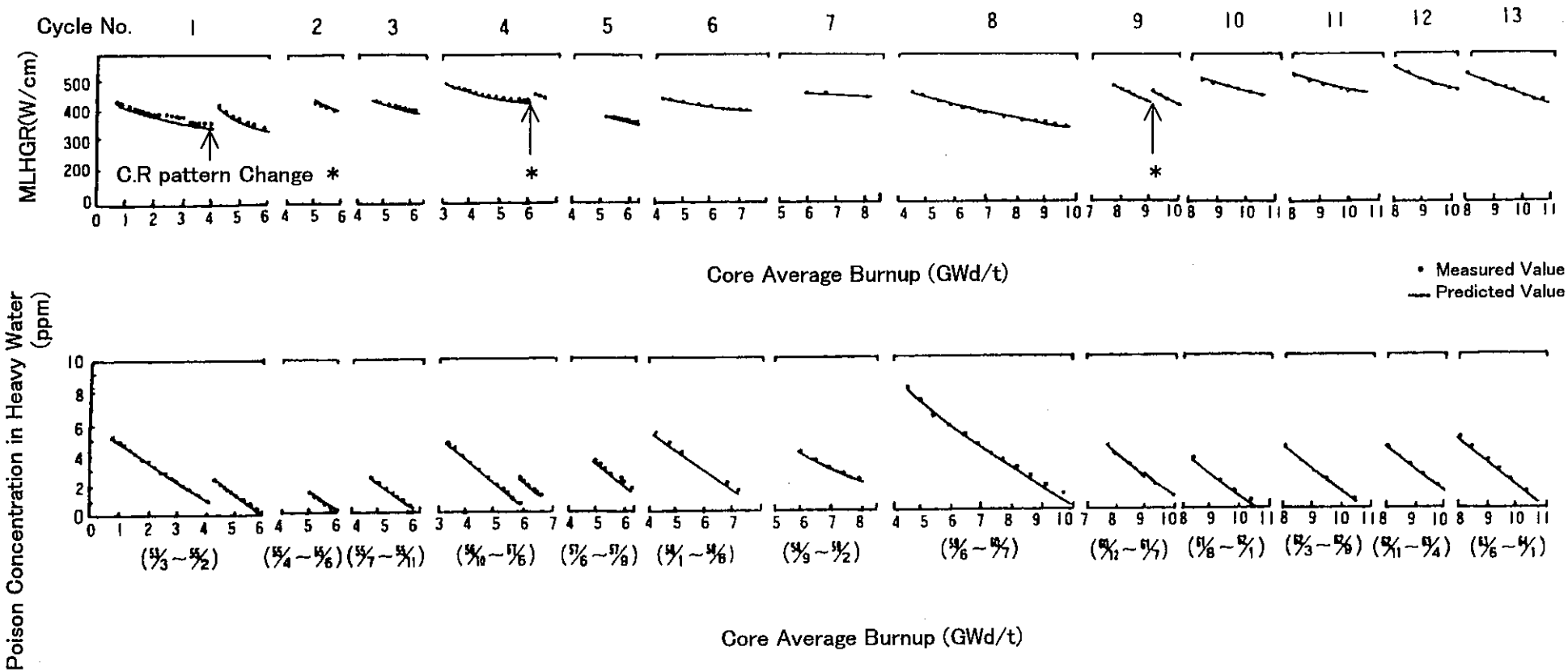


Fig.I.3-26 Change in Maximum Linear Heat Generation Rate and Poison (^{10}B) Concentration with Fuel Burnup

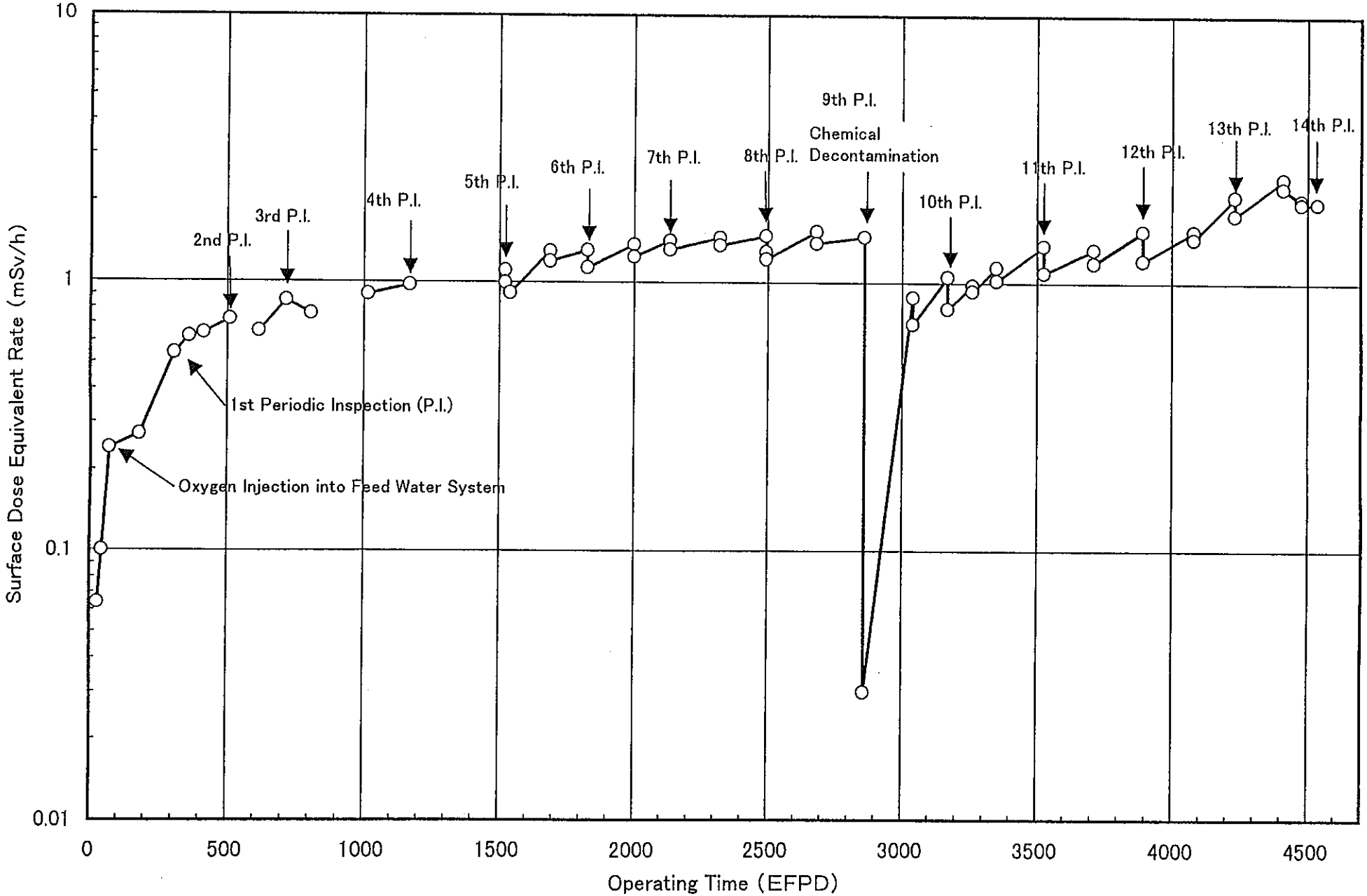


Fig.I.3-27 Surface Dose Equivalent Rate of RCP Outlet Piping

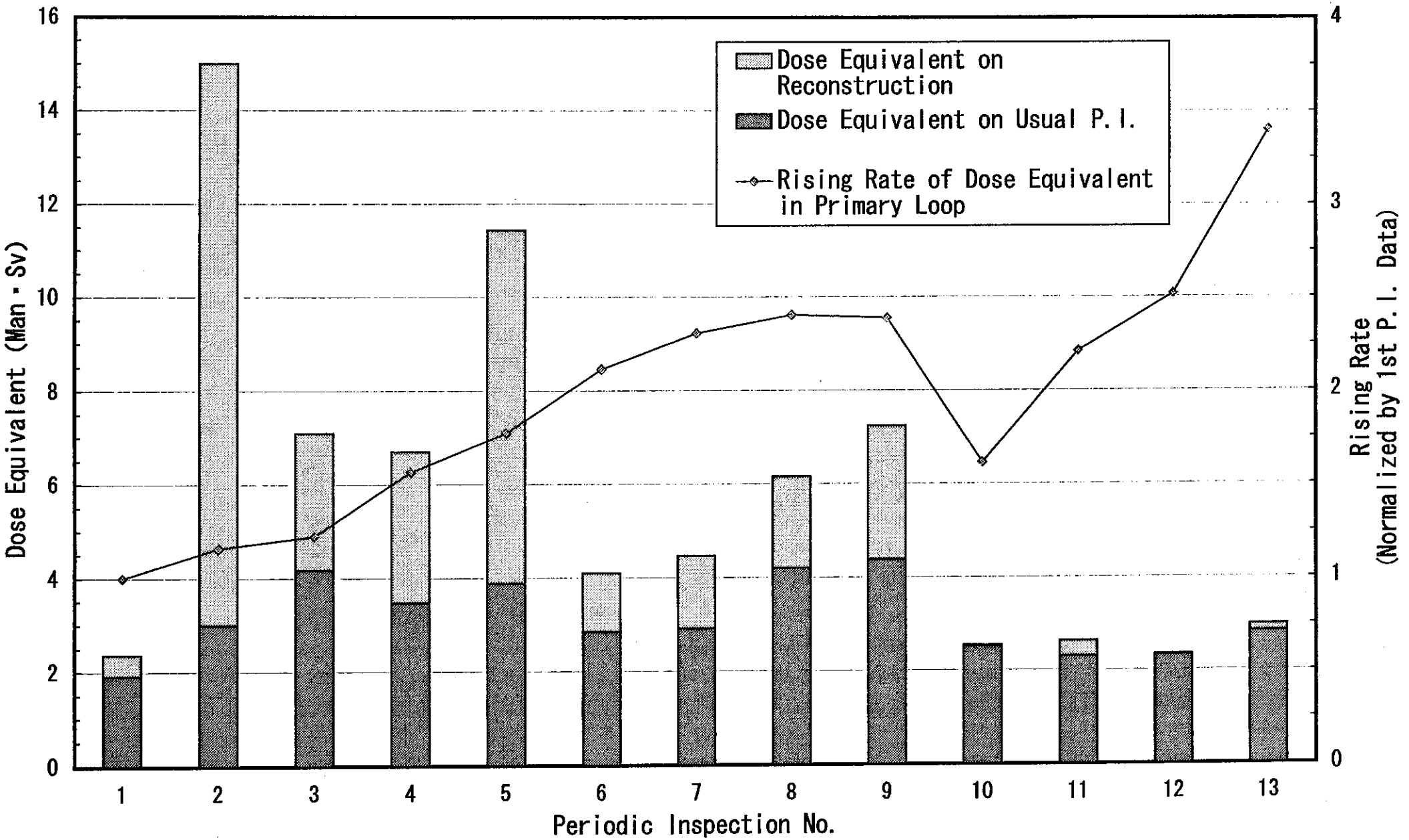


Fig.I.3-28 Radiological Safety Management in Fugen

Table I.3-1 Plant Data of "Fugen"

Location	Tsuruga, Fukui prefecture
Owner and operator	Power Reactor & Nuclear Fuel Development Corporation
Designers	Power Reactor & Nuclear Fuel Development Corporation
Number of Reactors in Plant	One
Rated output per reactor	Gross Thermal power 557MW Gross electrical 165MW Net electrical 148MW Self consumption 17MW
Fuel	Slightly enriched (1.5-1.9%) UO_2 and MOX (0.7-1.3% Puf)
Moderator and reflector	Heavy water (D_2O)
Heat Transport Fluid	Boiling light water
Type	Vertical pressure tube
Construction schedule	Start of construction December, 1970
In-service dates	March, 1979
Buildings and structures	
Reactor building	
Form	Cylindrical wall with elliptical concrete dome
Material	Reinforced concrete
Design pressure	1.35kg/cm^{2}G
Internal diameter	36m
Height of cylindrical portion	67.5m
Wall thickness	1,050mm
Reactor auxiliary building	
Form	Five-storey steel frame
Materials (floors)	Reinforced concrete
Length of building	40m
Turbine building	
Form	Three storey Steel frame
Length	84m
Width	32m
Height	30.4m
Service wing	
Form	Five-storey steel frame
Length	84m
Width	21m
Reactor vessel	
Pressure tubes	
Quantity	224
Material	Zr-2.5%Nb
Inside diameter	117.8mm
Wall thickness	4.3mm
Length	4,770mm

Calandria tank	
Form	Cylindrical vessel
Material	Austenitic Stainless Steel
Height	5,040mm
Outside diameter	
Inner zone	4,910mm
Outer zone	7,950mm
Plate thickness	
Inner zone	25mm
Outer zone	27mm
Radiation shields	
Form	Cylindrical vessel
Material	Carbon steel and H_2O
Height	6,980mm
Outside diameter	9,450mm
Calandria tubes	
Quantity	224
Material	Zircaloy-2 seam welded
Inside diameter	156.4mm
Wall thickness (nominal)	1.9mm
Dump tank	
Form	Cylindrical vessel
Material	Austenitic stainless steel
Inside diameter	2,700mm
Length	6,266mm
Wall thickness	18mm
Reactor physics	
Core data	
No. of fuel channels	224
Cell array	approximately circle 16x16
Lattice pitch	240mm
Core radius	4,050mm
Core length	3,700mm
Average reflector thickness at midpoint	400mm
Fuel load (MOX)	38.3/t
Average thermal flux in core	$5x10^{13}$n/cm^2・sec
Burnup (thermal energy)	20,000MWd/t (for MOX fuel)
Blanket gas	Helium
Control Rods	
Material	Born carbide in stainless steal tube
Purposes	Power regulating, neutron flux flattening and reactor shutdown
Number	49 (4 for power regulating 0-5 for neutron flux flattening and residual rods for reactor shutdown)
Orientation	Vertical

Liquid poison	
Material	Enriched-boron dissolved in moderator
Purposes	Suppression of excess reactivity for burnup
Concentration	Below 10ppm as ^{10}B
Other shutdown provisions	Dumping of D_2O moderator by gas balance scheme
Moderator	
Type	Heavy water
Total volume in core	81.5m^3
Purity	99.75%D_2O by weight
Temperature, maximum	70℃
Heat produced in moderator	33MWth
Flow through Calandria	1,320m^3/hr
Moderator temperature rise	13℃
Helium pressure in dump tank	0.54kg/cm^2g
Heat transport fluid	
Type	Boiling light water
Quantity	56t
Total mass flow	7,600t/hr
Fluid temperature	
at reactor	277℃
at reactor outlet	284℃
No. of pumps	4 in 2 groups of 2
Steam drum	
Number of boilers	2
Stream pressure at drum	68kg/cm^2g
Steam temperature at drum	284℃
Total steam flow	910t/hr
Steam quality at reactor outlet	14%
Feed water temperature	182℃
Fuel	
Type	28 bundles 4100mm long
Diameter of fuel pin	16.5mm including Zircaloy-2 sheath
Pellet diameter	14.4mm
Minimum cladding thickness	0.8mm

Total weight of PuO_2-UO_2 per bundle	174kg
Bundle diameter	111.6mm
Heat transfer area per bundle	5.36m^2
Heat flux average	3.7x10^5kcal/m^2·hr
Maximum	9.5x10^5kcal/m^2·hr
Fuel temperature maximum	2,200℃
Maximum cladding temperature	350℃
Turbine Generator	
Turbine	One tandem compound unit per reactor with external moisture separation
Number of high-pressure cylinders	1 single flow
Number of low-pressure cylinders	2 single flow
Speed	3,600rpm
Throttle pressure	65kg/cm^2g
Type of condenser	One pass
Cooling water flow	35,580m^3/hr
Generator	One per turbine-200,000kVA 3 phase, 60 cycle, 16kV
Transformers	
Main Power	178,000kVA, 3 phase, 60 cycle, 16/275kV
Station service unit	27,500kVA, 3 phase, 60 cycle, 16/6,9kV
System	22,000kVA, 3 phase, 60 cycle, 275/6,9kV
Circuit breakers	
275kV	2-3 cycle, gasblast(SF_6), 2,000amp, 40,000MVA
6,9kV	390MVA
Standby generation and switching	
Diesel Generator	6,000kVA, 6.9kV
Circuit Breakers	7.2kV, 390MVA

Table I.3-2 Classification and Qualification of Operators

Classification	Qualification
Trainee (experience : less than 1 yr)	(1) New personnal (2) Personal from other Section (unexperienced operator) (3) New personal from other organizations (N.B.) As to (2) and (3), training period can be shortened depending on personal history.
Junior Operator (experience : 1 - 4 yr)	Operators having 1 yr or more experience after joining PNC and finishing the training goal as trainee, or recognized by Power Generating Section General Manager.
Operator (experience : 4 - 7 yr)	Operators having 4 yr or more experience after joining PNC and finishing the training goal as junior operator, or recognized by P.G.Section General Manager.
Senior Operator (experience : 7 yr or more)	Operators having 7 yr or more experience after joining PNC and finishing the training goal as operator, or recognized by P.G.Section General Manager.
Assistant Shift Leader	Senior operators having satisfactory knowledge of nuclear energy and skill and ability of instructing followers and ability as supervisor, or recognized by P.G.Section General Manager.
Shift Leader	Those who have sufficient knowledge of nuclear energy and skill, and ability of instructing followers and leadership as leader.

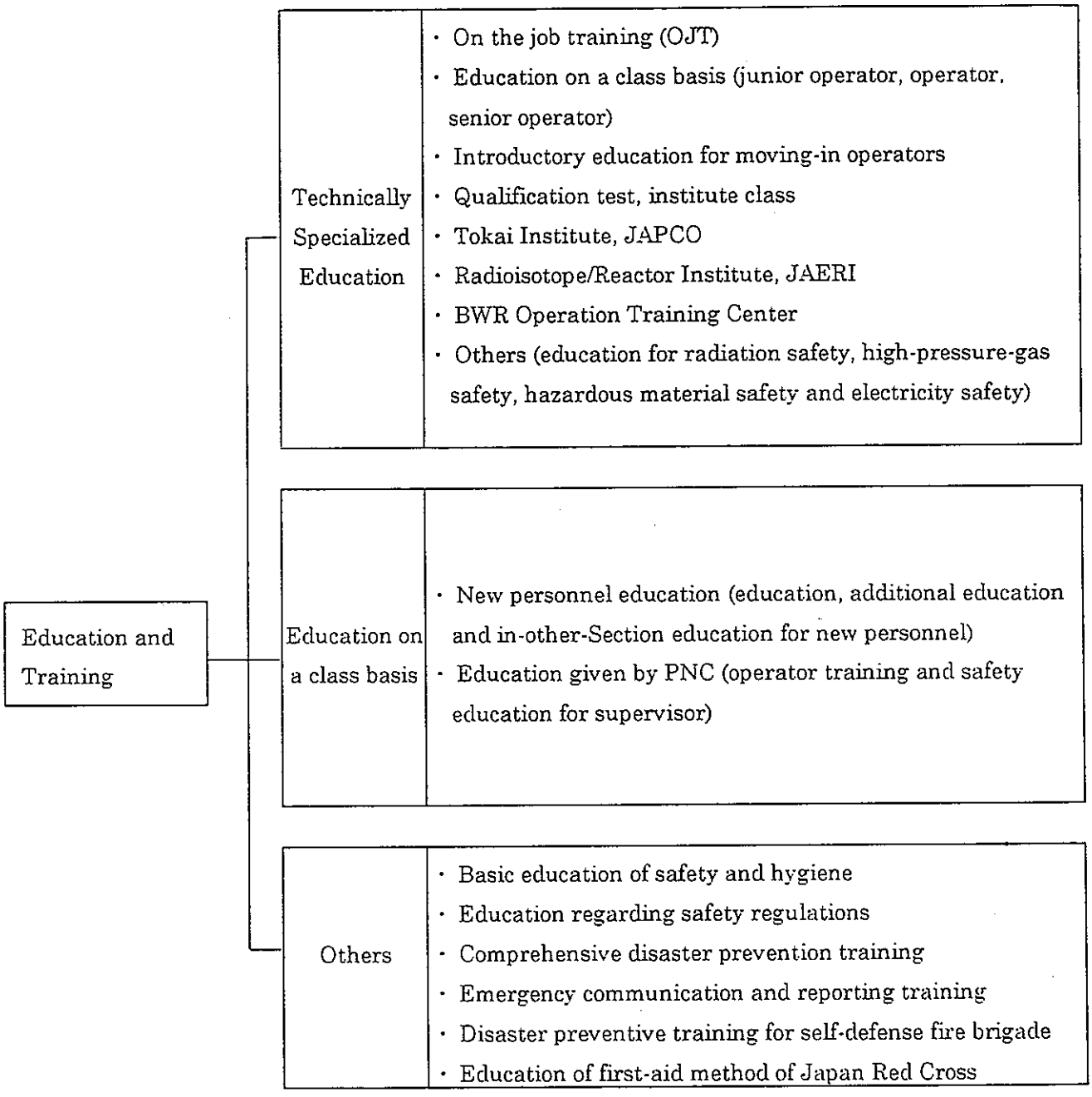


Table I.3-3 Education and Training System of "Fugen"

Table I.3-4 Education/Training System for "Fugen" Operators

Classification	Year of experience	In-Section Education				Outside Training	
		On-the-Job Training	Desk Education	Emergency Training	Training Using Operation-Grade-up Supporting Device	BTC Training	Others
Trainee	0	(1) Outlines of equipment/ system (2) Arrangement of field equipment (3) Practice of patrol (4) Operational practice of auxiliary equipment	(1) Introductory education (scout a week) • Mental attitude of operators • Surveillance/Check procedures (2) Junior operator education •System education(I) •System education(II) (10 days each)	(1) Study of emergency operating procedures	(1) Normal operation training		(1) New personal education •Self study report •Training in other Sections •Results presentation meeting
Junior Operator	1 • 3	(1) Operation of field equipment and system (2) Patrol (3) Simple periodic test (4) Simple isolation (5) Radwaste treatment practice	(3) Operator education •Control system education (5days)	(1) Confirmation of emergency operating Procedures (2) Simulation training for emergency (3) Discussion and reflection of result of training •Emergency simulation training items ① Reactor scram/ turbine trip accident ② Loss of external Power accident ③ Loss of coolant accident ④ Fuel handling accident ⑤ Heavy water/ helium accident ⑥ Others	(2) Minor trouble training (1) Normal operation training (2) Plant startup/ shutdown training (3) Emergency operation training	(1) Family training course	(1) Reactor training in JAERI general course (5 months) (2) Training in JAPCO (3.5 months) (3) Training of fuel cycle (4) Technical training in equipment makers (5) Voyage training
Operator	4 • 6	(1) Operation of equipment/ system (2) Startup/shutdown of plant (3) Periodic test (4) Isolation at periodic inspection (5) Training of heavy water purification	(4) Senior operator education •Operation control education(5days)			(2) Standard training course (12 weeks) (3) Short basic course (10 days)	
Senior Operator	more 7	(1) Operation and instruction of plant startup/shutdown (2) Operation and instruction during plant emergency (3) Operation control (4) Operation management				(1) Retraining course (10days) (2) Senior operator training course (5 days)	

Table I.3-5 Key Contents of Periodic Inspection for "Fugen"

Classification	Equipment	Inspection Item
Reactor Proper	Inlet pipe Outlet pipe Pressure tube	In-service inspection (nondestructive inspection and leak testing)
	Calandria tube	Leak testing
	Fuel	Appearance/dimension/leak testing Fuel loading test Reactor shutdown margin test Maximum excess reactivity test
Fuel handling facility and storage	Refueling machine	Scattering test, Functional test
	Fuel handling equipment pool water cooling/purification system	Functional test
	Spent fuel storage rack Fresh fuel storage rack	Storage capacity test
Reactor cooling system facility	Reactor recirculation system steam drum Steam drum relief valve	In-service inspection (non-destructive inspection and leak testing) Scat-tering test, Actuation test, Leak testing
	Main steam isolation valve	Scat-tering test, Actuation test, Leak testing
	High-pressure injection system Low-pressure injection system Rapid water injection system	Functional test
	Isolation cooling system	Functional test
	Equipment/piping, pump, valves	In-service inspection (nondestructive inspection and leak testing)
Reactor auxiliary facility	Heavy water cooling system Helium recirculation system	Functional test

Facility	Equipment	Inspection Item
Instrumentation control facility	Control rod drive mechanism	Overhaul/operation inspection, Scram test
	Nuclear instrumentation	Set value confirmation Interlock confirmation
	Safety system	Set value verification Scram, heavy water dump test Comprehensive interlock function test
	Controlling air system	Functional test
Radwaste facility	Solid waste treatment system Liquid waste treatment system	Treatment capacity test Storage capacity inspection
	Solid waste storage	Storage capacity/state of storage inspection
Radiation control facility	Area monitor Process monitor Monitoring post Devices mounted on radiation monitoring car	Check calibration Set value verification Function confirmation
	Emergency filter Containment vessel air recirculation system Annulus exhaust system Control room ventilation system	Function confirmation Filter performance test
Reactor containment facility	Reactor containment Isolation valve	Leak testing Isolation function test
	Containment vessel spray system	Functional test
Emergency power, supply installation	Emergency diesel generator D.C. power supply system	Functional test
Steam turbine	Turbine body & attachment	Opening/performance tests
Reactor facility	Whole reactor facility	Comprehensive load test

Table I.3-6 Outline of Modification Works

Object	Outline of Modification Work	Implementation Period
Measures for SCC	Installation of desertion system	1st periodic inspection
	Replacement of cooling system piping material with SCC-resistant one	2nd-9th periodic inspection
	Residual stress improvement heat treatment by high-frequency induction heating for cooling piping welds	4th-5th periodic inspection
	Installation of water quality improvement system by hydrogen injection	5th periodic inspection
Measures for corrosion and erosion	Replacement of turbine steam drain piping with erosion-resistant piping	3rd-7th periodic inspection
	Replacement of tube bundle material with corrosion-resistant one	3rd-5th periodic inspection
	Replacement of feed water control valve with corrosion-resistant valve	4th periodic inspection
	Replacement of No. 5 heating pipe material (STPT38) with low-alloy carbon steel (STPA23)	7th periodic inspection
	Replacement of exhaust gas condenser tube bundle material (SUS 304) with SUS 316Z	7th periodic inspection
	Overlaying for moisture separator drum of turbine and auxiliary device and partial replacement of main piping	8th periodic inspection
Measures for water quality improvement (measures for crud)	Installation of oxygen injection equipment for feed water system	Startup test
	Improvement of clean-up line of feed water system	1st periodic inspection
	Addition at washing piping for bearing of refueling machine	2nd periodic inspection
	Installation of heavy water cooling and heating equipment for heavy water purification system	5th periodic inspection
Measures for low radiation exposure	Replacement of supporting system of reactor outlet pipes with mechanical snubber	1st periodic inspection
	Structural change of drain sump pump of reactor building	5th periodic inspection
Others	Increase in capacity of spent fuel storage rack	1979 fiscal year
	Installation of high-range monitor for emergency	4th periodic inspection
	Construction of 2nd solid waste storage	1983-1984 fiscal year
	Increase in spent ion exchange resin storage tank and installation of solid waste incinerator	1986-1989 fiscal year
	Change of waste liquid filter from precoat type to hollow fiber filter	8th periodic inspection
	Replacement of dry cleaning machine for liquid waste treatment system with bigger one	8th periodic inspection

Table I.3-7 Outline of Incidents/Troubles

Date of occurrence	State	Incidents/Troubles	Outline	Cause
1980. 7.19	During power up	Reactor shutdown by trouble of 2B heater of reactor feed water system	During ~40% electric power operation, turbine tripped by false signal "high high water level" of the 2B feed water heater in feed water system, and reactor shutdown automatically.	Lower part of connecting tube for detecting switch of feed water heater water level (float type) was plugged by rust; and switch actuated at different level from actual level.
1980. 7.23	During power up	Turbine trip and reactor automatic shutdown by signal "high high steam drum water level"	During forming control rod pattern at ~40% electric power, power lowered and accordingly water level of the steam drum rose while maintaining power, and reactor automatically stopped by turbine trip.	Of 4L control rods under withdrawal, one went down because of one phase loss of motor power supply, so that power changed too great and feed water control could not follow, leading to rise in the water level of steam drum.
1980. 11.18	During scheduled shutdown	Anomaly of cooling system piping	SCC (stress corrosion cracking) were found near the welds of piping in remaining heat removal system by checking main piping under scheduled shutdown.	The stemmed from SCC of grain boundary of the austenitic stainless steel in high-temperature demineralized water.
1981.2.18	During scheduled shutdown	Trouble of fuel exchange Machine	During fuel withdrawal, anomaly took place in grab torque when grab of refueling machine was risen to withdraw the seal plug.	Movement of grab was hindered, because grad guide roller shaft of chain setting stand for grab up-and-down machine was projected and came in contact with the chain case.
1981.11.10	During operation	Reactor automatic shutdown by signal "high high neutron flux"	During power operation, local power monitor (LPM) made false working, and regional power monitor signaled "high high neutron flux" leading to automatic reactor shutdown.	Of 64 channels of LPM measuring neutrons, one channel failed to give signal "high high neutron flux (120%)", resulting in reactor automatic shutdown.
1981.11.22	During power up	Reactor manual scram by failure of reactor feed water system and flow control valve	During 35% electric power operation, main feed water flow control valve of the B loop malfunctioned and steam drum water level rose, resulting in manual reactor shutdown.	Misalliance between valve stem and valve driving rod led to galling between driving rod and rod guide bushing.
1982.2.8	During operation	Leak of reactor feed water system and moisture separation drain piping	During rated power operation, leak from drain piping of moisture separator was found by surveillance/check, so that reactor was shutdown manually.	Inside surface of pipe was eroded by high-temperature 2 phase flow, which caused small hole.
1982.11.10	During periodic inspection	Trouble of high-pressure water injection piping.	During in-service inspection of reactor coolant piping, abnormal echo was detected in B piping of high-pressure water injection piping on the steam drum side.	Impurities, i.e. chlorine, contained in grand packings of water injection valves were dissolved out, and locally caused corrosion of pipe.
1984.6.30	During power up	Unscheduled reactor shutdown by power up at the time of reactor recirculation pump speed change over.	After paralleling the generator, reactor automatically shutdown when recirculation pump speed was changed over from low to high at 40% electric power.	Reactor reciculation pump speed was changed over in the control pattern different from normal one for reactor core characteristic test, so that power great changed due to void fraction change in core, leading to automatic shut down because of high RPM and low low flow rate.

Date of occurrence	State	Incidents/Troubles	Outline	Cause
1984.12.16	During operation	Unscheduled reactor shutdown by signal of "high high heavy water temperature"	During rated power operation, cooling water decreased due to temporary trouble of heavy water temperature control valve, and heavy water temperature rose to "high high heavy water temperature" level, resulting in reactor automatic shutdown.	Foreign substance temporarily entered into driving zone of pneumatic operation device of outlet temperature control valve of heavy water cooler, and interrupted smooth motion of valve.
1985.11.9	During periodic inspection	Damage of fresh fuel assembly during its transportation	When fresh fuel assemblies (36 for irradiation) were transferred by the fuel transfer machine, they came into contact with transfer vessel lever in fuel delivery pool, thereby denting and bending occurred in the assembly.	These defects took place because of poor position adjustment of grippers of refueling machine.
1985.12.18	During power up	Unscheduled reactor shutdown during periodic inspection	Reactor was automatically shut down when reactor operation mode switch was changed from "startup" to "operation" at 8% power.	Signal "main steam stopping valve shut" could not be released, because contact failure took place in "low turbine 1st steam room pressure low" detecting relay which should be released when reactor output is low.
1986.1.13	During operation	Coolant leak from reactor coolant purification system piping	Since small amount of steam leaked out near piping weld in reactor coolant purification system in the main steam piping room located in 1st basement of turbine building, and reactor was manually shutdown.	Leak occurred due to intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel.
1986.2.13	During power up	Unscheduled reactor shutdown by "low low steam drum water level"	When water level of steam drum was manually adjusted during reactor power up, its water level lowered at ~15% power, and signal of "low B steam drum water level (NWR-230mm) made reactor automatically shut down.	Feed water flow rate was manually adjusted during reactor power up, operation of the low flow rate control valve was not enough to open, so necessary amount of feed water was not injected and reactor automatically shut down by signal "low low1 B steam drum water level".
1987.10.2	During scheduled shutdown	Trouble at refueling	During fuel loading, it was intended to insert the fuel manually into refueling, but seal plug and grab could not be separated. It was considered that seal plug guide head adhered to pressure tube near normal loading position. Moreover, grab of refueling machine could not be lowered below snout head.	Cause of seal plug adhesion was that crud entered into gap between wear ring of seal plug guide head and pressure tube, and guide head inclined upon detaching seal plug. As a result, sliding resistance between guide head and pressure tube increased, and wear ring adhered to the pressure tube.
1988.6.27	During power up	Unscheduled reactor shutdown by decrease of vacuum degree of condenser associated with failure of auxiliary steam pipe safety valve	During about 35% electric power (58MW) operation, generator output decreased to 52MW. Afterwards, vacuum degree of condenser began to decrease, power was manually reduced to shut down reactor. Decrease of degree of vacuum continued, and reactor was automatically shutdown by signal "low low 1 the degree of vacuum of condenser".	Generator output decreased by following reasons : Cracks in upper diaphragm of airlock valve for auxiliary steam control valve of air ejector and deformation of O-ring led to control air leakage, and made valve fully open; accordingly, steam pressure of the line rose to actuate safety valve to reduce generator output.

Date of occurrence	State	Incidents /Troubles	Outline	Cause
				As to decrease in degree of vacuum, lock screw of safety valve became loose and was lost when control valve actuated. After action of safety valve stopped, air was sucked into condenser through screw hole to reduce degree of vacuum.
1989.11.6	During operation	Unscheduled reactor shutdown by signal "main steam stop valve"	During rated power operation, main steam stop valve shut, and reactor automatically shut down by signal "main steam stop valve closure".	Intercept valve rapidly closed because of temporary poor contact of emergency oil pressure timer relay for turbine control oil pressure system. Upon its resetting instantaneous pressure drop occurred in emergency oil pressure system, and opening of 2 main steam stop valves became 90% or less.
1990.10.21	During operation	Unscheduled reactor shutdown by signal "high high steam drum water level"	During rated power operation, alarm of "abnormal feed water control system" sounded, and decrease in steam drum water level was confirmed, so reactor power was decreased and tried to stabilize steam drum water level. However, turbine automatically stopped by signal "high high B steam drum water level".	Feed water flow rate changed with decrease in control air pressure, and steam drum water level fluctuated. This caused automatic reactor shutdown. Decrease in control air pressure was due to incomplete periodic test manual.
1992.11.2	During scheduled shutdown	Trouble of refueling machine	During refueling, fuel assembly was taken out of core, placed onto refueling machine, and seal plug was attached to pressure tube. While the grab was hung down, alarm "grab torque is abnormal" sounded, and grab stopped.	Sticking of triple chain tightener shafts for upper driving system could not absorb sagging of chains, leading to their sagging. Therefore, engaging chains with sprocket became poor, making driving impossible.
1993.10.18	During operation	Reactor manual shutdown by steam leak from high pressure turbine outlet piping	During rated power operation, reactor was manually shut down, because steam was slightly leaked out near pressure detection piping of high-pressure turbine outlet piping below the ceiling of 1st floor of turbine building. Leaking place was plug for non-destructive inspection as result of on-the-spot inspection	The cause of steam leakage is considered to be erosion/corrosion due to material of 1 plug and moisture content of steam.
1993.9.30	During periodic inspection	Injury of operator	Rust was found on piping of "emergency gas treatment system" installed in degraded heavy water storage room located on 1st basement of reactor auxiliary building. When an operator went up near end plate of A degraded heavy water storage tank to check rust in detail, he missed his footing and fall down about 3 m and was wounded.	A degraded heavy water storage tank is about 2.2 m diameter stainless steel cylinder. Although its upper part seemed to be slippery, contents of his work was very simple, which made him not careful to keep safety.
1993.11.25	During operation	Unscheduled reactor shutdown by signal "high high steam drum pressure "	During test run after periodic inspection (rated power operation) reactor automatically shutdown by signal "high high steam drum pressure".	Lead wire crimp junction of amplifier of the maximum flow rate control circuit in EHC control circuit was not tightly fastened on manufacture, thereby having poor contact due to increase in oxide film.

Date of occurrence	State	Incidents/Troubles	Outline	Cause
1996.12.24	During operation	Unscheduled reactor shutdown by signal "high high steam drum pressure"	During rated power operation, reactor automatically shutdown by signal "high high steam drum pressure".	Of turbine control system, capacitor element of differnciator giving main speed differencial signal of speed control circuit underwent breakdown and was internally short-circuited, leading to false signal.

Table I.3-8 Results of Water Chemistry

	Item	Unit		Result (Average)									
				1979	1980	1981	1982	1983	1984	1985	1986	1987	1988
Coolant	pH	–	5.5～6.5	6.7	5.6	6.5	5.7	6.5	6.3	7.2	7.2	7.3	7.2
	Conductivity	μS/cm	≦1	0.29	0.37	0.30	0.22	0.26	0.38	0.10	0.14	0.16	0.18
	Chrorine ion	ppm	≦0.2	0.002	0.002	0.001	0.001	0.001	<0.001	<0.001	<0.001	<0.001	<0.001
	SiO_2	ppm	≦2	0.14	0.025	0.014	0.023	0.051	0.009	0.026	0.042	0.022	0.011
	Dissolved Oxygen	ppm	≦0.4	0.12	0.14	0.19	0.17	0.13	0.096	0.035	0.026	0.024	0.025
Moderator	pD	–	4.5～6.5	5.4	5.3	6.0	5.5	5.7	6.1	5.6	7.0	6.9	7.0
	Conductivity	μS/cm	≦5.0	3.2	3.7	3.3	3.8	2.9	2.4	2.3	0.46	0.46	0.43
	Chrorine ion	ppm	≦1	0.067	0.075	0.076	0.098	0.050	0.14	0.12	0.006	0.006	0.005
	SiO_2	ppm	≦1	0.081	0.040	0.042	0.035	0.04	0.076	0.091	0.013	0.018	0.035
	Insoluble impurity	ppm	≦0.5	0.16	<0.1	0.16	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

	Item	Unit		Result (Average)							
				1989	1990	1991	1992	1993	1994	1995	1996
Coolant	pH	–	5.5～6.5	7.1	7.1	7.1	7.1	7.0	7.0	7.0	7.2
	Conductivity	μS/cm	≦1	0.15	0.17	0.10	0.087	0.078	0.083	0.080	0.075
	Chrorine ion	ppm	≦0.2	<0.001	<0.001	<0.001	<0.001	<0.001	<0.001	<0.001	<0.001
	SiO_2	ppm	≦2	0.016	0.027	0.020	0.017	0.026	0.028	0.014	0.009
	Dissolved Oxygen	ppm	≦0.4	0.019	0.011	0.011	0.010	0.009	0.006	0.005	0.001
Moderator	pD	–	4.5～6.5	6.8	6.6	6.4	6.2	6.1	6.3	6.5	6.5
	Conductivity	μS/cm	≦5.0	0.43	0.52	0.59	0.69	0.74	0.70	0.50	0.49
	Chrorine ion	ppm	≦1	0.005	0.005	0.004	0.005	0.006	0.006	0.005	0.004
	SiO_2	ppm	≦1	0.048	0.044	0.032	0.005	0.008	0.006	0.030	0.022
	Insoluble impurity	ppm	≦0.5	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Table I.3-9 Results of Discharged Readioactive Wastes

(Unit:Bq)

Fisical Year	Radioactive Wasts Exept Tritium	Tritium
1978	3.3×10^{7}	2.6×10^{11}
1979	5.3×10^{7}	2.7×10^{11}
1980	3.7×10^{7}	7.7×10^{11}
1981	2.9×10^{7}	8.5×10^{11}
1982	3.1×10^{7}	1.2×10^{12}
1983	4.8×10^{7}	1.3×10^{12}
1984	1.9×10^{7}	2.6×10^{12}
1985	1.0×10^{7}	3.6×10^{12}
1986	4.8×10^{7}	2.2×10^{12}
1987	1.9×10^{7}	1.9×10^{12}
1988	4.8×10^{7}	4.4×10^{12}
1989	5.8×10^{7}	7.0×10^{12}
1990	1.4×10^{7}	3.3×10^{12}
1991	4.7×10^{7}	1.8×10^{12}
1992	1.1×10^{7}	3.9×10^{12}
1993	1.6×10^{6}	3.5×10^{12}
1994	ND	4.7×10^{12}
1995	ND	4.1×10^{12}
1996	ND	5.9×10^{12}
Control Goal of Released Radioactive Wastes	7.4×10^{9}	1.1×10^{13}

4. Safety Control System and Safety Regulations

4.1 Bases of Safety Control

The rules of safety control of "Fugen" are basically to ensure the safety of the reactor facility, to control the release of radioactive materials below the target values, and thereby reducing the effect on the environment as low as practicable. Therefore, the safety control system has been established.

To reduce the radiation exposure of operators as low as possible, necessary measures have been taken.

For safety control, the safety regulations for the reactor facility are established to control the safety control system, operation of the reactor facility, maintenance, radioactive wastes, and radiation. Operation control, maintenance control and radiation control are also carried out by determining the basic items for ensuring safety regarding the emergency measures.

"Fugen" is not only a reactor facility, but also equipped with power generating installations such as turbines and generators; accordingly, the safety regulations for electric equipment was enacted to ensure safety, and the basic items regarding works, maintenance and operations of the electric equipment were established to control operation and maintenance.

The owner of the reactor must nominate the Licensed Reactor Engineer to supervise the safety of reactor facility operation. He gives instructions to the operators, offers his opinion to the plant Director, any gives advices to the related Section Chiefs, if necessary.

The owner also nominate the Licensed Engineer for Boiler/Turbine and Licensed Electrical Engineer. They give instructions and advices to the Safety Managers, if necessary.

For safety control of the facility, various manuals and standards must be prepared, such as operating procedures, maintenance control manuals and emergency manuals, and safety control should be performed in accordance with them.

As to the use of radioisotopes and handling of wastes and radioisotope-contaminated materials, the regulations for preventing radiation hazard from occurring must be established for safety control, and the Licensed Engineer for Radiation Safety must be nominated to supervise the prevention of radiation hazards.

4.2 Safety Management Activity

(1) Safety Management Organization

Positions and duties of the Director, Safety Control General Manager, Technical Section General Manager, Power Generating Section General Manager and Maintenance Section General Manager relating to the safety of

reactor facility are stipulated.

Table I.4-1 shows the organization of safety control.

(2) Operation Management

Operation is carried out for 24 hr continuously on 6-group 3-shift basis. The main shift jobs are operation, plant operation monitoring, surveillance/check and periodic tests.

Operators operate the plant in accordance with the operating limit values and operating standards written in the operation manuals stipulating the plant startup/shutdown procedures, etc. To monitor the operations of the facilities and equipment, moreover, surveillance and check are carried out periodically.

As to the engineered safety features important for safety, surveillance tests are periodically performed to confirm the integrity of their operating functions.

When an operator finds anomaly of the facility during operation, he informs the shift leader of it, and the leader confirms the anomaly for report, and takes measures to remove the cause and to mitigate the anomaly.

(3) Core Management

The refueling plan is determined on an operating cycle basis taking into consideration of the muclear-thermal-hydraulic design criteria, shutdown margin of the reactor and burnup of the fuels to be withdrawn.

The fuels are exchanged at the time of the periodic inspection and scheduled shutdown in accordance with the refueling plan. After the refueling is completed, maximum excess reactivity and shutdown margin of reactivity are evaluated at the time of reactor startup to confirm safety. During operation, moreover, the state of the core are periodically evaluated and monitored on the basis of measured neutron flux and other operating data of the reactor.

(4) Maintenance Management

Maintenance of installations and equipment is carried out in accordance with the check plans such as everyday and/or periodic check and the modification/exchange plans of the installations and equipment.

In the daily checks, the operators and maintenance personnel carry out surveillance and check to find out troubles as early as possible and to take preventive measures.

In the periodic inspection, the periodic inspection for the reactor facility and that for the power generating installations are carried out annually. As regards the periodic inspection, the long-term periodic inspection plan is first prepared, followed by establishment of the periodic inspection plan, taking into account the results of everyday check and periodic inspection and

troubles occurring in the other plants.

At the time of the periodic inspection, moreover, the measures for the aged deterioration of installations and equipment are taken and the modification works for improving their functions are performed.

(5) Radiological Safety Management

Radiological safety management is carried out on the basis of the ICRP's recommendation that ① radiation exposure dose equivalent shall not exceed the limit values and that ② radiation exposure dose equivalent shall be as low as possible.

Because "Fugen" uses heavy water as the moderator, the emphasis in radiological safety management is placed on exposure by tritium; accordingly, for checking installations and equipment handling heavy water, exposure reduction measures must be taken by installing local ventilation devices and by wearing protective clothing. To confirm the state of exposure, tritium exposure is evaluated by bioassay.

4.3 Safety Regulations and Others

For safety control of reactor facilities the safety regulations for them and those for electric installations as the power generating facilities for safety control are stipulated.

In addition, the Radiation Injury Prevention Act is stipulated to prevent radiation injury from occurring.

The approval of the safety regulations of reactor facilities must be applied to the Minister of STA on the basis of the Law Regulating Nuclear Reactors etc. As to the safety regulations of electric installations must be applied to the Minister of MITI for approval based on the Electric Enterprise Act, whereas the Radiation Injury Prevention Regulation must be notified to the Minister of STA based on the Law for Preventing Radiation Injury by Radioactive Materials.

(1) Safety Regulations of Reactor Facilities

The safety regulations of reactor facilities stipulate the basic matter relating to the safety of reactor facilities to prevent disasters due to materials contaminated with nuclear raw materials and/or nuclear materials and due to nuclear reactors.

The safety regulations stipulate the following necessary for safety control :

(a) Organization and Duty

① Organization and Duty Regarding Safety

② Nomination and Duty of Licensed Reactor Engineer

(b) Operation Management

① Matters to Be Attended for Operation

② Limits in Operation

③ Operational Conditions

④ Emergency Procedures

(c) Fuel Management

① Transfer and Storage of Fresh Fuels

② Fuel Exchange

③ Storage of Irradiated Fuels

(d) Radwaste Management

① Management of Radioactive Solids, Liquids and Gases

② Measures When Release Target Values Are Exceeded

(e) Radiation Control

① Area Control

② Exposure Control

③ Measurement of Radiation Dose Equivalent Rate, etc.

(f) Maintenance Control

① Periodic Tests and Inspection

② Repair and Modification

(g) Emergency Measures

① Preventive Measures

② Initial Activity

③ Emergency Activity

(h) Education and Training

(i) Record and Report

① Attachment of Safety Control Organization Chart (Fig.I.4-1)

(2) Safety Regulation of Electric Installations

The safety regulations of electric devices stipulate basic matters regarding works, maintenance and operations of private electric devices to ensure their safety.

The safety regulations stipulate the following matters necessary for safety control.

(a) Safety Control System

① Basic Duties and Safety Organization

② Nomination of Licensed Engineers and Their Duties

(b) Safety Education

① Contents and Method of Education

(c) Surveillance, Check and Inspection of Private Electric Devices

① Execution of Surveillance, Check and Inspection

② Procedures for Results of Surveillance, Check and Inspection

(d) Operation of Private Electric Facility

① Basic Operation

② Procedures for Accidents and at Emergencies

③ Procedures at Hazards and Other Emergencies

(e) Record

① Safety Control Organization Chart to Be Attached (Fig.I4-1)

(3) Radiation Hazard Prevention Regulation

The radiation hazard prevention regulation stipulates the basic matters regarding the use of radioactive isotopes in the facility and the handling of wastes and materials contaminated with radioactive isotopes for preventing radiation injury from occurring.

(a) Safety Control System

① Organization and Duties

② Licensed Engineer Radiation Safety

(b) Maintenance of Radiation Facilities

(c) Use of Radioactive Isotopes

(d) Radiological Safety Management

① Controlled Area

② Exposure Control

③ Measurement of Dose Equivalent Rate, etc.

④ Waste Management

(e) Procedures for Accidents and Dangers

(f) Report, Education, Training, etc.

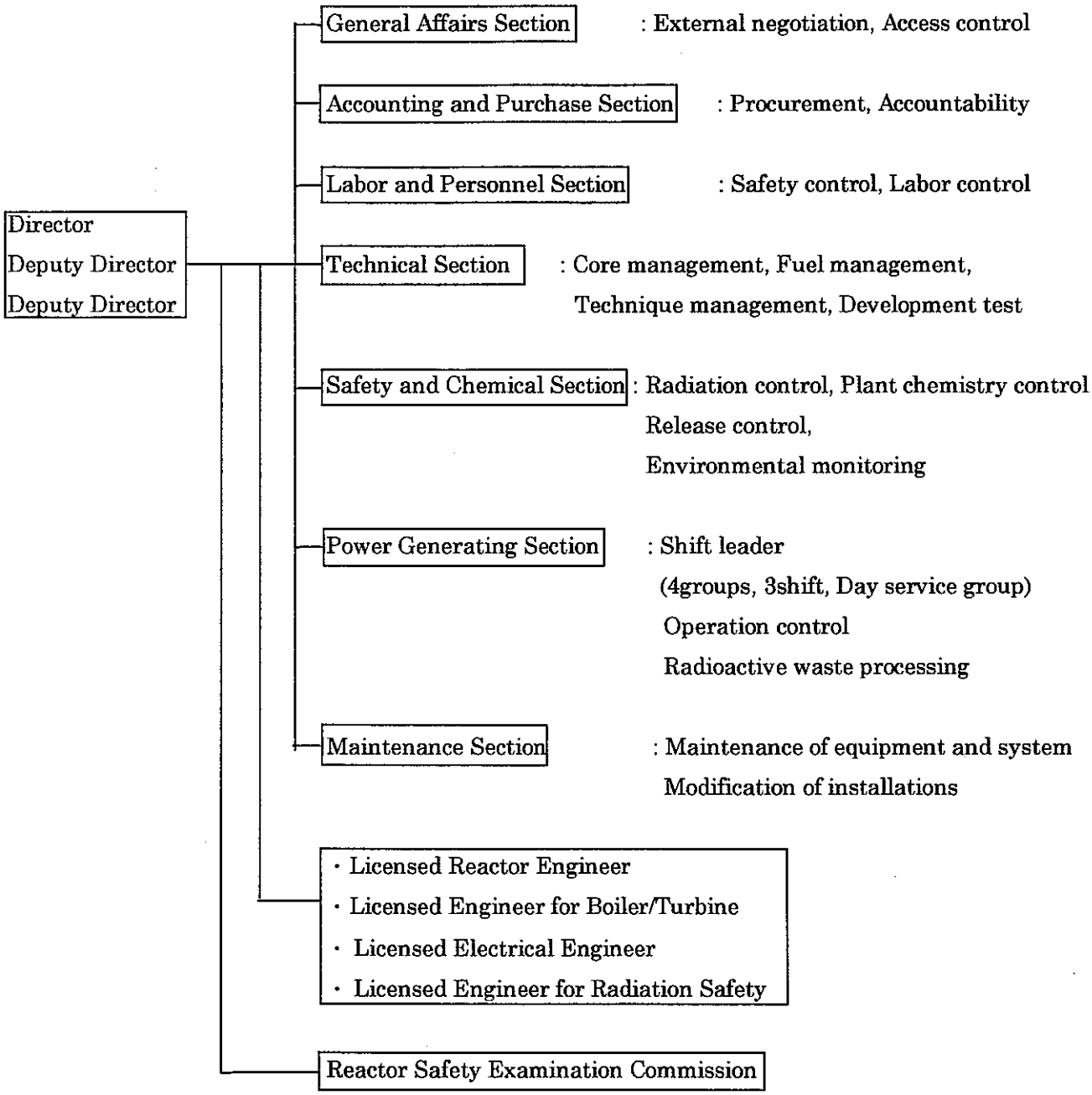


Fig.I.4-1 Safety Management Organization of "Fugen"

5. Quality Assurance [I-33]

5.1 Necessity of Quality Assurance

In order to attain the specified quality and ensure safety as a power generating plant in each stage from design to operation, the matters to be implemented in each stage are planned in advance, executed and confirmed. These plans are improved, as required, to maintain quality through systematic activity, i.e. quality assurance (hereinafter referred to as "QA") activity.

For this purpose, the design and construction works of "Fugen" basically followed the laws and standards in force at that time. To conduct PNC's peculiar QA activity, the following standards (ATR Standards) were stipulated for quality assurance. These ATR standards were partly revised, and are used at present.

① Criteria for Aseismatic Support Design of Piping Systems
② Execution Management Manual for Piping Supports
③ Aseismatic Supporting Policy for Ducts
④ Quality Assurance Standards for Piping and Hardware Buried in Concrete Structures
⑤ Execution Management Manual for Buried Hardware
⑥ Execution Management Manual for Concrete Anchor
⑦ Quality Assurance Standards for Materials and Welds
⑧ Shield Design Standards for ATR
⑨ General Specifications for Distribution Panels of "Fugen"
⑩ General Specifications for Control Center
⑪ Standard Specifications for Electric Motors
⑫ Specifications for Separating Electric Equipment of Safety Systems
⑬ Standards for Logical Circuit Structures
⑭ Installations Standards for Detectors of "Fugen"
⑮ Design Standards for Process Instrumentation Piping
⑯ Standards for Multi-Core Cables for Instrumentation Control
⑰ Manufacturing Specifications for Cable Train

The QA activity follows the PNC's "Quality Assurance Regulation for Reactor Facilities" and the "Quality Assurance Plan for Fugen" stipulated on December 10, 1981 with reference to the Japan Electric Associations Electric Technology Guides JEAG 4101-1981 "Quality Assurance Guides for Nuclear Power Generating Stations" (revised to JEAG 4101-1993).

5.2 Basic Policies and Results of QA Activity

The basic policy of QA activity of "Fugen" is to perform actions necessary for

giving enough reliability systematically, so that each system and equipment may fulfill their functions.

(1) Organization

The organization necessary for the QA actions for "Fugen" is based on the "Organization Regulations" of PNC, and further on the committees necessary for quality assurance, for prevention of disasters and radiation injuries caused by reactor operation and incidents.

(a) Quality Assurance Committee

This Committee promotes the QA activity of "Fugen" and examines and investigates important matters regarding QA activity.

(b) Reactor Safety Examination Committee

The Reactor Safety Examination Committee examines matters important for reactor facilities and those important for prevention of radiation injuries.

(c) "Fugen" Operation Management Meeting

The "Fugen" Operation Management Meeting (hereinafter referred to as "Operation Management Meeting") investigates and adjusts the basic matters relating to the operation and development of "Fugen".

(d) Human Error Prevention Committee in "Fugen"

The Human Error Prevention Committee in "Fugen" (hereinafter referred to as "Committee") investigates and examines the measures for preventing the recurrence accidents due to human errors in "Fugen".

(2) Document Control [I-34]

For doucument control, it is necessary to clarify the division in charge of preparation, examination, approval, distribution, revision and deposit important to quality assurance (quality assurance plan, quality record, specification, instructions and drawings) and the control method; accordingly, documents associated with "Fugen" are classified into general documents, design/manufacture documents and operation/maintenance documents, and these are further classified into internal, approved and submitted documents.

(a) Preparation, Distribution and Revision of Internal Documents

The internal documents (prepared by PNC) regarding QA activity are prepared in accordance with the Technical Document Control Regulation (No.3 Regulation enacted in 1982), approved by the General Manager of the Section concerned (hereinafter referred to as "General Manager in change"), and then distributed to the related sections.

The documents submitted to the government such as the application for approval are confirmed by the Power Reactor Construction and Operation Project, and approved by the Station Director before submission.

(b) Handling of Document for Approval and for Submission

As to documents for approval submitted from the order-received companies, they must be classified into the documents relating to the reactor facilities and those relating to the fuels.

(3) Design Management [I-35]

Upon designing the systems and the equipment, requirements stipulated in laws, regulations and standards must be reflected on the design, in addition to the those written in the specifications. The General Managers in change, therefore, should understand and decide the requirements for quality assurance shown in the following in accordance with the grade of the components in addition to the above. After obtaining confirmation of the Director, the General Manager in charge should clearly show the above-mentioned matters to the designers to reflect them on the design.

① Safety Analyses (transient analysis, reactor physics analysis, dynamic flow analysis, stress analysis, accident analysis and evaluation of adequacy of materials)

② Evaluation of Operational Reliability

③ Evaluation of In-Service Inspection, and Easiness of Operation/ Maintenance and Repair

④ Evaluation of Radiation Protection and Exposure

⑤ Establishment of Testing and Inspection Criteria

⑥ Confirmation of Reflection of Results of Tests and Research

(4) Procurement Management [I-36]

Upon the procurement of equipment, specified-quality products and services must be supplied; consequently, the General Manager in charge makes clear the following necessary for quality assurance of goods and services such as the general requirements, technical requirements and basic design conditions.

① Scope of Work to Be Done by Order Receiver

② Conditions for Permission and Approval, Design Conditions and Law and Regulations to Be Applied, and Standards and Manuals

③ Matters Relating to Tests and Inspections

④ Access to Order Receivers for Inspection and Observation

⑤ Documents such as Specifications, Manuals, Drawings and Quality Records to Be submitted to Orderer, and Method and Time of Submission

⑥ Matters Relating to Taking Custody and Disposition

⑦ Matters Relating to Report and Approval of Measures Taken for Disposition of Inadequate Products or Services

⑧ Matters Relating to the Submission of Quality Assurance Plan

⑨ Evaluation to Make Requirements Written in Procurement Document Effective to Subcontractor of Order-Receiver

(5) Materials and Equipment Control

Because the confirmation of specifications and number of items of materials and equipment (including those supplied by PNC) should be properly performed, the General Manager in charge makes the order receivers of the materials and equipment important for safety to submit the manuals describing the methods of control, handling, storage and shipping for examination, and manages them based on following ways of thinking:

① As to the specifications of materials and equipment, it is necessary to confirm that the specifications, materials, number of items and performance are acceptance-inspected by measurement and/or records.

② Parts accepted by the inspection must be registered in the "spare/parts ledger" by the computer. Machine parts are stored in the shelves in the warehouse, whereas electric parts in the instrument repair room in the main building equipped with an air conditioning system.

③ Upon shipping parts, documents are prepared by the computer to manage their numbers correctly.

④ When small parts for replacement are delivered to the field, use bags with seal to avoid troubles such as inclusion of foreign substances and/or loss during transport and open the seal in the field.

(6) Management During Manufacture and Installation [I-37] [I-38]

For manufacturing and installation, manuals, procedures and instructions must be prepared in advance, and the actions must be done in accordance with them. The General Managers in charge make the order receivers submit manufacture/execution manuals describing the execution methods, working procedures, working environment, installations, tools, special process, operator's technique, and approve them to confirm the adequacy of the manufacture/execution manuals.

(7) Management of Inspection and Testing [I-39]

The inspections and testing carried out for "Fugen" are classified into those performed by PNC (hereinafter referred to as "Internal Inspection") and those performed by the government authorities (hereinafter referred to as "Government Inspection").

As regards the inspection and testing of equipment, etc., the plans and manuals for them regarding the scope, items, attendance and record submission are prepared to perform the inspection and testing in accordance with them.

(8) Management of Operation and Maintenance [I-40]

The works necessary for ensuring safety, such as operation management, fuel management, radwaste management, radiation control system, safety education and training, emergency system and measures, safety-related records and report, and should be clearly defined and smoothly carried out with them.

(9) Measures for Preventing Recurrence

In "Fugen", in oder to improve reliability, troubles occurring in plants in Japan and abroad have been reflected on its design and operation, plant anomalies have been found by LTT (Long Term Trend) as early as possible, and the reliability of installations/equipment have been evaluated; these activities are shown as follows:

① Reflection of Troubles Occurring in Japan and Abroad

The troubles occurring in Japan (including "Fugen") and abroad are fully investigated as to their facts, causes and measures taken, and the results are reflected on "Fugen".

② Early Detection of Plant Anomalies by Surreillance LTT

In "Fugen", the state of the plan has been always observed by the computers, and a technique to monitor and evaluate various plant process parameters from long-term trend point of view has been investigated and implemented to detect plant anomalies as fast as possible.

③ Evaluation of Reliability of Installation/Equipment by FTA

For this purpose, the importance of each installation/equipment for plant shutdown probability have been classified by the usual Fault Tree Analysis (FTA) as reliability analysis since the fiscal 1985 by using the "Fugen"'s operation/maintenance experience and maintenance data, as well as LWRs' and fossil fuel power generating plants' data.

5.3 Quality Assurance Audit [I-41]

In "Fugen", quality assurance independent audit is carried out once a year to verify the state of implementation of the quality assurance plan and its effectiveness.

Audit is performed on the basis of the Quality Assurance Audit Manual (established on June 25, 1987) for education, training, document management, operation management, maintenance management, procurement management, measures for preventing recurrence and record management, all of which are very important in operating "Fugen".

6. Nuclear Regulation System

Nuclear facilities have been regulated on the basis of the "Laws for Regulating Nuclear Raw Materials, Nuclear Materials and Nuclear Reactors" (Laws for Regulating Nuclear Reactors etc.)

To construct nuclear facilities, their owners must apply the construction permit and design & construction method approval, and then must undergo inspection before use and that of welds. For operating the reactor facilities, moreover, the safety regulations must be approved, the Licensed Reactor Engineer must be nominated and the nuclear safeguards regulation must be approved. After starting operation, the performance of the nuclear facilities must be subjected to the periodic inspection once a year.

On the other hand, as to the facilities generating power by nuclear reactors, construction and construction plans of electric equipment must be approved and undergo inspection before use.

Since "Fugen" is a prototype reactor, it is classified under the developing reactor, and follows the regulation stipulated for this category. "Fugen" is a power generating facility, but is not a commercial power reactor, so that it must follow the Electric Enterprise Act as a private electric facility.

6.1 Regulation by Laws for Regulating Nuclear Reactors, etc.

(1) Application of Nuclear Reactor Construction Permit

Those who want to apply the construction permit of a nuclear facility must submit the application of the construction permit of the nuclear reactor describing the following items to the Minister in charge (Minister of STA).

① Name, Address and Representative of the Juridical Person
② Objectives of Use
③ Type, Thermal Output and Number of Nuclear Reactors
④ Name and Address of Works in Which Nuclear Reactor Is Constructed
⑤ Site, Structure and Installations of Reactor and Its Auxiliary Facilities
⑥ Construction Plan of Nuclear Reactor Facility
⑦ Type and Annual Consumption of Nuclear Materials Used in Reactor
⑧ Method of Disposal of Spent Fuels

Safety examination of nuclear reactor construction permit is first performed in the government in charge (primary examination). The results are submitted to the Nuclear Safety Commission and Atomic Energy Commission for further safety examination (secondary examination).

Upon approving the construction of a nuclear reactor, the following must be met:

① There should be no fear of being used for the purposes other than peaceful

use.

② There should be no fear of giving troubles for planned implementation of development and utilization of nuclear energy.

③ There should be enough technical capability and financial base necessary for constructing the nuclear reactor.

④ The site, structure and facility of the nuclear reactor should be satisfactory for preventing disasters due to nuclear materials and substances contaminated with nuclear materials or the reactor (safety examination).

In the safety examination of a nuclear reactor, natural and social conditions, engineered safety of it, exposure of operators at normal operation, various possible accidents of the reactor and disasters are evaluated on the basis of the safety examination criteria stipulated by the Nuclear Reactor Safety Commission.

(2) Approval of Design and Construction Methods

The founder of the reactor performs the detailed design in accordance with the application of construction approval of the nuclear reactor, and applies the approval of its design and construction method to the Minister in charge. The founder must undergo examination regarding whether or not the design and method of construction meet the technical criteria.

(3) Pre-Service Inspection and Testing

Based on the approval on the design and construction method, the founder must undergo the government inspection regarding whether or not the design and construction method are adequate and whether or not the strength, structure and performance of structures meet the technical criteria.

(4) Welding and Inspection

Those who will to do welding in nuclear facilities must apply approval on welding method, and undergo government inspection on whether the welding meets the technical criteria.

(5) Application of Approval of Safety Regulations

Prior to operation of nuclear power facilities, application of approval of the safety regulations of the reactor facilities stipulating the organization, operation and maintenance management and radiological safety management relating to the safety of the reactor facilities should be done.

(6) Nomination of Licensed Reactor Engineer

To supervise the safety of reactor operation, the Licensed Reactor Engineer must be nominated among those who have the Licensed Reactor Engineer certificates and notifies to the Minister in charge.

(7) Notification of Operation Plan

The reactor founder must prepare the operation plan of the reactor, and notifies it to the Minister in charge.

(8) Periodic Inspection

The reactor founder is subjected to periodic government inspection (periodic inspection) once a year regarding the performance of the reactor facility to confirm whether or not the performance meets the technical criteria.

(9) Establishment of Physical Protection Regulation and Nomination of Physical Protection Superintendent

The reactor founder must establish the physical protection regulation and obtain its approval from the Minister in charge before the start of handling special nuclear fuel materials.

To systematically manage the business regarding the physical protection of special nuclear fuel materials, moreover, the Physical Protection Superintendent must be nominated and notified to the Minister in charge.

6.2 Regulation by Electric Enterprise Act

(1) Application of Construction Plan Approval

As the construction plan approval of a private electric equipment, approval of the construction methods of the reactor proper, reactor cooling system, reactor containment facility, steam turbine and electric installation as nuclear installations is applied to the Minister of MITI (Ministry of International Trade and Industry).

The construction methods must be examined from the viewpoint of whether or not they meet the technical criteria.

(2) Application of Fuel Design Approval

Fuel design approval must be applied to the Minister of MITI.

The design approval of the fuel must be examined from the viewpoint of whether or not the design meets the technical criteria.

(3) Pre-operational Inspection and Testing

Based on the approved construction method, whether or not the construction method the founder adopted was adequate undergoes government inspection. Moreover, government inspections are carried out on whether or not the construction methods are pertinent and on whether or not structural strength, structures and performance meet the technical criteria when tests on structures, strength or leak can be done, when the fuels can be loaded into the core, when the reactor reaches criticality and when all the construction plan is completed.

(4) Inspection of Fuels

Government inspection is to be carried out for the fuels from the viewpoint of

whether or not the design of the fuels meets the approved one and meets the technical criteria when the tests on the structure, strength and leak can be done and when fabrication of fuel elements and fuel assemblies are completed.

(5) Welding Inspection

The equipment and piping for which welding is performed undergo the government inspection from the viewpoints of whether or not welding follows the approved welding method and meets the technical criteria when nondestructive test can be applied, when mechanical tests can be applied and when pressure tests can be performed.

(6) Safety Regulations for Electric Equipment

The safety regulations for electric equipment stipulates the basic matters regarding the works, maintenance and operation of private electric equipment to ensure the safety. The safety regulation for them must be stipulated and apply its approval to the Minister of MITI.

(7) Periodic Inspection

Of those who have the certificates, Licensed Engineer for Boiler/Turbine and Licensed Electrical Engineer must be nominated and notify the Minister of MITI of them.

"Fugen" is classified as "a reactor in the R&D stage" by the Law for Regulating Nuclear Reactors, and the matters relating to installation and operation are regulated by the "Test Reactor Installation and Operation Act" (hereinafter referred to as "Test Reactor Act").

On the other hand, LWRs are classified as "commercial power generating reactors", and the matters relating to installation and operation are regulated by the "Act for Installation and Operation of Commercial Power Generating Reactors", but "Approval of Design and Construction Work Method, Pre-Operational and Periodic Inspection" of LWRs are so stipulated as to follow the related provisions in the Electric Enterprise Act to avoid double regulation by the "Law for Regulating Reactors etc." and "Electric Enterprise Act".

Meanwhile, "Fugen" was not installed mainly to provide electricity, but its each installation is classified under "private electric equipment" according to the definition of the Electric Enterprise Act; accordingly, the installations have been regulated by this Act.

Of the matters relating to the installation and operation of nuclear reactors, therefore, those relating to the modification and repair of equipment (application of construction plan approval, notification of construction plan and pre-operational inspection), periodic inspection and report are regulated

by both the Law for Regulating Reactors and Electric Enterprise Act.
To make the operation and maintenance of "Fugen" smooth, the Minister in charge of "Fugen", i.e. the Minister of STA, and the Minister of MITI exchanged the memorandum on the concrete working method to avoid duplication of regulation.

6.3 Other Regulations

(1) Radiation Injury Prevention Act

The purpose of this Act is to prevent radiation injury from occurring, and to ensure safety of the public by regulating, on the basis of the Nuclear Energy Basic Law, the use, sales and disposal of radioactive isotopes, use of radiation-generating equipment and handling and disposal of materials contaminated with radioactive isotopes.
Because "Fugen" is provided with the heavy water upgraders, hot chemistry laboratory, radiation source storage, high radiation dose calibration room and low radiation dose calibration room, the usage permit of related devices (Article 3 of Law), notification of starting use (Article 3-2, Law) and nomination of Licensed Reactor Engineer (Article 34-2, Law) are necessary in addition to the regulation by the transportation of radioactive isotopes or materials contaminated with radioactive isotopes (Article 18-2, Law) and their disposal (Article 19-2, Law).

(2) Labor Safety and Hygiene Law

The Labor Safety and Hygiene Law, with the Labor Standards Act, is established to ensure the safety and health and to form comfortable working environment by promoting the preparation of comprehensive measures for the "Injury Prevention Standards", clarification of a responsible system and by taking measures for enhancing independent activity.
In "Fugen", there are maintenance works for Class-1 and Class-2 pressure vessels of the office building boiler, the heavy water upgraders and hydrogen-oxygen generation system, as wel as works on crane, hoist and elevator in addition to the handling of special chemical substances and organic solvents and oxygen-deficient operation; consequently, these operations must follow the related provisions of this law.

(3) High Pressure Gas Control Law

To prevent disasters by high-pressure gases, this law regulates production, sales, storage and transfer of high-pressure gases. Other handling, consumption of high-pressure gases together with the production and handling of the vessels are also regulated. This law also promotes activity regarding high-pressure gas safety established by the High-Pressure Gas

Safety Association to ensure safety of the public.

Of the equipment installed in "Fugen", necessary procedures are taken for following systems under this law.

(a) Refrigeration Safety Regulation

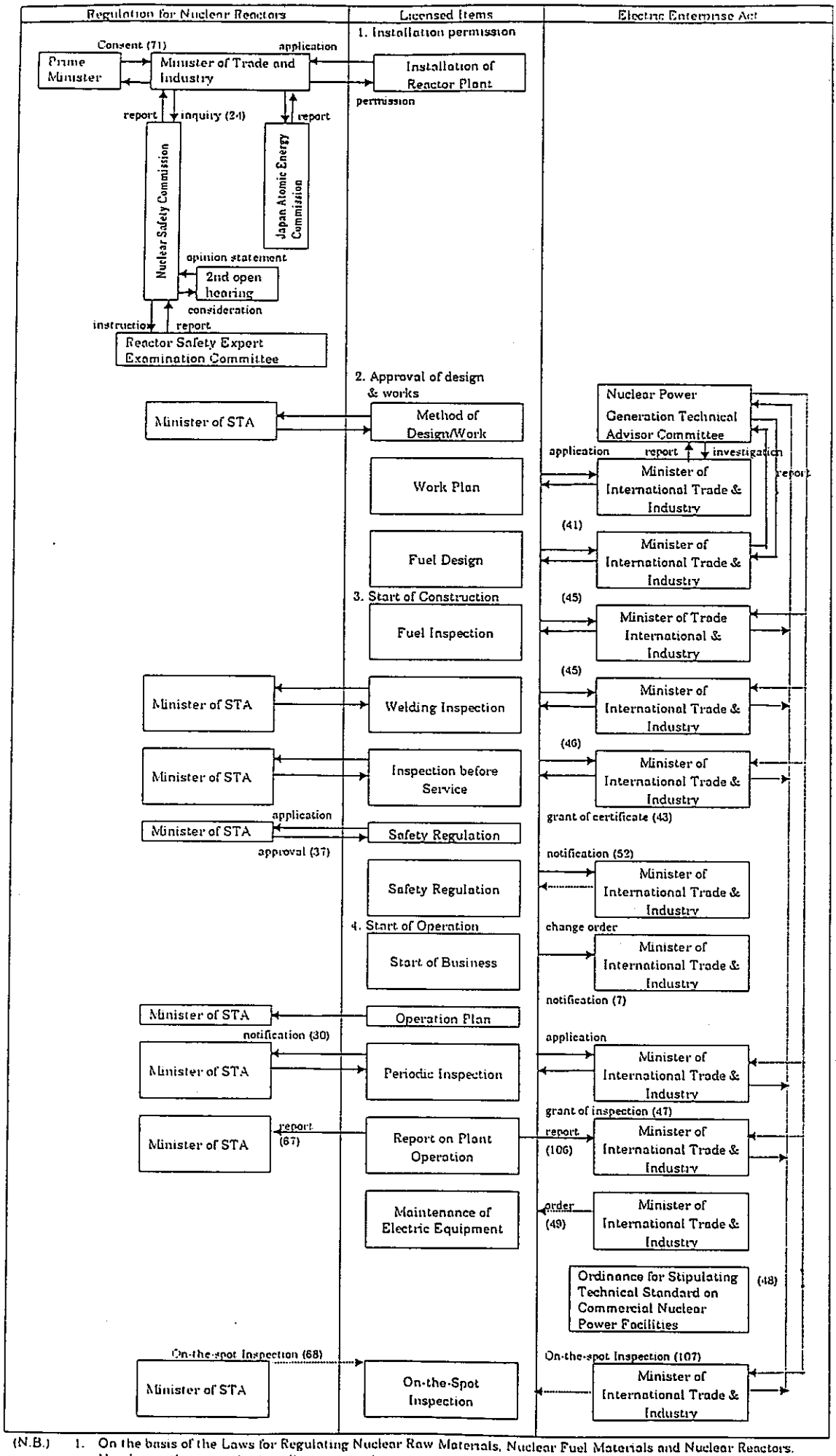
① Turbo-Refrigerator for Containment Air Recirculation System 2units

(b) General High-Pressure Gas Safety Regulation

① Hydrogen-Oxygen Generating System 1 unit

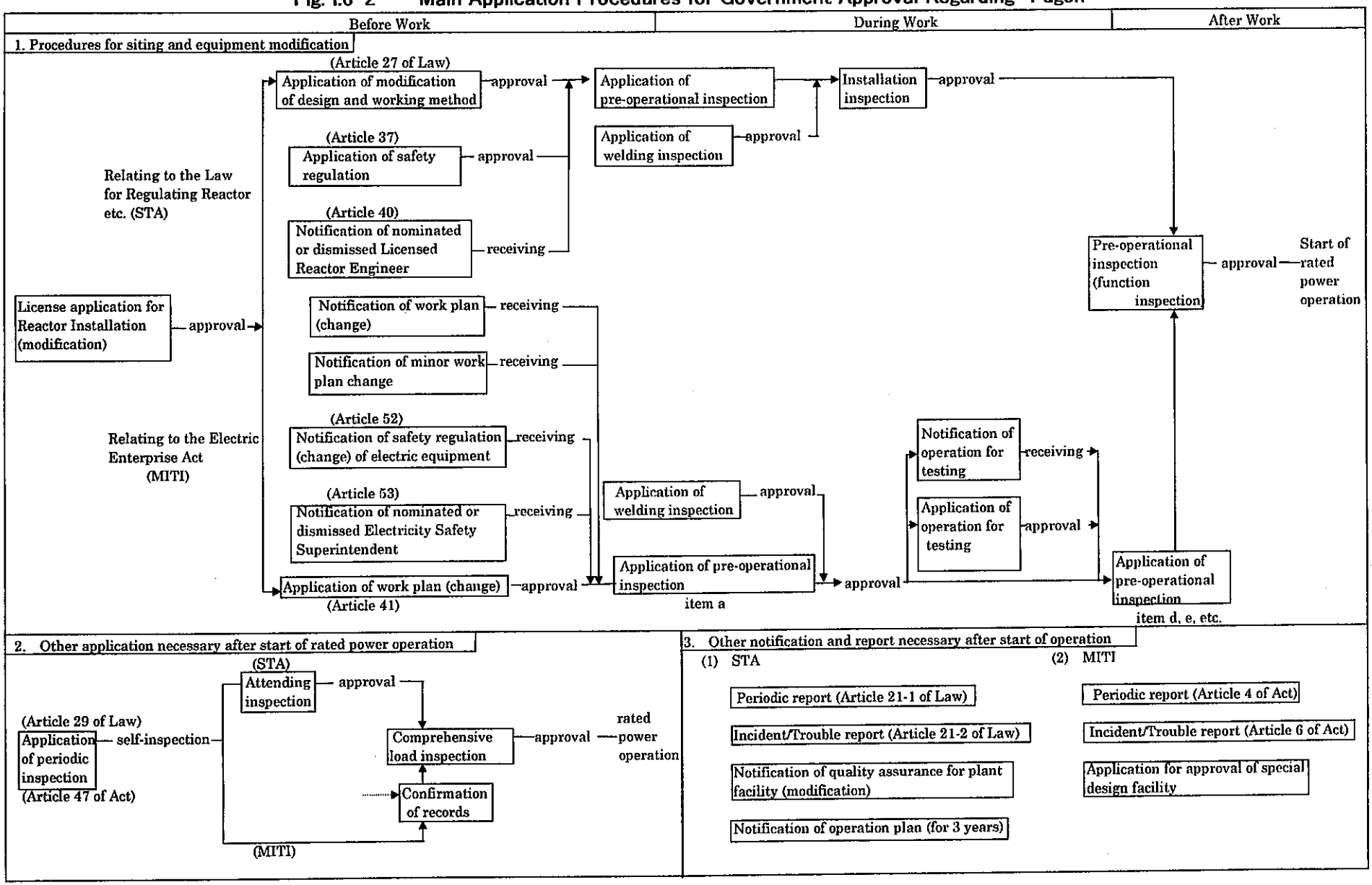
② High-Pressure Gas Storage 1 unit

Fig.I.6-1 License Approval Procedures of Reactors in Development Stage



(N.B.) 1. On the basis of the Laws for Regulating Nuclear Raw Materials, Nuclear Fuel Materials and Nuclear Reactors. Number in the parenthesis indicates provision.
2. The solid lines indicate normal business lines, and the broken lines those carried out if necessary.
3. For the Atomic Energy Commission, inquiry is done for No.1, 2 and 3 (relating to accounting bases), whereas for the Nuclear Safety Committee for No.3 (relating to technical matter) and No.4.

Fig. I.6-2 Main Application Procedures for Government Approval Regarding "Fugen"



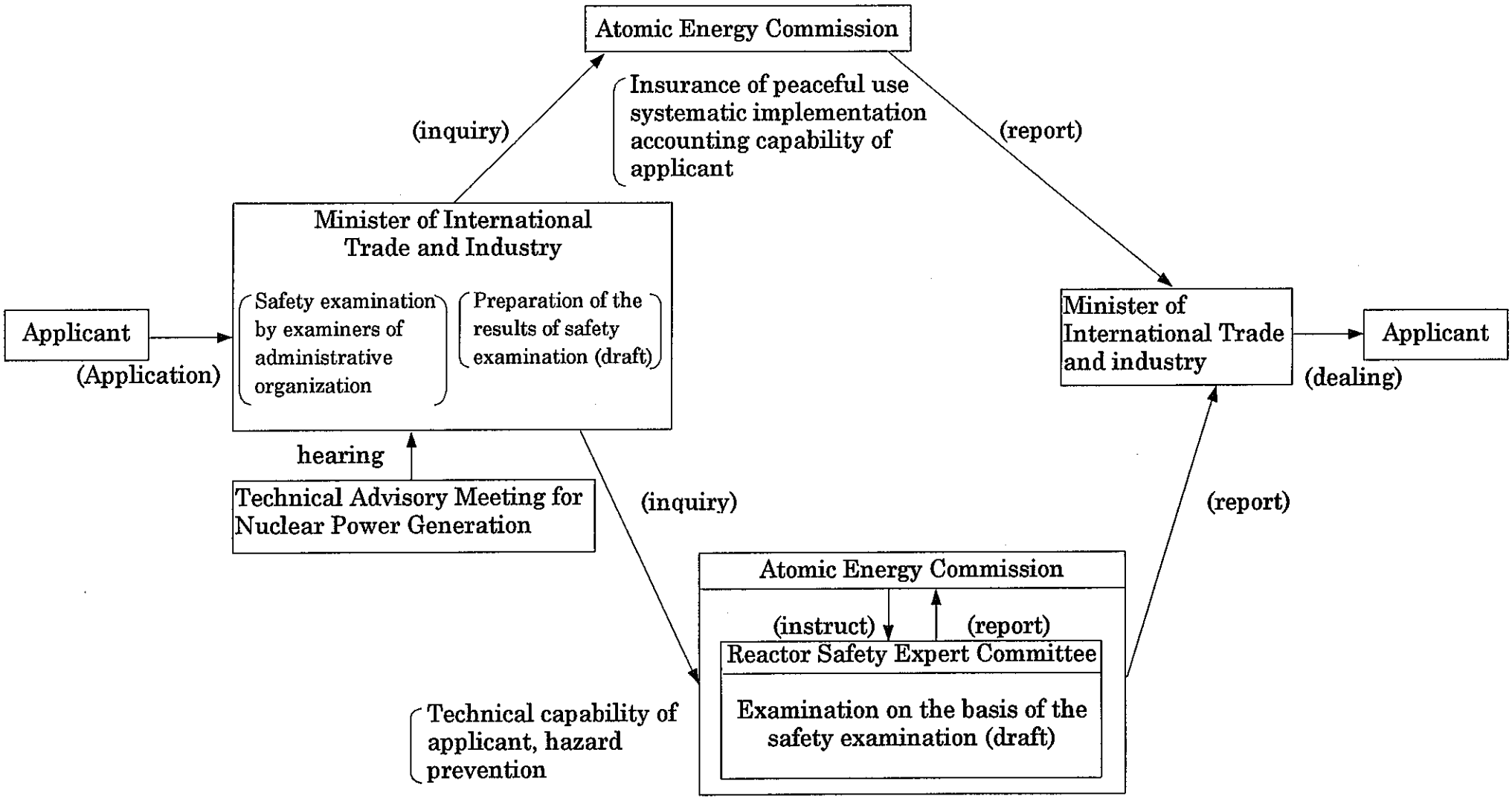
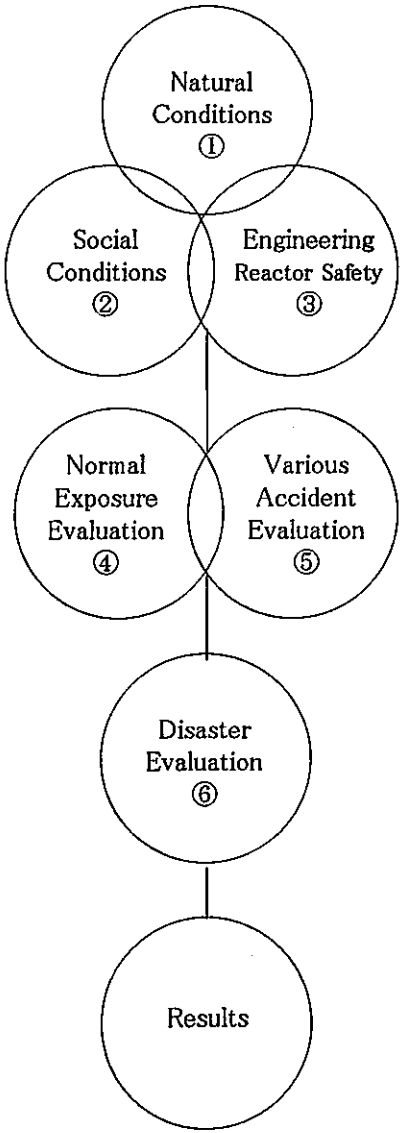


Fig.I.6-3 Procedures for Reactor Safety Examination



① Natural Conditions ——— Terrain, Geography, Land Water, Hydrography Earthquake, Meteorology, etc.

② Social Conditions ——— Population distribution, Industrial activity, Public facilities, Transportation, etc.

←Adequacy of standards (A), (B) and (C)

③ Engineering Reactor Safety — Reactor & core, Fuel handling system, Cooling system, Reactor auxiliary system, Instrumentation & control, Radwaste storage, Radiation control facility, Reactor containment, Nuclear & thermal design & dynamic analysis, seismic design, etc.

←Adequacy of standards (B)- (C), (E) and (F)

④ Normal Exposure Evaluation ——— ←Adequacy of standards (C) – (G), (H) and (I)

⑤ Various Accident Evaluation ——— ←Adequacy of standards (B) – (D), and (E)

⑥ Hazard Evaluation — Major Accident — ←Adequacy of standards (A) – (G)

— Hypothetical Accident

N.B.) Refer to the "Safety Examination Standards of Atomic Energy Safety Committee" (Fig.I.6-5).

Fig. I.6-4 Contents of Reactor Safety Examination

Fig.I.6-5 Safety Examination Standards of Atomic Energy Safety Committee

Primary Guidelines

A Yardsticks for reactor site examination and its application (site examination guideline)

B Safety design guideline for LWRs (safety design guideline)

C Guideline for target dose near LWR facility (ALAP guideline)

Secondary Guidelines

D Examination Guideline for safety evaluation of LWRs

E Safety evaluation guideline for emergency core cooling system of LWR

F Aseismic design guideline for LWRs

G Meteorological guideline for safety analysis of power generating reactors

H Guideline for measuring released radioactive materials near LWR facilities

I Guideline for evaluating target dose near LWR facilities

7. Education and Training

7.1 Education and Training System and Structure

(1) Education and Training System

As to education and training, based on the job training (OJT) and desk education, technically specialized education outside PNC, such as qualification tests and various and short courses of study, and PNC's education on a class bases have been carried out. Fig.I.7-1 shows the education and training system in "Fugen". These education and training have been performed in accordance with the "Annual Education and Training Plan" established by the Labor and Personnel Section on the basis of "Education and Training Plan" prepared by taking into account the work plans, lineup and lineup plan of each Section.

(2) Education and Training Structure

Education and training have been executed in the normal plant system. The persons in charge of education are nominated in each Section, and they adjust and prepare the contents and plans of in-Section education. In the plan, the "Education Review Working Group" (secretariat : Labor and Personnel Section) consisting of the persons in charge of each Section including the Labor and Personnel Section has been organized for investigating education and training.

7.2 Education Method

As mentioned in the preceding subsection, education and training have been carried out through taking various qualification tests and joining short course studies, with OJT and desk education being the center. For fresh personnel, in particular, the "Fresh Personnel Education Plan" is established in "Fugen", and the basic knowledge of the plant as a whole given to them, mainly with OJT, for a year after employed, and then they are attached to each Section.

7.3 Execution of Education and Training

The results of education and training (of the Power Generating Section) in "Fugen" are shown below, whereas those by the safety regulation is given in Table I.7-2.

(1) Result of Education by Safety Regulation

Education stipulated by the safety regulation has been performed annually for all the personnel; the contents are the education of the safety regulation, and that of high-pressure safety, hazardous material safety and electric safety stipulated by the disaster prevention regulation.

(2) Education in Fuel Cycle Training Center of Tokai Works

Training in Tokai Works had been carried out before the Fuel Cycle Training Center was established there, and many personnel were trained in the fields of radiation protection, quality assurance and safety training for supervisors.

(3) Training in Outside Organizations

To obtain technically specialized knowledge for operators, many personnel were dispatched to the RI/Reactor Training Center of JAERI (Japan Atomic Energy Research Institute) and the Comprehensive Training Center of JAPCO (Japan Atomic Power Company) for education.

Table I.7-1 Education and Training Program (in-Section Education) for Operators

Classification	Goal of Education	On the Job Training	Comprehensive Education (for a week)	Operation Training for Emergency (once/month)	In-shift Education (in PNC, any time)
Trainee	(1) Capable of Surveillance / check (2) Capable of simple operations (3) Capable of understanding sequence	(1) Outlines of equipment & systems, and field arrangement (2) Surveillance/check (route, basic items of confirmation) (3) Operation (with monitoring) of auxiliary equipment	Junior (I) course (basic knowledge of nuclear energy, surveillance/ check procedures, mental attitude of operators)	Field trip	On a shift basis
Junior Operator	(1) Capable of surveillance/check (2) Capable of field operations (3) Capable of doing simple periodic tests (4) Capable of doing simple isolation	(1) (field) Operation of equipment/system (2) Surveillance/check (3) Simple periodic test (4) Simple isolation	Junior (II) course (education for each system & equipment --- objectives, interlock, operation procedures)	(1) Confirmation of emergency operating procedures (2) Simulated operation training (3) Discussion on results and reflection (4) Training items Scram, Turbine trip, LOPA, LOCA, Loss of feed water etc. (N.B.)Basically, in-shift training, but under the attendance of Licensed Reactor Engineer once a year	
Operator	(1) Capable of instructing field operations (2) Capable of operations in control room (under the attendance of senior operator) (3) Capable of doing periodic tests (4) Capable of doing isolation	(1) Operation of equipment/system (in control room) (2) Periodic test (3) Isolation during periodic inspection	Intermediate education (control & instrumentation, trouble training)		
Senior Operator	(1) Capable of doing plant startup/shutdown (2) Capable of response to plant emergency (3) Capable of isolation at the time of periodic inspection (4) Capable of doing operation control	(1) Startup/Shutdown of plant (2) Operation at the time of emergency (3) Operation control (4) Operation management	Senior Education (operation control)		
Assistant Shift Leader	(1) Capable of doing plant startup/shutdown (2) Capable of instruction at the time of plant emergency (3) Capable of doing plant control	Jobs of shift leaders and their assistants			

Table I.7-2 Education and Training Program (outside-Section Education) for Operators

<table>
<tr><th rowspan="2"></th><th rowspan="2">In-PNC Education</th><th rowspan="2">In-PNC Education</th><th colspan="3">Outside-PNC Training</th></tr>
<tr><th>BTC Training</th><th>Training in JAERI, JAPCO</th><th>Qualification Test, etc.</th></tr>
<tr><td>Trainee</td><td rowspan="4">(1)Radiation control education
(2)Labor safety education
(3)Damage prevention training
(4)Early stage fire fighting training
(5)First did training

For all shift operators, once a year or more</td><td>(1)New personnel training
(2)Additional new personnel training</td><td></td><td></td><td></td></tr>
<tr><td>Junior Operator</td><td rowspan="2">(1)Leading personnel Training
(2)Others</td><td rowspan="3">Dispatch every year systematically.
(1)Standard training course (3 months)
(2)Shout training course (3 weeks)
(3)Special training course (10 days)
(4)Family training course (3 days)</td><td>JAPCO training course(A): Give knowledge necessary for trainees and operators.</td><td rowspan="2">(1)2nd-class boiler engineer
(2)Hazardous material engineer
(3)3rd-class refrigerator engineer
(4)Others</td></tr>
<tr><td>Operator</td><td>JAPCO general curriculums (B): Give knowledge necessary for senior operators.</td></tr>
<tr><td>Senior Operator</td><td>(1)Supervisor training
(2)Safety training of supervisor
(3)Others</td><td>(1)Licensed Reactor Engineer
(2)Licensed Engineer for Radiation Safety
(3)Licensed Engineer for Nuclear Fuel
(4)Licensed Engineer for Class-1 Refrigerator
(5)Licensed Electrical Engineer
(6)Licensed Engineer for Boiler/Turbine</td></tr>
<tr><td>Assistant Shift Leader</td><td></td><td></td><td></td></tr>
</table>

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Development of Advanced Thermal Reactor "Fugen"

Japan Nuclear Cycle Development Institute
Fugen Nuclear Power Station

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Ⅱ. Operation Management [II-1][II-2][II-3] [II-4] [II-5]

1. Outline of Operation Management

"Fugen" is a prototype advanced thermal reactor (heavy water moderated boiling light water cooled reactor) developed by Japan, with an unique operation control method established. This operation method unique to "Fugen" has been developed by making reference to those of other commercial nuclear power plants and IAEA Safety Guides, and thereby achieving good operational results.

Since "Fugen" uses heavy water as its moderator and is of pressure-tube type, its features lie in the operational characteristics and in fuel handling, but its boiling-light-water-cooling system makes its operational characteristics of the turbine system resemble those of BWRs.

The operational characteristics of the plant have been revealed to be stable and very well as designed. This fact was verified by the one-year startup test [II-6] started from March, 1978 and subsequent 19-year operational results.

For safe and reliable operation of the plant, establishment of an operation control system and operation technique, especially preparation of operating procedures and education and training of operators are essential. As to the operating procedures, we have the "Startup/Shutdown Procedures", "Emergency Operating Procedures", etc. Those present procedures were established in accordance with the experience obtained through the "system tests", "startup tests" and "comprehensive functional tests"; afterwards, operational experiences, results, improved operability and various modifications were reflected on these procedures for further exact operations. On the basis of these operating procedures, startup/shutdown operations, periodic tests for confirming the functions of equipment important to safety and surveillance of equipment have been carried out to keep safe operation and improve reliability.

As to education and training of operators, some of them were first dispatched to a CANDU reactor plant in Canada to learn its operation, because "Fugen" is the first heavy water reactor in Japan; moreover, many operators were trained in the domestic thermal power and nuclear power plants for operation. After the startup of "Fugen", education and training in outside organizations have been systematically performed in addition to the basic desk training and on the job training. In particular, in view of the importance of operations at the time of plant emergency, the emphasis has been placed on the operational training using the operation simulator in the BWR Training Center. Since 1990, moreover, the site-type compact simulator was introduced for more effective training.

Besides the above, in order to strengthen operation control, the operation data of the plant have been collected and processed by the computer using the comprehensive evaluation system for early detection of plant anomalies and the state of operation [II-4].

Operational work management, moreover, have been exactly performed by using the maintenance management and isolation-control systems developed by PNC.

2. Operation Management System

2.1 Basic Policy of Operation Management

The operation management of "Fugen", a prototype reactor in the R&D stage, should be established. On the other hand, it is equipped with the power generating system and transmits power to the general transmission network; accordingly, it is considered as a commercial power generating nuclear reactor, so that the same strict operation management as that of electric power generating company is required.

Therefore, "Fugen" has duties of following various regulations and agreements with the local governments and safety assurance of the public living around the reactor facilities; accordingly, PNC has followed these upon operating the reactor to secure their safety.

The safety regulations of reactor facilities were stipulated upon operation to realize plant safety, based on which the practices, various operating procedures, and various plant regulations and operation manuals were prepared for management.

2.2 Operation System

Since "Fugen" has supplied electricity to general users by using the transmission lines of private power generating companies, it has performed 24-hr continuous operation like commercial nuclear power plants although it is a prototype reactor.

Accordingly, the Power Generating Section has operated the reactor on 4-group 3-shift basis with a supporting operation group. The shift operation consists of 5 groups, i.e. A, B, C, D and E groups. Among them, 4 groups perform shift operation, and the rest one carries out day service.

The working cycle of each group consists of 70 days; namely, 56-day shift operation and 14-day day service.

When the periodic inspection is performed, the "periodic inspection management group" is organized to fulfill the service.

The organization of the Power Generating Section is shown in Fig.II.2-1.

(1) Shift Operation Services

The main shift operation services include monitoring of plant operating state, surveillance/check of the reactor facilities, periodic test and change over of the systems, equipment control during operation and shutdown, work management and isolation.

The shift operation is carried out by the shift leader, assistant shift leader, senior operators (2-3 persons), operators (2-3 persons), junior operators (2-3 persons) and

trainees (0-2 persons); total number is about 10-12 persons.

The number of shift members depends on the state of plant management.

(a) During Normal Operation

The normal operation is usually performed by a shift leader, a work supervisor, 3 persons in the central control room, 4 patrolmen and 1-2 routine periodic testing persons, i.e. 10-12 operators in total.

In an emergency, the central control room is operated by a leader, a electricity operator, a turbine operator, a reactor operator and an assistant, i.e. 5 persons in total.

(b) During Plant Shutdown

During plant shutdown, it is maintained by a shift leader, an assistant shift leader, 3-4 persons for normal patrol, and 2-3 persons for refueling (a leader, 1-2 persons for operating the refueling machine), i.e. 8-10 persons in total.

(c) Normal Plant Startup and Shutdown Operations

Normal plant startup and shutdown operations are carried out by a shift leader, an assistant shift leader, 3 reactor operators, a turbine operator, an electricity operator, a monitor of power-up rate and 2-3 field operators, i.e. 10-12 persons in total.

Classification of the shift leader and various-class operators and their jobs are given in Table II 2-1.

(2) Jobs of Day Service Group

The jobs of the day service group are to support operation shifts. The main jobs include in-group education and changeover of auxiliary boilers, etc.

(3) Jobs of Operation Control Group

The jobs of the operation control group include comprehensive management of operation, planning of operation schedules, preparation of refueling program, power demand/supply plans and consolidating operation-related documents, planning and execution of operation budget, and procurement management and preservation of operation records. The latter two are important for operation management, and the obligation of preserving important records is imposed by the law.

Besides the above, as part of the operation control of the plant, the heavy water purification and waste treatment systems, and the installations for raw water and water for miscellaneous use are operated and controlled.

Operations of these systems and installations have been carried out by the Tsuruga Atomic Power Service Co., Ltd. under a consignment contract, but

operation and monitoring of them are performed under the management of PNC's shift operators.

In addition, preparation of operation documents (operation procedures and documents for training), and revision and discussion of their contents are carried out in the "Operation Technique Investigation Meeting" by senior operators (or operators) (6 normal members and 5 assistant members) selected by each shift leader.

Preparation and revision of operating procedures are mainly performed by the senior operators.

2.3 Establishment and History of Operation System

(1) Operation System

When the comprehensive functional test ended and moved to the startup test in March, 1978, the 4-group (1 group consisting of 12-14 persons) 3-shift (with the shift leader being in charge) system was introduced. Since then, this operation system had been continued up to March, 1985.

Meanwhile, in private electric power generating companies 2-holidays-per-week system had been introduced since about 1982, thereby specified holiday of the shift operators greatly increased, so that the 5-group 3-shift system was introduced instead of the conventional 4-group 3-shift system.

Based on such a background, "Fugen" introduced the 4-group 3-shift one-day service system since April, 1985 to give adjusting holidays to the shift operators and to improve their working conditions.

(2) Management System During Inspection

"Fugen" has undergone 14-times of periodic inspection from March, 1979 to October, 1997.

In the 1st to 7th periodic inspection, work coordination with other Sections was performed in the "periodic inspection supporting group" consisting of 4-5 persons, while the isolation, test runs and so on, associated with the periodic inspection works, were carried out as the shift operation.

However, when measures should be taken by limited number of shift operators, and when the operation continues from the 1st to the 2nd shift or for several days, there was a room for improvement from the viewpoint of working efficiency because of necessary suspension of works or reconfirmation of operation status at the time of shift changeover.

After the 8th periodic inspection, therefore, the "periodic inspection management

group" consisting of 12-17 persons has been organized to perform the periodic inspection, such as to coordinate works with other Sections, to make isolation of the periodic inspection works, to conduct test runs, etc.

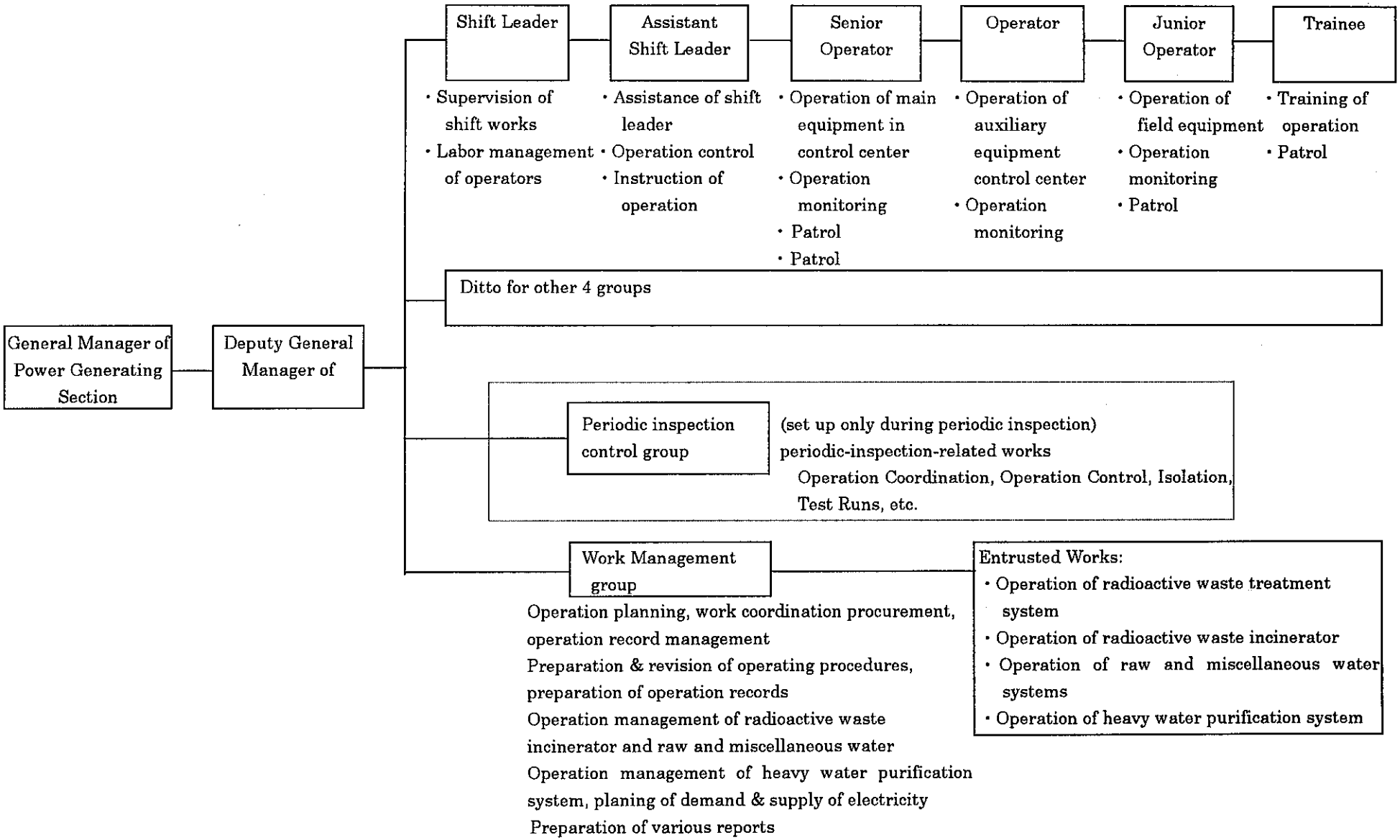


Fig.II.2-1 Organization of Power Generating Section

Table II.2-1 Main Works of Power Generating Shift Members

Class	Operation Experience	Main Work
Shift Leader	Person having ability to command his followers and sufficient nuclear-energy-related knowledge and technique	Instruction of operation, and labor management of shift operators
Assistant Shift Leader	Person having ability to command his followers and sufficient nuclear-energy-related knowledge and technique	Assistance of the shift leader, Instruction of plant operation, and Operation control
Senior Operator	Person having operation experience of 7 years or more	Operation of main equipment in the control room, Operation monitoring, and Patrol
Operator	Person having operation experience of 4 years to 7 years	Operation of auxiliary equipment in the control room, Operation monitoring, and Patrol
Junior Operator	Person having operation experience of a year to 4 years	Operation of field equipment, and Patrol
Trainee	Person having operation experience of less than a year	Training of operation (operation by himself is prohibited) Patrol

3. Operation Technique [II-8] [II-9] [II-10] [II-11]

3.1 Establishment of Plant Operating Procedures

Operation of "Fugen" has been carried out in accordance with various operating procedures prepared on the basis of the "Reactor Facility Safety Regulation" and "Reactor Facility Safety Regulation Practices".

As to the operating procedures, "Fugen" has the "Startup/Shutdown Procedures", "Emergency Operating Procedures", "Operating Procedures for Each System", "Periodic Test Procedures", "Surveillance/Check Procedures", "Special Handling Procedures", "Alarm Responding Procedures" and "Emergency Operating Procedures (EOP)".

The operating procedure system of "Fugen" is shown in Fig. II 3-1.

The origin of the present operating procedures were set up in accordance with the results of plant safety analysis, experiences obtained by the system tests, startup tests and comprehensive functional tests and with the operation techniques of the precedent LWRs. Afterward, these operating procedures were reviewed by reflecting the troubles encountered in "Fugen", operational achievements and modification of equipment and systems for improvement.

3.2 Outline of Plant Startup Precedures and Main Improvements

Typical plant startup and shutdown curves are given in Fig.s II 3-2 and II 3-3, respectively.

About 20 hr are required from reactor startup to parallel in, and about 8 days to reach rated power.

The operation techniques regarding reactor startup and shutdown were first set up by referring to those of LWRs and adding knowledge obtained by the startup test of "Fugen", and then gradually improved by reflecting operational experiences gained afterward.

The current procedures and main improved points are described in the following paragraphs.

(1) Deaeration Operation

Since dissolved oxygen of reactor coolant give great influence on the integrity of reactor coolant systems and piping of engineered safety protection systems, reactor cooling water, feed water and condensate are deaerated by vacuum before plant startup. Vacuum deaeration is carried out by evacuating the gas phase of steam drums and condenser by the startup vacuum pumps and replacing reactor cooling water with deaerated condensate.

Replacement with dearated water is performed by using the deaerator operating system modified from the residual heat removal system. The deaerator operating system is shown in Fig. II3-4.

The target of deaeration is to decrease the dissolved oxygen concentration in the reactor coolant to 0.2 ppm or less. The dissolved oxygen concentration in the steam drum decreases to 0.2 ppm or less, about 6 hr after start of deaeration of the steam drum, and is finally stabilized at about 0.1 ppm. In parallel with vacuum deaeration, water in the injection nozzle of the engineered safety features is replaced with deaerated water. There are 28 injection nozzle zones in total, and most of deaeration time (about 30 hr) are spent for replacement of water in the nozzles. After deaeration, about 1-kg/cm^2 g N_2 gas is enclosed to keep dissolved oxygen concentration of the coolant low.

(2) Reactor Startup, Criticality and Nuclear Heating

(a) Present Procedures

For the reactor startup, changeover the reactor mode switch from "shutdown" to "startup", and then withdraw the control rods in accordance with the control rod withdrawal sequence.

About 3 hr after the reactor startup, it reaches criticality. Judgment of reaching criticality is carried out on the basis of the degree of increment in the value indicated by the Start Up range Monitor (hereinafter referred to as "SUM").

When the range of the Power Up range Monitor (hereinafter referred to as "PUM") exceeds 3, SUM becomes unnecessary and is taken out of the reactor.

About 30 minutes after reaching criticality, nuclear heating starts. The output in the beginning of nuclear heating is about 8 (about 0.5% of total output) in the PUM range.

The temperature rising rate of the primary cooling water is set at 30°C/h for the limit value of 55°C/h. For around the rated pressure (68 kg/cm^2 at the steam drum), however, it is changed to 15°C/h.

(3) Changeover of Rated Pressure and Mode Switch

The rated pressure is reached at about 5% reactor power. Temperature rising rate near the rated pressure is 15°C/h as mentioned above.

Power-up rate up to 18% reactor power after reaching the rated pressure is about 0.2%/min, taking into account the controllability and stability of the reactor.

At the reactor power of about 7 to 8%, the mode switch is changed over from "startup" to "operation", where neutron flux (power) is monitored by the power

range monitor and PUM is taken out of the reactor.

(4) Changeover of Feed Water Flow Control Valve (Changeover from Low Flow Control Valve to Main Feed Water Flow Control Valve)

(a) Present Procedures

About 12 hr after reactor startup, it reaches the rated pressure (68 kg/cm^2 g at about 7% rated power) and the turbine bypass valve opens. Then, power continues to rise (about 18% rated power) until the opening of the turbine bypass valve becomes about 80 %. During this period, the feed water flow control valve is changed over (the low flow control valve is "completely closed" and the main feed water flow control valve is "automatically opened and controlled").

The feed water flow rate (= steam flow rate) at about 80% opening of the turbine bypass valve is about 45 T/h (for a loop) and the opening of the low flow control valve becomes about 80%. The reason why changeover is carried out at this power is that greater feed water flow rate makes the influence given to flow rate at one valve operation smaller and thereby stable changeover becomes possible.

(b) Main Improvements

Initially, the changeover of the feed water flow control valve was performed at the reactor power of 7-8 % by the capacity limit of the low flow control valve. At this power, however, feed water flow change at the feed water changeover greatly affected the void fraction in the core which made water level in the steam drum change much. Accordingly, the capacity of the low flow control valve was increased, and the reactor power at the changeover was changed to the current power (about 18% rated power).

In "Fugen", to reduce the work load of the operators in monitoring and handling by improving the controlling performance of reactor feed water flow control system at reactor startup/shutdown, a fuzzy control system was employed for operating the low flow control valve.

(5) Turbine Startup and Generator Parallel-in

(a) Present Procedures

The turbine is started at the reactor power of about 18% and turbine bypass valve opening of about 80%. The turbine startup is classified into cold, warm and hot start depending on temperatures of each part of the turbine at startup and temperature of lubricating oil. In cold start, speedup rate of the turbine is 120 rpm/min, and it takes about 30 min to reach the rated speed (3600 rpm). During this period, rub- check is performed at 200 rpm. After reaching the

rated speed, no-load operation is carried out for 30min to 1 hr for warming up the turbine, and then the generator is paralleled in. Upon startup of the turbine after its overhaul, great care must be paid for the vibration of each bearing. After the generator is paralleled in, operation is performed with initial load (about 8MWe) for 15 min to confirm no anomalies in each part, and then power is raised.

Table II 3-2 shows the conditions for turbine startup and paralleling of the generator.

(6) Formation of Control Rod Pattern

The control rods are withdrawn sequentially in accordance with the control rods withdrawal procedure up to generator output of 58MWe, i.e. reactor power of about 40%. Reactor power is kept constant at 40%, and the control rod pattern for rated operation is formed with the build-up of xenon being compensated. Time required for forming the control rod pattern is about a day.

The control rod pattern during the rated power operation, except special cases, consists of 70% position of automatic power control rods "E control rods" (60 ~ 70 % position after overlapping the envelope described later) and over 65% position of power fine control rods ("C control rods") with the other control rods withdrawn completely.

For 18 hr from power-up after forming the rod pattern to the rated power, operation of the control rods is prohibited; accordingly, power-up after forming the control rod pattern is realized by removing poisons.

(7) Changeover of Recirculation Pump Speed

After forming the control rod pattern, the speed of the reactor recirculation pump is changed from low (450 rpm) to high (900 rpm). The changeover of speed is carried out by pole changing of the pump motor. Since the changeover of recirculation pump speed decreases the water level of the steam drum by about 200mm, it is set up at (NWL + 140mm) in advance.

The changes in water level of the steam drum at the time of changeover of the recirculation pump are shown in Fig.II.3-5 and Fig.II.3-6.

Decrease in the water level of the steam drum stems from disappearance of coolant void due to increase of core flow rate and from the increase in suction flow rate. This phenomena is a transient change and recovered after several seconds, so that the feed water control system is operated "manually". Moreover, as the coolant void reactivity coefficient is as small as zero, the effect of coolant void change is small on the reactor power. In addition, changeover in the

recirculation pump speed is performed on a single loop basis because of rather large fluctuation of water level of the steam drum.

(a) Reactor Power at Changeover of Recirculation Pump Speed

Initially, reactor power level at the changeover of recirculation pump speed upon reactor startup and shutdown had been at about 45% reactor power (generator output of 66 MWe). However, when the reactor was started up in June 1984, the reactor pump speed was changed to confirm the coolant void reactivity coefficient by the position of the control rod. But regional power (RPM) instantaneously exceeded 60%, thereby the interlock was actuated, resulted in automatic shutdown of the reactor. In this period, the test was carried out with a special control rod pattern different from the normal one. From the viewpoint of securing safety operation, hereafter, changeover of the recirculation pump speed from low to high (at reactor startup) has been carried out at 58MWe (about 40% of reactor power) and from high to low (at reactor shutdown) at 66MWe (about 45% of reactor power). (revised in June, 1984)

(b) Water Level of Steam Drum at Recirculation Pump Speed Changeover [II-12]

Since the water level of the steam drum temporarily greatly fluctuates when the recirculation pump speed is changed, its water level is usually set up at a high level before the changeover. In "Fugen", the water level of the steam drum is set up, taking into account its setting error, the results of water level change at the changeover of the recirculation pump speed including in the startup tests and the difference in detection of water levels between the feed water control system and the reactor protection system (revised in June, 1982)

(8) Power-up after Recirculation Pump Speed Changeover

The power-up rate after the changeover of the recirculation pump speed is 0.2% /min up to generator power 88MWe (reactor power of about 58%) and 0.6%/h after that to rated power.

The power-up rate during reactor startup is set up from the viewpoints of the integrity of fuels and the limit (83°C/h) of the metal temperature change rate of the 1st stage steam room of the turbine, but the power-up rate corresponding to this limit is as fast as 1.3%/min; consequently, the actual power-up rate is primarily limited by the integrity of fuels (PCIOMR).

Initially, the rising rate of linear heat generation rate for all loaded fuels was uniformly set up in the region exceeding 8 kW/ft, but evaluation of the results of operation of "Fugen", the irradiation data of "Fugen" fuels in Halden, SGHWR and Studsvik R-2 reactor, and the published data of power-up tests led us to set the

limit of rising rate of linear heat generation rate on a fuel burnup basis. Namely, for low burnup fuel the PCIOMR (Pre-Conditioned Interim Operating Management Recommendation) application range in which the power-up rate is limited to 12 kW/ft at a maximum was raised, and the limit of rising rate of linear heat generation rate was defined as the function of burnup. In parallel with this, the detailed linear heat generation rate of each loading fuel assembly were analyzed in advance and the PCIOMR application power level was raised from 45% to 58% power, and the power-up rate after this point was increased from 0.4%/hr to 0.6%/hr. Accordingly, the time required from reactor startup to rated power was decreased by about 70hr, i.e. from about 200 hr to about 130 hr. (revised in August, 1986).

(9) Pre-conditioning

Rapid change in reactor power causes mechanical interaction between fuels and cladding (PCI: Pellet Clad Interaction) and results in great stress on the cladding, leading to fuel failure. In LWRs, to mitigate PCI, power-up rate is limited.

In "Fugen" the results of precedent fuel irradiation tests and the operation experience of precedent LWRs were evaluated and investigated, and the operation methods considering PCI have been employed. At present, power-up rate over the linear heat generation rate of 8 kW/ft is limited to less than 0.6%/h. After reaching rated power, power is kept constant for 18 hr, i.e. preconditioning for running-in of the fuels and claddings.

Preconditioning is effective during the period when the reactor is effectively operated. When the state of core is changed by the change in the control rod pattern or fuel replacement and when reactor shutdown period exceeds 3 weeks, preconditioning should be done again.

To make automatic operation by the E control rods possible, envelope is formed in the 60% position of the E control rods and in the 70% position. (As the definition in "Fugen", the first envelope is said to "Formation" and the second envelope is said to "Overlapping" for the purpose of distinction.)

(a) Power Holding Time for Envelop Formation

Initially, the power holding time for envelop formation after reaching rated power was 24 hr, but was reduced to 18 hr after evaluating the operational results of "Fugen" and LWRs. (revised in August, 1986)

(b) Control Rod Pattern Cahnge Method During Envelop Overlapping

Initially, to change the control rod pattern during envelop overlapping, reactor power was decreased to 85%, then the position of the E control rods was changed

from 70% to 60% and finally power was increased again to rated power. As the results of improved measurement accuracy of fuel power density and investigation of operability, a new technique was devised on the control rod pattern change which is made by slowly adjusting poison concentration while the reactor power is kept at rated power (revised in August, 1986).

3.3 Power Control Method during Normal Operation

Power control is carried out by adjusting control rods and poison concentration of heavy water. During normal operation, the E control rods are set at "automatic position" to respond to short-term and minor power change, while long-term power change due to fuel burnup is adjusted by the poison concentration in heavy water.

The poison concentration in heavy water is about 5 ppm in the Beginning of Operating Cycle (BOC), and about 0.1-0.5 ppm at the End of Operating Cycle (EOC).

E control rods are to be operated in the range of 60% ~ 70% position for automatic reactor power control which can be well done by only one of 4 control rods. The E control rods are gradually withdrawn as fuel burnup proceeds, and when they reach near 70%, poisons in heavy water are removed and the E control rods are inserted to about 60%. This operation is carried out once per 3-10 days during reactor operation (Fluctuation in days stems from the change in the withdrawal rate of control rods due to the influence of change in boric acid concentration in heavy water originated from fuel burnup and neutron absorption).

Fine control of reactor power is so designed as to be done by the C control rods, but because of excellent controllability of the E control rods in charge of automatic control, handling of the C control rods is almost unnecessary.

3.4 Outline of Plant Shutdown Operation and Main Improvements

(1) Outline of Shutdown Operation

Plant shutdown operation is as important as startup operation. The plant shutdown should be performed in accordance with the operating procedures, core management notification and instruction.

Time required for the start of shutdown operation to parallel off the grid is about 2 hr, and time required for cooling water in the steam drum decreasing to less than 100℃ (cold shutdown) is about 9 hr.

Usually, when the reactor is shut down as scheduled, the power reactivity coefficient is measured by utilizing the power change of the reactor. The power reactivity coefficient is measured by inserting the control rod (1E rod) by about 5%

and calculated from the amount of change in power at this time. Then, power is decreased at a rate of about 0.9%/min. Power-down rate is adjusted by the insertion and withdrawal of the control rods. Recirculation pump speed is changed from high to low speed at the reactor power of about 45%. The power-down rate after changeover of the reciculation pump is about 0.6%/min. Unlike reactor startup, limitation is not imposed on power-down rate by fuels, but by rate of change in the internal metal temperature of a high-pressure turbine (83℃/h). Moreover, the reason why power-down rate is changed before and after RCP changeover is that the change rate of the internal metal temperature of the high-pressure turbine is not in linear relation with power-down rate depending on the inlet steam condition of high-pressure turbine. As the countermeasure for SCC (Stress Corrosion Cracking), during temperature-decreasing operation, 1-2 kg/cm^2 N_2 gas is inserted into the steam drum to prevent dissolved oxygen concentration of reactor water from becoming high. In short-term shutdown, moreover, the steam drum stands by with being filled with nitrogen, so that deaeration at the plant re-startup is omitted to shorten startup time.

In the early stage of operation, reactor power-down rate was 0.6%/min. If power-down rate is small, negative reactivity caused by formation of xenon in the course of reactor shutdown should be compensated with withdrawal of the control rods, so that power-down rate from rated power to a power of 45% was changed to 0.9%/min. (under 45% power, power-down rate is still 0.6%/min). The reason for this is that control time by the control rods could be got longer by increasing power-down rate, thereby giving enough time for changeover of the feed water flow control valves, 30-min changeover time.

3.5 Operation Methods of Installations Peculiar to "Fugen" and Main Improvements

(1) Heavy Water-Helium System

Heavy water is used as the moderator, whereas helium as the cover gas. The level of heavy water in the calandria tank is adjusted by the pressure difference of helium between the overflow space of the calandria tank and dump space. The heavy water system also has a function of back-up reactor shutdown "heavy water dump". The helium communicating valve is rapidly opened, in case of emergency and eliminates the above pressure difference and then decreases heavy water level rapidly to shutdown the reactor. The design criterion of the heavy water dump is to decrease heavy water level by 2 m or more within 5 sec., but actually 3 sec. To maintain the quality of heavy water above the specified value, impurities have

been removed by passing heavy water periodically through ion exchange resins.

Helium gas is entangled and pressurized when heavy water overflows from the calandria tank and falls down into the dump tank, and thereby obtains recirculation force. The helium recirculats at the designed flow rate by the entanglement.

The features of the heavy water-helium system operation are found in the helium purification and exchange of ion exchange resins in the heavy water purification columms. The outline of these operations are explained in the succeeding subsections.

(a) Helium Purification

To remove impurities (mainly nitrogen and carbon dioxide gas) contained in helium, helium is passed through activated carbon to absorb impurities only.

The origin of impurities mostly comes from inclusion of air at the time of open checks during periodic inspection; therefore, helium purification is usually carried out after the open checks during periodic inspection.

Main improvement in the helium purification system lies in the addition of the decarbonation line. Initially, plugging occurred in the helium purification line during helium purification. This was because carbon dioxide gas contained in helium was compressed and cooled to dry ice, which remained in the activated carbon absorption column and caused plugging. Addition of the decarbonation line has improved helium purification operation.

Decarbonation is performed by absorbing carbon dioxide gas in the dehumidifier of the helium purification system. This improvement was carried out by partial modification of the helium dehumidifier line and adoption of molecular sieves having higher absorption capability of carbon dioxide.

(b) Exchange of Ion Exchange Resins for Heavy Water

The heavy water purification system consists of 2 heavy water purification columns and 2 poison removal columns: The former contains ion exchange resins to remove impurities such as corrosion products in heavy water, whereas the latter removes boric acid added to heavy water to absorb excess reactivity in the beginning of operating cycle in accordance with the burnup of fuels.

When the performance of the ion exchange resins are degraded, they are exchanged with new ones. To exchange resins, they are first dedeuterated (to replace heavy water in the used resins to light water) prior to discard, and transferred to the granular waste resin tank for storage.

Fresh resins are first rinsed with light water, and then deuterated and

transferred to the heavy water purification column or the poison removal column depending on the kinds of resins.

The greatly different points of exchange operation of ion exchange resins in the heavy water purification columns from normal one lie in deuteration of resins. In deuteration of resins, light water contained in normal ion exchange resins is replaced with high-concentration heavy water (99.7%).

On the other hand, the aim of replacing heavy water in used resins to light water is to recover heavy water and tritium contained in used resins before discarding them and to remove tritium completely from the resins. The aim of deuterating resins is to substitute light water in fresh resins with heavy water so as not to degrade the concentration of heavy water of the moderator.

(2) Carbon Dioxide Gas System

Carbon dioxide gas flows through the annulus between the calandria tube and the pressure tube to detect rupture of a pressure tube and a calandria tube, as well as the thermal shields.

The operation of the carbon dioxide gas system was initially of once-through type; i.e., CO_2 gas flow from "CO_2 gas supply system" → "core" → "exhaust system". To reduce the CO_2 gas consumption, it was recycled to the core recirculation line by using the compressor, thereby reducing CO_2 gas consumption from 430 kg/day to 33 kg/day, i.e., to 1/13.

(3) Refueling System

Refueling is carried out from the bottom of the reactor by using the refueling machine in accordance with the following procedures:

(a) As preparatory operation, a fresh seal plug is placed on the refueling machine.

(b) Move the refueling machine to the lower part of the core, and connect it with a designated pressure tube.

(c) The seal plug is disconnected by the refueling machine, and the seal plug is placed in the refueling.

(d) A spent fuel combined with the shield plug are accommodated in the refueling machine.

(e) A new seal plug is attached to the pressure tube.

(f) After separating the refueling machine from the pressure tube, the refueling machine is moved to the transfer port of the exchange pool.

(g) Combine the refueling machine with the transport.

(h) The spent fuel is transferred to the exchange pool by using the fuel loading machine through the transfer port. The spent fuel and shield plug are

disconnected in the transfer port, and is sent to each exchange pool by the loading machine. (Sometimes they are sent directly to the spent fuel storage pool).

(i) The spent fuel transferred from the exchange pool to spent fuel storage pool using the transfer chute.

(j) The spent fuel transferred to the spent fuel storage pool is placed on a designated rack by the fuel transfer machine.

Fresh fuels are loaded in nearly the inverse procedure for spent fuel withdrawal. The fresh fuels after inspection are temporarily placed in the spent fuel storage pool, and then successively transferred to the exchange pool through the transfer chute. The fresh fuels are combined with the shield plug in the transfer port, and then loaded into the designated pressure tube by the refueling machine.

Handling of the refueling machine is remotely carried out in the central control room.

Spent fuels are withdrawn at a rate of 5-6 assemblies a day, whereas fresh fuels are loaded at a rate of 9-10

In a refueling operation of "Fugen", it is essential to precisely handle fuel assemblies from the standpoint of quality assurance and safeguards. In order to plan and execute efficient operation, knowledge of experienced operators based on experiences is required. Therefore, an automatic control system based on the "Artificial Intelligence (AI)" technology has been developed and put into operation in the refueling operations [II-13].

This system was developed as an off-line system in 1989 firstly and then the development of the on-line system based on the same method started. The off-line system was put into practice in 1991 and the automatic control system was applied to the actual operation in 1995.

The development of this system has made it possible to optimize refueling schedule, to save control work and to improve reliability of refueling.

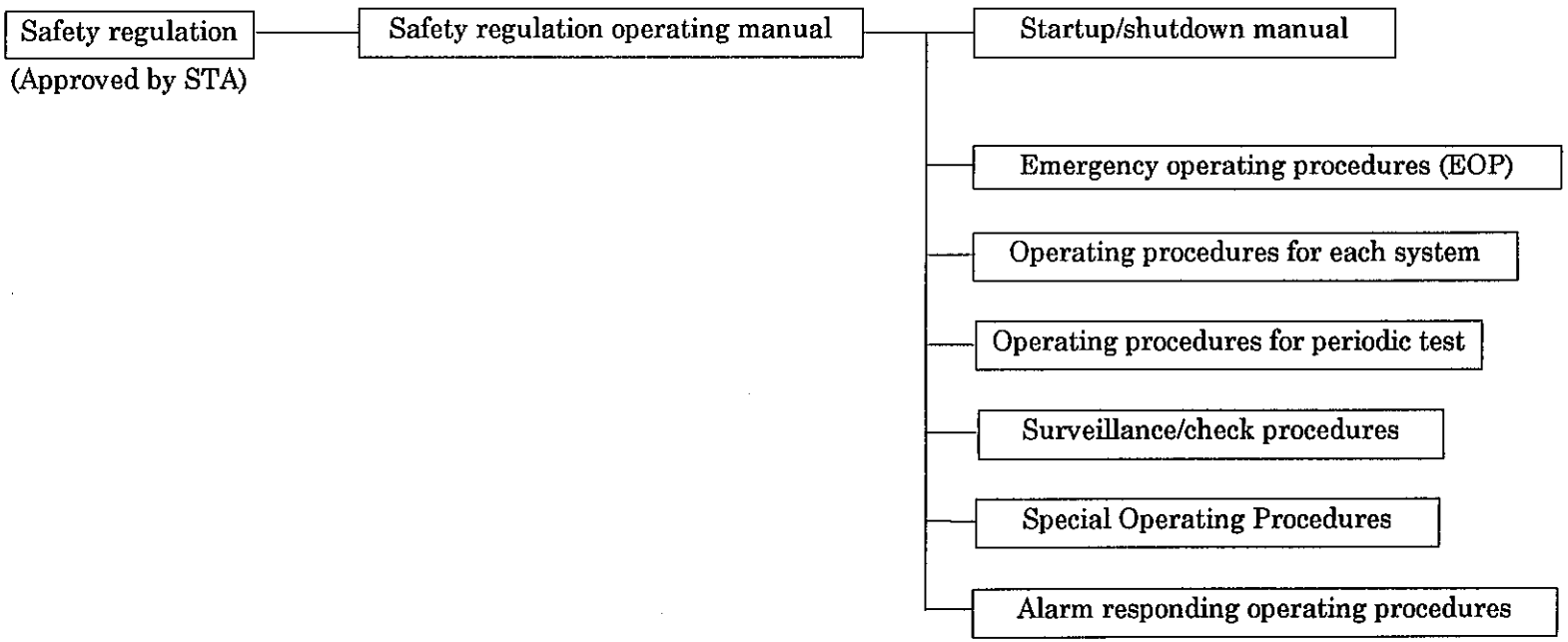


Fig.II.3-1 System of Operating Procedures of "Fugen"

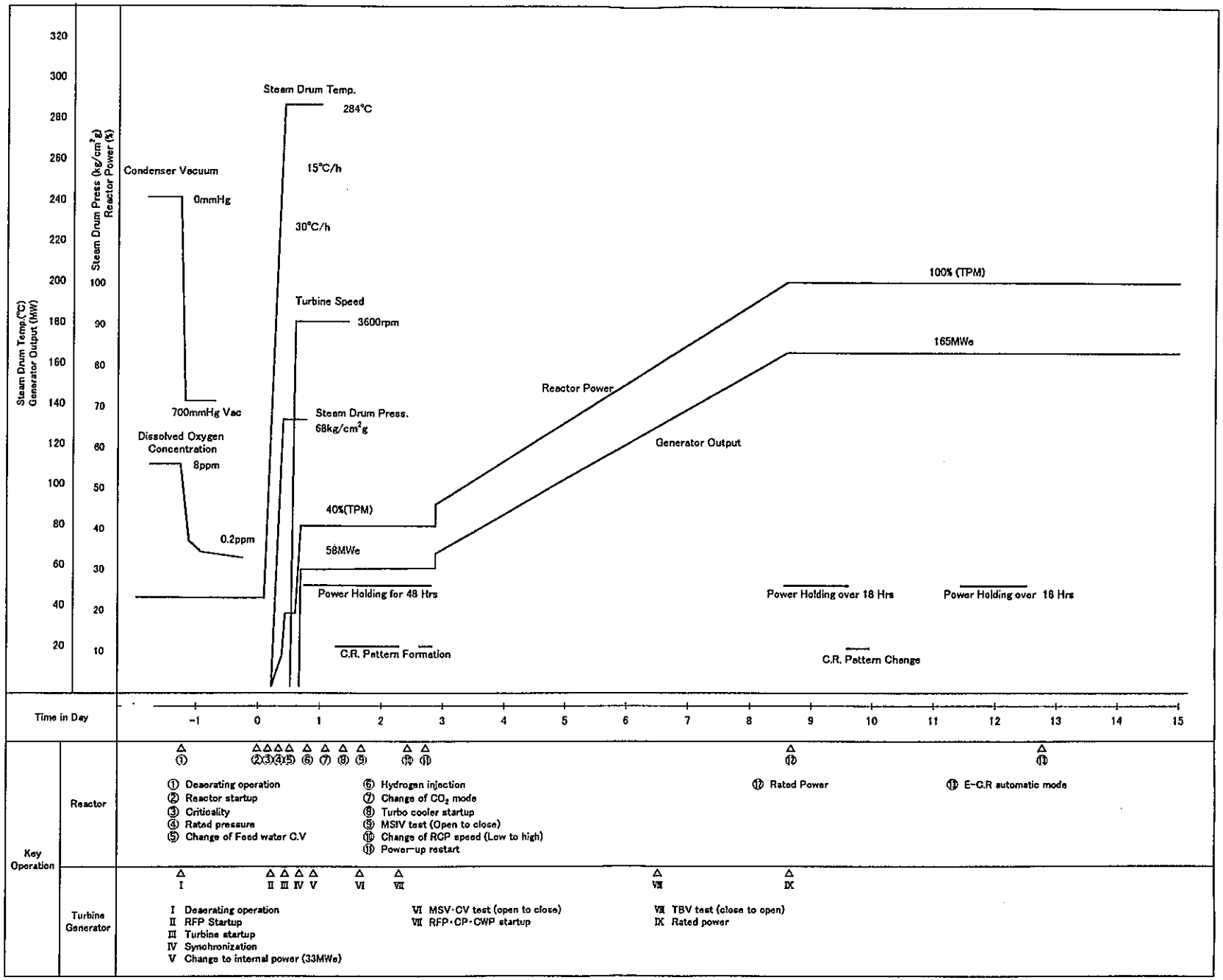


Fig.II.3-2 Plant Startup Procedures

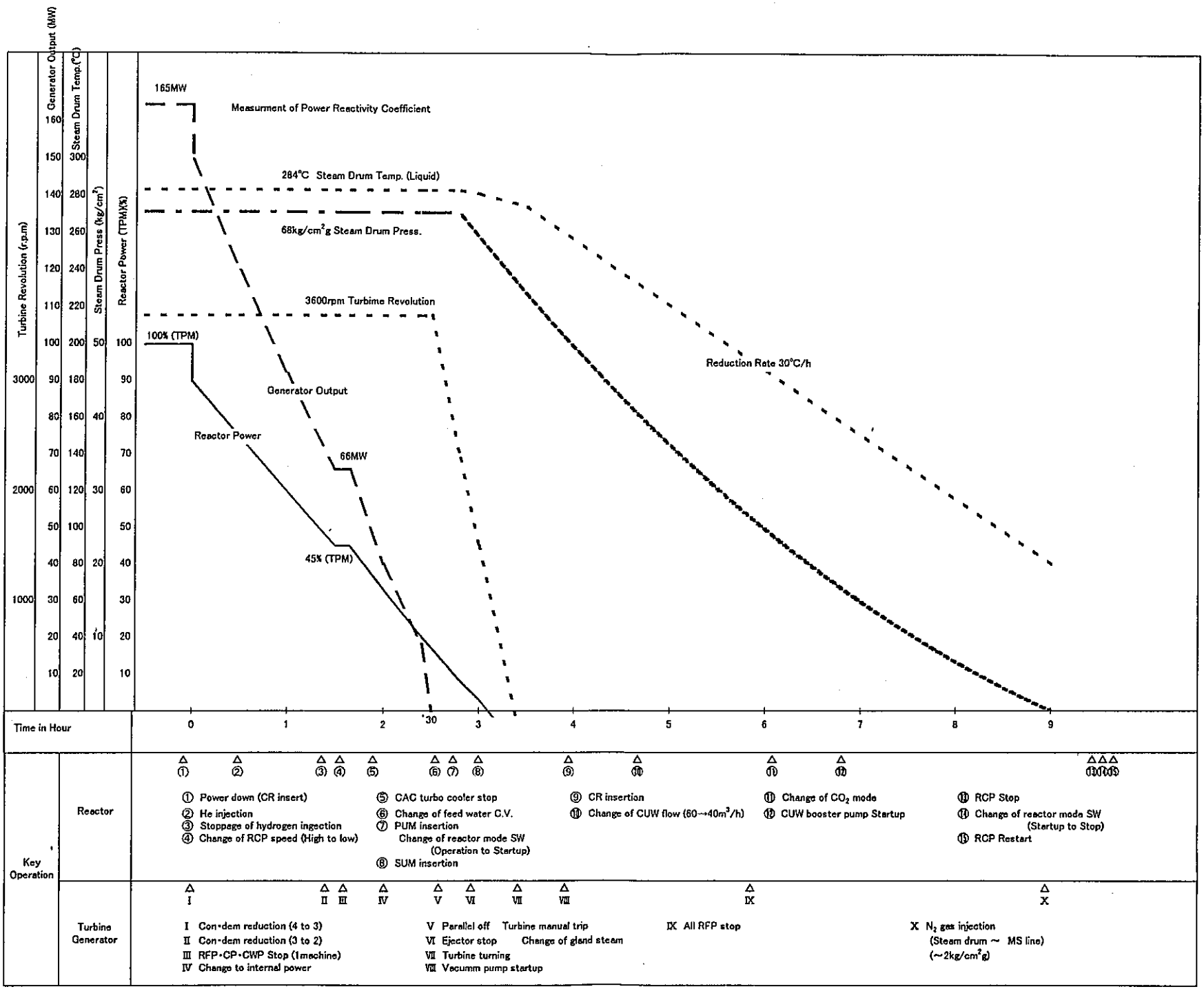


Fig.II.3-3 Plant ShutDown Procedures

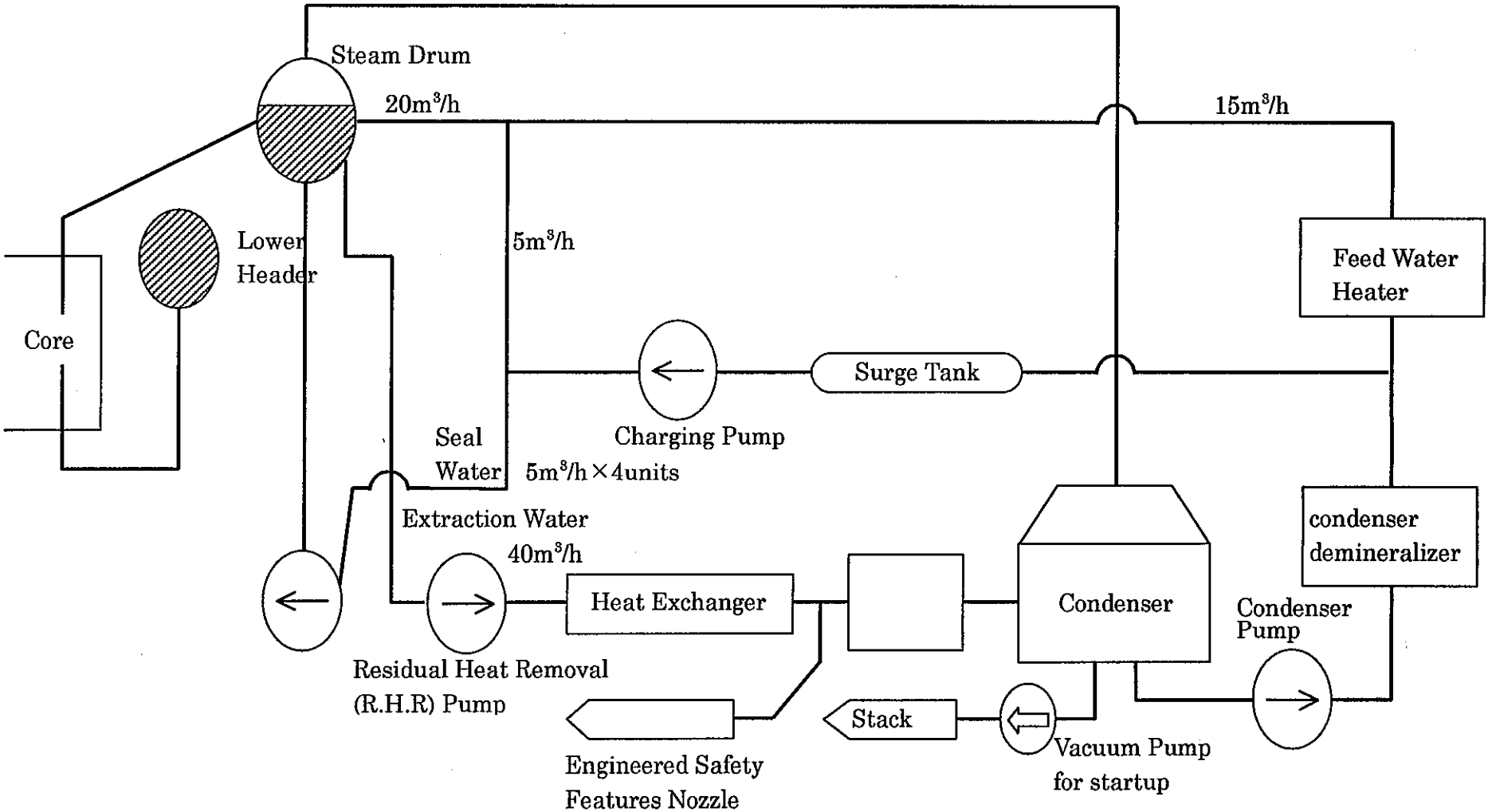


Fig.II.3-4 Deaerator System

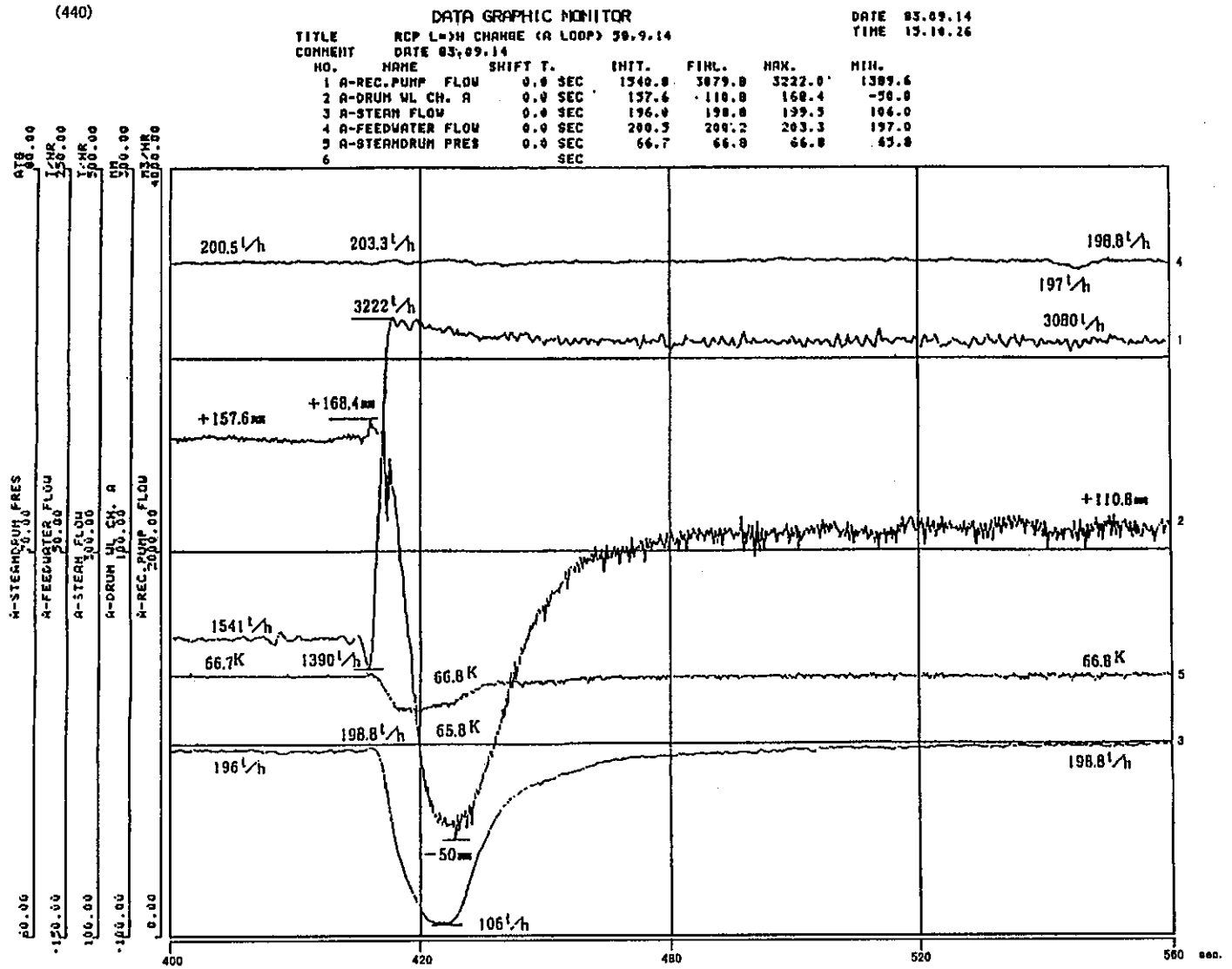


Fig.II.3-5 Changeover of Recirculation Pump Speed (Low to High)

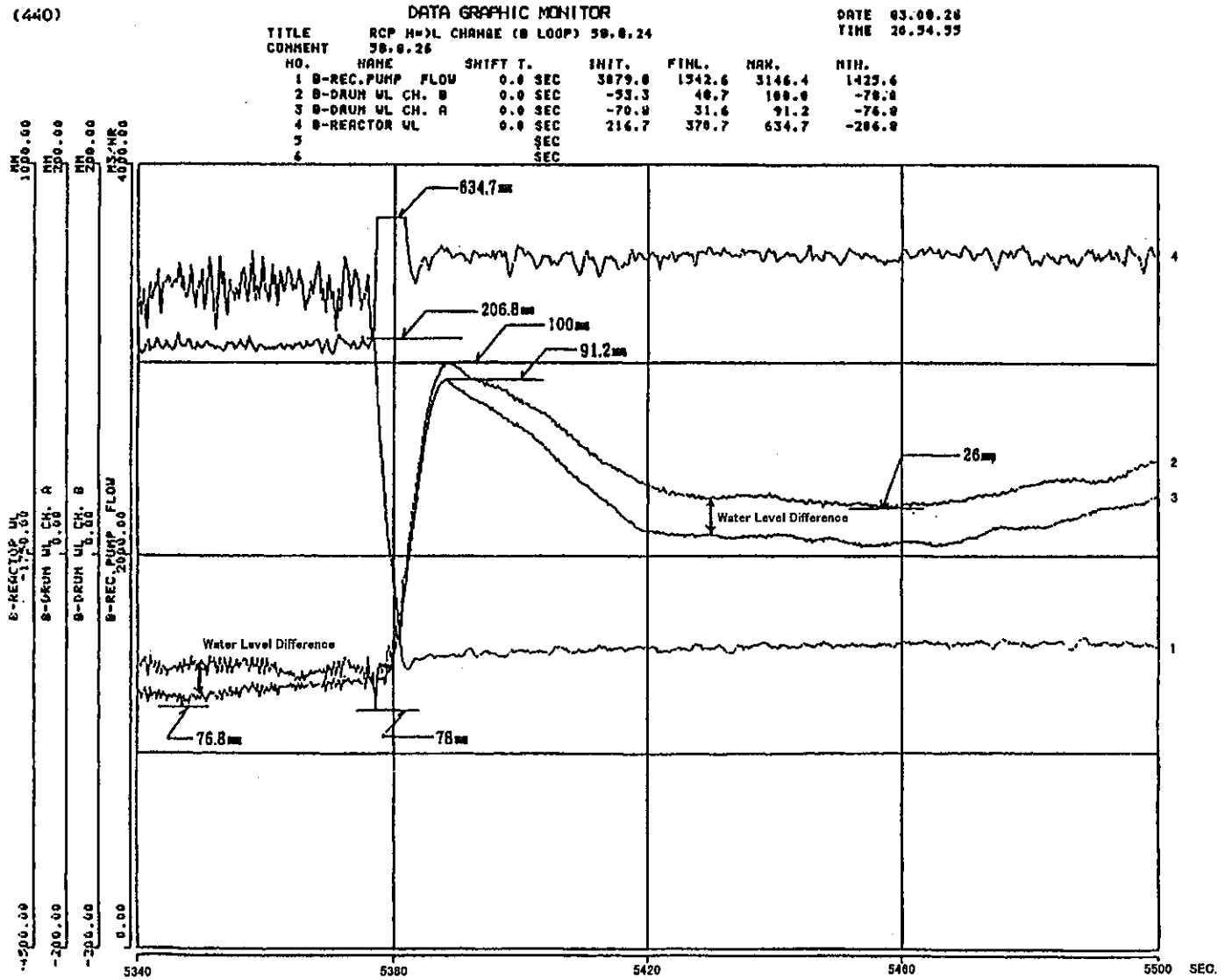


Fig.II.3-6 Changeover of Recircuration Pump Speed (High to Low)

Table II.3-1 Relations of Reactor Power to Feed Water Control System and Control Valve

Reactor Power (%)	0～18	18～30	over 30
Control Mode	Single-element control	Single-or three-element control	Three-element control
Low-flow control valve	Automatic	Automatic or totally closed	Totally closed
Main feed water control valve	Manually totally closed	Manual or automatic	Automatic

Table II.3-2 Conditions of Turbine Startup and Generator Synchronization

	Cold Start	Warm Start	Hot Start
Inner metal temperature of high-pressure first stage inlet	Less than 90℃	90～150℃	over 150℃
Selection of rpm rising rate	Low speed 120rpm	Medium speed 180rpm	High speed 360rpm
No-load operation after reaching rated rpm	About 1 hr	No	No
Initial load of generator	About 5 MWe	About 5 MWe	About 5 MWe
Initial load holding time	About 5 min.	About 5 min.	About 5 min.

4. Operations

To secure the safety and stability of the plant, operation monitoring, surveillance /check, periodic tests and operational work management are very important jobs. The operation shifts have made much effort to confirm that plant-composing equipment and systems are normally functioning and to find an anomaly as fast as possible.

Therefore, the operators perform field surveillance/check, 4 times a day (3 times a day for some areas) in addition to operation monitoring in the central control room to confirm that there is no anomaly in the field equipment /systems. For the engineered safety features, moreover, the integrity confirmation tests for normally standing-by equipment and stand-by main pumps and the functional tests for reactor protection systems have been periodically carried out.

As to daily works, since management of the plant operating systems has been centralized to the Power Generating Section, they have been performed on the basis of the shift leader's permission.

On the other hand, operational work management during periodic inspection, such as work coordination and safety measures (hereinafter referred to as "isolation") and test runs, is completely performed by the periodic inspection managemaent group. Issuance and reception of work sheets, planning of isolation, issuance of tags to physically confirm isolation (hereinafter referred to as "handling prohibition tag") and confirmation of the state of isolation are all controlled by computers.

4.1 Surveillance/Check

Surveillance/check are indispensable jobs to keep safe and stable operation, to find out anomalies of each equipment and system as early as possible to prevent accident from occurring.

Like in LWRs, the Article 9 of the "Regulations for Test Reactors" stipulates that surveillance of the reactor facilities must be performed once or more per day to check the following:

① Reactor cooling system

② Control rod driving system

③ Power supply, water supply, sanitary and air exhaust systems

In "Fugen", daily once-or-more surveillance/check is specified in its safety regulations as determined in the "Regulations for Test Reactors", and they are performed in accordance with the "Surveillance /Check Procedures".

Surveillance/check is classified into two categories; "normal" and "special".

"Normal surveillance/check" relates to equipment/systems existing in the places except high-dose rate and highly contaminated areas, and is carried out 4 times a day (3 times a day for some areas).

"Special surveillance/check" related to equipment/systems existing in the special access prohibited areas such as high-dose rate and highly contaminated areas, and is performed once per two weeks.

On the other hand, for the reactor building and so on to which no access is made during operation, television cameras are installed in the key positions for indirect monitoring.

Surveillance/check are performed for the equipment, regular routes and matters to be checked specified in the "Surveillance/Check Procedures".

The matters to be checked for the main equipment are shown in Table II 4-1.

The areas for surveillance/check are as follows:

① Reactor auxiliary building (including reactor building), and fuel storage pool building

② Turbine building, and waste treatment room (partly including the miscellaneous solid waste treatment facility)

③ Intermediate building (partly including the turbine building)

④ Outdoors(partly including the miscellaneous solid waste treatment facility)

The aims of surveillance/check are to confirm the integrity of equipment/systems and to find out an anomaly as fast as possible; for this purpose, besides visual appearance check, related key parameters are recorded and the results are compared with the standard values to judge the state of the equipment. Besides the above, about 1500 key operation parameters of "Fugen" are collected in the "data collector", and the data is periodically investigated. When a minor change is observed, it is fed back to the operation shift to strengthen monitoring of the related equipment and surveillance /check.

Table II .4-2 and Fig. II .4-1 show the number and ratio of detected troubles by surveillance/check.

4.2 Periodic Routine Tests

The aims of the periodic routine tests are to confirm the integrity of the functions of stand-by engineered safety features and stand-by equipment, and to confirm whether or not the functions of the interlocks and alarms for reactor protection system which must shut down the plant in safe in an emergency are normal.

The test items can be classified into two types; one is the items whose testing

contents and frequency are stipulated by the safety regulations such as the scram test for reactor protection systems and the pump running tests for engineered safety features, and the other is those for which tests are carried out independently in the plant such as the testing items relating to the turbine and electric devices. The testing frequency is shown as follows:

(1) The test items carried out during normal operation

① once a day ·····5% closing test for main steam isolation valve, turbine main valve inching test, etc.

② once a week ·····control rod actuation test, nuclear instrumentation test, etc.

③ once a month ·····manual pump startup test for engineered safety features, diesel generator manual startup test, etc.

④ once per 6 months ·····complete closing test for main steam isolation valve, injection water valve switching test for rapid water injection system, etc.

(2) The test items carried out during periodic inspection

once a year ··· automatic pump startup test for engineered safety features, automatic diesel startup load test, etc.

The periodic test items are shown in Table II 4-3.

The test items peculiar to the heavy water reactor "Fugen" include the heavy water dump circuit test, helium communicating valve switching test and stand-by heavy water recirculation pump running test. These test have been carried out once a month in accordance with the reactor protection system test and pump running test for engineered safety features.

Of the periodic tests, main tests have been performed under the attendance of the operation control competent officers dispatched from the Science and Technology Agency.

Table II 4-3 shows the main periodic test items and changeover of auxiliary equipment.

4.3 Work Management

Work management is, like plant operations, one of the important operation jobs.

In particular, works during operation may lead to plant shutdown due to insufficient investigation of isolation, etc. so that all works are controlled by the work sheets and performed after obtaining the permission from the shift leader. The contents of isolation investigated by the Section in charge are confirmed again by the Power

Generating Section, and only after being recognized that there is no problem, implementation of the job is permitted. The work sheet issued for works, isolation plan, handling prohibition, tag issuance, cancellation work sheet issuance, receipt of work/continuation/completion of work procedures and work report have been completely controlled by the computers since fiscal 1988.

Operation control is generally carried out during plant operation when the amount of maintenance is fairly small and during periodic inspection when the amount of maintenance is rather great.

Upon executing the job, the operator in charge of the job issues the "work sheet", taking into account the matters to be noted from the view points of radiation exposure control and safety.

Figure II4-2 shows the flow of work sheet procedures.

In the initial stage of operation, all work procedures were written by hand, but since fiscal 1988 the "maintenance control system" has been introduced and all procedures have been done by computers.

In the following subsections, work management during plant operation and during plant shutdown are described.

(1) Work Management during Plant Operation

Work management has been carried out by the General Manager of the Power Generating Section and shift leaders, regardless of during plant operation or plant shutdown. The Work Management Group of Power Generating Section mainly coordinates the contents of jobs and their duration and confirms isolation, whereas the operation shift understands the contents of everyday jobs in addition to confirmation and execution of isolation. Main jobs during plant operation are the isolation of maintenance and check of equipment that caused malfunction, and the number of these jobs is small.

For work management during plant operation, consideration of the effects of isolation on the plant, and implementation of labor safety measures for operations associated with electricity, high temperature and high pressure are important.

These are prepared by the Sections in charge of these jobs at the time of preparing the work sheets. In addition this the Work Management Group of the Power Generating Section and the operation shifts confirm the contents from the viewpoint of work management, and revise the contents, if necessary.

(2) Work Management during Periodic Inspection

There are many equipment and systems to be maintained and checked during the periodic inspection, and work management inevitably becomes complicated.

Therefore, since fiscal 1989 (the 8th periodic inspection) the Periodic Inspection Management Group consisting of 12-17 persons (including a leader and 3 sub leaders) has been organized for work management. The Group, under the guidance of the General Manager of Power Generating Section, makes centralized management on the job period, investigation and confirmation of safety-related matters, operation of equipment and systems, and isolation, restoration and test run of systems.

The organization of the Power Generating Section during periodic inspection is shown in Fig. II4-3.

The operation system of the Power Generating Section was Changed to 4 groups - 3 shift - a day service shift system in April, 1985. Work management was also reviewed, and investigation of isolation, etc. were changed to be done in the day service and operation shifts.

So far, the isolation handling prohibition tag was put on each job basis. In order to mitigate the load imposed on the operators due to overlapping of handling-prohibited jobs and to centralize work management, "centralized isolation" management system has been introduced since fiscal 1985. This system has made the jobs rational and effective, and work management has become smooth.

Meanwhile, introduction of "maintenance control system" is effective like periodic inspection where the number of jobs is great and complicated management is required.

The composition of the operators and the tendency of becoming full- time job for refueling made investigation, execution and restoration of isolation in the operation shifts a little difficult, so that the work management has been performed, as shown above, by the Periodic Inspection Management Group since the 8th periodic inspection in 1989.

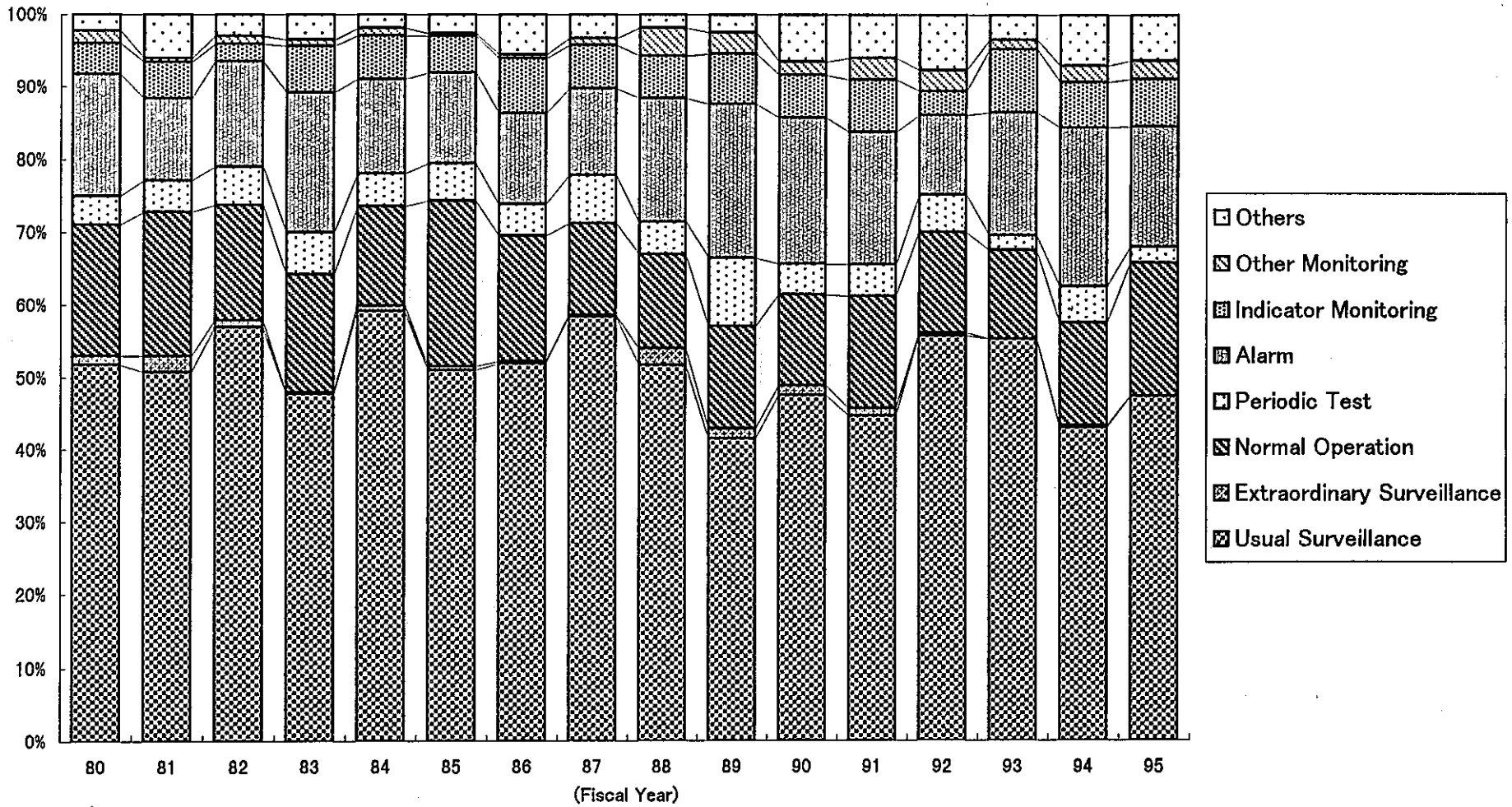


Fig.II.4-1 Histogram of Troubles on Basis of Detection Means

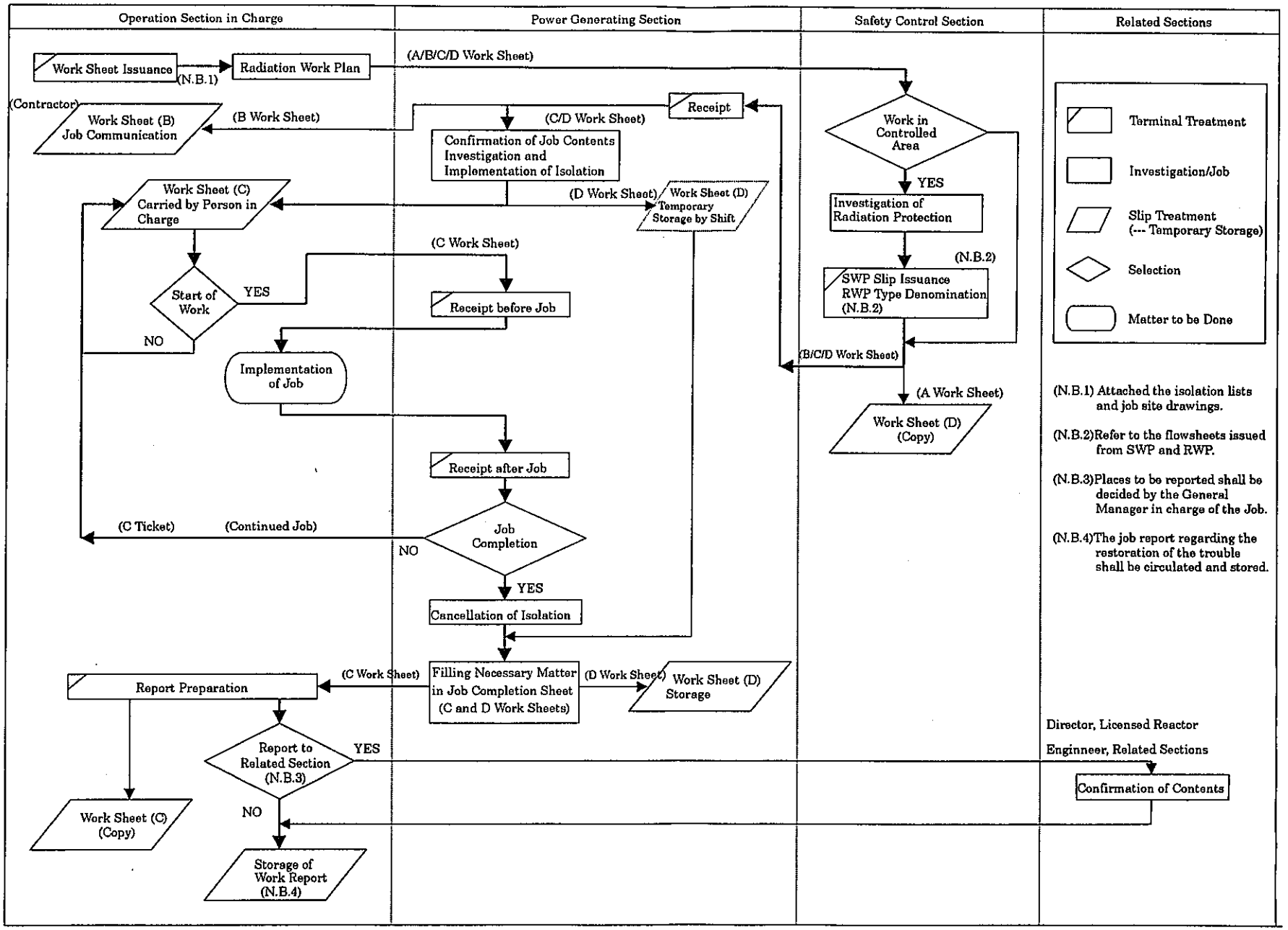


Fig.II.4-2 "Work Sheet" Process Flowsheet

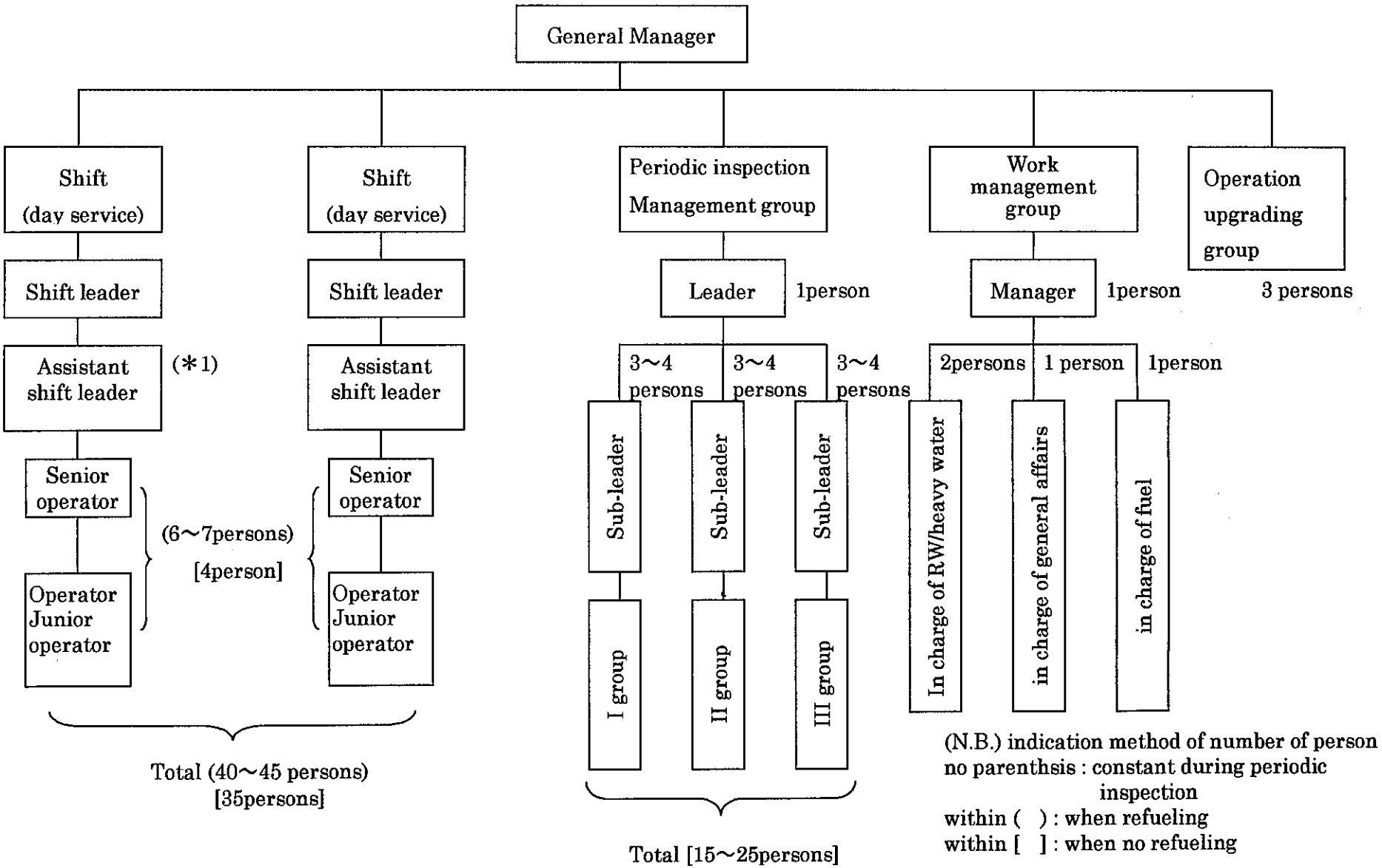


Fig.II.4-3 Organization of Power Generating Section During Periodic Inspection

Table II.4-1 Inspection Items for Main Installations

Installation	Inspection Item
Rotating machine	1. Abnormal noise, smell, vibration, temperature 2. Level of lubrication oil 3. State of belt (fan, compressor, etc.) 4. Cooling water flow rate 5. Seal water flow rate 6. Leak from seals and flanges 7. Appearance (fouling, cracks) 8. Process parameters (temperature, pressure, flow rate, etc.)
Piping, Duct	1. Leak 2. Interference 3. Anomaly of supporting structure
Valve	1. Opening and closing positions of valve 2. Leak from gland and bonnet 3. Sheet leak of drain and vent valves 4. Oil leak from electric valve & gear box 5. Leak from air-actuated valve and air piping 6. Safety valve: seat leak
Straighter, Filter	1. Differential pressure
Draintrap	1. Confirmation of operation by checking downstream temperature and inside of funnel
Heat exchanger	1. Leak 2. Differential pressure 3. Inlet & outlet temperatures
Tank	1. Leak 2. Liquid level of tank 3. Liquid level control valve 4. Cumulative supply water flow meter
Water purification system	1. State of flowing operation 2. Alarm window 3. Differential pressures, flow rates and electric conductivities of inlets and outlets of each filter 4. Differential pressure of outlet strainer 5. Electric conductivities of inlet & outlet of header 6. Leak

Installation	Inspection Item
Drain Sump	1. Floating substance, such as dusts 2. Oil & grease
Drain fannel	1. Water seal 2. Plugging by foreign substance
Air compressor piping	1. Leak at connecting part of devices 2. Moisture inclusion (state of actuation of drain trap)
Breaker control center etc.	1. Abnormal noise & smell 2. Water leakage 3. Display of operation or trip 4. Instrument, relay 5. Fouling
Transformer	1. Temperature 2. Oil surface 3. Operational state of oil pump, blower 4. Abnormal noise, smell, vibration, oil leak
Armature brush	1. Abrasion of brush 2. Spark generation 3. Carbon dust adhesion
Field control panel	1. Display of instruments 2. Operation display lamp 3. Alarm window
Instrumen-tation rack	1. Display of instruments 2. State of instrumentation valve 3. Leak

Table II.4-2 Number of Troubles on Detection Means Basis (in Power Generating Section)

as of the end of fiscal 1995

Motive \ Fiscal Year	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	Total
Usual Surveillance	429	406	426	384	475	397	338	362	265	155	184	135	153	82	130	129	4450
Extraordinary Surveillanc	10	18	7	2	6	4	2	1	12	5	5	3	1	0	1	0	77
Normal Operation	149	158	118	130	109	177	113	78	66	53	49	47	38	18	43	50	1396
Periodic Test	33	35	39	47	36	40	28	41	23	35	16	13	14	3	15	6	424
Alarm	139	90	109	154	105	98	81	74	87	79	78	55	30	25	66	45	1315
Indicator Monitoring	35	41	18	52	48	39	50	37	30	26	23	22	9	13	19	18	480
Other Monitoring	15	4	8	7	8	3	3	6	20	11	7	9	8	2	7	7	125
Others	18	48	22	28	15	20	36	20	9	9	25	18	21	5	21	17	332
Total	828	800	747	804	802	778	651	619	512	373	387	302	274	148	302	272	8599

Table II.4-3 Periodic Tests (1/3)

Frequency	Number	Item
Once a Day	R-101	Main steam isolation valve 5% closing test
Once a Week	R-203	Control rod actuation test
	E-203	115V pilot battery check
	E-204	±24V instrumentation plot battery check
Once a Month	R-306	Annulus exhaust air system emergency filler unit running test
	R-307	High-pressure water injection system manual startup test
	R-308	High-pressure water injection system valve switching test
	R-309	Low-pressure water injection system pump manual startup test
	R-310	Low-pressure water injection system valve switching test
	R-312	Isolation cooling system pump manual startup test
	R-313	Isolation cooling water system valve switching test
	R-314	Residual heat removal system pump running test
	R-315	Residual heat removal system feed water pump running test
	R-316	Steam discharge pool cooling system pump manual startup test
	R-317	Steam discharge pool cooling system valve switching test
	R-318	Containment vessel spray system pump manual startup test
	R-319	Containment vessel spray system valve switching test
	R-321	Reactor protection system test
	E-303	Diesel generator manual startup test
	E-304	Diesel generator auxiliary equipment test

Frequency	Number	Item
Once / 3 Months	E-401	Total inspection of 115V batteries
	E-402	Total inspection of ±24V instrumentation batteries
Once / 6 Months	R-501	Main Steam Isolation Valve Complete Closing Test *1
	R-502	Rapid water injection system pressure accumulator valve switching test *2
		*1 : Execution at plant startup
		*2 : Execution at plant shutdown
Once a Year		

Table II.4-3 Periodic Tests (2/3)

Frequency	Number	Item
Once a Day	T-101	Turbine main valve inching test
	T-102	Auxiliary boiler drum water level meter check
	E-101	Alarm test in control rod panel
	E-101	Alarm test in control rod panel (field panel)
Once a Week	R-201	Seal leak check
	R-202	RCWS strainer electric valve actuation test
	R-204	SUM/PUM driving test
	R-205	SUM test
	R-206	PUM test
	R-207	PRM test
	R-208	TPM test
	R-209	RSM test
	T-201	Turbine auxiliary equipment cooling sea water pump running test
	T-202	Turbine control oil/lubrication oil system pump automatic start-up test
	T-203	Alarm test on abnormal oil level of main oil tank
	T-204	turbine rotor potential measurement test
	T-205	Turbine safety circuit test
	T-206	Turbine safety device test
	T-207	Control air device test
	T-208	Control air device drain check
	T-209	Miscellaneous air device drain check
	E-201	Seal oil differential pressure alarm test
	E-202	Emergency seal oil pump automatic startup test

Frequency	Number	Item
Twice a Month	—	Special patrol (1/2 shift)
	—	Patrol car washing and check
	—	Condemi resin a scrabbing (after 8th scrubbing →M column)
	—	Condemi resin regeneration (after regeneration →M column)
	—	Condemi resin replacement (M column →Condensate polisher)
Once a month	R-301	C-heavy water circulation pump running test
	R-302	Helium communication valve switching test
	R-303	Reactor auxiliary equipment cooling water pump running test
	R-305	Containment vessel air circulation system C-brine recirculation pump running test
	R-311	Rapid water infection system pressure accumulator feed pump running test
	R-302	LPM test
	T-301	Startup vacuum pump start-up test
	T-302	Actuation test of recirculation pump lubrication water back up magnetic valve
	T-303	Motor driven fire extinguishing pump/engine fire extinguishing pump manual startup test
	T-304	Auxiliary boiler soot blower
	T-305	Ventilation system compressor/dehumidifier manual startup test
	T-306	drain check of H_2 /O_2 generator densitometer drain check
	E-301	Seal oil treatment system vacuum pump running test
	E-302	Check of generator brush abrasion
	E-305	Diesel generator sea water piping vacuum break valve actuation test
	—	LPM calibration (Maintenance Section)
	—	Check of tools, etc. for exclusive use of Power Generating Section, check of goods in warehouse

Frequency	Number	Item
Once per 2 months (Feb., Apr., June, Aug., Oct., Dec.)	R-401	Fuel handling equipment running test
	R-402	Fuel transfer equipment running test
	T-402	Disposer/wastes treatment room drain pump running test
Once per 3 months (Jan., Apr., July, Oct.)	T-401	Auxiliary boiler safety test
Once per 6 monthes	T-501	Main steam stop valve complete closing test
	T-502	Steam control valve complete closing test
	T-503	Combined intermediate valve complete closing test
	T-504	Turbine bypass valve complete open test
Once a year	R-602	High-pressure water injection system valve automatic switching test
	R-604	Low-pressure water injection system automatic switching test
	R-608	Isolation cooling system automatic switching test
	R-609	Isolation cooling system double piping breaking test
	R-612	Containment vessel spray system valve automatic switching test
	T-601	Turbine overspeed trip test
	T-602	Turbine backup overspeed trip test

Table II.4-3 Periodic Tests (3/3)

Changeover of auxiliary equipment

Frequency	Number	Item
3 times a month	228	H_2/O_2 generator circulating liquid pump
Twice a amonths	228	H_2/O_2 generator cooling water tower pump
Once a month	44	Annulus exhauster
	67	Noble gas refrigerator
	84	Portable steam exhauster
	90	high-pressure oil pump
	105	Phase separation bus cooling fan
	106	Main transformer cooling fan (odd ←→even number)
	106	In-plant transformer cooling fan (1 ←→ 2 groups)
	106	Power transformer cooling fan (working ←→standby)
	106	Startup transformer cooling fan (1 ←→ 2 groups)
	156	Auxiliary boiler feed water strainer change over / cleaning
	204	Chilling unit (No.1←→No.2)
	228	H_2/O_2 generator H_2Compressor
Once per 2 months	35	He purification system refrigerator
	43	Instrumentation system temperature adjusting unit blower
	61	R/B equipment drain sump pump
	61	R/B floor drain sump pump
	61	A/B equipment drain sump pump
	61	A/B floor drain sump pump
	61	T/B equipment drain sump pump
	61	T/B floor drain sump pump
	61	RW/B equipment drain sump pump
	61	RW/B floor drain sump pump
	61	New RW/B equipment drain sump pump
	61	New RW/B floor drain sump pump

Frequency	Number	Item
Once per 2 months	67	Rare earth equipment drain sample pump
	67	Rare earth floor drain sample pump
	84	Condensate transfer pump
	151	Raw/Miscellaneous water system pumps
	152	Demineralized water transfer pump
	154	Control air compressor
	155	Miscellaneous air compressor
	158	Sea water cleaning pump
	167	CO_2 Supplier carburetor heater
	202	Ventilating system for fuel storage pool building
	203	Ventilation system for waste treatment building
	204	Ventilation system for control room
	205	Ventilation system for reactor auxiliary building
	207	Ventilation system for battery room
	208	Ventilation system for health physics room
	209	Turbine building ventilation system
	211	Local cooling system cooling water pump
	228	Demineralized water pump for H_2/O_2 generator
Once per 3 months	18	Shield cooling system recirculation pump
	103	Seal oil treatment system vacuum pump
	156	Auxiliary boiler
	158	Sea water cleaning pump strainer change over

Table II.4-4 Change in Main Periodic Test Cycle

No.	Testing Item	Contents of Change	Reason of Change	Date after Change
1	Completely Closing Test for Main Steam Isolation Valve	1／3M → 1／6M 1／6M → 1／Y	The changes were done on the basis of good operating result since the start of rated power operation (Mar., 1979) and the testing frequency of LWRs.	Jan., 1983 Feb., 1991
2	Completely Closing Test for ・Main Steam Stop Valve ・Steam Control Valve ・Turbine Bypass Valve	1／3M → 1／6M	The test is carried out under the same conditions as those in the completely closing test for the main steam isolation valve. As the frequency of test for the main steam isolation valve was changed, review was made and the frequency of the test was changed because the reliability of the valve was considered to be ensured due to ①good operation results and ② everyday implementation of inching tests.	Apr., 1983
3	No-Load Operation Test for Diesel Generator	1／W → Cancel	Because 2 years have passed since the start of full power operation and operation results have been good, no-load operation test once a week was canceled, and load test once a month has started	Dec., 1984
4	Completely Closing Test for Turbine Combination Intermediate Valve	1／W → 1／6M	The rather great influence of this test on the plant (excessive fluctuation) leads to make review. As a result, the test frequency was changed because the reliability of the valve was considered to be ensured in accordance with ① good operation results obtained hitherto and② everyday implementation of inching tests.	Jun., 1989

5. Education and Training [II-14]

5.1 Basic Policy

Except that "Fugen" is a heavy water-moderated pressure-tube-type reactor, it has basically the same system structure as those of BWRs, and its operational characteristics are very close to those of BWRs. Education and training of the "Fugen" operators, therefore, have been carried out in the same way for the precedent BWRs.

However, as "Fugen" is the first heavy water moderated power reactor in Japan, the main member of initial-stage operation shift leaders were trained in the heavy water reactor in Canada (13 persons in the Reactor Training Center of Ontario Hydoro and in NPD from 1974 to 1975) and learned field techniques in the domestic thermal power plants.

As to education and training, desk-type education and OJT (On-the-Job Training) have been basically focussed, with training in the external training organizations such as BWR Operation Training Center (BTC), JAPCO and JAERI.

This basic policy has not been changed yet, but the contents of education and training will be reviewed, taking into account the operation system, changes in composing operators and social environment.

The education and training systems of "Fugen" are shown in Table II 5-1.

The results of education and training after March, 1979 are described in the following subsection.

5.2 Contents of Education and Training

(1) In-plant Education and Training

In-plant education and training center on desk education, which is shown in Table II 5-2.

The contents of education shown above are carried out in accordance with the schedules ("Basic Operator Training Plan") determined for each level operators.

(a) Education of Fresh Personnel

When fresh personnel are nominated as operators, they are educated regarding the outline of "Fugen", related laws and regulations, basic knowledge on radiation and basic action of operators.

The contents of education for fresh personnel are completely reviewed in April, 1987.

The following items were added to the education program.

① Execution of Training in Other Sections

The fresh personnel are dispatched to other Sections of "Fugen" so as to understand the jobs performed in relating Sections of "Fugen", and thereby deeping the knowledge on the organization of "Fugen".

② Execution of Basic Education

In addition to the education regarding the facilities of "Fugen", the textbooks relating to basic knowledge of nuclear power are distributed to them to support their self-study on wide knowledge of nuclear power.

③ Execution of Briefing Session

At the end of the training period (on year) a briefing session is held to bring their one-year experience to conclusion.

(b) Education for Junior Operators (System Education (I) and (II))

This type of education is carried out within 2 years after assignment, and the contents are related to the basic matters of the power generating plant (motors, pumps, valves etc.), but have changed since fiscal 1981 to the system-related knowledge of "Fugen", interlocks, operation techniques, etc.

(c) Education for (Middle-class) Operators

This education is given basically to the junior operators before they are graded up to the (middle-class) operators (4 years after assignment). It includes the main control systems of "Fugen" (such as reactor control, feed water control, turbine control, etc.).

(d) Education for Senior Operators

This education is given basically to the (middle class) operators before they are graded up to the senior operators (7 years after assignment). Initially, the education was made on cultivating the comprehensive knowledge of the plant and on training circumstantial judgment ability (design criteria of ECCS, safety analysis and circumstantial judgment at the time of emergency), but the operation management (core management, related laws and regulations, safety regulations, safety analysis, chemistry control, radiological safety management, process computers, etc.) have been added since fiscal 1986.

(e) Field Education of Heavy Water and Radioactive Wastes

Education of these fields by OJT became difficult after the operation of the radioactive wastes treatment system and heavy water purification system had been entrusted to the external company, and so it was decided to do it by the separate training.

Its contents are as follows:

① Modification histories and reasons for modified installations and parts

② Notices on operation and operational limits

③ Basic interlocks

④ Encountered and experienced troubles since commissioning (plant functional test and startup test)

(f) Simulator Training for Upgrading ATR Operation [II-15]

The simulator is compact simulator for study.

The results of in-plant education are shown in Table Ⅱ5-5 and Fig. Ⅱ5-1.

(g) In-group Education

Each group establishes education programs peculiar to them, and they have executed the system education, study of structures of equipment, KY training (training on risk prediction), first aid training and fire fighting training.

Moreover, "Emergency operation simulated training" (reactor scram / turbine trip accident, loss of external power supply accident, loss of coolant accident, etc.) is carried out once a month. This training is performed under the attendance of the Licensed Reactor Engineer twice a year, and his criticisms are reflected on the later simulated training.

(2) In-plant Education

The results of in-PNC education are shown in Table Ⅱ5-6. To supplement the in-plant education, operators are dispatched to PNC's Tokai Works, mainly for training to educate them safety and to improve their capabilities.

(3) Simulator Training in BTC

In "Fugen", as a part of education and training of the operators, the operation techniques and quality of the operators are improved by using BTC. This training consists of the following 6 courses:

(a) Initial Training Course (6 weeks)

This course enables operators to learn effectively the basic knowledge and techniques, and so the operators are dispatched to this course intentionally. The course consists of 3 phases; from the phase 1 to 3. Persons dispatched can take part in any phase depending on their levels to learn plant installations /operation knowledge, normal operations and operations in an emergency.

The operators to be dispatched are middle/senior operators 5 years after entering PNC (senior high school graduates).

(b) Retraining II Course (8 days)

This course gives retraining of operation techniques. As this course is considered to be necessary for retraining of the techniques and knowledge, operators are intentionally dispatched to take the course.

Those who are dispatched consist of middle/senior operators, and they are dispatched 2-3 years after finishing the standard or the initial training course.

(c) Retraining III Course (8 days)

This course is for maintaining and improving the comprehensive abilities including knowledge/technique necessary for operation and instructive ability for lower operators. The curriculum includes EPG (Emergency Procedures Guide) training.

Those who are dispatched are senior operators or assistant shift leaders who finished the retraining II course and EPG training. They are dispatched to this course 3-4 years after finishing the above-mentioned training.

(d) Senior Operator Training Course (6days)

As the course provides education and training items necessary for shift leaders and assistant shift leaders, such as necessary operations, judgment of state at the time of emergency and training of instructive ability for operators, the operators are intentionally dispatched to take the course. Those who are dispatched are shift leaders or assistant shift leaders who finished EPG training (This course has included EPG training since fiscal 1994).

(e) Retraining Refresh Course (3 days)

In this training course, operation techniques are refreshed by the simulator training. Those who are trained are senior operators who will be dispatched 3-4 years after finishing the retraining III course.

(f) EPG Training Course (2 days)

The aim of this course is to train emergency operating procedures to improve judgement ability at the time of emergency. This course is given in addition to the retraining II course, as a rule.

Those who finished the retraining III or senior operator training course must finish this training as a rule.

(g) Family Training Course (3 days)

Since this course is effective for team cooperated operation, etc., and dispatch of operators is done intentionally on a shift basis with improving the contents of training; namely, once a year x 5 groups, and those who are dispatched on a group basis are about 12 persons/group with the shift leader being the leader.

To make the team training in this course satisfactory, the shift leader and the BTC side must sufficiently negotiate the plan in advance.

(h) Special Training Course (since fiscal 1994) (6 days)

This course is given for a team consisting of junior operators having 2-3 years operation experience to provide them with knowledge/operation techniques on basic operations responding to emergency, and thereby maintaining and improving comprehensive abilities of the operation shifts as a whole.

The training period is 6 days (4 days simulator and 2 days lecture), and training is held once a year, dispatching a team of about 8 operators.

To perform the satisfactory team training in this course, the training program must be closely negotiated in advance with the BTC side.

The contents of BTC training and the results of BTC training are shown in Table II 5-7 and Fig. II 5-2, respectively. The total number of operators dispatched to BTC training by the end of fiscal 1995 is about 780.

(4) Basic Reactor Training in JAPCO and JAERI

The operators who were employed 3-6 years ago have been dispatched to JAPCO or JAERI to study the general knowledge on nuclear energy and special knowledge on nuclear reactors.

So far, 59 persons have been dispatched to JAPCO and 8 persons to JAERI. The results of training are shown in Table II 5-8.

(a) Reactor Training in JAPCO

This is a training course to study the basic knowledge on nuclear energy as a whole.

(b) Reactor Training in JAERI

This is a training course to study the knowledge of more advanced nuclear engineering for the operators having the basic knowledge on nuclear energy.

(5) Short Training Course in Makers

Besides the above-mentioned education and training, "Fugen" has dispatched operators to short training courses sponsored by equipment makers to let them be trained on the structures of equipment and the principles of instruments, necessary for operation. The results of training are shown in Table II 5-9.

(6) Short Training Courses for Obtaining Qualifications

Not only to secure the legally necessary number of superintendents and leaders to maintain, to control and to operate equipment and systems but also to let operators make self-enlightenment, "Fugen" has positively made operators take

short training courses and tests for obtaining qualification regarding plant operation.

(7) Utilization of "Education Pocket Book for Operators"

The utilization of "Education Pocket Book for Operators" is started to record the results of the above education and training of each operator and to check their progress. The "Education Pocket Book for Operators" is so made that the state of desk education on a system basis and the state of study of handling methods (periodic tests, normal operations, alarm responding procedures, etc.) can be recorded, thereby clarifying the matters to be studied for each operator class in each year.

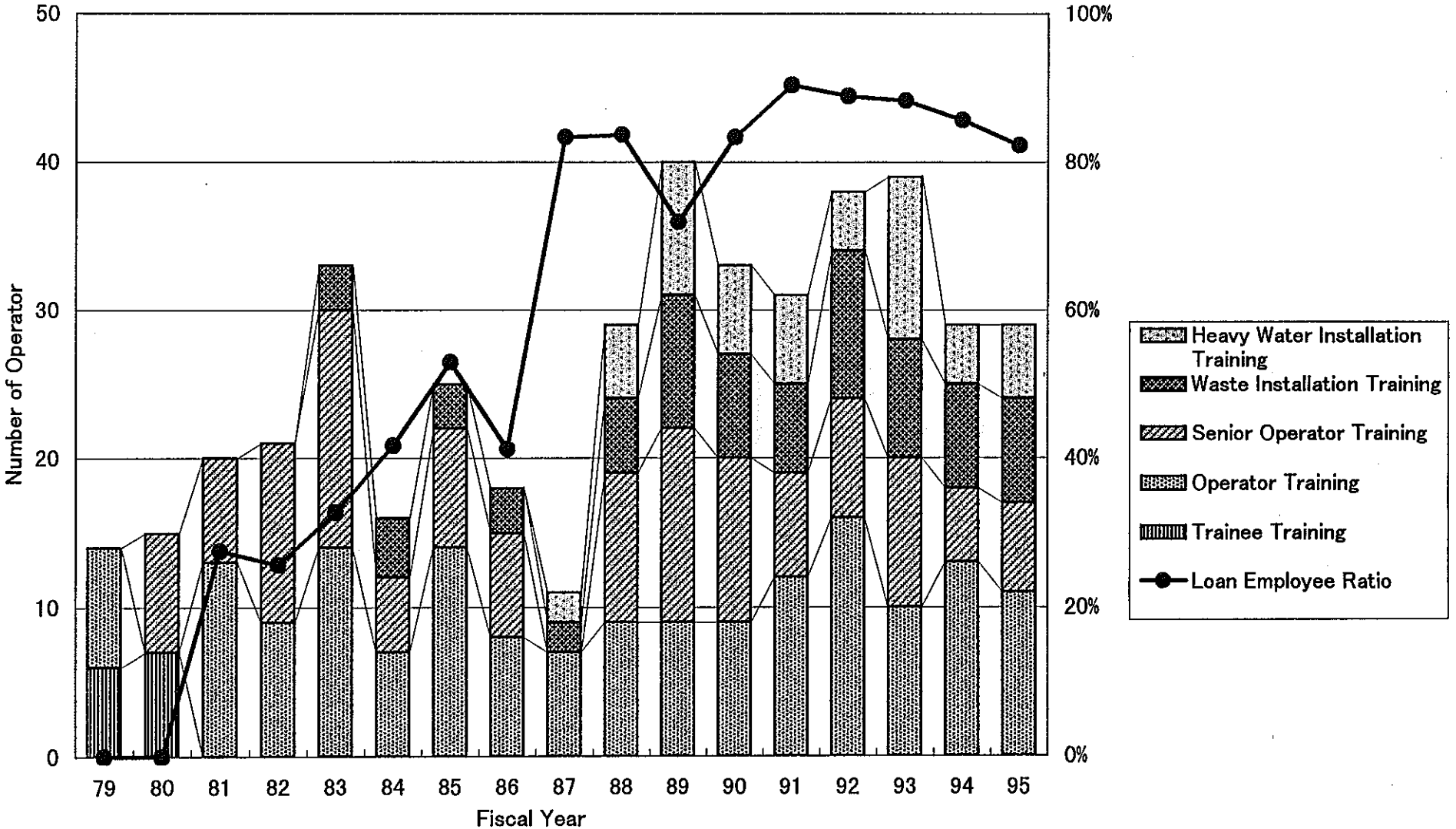


Fig.II.5-1 Change in Results of Desk Training on Fiscal Year Basis

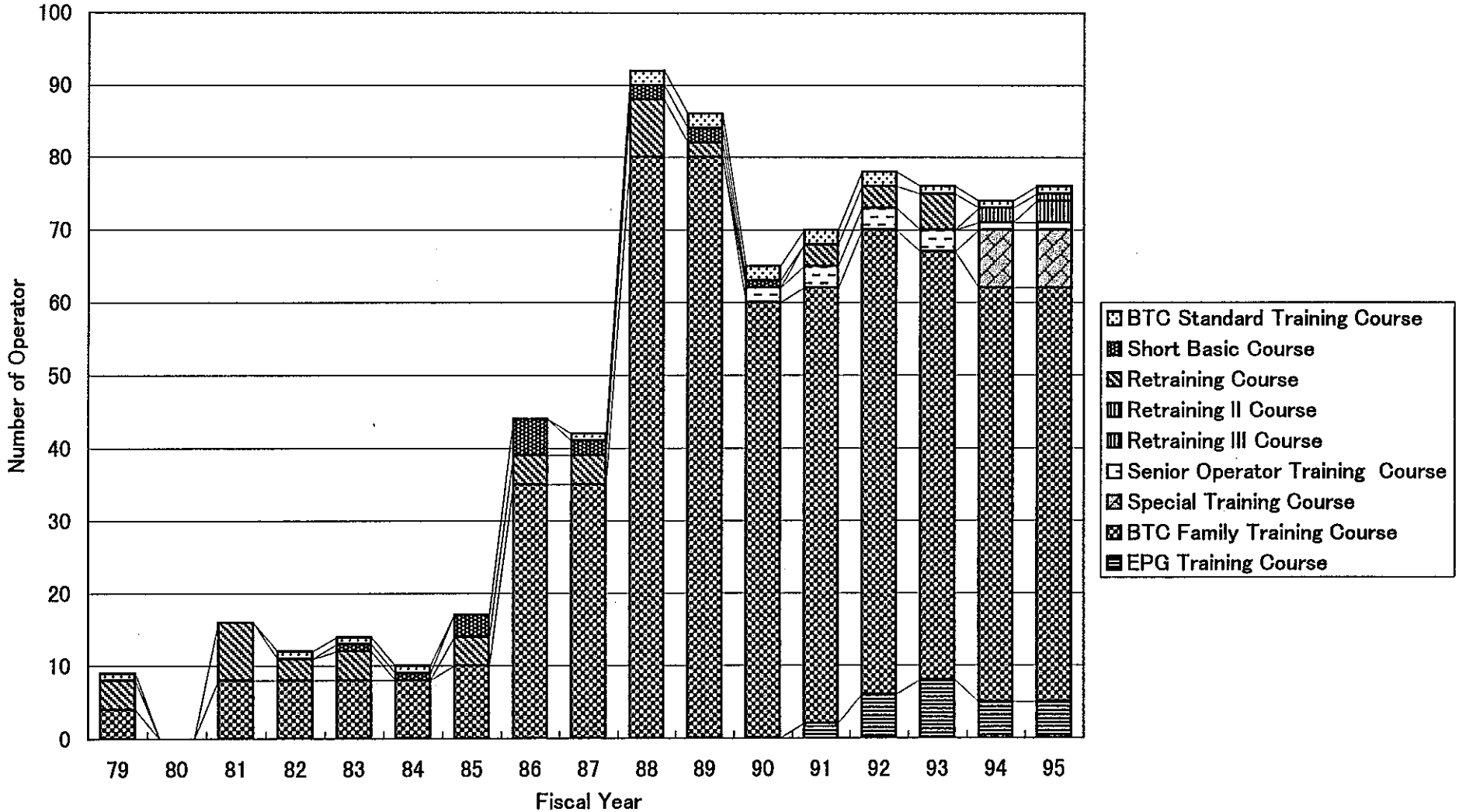


Fig.II.5-2 Change in Results of BTC Training on Fiscal Year Basis

Table II.5-1 Training Course for "Fugen" Operators

as of fiscal 1990

Classification	Experience (year)	In-Section Training			External Training		
		On the Job Training	Desk Training	Emergency Operation Training	Training in JAERI & JAPCO	BTC Training	Others
Trainee	0	(1)Outline of equipment and system (2)Layout of field and equipment (3)Patrol exercise (4)Operational exercise of auxiliary equipment	(1)Introductory training (about a week) ・Attitude of operators ・Surveillance/check procedures (2)Junior operator training ・System (I) ・System (II) (10 days for each)	(1)Study of emergency operating procedure			(1)Basic freshman training ・Self study & report ・Training in other Sections ・Result reporting meeting
Junior Operator	1 | 3	(1)Operation of field equipment and system (2)Patrol (3)Simple periodic test (4)Simple isolation (5)Training in waste treatment	(3)Operator training ・Control system training (5 days)	(1)Confirmation of emergency operating procedure (2)Simulated emergency operation training (3)Discussion and reflection of training results		(1)Family training course	
Operator	4 | 6	(1)Operation of equipment and system (2)Operation of plant startup and shutdown (3)Periodic test (4)Isolation in periodic test (5)Training on heavy water purification	(4)Senior operator training ・Operation control training (5 days)	・Simulated emergency operation items ①Reactor scram/turbine trip accidents ②Loss of external power supply accident ③Loss of coolant accident ④Fuel handling accident ⑤Heavy water/helium accident ⑥Others	(1)JAERI's general reactor course (5 months) (2)JAPCO's training (3 months)	(2)Standard training course (12 weeks) (3)Short-term basic course (4 weeks)	(1)Safety Training Center of PNC's Tokai Works (2)Technical training in equipment manufactures
Senior Operator	over 7	(1)Operation of plant startup and shutdown, and its instraction (2)Operation of emergency operating procedures, and its instruction (3)Operating work management (4)Operation control				(1)Retraining course (10 days) (2)Senior operator training course (5 days)	

Table II.5-2 Contents of Desk Training

as of end of 1989

	Object	Contents of Training	Training Period
Education for Trainee	Fresh shift operator	Study of basic matters as an operator of "Fugen"s Power Generating Section ・Safety regulation ・Radiation control ・Nuclear power in general ・Outline of ATR system	5.5 days
System Training (I)	Experience; less than a year	Study of system, interlock and operation methods ・Turbine system ・Generator system ・Electricity system ・Diesel generator system ・Other systems (ventilation, heavy water purification, H_2O/O_2 generating system, raw/demineralized/miscellaneous water systems, auxiliary steam system)	10 days
System Training (II)	Experience; a year	Study of systems, interlock and operation methods ・Reactor proper ・Reactor cooling system ・Heavy water cooling/purification system ・Engineered safety features ・Fuel handling system ・Reactor auxiliary equipment cooling system ・Reactor containment vessel system ・Neutron measurement device ・Feed water control device ・Radiation monitoring device	10 days
Operator training	Experience; 3 years	Study of control systems ・Reactor control system ・Reactor protection system ・Feed water control system ・Turbine control system (EHC) ・Generator control system ・Radiation monitoring system ・Reactor physics	5 days
Senior operator training	Experience; 6 years	Study of operation control techniques ・Core characteristics of ATR ・Safety examination ・Plant safety analysis, dynamic characteristics ・Calculation of core performance ・Process computers ・Nuclear-power-related laws, safety regulations, operation control manual, etc. ・Large computer system ・Radiation control ・Chemistry control ・Discussion on accident examples	5 days

Table II.5-3 Documents for Education and Training

(prepared in 1976)

Basic Document (I)	Basic Document (II)
・Reactor in general (reactor physics, heat transfer) ・Reactor in general (materials, fuels) ・Radiation protection ・Electricity in general	・Chemistry in general ・Machinery in general ・Instrumentation/control ・Reading/writing of drawings ①Reading of sequence ②Reading of interbloc diagram ③Reading of P&I
System (I)	System (II)
・Reactors (10-38, 70-77)	・Electricity (100-118, 143)
System (III)	System (IV)
・Turbine (66, 67, 81-91) (151-169) ・Instrumentation (131-133, 135, 136, 138, 139)	・Refueling machine (19, 50-56, 174, 175, 181, 189, 192) ・Air condition system (41, 43, 44) (200-211, 213) ・Waste treatment system (61-65, 68)

Table II.5-4 Documents for Operators

(prepared in 1982)

Control-related Training Documents
・Reactor control system ・Reactor protection system ・Feed water control system ・Turbine control system ・Generator control system ・Reactor physics

Table II.5-5 Results of Desk Training

as of end of fiscal 1995

	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	Total
Trainee Training	6	7																13
	(3)	(3)																(6)
Trainee Training (I)				7	5	3	5											20
					(2)		(4)											(6)
System Training (I)								8	7	10	8	16	25	17	19	17	17	144
								(3)	(7)	(8)	(7)	(15)	(22)	(16)	(17)	(15)	(14)	(124)
Trainee Training (II)			9	7	14	5	4											39
			(3)	(4)	(5)	(2)	(2)											(16)
System Training (II)								8	6	10	9	17	27	17	19	17	16	146
								(3)	(6)	(8)	(8)	(16)	(24)	(16)	(17)	(15)	(13)	(126)
Operator Training	8		13	9	14	7	14	8	7	9	9	9	12	16	10	13	11	169
	(2)		(3)	(2)	(7)	(5)	(7)	(5)	(7)	(8)	(7)	(8)	(10)	(14)	(9)	(11)	(9)	(114)
Senior Operator Training		8	7	12	16	5	8	7		10	13	11	7	8	10	5	6	133
		(4)	(2)	(3)	(3)	(3)	(5)	(3)		(7)	(9)	(8)	(7)	(6)	(9)	(4)	(5)	(78)
Subtotal	14	15	29	35	49	20	31	31	20	39	39	53	71	58	58	52	50	664
	(5)	(7)	(8)	(9)	(17)	(10)	(18)	(14)	(20)	(31)	(31)	(47)	(63)	(52)	(52)	(45)	(41)	(470)
Waste Installation Training					3	4	3	3	2	5	9	7	6	10	8	7	7	74
										(5)	(5)	(4)	(6)	(8)	(6)	(7)	(5)	(46)
Heavy Water Installation Training									2	5	9	6	6	4	11	4	5	52
										(5)	(5)	(4)	(6)	(4)	(10)	(2)	(5)	(41)
Total	14	15	29	35	52	24	34	34	24	49	57	66	83	72	77	63	62	790
	(5)	(7)	(8)	(9)	(17)	(10)	(18)	(14)	(20)	(41)	(41)	(55)	(75)	(64)	(68)	(54)	(51)	(557)
Loan Employee Ratio	35.7%	46.7%	27.6%	25.7%	32.7%	41.7%	52.9%	41.2%	83.3%	83.7%	71.9%	83.3%	90.4%	88.9%	88.3%	85.7%	82.3%	70.5%

( ):number of loan employees

Table II.5-6 Results of Nuclear Fuel Cycle Engineering Training

as of end of fiscal 1995

	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	Total
Safety Education for Supervisor			1		4 (3)	4 (1)	4	12 (1)	3	2	3 (1)	2	4	3	4	1		47 (6)
Radiation Protection Course										3 (3)	2 (1)	1 (1)		1 (1)	2 (2)			9 (8)
Basic Course for Nuclear Fuel Technology			2	1	1						1	2 (1)	1 (1)					8 (2)
Retraining of Radiation Operator					3 (2)		3 (2)	4 (2)	1 (1)		2 (1)	1	1		1			16 (8)
Radiation Measurement Course					1 (1)	2		2 (2)	1 (1)	1 (1)	3 (2)		3 (1)	3 (2)	2 (2)			18 (12)
PSA Course											1	2 (1)	3 (1)	3 (2)				9 (4)
Computer Course											1	2	4					7 ()
Total			3	1	9 (6)	6 (1)	7 (2)	18 (5)	5 (2)	6 (4)	13 (5)	10 (3)	16 (3)	10 (5)	9 (4)	1 ()	() ()	114 (40)

( ):number of loan employees

Table II.5-7 Results of Training in BTC

as of end of fiscal 1995

	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	Total
BTC Standard Training Course	1			1	1	1			1	2	2	2	2	2	1	1	1	18 (0)
Short Basic Course					1	1	3	5	2	2	2	1						17 (0)
Retraining Course	4		8	3	4		4	4	4	8	2		3	3	5			52 (0)
Retraining II Course																2	1	3 (0)
Retraining III Course																	3	3 (0)
Senior Operator Training Course												2	3	3	3	1	1	13 (0)
Special Training Course																8 (7)	8 (6)	16 (13)
BTC Family Training Course	4		8	8 (1)	8 (2)	8	10	35 (11)	35 (5)	80 (16)	80 (18)	60 (33)	60 (37)	64 (45)	59 (41)	57 (38)	57 (36)	633 (283)
EPG Training Course													2	6	8	5	5	26 (0)
Total	9 ()	0 ()	16 ()	12 (1)	14 (2)	10 ()	17 (0)	44 (11)	42 (5)	92 (16)	86 (18)	65 (33)	70 (37)	78 (45)	76 (41)	74 (45)	76 (42)	781 (296)

( ):number of loan employees

Table II.5-8 Results of Training in JAPCO/JAERI

as of end of fiscal 1995

	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	Total
JAERI Reactor in General Course		1	1	1		1	1	1		1					1			8 (0)
JAPCO Training Center (A Course)	3	2	2	2	3	3	3	2		3	3	3	11 (9)	8 (7)	9 (8)	1	1	59 (24)
JAPCO Training Center (A Course)	1																	1 (0)
Navigation Training (Ministry of Transport)							1	1		1	1	2	1	1		1		9 (0)
Total	4 (0)	3 (0)	3 (0)	3 (0)	3 (0)	4 (0)	5 (0)	4 (0)	0 (0)	5 (0)	4 (0)	5 (0)	12 (9)	9 (7)	10 (8)	2 (0)	1 (0)	77 (24)

( ):number of loan employees

Table II.5-9 Results of Training in Makers

as of end of fiscal 1989

	'79	'80	'81	'82	'83	'84	'85	'86	'87	'88	'89	Total
Nuclear Reactor Control Course (Toshiba)					2				1	2		5
Turbine Course (Toshiba)							2	1	1	1	4	9
Turbine Control Course (Toshiba)							1		2	2		5
Generator Course (Toshiba)							2	3	1	2	2	10
Relay Course (Toshiba)			2	2	1	1	2		1	3	1	13
Basic Automatic Control Course (YEW)						2	2	3	3	3		13
Basic Industrial measurement Course (YEW)					6 (1)					2		8 (1)
Basic Air-type Instrument Course (YEW)						1	2					3
Basic Electronic Instrument Course (YEW)						1		1				2
Advanced Control (Basic) (YEW)											1	1
Reactor Safety Training (Hitachi)					1	1			1			3
Diesel Technology Course (Nigata Steel)								2		1		3
Limitorque Technology Course (Nippon gear)					2	2	1	1	2			8
Electromagnetic Valve Technology Course (Asco)				2	2	2		1	2			9
Control Valve Technology Course (Gadelius)							2	1	1			4
Instrumentation Technology Course (Fuji Electric)			2 (1)									
Total			4 (1)	4	14 (1)	10	14	13	15	16	8	98 (2)

( ):number of loan employees

Reference

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[II-2] IAEA Safety Series No.50-SG-O3, Operational limits and conditions for nuclear power plants, 1979

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III. Maintenance Management [III-1] [III-2] [III-3] [III-4] [III-5] [III-6] [III-7]

1. Outline of Maintenance Management

The basic principle of maintenance management in "Fugen" is to make preventive maintenance, which aims at improvement of the reliability of the plant by preventing occurrence of trubles. Efforts are made to maintain and improve functions and integrity of each equipment/component constituting of the plant. In addition, the contents of the periodic inspection and the inspection cycle are so determined as to realize preventive maintenance, by taking into account the degree of importance on safety, the operating conditions (temperature, pressure, type and flow rate of the fluid, installation environment, etc.), the operating hours (consecutive operation, standby, etc.), the technical standards (safety regulations, JEAC guidelines, etc.) and the inspection results. Furthermore, the inspection methods, the inspection cycle, etc. are reviewed and improved, while the aging deterioration status (abrasion quantity, changes in dimensions, changes in characteristics, etc.), the event occurrence result, etc. are reflected, so that the reliability of the plant can be improved. With regard to unscheduled shutdown of the plant including a reactor scram or damages in equipment and components, examination and evaluation are made on the structure, the materials, the operational status, and environment, the design and construction records, etc.; based on which the causes are investigated, tests and inspections are performed, and appropriate countermeasures and the scope of countermeasures are determined. At the same time, considering the above evaluation the similar portions are inspected and countermeasurs are taken when necessary, so that recurrence can be prevented.

The daily inspection is made for detecting faults in the early stage by tour inspection, etc. of operators and maintenance personnel, thereby preventive maintenance actions can be taken. In addition, minute fluctuations in the plant operation status are monitored and evaluated on the long-term basis using the plant data processing system, etc. so that anomaly can be detected in the early stage and preventive maintenance can be securely performed.

The periodic inspection is made on the basis of the "long-term periodic inspection plan". The experiences of daily inspection and previous periodic inspections and trouble experiences occurred in "Fugen" and other nuclear power plants are reflected on the periodic inspection plan of the year, based on which inspection, testing and maintenance are made on the equipment and components. Furthermore, during the periodic inspection, the works to cope with aging deterioration of the equipment and

components, the works to improve the functions and the reliability and the works to save the labor and mitigate exposure are performed so that the operability and the maintainability can be improved.

When maintenance works of equipment/components and works to recover from a failure, etc. are performed, a work sheet is issued each time so that the works, isolation, etc. can be correctly managed. And the work results, the parts replacement results, the failure experiences, etc. are totally managed using the maintenance management system with computers so that the maintenance data can be accumulated and maintenance management can be sophisticated.

With regard to the plant-specific equipment such as the pressure tube assembly, the seal plug, the fuel handling equipment, etc., and important equipment such as the reactor recirculation pump, the main steam isolation valve, etc., the development demonstration tests were performed in the design stage and the manufacture stage to confirm the function and the performance, and the endurance tests were performed in the Oarai Engineering Center to reflect the test result on design and manufacture of the actual equipment and components. As the result, though several troubles including several times of shutdown of the plant occurred during the operation period so far, troubles caused by the nature of design characteristics of a heavy water reactor have not occurred at all. It means that the designed performance and the reliability of each equipment/components are demonstrated. We have performed the periodic inspection 14 times so far, and established the overhaul inspection techniques for the specific equipment such as the heavy water circulation pump, the control rod drive mechanism, the refueling machine, etc. and the in-service inspection techniques for the pressure tube assembly, the inlet pipe and the riser pipe.

2. Maintenance Management Organizational System

2.1 Outline

The Maintenance Section is in charge of the jobs including maintenance, inspection and repair of each equipment in the power plant, extension, modification and removal of equipment as well as repair and extension of civil engineering and buildings. These maintenance jobs in the power plant are performed by approximately 40 people belonging to the Maintenance Section, by approximately 30 people to which works are consigned and through contracts to external organizations including five companies which designed and manufactured major equipment and components.

When commercial operation of "Fugen" was started, the Maintenance Section consisted of groups classified by the type of job (machinery, electricity and instrumentation). From 1974 (before the fifth periodic inspection), however, the groups have been classified for each equipment so that maintenance can be performed efficiently and the personnel can be stationed more flexibly.

The maintenance jobs of "Fugen" are expected to increase as the equipment is subject to aging. The maintenance jobs will never be decreased. On the other hand, it seems difficult to secure and increase well-skilled maintenance personnel. Considering these situations, it would be necessary to reinforce the system and the organization by rationalizing and standardizing the routine jobs, promoting the consignment of jobs and improving the technical level of the maintenance personnel through fulfillment of education and training.

2.2 History of Maintenance Management Organizational System

Until the current point after operation is started, the organizational system and the personnel configuration of the Maintenance Section have been mainly modified three times.

In the first stage (1979 to 1984), the Maintenance Section consisted of approximately 40 people. Many of them were from the Electric Power Development Company (EPDC) who were administrators and engineers having experience of thermal and hydro power plants and experience of construction management of "Fugen".

The maintenance management organizational system in this stage was classified by the type of job (machinery, electricity, instrumentation and engineering works). It was tried to foster and reinforce the staff members of the Power Reactor and Nuclear Fuel Development Corporation (PNC) through on-the-job training

instructed by engineers of EPDC.

In the second stage (1984 to 1990), the Maintenance Section consisted of approximately 40 to 45 people. Many PNC staff members occupied the administrative posts and worked as technical instructors in the Maintenance Section. In the 1990's, skilled maintenance personnel were transferred to other sections including "Monju", etc., and the personnel became insufficient, as well as well-skilled personnel. The personnel was decreased to 37 people at the end. Insufficiency in the personnel quantity caused by transfer of personnel was partially filled up by the personnel in charge of operation of the ATR demonstration reactor from EPDC. But the personnel quantity was still insufficient and the well-skilled personnel was not still fulfilled, and the works and responsibility assigned to each person were increased. Moreover, failures and troubles in the equipment were getting to increase due to aging of the equipment, modifications and replacement of the equipment increased, and work for education of the operators of the ATR demonstration reactor were added. Fig. III.2-1 shows the history of the personnel in the Maintenance Section and the major modifications and extensions.

2.3 Current Maintenance Management Organizational System

In the current maintenance management organizational system, the maintenance works for the power plant are classified for each equipment.

Figs. III.2-2 and III.2-3 show the management system of the Maintenance Section. Maintenance of "Fugen" is performed by four groups in the Maintenance Section in cooperation with Tsuruga Atomic Service (TAS) and plant manufacturers. The contents of the major jobs assigned to each group in the Maintenance Section are described below.

(1) Superintendence Group

This group is in charge of the common jobs in the entire Maintenance Section, superintendence of the Maintenance Section, and functions as the point of contact with other sections.

The major jobs are job planning, budget management, property management, process management, quality control, response to the authorities, document control, education management, etc. This group is also in charge of development and operation of the computer system to upgrade operation /maintenance management.

(2) Equipment I Group

This group is in charge of maintenance management of the waste treatment equipment and the utility equipment in the site, planning and execution of the modification works, etc. and documentation of the maintenance results, etc. This group is also in charge of maintenance of the equipment except the power generation equipment (telephones, structures in the site, company dormitories, etc.).

In addition, this group adjusts and superintends the daily inspection works performed by TAS, such as repair of equipment, monthly inspection, cleaning and decontamination of main building, instrument management, warehouse management, etc.

(3) Equipment II Group

This group consists of the Fuji Subgroup which is in charge of the fuel handling equipment, the engineered safety features, the radiation monitor, the seal plug, the shielding plug, etc. and the Sumitomo Subgroup which is in charge of the heavy water system, the helium system, the auxiliary cooling system, the heavy water upgrander-I and -II, etc. This group is in charge of maintenance management of the equipment in charge, planning and execution of the modification works and documentation of the maintenance results, etc.

(4) Equipment III Group

This group consists of the Hitachi Subgroup which is in charge of the reactor proper, the CRD, the nuclear instrumentation, the reactor control system, the feed water control system, the reactor protection system, the electric equipment, etc., the Toshiba Subgroup which is in charge of the reactor containment equipment, the turbine system equipment, the feed water/condenser system equipment, etc. and the Mitsubishi Subgroup which is in charge of the reactor recirculation system, the reactor coolant purification system, etc.

This group is in charge of maintenance management of the equipment in charge, planning and execution of the modification works and documentation of the maintenance results, etc.

2.4 Status of Jobs Consigned to Outside of PNC

A part of maintenance/modification jobs for the equipment in the power plant are consigned to the outside of PNC.

The contents of the jobs consigned to the outside are described below.

(1) Consignment of Maintenance Works of Power Plant Equipment (to TAS)

A part of the maintenance management works for machinery, electricity,

instrumentation, buildings, etc. inside the plant, the cleaning/decontamination works inside the main building, the instrument/warehouse management works are consigned.

(2) Consignment of Maintenance (to companies of consigned maintenance)

As to the common equipment among the power generation equipment, maintenance and management are consigned to the external companies.

The major equipment consigned are shown below.

① Various computer equipment (process computer, computer for fuel handling, computer for ATLB, etc.)

② Elevators (R/B, A/B, new RW/B, D/B)

③ Battery equipment (115 V battery, 24 V battery, battery for communication equipment and battery for fire extinguishing equipment)

④ Package coolers (inside main building, computer room, etc.)

⑤ Telephone exchange equipment, etc.

(3) Contract Works (to many companies)

The periodic inspection works, modification works, repair works, etc. are put out to contract to many companies.

Among these works, the periodic inspection works for the major equipment inside the power plant are put out to contract mainly to five companies which designed manufactured and installed "Fugen". Assignment of the major equipment is described below.

① Hitachi: Reactor proper, CRD, nuclear instrumentation equipmwnt, reactor protection system, in-site current feeding equipment, power supply for instrumentation, switchyard equipment, rare gas equipment, IA and SA, hydrogen/oxygen generation equipment, etc.

② Toshiba: Main steam system, reactor auxiliary cooling sea water system, reactor containment vessel equipment, turbine system, feed water/condensed water system, etc.

③ Mitsubishi: Reactor cooling system

④ Fuji: Seal leak system, fuel handling system, engineered safety features, radiation monitor, etc.

⑤ Sumitomo: Heavy water/helium system, reactor auxiliary cooling system, carbon dioxide gas system, etc.

⑥ Orugano: Condenser demineralizer

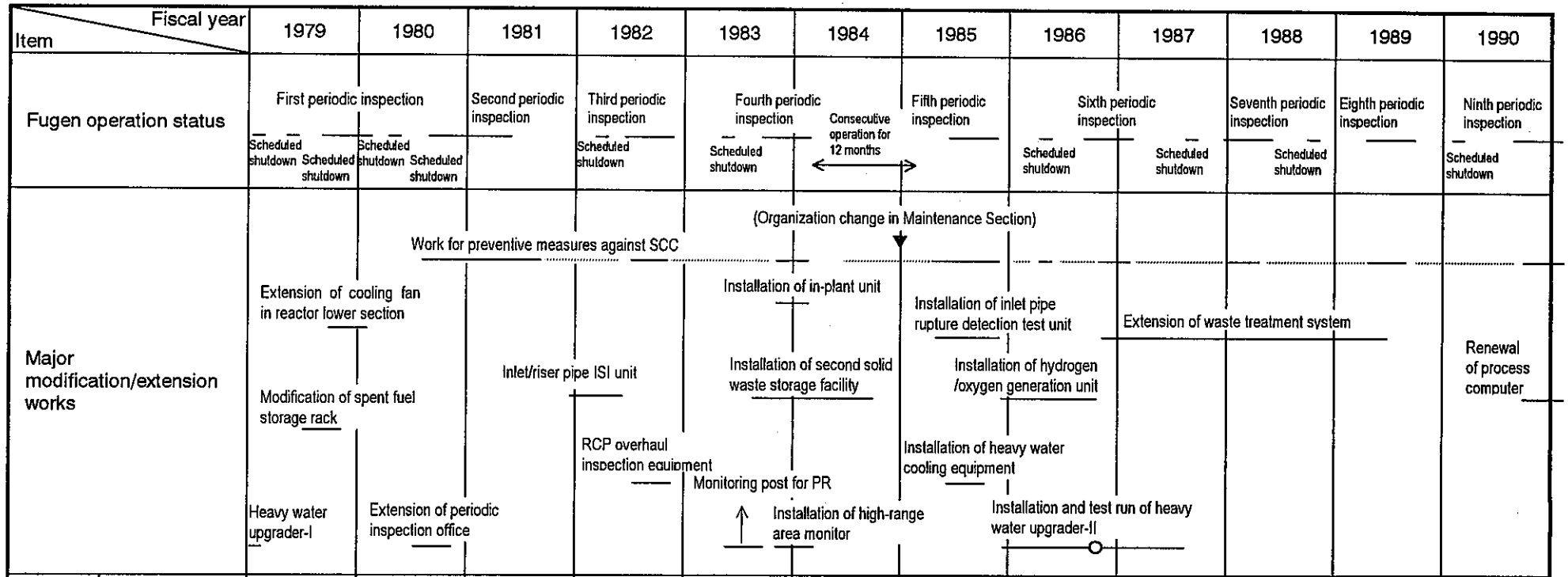
⑦ Niigata Tekko: Emergency diesel engine

⑧ Yamabu Honeywell: Turbine system regulation valve

⑨ Ebara: Part of waste treatment equipment

⑩ TAS: Inspection of seal plugs and shielding plugs, on-site utility equipment, etc.

Fig. III.2-1 History of Maintenance Section Personnel



Item		1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990
Administrators	PNC	0	0	0	0	0	1	1	1	3	2	2	0
	EPDC	3	3	2	2	2	1	1	1	1	1	2	4
	Others	2	2	2	2	3	2	1	1	0	0	0	0
Subtotal		5	5	4	4	5	4	3	3	4	3	4	4
Non-administrators		27	30	32	33	34	35	34	34	34	28	26	21
		6	7	4	3	2	2	4	6	6	6	10	5
	Utility companies	0	0	0	0	0	0	0	0	1	1	2	2
	Manufacturers	0	0	0	0	0	0	0	0	0	0	0	1
Subtotal		33	37	36	36	36	37	38	40	41	35	38	29
Total		38	42	40	40	41	41	41	43	45	38	42	33

Fig.III.2-2 Maintenance Section Organizational System

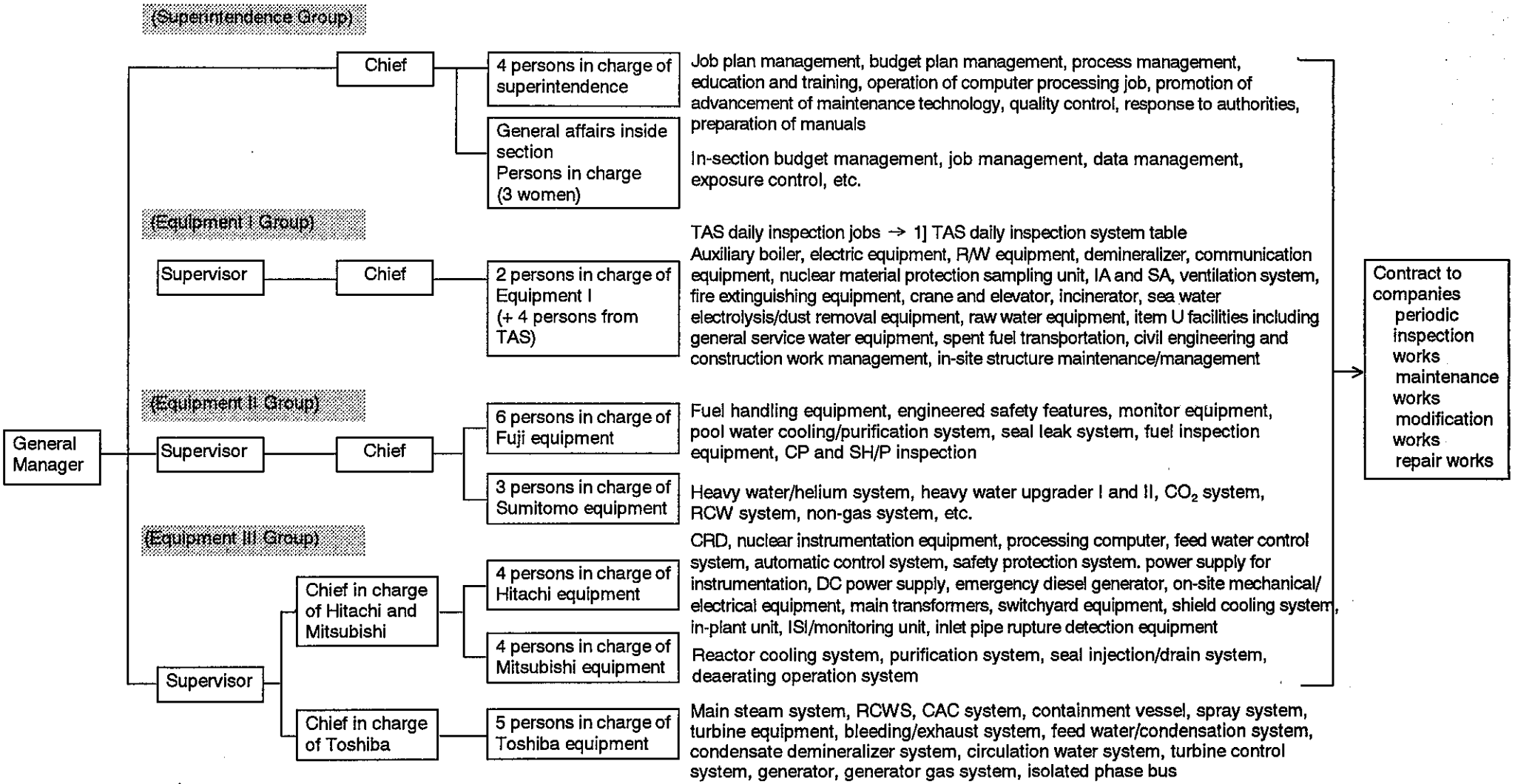
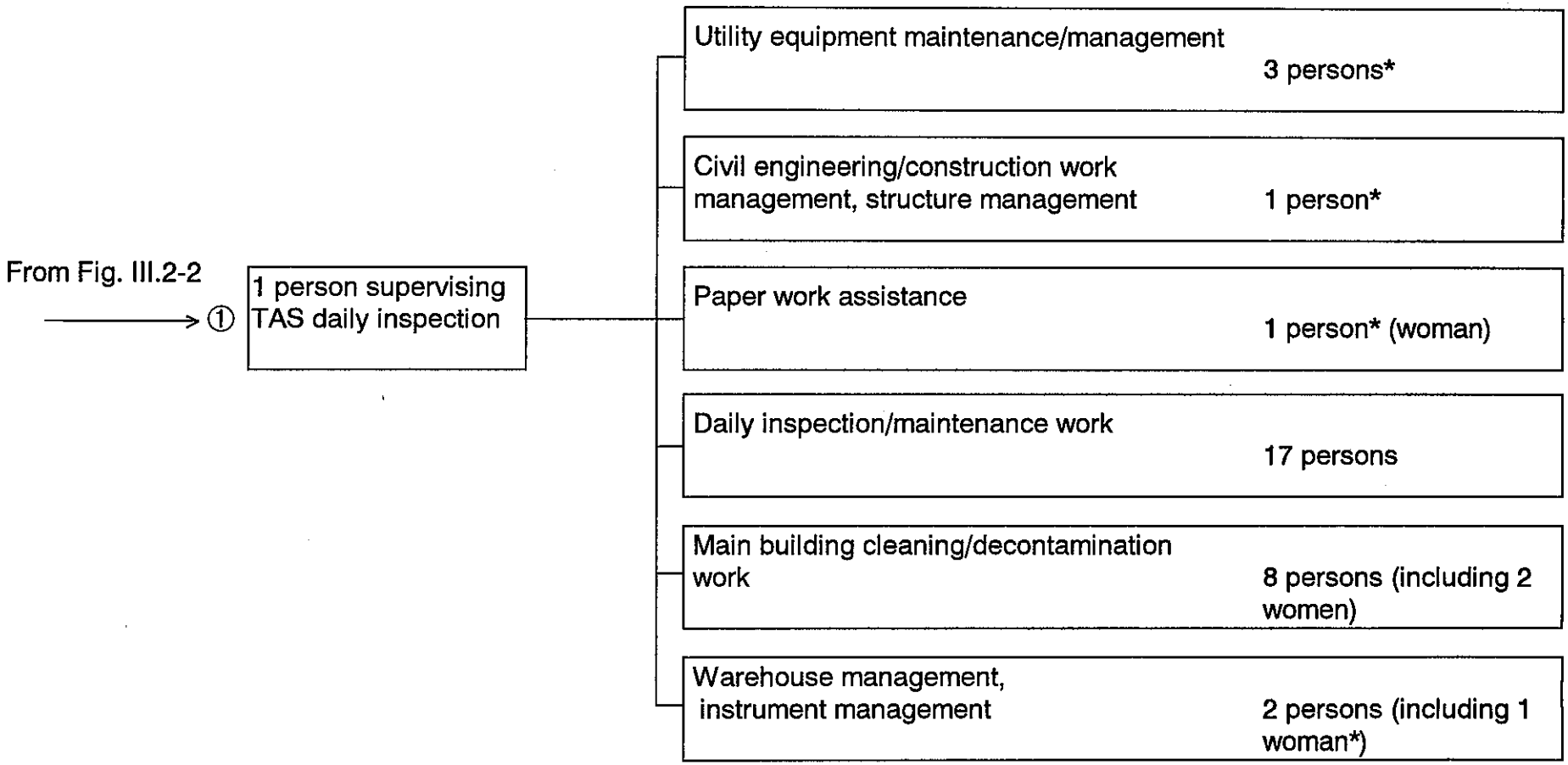


Fig. III.2-3 TAS Daily Inspection Work System



(Total: 33 persons including 6 persons marked with * always working in Maintenance Section)

3. Basic Policies of Maintenance Management [III-8]

3.1 Basic Policies

The Maintenance Section performs maintenance, modification, etc. of the equipment in accordance with the schedule so that safe and stable operation of "Fugen" is assured. In addition, the Maintenance Section has a basic policy to establishe and enhance the technologies including the operation/maintenance technology as described below.

① To perform inspection, repair, modification, etc. of the equipment in accordance with the schedule to assure stable operation of "Fugen".

② To establish and enhance the maintenance management technology.

③ To document the maintenance experience and result of "Fugen" as much as possible so that the data documented can be reflected on development of the demonstration reactor.

④ To execute the demonstration test required to develop the ATR demonstration reactor, and perform modification of the equipment required.

⑤ To introduce positively new technology, and make efforts to develop new technology by itself.

⑥ To promote consignment of the established routine jobs.

⑦ To improve the technical level and the ability through execution of systematic education.

⑧ To utilize efficiently the fund, the personnel, etc. by executing the jobs in accordance with the schedule.

The "Maintenance Section Medium/Long-Term Job Plan" and the "Yearly Maintenance Plan" are established so that maintenance management of "Fugen" can be systematically executed in accordance with the basic policies described above.

3.2 Jobs Executed

(1) Maintenance Management of Equipment and Components

The reliability is required to be maintained and improved through enhancement of maintenance management and quality control to assure stable operation of "Fugen". To realize this, the following points are examined especially.

① Countermeasures against aging deterioration of the equipment and examination of measures to realize long service life

② Rationalization of the periodic inspection

③ Examination of efficient maintenance methods

④ Improvement of the maintainability

⑤ Countermeasures to mitigate exposure

⑥ Countermeasures to mitigate the waste generated and mitigate the radiation released

(2) Extensions and Modifications of Equipment and Components

In "Fugen", the operation result and the maintenance result of "Fugen" as well as experiences of light water reactors are reflected, and the equipment has been extended and modified using mainly the periodic inspections and the scheduled shutdown period in each fiscal year. The major extensions and modifications of the equipment in "Fugen" are preventive countermeasures against SCC and enhancement in the ability and the capacity of each equipment.

(3) Research and Development

The following technology development required for maintenance management is performed while specificity of "Fugen" is taken into account.

① Pressure tube repair method (replacement, etc.)

② Establishment and enhancement of the ISI technology (for pressure tube assemblies, for inlet and riser pipes, general piping)

③ Establishment and enhancement of the heavy water handling technology

④ Development of new instruments for heavy water (development of double diaphragm type transmitter)

⑤ Establishment and demonstration of the primary system decontamination technology

⑥ Establishment of the ice plug execution technique

⑦ Development of equipment and parts of long service life

⑧ Development of repair robots

⑨ Demonstration of the hydrogen injection technique (in-plant material test)

⑩ Development of the work isolation system

⑪ Enhancement of the maintenance management system (expert system, etc.)

⑫ Development of diversified anomaly diagnosis systems and techniques

4. Maintenance Management Result

4.1 Maintenance Plan

The "Maintenance Plan of the Fiscal Year" is established in each fiscal year while the "Maintenance Section Medium/Long-Term Job Plan", diversified basic plans, the "Budget Plan of the Fiscal Year", the job results of the previous fiscal year, etc. are taken into account.

In the "Maintenance Plan of the Fiscal Year", the contents to be executed in the current fiscal year are determined such as the maintenance program including inspection of the equipment and components, the modification/replacement program for the equipment and components required to maintain/improve the equipment, the technology development program for ATR commercialzation, the budget plan, the job execution program, etc.

4.2 Daily Inspection/Maintenance

The monthly inspection is performed to each equipment to prevent failures and assure the reliability. The monthly inspection includes inspection and replacement of the rotator brush, calibration of the nuclear instrumentation, analysis of vibration of the rotation equipment, inspection of the Item U equipment, etc. so that the integrity and the reliability of the components whose inspection and calibration are required to be performed in a short time can be maintained and confirmed.

Against faults detected by daily inspection and tour inspection by operators, repair, preventives maintenance actions, etc. are performed so that the integrity and the reliability of the equipment and components can be maintained.

In addition, the trend of the plant operation status quantity is analyzed and evaluated by utilizing statistical/analytical process of the maintenance data accumulated in the Maintenance Management System (MMS), the plant data processing system, etc. so that the trend evaluated can be reflected on daily inspection, etc. and anomalies can be detected in the early stage or prevented.

4.3 Periodic Inspection [III-9] [III-10]

(1) Necessity of the Periodic Inspection

The periodic inspection is performed in every year in accordance with the "Law for Regulation of Nuclear Fertile Material, Nuclear Fuel Material and Reactors" (hereinafter referred to as "law for regulating nuclear installations"). And because "Fugen" is a reactor facility having power generating installations, the

inspections in accordance with the "Utility Industry Law" are also performed. Accordingly, the procedures, the execution methods, etc. of the periodic inspection are so made that regulation by the "law for regulating nuclear installations" and the "Utility Industry Law" may not function as double regulation.

As to the equipment to which other laws and regulations are applied, the periodic inspection is systematically performed. As to the equipment and facilities to which target the periodic inspection is not applied in accordance with the laws described above, efforts have been made to assure their integrity and reliability by performing self-inspection.

In the periodic inspection, it is confirmed that the technical standards specified by the law, the safety review guideline, the Japan Electric Technology Code (JEAC), etc. are satisfied and that the requirements specified in the reactor facility safety regulations are also satisfied so that the safety and the integrity of the function and the performance of the equipment and components can be secured and that stable operation can be maintained.

(2) Periodic Inspection Items and Inspection Cycle

The items in each periodic inspection are performed in the inspection cycle determined preliminarily by the "Long-Term Periodic Inspection Plan" established in May, 1979. In concrete, the components are overhauled, the functional inspection is performed, and the instrumentation and control equipment and the radiation control equipment are inspected and calibrated. In addition, the safety relief valve and the main steam isolation valve are inspected by performance inspection and leak testing, the emergency core cooling system is inspected by functional inspection, the fuel assembly is inspected by appearance inspection and sipping inspection, and in-service inspection is made on the Classes 1 and 3 components, etc.

The periodic inspection consists of the authorities inspection performed by the government and the in-house self-inspection. The items of the authorities inspection are determined so that the requirements specified in the "law for regulating nuclear installations" and the "Utility Industry Law" are satisfied and that the contents of the "Light Water Reactor Periodic Inspection Standard Procedures" are compatible.

For the equipment and facilities specific to the ATR, the inspection items are added. Table III.4-1 shows the outline of the current authorities inspection. Table III.4-2 shows comparison between the "Light Water Reactor Periodic

Inspection Standard Procedure" and the "Fugen Periodic Inspection Procedure". The inspection cycle is determined based on the technical standards (Safety Regulation, JEAC, etc.), the in-house standards of manufacturers, the importance of the equipment, the operating conditions (temperature, pressure, flow rate, fluid type, etc.) and the operating hours (consecutive operation, stand-by, etc.). The inspection methods and the inspection cycle are reviewed by referring to the aging deterioration status (abrasion quantity, changes in dimensions) of the parts and the operation/maintenance results including fault experience so that inspection and maintenance can be rationalized further more. Table III.4-3 shows a part of the reactor cooling system described in the Ten-Year Plan for Periodic Inspection in "Fugen".

(3) Plan, Management and Report of Periodic Inspection

When the periodic inspection is performed, the "Periodic Inspection Plan" (to be distributed to the site, the authorities, etc.) including the inspection items, the contents, the processes, the radiation control, the waste management, etc. is preliminarily established. Diversified preparations including explanation of the "Periodic Inspection Plan" to the authorities, application for licenses related to modification works, application for the pre-service inspection, jobs for contracts related to the periodic inspection, etc. are started 6 to 8 months before the periodic inspection is started.

Within the period from parallel off to parallel in of the generator, each equipment described in the "Periodic Inspection Plan" is inspected and tested to confirm that no anomaly is detected in any equipment. After parallel in, "Fugen" is operated at the rated power for a certain period (approximately one month), then undergoes the total load test. When "Fugen" has passed the total load test, the periodic inspection is completed. The next periodic inspection should be performed within one year before the month and day on which the previous periodic inspection was completed.

When the periodic inspection is completed, the periodic inspection report is made describing the inspection results, the modification contents, the core management, the operation management, the chemistry control, the radiation control, the waste management, the quality control, the schedule of management result, etc. The report is submitted to the authorities and distributed in PNC and related organizations.

(4) Determination and Reduction of Periodic Inspection Schedule

(a) Periodic Inspection Schedule

The periodic inspection schedule is determined based on the major inspection items described in the "Long-Term Periodic Inspection Plan", the refueling program, the major modifications items in the periodic inspection, the authorities inspection, etc. The basic schedule is so determined that the schedule can be minimized, taking into account the major works in the periodic inspection, the modification works, the works for refueling, etc. and the authorities inspection.

(b) Examination to Reduce Standard Schedule and Present Periodic Inspection Schedule Based on Results

After the second periodic inspection, the works for countermeasures to prevent SCC were performed in a long period (up to the ninth periodic inspection), and considerable time was required to perform the periodic inspection. However, in the first periodic inspection (84 days) in which the works for countermeasures to prevent SCC were not started yet and the seventh periodic inspection (73 days) in which the work quantity for countermeasures to prevent SCC was small, the schedule almost equivalent to the one required for a light water reactor was achieved.

It is concluded based on the results so far that a standard periodic inspection of "Fugen" can be performed in approximately 90 days (from parallel off to parallel in) if the works for countermeasures to prevent SCC are not performed.

The breakdown of 90 days required for a standard periodic inspection contains measurement of the iodine release quantity (1 day), fuel unloading (11 days), inspection of the fuel handling equipment (38 days), monitoring of the pressure tubes (10 days), fuel loading (8 days) and the system functional inspections (22 days).

The works to inspect the refueling machine and the works using the refueling machine occupy more than a half of 90 days of the periodic inspection. Accordingly, to reduce the schedule, it is essential to rationalize the refueling machine inspection works and utilize efficiently the refueling machine.

There are inspection works under subcriticality for the CRD, the LPM and the equipment related to heavy water including the heavy water circulation pump. It is required, therefore, to examine reduction of the schedule for these works together with rationalization for the works related to the refueling machine.

Fig. III.4-1 shows the fourth periodic inspection result which indicates a representative periodic inspection schedule in which countermeasures to

prevent SCC are performed. Fig. III.4-2 shows the seventh periodic inspection result which indicates a representative reduced periodic inspection schedule.

(5) Periodic Inspection Results

The first periodic inspection was a standard periodic inspection of "Fugen", and the period from parallel off to parallel in was 84 days which was almost equivalent to the periodic inspection of a light water reactor. This was the first periodic inspection after the commercial operation had been started, and the integrity of the components specific to the ATR was confirmed and the inspection methods were established.

In the second periodic inspection, the works to prepare for repair, such as unloading all the fuel assemblies, etc., were performed preliminarily, then the periodic inspection was started, and the works for countermeasures to prevent SCC were performed, because stress corrosion cracking (hereinafter referred to as "SCC") had been detected in the weld area of the piping (5B, 6B, 8B) in the residual heat removal system, the low pressure injection system, etc. among the reactor cooling system piping in the scheduled shutdown for inspection in November, 1980. As the result, it took 174 days until parallel in after the periodic inspection was started. The piping replacement length related to the countermeasures against SCC at this time was approximately 300 m.

In the third periodic inspection, the overhaul inspection was performed to the reactor recirculation pump (Unit No. D) which is a major equipment in the reactor cooling system to confirm the integrity and establish the inspection method. As the works for countermeasures to prevent SCC, the piping in the reactor coolant clean-up system and the high pressure injection system was replaced, and the Induction Heating Stress Improvement (IHSI) was performed to the piping in the reactor cooling system (the steam drum nozzle safe end, the lower header nozzle, etc. in the high pressure injection system, the low pressure injection system and the residual heat removal system). It took 123 days to complete the periodic inspection from start to parallel in.

In the fourth periodic inspection, monitoring was performed to five pressure tubes using the pressure tube monitoring equipment No. 1 to confirm the irradiation characteristics of the pressure tube. By making the inner diameter measurement, the UT, etc., the integrity of the pressure tubes was confirmed, and the function of the pressure tube monitoring equipment was confirmed. As the works for countermeasures to prevent SCC, the materials of the reactor feed

water system piping was replaced, and the IHSI was performed to the B loop piping in the reactor recirculation system. It took 135 days to complete the periodic inspection from start to parallel in.

In the fifth periodic inspection, careful inspections were performed to the heavy water system equipment (heavy water dump tank, heavy water cooler, etc.) by utilizing the processes prolonged due to large-scale works for countermeasures to prevent SCC, then the integrity of the components was confirmed and the inspection methods were established. As the works for countermeasures to prevent SCC, the materials of the A loop piping in the reactor recirculation system were replaced, the IHSI was performed to the loop B of the reactor recirculation system, and the hydrogen injection equipment was installed to prevent SCC in the inlet pipes and the riser pipes. It took 172 days to complete the periodic inspection from start to parallel in.

In the sixth periodic inspection, the overhaul inspection of the generator which is to be performed once in five years was performed, then the integrity of the equipment was confirmed, and validity of the inspection cycle was confirmed. As the in-service inspection, the elongation of the pressure tubes in the axial direction was checked using the pressure tube monitoring equipment No. 2, and the UT was performed to the area around the upper rolled joint, then the integrity of the pressure tubes was confirmed. By using the inside-inlet/riser pipe inspection equipment, the ISI range in the inlet/riser pipes was extended, and the inspection was improved. As the works for countermeasures to prevent SCC, the materials of a part of the main steam system piping, the steam pipe nozzle safe end and the safety relief valve nozzle safe end were replaced. It took 84 days to complete the periodic inspection from start to parallel in.

In the seventh periodic inspection, the inspection items were rationalized and the period was reduced because the works for countermeasures to prevent SCC was of small scale. As the result, it took only 73 days to complete the periodic inspection, which is the shortest among the periodic inspections of "Fugen". As the works for countermeasures to prevent SCC, the nozzle safe ends in the high pressure injection system and the low pressure injection system in the steam drum were replaced.

In the eighth periodic inspection, two recirculation pumps (units A and C) were overhauled, and the integrity was confirmed. As the works for countermeasures to prevent SCC, the center part of the lower header and the corresponding inlet pipe of the A loop as well as the nozzle safe end in the residual heat removal

system in the steam drum were replaced. To reduce exposure caused by the works for countermeasures to prevent SCC and the periodic inspection, the A loop in the recirculation system was decontaminated. As the result, exposure was considerably reduced. It took 104 days to complete the periodic inspection from start to parallel in.

Table III.4-4 shows the results of the first to thirteenth periodic inspections in "Fugen".

(6) Periodic Inspection Result of Major Equipment

(a) Overhaul Inspection of Refueling Machine

The refueling machine is so designed that refueling could be performed from the bottom of the reactor while the reactor is operating. However, refueling is actually performed while the reactor is shut down due to fear of fuel failure caused by PCI (Pellet Clad Interaction).

As the spent fuel assemblies are handled, crud, etc. of high radioactivity are brought into and accumulated inside the handling equipment. In the early stage of operation, malfunction occurred in the drive unit and the slide unit caused by effects of clad, etc. To cope with this type of malfunction, the countermeasures against clad have been taken by installing the purified water injection system in the bearing and modifying the clearance and the material of the slide unit. In addition, the inspection cycle and the parts replacement criteria have been reviewed, and the operating procedures have been improved. As the result, after the third periodic inspection, refueling can be performed smoothly by replacing consumable parts in accordance with the schedule.

Because the exposure related to inspection of the refueling machine occupies approximately 15% (approximately 50 man/rem) of the total exposure dose (approximately 350 man/rem) related to inspection and testing in the periodic inspection, chemical decontamination has been performed since the fifth periodic inspection to reduce exposure.

(b) In-Service Inspection (ISI)

The ISI of "Fugen" is planned based on the "Fugen ISI Ten-Year Plan" established in accordance with the JEAC 4205 "In-Service Inspection of Systems and Components for Light Water Type Nuclear Power Generation". However, since the second periodic inspection, the flaw detection angle has been improved so that the UT can be performed to the stainless steel tube welded area of the inlet pipe (2B) and the riser pipe (3B) because stress corrosion cracking (SCC) was detected. Table III.4-5 shows the ten-year plan

of in-service inspection for the reactor coolant pressure boundary in "Fugen". Table III.4-6 shows the ten-year plan of in-service inspection for the outside of the reactor coolant pressure boundary.

The core of "Fugen" consists of 224 pressure tubes, and the piping of inlet pipes (2B) and outlet pipes (3B) connected to the pressure tubes has the pipe bundle structure. Accordingly, the UT can be performed to the outer circumference of the pipe bundle using the normal manual flaw detection technique, but the UT with the manual flaw detection technique cannot be performed in many cases to the welded area inside the pipe bundle because the inspection space cannot be assured. Therefore, UT unit for the inside pipe bundle was developed to extend the inspection area.

As to the pipe in which erosion by wet steam is expected such as the steam bleeding system and the heater drain system around the steam turbine, the wall thickness is measured to monitor the piping replacement timing.

(c) Pressure Tube Monitoring Inspection

The pressure tube material is Zr-2.5%Nb alloy, and subject to irradiation creep in the operation status because it is exposed to neutron irradiation as a core structural material. Though irradiation creep is taken into account in the design of the pressure tube, there are a few data on creep of the Zr-Nb material. Accordingly, the follow-up monitoring is performed on aging changes in the inner diameter, presence of defects and the inner surface status. Fig. III.4-3 shows the pressure tube monitoring equipment.

In the fourth periodic inspection, the pressure tube monitoring inspection was performed for the first time. As the result, the integrity of the pressure tube was confirmed by detecting that the creep of the pressure tube in the circumferential direction is almost equivalent to the design predicted value and that no anomaly is detected by the UT in the pressure tube itself.

After that, the UT, the VT and the inner diameter measurement were performed to five pressure tubes. The five pressure tubes monitored consisted of four follow-up pressure tubes and one pressure tube of the maximum integrated neutron irradiation. The VT was performed to other 10 pressure tubes as the in-service inspection. No anomaly was detected by any inspection. By comparing the inner diameter measurement data with the pre-service inspection (PSI) and the first inspection (performed in the fourth periodic inspection), it is concluded that the values obtained are almost equivalent to the values predicted using the calculation formula and that the

calculation formula used for design would be valid.

(d) Overhaul Inspection of Reactor Recirculation Pump

The recirculation pump of "Fugen" is of vertical axis, single-step axis seal type, and was manufactured for the first time in Japan as a pump used in the primary cooling system of nuclear power plants.

The shaft seal unit is the most important part of the recirculation pump, and consists of the No. 1 seal of non-contact type, the No. 2 seal of mechanical seal type and the No. 3 seal for emergency. The shaft seal unit has been overhauled once in every two years to confirm the operation status and replace the seals. In the third periodical inspection, the internal of one (unit D) among the four pumps was overhauled, and the impeller, the main shaft, the thermal shielding unit and the casing were inspected and serviced. Though reddish brown crud was evenly adhered on every part of the internal, no anomaly was detected.

In the overhaul inspection, the internal was chemically decontaminated to reduce exposure and assure good inspection status. As the result, the dose rate on the surface of the internal was reduced to approximately 400 mR/h from approximately 2.5 R/h.

The rubber O ring inside the non-contact type No. 1 seal may cause a trouble when reused. The endurance test was performed on the rubber O ring under the condition similar to the condition of the actual unit. Based on the results, the No. 1 seal has been reused from the fifth periodical inspection so that the cost can be reduced.

(e) Inspection of Heavy Water System

The heavy water circulation pump is equipment specific to "Fugen", and the canned structure in which the pump and the motor are integrated each other is adopted to prevent leak of heavy water.

Before the overhaul inspection, the inside of the pump is dried completely with dry helium and dry air to reduce completely the tritium concentration so that internal exposure caused by tritium contained in heavy water can be prevented.

All the inspection works are performed inside the temporary clean house so that extension of pollution can be prevented, and the workers in charge wear tritium protective clothes upon necessity. Biological assay through urine analysis is performed to the workers in charge so that existence of intake of tritium caused by the works can be checked.

In the periodic inspections performed so far, the integrity of the heavy water circulation pump has been assured by the overhaul inspection performed once in two years. The wear of the bearing is considerably less than what is predicted, so the inspection cycle may be able to be prolonged in the future. Table III.4-7 shows the maintenance result and evaluation of the heavy water circulation pump.

In the fifth periodic inspection, heavy water was collected into the heavy water tank so that the heavy water equipment could be opened, and the internal inspection was performed. As the result, no anomaly was detected in the dump tank and the heavy water cooler, and the long-term integrity for the future was confirmed.

(f) Inspection and Servicing of Control Rod Drive Mechanism (CRD)

The CRD is installed in the upper part of the reactor, and controls the reactor power by withdraw and inserting the control rods into heavy water (moderator) of low temperature and low pressure using the wire drum motor drive method.

When the control rods are separated from the CRD in heavy water to inspect the CRD, a dedicated CRD replacement unit is installed and the required works are performed in the grove box to prevent internal exposure caused by tritium contained in heavy water. After inspection and servicing are finished, the CRD is installed again using the CRD replacement unit, and existence of leak is checked by the helium leak inspection.

Four CRD mechanisms (for the E control rods) for automatic power control, among 49 CRD mechanisms in all, are overhauled every year. Other 45 CRD mechanisms are overhauled once in five years. With regard to the CRD mechanisms installed in the reactor, the performance inspection and the scram inspection are done every year (100% inspection), and the integrity of the performance and the function is confirmed. Table III.4-8 shows the overhaul result for 55 CRD units including spare CRD units.

(g) Inspection of Turbine Equipment

The steam turbine of "Fugen" offers the number of rotations of 3,600 rpm. When the turbine is opened for inspection, attention is paid to erosion caused by wet steam specific to the nuclear turbine. The anti-erosion plate (stellite plate) claded on the final stage blades in the low pressure turbine was checked by the open inspection of the steam turbine performed in every periodic inspection, and was replaced if a partial erosion or peel-off was detected.

Erosion was detected in some parts (joint face, etc.) of the casing, etc., and repair welding, etc. were performed and reassembly was performed. Table III.4-9 shows the steam turbine periodic inspection of "Fugen".

In every periodic inspection, the eddy current test is performed to all of 11,888 tubes (made of BsTF-2) in the main condenser. As the result, 73 end plugs were attached, but sufficient margin is present against the degree of margin 5% (592 tubes).

4.4 Equipment Modification/Replacement Works

Modification/replacement works required to maintain and improve the equipment are performed mainly in each periodic inspection based on the operation result and the maintenance result of "Fugen" and the experience obtained in light water reactors.

The equipment modification works are classified as follows.

① Replacement/modification works from the viewpoint of preventive maintenance

② Replacement/modification works to improve or maintain the operability and the maintainability

③ Works to improve the workability and reduce exposure

④ Improvement works accompanied by revision of technical standards and guidelines

⑤ Repair works to eliminate faults

Among the works above, especially the works for countermeasures against stress corrosion cracking and countermeasures against corrosion and erosion are systematically performed and the in-service inspection is enhanced so that faults can be eliminated and the integrity and the reliability can be assured .

Before the countermeasures are taken, the reproducibility test, the mock-up test, etc. are performed preliminarily so that the countermeasures can be well done without fail.

The execution status of the major modification/replacement works performed so far is described below.

(1) Countermeasures against Stress Corrosion Cracking

In "Fugen", stress corrosion cracking (SCC) was detected around the weld area in the SUS 304 piping in the residual heat removal system and the emergency core cooling system inside the reactor containment vessel in 1980. To cope with this, efforts were made to examine and establish the total countermeasures against

SCC for the piping system in "Fugen" and take the countermeasures systematically.

As the countermeasure technology, piping material Stainless Steel (SUS 304) was changed to low carbon stainless steel (sus 316 or SUS 316 L) and Induction Heating Stress Improvement (IHSI) was performed to the welds, which had been already adopted in BWR in Japan. These countermeasures were finished in April, 1991.

On the other hand, as the countermeasures for the parts to which application of the existing technology was difficult such as the inlet pipes, the riser pipes, etc. of small diameter and pipe bundle structure specific to the pressure tube type reactor, the water quality improvement technology with hydrogen injection was determined to be adopted. Based on the test result of hydrogen injection in the plant for approximately 1 year, continuous hydrogen injection was started in December, 1985.

(a) Countermeasures against SCC at Construction

Around 1974 when the reactor cooling system piping of "Fugen" was installed on the site, cracking regarded as SCC was detected in the weld area of the reactor cooling system piping in boiling water type reactors (BWR) in Japan. Accordingly, the following countermeasures were taken based on the knowledge at that time in manufacture and installation of the piping in the reactor coolant pressure boundary in "Fugen" which is in the condition almost equivalent to that of a BWR.

For the welds in the SUS304 piping, the welding control was thoroughly performed including the welding heat input control, the inter-layer temperature control, etc.

In addition, to the welds with relatively high stress value in the inlet pipe and the riser pipe specific to the pressure tube type reactor, the solid solution heat treatment or water-cooled welding was performed. And the drain piping of the inlet pipe was changed from SUS304 to SUS304L.

(b) Countermeasures After SCC Occurred (November, 1980)

In November, 1980 during scheduled shutdown, penetration cracks were detected in the weld heat affected area in the lower header injection piping in the residual heat removal system inside the reactor containment vessel. In response to this fact, the ultrasonic test was performed to the weld area in the piping in the reactor coolant pressure boundary under the condition similar to that of the above parts, and anomalies were detected in 20 positions in all --- 4

positions in the residual heat removal system, 11 positions in the low pressure injection system of the emergency core cooling system and 5 positions in the high pressure injection system of the emergency core cooling system. The representative piping was cut, and investigation was performed by observing the ruptured surface, the metal structure, etc. As the result, the cause of these anomalies was evaluated as SCC in the weld area in the SUS304 piping in high temperature demineralized water.

It is thought that SCC occurs by combination of three elements, stress, environment and material. It is found based on the test result related to SCC that SCC occurs easily in the weld area in the stainless steel piping such as SUS304 piping whose carbon content is large (approximately 0.08%) and the temperature exceeds 100℃ in the water quality environment in which the dissolved oxygen concentration of the reactor water is approximately 0.2 ppm such as in "Fugen". Though the time leading to occurrence of SCC varies depending on the piping diameter, the heat input during welding, existence of flow of fluid, the stress status, etc., it is thought that SCC occurs some day.

Based on the background describe above, the countermeasures against SCC have been systematically taken so far including replacement of the piping material, the IHSI, etc. in BWR in Japan. In the piping and equipment in the reactor coolant pressure boundary in "Fugen", the countermeasures have been also taken with the latest technology of countermeasures against SCC.

① Execution Range of Countermeasures against SCC

After the residual stress, the environment and the material (factors of SCC occurrence) had been examined with consideration of the experience and knowledge accumulated in BWR reactors, the piping and the equipment of the reactor coolant pressure boundary which satisfy all the conditions below were determined as the execution range of the countermeasures against SCC.

(i) Stress

Not only the stress caused by the pressure, etc. of the fluid but also the residual stress generated during welding should be taken into account. Accordingly, all the weld areas are regarded as targets (except the position in which water-cooled welding is performed at the construction).

(ii) Environment [III-11]

It was confirmed by experiments that SCC occurs depending on the dissolved oxygen concentration, the corrosion potential, the temperature,

etc. Accordingly, the environment where the dissolved oxygen concentration is 20 ppb or more or the corrosion potential is -300 or more mVSHE at 100℃ or more of operating temperature is regarded as target.

(iii) Material

Among austenitic stainless steel, the materials in which carbide is not stabilized and whose carbon content exceeds 0.03% become sensitive during welding, and SCC can easily occur. Accordingly, the stainless steel material SUS304 is determined as target (except the parts in which solid solution heat treatment is performed at the construction).

② Technology of Countermeasures Against SCC

The technology of countermeasures against SCC in "Fugen" were examined in principle while experience and countermeasures in BWR in Japan were referred to. In addition, as the countermeasures for the parts to which application of the existing technology is difficult such as the inlet pipe, the riser pipe, etc. of small diameter and pipe bundle structure specific to the pressure tube type reactor, the water quality improvement technology with hydrogen injection was adopted. The countermeasures against SCC were determined to be performed based on the following basic policies.

(i) Replacement to materials resistant to SCC

Materials resistant to SCC are to be adopted in the following positions.

- Piping in which SCC occurred before or SCC is liable to occur
- Piping whose material can be replaced relatively easily such as piping of small or medium diameter (up to approximately 10 inches)
- Positions in which materials can be replaced in the components such as the steam drum, the inlet header, etc.

(ii) Execution of IHSI

The IHSI is to be performed in the positions in which materials cannot be easily replaced due to the structure of the piping/equipment but the IHSI can be performed.

(iii) Adoption of the water quality improvement method by hydrogen injection

The water quality improvement method to reduce the dissolved oxygen concentration by injecting hydrogen into the reactor water is to be adopted in the positions in which replacement of the materials and execution of the IHSI are difficult such as the inlet pipe, the riser pipe, etc. of small diameter and pipe bundle structure specific to the pressure tube type reactor.

Adoption of the water quality improvement technology by hydrogen

injection in "Fugen" is the first time in Japan, and there are several execution examples all over the world.

(c) Results of Material Replacement and IHSI

Material replacement and the IHSI are to be performed during shutdown for periodic inspection, and have been performed so far with the priority shown below.

① Parts contacted with the stagnant water (residual heat removal system, high pressure injection system, low pressure injection system, drain tube, etc.)

② Parts where SCC occurred in BWR (piping in reactor coolant purification system)

③ Parts where the liquid flows (reactor recirculation system, piping in feed water system)

④ Parts where the steam flows (piping in main steam system)

⑤ Weld areas in components (steam drum nozzle safe end, lower header, etc.)

Table III.4-10 shows the plan and the result of material replacement and the IHSI.

In case of the replacement of the parts welded with SUS304 stainless steel of outside the replacement target, water-cooled welding or the ISHI was performed.

(2) Countermeasures against Corrosion and Erosion

Table III.4-11 shows the major countermeasures against corrosion and erosion.

Repair and replacement to cope with intergranular penetration corrosion in stainless steel, the countermeasures against corrosion in the carbon steel heat exchanger and the countermeasures against erosion in the piping in the turbine system caused by steam or drain have been performed and taken so far.

In addition, countermeasures were taken, such as replacement of the sea water system equipment caused by corrosion by sea water including the sea water piping for the emergency diesel generator, the sea water piping in the turbo refrigerator for the containment vessel air recirculation system, etc., as well as attachment of end plugs to the heat exchanger tubes in the main condenser, the auxiliary cooling system, etc., and the countermeasures against corrosion in each equipment in the ventilation system and in the general service water piping. Among them, the major works related to repair and replacement are described below.

① Repair of solidification supply tank

② Replacement of the recombiner in the gaseous waste system, etc.

③ Countermeasures against corrosion in the carbon steel heat exchanger tubes.

④ Countermeasures against erosion in the turbine system piping

(3) Exposure Reduction by Decontamination

System decontamination based on the "diluted solution chemical decontamination method with KRIDECON 203" which has been developed since 1977 was performed to independent equipment such as the refueling machine in the fifth periodic inspection and the heat exchanger in the residual heat removal system in the sixth periodic inspection, and the effects were confirmed.

Because large-scale works for countermeasures against SCC including replacement of the center part of the lower header in the A loop in the reactor cooling system can be hardly performed if system decontamination is not performed, the fuel in the A loop in the reactor cooling system was unloaded in the eighth periodic inspection and system decontamination was performed to reduce exposure.

The system decontamination work was performed on the 24-hour basis by utilizing the shift system, and temperature rise, decontamination, cleanup and flashing were performed in 7 days. The decontamination result was DF 2.4 to 8 which was much better than DF 2 planned at first, and the dose equivalent related to the modification work, the periodic inspection work, etc. in the reactor building was reduced by approximately 80%.

In addition, exposure reduction by decontamination was mode in the seal plug with different decontamination method, the shielding plug, components in the waste treatment equipment, etc. and gave good effects.

(4) Improvement/Extension Works Accompanied by Revision of Technical Standards and Guidelines

The following equipment was extended to satisfy the "Guidelines for the radiation measurement of LWR power plants at accidents" issued by the Atomic Energy Commission.

① High range area monitor in the containment vessel (installed in the fourth periodic inspection performed in 1984)

② High range monitoring post (installed in the fourth periodic inspection performed in 1984)

③ High range stack area monitor (installed in the fifth periodic inspection performed in 1985)

④ Containment vessel sampling equipment (to be installed in the eleventh periodic inspection to be performed in 1992)

(5) Works for Extensions and Modifications

The following equipment was extended/modified to continue operation safety and to improve the reliability and the function.

(a) Installation of Heavy Water Upgrader-I (FY 1979)

The equipment was installed to upgrade the deteriorated heavy water generated by inspection of the equipment, replacement of resin in the heavy water system, etc.

(b) Extension of Spent Fuel Storage Rack (FY 1979)

The spent fuel storage rack was modified to increase the spent fuel storage capacity so that the effects of external factors including operation of the reprocessing plant, transportation of the spent fuel, etc. on operation of "Fugen" can be reduced.

(Spent fuel to be accommodated: 570 assemblies → 730 assemblies)

(c) Extension of Cooling Equipment in Lower Part of Reactor (FY 1979 during the first periodic inspection)

The supply fan, the duct, etc. were extended so that cooling of the secondary seal area of the seal plug in the pressure tube lower extension and cooling of the biological shielding can be reinforced.

(d) Installation of the Mechanical Snapper to the Pressure Tube Inlet/Outlet Pipe (FY 1970, during the first periodic inspection)

The existing hydraulic vibration isolator for the pressure tube inlet/outlet pipe was updated to the mechanical snapper so that the reliability of the vibration isolator can be improved, necessity of maintenance can be eliminated, thereby reducing exposure.

(e) Extension of Heavy Water Tank to Replace Resin (FY 1983)

The tank to store a large quantity of deteriorated heavy water of low concentration generated by replacement of resin in the heavy water system, the isolation chamber, etc. were installed.

(f) Extension of Solid Waste Storage Facility (FY 1984)

The storage facility was extended to assure the storage capacity of the miscellaneous solid wastes.

(g) Improvement of Reactor Feed Water Regulation Valve (FY 1984, during the fourth periodic inspection)

The reactor feed water regulation valve was replaced with a one with excellent

cavitation resistance to improve the operability and the maintainability.

(h) Installation of Heavy Water Cooling Equipment (FY 1985, during the fifth periodic inspection)

The equipment to cool down the heavy water flowing in the purification system was extended as the countermeasures against deterioration of resin in the heavy water purification system caused by radiation decomposition of the heavy water.

(i) Modification of Carbon Dioxide Gas System (FY 1985, during the fifth periodic inspection)

The carbon dioxide gas system was modified from the once-through method to the recirculation method to reduce radiation released.

(j) Extension of Heavy Water Upgrader-II (FY 1986) [III-12] [III-13]

The heavy water upgrader was extended to upgrade the deteriorated heavy water of low concentration.

(k) Installation of Dry Cleaning Unit (FY 1989)

The dry cleaning unit was installed to reduce radiation released from the laundry drain.

(l) Improvement of Drain Filter in Liquid Waste Treatment Equipment (FY 1989)

The drain filter in liquid waste treatment equipment was improved from the pre-coated filter to the hollow yarn film filter.

(m) Extension of Radioactive Waste Storage Facility (FY 1989)

The tank, the equipment and the building were extended to reinforce the storage capacity of used ion exchange resins.

(n) Installation of Miscellaneous Solid Waste Incineration Equipment (FY 1989)

The incineration equipment was installed to reduce the capacity of flammable waste which occupied 60% of miscellaneous waste.

4.5 Work Management

The following work management is performed for inspection, repair, modification, etc.

Before a work is started, the work contents, the safety action, the radiation control, etc. are examined based on the operating procedures, work procedures, etc. of the contractors, and adjustment is performed between the related sections and companies upon necessity.

When the work is performed, the required items are input at a computer terminal (CRT) and the work sheet is issued so that the work can be performed based on the

work sheet. The work contents, the work term, the work place, the radiation control type (The radiation control sheet is issued separately for the radiation control.), the isolation, the related failure sheet No., etc. are input on the work sheet so that communication and confirmation can be performed between the section in charge of execution of the work, the Power Generating Section and the Safety and Chemistry Section. Fig. III.4-4 shows the work sheet processing flow.

4.6 Property Management [III-11]

① In the equipment management, computers were introduced for the equipment ledger management processing when computers were introduced for the plant maintenance record management.

② A certain quantity of replacement parts are required to be always stored as spare parts, so that unexpected failures of the equipment can be immediately handled. The spare parts to be always stored is determined based on the importance of each equipment, the replacement frequency, the period from order to delivery, etc. There are approximately 10,000 items of spare parts in "Fugen". Computers are used to manage the stock status of these large quantity of spare parts.

4.7 Technical Information Management [III-14]

Data on failures and diversified data related to maintenance works are accumulated in the Maintenance Management System (MMS). The piping & Instrument Diagrams (P & ID) of "Fugen" are entered in the isolation management support system, and the technical information such as system diagrams, data on equipment, data on periodic inspection, drawings, documents, etc. is managed unitarily.

4.8 Maintenance Management System (MMS)

The maintenance management system consists of the following systems.

① Maintenance work budget management system

② Periodic inspection work estimation management system

③ Periodic inspection schedule management support system

④ Spare parts management system

⑤ Equipment ledger management system

⑥ Work/failure management system

⑦ Statistics/retrieval system

⑧ Isolation management support system

⑨ Maintenance support system

4.9 Education [III-15]

The "Education Plan of the Fiscal Year" is established every year based on the "Maintenance Section Education Basic Plan" so that the talented personnel with wide knowledge, advanced technology and ability to cope with diversification and upgrading of jobs can be fostered. Education is performed with the major targets to improve the job execution ability, foster the adaptability to new jobs, to give opportunities for ability development and to raise the self-development will. Table III.4-12 shows the contents of education performed in the Maintenance Section.

The examination items and the education items related to fostering of talented personnel in the Maintenance Section are described below.

(1) Execution/Examination Items Related to Fostering of Talented Personnel

① Establishment of the basic education plan in the Maintenance Section

② Preparation of the data on education in the Maintenance Section

③ Transition of external training to internal training

④ Preparation of the education environment (education place, utilization of VTR and CAI, etc.)

⑤ Installation of the maintenance training center

⑥ Education execution items in the Maintenance Section

(2) Education Items in Maintenance Section

① Desk training in the Maintenance Section (beginner's course, intermediate course, senior course)

② On-the-job training in the Maintenance Section (OJT for work procedures, field works, etc.)

③ In-site education (labor safety, radiation control, first-aid method, high pressure gas safety)

④ In-house education (Hierarchical education, supervisor's safety education, radiation control, quality control)

⑤ Education outside company (training center of Japan Atomic Energy Research Institute, training center of the Japan Atomic Power, diversified lectures/ seminars, technical course by manufactures, qualification tests, etc.)

4.10 Evaluation of the Maintenance Management Result

The failures, troubles, etc. reported to the government in accordance with the law are analyzed and compared with those occurred in light water reactors, and the maintenance management result is evaluated from the viewpoint of the operation reliability.

It is obliged by the law that the failures, troubles, etc. occurred in nuclear power plants shall be reported soon by the corresponding utility company, etc. to the government. Three report levels as follows are specified.

① Report in accordance with the law

② Report in accordance with an ordinance issued by the Minister for International Trade and Industry or the Director-General for the Science and Technology Agency

③ Other reports

Among them, the failures and troubles reported from utility companies to the Minister for International Trade and Industry in accordance with the law are analyzed by the Agency of Natural Resources and Energy and described in the "Yearly Report on Nuclear Power Station Operation Management".

The status of failures and troubles were reported to the Director-General for the Science and Technology Agency and the Minister for International Trade and Industry in accordance with the law from start of operation of "Fugen" (March, 1979) are shown in Table III.4-13.

Fig. III.4-5 shows history of the number of reports related to "Fugen" and transition of the average number of reports per commercial power generating plant (including one GCR plant). Many initial failures occurred in commercial power generating plants introduced in the initial stage. The number of failures and troubles reached the peak in the third year after commercial operation had been started, then the number of failures and troubles decreased. In "Fugen", the number of failures and troubles reached the peak also in the third year after operation had been started. By analysis of causes, many of these failures and troubles are regarded as initial failures/troubles.

Fig. III.4-6 shows the rate of each type of failures and troubles in "Fugen" classified by effects given on the operation, as well as in light water reactor plants (BWR, PWR). In light water reactors, the number of automatic shutdowns during operation is almost equivalent to the number of manual shutdowns during operation. In "Fugen", however, the number of automatic shutdowns during operation is approximately 3 times the number of manual shutdowns during operation.

Fig. III.4-7 shows history of the average number of reports on failures and troubles leading to automatic shutdown occurred while the reactor is operating. Fig. III.4-8 shows the occurrence rate classified by failed/troubled systems. Both in light water reactors and in "Fugen", the largest number of automatic shutdowns occurred caused by components belonging to the instrumentation and control system equipment and the steam turbine equipment.

Fig. III.4-9 shows history of the average number of reports on failures and troubles leading to manual shutdown occurred while the reactor is operating. Fig. III.4-10 shows the occurrence rate classified by failed/troubled systems. In light water reactors, the largest number of manual shutdowns occurred caused by leak in components belonging to the reactor cooling system equipment (damages in heat transfer tubes in the steam generator in PWR, fatigue cracking, stress corrosion cracking, etc. in the piping in BWR). In "Fugen", the largest number of manual shutdowns occurred caused by erosion thinning or stress corrosion cracking in the piping and malfunction of the feed water flow regulation valve.

Fig. III.4-11 shows history of the average number of reports on troubles and failures found during shutdown for periodic inspection, etc. Fig. III.4-12 shows the occurrence rate classified by failed/troubled systems. In light water reactors, the largest number of reports were issued caused by thermal fatigue cracking in the reactor pressure vessel nozzle (BWR), stress corrosion cracking in the piping (BWR) and damages in support pins/flexible pins in the control rod guide tube (PWR). In "Fugen", the largest number of reports were issued on failures/troubles in components belonging to the fuel handling equipment.

Figs. III.4-13 to III.4-16 show history of the number of failure sheets issued for each system. History of the failure sheet issue status in each system is described below.

(1) Reactor Proper

The smallest number of failure sheets have been issued so far. Eight failure sheets (5 for machinery and 3 for instrumentation) have been issued.

The major failures in the machinery were malfunction in the selector valve in the seal leak detection unit and corrosion in heat transfer tubes in the heat exchanger in the shielding cooling system.

(2) Reactor Cooling System Equipment

Relatively a small number of failure sheets have been issued so far. Twenty-two failure sheets (9 for machinery, 4 for electric appliance and 9 for instrumentation) have been issued.

The major failures in the machinery were related to the drive unit of the main

steam isolation valve, the solenoid valve, the pump and the partition of heat exchanger header in the reactor coolant clean-up system, etc. In the electric appliance, failures occurred in relays in the pole chamge panel of the reactor recirculation pump. Approximately 70% of these failures were detected by inspections and testings performed during shutdown for periodic inspection, etc.

(3) Reactor Auxiliary System Equipment

The third largest number of failure sheets have been issued so far for this system next to the turbine system equipment and the fuel handling/storage facility equipment. Sixty failure sheets (37 for machinery, 2 for electric appliance and 21 for instrumentation) have been issued.

The number of failures reached the peak in the third year caused by initial failures, and many failures continued to occur as shown in Fig. III.4-14. These failures occurred in the machinery as shown in Table III.4-14.

In the machinery, most failures occurred in dynamic components such as pumps, refrigerators, air-operated valves, control valves, etc. With regard to pumps, many failures were caused by cracking in the weld area in the stator can of the heavy water circulation pump, corrosion in the casing of the reactor auxiliary cooling sea water pump, especially corrosion in the piping and valves in the auxiliary cooling sea water system. With regard to valves such as air-operated valves, control valves, etc., many failures occurred in the heavy water system and the helium system.

(4) Reactor Containment Equipment

Relatively a small number of failure sheets have been issued so far. Twelve failure sheets (7 for machinery, 2 for electric applianc and 3 for instrumentation) have been issued.

The major failures in the machinery were related to the cooling water (sea water) pump in the containment vessel air recirculation system and the reciprocating refrigerator. The major failures in the instruments occurred in pressure switches and timer relays in the annulus ventilation system.

(5) Fuel Handling/Storage Facility Equipment

The second largest number of failure sheets have been issued so far next to the turbine system equipment. Sixty-one failure sheets (32 for machinery, 1 for electric appliance and 28 for instrumentation) have been issued as shown in Fig. III.4-15.

Especially in the fuel handling equipment, 10 failures occurred intensively in FY 1980 and 1981. Seven failures occurred in FY 1980 caused by defective control

due to bugs in programs for computers to control the fuel handling equipment.

(6) Radioactive Waste Treatment Equipment

Forty-six failure sheets (32 for machinery and 14 for instrumentation) have been issued so far. The number of failures reached the peak in the third year due to initial failures as shown in Fig. III.4-15.

In the machinery, many failures occurred caused by clogging in the piping, etc. in the solid system and the liquid system. In the gas system, corrosion was detected in the recombiner, the exhaust gas condenser, etc.

(7) Engineered Safety Features Protection Equipment

A relatively small number of failure sheets have been issued so far. Sixteen failure sheets (3 for machinery and 13 for instrument) have been issued.

A major number of failures occurred in the instrumentation. In the machinery, stress corrosion cracking was detected in the weld area in the pressure boundary piping in FY 1980, and a failure occurred in the drive unit in each of an air-operated valve and a solenoid valve respectively.

(8) Turbine System Equipment

The largest number of failure sheets have been issued so far. Sixty-three failure sheets (47 for machinery, 3 for electric appliance and 13 for instrumentation) have been issued. Many number of failures occurred in the machinery in FY 1985 and 1986 as shown in Fig. III.4-15.

In the machinery, relatively many failures occurred in dynamic components such as pumps, air-operated valves, control valves, solenoid valves, relief valves, etc. though the number of failures also dispersed in each fiscal year. And many failures also occurred in static equipment such as defect in the rubber expansion and the internal piping of the condenser, corrosion in the sheet face of the water chamber manhole in the feed water heater, etc.

Because there are many places of high ambient temperature in the basement of the turbine building, every effort should be made in managing the rubber parts used in the drive system of valves. Damages caused by deterioration occurred in rubber parts of the control air valve of the control valve in FY 1988. After that, it was determined to replace such parts once in 3 to 5 years.

Defective control occurred twice in the turbine control (EHC) circuit. In one case between them, the plant was automatically shut down. Because defect in this system can lead to automatic shutdown of the plant, the single conventional analog circuit: was replaced with a digital multiplexed circuit.

(9) Electrical Equipment

Forty failure sheets (19 for machinery, 16 for electric appliance and 5 for instrumentation) have been issued so far. The number of failures reached the peak in the third year due to initial failures, but decreased after that as shown in Fig. III.4-16.

In the machinery, many initial failures were related to the emergency diesel engine, Such as failures in the air compressor, corrosion by sea water in the air cooler, etc.

(10) Instrumentation and Control Equipment

Fifty-five failure sheets (6 for machinery, 1 for electric appliance and 48 for instrumentation) have been issued so far. The number of failures (initial failures) reached the peak in the fourth year as shown in Fig. III.4-16. These failures at the peak were caused by the instrumentation.

In the electric appliance, only one failure occurred in the power supply unit of the control rod drive mechanism. In the instrumentation, 21 failures occurred in the nuclear instrumentation equipment, 13 failures occurred related to the radiation monitor, and 8 failures occurred related to the control panel. Especially, the indication sensitivity of the local power detection unit got defective in the nuclear instrumentation equipment, the radiation detection unit got failed in the radiation monitoring, and relays got failed in the control panel.

(11) Ventilation Equipment

Fifty-five failure sheets (46 for machinery, 2 for electric appliance and 9 for instrumentation) have been issued so far. The number of failures (initial failures) at the peak is relatively small, and many failures occurred in the machinery as shown in Fig. III.4-16.

As described above, the number of reports on failures and troubles, in accordance with the law reached the peak in FY 1980 (the third year after commercial operation had been started) caused by initial failures, then decreased after that. It is considered that appropriate measures to prevent reoccurrence were taken against initial failures for the corresponding equipment and also for similar equipment.

After FY 1986, history of the number of reports in "Fugen" is almost equivalent to that of the average number of reports in commercial power generating reactors. However, the rate of automatic shutdowns is higher in "Fugen" after FY 1984. The major causes of automatic shutdowns are failures in the control circuit, the protection circuit and the pressure/flow control valves.

Though the number of failures occurred in the control circuit or the protection circuit is relatively small, considerable effects would be given on the plant operation if such a failure occurred. Accordingly, it is required to enhance the preventive maintenance measures. In addition, with regard to the turbine system equipment, the sea water equipment installed in severe corrosive environment, the waste treatment equipment, etc., it is required to check the results of the wall thickness measurement and the overhaul inspection and perform appropriate replacement or repair.

Fig. III.4-1 Fourth Periodic Inspection Results

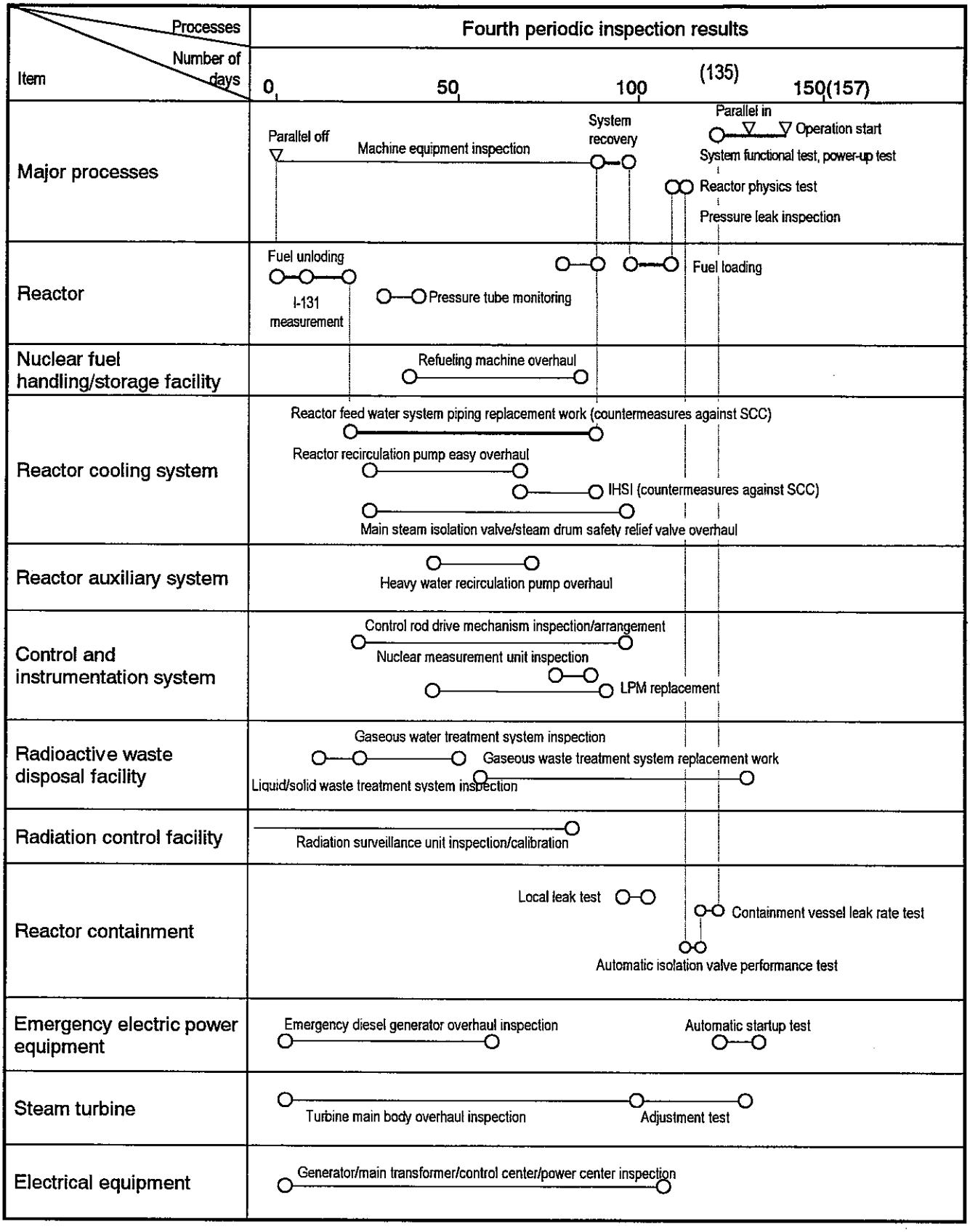


Fig. III.4-2 Seventh Periodic Inspection Results

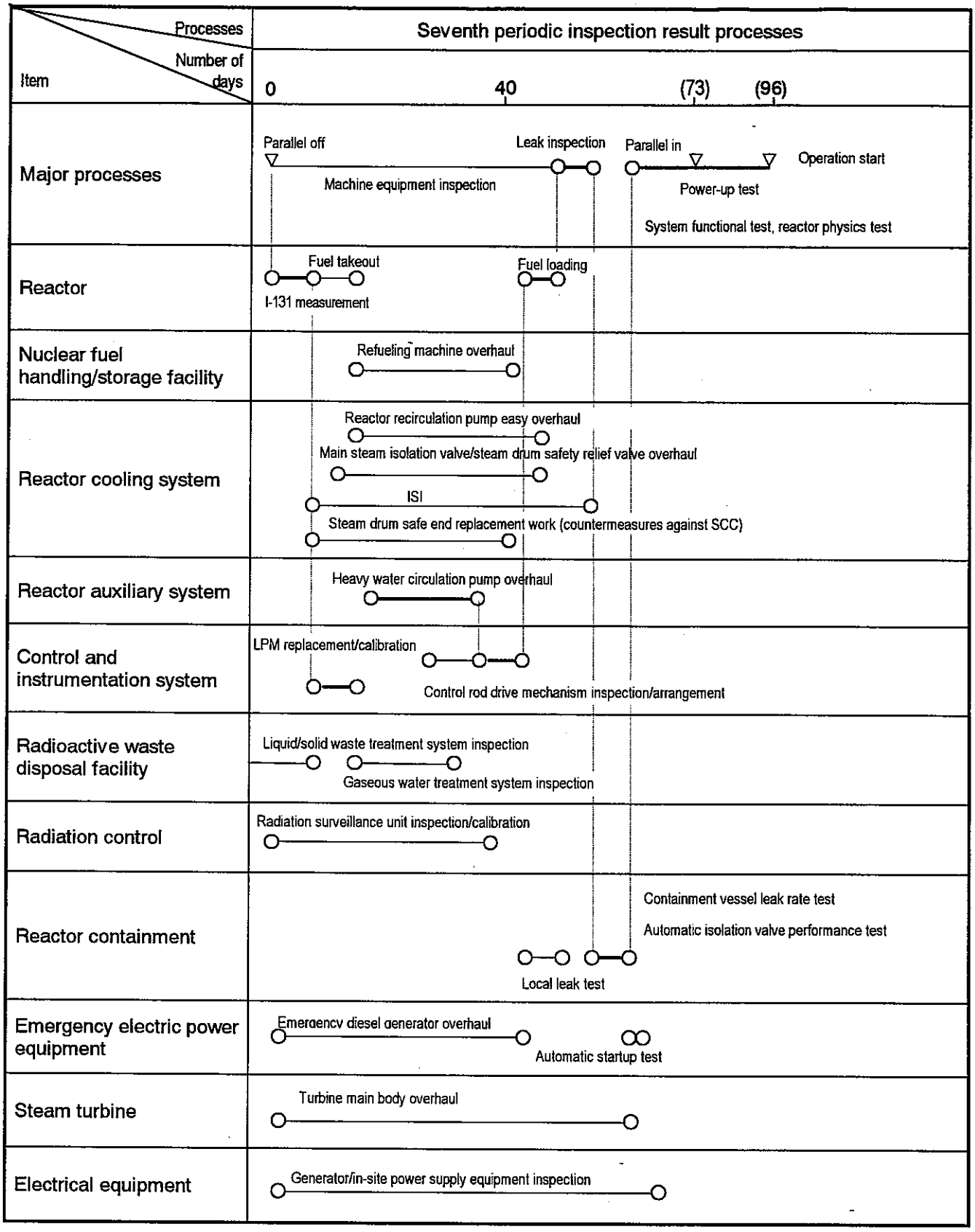
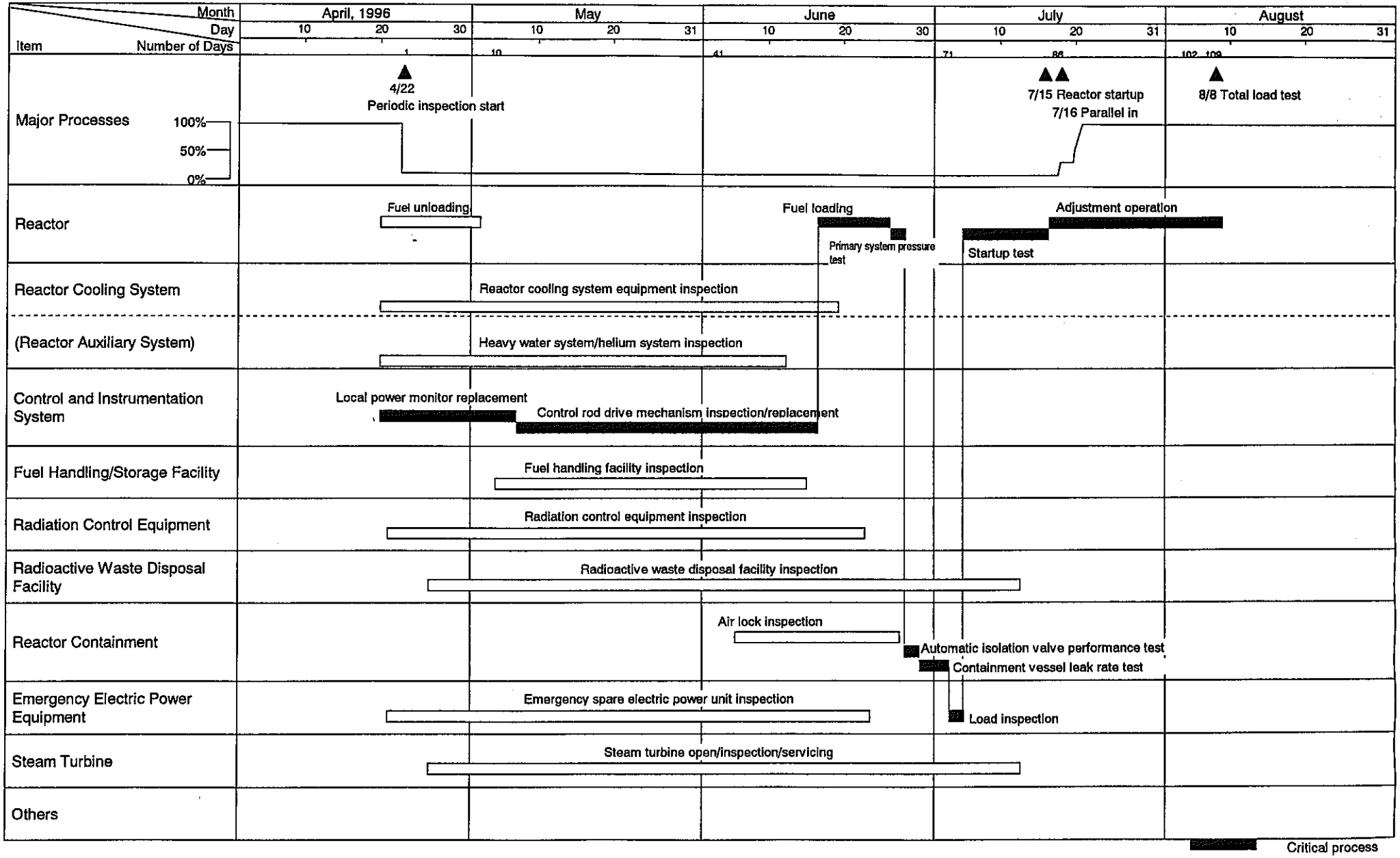


Fig. III.4-2(2) Thirteenth Periodic Inspection Results



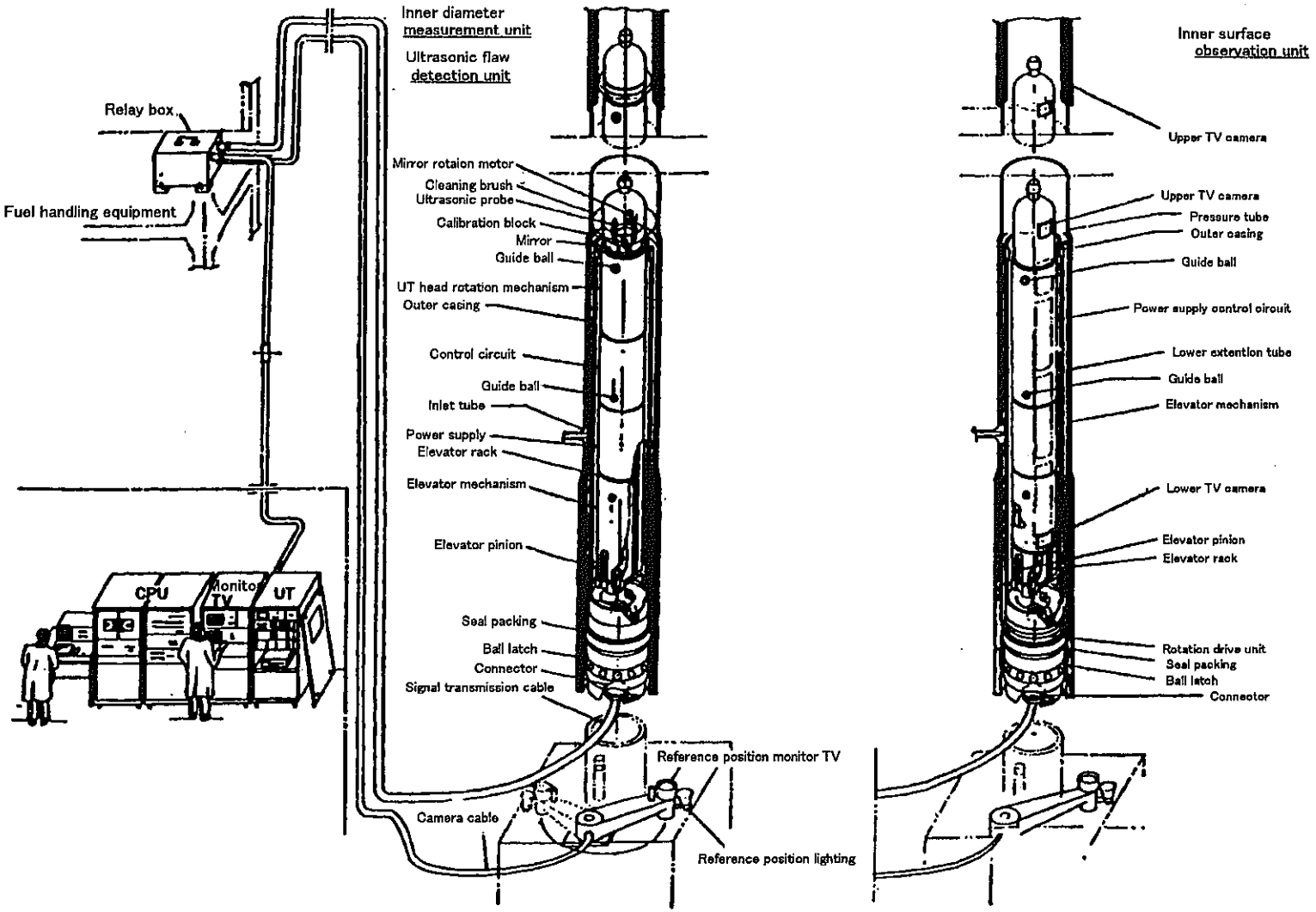
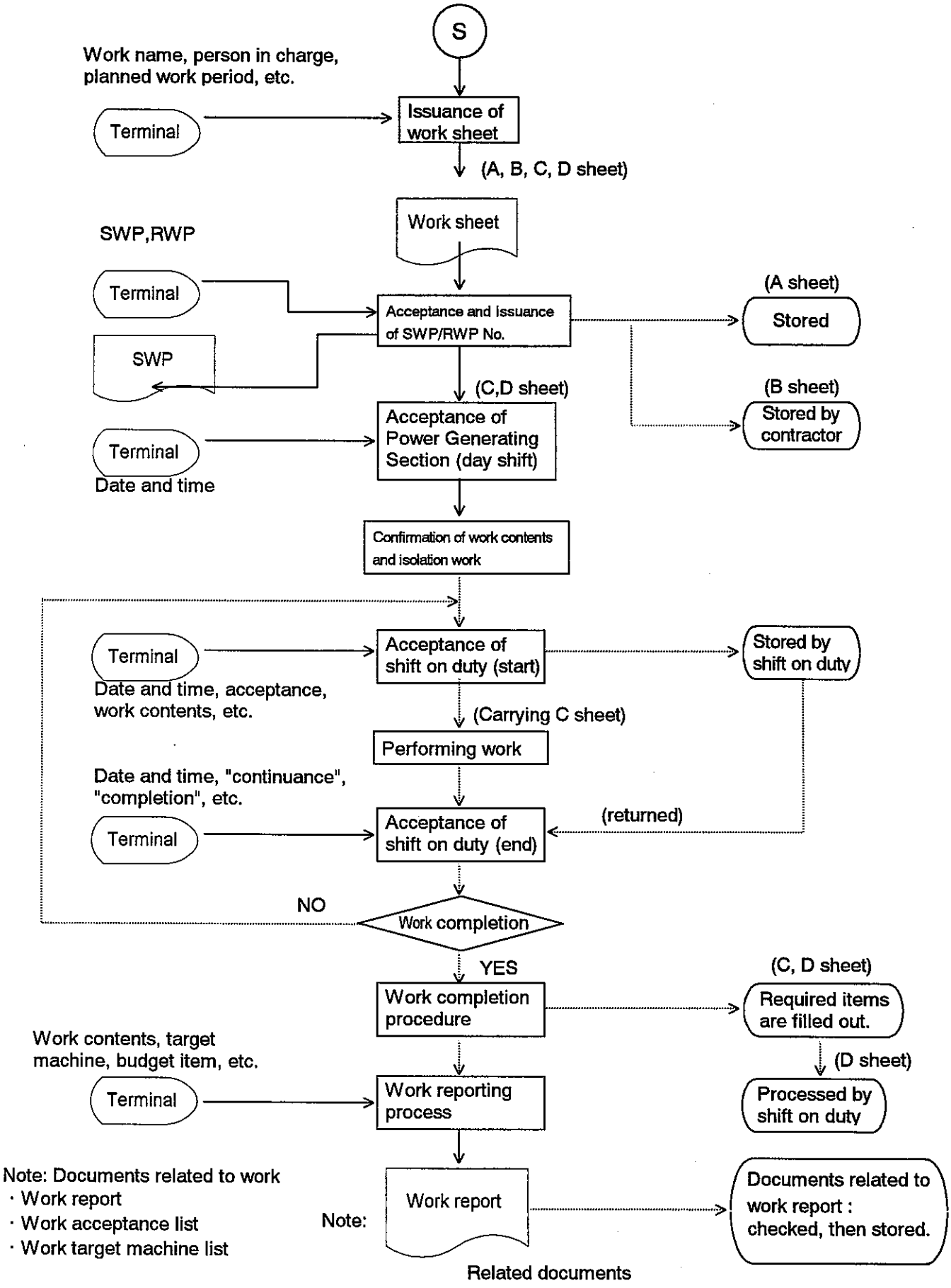


Fig. III.4-3 Pressure Tube Monitoring Unit

Fig. III.4-4 Work Sheet Process Flow



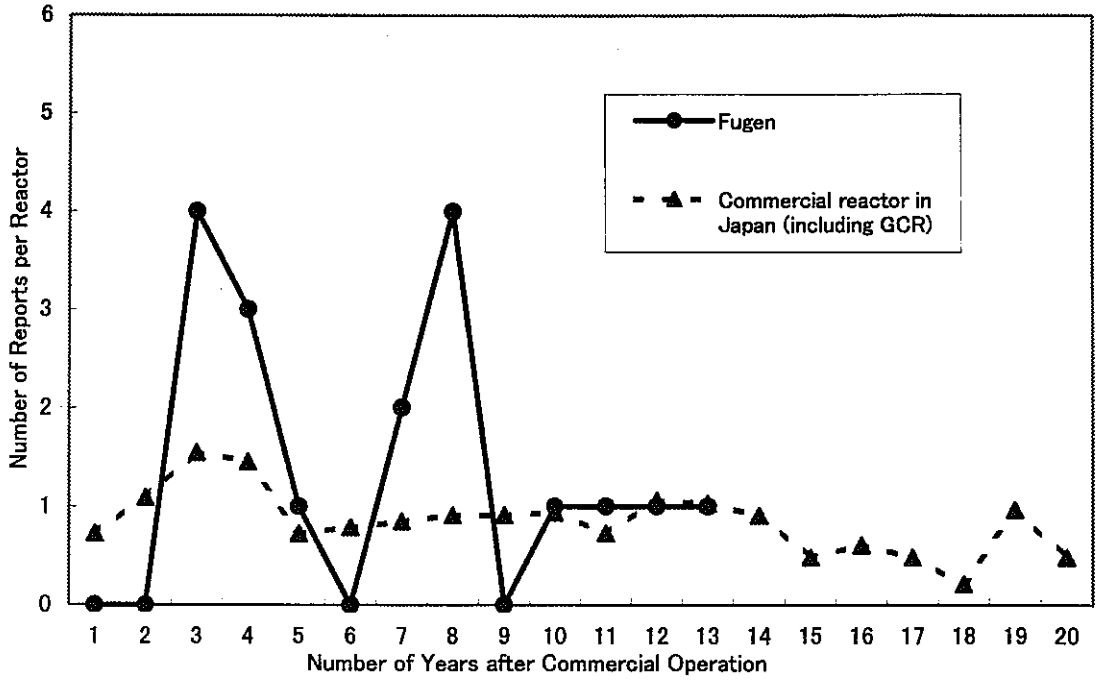


Fig. III.4-5 History of Number of Reports of Failures and Troubles after Commercial Operation

Notes:

1. Number of reports per reactor = Number of reports / Number of reactors at end of fiscal year
2. When counting the "number of years after commercial operation", the fiscal year in which commercial operation is started is regarded as "1". After the 1st fiscal year, each fiscal year is regarded as 1 year

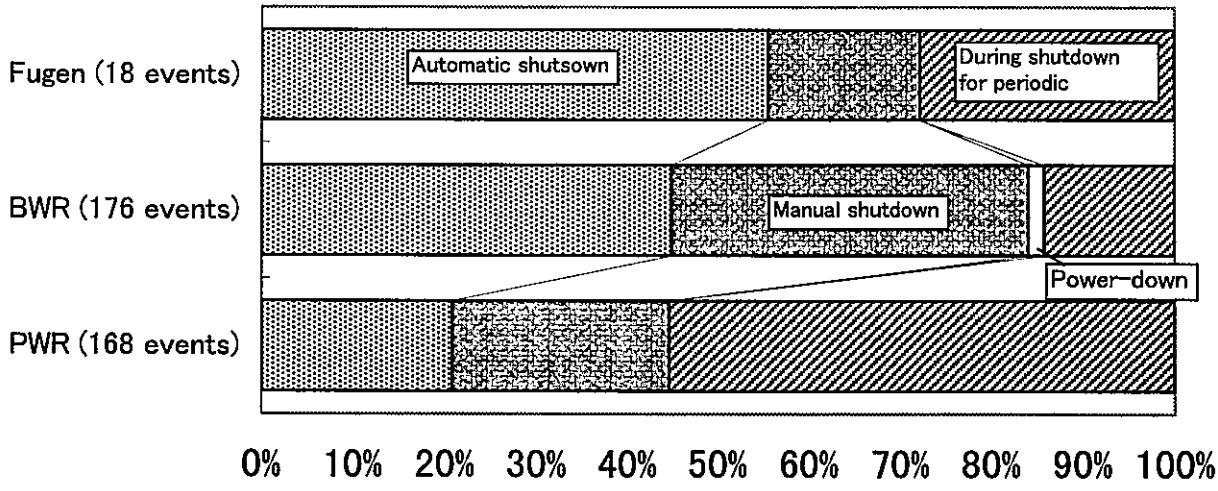


Fig. III.4-6 Occurrence Rate of Failures and Troubles Classified by Effects on Plan Operation

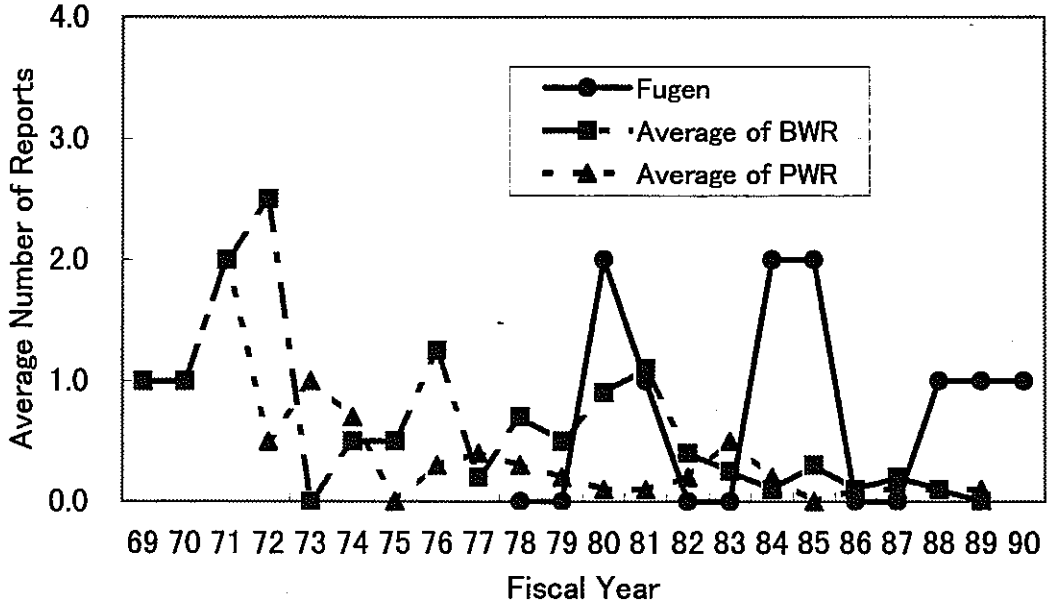


Fig.III.4-7 Histry of Average Number of Reports on Failures and Troubles Leading to Automatic Shutdown (Number of Failures and Troubles/Number of Reactors)

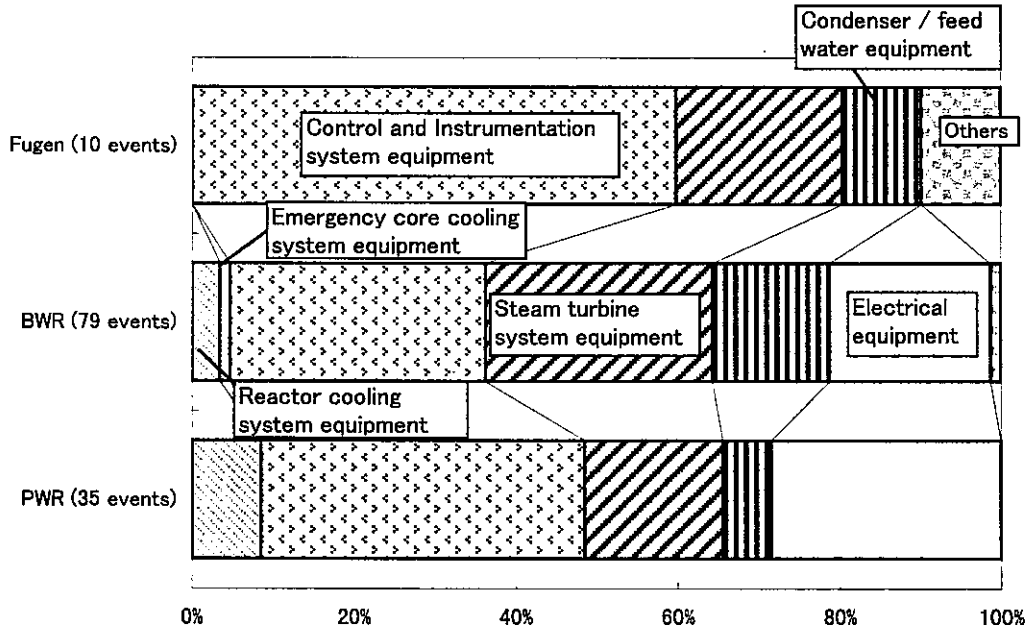


Fig. III.4-8 Rate of Failures and Troubles Leading to Automatic Shutdown Classified by Failed/Troubled Systems

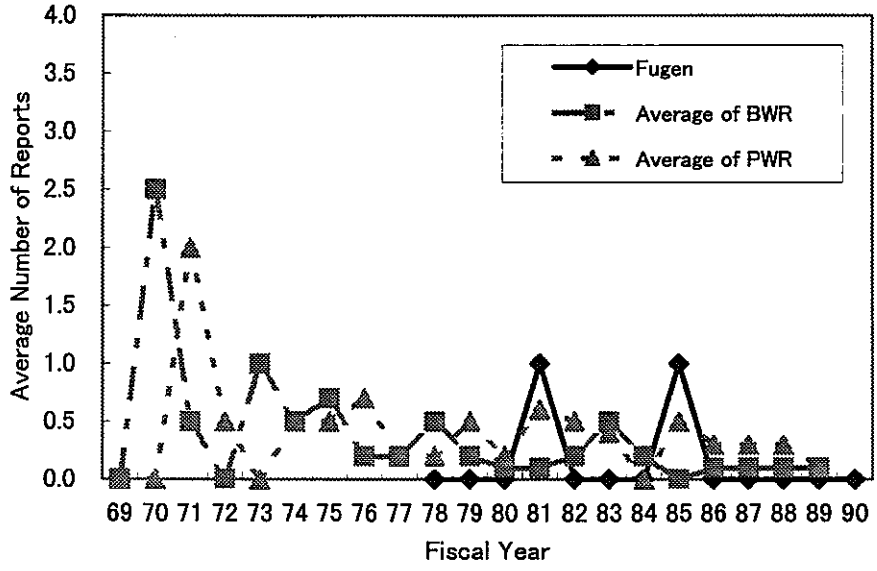


Fig III.4-9 History of Average Number of Reports on Failures and Troubles Leading to Manual Shutdown (Number of Failures and Troubles/Number of Reactors)

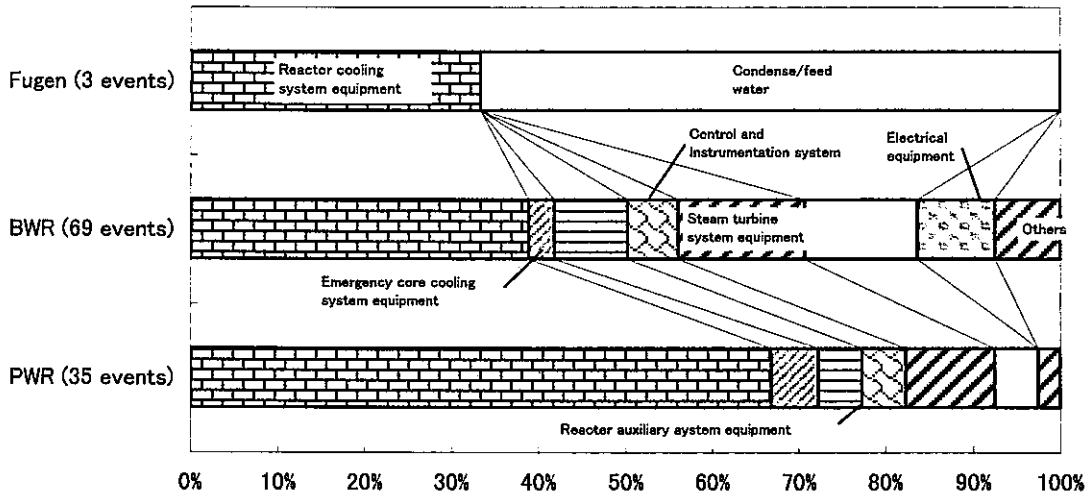


Fig. III.4-10 Rate of Failures and Troubles Leading to Manual Shutdown Classified by Failed/Trouble Systems

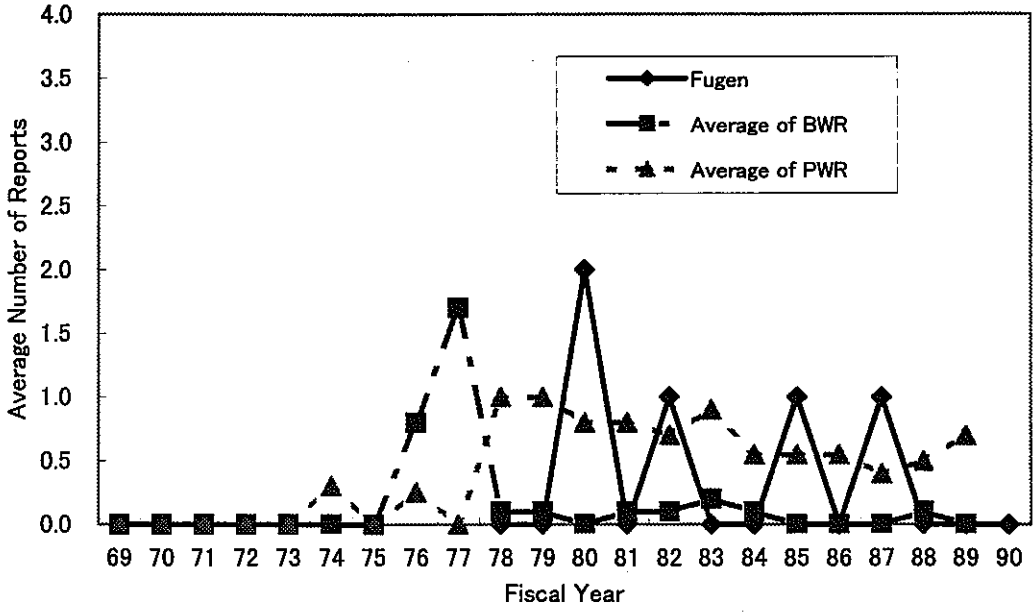


Fig III.4-11 History of Average Number of Reports on Failures and Troubles Found During Shutdown for Periodic Inspection, etc.(Number of Failures and Troubles /Number of Reactors)

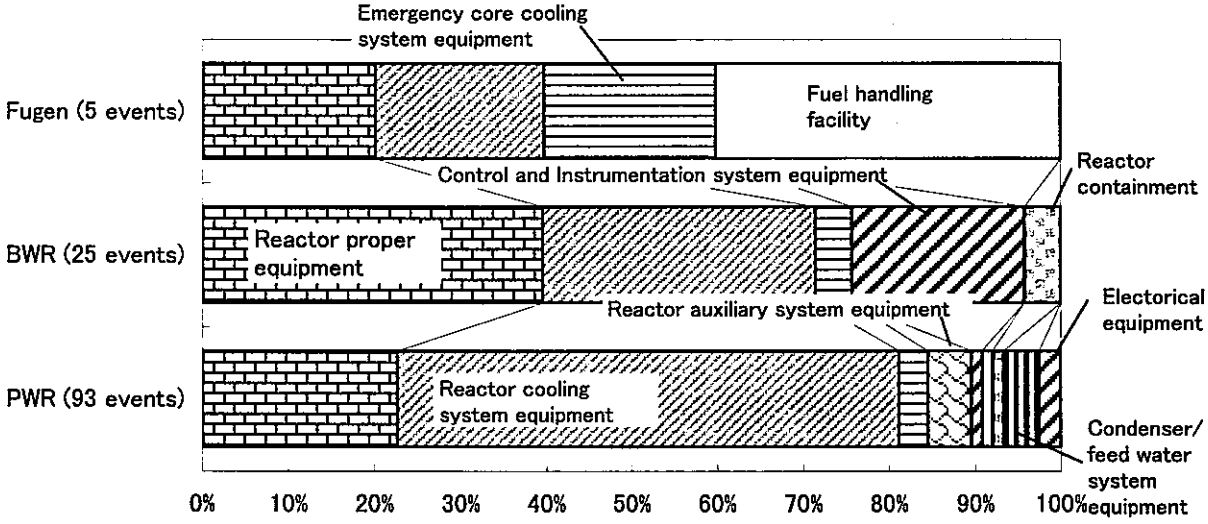


Fig. III.4-12 Rate of Failures and Troubles (Found During Shutdown for Periodic Inspection, etc.) Classified by Failed/Troubled Systems

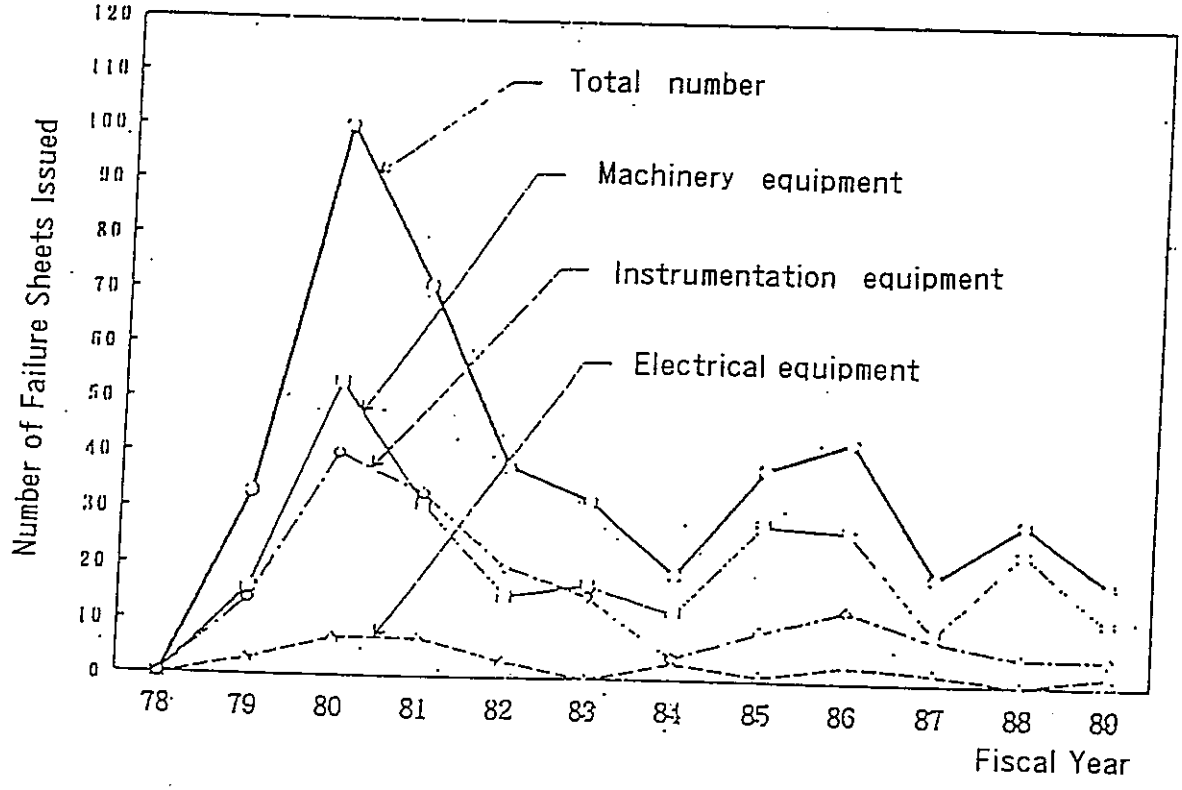


Fig.III 4-13 History of Number of Issued Failure Sheets Classified by System Equipment

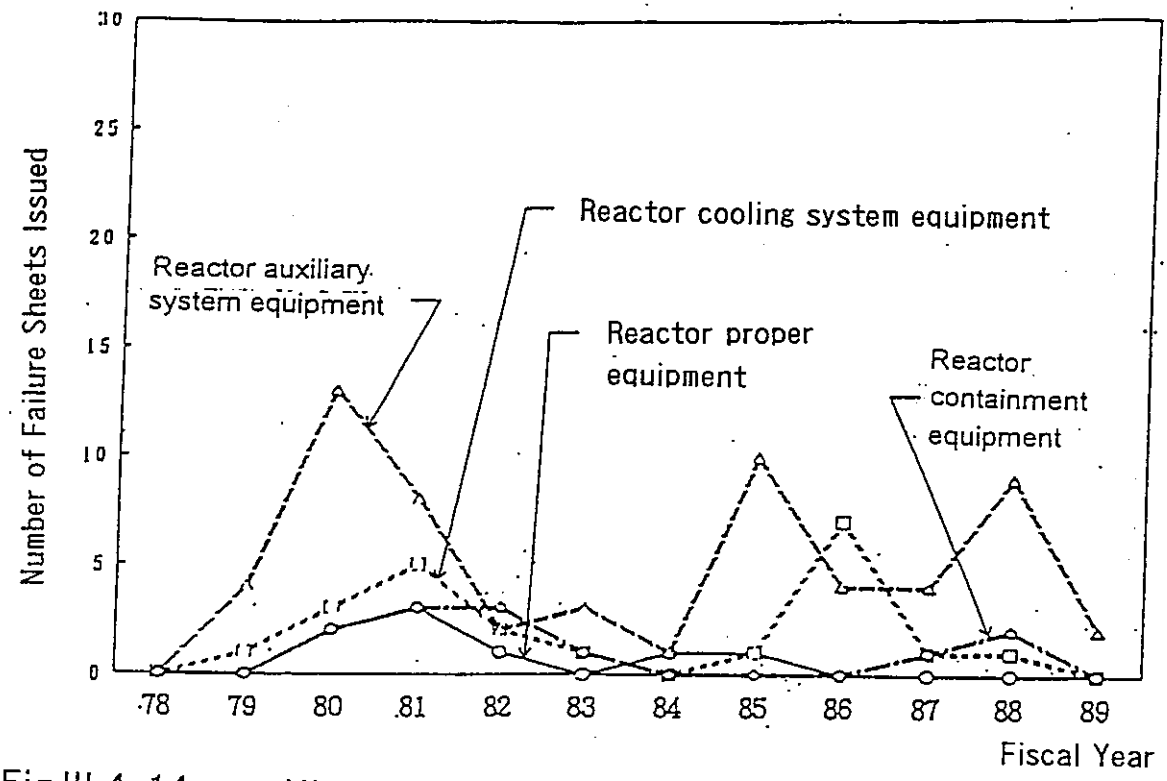


Fig.III 4-14 History of Number of Issued Failure Sheets Classified by System Equipment (1)

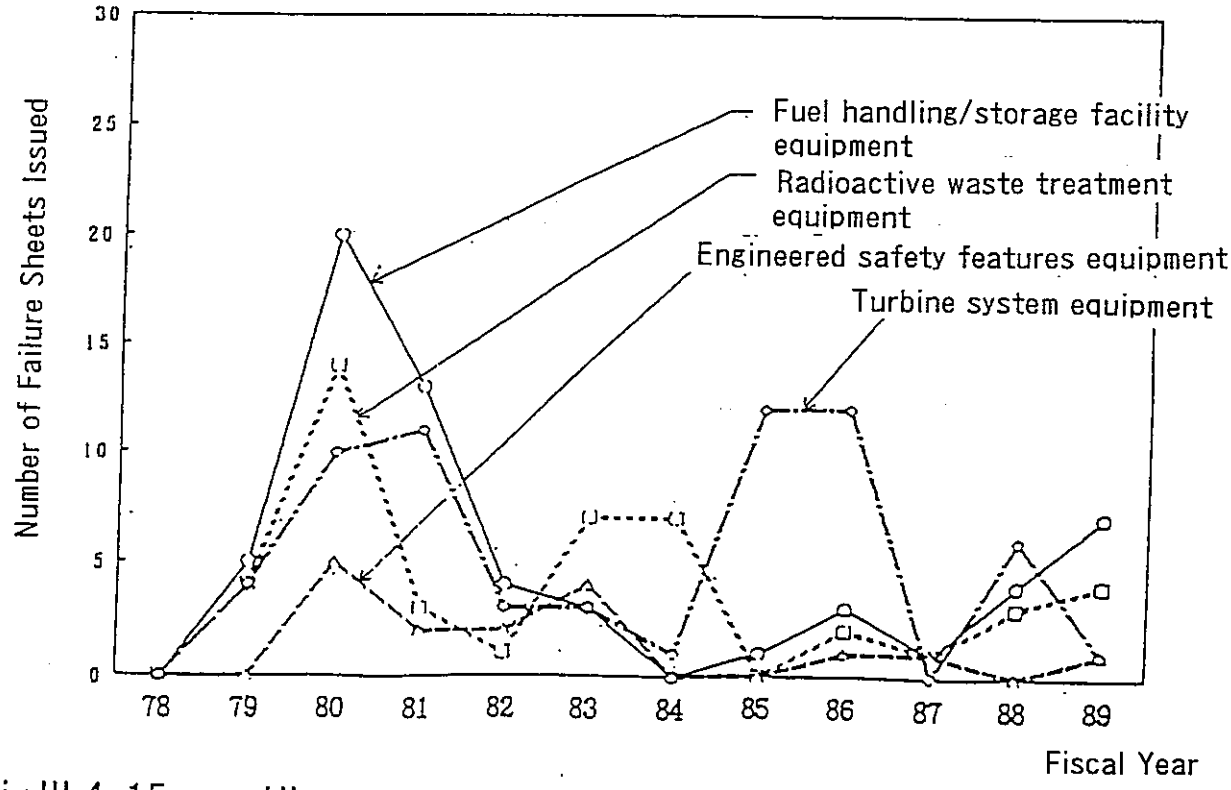


Fig.III 4-15 History of Number of Issued Failure Sheets Classified by System Equipment (2)

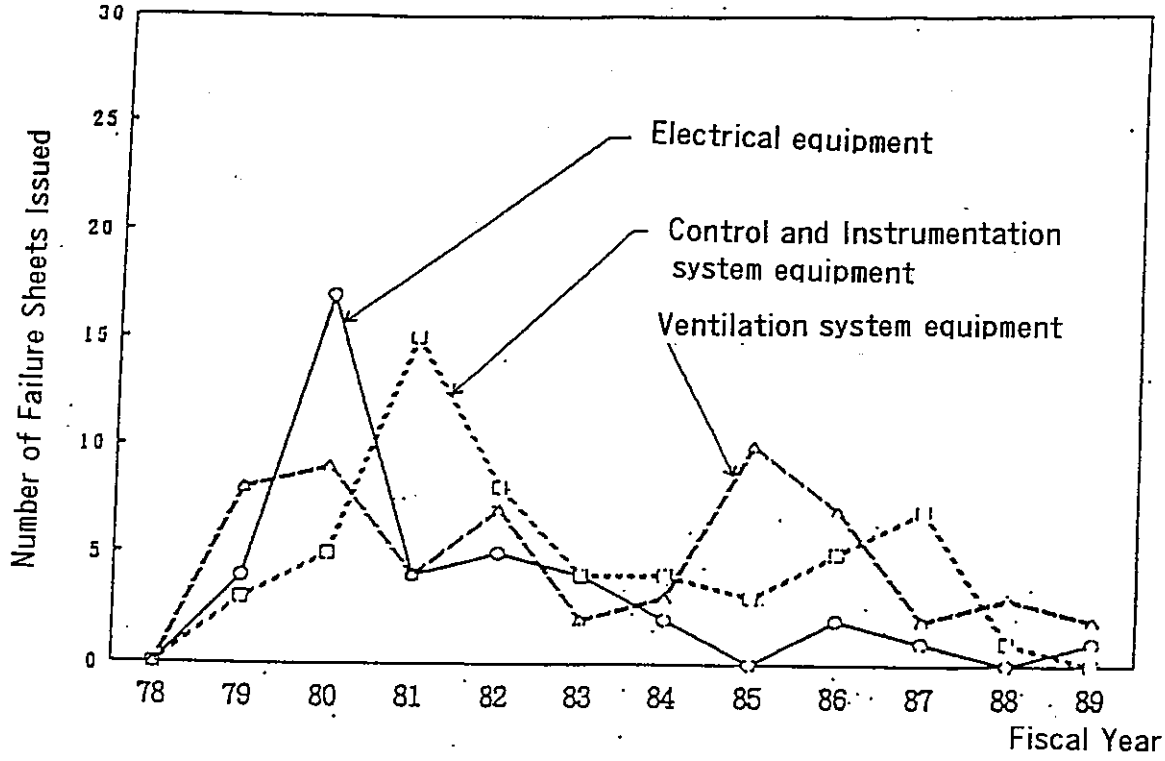


Fig.III 4-16 History of Number of Issued Failure Sheets Classified by System Equipment (3)

Table III.4-1 Outline of Periodic Inspection Execution

[I] Reactor Proper

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-1-1	Calandria tank Leak inspection	111	Analyze the quality of the heavy water flowing in the calandria tank, and check leakage.	Witnessed inspection
SF-1-2	Calandria tube Pressure tube Leak inspection	121	Measure the moisture of carbon dioxide gas flowing at low speed between a calandria tube and a pressure tube during operation at 100% power, and check leakage based on change in the moisture.	Witnessed inspection
SF-1-3	Fuel assembly Sipping inspection	131	Evaluate the concentration record of iodine 131 contained in the reactor coolant measured during operation after the previous periodic inspection as well as the measured value of additional iodine 131 release accompanied by shutdown of the reactor, then make sure that fission products do not leak from the fuel assemblies. In addition, acquire the sample water from the steam drum and each pressure tube if necessary, measure the concentration of iodine 131 contained in the sample water, then confirm that fission products do not leak from the fuel assemblies.	Record check
	Appearance inspection	132	Check the surface of the fuel assemblies using submerged cameras, and make sure that harmful defects are not detected.	Witnessed inspection
	Dimensional inspection	133	With regard to the fuel assemblies for which the dimensional inspection is required based on the result of the appearance inspection, check the dimensions using the light transmission technique or the fuel inspection equipment if required, and make sure that anomaly is not detected in the clearance among fuel elements.	
SF-1-4	Fuel assembly Fuel assembly loading inspection	141	Make sure that each fuel assembly is loaded safely, securely and correctly in the specified position in the reactor in accordance with the loading program.	Record check
SF-1-5	Reactor Maximum excess reactivity inspection	151	Restrain the excess reactivity in the core after refueling using the poison (^{10}B) contained in the heavy water, and measure the critical poison (^{10}B) concentration while the M control rods are located at the 40% position or more and all other control rods are withdrawn out. Evaluate the excess reactivity based on the nuclear design code with the poison (^{10}B) concentration measured, and make sure that any hindrance in operation is not detected.	Record inspection

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-1-6	Reactor Reactor shutdown margin inspection	161	Set the poison (^{10}B) concentration contained in the heavy water in the core after refueling to the concentration at the time when nuclear heating of the reactor is started, withdraw the control rod with the maximum reactivity worth, and confirm subcriticality. In addition, calculate and evaluate the reactivity worth held by all the control rods, and calculate and evaluate the maximum control rod reactivity addition rate	Witnessed inspection Record check
SF-1-10	Reactor coolant pressure boundary Pressure tube assembly Inlet/outlet tube In-service inspection	1101	Perform the visual inspection, the penetrant test and the ultrasonic test to the pressure tube assembly, the piping of inlet/outlet tubes and the support unit contained in the reactor coolant pressure boundary, and make sure that harmful defects are not detected. The inspection items at this periodic inspection shall be those specified in the "Ten-Year Plan for Reactor Coolant Pressure Boundary In-Service Inspection (including the parts specified for the in-service inspection enhancement plan for the countermeasures against stress corrosion cracking)".	Witnessed inspection (UT10%) Record check
SF-1-20	Reactor coolant pressure boundary Pressure tube Pressure tube monitoring	1201	Measure the inner diameter to evaluate the creep characteristics of the pressure tube, and make sure that harmful defects are not detected by performing the visual inspection using optical units and the ultrasonic test.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-2-1	Fuel handling equipment Functional inspection	211	Make sure that the dummy fuel is loaded and unloaded without any problem using refueling machine, the fuel handling equipment and the fuel transfer unit and that the dummy fuel is surely hold when the electric power is lost.	Witnessed inspection
SF-2-2	Fuel handling equipment Refeuling machine Upper head Overhaul	221	Disassemble the upper head of the refueling machine, and make sure that no anomaly is detected in any part.	Record check
	Lower head Overhaul	222	Disassemble the lower head of the refueling machine, and make sure that no anomaly is detected in any part.	Record check
SF-2-3	Fuel handling equipment Transfer unit Transfer chute Leak testing	231	Pressurize the space between the isolation valves of the transfer chute with water, and make sure that drastic pressure reduction is not detected.	Record check
SF-2-4	Fuel storage equipment Functional inspection	241	Operate the new fuel handling crane, the new fuel transfer equipment, the new fuel acceptance unit and the new fuel transportation vehicle, and make sure that no anomaly is detected in the function and the operation.	Record check
	Storage ability confirmation test	242	Check the new fuel storage rack by the visual inspection, and make sure that no anomaly is detected in any part.	Witnessed inspection
SF-2-5	Spent fuel storage facility Functional inspection	251	Operate the cask handling crane and the fuel transfer equipment, and make sure that no anomaly is detected in the function and the operation status. With regard to the fuel transfer equipment, make sure that the dummy fuel is surely hold when the electric power is lost.	Record check
	Storage capacity confirmation test	252	Check the spent fuel storage rack by the visual inspection, and make sure that no anomaly is detected in any part.	Witnessed inspection
SF-2-6	Pool water cooling/purification system Functional inspection	261	Operate the pool water cooling/circulation pump, and make sure that stable operation can be performed at the rated flow rate.	Witnessed inspection (1 systems) Record check (2 systems)

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-3-1	Reactor coolant recirculation system Functional inspection	311	Operate the recirculation pump at a low speed, and make sure that it can operate stably.	Record check
SF-3-2	Reactor coolant recirculation system Steam drum safety relief valve Overhaul	321	Remove the steam drum safety relief valve from the steam drum, disassemble it, and make sure that no abnormality is detected in any portion such as the main valve and the pilot valve, etc.	Record check
	Performance inspection	322	After the overhaul inspection, attach the steam drum safety relief valve to the test unit, pressurize the valve with nitrogen gas, and check the operation of the main valve and the blowout pressure of the pilot valve.	Witnessed inspection (2 units) Record check (4 units)
	Leak testing	323	After the performance inspection, attach the shutoff plate on the exhaust side, pressurize the primary side with nitrogen gas, and measure the leak from the primary side.	Witnessed inspection (2 units) Record check (4 units)
SF-3-3	Reactor coolant recirculation system Reactor coolant recirculation pump Easy overhaul	331	Disassemble the shaft seal unit of the reactor coolant recirculation pump, and make sure that no anomaly is detected in any part.	Record check
	Overhaul	332	Disassemble the reactor coolant recirculation pump, and make sure that no anomaly is detected in any part.	Record check
SF-3-4	Main steam system Main steam isolation valve Overhaul	341	Disassemble the main steam isolation valve, and make sure that no anomaly is detected in any part.	Record check
	Leak inspection	342	Pressurize the main steam isolation valve with air, and measure the leak rate of the valve seat.	Witnessed inspection
	Performance inspection	343	Open and close the main steam isolation valve, and measure the rapid closing time.	Witnessed inspection
SF-3-5	Emergency core cooling system High pressure injection system Functional inspection	351	Startup automatically the high pressure injection pump using the pump automatic startup signal, measure the time until the rated flow rate is achieved, and make sure using the test line that stable operation can be performed at the rated flow rate.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	352	Open and close the inspection valve in the high pressure injection system, and make sure that no anomaly is detected during the opening/closing and the operation.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-3-6	Emergency core cooling system High pressure injection system High pressure injection pump Overhaul	361	Disassemble the high pressure injection pump, and make sure that no anomaly is detected in any part.	Record check
SF-3-7	Emergency core cooling system High pressure system Main valve Overhaul	371	Disassemble the inspection valve in the high pressure injection system, and make sure that no anomaly is detected in any part.	Record check
SF-3-8	Emergency core cooling system Low pressure injection system Functional inspection	381	Startup automatically the low pressure injection pump using the pump automatic startup signal, measure the time until the rated flow rate is achieved, and make sure using the test line that stable operation can be performed at the rated flow rate.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	382	Open and close the inspection valve in the low pressure injection system, and make sure that no anomaly is detected during the opening/closing and the operation.	Record check
SF-3-9	Emergency core cooling system Low pressure injection system Low pressure injection pump Overhaul	391	Disassemble the low pressure injection pump, and make sure that no anomaly is detected in any part.	Record check
SF-3-10	Emergency core cooling system Low pressure injection system Main valve Overhaul	3101	Disassemble the inspection valve in the low pressure water injection system, and make sure that no anomaly is detected in any part.	Record check
SF-3-11	Emergency core cooling system Rapid injection system Functional inspection	3111	Open automatically the accumulator outlet valve using the automatic injection signal, and make sure using the test line that injection can be correctly performed.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	3112	Open and close the inspection valve in the rapid injection system, and make sure that no anomaly is detected during the opening/closing and the operation.	Record check
SF-3-12	Emergency core cooling system Rapid injection system Main valve Overhaul	3121	Disassemble the inspection valve in the rapid injection system, and make sure that no anomaly is detected in any part.	Record check

[III] Reactor Cooling System (6/18)

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-3-13	Isolation cooling system Functional inspection	3131	Startup automatically the isolation cooling pump using the pump automatic startup signal, open the test valve after startup, and make sure using the test line that stable operation can be performed at the rated flow rate.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	3132	Open and close the inspection valve in the isolation cooling system, and make sure that no anomaly is detected during the opening/closing and the operation.	Record check
SF-3-14	Isolation cooling system Isolation cooling pump Overhaul	3141	Disassemble the isolation cooling pump, and make sure that no anomaly is detected in any part.	Record check
SF-3-15	Isolation cooling system Main valve Overhaul	3151	Disassemble the inspection valve in the isolation cooling system, and make sure that no anomaly is detected in any part.	Record check
SF-3-16	Residual heat removal system Functional inspection	3161	Operate the residual heat removal pump, and make sure that stable operation can be performed at the rated flow rate.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	3162	Open and close the inspection valve in the residual heat removal system, and make sure that no anomaly is detected under the opening/closing and the operation.	Record check
SF-3-17	Steam discharge pool cooling system Functional inspection	6171	Operate the steam discharge pool cooling pump, and make sure that stable operation can be performed at the rated flow rate.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	3172	Open and close the main valve in the steam release pool cooling system, and make sure that no anomaly is detected during the opening/closing and the operation.	Record check
SF-3-18	Steam discharge pool cooling system Steam discharge pool cooling pump Overhaul	3181	Overhaul the steam discharge pool cooling pump, and make sure that no anomaly is detected in any part.	Record check
SF-3-19	Steam discharge pool cooling system Main valve Overhaul	3191	Disassemble the inspection valve in the steam discharge pool cooling system, and make sure that no anomaly is detected in any part.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-3-20	Reactor feed water system Reactor feed water pump Overhaul Functional inspection	3201 3202	Disassemble the reactor feed water pump, and make sure that no anomaly is detected in any part. Set two among three reactor feed water pumps operating (or simulation) , input the pump trip signal to either one pump for simulation, make sure that the corresponding pump is tripped, and make sure by checking the indicator lamp that the spare pump is automatically started up.	Record check Witnessed inspection (1 unit) Record check (2 units)
SF-3-21	Reactor feed water system Main feed water flow regulating valve Functional inspection	3211	Open and close the main feed water flow regulation valve in the reactor feed water system, and make sure that no anomaly is detected in the operation.	Record check
SF-3-22	Reactor auxiliary cooling system Functional inspection	3221	Issue the dummy signal for external electric power loss, and make sure that the reactor auxiliary cooling water pump and the reactor auxiliary cooling sea water pump are stopped and that these pumps are automatically started up after the specified time has passed since the emergency diesel generator breaker is turned on.	Record check
SF-3-23	Steam turbine and accessory equipment Appearance inspection	3231	Perform the visual inspection of the steam turbine as well as the components, the piping and the valves of the accessory equipment, and make sure that no anomaly is detected in any part.	Record check
SF-3-24	Turbine bypass valve Performance inspection	3241	Open and close the turbine bypass valve, and make sure that no anomaly is detected in the valve operation.	Witnessed inspection
SF-3-50	Reactor coolant pressure boundary Leak testing	3501	Pressurize the reactor coolant pressure boundary above the working pressure (68 kg/cm2) using the reactor coolant cleanup system charge pump, and make sure that considerable leak and deformation are not detected in the components, the piping and the valves.	Witnessed inspection
SF-3-60	Reactor coolant pressure boundary Reactor coolant recirculation system Reactor coolant clean-up system Main feed water system In-service inspection	3601	Perform the visual inspection, the penetrant test and the ultrasonic tests of the components, the piping, the valves and the support units in the reactor coolant recirculation system, the reactor coolant clean-up system and the main feed water system contained in the reactor coolant pressure boundary, and make sure that harmful defects are not detected. The inspection items at this periodic inspection shall be those specified in the "Ten-Year Plan for Reactor Coolant Pressure Boundary In-Service Inspection (including the parts specified for the in-service inspection enhancement plan for the countermeasures against stress corrosion cracking)".	Witnessed inspection (UT10%) Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-3-70	Reactor coolant pressure boundary Main steam system In-service inspection	3701	Perform the visual inspection and the ultrasonic tests of the components, the piping, the valves and the support units in the main steam system contained in the reactor coolant pressure boundary, and make sure that harmful defects are not detected. The inspection items at this periodic inspection shall be those specified in the "Ten-Year Plan for Reactor Coolant Pressure Boundary In-Service Inspection (including the parts specified for the in-service inspection enhancement plan for the countermeasures against stress corrosion cracking)".	Witnessed inspection (UT10%) Record check
SF-3-80	Reactor coolant pressure boundary Emergency core cooling system Isolation cooling system Residual heat removal system In-service inspection	3801	Perform the visual inspection, the penetrant test and the ultrasonic tests to the piping and the support units in the emergency core cooling system, the isolation cooling system and the residual heat removal system contained in the reactor coolant pressure boundary, and make sure that harmful defects are not detected. The inspection items at this periodic inspection shall be those specified in the "Ten-Year Plan for Reactor Coolant Pressure Boundary In-Service Inspection (including the parts specified for the in-service inspection enhancement plan for the countermeasures against stress corrosion cracking)".	Witnessed inspection (UT10%) Record check
SF-3-90	Outside reactor coolant pressure boundary In-service inspection	3901	Perform the visual inspection, the penetrant test and the ultrasonic tests of Class 3 components outside the reactor coolant pressure boundary, and make sure that harmful defects are not detected. Perform the wall thickness inspection and the penetrant test of the major components, the piping and the valves in the normal service system outside the reactor coolant pressure boundary, and make sure the integrity. The inspection items at this periodic inspection shall be those specified in the "Ten-Year Plan for Reactor Coolant Pressure Boundary In-Service Inspection (including the psats specified for the in-service inspection enhancement plan for the countermeasures against stress corrosion cracking)".	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-4-1	Heavy water cooling system Functional testing	411	Operate the heavy water circulation pump, and make sure that no abnormality is detected in the specified function and any part.	Record check
	Leak testing	412	Operate the heavy water circulation pump, and make sure by the visual inspection that heavy water does not leak from the components, piping, valves, etc. Measure the tritium concentration in the air in the room in which the components, piping, valves, etc. are installed, and make sure that no leak is detected.	Witnessed inspection
SF-4-2	Helium circulation system Communicating valve for heavy water dump Performance inspection	421	Open and close the communicating valve for heavy water dump, and make sure that no anomaly is detected during the fully opening and the operating.	Witnessed inspection (2 units) Record check (6 units)

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-5-1	Control rod drive mechanism Overhaul inspection	511	Disassemble the control rod drive mechanism, and make sure that no anomaly is detected in any part.	Record check
	Performance inspection	512	Withdraw and insert the control rods after their installation in the reactor, and make sure that the drive speed is within the criteria range.	Witnessed inspection (5 units) Record check (44 units)
	Scram inspection	513	Measure the scram characteristics while they are installed in the reactor, and make sure that the scram time is within the criteria range.	Witnessed inspection (5 units) Record check (44 units)
SF-5-2	Nuclear instrumentation equipment Set value confirmation testing	521	Input dummy signals to the instrumentation loop of the nuclear instrumentation equipment, and make sure that no anomaly is detected in the alarm function and any set value in the trip circuit.	Witnessed inspection SUM: 1 channel PUM: 1 channel RPM: 1 channel TPM: 1 channel RSM: 1 channel Record check SUM: 3 channels PUM: 5 channels RPM: 7 channels TPM: 1 channel RSM: 1 channel
	Interlock inspection	522	Input dummy signals to the nuclear instrumentation equipment, etc., and make sure that no anomaly is detected in the control rod withdrawal stoppage function.	Witnessed inspection (1 item) Record check (6 items)
SF-5-3	Safety protection system Calibration inspection	531	Input dummy signals to the instrumentation loop in the safety protection system, perform calibration inspection, and make sure that no anomaly is detected in the instrumentation loop.	Record check
	Set value confirmation testing	532	Input dummy signals to the instrumentation loop in the safety protection system, and make sure that no anomaly is detected in the alarm function and any set value in the trip circuit.	Witnessed inspection (5 items) Record check (6 items)

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
	Performance inspection	533	(1) Scram exclusion test Input dummy signals to the instrumentation loop in the safety protection system, and make sure that no anomaly is detected in the operation of the wipeout circuit. (2) Logical circuit test Input dummy signals to the instrumentation loop in the safety protection system, and make sure that no anomaly is detected in the logical circuit. (3) Scram test Input dummy signals to the instrumentation loop in the safety protection system to actuate the logical circuit related to reactor scram and scram the control rod, then make sure that no anomaly is detected in the operation of the scram function. (4) Heavy water dump function confirmation test Input dummy signals to the instrumentation loop in the safety protection system to actuate the logical circuit related to heavy water dump and open completely the helium communicating valve, then make sure that no anomaly is detected in the heavy water dump function.	Record check Witnessed inspection (3 items) Record check (81 items) Witnessed inspection Witnessed inspection
SF-5-4	Safety protection system Total interlock inspection	541	Actuate the operation elements of the reactor, the turbine and the motor protection system inputting dummy signals, and make sure that no anomaly is detected in the shutdown interlock function among the reactor, the turbine and the generator.	Witnessed inspection
SF-5-5	Air compressor equipment for control Functional inspection	551	(1) Air compressor spare unit automatic startup test Simulate low air pressure of the pneumatic system for control, and make sure that the spare unit of the air compressor is automatically started up. (2) Common air backup valve performance test Simulate low air pressure in the pneumatic system for control, and make sure that the air backup valve is automatically opened. (3) Alarm confirmation test Simulate low air pressure of the pneumatic system for control, and make sure that alarm transmitted.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-6-1	Gaseout waste treatment system Functional inspection	611	During rated power operation, make sure based on the operation data that each equipment/system is stable.	Record check
SF-6-2	Liquid waste treatment system Alarm/interlock inspection	621	Input dummy or actual signals to the instrumentation loop related to alarm and interlock in the liquid waste treatment system, and make sure that no anomaly is detected in the alarm function and the interlock function.	Record check
	Storage capacity confirmation testing	622	Perform the visual inspection of various tanks in the liquid waste treatment system, and make sure that no anomaly is detected in any part.	Witnessed inspection (3units) Record check (10 units)
SF-6-3	Liquid waste treatment system Evaporation/ concentration equipment Leak testing	631	Perform the leak testing of the evaporation/ concentration equipment (heating basket), and make sure that considerable leak is not detected.	Record check
	Treatment capacity confirmation testing	632	Supply liquid waste to the evaporator-concentrator from the liquid waste neutralization tank, operate the evaporator-concentrator at the rated condition, and make sure that the treatment capacity satisfies the criteria.	Witnessed inspection (1 unit) Record check (1 unit)
SF-6-4	Solid waste treatment system Treatment capacity confirmation testing	641	Operate the solidification unit at the rated condition, and make sure that the treatment capacity satisfies the criteria.	Record check
	Storage capacity confirmation testing	642	Perform the visual inspection of various storage tanks in the solid waste treatment system, and make sure that no anomaly is detected in any part.	Witnessed inspection (2units) Record check (4 units)
SF-6-5	Solid waste storage facility Radiation control status inspection	651	(1) Spatial dose rate inspection Measure the spatial dose rate in the controlled area boundary in the solid waste storage facility, and make sure that no anomaly is detected. (2) Surface contamination density inspection Measure the density of the radioactive materials adhered on the surface of the solid waste storage facility using the smear method, and make sure that no anomaly is detected.	Record check
	Storage capacity confirmation testing	652	Perform the visual inspection to the solid waste treatment system, and make sure that no anomaly is detected in the storage status and any part.	Witnessed inspection

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-6-6	Leak detection unit and alarm unit for liquid radioactive wastes Functional inspection	661	Actuate the leak detection unit, and make sure that alarm is transmitted and the indicator lamp is lighted at an alarm set value.	Record check
SF-6-7	Solid waste treatment system Miscellaneous waste incineration equipment Treatment capability confirmation test	671	Put flammable miscellaneous solid wastes into the incinerator, operate the incineration equipment at the rated condition, and make sure that the treatment capacity satisfies the criteria.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-7-1	Instrumentation equipment for radiation control Area monitor Process monitor Radiation source calibration inspection	711	Make sure that the value indicated by the inspection monitor of the instrumentation equipment for radiation control is within the criteria range with regard to the reference dose rate or the reference count rate.	Record check
	Set value confirmation test	712	Make sure that the alarm set value of the inspection monitor of the instrumentation equipment for radiation control is within the criteria range.	Witnessed inspection
	Interlock functional inspection	713	Input dummy signals to the instrumentation loop related to interlock, and make sure that no anomaly is detected in the alarm function and the interlock function.	Record check
SF-7-2	Instrumentation equipment for radiation control Monitoring post Machine mounted radiation observation vehicle Radiation source calibration inspection	721	Make sure that the values indicated by the monitoring post and the monitor mounted the radiation observation vehicle are within the criteria range with regard to the reference dose rate.	Record check
	Set value confirmation test	722	Make sure that the alarm set value of the monitoring post is within the criteria range.	Witnessed inspection
SF-7-3	Containment vessel air circulation system Functional inspection	731	Operate the circulation fan, and make sure that no anomaly is detected in any part of the system and that operation is changed over to the system for emergency.	Record check
	Filter performance inspection	732	Calculate the total efficiency of charcoal filter based on the radioactive iodine removal efficiency test using sample active carbon and the leak test of the charcoal filter in the actual system, and make sure that no anomaly is detected in the filter performance.	Record check
SF-7-4	Annulus exhaust system Functional inspection	741	Operate the exhaust fan in the annulus exhaust system, and make sure that no anomaly is detected in the operation status of each part.	Record check
	Filter performance inspection	742	Calculate the total efficiency of charcoal filter based on the radioactive iodine removal efficiency test using sample active carbon and the leak test of the charcoal filter in the actual system, and make sure that no anomaly is detected in the filter performance.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-7-5	Main control room ventilation system Functional inspection	751	Manipulate the A (B) main control room air conditioning selector switch, and make sure that the main control room ventilation system is changed over to the filter for emergency.	Record check
	Filter performance inspection	752	Calculate the total efficiency of charcoal filter based on the radioactive iodine removal efficiency test using sample active carbon and the leak test of the charcoal filter in the actual system, and make sure that no anomaly is detected in the filter performance.	Record check
SF-7-6	Emergency gas treatment system Functional inspection	761	Input dummy signals for high tritium concentration, and make sure that the blower is automatically started and that no anomaly is detected in the specified function and each part.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-8-1	Reactor containment vessel Total leak rate testing	811	Pressurize the reactor containment vessel with air to the inspection pressure, measure the temperature, the humidity and the pressure of the air inside and outside the reactor containment vessel, obtain the leak quantity (%), and make sure the leak rate per day based on the leak quantity. The inspection method and the leak calculation method shall be in accordance with the JEAC 4203-1974 "Leak testing for reactor containment vessel" (absolute pressure method).	Witnessed inspection
SF-8-2	Reactor containment vessel penetration Automatic isolation valve overhaul	821	Disassemble the automatic isolation valve, and make sure that no anomaly is detected in any part.	Record check
SF-8-3	Reactor containment vessel penetration Automatic isolation valve performance inspection	831	(1) Performance inspection using general isolation signal Generate general isolation signals using dummy signals for high-high reactor containment vessel pressure, and make sure that the isolation valve is completely closed. (2) Performance inspection using rupture isolation signal Generate rupture isolation signals caused by rupture dummy signals of low-low 3 steam drum level and low-low 1 steam drum pressure, and make sure that the isolation valve is completely closed	Witnessed inspection
SF-8-4	Reactor containment vessel spray system Functional inspection	841	Start up automatically the reactor containment vessel spray pump using the pump automatic startup signal, open the test valve, and make sure that stable operation can be obtained at the rated flow rate in the test line.	Witnessed inspection (1 system) Record check (1 system)
	Main valve performance inspection	842	Open and close the valves, and make sure that no anomaly is detected during the opening/closing and the operation.	Record check
SF-8-5	Reactor containment vessel spray system Spray pump Overhaul	851	Disassemble the reactor containment vessel spray pump, and make sure that no anomaly is detected in any part.	Record check
SF-8-6	Reactor containment vessel spray system Main valves Overhaul	861	Disassemble the inspection valves of the reactor containment vessel spray system, and make sure that no anomaly is detected in any part.	Record check
SF-8-7	Reactor containment vessel pressure relief equipment Functional inspection	871	Open and close the isolation valve in the reactor containment vessel pressure relief equipment, and make sure that the valve is correctly operated.	Record check

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-9-1	Emergency diesel generator Overhaul	911	Disassemble the emergency diesel generator (engine), and make sure the integrity.	Record check
SF-9-2	Emergency diesel generator Automatic startup load inspection	921	Make sure that the emergency diesel generator is automatically started up by the external electric power loss signal and that the specified load is picked up in accordance with the specified sequence after the voltage is established.	Witnessed inspection (1 system) Record check (1 system)
SF-9-3	DC power supply unit Automatic changeover inspection	931	Make sure that the AC power is changed over to the battery immediately after the loss of AC power supply.	Witnessed inspection (1 system) Record check (1 system)

[X] Entire Reactor Facility

Management No.	Inspection item	Inspection No.	Outline of inspection	Remarks
SF-10-1	Reactor facility Total load inspection	1011	Operate the plant at the rated power, and record various plant data required for the total load inspection. Based on the data recorded, make sure that stable continuous load operation can be performed in the plant.	Witnessed inspection

Table III.4-2 Comparison of Standard Periodic Inspection Procedures between in LWR and in "Fugen"

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedureas (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-1	Class 1 commponents in-service inspection	F-1-1	Reactor coolant pressure boundary in-service inspection (pressure tube assembly, inlet/outlet pipe)	SF-1-10	1101	
		F-1-2	Reactor coolant pressure boundary leak inspection	SF-3-50	3501	
	No corresponding item	F-1-3	Pressure tube monitoring	SF-1-20	1201	
		F-1-4	Calandria tank leak inspection	SF-1-1	111	
		F-1-5	Calandria/pressure tube leak inspection	SF-1-2	121	
B-2-1	Fuel assembly sipping inspection	F-1-6	Fuel assembly sipping inspection	SF-1-3	131	
B-2-2	Fuel assembly appearance inspection and other inspections		Fuel assembly appearance inspection		132	
			Fuel assembly dimensional inspection		133	
B-2-3	Fuel assembly in-core arrangement (refueling) inspection	F-1-7	Fuel assembly loading inspection	SF-1-4	141	
	No corresponding item	F-1-8	Reactor maximum excess reactivity inspection	SF-1-5	151	
B-3	Reactor shutdown margin inspection	F-1-9	Reactor shutdown margin inspection	SF-1-6	161	
(B-1)	Class 1 commponents in-service inspection	F-2-1	Reactor coolant pressure boundary in-service inspection (Coolant recirculation system, coolant clean-up system, feed water system)	SF-3-60	3601	
			Reactor coolant pressure boundary in-service inspection (main steam system)	SF-3-70	3701	
			Reactor coolant pressure boundary in-service inspection (emergency core cooling system, residual heat removal system, isolation cooling system)	SF-3-80	3801	
	No corresponding item	F-2-5	Reactor recirculation pump easy overhaul	SF-3-3	331	

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-4	Reactor coolant recirculation pump overhaul	F-2-5	Reactor coolant recirculation pump overhaul	SF-3-3	332	
B-5	Class 3 commponents in-service inspection	F-2-2	Outside reactor coolant pressure boundary in-service inspection	SF-3-90	3901	
	No corresponding item	F-2-3	Reactor recirculation system functional inspection	SF-3-1	311	
B-6-1	Main steam safety valve functional inspection	F-2-4	Steam drum safety relief valve performance inspection	SF-3-2	322	
			Steam drum safety relief valve leak testing		323	
B-6-2	Main steam safety valve overhaul inspection		Steam drum safety relief valve overhaul		321	
B-7-1	Main steam safety relief valve safety valve functional inspection	—	No corresponding equipment			
B-7-2	Main steam safety relief valve relief valve functional inspection	—				
B-7-3	Main steam safety relief valve overhaul					
B-8-1	Main steam isolation valve functional inspection	F-2-6	Main steam isolation valve performance inspection	SF-3-4	343	
B-8-2	Main steam isolation valve leak rate test (after shutdown)		Main steam isolation valve leak testing		342	
B-8-3	Main steam isolation valve leak rate test (before startup)		Main steam isolation valve leak testing		342	
B-9	Turbine bypass valve functional inspection	F-2-25	Turbine bypass valve performance inspection	SF-3-24	3241	
B-10	Reactor auxiliary cooling system functional inspection	F-2-24	Reactor auxiliary cooling system functional inspection	SF-3-22	3221	
B-11	Emergency condenser system functional inspection	—	No corresponding equipment			
B-12	Isolation cooling system functional inspection	F-2-15	Isolation cooling system functional inspection	SF-3-13	3131	
			Isolation cooling system main valve performance inspection		3132	
	No corresponding item	F-2-16	Isolation cooling pump overhaul	SF-3-14	3141	

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
	No corresponding item	F-2-17	Isolation cooling system main valve overhaul	SF-3-15	3151	
		F-2-18	Residual heat removal system functional inspection	SF-3-16	3161	
			Residual heat removal system main valve performance inspection		3162	
		F-2-19	Steam discharge pool cooling system functional inspection	SF-3-17	3171	
			Steam discharge pool cooling system main valve performance inspection		3172	
		F-2-20	Steam discharge pool cooling pump overhaul	SF-3-18	3181	
		F-2-21	Steam discharge pool cooling system main valve overhaul	SF-3-19	3191	
B-13-1	High pressure injection system functional inspection	F-2-7	High pressure injection system functional inspection	SF-3-5	351	
			High pressure injection system main valve performance inspection		352	
B-13-2	High pressure injection system pump overhaul	F-2-8	High pressure injection system pump overhaul	SF-3-6	361	
B-13-3	High pressure injection system main valve overhaul	F-2-9	High pressure injection system main valve overhaul	SF-3-7	371	
B-14-1	Emergency diesel generator/core spray system/low pressure injection system functional inspection	F-2-10	Low pressure injection system functional inspection	SF-3-8	381	
			Low pressure injection system main valve performance inspection		382	
B-14-2	Residual heat removal system pump overhaul inspection	F-2-11	Low pressure injection pump overhaul	SF-3-9	391	
B-14-3	Residual heat removal system main valve overhaul	F-2-12	Low pressure injection system main valve overhaul	SF-3-10	3101	
P-14-1	Accumulator injection system functional inspection	F-2-13	Rapid injection system functional inspection	SF-3-11	3111	
			Rapid injection system main valve performance inspection		3112	
P-14-2	Accumulator injection system main valve overhaul	F-2-14	Rapid injection system main valve overhaul	SF-3-12	3121	

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-15-1	Injection system functional inspection	—	No corresponding equipment			
B-15-2	Core spray system pump overhaul	—				
B-15-3	Core spray system main valve overhaul	—				
B-15-4	Low pressure core spray system pump overhaul	—				
B-15-5	Low pressure core spray system main valve overhaul	—				
B-15-6	High pressure core spray system pump overhaul	—				
B-15-7	High pressure core spray system main valve overhaul	—				
B-16	Automatic pressure reduction system functional inspection	—				
	No corresponding item	F2-22	Reactor feed water pump overhaul	SF-3-20	3201	
B-17	Feed water pump functional inspection		Reactor feed water pump functional inspection		3202	
	No corresponding item	F-2-23	Reactor feed water system main feed water flow regulation valve performance inspection	SF-3-21	3211	
	No corresponding equipment	F-3-1	Heavy water cooling system functional testing	SF-4-1	411	
			Heavy water cooling system leak testing		412	
		F-3-2	Heavy water dump communicating valve performance inspection	SF-4-2	421	
B-18-1	Control rod drive hydraulic system functional inspection	F-4-1	Control rod drive mechanism performance inspection	SF-5-1	512	
			Control rod drive mechanism scram inspection		513	
B-18-2	Control rod drive mechanism open inspection		Control rod drive mechanism overhaul		511	
B-18-3	Control rod drive hydraulic system scram valve overhaul	—	No corresponding equipment			

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-19	Boric acid water injection system functional inspection	—	No corresponding equipment			
B-20	Control air compressor system functional inspection	F-4-5	Control air compressor system functional inspection	SF-5-5	551	
B-21-1	Safety protection system protection detection element performance (calibration) inspection	F-4-3	Safety protection system check/calibration inspection	SF-5-3	531	
B-21-2	Safety protection system set value confirmation testing	F-4-2	Nuclear instrumentation equipment set value confirmation testing	SF-5-2	521	
			Nuclear instrumentation equipment interlock inspection		522	
		F-4-3	Safety protection system set value confirmation testing	SF-5-3	532	
B-22	Reactor protection system interlock functional inspection		Safety protection system performance inspection		533	
		F-4-4	Safety protection system total interlock inspection	SF-5-4	541	
B-23	Fuel handling equipment functional inspection	F-5-1	Fuel handling equipment functional inspection	SF-2-1	211	
No corresponding item		F-5-2	Fuel handling equipment upper head overhaul	SF-2-2	221	
			Fuel handling equipment lower head overhaul		222	
		F-5-3	Transfer chute leak testing	SF-2-3	231	
		F-5-4	Fuel storage facility functional inspection	SF-2-4	241	
			Fuel storage facility capacity confirmation testing		242	
		F-5-5	Spent fuel storage facility functional inspection	SF-2-5	251	
			Spent fuel storage facility storage ability confirmation testing		252	
		F-5-6	Pool water cooling cleanup system functional inspection	SF-2-6	261	

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-24-1	Environmental radiation monitor functional inspection	F-6-2	Monitoring post, etc. radiation source calibration inspection	SF-7-2	721	
			Monitoring post set value confirmation testing		722	
B-24-2	Area process monitor functional inspection	F-6-1	Area process monitor radiation source calibration inspection	SF-7-1	711	
			Area process monitor set value confirmation testing		712	
			Area process monitor interlock functional inspection		713	
	No corresponding item	F-6-3	Reactor containment vessel air recirculation system functional inspection	SF-7-3	731	
			Reactor containment vessel air recirculation system filter performance inspection		732	
B-25-1	Emergency gaseous treatment system functional inspection	—	No corresponding equipment			
B-25-2	Emergency gaseous treatment system filter performance inspection	—				
B-26-1	Main control room emergency ventilation system functional inspection	F-6-5	Main control room ventilation system functional inspection	SF-7-5	751	
B-26-2	Main control room emergency circulation system filter performance inspection		Main control room ventilation system filter performance inspection		752	
	No corresponding equipment	F-6-6	Emergency gaseous waste treatment system functional inspection*	SF-7-6	761	*Tritium removal system
B-27	Gaseous waste treatment system functional inspection	F-7-1	Gaseous waste treatment system functional inspection	SF-6-1	611	
P-27-1	Annulus circulation exhaust system functional inspection	F-6-4	Annulus exhaust system functional inspection	SF-7-4	741	
P-27-1	Annulus circulation exhaust system filter performance inspection		Annulus exhaust system filter performance inspection		742	
	No corresponding item	F-7-3	Evaporation/concentration equipment leak testing	SF-6-3	631	
B-28-1	Liquid waste treatment system functional inspection		Evaporatton/concentration equipment processing capacity confirmation testing		632	

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-28-2	Liquid waste treatment system liquid waste storage facility/treatment equipment interlock functional inspection	F-7-2	Liquid waste treatment system alarm/interlock inspection	SF-6-2	621	
	No corresponding item		Liquid waste treatment system storage capacity confirmation test		622	
	No corresponding item	F-7-4	Solid waste treatment system treatment capacity confirmation testing	SF-6-4	641	
			Solid waste treatment system storage capacity confirmation testing		642	
B-29-1	Solid waste treatment system incinerator functional inspection	F-7-7	Solid waste treatment system incineration equipment processing capacity confirmation testing	FS-6-7	671	
B-29-2	Solid waste treatment system storage building management status inspection	F-7-5	Solid waste treatment system storage building radiation control status inspection	SF-6-5	651	
			Solid waste treatment system storage building storage ability confirmation testing		652	
B-30	Fluid radioactive waste leak detection unit/alarm unit functional inspection	F-7-6	Fluid radioactive waste leak detection unit/alarm unit functional inspection	SF-6-6	661	
B-31	Reactor containment vessel leak testing	F-8-1	Reactor containment vessel total leak rate inspection	SF-8-1	811	
B-32-1	Reactor containment vessel isolation valve functional inspection	F-8-3	Reactor containment vessel penetration automatic isolation valve performance inspection	SF-8-3	831	
B-32-2	Reactor containment vessel isolation valve overhaul	F-8-2	Reactor containment vessel penetration automatic isolation valve overhaul	SF-8-2	821	
B-33	Reactor containment vessel vacuum break valve functional inspection	F-8-7	Reactor containment vessel pressure relief unit functional inspection	SF-8-7	871	
B-34-1	Reactor containment vessel spray system functional inspection	F-8-4	Reactor containment vessel spray system functional inspection	SF-8-4	841	
			Reactor containment vessel spray system main valve performance inspection		842	
B-34-2	Reactor containment vessel spray system pump overhaul	F-8-5	Reactor containment vessel spray pump overhaul	SF-8-5	851	

LWR Standard Periodic Inspection Procedures		"Fugen" Periodic Inspection Procedures (Manuals)				Remarks
Procedure No. (standard)	Inspection item	Procedure No. (MITI)	Inspection item	Procedure No. (Science and Technology Agency)	Inspection No.	
B-34-3	Reactor containment vessel spray system main valve overhaul	F-8-6	Reactor containment vessel spray system main valve overhaul	SF-8-6	861	
B-35-1	Flammable gas concentration control system functional inspection	—	No corresponding equipment			
B-35-2	Flammable gas concentration control system main valve overhaul inspection					
B-36	Reactor building leak testing	—				
B-37-1	Emergency diesel generator/core spray system/low pressure injection functional inspection	F-9-2	Emergency diesel generator automatic startup load inspection	SF-9-2	921	*Equivalent to B-14-1
B-37-2	Emergency diesel generator overhaul	F-9-1	Emergency diesel generator overhaul	SF-9-1	911	
B-37-3	High pressure core spay system diesel generator overhaul	—	No corresponding equipment			
	No corresponding item	F-9-3	DC power supply automatic changeover inspection	SF-9-3	931	
B-38	Total load inspection	F-10-1	Reactor facility total load inspection	SF-10-1	1011	

Table III.4-3 Ten-Year Plan for Periodic Inspection

Periodic inspection ten-year plan

System No.: 21 System name: Reactor cooling system

S55

Manufacturer: Mitsubishi Heavy Industries, Ltd.

No.	Machine name	Machine No.	Inspection cycle	1	2	3	4	5	6	7	8	9	10	Remarks
1	Steam drum A	21-1A	1/10	○	○	●	○	○	○	○	△	○	○	△: Replace manhole flange and packing.
2	Steam drum B	21-1B	1/10	○	○	●	○	○	○	○	○	△	○	
3	Recirculation pump A	21-2A	1/10	△	△	○	△	○	△	○	◎	○	△	
4	Recirculation pump B	-2B	1/10	△	○	△	○	△*	○	△	○	◎	○	△*Reuse first seal.
5	Recirculation pump C	-2C	1/10	△	△	○	△	○	△*	○	◎	○	△*	
6	Recirculation pump D	-2D	1/10	△	○	◎	○	△	○	△*	○	△	○	
7	Inlet header A (including check valve)	21-3A	1/10	○	○	○	○	○	○	○	○	○	○	Perform inspection at unloading a half of core fuels. Overhaul ckeck valves at unloading core fuels.
8	Inlet header B (including check valve)	-3B	1/10	○	○	○	○	○	○	○	○	○	○	
9	Steam drum safety relief valve	RV21-101A	1/1	◎	◎	◎	◎	◎	◎	◎	◎	◎	◎	
10	Steam drum safety relief valve	-102A	1/1	◎	◎	◎	◎	◎	◎	◎	◎	◎	◎	
11	Steam drum safety relief valve	-103A	1/1	◎	◎	◎	◎	◎	◎	◎	◎	◎	◎	
12	Steam drum safety relief valve	-101B	1/1	◎	◎	◎	◎	◎	◎	◎	◎	◎	◎	
13	Steam drum safety relief valve	-102B	1/1	◎	◎	◎	◎	◎	◎	◎	◎	◎	◎	
14	Steam drum safety relief valve	-103B	1/1	◎	◎	◎	◎	◎	◎	◎	◎	◎	◎	
15	Piping (including feed water system inside R/B)		1/1	○	○	○	○	○	○	○	○	○	○	Perform inspection except parts specified in ISI
16	Steam drum support equipment		1/1	○	○	○	○	○	○	○	○	○	○	

Symbols ○: Appearance inspection ◎: Overhaul inspection ●: Open inspection X: Replacement △: Easy overhaul inspection

Table III.4-4 Periodic Inspection and Improvement Work Result in "Fugen"

Periodic Inspection	First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth	Eleventh	Twelfth	Thirteenth
Date of Start (parallel off)	Feb. 1, 1980	Apr. 20, 1981	Sep. 24, 1982	Feb. 15, 1984	Jul. 3, 1985	Jan. 7, 1987	Apr. 16, 1988	Jul. 19, 1989	Dec. 14, 1990	Apr. 26, 1992	Aug. 18, 1993	Jan. 6, 1995	Apr. 22, 1996
Date of Parallel in	Apr. 24, 1980	Oct. 10, 1981	Jan. 24, 1983	Jun. 28, 1984	Dec. 21, 1985	Mar. 31, 1987	Jun. 27, 1988	Oct. 30, 1989	Apr. 30, 1991	Jul. 21, 1992	Nov. 16, 1993	Apr. 25, 1995	Jul. 16, 1996
Date of Completion	May 15, 1980	Nov. 5, 1981	Feb. 17, 1983	Jul. 19, 1984	Jan. 10, 1986	Apr. 17, 1987	Jul. 20, 1983	Dec. 21, 1989	May 23, 1991	Aug. 20, 1992	Jan. 7, 1994	May 18, 1995	Aug. 8, 1996
Periodic Inspection Period*	84 days (105 days)	174 days (200 days)	123 days (147 days)	135 days (156 days)	172 days (192 days)	84 days (101 days)	73 days (96 days)	104 days (156 days)	138 days (161 days)	87 days (117 days)	91 days (143 days)	110 days (133 days)	86 days (109 days)
Major Features of Periodic Inspection	· This was the first periodic inspection after commencement of commercial operation, and as many components and equipment as possible were inspected. · Servicing was performed.	· SCC was detected in the residual heat removal system piping, etc., and the work for countermeasures against SCC was done systematically.	· D-recirculation pump was overhauled, and the integrity of the pump manufactured in Japan was confirmed.	· Pressure tube monitoring was performed for the first time. Inner diameter measurement, inner surface inspection, ultrasonic test, etc. were performed.	· Inside of the heavy water system components was inspected with priority, and the integrity was confirmed.	· Generator was overhauled. · Elongation of pressure tubes in axial direction was measured.	· This periodic inspection was shortest.	· A/C-recirculation pump was overhauled. · Reactor cooling system A loop was decontaminated. · Pressure tube monitoring was performed for the 2'nd time.	· Reactor cooling system B loop was decontaminated. · The work for countermeasures against SCC was completed.	· Pressure tube monitoring was performed for the third time. · Reactor pre-startup system integrity confirmation test was performed.	· This periodic inspection was performed in accordance with the new long-term plan.	· Pressure tube monitoring was performed for the fourth time. · EHC was updated (to digital one).	· Turbine precise inspection was performed.
Major Improvement Works	(1) Reactor coolant clean-up system was modified. (2) Heavy water purification system was modified. (3) Feed water purification system was modified. (4) Piping supports of riser pipes were replaced.	(1) Residual heat removal system piping was modified. (2) Refueling machine was modified. (3) Solidification supply tank piping material was replaced. (4) Monitoring posts were modified.	(1) High pressure injection system piping was replaced. (2) Feed water system/drain vent system piping material was replaced. (3) Shield cooling system tube bundle was replaced. (4) Waste treatment building leak detection unit was installed. (5) B-steam concentrator heating basket was replaced.	(1) High range area monitor was installed. (2) High range monitoring post was installed. (3) Main feed water flow control valve was replaced. (4) Gaseous waste system recombiner/exhaust gas condenser material was replaced.	(1) Steam discharge pool cooling system heat exchanger/No. 3-4 feed water heater material was replaced. (2) Heavy water cooler was installed. (3) Carbon dioxide gas system was modified. (4) Reactor building floor drain/equipment drain sump pump was improved. (5) Shape of solidification evaporator was modified. (6) High range stack gas monitor was installed.	(1) Steam turbine low pressure internal casing rib weld area was repaired.	(1) No. 5 feed water heater nipple was replaced. (2) Exhaust gas system condenser tube bundle was replaced. (3) Annulus seal was replaced. (4) Machines mounted in portable radiation observation vehicle were replaced.	(1) Moisture separator unit shell fillet weld was repaired. (2) Liquid waste filter was changed. (3) A-evaporation/concentration equipment heating basket was modified. (4) Dry cleaning unit was changed. (5) Waste treatment equipment was enhanced.	(1) Process computer was replaced.	(1) Feed water heater vent system piping material was replaced. (2) Moisture separator unit internal structure support frame material was replaced. (3) Auxiliary panel for feed water flow control was updated (Fuzzy control was adopted.).	(1) Refueling machine was modified (Pressure vessel structure was changed, etc.). (2) Reactor coolant purification system regenerative heat exchanger/non-regenerative heat exchanger material was replaced. (3) Feed water heater drain system piping material was replaced.	(1) No. 5 feed water heater shell material was replaced. (2) Seat to mount moisture separator unit instrument was installed. (3) Main steam pipe system monitor was replaced.	(1) Heavy water system sump pump spare unit was installed and replaced. (2) Primary coolant instrumentation equipment was updated. (3) Vertical direction earthquake acceleration detection unit was installed. (4) Main control room ventilation system fan spare unit was installed and replaced.
Work for Countermeasures against SCC		(5) High pressure injection system/low pressure injection system/residual heat removal system piping was replaced.	(6) Reactor coolant clean-up system/recirculation system piping material was replaced. (7) IHSI was performed to engineered safety features nozzle and adjacent area.	(5) Feed water piping material was replaced. (6) Reactor coolant clean-up return piping material was replaced.	(7) Recirculation system A loop piping/check valve material was replaced. (8) IHSI was performed in recirculation system B loop.	(2) Main steam system piping material was replaced.	(5) Steam drum nozzle safe end material was replaced	(6) Part of inlet pipe (8 pipes) material was replaced. (7) Inlet header shell central part material was replaced. (8) Steam drum nozzle safe end material was replaced	(2) Part of inlet pipe (8 pipes) material was replaced. (3) Inlet header shell central part material was replaced. (4) Steam drum nozzle safe end material was replaced				
Total Dose Equivalent	2.36 man · Sv	15.0 man · Sv	7.10 man · Sv	6.70 man · Sv	11.44 man · Sv	4.10 man · Sv	4.47 man · Sv	6.16 man · Sv	7.24 man · Sv	2.54 man · Sv	2.63 man · Sv	2.35 man · Sv	3.06 man · Sv

* It indicates the term from the date on which the periodic inspection was started (usually the date of parallel off) to the date of parallel in. A value inside ( ) indicates the period from the date on which the periodic inspection was started (usually the date of parallel off) to the date on which the periodic inspection was completed.

Table III.4-5 Ten-Year Plan for Reactor Coolant Pressure Boundary In-Service Inspection

Inspection Equipment: Pressure Tube Assembly

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-B	Pressure-resistant weld area inside containment ① Circumferential weld area of reducer	Volume	5%	UT	5%		224				5		5(5)		5		2
B-D	Complete penetration weld area of containment nozzle ① Inlet tube nozzle	Volume	100%	VT	10%	VT is to be performed until remote UT/PT technology is established.	224						10		10		3
B-F	Pressure-resistant weld area in dissimilar metal ① Upper extension tube weld area	Volume and surface	100%	×	×	These places could not be accessed because upper shielding plug was installed during construction.	224										
	② Lower extension tube weld area	Volume and surface	100%	VT	10%	VT is to be performed until remote UT technology is established.	448				20				20		6
B-N	Inside of reactor vessel ① Inner surface of upper extension tube	Visual inspection	Accessible range only	×	×	These places could not be accessed because upper shielding plug was installed during construction.	224 tubes										
	② Inner surface of pressure tube main body	Visual inspection	Accessible range only	VT	10%		224 tubes				10 tubes				10 tubes		3 tubes
	③ Inner surface of lower extension tube	Visual inspection	Accessible range only	VT	10%		224 tubes				10 tubes				10 tubes		3 tubes
	④ Upper rolled joint area	Visual inspection	Accessible range only	VT	10%		224 places				10 places				10 places		3 places
	⑤ Lower rolled joint area	Visual inspection	Accessible range only	VT	10%		224 places				10 places				10 places		3 places

Pressure tube monitoring

Unit: Tube

Inspection Target Place	Inspection Method	Inspection Range	Remarks	Number of Equipment	Periodic Inspection Result								Inspection Plan	
					First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
① Pressure tube main body (Zr-2.5% Nb)	UT, VT, dimension measurement	5		224				5				5.		5

A value inside ( ) is related to the ISI enhancement plan out of the total value.

Inspection Equipment: Steam Drum (1/2)

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-B	Pressure-resistant weld area inside containment ① Weld area in longitudinal direction	Volume	10%	UT	10%	Total length of welded Line: 57,600mm	12	1,920 mm				1,920 mm				1,920 mm	
	② Weld area in circumferential direction	Volume	5%	UT	5%	Total length of welded Line: 50,136mm	8	836 mm				836 mm				836 mm	
B-D	Complete penetration weld area in containment nozzle																
	① Manhole flange weld area	Volume	100%	UT	50%		4	1								1	
	② Downcomer nozzle	Volume	100%	UT	50%		14B×8	1				2				1	
	③ Feed water pipe nozzle	Volume	100%	UT	50%		10B×2					1					
	④ Low pressure injection system nozzle	Volume	100%	UT	50%		6B×8	2				2					
	⑤ Safety relief valve nozzle	Volume	100%	UT	50%		4B×6	2								1	
	⑥ Main steam pipe nozzle	Volume	100%	UT	50%		6B×32	6				5				5	
	⑦ Residual heat removal system nozzle	Volume	100%	UT	50%		8B×4	1								1	
	⑧ High pressure injection system nozzle	Volume	100%	UT	50%		4B×4					2					
	⑨ Reactor coolant clean-up system nozzle	Volume	100%	UT	50%		3B×2	PT1						1			
	⑩ Vent nozzle	Volume	100%	UT	50%	PT is performed because diameter is small.	2B×4	1								1	
	⑪ Riser pipe nozzle	Volume	100%	UT	10%		3B×224	PT8				8		8		7	
	⑫ Pressure gage nozzle	Volume	100%	PT	50%	PT is performed because diameter is small.	1½B×2								1		
	⑬ Level gage nozzle	Volume	100%	PT	50%	PT is performed because diameter is small.	1½B×12							2	2	1	1
	⑭ Thermometer nozzle	Visual inspection	100% at hydraulic test	VT	100% at hydraulic test		1B×6	6	6	6	6	6	6	6	6	at every hydraulic test	at every hydraulic test

Inspection Equipment: Steam Drum (2/2)

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-D	Pressure-resistant weld area in dissimilar metal																
	① Downcomer nozzle	Volume and surface	100%	UT,PT	50%		14B×8	PT1				2Replacement A				2Replacement B	2
	② Feed water pipe nozzle	Volume and surface	100%	UT,PT	50%		10B×2				Replacement	1					
	③ Low pressure injection system nozzle	Volume and surface	100%	UT,PT	50%		6B×8	PT2				2	6(6)	Replacement		2	2
	④ Safety relief valve nozzle	Volume and surface	100%	UT,PT	50%		4B×6	PT2	3(3)	3(3)	1(1)		Replacement	1	1	1	
	⑤ Main steam pipe nozzle	Volume and surface	100%	UT,PT	50%		6B×32	PT6				5	Replacement	4	4	4	4
	⑥ Residual heat removal system nozzle	Volume and surface	100%	UT,PT	50%		8B×4	PT1						1(1)	1(1)Replacement A	1Replacement B	1
	⑦ High pressure injection system nozzle	Volume and surface	100%	UT,PT	50%		4B×4					2	2(2)	Replacement		1	1
	⑧ Reactor clean-up system nozzle	Surface	100%	UT,PT	50%		3B×2	PT1		Replacement				1			
	⑨ Riser pipe nozzle	Surface	100%	UT,PT	10%		3B×224	PT8				8	8(8)	8(8)	7(7)	7	7(7)
	⑩ Vent nozzle	Surface	100%	UT,PT	50%		2B×4	PT1		Replacement				1		1	
	⑪ Pressure gage nozzle	Surface	100%	PT	50%	PT is performed because diameter is small.	1½B×2									1	
	⑫ Level gage nozzle	Surface	100%	PT	50%	PT is performed because diameter is small.	1½B×8	2								2	
	⑬ Thermometer nozzle	Visual inspection	100% at hydraulic test	VT	100% at hydraulic test		1B×6	6	6	6	6	6	6	6	6	at every hydraulic test	at every hydraulic test
B-G-2	Pressure holding bolt tightening area of diameter 50 mm or less ① Manhole flange bolt	Visual inspection	100%	VT	100%	Perform at overhauling	M48 x 16 bolts x 2 places x 2 units								32 bolts in A drum	at every overhauling	at every overhauling
B-H	Containment support area ① Lug	Surface	100%	PT	100%		10	1			3	1	1	1	1	1	1

A value inside ( ) is related to the ISI enhancement plan out of the total value.

Inspection Equipment: Inlet Header

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-B	Pressure-resistant weld area inside containment ① Weld area in circumferential direction	Volume	5%	UT	50%		6			192mm	IHSI was performed to a part of B loop.	Loop A was partially replaced.	384mm (192mm)	2(2)	2(1) Loop A was partially replaced.	1 Loop B was partially replaced	2
B-D	Pressure-resistant weld area in dissimilar metal																
	① Low pressure injection system nozzle	Volume	100%	UT	50%		8B×8			2				5(4)	2(2)	2(1)	2(2)
	② Residual heat removal system nozzle	Volume	100%	UT	50%		6B×4			1				2(2)	1(1)	1	1(1)
	③ Drain nozzle	Volume	100%	PT	50%	PT is performed because diameter is small.	1½B×4			1						1	
	④ Inlet pipe nozzle	Volume	100%	UT	10%		2B×224			8			9(1)	9(9)	9(9)	9(2)	9(9)
	⑤ Thermometer nozzle	Visual inspection	100% at hydraulic test	VT	100% at hydraulic test		1B×4	4	4	4	4	4	4	4	4	at every hydraulic test	at every hydraulic test
	⑥ Pressure gage nozzle	Visual inspection	100% at hydraulic test	VT	100% at hydraulic test		½B×2	2	2	2	2	2	2	2	2	at every hydraulic test	at every hydraulic test
	⑦ Flow rate measurement nozzle	Visual inspection	100% at hydraulic test	VT	100% at hydraulic test		3/8B×2	2	2	2	2	2	2	2	2	at every hydraulic test	at every hydraulic test
B-H	Containment support area ① Lug	Surface	100%	PT	100%		36			2	9	Loop A was partially replaced.	4	8	8	8	7

A value inside ( ) is related to the ISI enhancement plan out of the total value.

Inspection Equipment: Piping (1/3)

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-F	Pressure-resistant weld area in piping (dissimilar metal)																
	① Main steam system	Volume and surface	100%	UT,PT	50%		32		5		5	3(3)	Replace-ment	4	4	4	4
	② Feed water system	Volume and surface	100%	UT,PT	50%		2			Replace-ment				1			
B-J	Pressure-resistant weld area in piping																
	① Recirculation system (14B, 16B, 20B, 22B)	Volume	25%	UT	25%		134	PT8	17(17)	17(9)	4(4) B IHSI	13(6)A Replacement B IHSI	4	7	1	7	
	② Main steam system (4B, 6B, 10B, 12B, 16B)	Volume	25%	UT	25%		352		15	8(8)	24(4)	8(8)	13 Partial replacement	15	14	14	13
	Main steam system (2B)	Surface	25%		25%		4								1		
	③ Feed water system (10B)	Volume	25%	PT	25%		49	PT3	6(6)	5(2)	Replace-ment	4	4	3		2	
	Feed water system (3B)	Surface	25%	UT	25%		2									1	
	④ High pressure injection system (4B, 5B)	Volume	25%	PT UT	25%		57		Replace-ment		6		3		3		3
	⑤ Low pressure injection system (6B, 8B, 12B)	Volume	25%	UT	25%		253	PT13	Replace-ment	13	13			3Partial replacement		12	
	⑥ Residual heat removal system (6B, 8B)	Volume	25%	UT	25%		61		Replace-ment	3	3		3	3Partial replacement	3		4
	⑦ Isolation cooling system (4B, 5B)	Volume	25%	UT	25%		120		6	1(1)	6 Replacement		6		6		6
	Isolation cooling system (3B)	Surface	25%	UT	25%		2						1				
	⑧ Reactor cleanup system (3B)	Surface	25%	UT	25%		82	PT4	7(7)	Replace-ment	4	4	4	4		5	
	⑨ Riser pipe (3B)*	Surface	25%	UT	Possible range out of 20% of total tubes	173	3450	PT6	29(23)	29(23)	12(6)	12(6)	17(11)	17(11)	17(11)	17(11)	17(11)
	⑩ Inlet pipe (2B)*	Surface	25%	UT		308	4788	PT12	58(46)	57(45)	23(11)	23(11)	27(16)	27(16)	27(16)Partial replacement	27(16)	27(16)
	⑪ Steam drum vent piping (2B)	Surface	25%	UT	25%		8	PT1	8(8)	Replace-ment				1			1
	⑫ Inlet header drain piping (1-1/2B)	Surface	25%	PT	25%		16		Replace-ment	2		Partial replacement		3		1	
	⑬ Manifold drain piping (1-1/2B)	Surface	25%	PT	25%		12		Replace-ment			1 Partial replacement				2	
	⑭ Manifold drain nozzle (1-1/2B)	Surface	25%	PT	25%		4					Partial replacement				1	
	⑮ Sampling piping (1-1/2B)	Surface	25%	PT	25%		1					Attachment				1	
	⑯ Sampling nozzle(1-1/2B)	Surface	25%	PT	25%		1					Attachment				1	
	⑰ Instrumentation system piping (above 1B)	Surface	25%	PT	25%		20		Replace-ment					2	1	1	1

* Among the number of riser pipes and inlet tubes inspected in the sixth to tenth periodic inspections, 5 seams are for trace investigation. A value inside ( ) is related to ISI enhancement plan out of the total value.

Inspection Equipment: Piping (2/3)

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-K-1	Piping support member																
	① Recirculation system	Surface	25%	PT	25%		144	13		13		13 A Replacement		12		12	
	② Main steam system	Surface	25%	PT	25%		33		3		3	Partial replacement of piping					
	③ Feed water system	Surface	25%	PT	25%		22	4			Replace-ment	4		1		1	
	④ High pressure injection system	Surface	25%	PT	25%		25		Replace-ment				4		3		
	⑤ Low pressure injection system	Surface	25%	PT	25%		112	6	Replace-ment	6		6		5	6	5	
	⑥ Residual heat removal system	Surface	25%	PT	25%		50		Replace-ment	3	3		3		2		2
	⑦ Isolation cooling system	Surface	25%	PT	25%		29			3			3			2	
	⑧ Reactor coolant clean-up system	Surface	25%	PT	25%		40	3		Replace-ment		2		2	2	2	2
	⑨ Riser pipe	Surface	25%	PT	Possible range out of 10% of total tubes	Inaccessible	224										
	⑩ Inlet pipe	Surface	25%	PT		184	1792	19	19	19	19	18	18	18	18	18	18
B-K-2	Piping support structure																
	① Recirculation system	Visual inspection	100%	VT	100%		156 units	10 units	10 units	10 units	10 units	10 units	10 units	24 units	24 units	24 units	24 units
	② Main steam system	Visual inspection	100%	VT	100%		106 units	10 units	10 units	10 units	10 units	10 units	10 units	12 units	12 units	11 units	11 units
	③ Feed water system	Visual inspection	100%	VT	100%		60 units	7 units	7 units	7 units	Replacement of piping	7 units	7 units	7 units	6 units	6 units	6 units
	④ High pressure injection system	Visual inspection	100%	VT	100%		31 units	4 units	3 units Replacement of piping	3 units	3 units	3 units	3 units	3 units	3 units	3 units	3 units
	⑤ Low pressure injection system	Visual inspection	100%	VT	100%		129 units	13 units	13 units Replacement of piping	13 units	13 units	13 units	13 units	13 units	13 units	13 units	12 units
	⑥ Residual heat removal system	Visual inspection	100%	VT	100%		28 units	3 units	3 units Replacement of piping	3 units	3 units	3 units	3 units	3 units	3 units	2 units	2 units
	⑦ Isolation cooling system	Visual inspection	100%	VT	100%		51 units	6 units	5 units	5 units	5 units	5 units	5 units	5 units	5 units	5 units	5 units
	⑧ Reactor coolant clean-up system	Visual inspection	100%	VT	100%		68 units	7 units	7 units	Replacement of piping	7 units	7 units	7 units	8 units	9 units	8 units	8 units
	⑨ Riser pipe	Visual inspection	100%	VT	100%		208 units	21 units	21 units	21 units	21 units	21 units	21 units	21 units	21 units	20 units	20 units
	⑩ Inlet pipe	Visual inspection	100%	VT	100%		138 units	14 units	14 units	14 units	14 units	14 units	14 units	14 units	14 units	13 units	13 units

Inspection Equipment: Piping (3/3)

Unit: Seam

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-K-2	① Drain/vent/instrumentation system piping	Visual inspection	100%	VT	100%		1 set	10%	10%	10%	10%	10%	10%	10%	10%	10%	10%
B-P	Excluded structure ① Instrumentation system piping of 1B or less	Visual inspection	100%	VT	100%		1 set	1 set	1 set	1 set	1 set	1 set	1 set	1 set	1 set	at every hydraulic test	at every hydraulic test

Inspection Equipment: Valves

Unit: Unit

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
	Pressure holding bolt tightening (ϕ 50 or less) ① B-M-2 tvalve	Visual inspection	100%	VT	100%	Perform at disassembling	1 set	1 set	1 set	1 set	1 set	1 set	1 set	1 set	1 set	at every disassembling	at every disassembling
B-M-2	Inner surface of valve main body (above 100A) ① Main steam isolation valve (16B)	Visual inspection	1 unit 100%	VT	1 unit 100%	Perform at disassembling	4	4	4	4	4	4	4	4	4	at every disassembling	at every disassembling
	② Inlet header check valve (16B)	Visual inspection	1 unit 100%	VT	1 unit 100%		8					A replace-ment				at every disassembling	at every disassembling
	③ High pressure injection system check valve (5B)	Visual inspection	1 unit 100%	VT	1 unit 100%		4			2	2	2				at every disassembling	at every disassembling
	④ High pressure injection system injection valve (5B)	Visual inspection	1 unit 100%	VT	1 unit 100%		4			2		2				at every disassembling	at every disassembling
	⑤ Low pressure injection system steam drum check valve (6B)	Visual inspection	1 unit 100%	VT	1 unit 100%		8				4					at every disassembling	at every disassembling
	⑥ Low pressure injection system inlet header check valve (8B)	Visual inspection	1 unit 100%	VT	1 unit 100%		8								2	at every disassembling	at every disassembling
	⑦ Low pressure injection system steam drum injection valve (8B)	Visual inspection	1 unit 100%	VT	1 unit 100%		4			2					1	at every disassembling	at every disassembling
	⑧ Low pressure injection system inlet header injection valve (12B)	Visual inspection	1 unit 100%	VT	1 unit 100%		4					2				at every disassembling	at every disassembling
	⑨ Residual heat removal system check valve (6B)	Visual inspection	1 unit 100%	VT	1 unit 100%		4						1			at every disassembling	at every disassembling
	⑩ Residual heat removal system intake valve (8B)	Visual inspection	1 unit 100%	VT	1 unit 100%		4						1		1	at every disassembling	at every disassembling
	⑪ Isolation cooling system check valve (5B)	Visual inspection	1 unit 100%	VT	1 unit 100%		2								2	at every disassembling	at every disassembling
	⑫ Isolation cooling system main steam takeoff valve (5B)	Visual inspection	1 unit 100%	VT	1 unit 100%		2					1	1			at every disassembling	at every disassembling
	⑬ Isolation cooling system heat exchange steam bypass valve (5B)	Visual inspection	1 unit 100%	VT	1 unit 100%		2									at every disassembling	at every disassembling
	⑭ Feed water system check valve (10B)	Visual inspection	1 unit 100%	VT	1 unit 100%		2				Replace-ment					at every disassembling	at every disassembling
	⑮ Communicating valve for main steam isolation valve test (10B)	Visual inspection	1 unit 100%	VT	1 unit 100%		2			2				2		at every disassembling	at every disassembling
	⑯ Main steam isolation valve (10B)	Visual inspection	1 unit 100%	VT	1 unit 100%		2									at every disassembling	at every disassembling

Inspection Equipment: Reactor Coolant Recirculation Pump

JEAC 4205-1986				Fugen			Number of Equipment	Periodic Inspection Result								Inspection Plan	
Category	Inspection Place	Inspection Method	Inspection Range	Inspection Method	Inspection Range	Remarks		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth
B-G-1	Pressure holding bolt tightening (above ϕ50) ① Recirculation pump bolt	Visual inspection and volume	100%	UT,VT	100%	at disassembling pump	M72 x 24 bolts x 4 units			24 bolts Unit D					48 bolts Units A and C	at every disassembling pump	at every disassembling pump
B-K-1	Pump support member ① Recirculation pump support member	Surface	25%	PT	25%		32 seams		2 seams		2 seams		2 seams		2 seams		
B-K-2	Pump support structure ① Recirculation pump support structure	Visual inspection	100%	VT	100%		32 units	8 units			8 units			8 units			8 units
B-L-2	Inner surface of pump casing ① Recirculation pump support inner surface	Visual inspection	1 unit 100%	VT	1 unit 100%	at disassembling pump	4 units			1 bolts Unit D					2 units Units A and C	at every disassembling pump	at every disassembling pump

Table III.4-6 Ten-Year Plan for Outside Reactor Coolant Pressure Boundary In-Service Inspection

Unit: Seam

Test Category		Inspection Range	Number of Equipment	Inspection Method			Inspection Range	Inspection Result								Inspection Plan		Remarks
				UT	PT	VT		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth	
C-A	Pressure-resistant weld area in pressure vessel	Rapid injection system accumulator Weld area between shell and shell end plate	2 units 31,520mm	UT			7%					368mm		368mm		368mm		Because there are two systems (A system and B system), the number of inspections is a half of the inspection range.
		Residual heat removal system heat exchanger Weld area between shell and shell end plate	2 units 7.838mm	UT			7%					92mm		92mm		92mm		
		Residual heat removal system heat exchanger Weld area between shell and shell flange	2 units 7,838mm	UT			7%					92mm		92mm		92mm		
C-B	Pressure-resistant weld area in pressure vessel and nozzle	Rapid injection system accumulator Weld area in injection tube nozzle	2 units 2	UT			7%									1		
		Residual heat removal system heat exchanger Weld area in inlet/outlet pipe nozzle	2 units 4	UT	PT		34%									1		
C-C	Support structure connected integrally to vessel	Rapid injection system accumulator Weld area in support saddle	2 units 4		PT		34%							1				
C-E-1	Piping/valve/pump support structure	High pressure injection system	75		PT		34%				6	5	5	5	5	5	7	
		Low pressure injection system	134		PT		100%				10	10	10	10	9	9	9	
		Rapid injection system	12		PT		100%				2		2			2		
		Residual heat removal system	334		PT		100%				24	24	24	24	24	24	23	

Unit: Seam

Test Category			Inspection Range	Number of Equipment	Inspection Method			Inspection Range	Inspection Result								Inspection Plan		Remarks
					UT	PT	VT		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth	
C-E-2	Piping/valve/pump support structure		High pressure injection system	106 units			VT	100%				8 units	8 units	8 units	8 units	7 units	7 units	7 units	Because there are two systems (A system and B system), the number of inspections is a half of the inspection range.
			Low pressure injection system	136 units			VT	100%				10 units	10 units	10 units	10 units	10 units	9 units	9 units	
			Rapid injection system	8 units			VT	100%				2 units					2 units		
			Residual heat removal system	268 units			VT	100%				20 units	19 units	19 units	19 units	19 units	19 units	19 units	
	Pressure-resistant weld area in piping/valve/ pump in which reactor coolant circulates	Piping weld area nominal wall thickness :12.7 mm or less	Residual heat removal system	578		PT		34%				14	14	14	14	14	14	15	
		Branch pipe weld area	Residual heat removal system	37		PT		34%				3			2		2		
C-G	Pressure-resistant weld area in piping/valve/ pump in which reactor coolant does not circulate	Piping weld area nominal wall thickness : 12.7 mm or less	High pressure injection system	186		PT		17%				8			4		4		
		Piping weld area nominal wall thickness : 12.7 mm or more	Low pressure injection system	252	UT			17%				9			7		6		
			Rapid injection system	16	UT			17%				1			1				
		Branch pipe weld area	High pressure injection system	9		PT		17%							1				
			Low pressure injection system	8		PT		17%							1				

Unit: Seam

Test Category			Inspection Range	Number of Equipment	Inspection Method			Inspection Range	Inspection Result								Inspection Plan		Remarks
					UT	PT	VT		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth	
C-G	Pressure-resistant weld area in piping/valve/pump in which reactor coolant does not circulate	Pump casing weld area	Low pressure injection pump	2		PT		17%							1		at every disassemb-ling	at every disassemb-ling	

ISI Enhancement Plan and Wall Thickness Measurement

Unit: Seam

Test Category	Inspection Range	Number of Equipment	Inspection Method			Inspection Range	Inspection Result							Inspection Plan			Remarks
			UT	PT	VT		First	Second	Third	Fourth	Fifth	Sixth	Seventh	Eighth	Ninth	Tenth	
Reactor coolant clean-up system regenerative heat exchanger	Nozzle weld area		UT			100%											ISI enhance-ment plan
Reactor coolant clean-up system non-regenerative heat exchanger	Nozzle weld area		UT			100%											
Among reactor cooling system, equipment not belonging to reactor coolant pressure boundary, bent, flow control part and their downstream main piping, inner surface of normally used system main valves/pumps as well as normally used system main vessel proper. Test method UT: Ultrasonic test (to measure weld thickness) PT: Penetrant test	Main steam system	83 places	UT	PT		25%			3 places	3 places	3 places	3 places	3 places	3 places	2 places	2 places	Wall thickness measure-ment, etc.
	Air extraction system	47 places	UT	PT		25%			2 places	2 places	2 places	2 places	2 places	2 places	2 places	2 places	
	Condenser/feed water system	255 places	UT	PT		25%			10 places	8 places	8 places	8 places	8 places	8 places	7 places	7 places	
	Heater drain system	173 places	UT	PT		25%			4 places	5 places	5 places	6 places	6 places	6 places	6 places	6 places	
	Feed water heater	7 units	UT	PT		25%			3 places	3 places	2 places	2 places	2 places	2 places	2 places	2 places	

Table III.4-7 Maintenance Result and Evaluation of Heavy Water Circulation Pump

Part Name	Deterioration/ Damage Phenomenon	Deterioration/ Damage Cause	Accident/Repair Result	Action in Periodic Inspection	Evaluation
(1) Casing	Erosion	Anormaly of flow	Accident : None Repair : None	Appearance inspection	[Cycle] Arbitrary [Reasons] · No accident/repair has occurred. · During overhaul, inspection (self-inspection), inspection is made from viewpoint of preventive maintenance.
(2) Impeller	Erosion	Anormaly of flow	Accident : None Repair : None	Appearance inspection Dimensional inspection	[Cycle] Arbitrary [Reasons] · No accident/repair has occurred. · During overhaul inspection (self-inspection), inspection is made from viewpoint of preventive maintenance.
(3) Shaft	Damage	Worn-out bearing	Accident : None Repair : None	Appearance inspection Dimensional inspection Liquid penetrant test	[Cycle] Arbitrary [Reasons] · No accident/repair has occurred. · During overhaul inspection (self-inspection), inspection is made from viewpoint of preventive maintenance.
(4) Stator	Damage	Worn-out bearing	Accident : None Repair : 1 case [Repair case] · Status : Penetrated crack of Stator can · Action : Peplacement	Appearance inspection Dimensional inspection Liquid penetrant test Insulation resistance measurement	[Cycle] Arbitrary [Reasons] · No accident has occurred . · Though there is repair, reoccurrence has been prevented by executing more strict quality control for manufacturing of replacement parts. However, from viewpoint of preventive maintenance, various tests and inspections should be performed during overhaul inspection (self-inspection).
(5) Rotor	Damage	Worn-out bearing	Accident : None Repair : None	Appearance inspection Dimensional inspection Liquid penetrant test	[Cycle] Arbitrary [Reasons] · No accident/repair has occurred. · During overhaul inspection (self-inspection), inspection is made from viewpoint of preventive maintenance.

Part Name	Deterioration/ Damage Phenomenon	Deterioration/ Damage Cause	Accident/Repair Result	Action in Periodic Inspection	Evaluation
(6) Journal Bearing	Worn-out	Radial load	Accident: None Repair: 2 cases [Repair cases] · Status : Journal leaf and journal pad were worn out. · Action : Replacement	Appearance inspection Dimensional inspection	[Cycle] Arbitrary [Reasons] · No accident has occurred. · Though there are repair results based on preventive maintenance, worn-out quantity was extremely small because radial load was extremely small. · During overhaul inspection (self-inspection), inspection is performed from viewpoint of preventive maintenance.
(7) Thrust Bearing	Worn-out	Thrust load	Accident: None Repair: 5 cases [Repair cases] · Status : Thrust disk and thrust pad were worn out. · Action : Replacement	Appearance inspection Dimensional inspection	[Cycle] Once in 3 years [Reasons] · No accident has occurred. · Based on worn-out quantity so far, appearance inspection and dimensional inspection is to be performed once in 3 years from viewpoint of preventive maintenance.

Table III.4-8 Control Rod Drive Mechanism Overhaul Result

Periodic Inspection	1	2	3	4	5	6	7	8	Periodic Inspection	1	2	3	4	5	6	7	8
1	○			○					29	○			●	L			
2		○				●	L		30		○			○			
3	○		○				●	L	31	●	L		○				
4			○					●	32		○			○		●	L
5	○			○					33		○				○		
6			○			○			34			○					●
7			○					○	35	●	L		○				
8			●	L					36			○					●
9			●	L					37			●	L				
10		○				●	L		38		○				○		○
11		○				●	L		39			○					○
12		●	L			○	●	L	40			○					○
13		●	L				○		41			○					●
14			○			○			42		○			○			
15		○			●	L			43			○					○
16			○					○	44	○			●	L			
17			○					○	45	○			●	L			
18		○			○				47	●	L	○	○	●	L	○	●
19	○			○					48	○	○	●	L	○	●	L	○
20		●	L					○	49	●	L	○	○	○	●	L	○
21		●	L				○		51	N			○				
22	○			○					52	N	●	N	●	N	○	●	L
23	●	L			●	L			53	N	●	N	●	N	○	●	L
24		○		●	L				54	N	○	●	L	●	L	○	●
25			●	L					55	N			○				
26		○			○				56		N			●	L		
27		○				●	L		57		N			●	L	●	L
28			○					○	Total	15	21	21	16	13	13	10	17

(126 in all)

Symbols: ○ Overhaul
● Overhaul, then spare for next year
L: Loading
N: Newly manufactured product

Note: CRD Nos. 47 to 49 and 52 to 54 are for adjustment rods. Others are for safety rods.

Table III.4-9 Steam Turbine Periodic Inspection Status (1/2)

Periodic Inspection		First	Second	Third	Fourth
Work Period		Feb. 1, 1980 to May 15, 1980	Apr. 20, 1981 to Nov. 5, 1981	Sep. 24, 1982 to Feb. 17, 1983	Feb. 15, 1984 to Jul. 19, 1984
Steam Turbine	Casing	· Erosion was detected in the horizontal bolt cover and on the gland packing casing horizontal joint face. Repair was performed by welding. · Adhered or deposited objects were not detected inside. · Erosion and deformation were not detected in the fitting part.	· Erosion was detected on the casing horizontal joint face and the gland packing casing horizontal joint face. Minimum maintenance was performed. · Adhered or deposited objects were not detected inside. · Minute erosion was detected in the nozzle engaged portion. Minimum maintenance was performed. · Deformation was not detected.	· Erosion was detected on the casing horizontal joint face and the gland packing casing horizontal joint face. Minimum maintenance was performed. · Minute cracks were detected in the structural weld area of the casing. Repair was performed by welding. · Erosion and performed were not detected in the fitting part.	· Stainless fillet welding was performed in the eroded part of the casing horizontal joint face (permanent countermeasures). · Adhered or deposited objects were not detected inside. · Erosion and deformation were not detected in the fitting part.
	Disc	· No anomaly was detected.	· No anomaly was detected.	· No anomaly was detected.	· No anomaly was detected.
	Diaphragm Nozzle	· Erosion was detected in the radial strip blade tip. All blades were replaced with new one. · Minute cracks were detected on the nozzle diaphragm horizontal joint face. Repair was performed by welding. · Erosion was detected in the nozzle labyrinth pins. It was replaced with new one. · Clearance between the rotor was within the criteria range.	· Minute cracks were detected on the nozzle diaphragm horizontal joint face. Repair was performed by welding. · Clearance between the rotor was within the criteria range. · Erosion was detected in the nozzle packing ring fitting part. Repair was performed by welding.	· Minute cracks were detected on the nozzle diaphragm horizontal joint face and in the inner ring part. Repair was performed by welding. · The bolt for radial strip holder was broken. It was replaced with new one. · Clearance between the rotor was within the criteria range.	· Minute cracks were detected on the nozzle diaphragm horizontal joint face. Repair was performed by welding. · The bolt for radial strip holder was broken. It was replaced with new one. · Clearance between the rotor: Clearance in the nozzle packing radial direction was outside the criteria range in some positions. It was adjusted during assembling
	Blade	· Low pressure final-stage blade stellite was peeled off. Stellite was replaced. Number of blades on which stellite replacement was required: 58 Number of blades on which stellite was replaced as preventive maintenance: 0 · No other anomaly was detected (with regard to shroud, lacing wire, engaged portion).	· Stellite was replaced. Number of blades on which stellite replacement was required: 20 Number of blades on which stellite was replaced as preventive maintenance: 37 · No other anomaly was detected.	· Stellite was replaced. Number of blades on which stellite replacement was required: 31 Number of blades on which stellite was replaced as preventive maintenance: 54 · No other anomaly was detected.	· No anomaly was detected in stellite. · No other anomaly was detected either.
	Rotor	· Rubbing was detected in the rotor. Minimum maintenance was performed. · Adhered objects were not detected. · No anomaly was detected in alignment. · Sliding flaw was detected in the thrust collar. Minimum maintenance was performed.	· No anomaly was detected.	· No anomaly was detected.	· No anomaly was detected.

Table III.4-9 Steam Turbine Periodic Inspection Status (2/2)

Periodic Inspection		Fifth	Sixth	Seventh	Eighth
Work Period		Jul. 3, 1985 to Jan. 10, 1986	Jan. 7, 1987 to Apr. 17, 1987	Apr. 16, 1983 to Jul. 20, 1983	Jul. 19, 1989 to Nov. 22, 1989
Steam Turbine	Casing	· Erosion was not detected on the casing horizontal joint face. · Minute cracks were detected in the weld area in the casing. After welding, repair was performed. · Adhered or deposited objects were not detected inside. · Erosion was detected on the gland packing casing horizontal joint face. Minimum maintenance was performed. · Erosion and deformation were not detected in the fitting part.	· Erosion was not detected on the casing horizontal joint face. · Stainless fillet welding was performed in the eroded portion of the gland packing casing horizontal joint face (permanent countermeasures). · Adhered or deposited objects were not detected inside. · Erosion and deformation were not detected in the fitting part.	· Erosion was detected on the casing horizontal joint face. · Adhered or deposited objects were not detected inside. · Erosion and deformation were not detected in the fitting part.	· Erosion was detected on the casing horizontal joint face. · Adhered or deposited objects were not detected inside. · Erosion and deformation were not detected in the fitting part. · Erosion was detected in the low pressure internal casing wrapper and the rib. Stainless fillet welding was performed (permanent countermeasures).
	Disc	· No anomaly was detected.	· No anomaly was detected.	· No anomaly was detected.	· No anomaly was detected.
	Diaphragm Nozzle	· Erosion was detected in the nozzle diaphragm. It was used as it was because erosion was minute. · Erosion was detected in the flush bolt (sealed key stopper). It was replaced with new one. · Clearance between the rotor was within the criteria range.	· Erosion was detected on the nozzle diaphragm horizontal joint face. Repair was performed by welding. · Erosion was detected in the radial strip holder bolt. It was replaced with new one. · Clearance between the rotor was within the criteria range.	· Erosion was detected in the nozzle diaphragm. It was used as it was because erosion was minute. · Erosion was detected in the radial strip holder bolt. It was replaced with new one. · Clearance between the rotor was within the criteria range.	· Erosion was detected in the nozzle diaphragm. Repair was performed by welding. · Erosion was detected in the radial strip holder bolt. It was replaced with new one. · Clearance between the rotor was within the criteria range.
	Blade	· Stellite was replaced. Number of blades on which stellite replacement was required: 5 Number of blades on which stellite was replaced as preventive maintenance: 43 · No other anomaly was detected.	· Stellite was replaced. Number of blades on which stellite replacement was required: 1 Number of blades on which stellite was replaced as preventive maintenance: 45 · No other anomaly was detected.	· Stellite was replaced. Number of blades on which stellite replacement was required: 3 Number of blades on which stellite was replaced as preventive maintenance: 0 · No other anomaly was detected.	· Stellite was replaced. · Number of blades on which stellite replacement was required: 3 · Number of blades on which stellite was replaced as preventive maintenance: 0 · No other anomaly was detected.
	Rotor	· Rotor outside the centering criteria range were corrected. · No other anomaly was detected.	· Wiring was measured. · Alignment was corrected. · Gland packing blade tip trace was detected in the rotor. Minimum maintenance was performed.	· No anomaly was detected.	· No anomaly was detected.

Table III.4-10 Execution Plan of Countermeasures against SCC in "Fugen"

As of December, 1987

	System Name	Piping Name	Material	Pipe Diameter	Steam Phase Existence	Fluid Flow Existence	Operating Temperature	Number of Seams	Number of Seams to Which Countermeasures against SCC Are Taken: Performed in Design or Construction	Number of Seams to Which Countermeasures against SCC Are Taken: Performed after Commercial Operation Start	Countermeasures: Contents Performed During Construction	Countermeasures: Contents Performed after Commercial Operation Start
Piping	Reactor Proper System	Inlet pipe	SUS 27 *	2B	×	○	200°C	4788	676	–	Solution heat treatment (partial) Water-cooled welding (partial) Weld heat input control	Water quality improvement by hydrogen injection and material change (SUS316NG)
		Riser pipe	SUS 27 *	3B	×	○	280°C	3450	2,248	–		
		Inlet pipe drain tube	SUS304L	2/3B	×	×	280°C	568	568	–	Low carbon stainless steel is used.	Countermeasures already taken
	Recirculation System	Downcomer	SUS 27 *	14B	×	○	280°C	134	–	134	Weld heat input control	Material change (A loop, SUS316L) IHSI (B loop)
		Discharge pipe	SUS 27 *	16B	×	○						
		Manifold	SUS 27 *	28B	×	○						
		Steam drum vent tube	SUS304	2B	○	×		8	–	8		Material change (SUS316L)
		Manifold drain tube	SUS304	1½B	×	×		12	–	12		
		Inlet header drain tube	SUS304	1½B	×	×		16	–	16		
		Recirculation system instrumentation piping	SUS304	1½B½B	×	×		76	–	76		
	Reactor Coolant Clean-up System	Reactor coolant clean-up system piping	SUS304	3B 4B	×	○	190~ 280°C	82 88	–	82 88	Weld heat input control	Material change (SUS316L)
	Seal Leak Detection System	Seal leak detection system piping	SUS304	¼B	×	×	100°C or less	2900	Outside target range	Outside target range	Weld heat input control	Outside target range (low pressure piping)
	Emergency Core Cooling System	Low pressure injection system steam drum	SUS304	4B,5B	△	×	280°C	35	–	36	Weld heat input control	Material change (SUS316NG) IHSI (partial)
		Low pressure injection system: Steam drum	SUS304	6B,8B	△	×	280°C	–	–	121		
		Low pressure injection system: Lower header	SUS304	8B,12B	×	×	280°C	–	–	132		
	Isolation Cooling System	Steam piping	STS42	4B,5B	○	×	280°C	120	Outside target range	Outside target range	–	Outside target range (carbon steel)
		Injection piping	STS42	3B	×	×	180°C	(2 positions in SUS304)	–	2		Material change (SUS316NG)
	Residual Heat Removal System	Steam drum	SUS304	8B	×	×	280°C	45	–	45	Weld heat input control	Material change (SUS316NG) IHSI (partial)
		Inlet header	SUS304	8B	×	×	280°C	54	–	54		
	Damaged Fuel Detection System	Damaged fuel detection system piping	SUS321	¾B	×	×	40~280°C	2700	Outside target range	Outside target range	Stabilizer is used.	Countermeasures already taken
	Main Steam System	Main steam system	SUS304	6B	○	○	280°C	80	–	80	Weld heat input control	Material change (SUS316NG)
			STS49	BB86,10 ,16	○	○	280°C	306	Outside target range	Outside target range	–	Outside target range (carbon steel)
	Feed Water System	Feed water system piping	SUS304	10B	×	○	180°C	49	–	49	Weld heat input control	Material change (SUS316L)
Machine	Inlet Header	Nozzle	SUS27	2B~6B	×	○	280°C	238	–	–	Weld heat input control	Water quality improvement by hydrogen injection IHSI (partial) Partial material change (SUS316L)
		Main body	SUS27	16B 24B	×	○	280°C	4	–	1		
	Steam Drum	Safe end	SUS27	2B~14B	○ ×	○	280°C	294	–	30	Weld heat input control	Water quality improvement Partial material change (SUS316L)
	Pressure Tube Assembly		Zr-Nb SUS27	ID ø117.6mm	×	○	280°C	1120	–	–	Weld heat input control	Water quality improvement by hydrogen injection
	Reactor Coolant Clean-up System	Regenerative heat exchanger nozzle	SA182 F316	3½B,4B	○	○	280~ 190°C	1	–	–	Weld heat input control	
		Non-regenerative heat exchanger nozzle	SA182 F316	4½B	○	○	190~40°C	1	–	–		

Execution Plan (Fiscal Year)

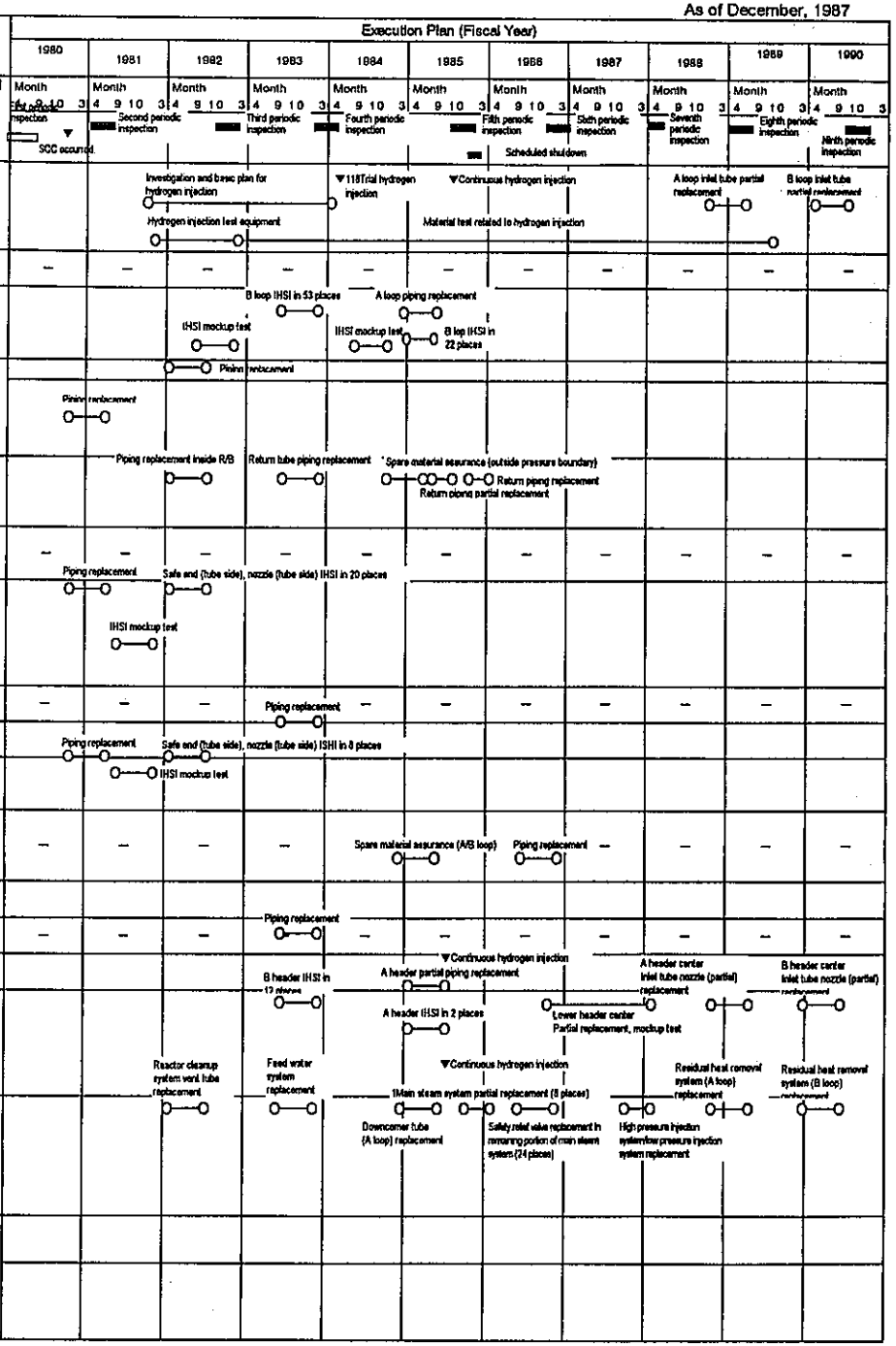


Table III.4-11 Countermeasures against Corrosion and Erosion in "Fugen"

	Item	Execution Reason	Contents	Execution Time
Countermeasures against Corrosion	Solid waste treatment system Solidification supply tank Steam nozzle material replacement	Repair and replacement caused by transgranular corrosion cracking	· Material of defective parts was changed from SUS304 to SUS316L. · Double nozzle structure was adopted in defective portion.	· Second periodic inspection (Apr. 20 to Nov. 5, 1981)
	Shield cooling system Heat exchanger tube bundle replacement	Preventive maintenance against corrosion in heat exchanger tube bundle	· Tube bundle material was changed from carbon steel (STB35) to SUS304 stainless steel.	· Third periodic inspection (Sep. 24, 1982 to Feb. 17, 1983)
	High pressure injection system Piping replacement	Repair and replacement caused by transgranular corrosion cracking	· Piping was replaced with new one of same material (SUS316TP). · Limiting values were determined for execution of welding (heat input quantity, inter-layer temperature). · Gland packing of injection source valve was replaced to low chlorine type.	· Third periodic inspection (Sep. 24, 1982 to Feb. 17, 1983)
	Gaseous waste treatment system Recombiner material replacement	Repair and replacement caused by transgranular and intergranular corrosion cracking	· Recombiner/exhaust gas / condenser/piping material was changed from SUS304 to SUS316L.	· Fourth periodic inspection (Feb. 15 to Jul. 19, 1984)
	Steam discharge pool cooling system Heat exchanger tube bundle replacement	Preventive maintenance	· Tube bundle material was changed from carbon steel (STB35) to SUS304.	· Fifth periodic inspection (Jul. 3, 1985 to Feb. 10, 1986)
	Solid waste treatment system Asphalt tank replacement	Preventive maintenance	· Corrosion was detected in lower shell plate of asphalt tank. It was replaced with new one.	· Seventh periodic inspection (Apr. 17 to Jul. 20, 1983)

	Item	Execution Reason	Contents	Execution Time
Countermeasures against Corrosion	Liquid waste treatment system Evaporation/concentration equipment heating basket replacement	Preventive maintenance	· Heating basket was replaced with new one of same material (SUS316L).	· (Aug., 1981) · Third periodic inspection (Sep. 24, 1982 to Feb. 17, 1983) · Fifth periodic inspection (Jul. 3, 1985 to Feb. 10, 1986) · Sixth periodic inspection (Jan. 7 to Apr. 17, 1987)
			· Material of heating basket was changed from SUS316 to corrosion-resistant, heat-resistant alloy steel (GNCF1) specified in Nuclear Power Generation Standard.	· Eighth periodic inspection (Jul. 19 to Dec. 21, 1989)

	Item	Execution Reason	Contents	Execution Time
Countermeasures against Erosion	Turbine system Air extraction drain/vent piping replacement	Repair and replacement caused by leak accompanied by thinning	· Piping material was changed from carbon steel (STPT38) to low alloy steel (STPA23).	· Third periodic inspection (Sep. 24, 1982 to Feb. 17, 1983)
	Reactor feed water system Feed water flow control valve structure change	Countermeasures against cavitation	· Plug/cage material was changed from SUS316 to SUS630	· Fourth periodic inspection (Feb. 15 to Jul. 19, 1984)
	No. 3/No. 4 feed water heater Nozzle material replacement	Preventive maintenance	· Nozzle material was changed from carbon steel (SS41, SB46) to low alloy steel (SCMV3).	· Fifth periodic inspection (Jul. 3, 1985 to Feb. 10, 1986)
	No. 5 feed water heater Nozzle material replacement	Preventive maintenance	· Nozzle material was changed from carbon steel (STPT38, SB46) to low alloy steel (STPA23, SCMV3).	· Seventh periodic inspection (Apr. 17 to Jul. 20, 1983)
	Moisture separator shell fillet-welded repair	Fillet-welded repair of eroded portion	· Thinned portion of moisture separator shell was fillet-welded with carbon steel, then furthermore fillet-welded with stainless steel. · A part of piping was changed from carbon steel (STPT38) to low alloy steel (STPA23).	· Eighth periodic inspection (Jul. 19 to Dec. 21, 1989)
	Steam turbine fillet-welded repair	Fillet-welded repair of eroded portion	· Steam turbine low pressure internal casing rib weld area and horizontal joint face were fillet-welded with carbon steel. · Eroded part and damaged rib weld area in steam turbine high pressure casing and low pressure casing were fillet-welded with stainless steel and carbon steel. · Damaged part of low pressure internal casing, rib, a part of wrapper and ring of steam turbine were fillet-welded with stainless steel.	· Third periodic inspection (Sep. 24, 1982 to Feb. 17, 1983) · Sixth periodic inspection (Jan. 7 to Apr. 17, 1987) · Fourth periodic inspection (Feb. 15 to Jul. 19, 1984) · Eighth periodic inspection (Jul. 19 to Dec. 21, 1989)

	Item	Execution Reason	Contents	Execution Time
Countermeasures against Corrosion	Countermeasures against corrosion for sea water system pump · Circulation water pump · Reactor auxiliary cooling sea water pump	Parts replacement caused by corrosion by sea water Parts replacement caused by corrosion by sea water	· Parts in contact with sea water were partially changed from epoxy-resin painted to stainless steel. · Pin holes and peeled off portions caused by corrosion were repaired with sea-water-resistant filler. · Suction cover, main shaft, sleeve, etc. were changed from SUS304 to SUS316L. · Bearing and protection tube were changed from SUS304 to FRP material.	· Second periodic inspection · Eighth periodic inspection
	Countermeasures against corrosion for sea water piping · Containment vessel air recirculation system sea water piping · Emergency diesel system sea water piping	Parts replacement caused by corrosion by sea water Parts replacement caused by corrosion by sea water	· Pipe lining was changed as repair for sea water leak during periodic inspection. · Up to third periodic inspection, piping was replaced with tar-epoxy lining tube of same specifications as those of existing piping. · In fourth, fifth and sixth periodic inspections, piping was replaced with rubber lining tube. · In seventh periodic inspection and later, piping was replaced with poly ethylene lining tube.	· First periodic inspection to eighth periodic inspection
	Countermeasures against corrosion for sea water system heat exchanger · Main condenser	Preventive maintenance for heat transfer tube against corrosion by sea water	· Integrity of all heat transfer tubes was confirmed by ECT (Eddy Current Test) in every periodic inspection. · End plugs were attached to heat transfer tubes. Plugging rate: 0.6%	· First periodic inspection to eighth periodic inspection

	Item	Execution Reason	Contents	Execution Time
Countermeasures against Corrosion	· Turbine auxiliary cooling sea water system heat exchanger · Reactor auxiliary cooling sea water system heat exchanger · Diesel sea water cooler	Preventive maintenance for heat transfer tube against corrosion by sea water Preventive maintenance for heat transfer tube against corrosion by sea water Repair and modification of heat transfer tubes caused by corrosion by sea water	· Integrity of all heat transfer tubes was confirmed by ECT in every periodic inspection. · Number of heat transfer tubes replaced A system: 25/584 tubes B system: 93/584 tubes · Integrity of all heat transfer tubes was confirmed by ECT in every periodic inspection. · End plugs were attached to heat transfer tubes. Plugging rate: 0.5% · Heat transfer tubes were replaced with new ones of same specifications caused by sea water leak. · End plugs were attached to heat transfer tubes, cansidering the results of ECT. · Mounting position and dimensions of anti-corrosive zinc plate were modified. · Heat transfer tube material was changed to cupronickel. · Heat transfer tubes and channel head were modified to reduce flow rate inside tubes.	First periodic inspection to eighth periodic inspection
	Countermeasures against corrosion for outdoor piping · Outdoor chemical injection piping · Piping around condenser storage tank · Reactor auxiliary cooling sea water piping · Heavy water circulation pump bearing wearing	Corrosion by rain water Corrosion by rain water Corrosion by sea water Countermeasures against corrosion	· Piping wall thickness was changed from sch40 to sch80. Corrosion-resistant and water-resistant measures were taken to outer surface. · Piping was replaced with one of same specifications and same material. · Corrosion-resistant and water-resistant measures were taken to outer surface. · Piping was replaced with one of same specifications and same material. · Corrosion-resistant and water-resistant measures were taken to outer surface. · Corrosion-resistant capability confirmation test is being performed using hot column test unit in system heavy water for material regarded as alternate of Colmonoy No. 4.	October, 1984 Eighth periodic inspection Scheduled shutdown in 1990

	Item	Execution Reason	Contents	Execution Time
Countermeasures against Erosion	Countermeasures against erosion for turbine system · Small-diameter tube	Replacement for preventive maintenance against erosion and corrosion	· Wall thickness was measured, remaining service life was evaluated, and integrity of weld area was confirmed in periodic inspection.	Second periodic inspection
	· Main condenser	Countermeasures against erosion	· Carbon steel material (STPT38) was changed to low alloy steel (STPA23). · Inlet nozzle structure was modified and baffle plate was attached in main condenser. · Internal piping (spray piping, drain tube) was changed from carbon steel to stainless steel. · Shell was fillet-welded and internal support structure was repaired in main condenser.	Eighth periodic inspection Sixth periodic inspection to eighth periodic inspection

Table III.4-12 List of Education Items Performed in Maintenance Section(1/2)

Desk Training inside Maintenance Section		
Education Item	Contents	Remarks
1. Beginner's Course (1) Organization inside Section, job assignment, job outline, job in charge (2) Outline of general work safety and periodic inspection (3) Regulations (4) Work management/QA and Maintenance Section job manuals (5) Maintenance management system and computer	(1) To understand the organization of and the jobs in the Maintenance Section as well as the outline of the jobs of the group to which assigned. (2) To obtain the knowledge required to manage the field works. To understand the outline of the periodic inspection which is the most important work in the Maintenance Section. (3) To learn the system of the laws including the law for regulating nuclear installations and the Utility Industry Law, the Safety Regulations and related regulations required in the jobs in the Maintenance Section, and learn the relationship between such laws and regulations and the jobs. (4) To understand the works performed in the Maintenance Section based on the Maintenance Section job manual. (5) To learn the maintenance management system essential to the jobs in the Maintenance Section. To obtain the knowledge on computer required to use the system.	New employees, employees transferred to the Maintenance Section and personnel transferred to the Maintenance Section are regarded as the targets.
2. Intermediate Course (1) Maintenance works for mechanical equipment (2) Maintenance works for electrical equipment (3) Maintenance works for instrumentation equipment (4) Inspection technology (5) Authorities inspection in "Fugen"	(1) To learn the outline of the mechanical equipment in "Fugen" and the basis of maintenance method for each machine. (2) To learn the outline of the electrical equipment in "Fugen" and the basis of maintenance method for each machine. (3) To learn the outline of the instrumentation equipment in "Fugen" and the basis of maintenance methods for each machine. (4) To learn the basis of the inspections performed in "Fugen" including material inspection, non-destructive inspection, hydraulic testing and leak testing. (5) To learn the outline of the authorities inspection performed in Fugen and how to receive the authorities inspection.	Personnel whose experience in the Maintenance Section is less than 5 years are regarded as the targets in principle.
3. Senior Course (1) "Fugen" Technical Standard (2) Ordinance No. 501 (3) Large-scale computer system (4) Status of ATR demonstration reactor development in Oma (5) New technology development in PNC	(1) To learn the contents of the "Fugen " Technical Standard which is treated as the standard in design, operation and maintenance of "Fugen". (2) To learn the contents and the revision status of the "Technical Standard of Structure, etc. Related to Power Generating Nuclear Facility (Ordinance No. 501)". (3) To learn the situation of the large-scale computer system in "Fugen" and examples of computer applications in other plants. (4) To learn the situation of demonstration reactor development performed in EPDC and PNC as well as the situation in Oma. (5) To learn the status of new technology development performed in Tokai and Oari including new material development, laser, AI.	All personnel are regarded as the targets in principle.
4. Tour of External Facilities	To tour facilities related to nuclear power or plant equipment around Tsuruga. (Example: Maintenance Training Center, NTC, Units Nos. 1 and 2 of Japan Atomic Power, Monju)	All personnel are regarded as the targets.

Table III.4-12 List of Education Items Performed in Maintenance Section(2/2)

On-the-Job Training inside Maintenance Section		
Education Item	Contents	Remarks
1. Beginner's Course (for new employees) (1) Understanding of work procedures (2) Execution of work procedures (3) Understanding of equipment in charge (4) Theme training	As a part of education by transfer to an other section whose targets are new employees, on-the-job training is performed inside the Maintenance Section. Each trainee belongs to one equipment group, and receives instruction directly from an instructor in that group. Contents: Instructor directly gives instruction from treatment of failure sheet to creation/execution of work sheet. A work sheet is issued for an easy work, and is in charge of the work (to the level of "TAS daily inspection"). Trainee understands the equipment from the viewpoint of maintenance and inspection through reading of instruction manual and learning in the field. Trainee picks up a theme from the jobs in the group, and make a report on it.	New employees in 1989
2. Intermediate/Senior Course Education for people/employees transferred to the Maintenance Sec.	On-the-job training is performed based on the "Education/Training Basic Plan in the Maintenance Section".	
Education inside PNC/plant		
1. Education inside Plant (1) Education on labor safety and hygiene (2) Education on radiological safety management (3) Education for first-aid personnel (4) Education on high pressure gas safety Education inside PNC (1) New employee training (2) Hierarchical training (3) Training/education on nuclear fuel cycle technology	 Performed once a year by the Labor and Personnel Section Performed once a year by the Safety and Chemistry Section First-aid personnel fostering education: 1/year (3 days) First-aid personnel periodic education: 1/yea (1 day) First-aid personnel reeducation: Once a year (3 days) Performed by the Labor and Personnel Section Performed once a year by the Labor and Personnel Section Performed by the PNC Personnel Division Performed by the PNC Personnel Division Performed by the Tokai Works	All members of all sections are regarded as the targets. All members of all sections are regarded as the targets. People selected from each section are regarded as the targets. All qualified people are regarded as the targets. Qualified people who have been first-aid members for 3 or 4 years are regarded as the targets. All members of all section are regarded as the targets. Introduction training and additional training Performed for each hierarchy. Supervisor safety education, reeducation for people engaged in radiation works, etc.
Education outside PNC		
Education Items	Contents	Remarks
1. Long-term Education (1) Japan Atomic Power Training Center A (2) Japan Atomic Energy Research Institute Training Center B 2. Lecture and Seminar (1) Short-term lecture on reactor theory (2) Short-term lecture on nuclear fuel engineering (3) Lecture on radioactive waste treatment (4) Diversified seminars a. ISU b. Techno-project c. Japan Atomic Industrial Forum d. Others 3. Training Course and Qualification Test	 May to September, 1990 April to September, 1990 (Japan Atomic Energy Research Institute) (Japan Atomic Energy Research Institute) (Japan Atomic Energy Research Institute) Regarded as necessary all through the year For other training courses and qualification tests, refer to "List of test/training course planned for each fiscal year" made by the Labor and Personnel Section.	

Table III.4-13 Reports of Failures/Troubles in "Fugen" Required by Law (up to FY 1990)

No.	FY of Occur-rence	Case Name	Effects on Operation	System Containing Machine in Which Failure/Trouble Occurred	Machine in Which Failure/Trouble Occurred	Cause	Operation Status	Detection Method
1	80	Reactor shutdown caused by failure in No. 2 B feed water heater	Automatic shutdown	Condenser/feed water equipment	Piping	Incorrect maintenance	During adjustment operation	Actuation of alarm and protection system
2	80	Turbine trip and reactor automatic shutdown caused by high-high steam drum level	Automatic shutdown	Reactor cooling system equipment	Motor	Incorrect design	During adjustment operation	Actuation of alarm and protection system
3	80	Abnormality (SCC) in cooling system piping	During shutdown for periodic inspection	Fuel handling equipment	Piping	Incorrect maintenance	During scheduled shutdown	Tour inspection
4	80	Failure in fuel handling equipment	During shutdown for periodic inspection.	Instrumentation and control system equipment	Others	Incorrect construction	During scheduled shutdown	Actuation of alarm and protection system
5	81	Reactor automatic shutdown caused by high-high neutron flux	Automatic shutdown	Instrumentation and control system equipment	Detector	Incorrect maintenance	During normal operation	Actuation of alarm and protection system
6	81	Reactor manual shutdown caused by malfunction in main feed water flow control valve	Manual shutdown	Condenser/feed water equipment	Valve	Incorrect construction	During adjustment operation	Monitoring in main control room and field
7	81	Leak in moisture separation unit drain piping	Manual shutdown	Condenser/feed water equipment	Piping	Incorrect maintenance	During normal operation	Tour inspection
8	82	Fault in high pressure injection system piping	During shutdown for periodic inspection.	Emergency core cooling equipment	Piping	Incorrect design	During periodic inspection	Periodic inspection
9	84	Reactor automatic shutdown caused by power increase at reactor recirculation pump speed change	Automatic shutdown	Instrumentation and control system equipment	No effects on machine	Incorrect operation	During adjustment operation	Actuation of alarm and protection system
10	84	Reactor automatic shutdown caused by high-high heavy water temperature	Automatic shutdown	Reactor auxiliary equipment	Valve	Unknown cause	During normal operation	Actuation of alarm and protection system
11	85	Damage in new fuel assembly while it was transported	During shutdown for periodic inspection	Fuel handling equipment	Others	Incorrect maintenance	During periodic inspection	During operation/work
12	85	Reactor automatic shutdown during periodic inspection	Automatic shutdown	Instrumentation and control system equipment	Relay	Incorrect maintenance	During adjustment operation	Actuation of alarm and protection system
13	85	Coolant leak from reactor coolant clean-up system piping	Manual shutdown	Reactor cooling system equipment	Piping	Incorrect maintenance	During normal operation	Tour inspection
14	85	Reactor automatic shutdown caused by low-low steam drum level	Automatic shutdown	Instrumentation and control system equipment	No effects on machine	Incorrect operation	During adjustment operation	Actuation of alarm and protection system
15	87	Fault during refueling work	During shutdown for periodic inspection	Reactor proper equipment	Others	Incorrect management	During scheduled shutdown	During operation/work
16	88	Reactor automatic shutdown caused by reduction of condenser vacuum degree accompanied by damage in auxiliary steam pipe safety valve	Automatic shutdown	Steam turbine equipment	Valve	Incorrect maintenance	During adjustment operation	Monitoring in main control room and field
17	89	Reactor automatic shutdown caused by main steam valve closure	Automatic shutdown	Steam turbine equipment	Relay	Incorrect design	During adjustment operation	Actuation of alarm and protection system
18	90	Reactor automatic shutdown caused by high-high steam drum level	Automatic shutdown	Instrumentation and control system equipment	Piping	Incorrect management	During normal operation	Actuation of alarm and protection system

Table III.4-14 Failure Sheet Issue Status in Each Year

Equipment \ Fiscal Year		78	79	80	81	82	83	84	85	86	87	88	89	total
Machinery/Equipment	Reactor main body	0	0	2	2	0	0	1	0	0	0	0	0	5
	Reactor cooling equipment	0	0	1	1	2	0	0	0	3	1	1	0	9
	Reactor auxiliary system equipment	0	1	4	5	0	1	1	8	3	4	8	2	37
	Reactor containment equipment	0	0	2	1	2	1	0	0	0	0	1	0	7
	Fuel handling/storage facility	0	4	9	8	0	1	0	0	2	1	4	3	32
	Radioactive waste treatment equipment	0	1	11	2	0	5	6	0	1	0	2	4	32
	Engineered safety features protection equipment	0	0	2	0	0	1	0	0	0	0	0	0	3
	Turbine system equipment	0	1	8	8	1	2	1	11	9	0	6	0	47
	Electrical equipment	0	2	9	2	2	4	0	0	0	0	0	0	19
	Instrumentation and control equipment	0	1	0	0	1	0	1	0	2	1	0	0	6
	Ventilation equipment	0	6	5	2	7	2	2	9	7	2	2	2	46
	Subtotal	0	16	53	31	15	17	12	28	27	9	24	11	243
Electric Appliance/Equipment	Reactor proper	0	0	0	0	0	0	0	0	0	0	0	0	0
	Reactor cooling equipment	0	0	0	3	0	0	0	0	1	0	0	0	4
	Reactor auxiliary system equipment	0	0	0	1	0	0	0	1	0	0	0	0	2
	Reactor containment equipment	0	0	0	1	0	0	0	0	0	1	0	0	2
	Fuel handling/storage facility	0	0	0	0	0	0	0	0	0	0	0	1	1
	Radioactive waste treatment equipment	0	0	0	0	0	0	0	0	0	0	0	0	0
	Engineered safety features protection equipment	0	0	0	0	0	0	0	0	0	0	0	0	0
	Turbine system equipment	0	1	1	0	1	0	0	0	0	0	0	0	3
	Electrical equipment	0	2	5	1	2	0	2	0	2	1	0	1	16
	Instrumentation and control equipment	0	0	0	0	0	0	1	0	0	0	0	0	1
	Ventilation equipment	0	0	1	1	0	0	0	0	0	0	0	0	2
	Subtotal	0	3	7	7	3	0	3	1	3	2	0	2	31
Instrumentation Equipment/Part	Reactor proper	0	0	0	1	1	0	0	1	0	0	0	0	3
	Reactor cooling equipment	0	1	2	1	0	1	0	1	3	0	0	0	9
	Reactor auxiliary system equipment	0	3	9	2	2	2	0	1	1	0	1	0	21
	Reactor containment equipment	0	0	0	1	1	0	0	0	0	0	1	0	3
	Fuel handling/storage facility	0	1	11	5	4	2	0	1	1	0	0	3	28
	Radioactive waste treatment equipment	0	3	3	1	1	2	1	0	1	1	1	0	14
	Engineered safety features protection equipment	0	0	3	2	2	3	0	0	1	1	0	1	13
	Turbine system equipment	0	2	1	3	1	1	0	1	3	0	0	1	13
	Electrical equipment	0	0	3	1	1	0	0	0	0	0	0	0	5
	Instrumentation and control equipment	0	2	5	15	7	4	2	3	3	6	1	0	48
	Ventilation equipment	0	2	3	1	0	0	1	1	0	0	1	0	9
	Subtotal	0	14	40	33	20	15	4	9	13	8	5	5	166
Total		0	33	100	71	38	32	19	38	43	19	29	18	440

Reference

[III-1] IAEA Safety Series No.50-C-O(Rev. 1), Code on the safety of nuclear power plants: Operation, 1988

[III-2] IAEA Safety Series No.50-SG-O2, In-service inspection for nuclear power plants, 1980

[III-3] IAEA Safety Series No.50-SG-O7(Rev. 1), Maintenance of nuclear power plants, 1990

[III-4] IAEA Safety Series No.50-SG-O8(Rev. 1), Survellance of items important to safety in nuclear power plants, 1990

[III-5] IAEA Safety Series No.50-SG-O9, Management of nuclear power plants for safe operation, 1984

[III-6] IAEA Safety Series No.50-C/SG-Q Q13, Quality assurence in operation, 1996

[III-7] No.50-P-3, Data collection and record keeping for the management of nuclear power plant ageing, 1991

[III-8] K. Kishiwada, et al.; "Maintenance Management in Fugen", ANS International Meeting on Nuclear Power Plant Maintenance, Salt Lake City, U.S.A., 1986

[III-9] IAEA Safety Series No.50-SG-O2, In-service inspection for nuclear power plants, 1980

[III-10] IAEA Safety Series No.50-SG-O8 (Rev. 1), Survellance of items important to safety in nuclear power plants, 1990

[III-11] K. Niizawa, et al.; "Effect of H_2 Addition on Stress Corrosion Cracking in Fugen", 4th International Conference on Water Chemistry, Bournemouth, UK, Oct. 1986

[III-12] M. Shimizu, et al.; "Numerical Analysis on Heavy Water Separation Characteristics for Pair of Bithermal Trickle-Bed Type H_2/H_2O-Exchange Columns" J. of Nuclear Science and Technology, P254～263, Vol.20, No.3, March 1983

[III-13] PNC Technical Review No.73, 1990. 3, P143～148, "R&D if ISI Equipments

in ATR" ;

[III-14] IAEA Safety Series No.50-C/SG-Q Q3, Document control and records, 1996

[III-15] N. Sakurai, et al.; "Operator Education Program in Fugen", (PNC TN3415 97-001), WANO Conference on Experience on Training NPP Personnel, Tokyo, Japan, Oct. 1997

IV. Radiological Safety Management [IV-1] [IV-2] [IV-3] [IV-4]

1. Outline of Radiation Control

Radiation control in Fugen has been carried out, with reference to the control techniques of heavy water reactors in Canada and other countries, and continuously updated for the purpose of safe plant operation and exposure dose reduction in line with "Safety Regulations on Nuclear Reactors".

Features of radiation control in "Fugen" includes control of external and internal exposure due to radioactive substances similar to light water reactors, i.e. corrosion products such as ^{60}Co and ^{54}Mn generated in system such as the reactor coolant system. Another feature is the need for strict control of internal exposure due to tritium generated in heavy water used as moderator.

An organization for promotion of exposure reduction has been established, in which "Specialized Radiation Exposure Reduction Team" and "Exposure Dose Reduction Promotion WG" conduct various studies and make plans for active promotion of exposure dose reduction. As a result of their efforts, dose equivalent during periodic inspection has been maintained around 3.0 manSv despite the growing up of dose equivalent rates of components and piping of the reactor coolant system.

Protective equipment and working procedures have been developed and improved for better control of tritium. The level of tritium in working area has been kept sufficiently low despite increase of tritium concentration in heavy water with operating time, and no abnormal internal exposure in working area has been experienced until today.

2. Organizational System for Radiation Control

Radiation Control Group in Safety Management Section is in charge of radiation control works.

Radiation control works are generally classified as follows:

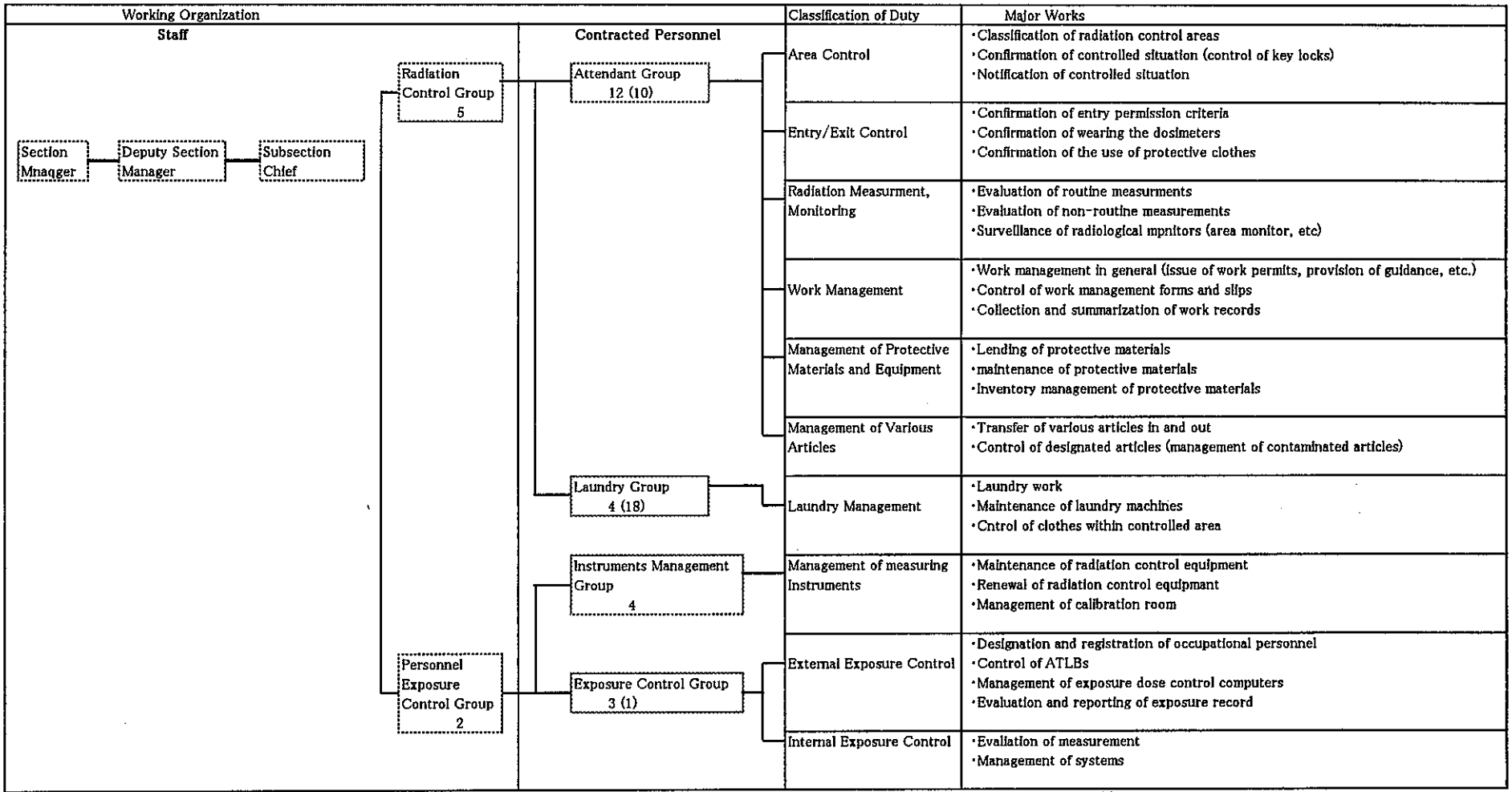
(1) Field Radiation Control: ensuring radiological safety of workers working within the nuclear facilities.

(2) Personnel Exposure Control: evaluate dose equivalent of each individual worker working in the radiation control areas.

(3) Instrument Maintenance: maintaining portable radiation measurement apparatus.

The organization for these works is shown in Fig. IV. 2-1 .

Changes of radiation control personnel engaged to each of the works are indicated in Table Ⅳ. 2-1 .



Notes: 1. As of Apr./ '97
2. Figures shown bellow the items are the numbers of people
3. Figures in ( ) are the numbers of people added for the 13th annual inspection

Fig.IV.2-1 Organization of Radiation Control and Classification of Its Work

Table IV.2-1 Changes of Radiation Control Perssonnel

Responsibilities		'77	'78	'79	'80	'81	'82	'83	'84	'85	'86	'87	'88	'89	'90	'91	'92	'93	'94	'95	'96
PNC's Staff	Sumarization			0	0	1	1	1	1	0	0	1	1	1	1	1	1	1	1	1	1
	Radiation Field Control			5	5	5	7	6	6	8	8	6	6	7	7	7	6	6	6	6	5
	Manag. of Instruments			1	1																
	Personal Exposure Control			4	4	4	4	4	4	4	3	3	3	3	3	3	3	3	3	2	2
	Sub Total			10	10	10	12	11	11	12	11	10	10	11	11	11	10	10	10	9	8
Contracted Personnel	Attendant Personnel	7	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	11	16	16
	Manag. of Instruments	2	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4	4		
	Laundry		4	5	5	5	5	5	4	4	4	4	4	4	4	4	4	4	4	7	7
	Personal Exposure Control	1	0	1	1	1	1	1	1	1	1	1	1	3	3	3	3	3	4		
	Sub Total	10	19	21	21	21	21	21	20	20	20	20	20	22	22	22	22	22	23	23	23
Total		10	19	31	31	31	33	32	31	32	31	30	30	33	33	33	32	32	33	32	31

3. Radiation Control Methods

Radiation control in Fugen is in conformity to the spirit and principle of ALARA [a recommendation of International Commission on Radiation Protection (ICRP) to maintain exposure "as low as reasonably achievable"].

In other words, the following are observed:

(i) Not to exceed specified exposure dose limits, and

(ii) To keep dose equivalent as low as reasonably achievable.

The targets are established as listed below:

(i) To avoid unplanned excessive exposure such as exposure due to an accident,

(ii) To acquire and analyze dose equivalent as promptly and as accurately as possible, provide the information to the workers, and promote workers' awareness of the importance of dose reduction,

(iii) To prevent or reduce internal exposure,

(iv) To maintain suitable environment for ensuring safe and smooth completion of works,

(v) To prepare radiation control plans, and to take appropriate protection measures,

(vi) To prevent propagation of contamination of radioactive substances,

(vii) To prohibit non-controlled persons to enter controlled areas, and

(viii) Not to take out anything contaminated from controlled areas without necessary pretreatment.

In line with "Safety Regulations on Nuclear Facilities" and "Rules on Prevention of Radiation Hazards", radiation control measures are taken based on these principles, and the details of these measures are stipulated in the "Implementation Manual and the Implementation Guidelines of Safety Regulations on Nuclear Reactors", and manuals are prepared for each type of works.

Fig. Ⅳ. 3-1 shows the overall system of various sets of radiation control rules and regulations, and Fig. Ⅳ. 3-2 shows the system of manuals for radiation control.

3.1 Field Radiation Control

(1) Area Control

Within the plant site, areas whose dose level exceeds or may exceed the value provided by the relevant ordinance (Article 2, Bulletin of Dose Equivalent) are designated as controlled areas. Criteria for designation of controlled areas depending on a situation of contamination level and dose equivalent are as outlined in Table Ⅳ. 3-1.

In addition, areas in which dose equivalent rates exceed or may exceed 1 mSv/h are designated as special areas of restricted access (Area 3). Areas whose surface contamination or contaminated density in air and water are, or may become, more than 10 times as high as the value stipulated by the relevant ordinance are also designated as restricted access area (Area D). Access to these special areas is controlled by signboards, fences, etc., and unnecessary exposure and propagation of contamination are prevented.

In practice, the following values are adopted:

① Dose Equivalent

The value of dose equivalent provided by the relevant ordinance for the controlled areas. 6.25×10^{-3} mSv/h, i.e. 0.3 mSv/week divided by the weekly working hours of 48, is used as the reference value.

② Surface Concentration (Bq/cm^2), Concentration in Air / in Water (Bq/cm^3)

3/10 of the value provided in the relevant ordinance is taken as the criteria.

(2) Entry / Exit Control

Implementation of Entry / Exit Control of controlled areas are as follows;

① At the entrance check point of each major controlled area, guards whole day confirm that each person entering the area has completed entrance procedures, has received and is carrying Controlled Area Entrance Permit issued by Safety Management Section, and is wearing a designated personal dose equivalent meter.

② At the attendant point that is the entry/exit point to a contamination controlled area, guards whole day monitor that each entrant has been duly permitted to enter and is wearing a personal dose equivalent meter, protective equipment and protective clothes.

③ Each person leaving a designated area takes off protective equipment and protective clothes, and confirms that his body-surface contamination is below 1/10 of the surface concentration limit provided in the relevant ordinance (Article 5, Bulletin of Dose Equivalent).

In practice, contamination concentration is controlled at 1/10 of the above mentioned values for safety margins.

(3) Work Management

(a) Classification of Works

Works carried out in the controlled areas are managed with the work permission system.

① Routine Works

Routine works are those done by established radiation work procedures (RWP: Routine Work Procedures), such as the patrol, etc..

② Special Works

Special works are those required special instructions on protective measures for radiation control. With consideration to radiological environment and contents of each work, a special work permit (SWP: Special Work Permit) is issued, in which radiation monitoring methods, assignment of protective equipment to be worn, etc. are instructed.

(b) Protective Equipment and Clothes

As protective equipment and clothes, the following items are selected and used according to the concentration level of radioactive substances;

① Prevention of Physical Contamination

Tybex suits, vinyl suits, etc.

② Prevention of Intake of Radioactive Substances

Dust respirators (half masks, full masks), airline masks, self-air sets, tritium protection clothes, or other breathing protection equipment.

Criteria for wearing breathing protection equipment are explained in Table Ⅳ. 3-2.

(c) Work Management

Actual work management is carried out in line with the principle of "radiological protection system (justification, optimization, and dose limitation)". As is indicated in Fig. Ⅳ. 3-3, work management plans are made, implemented, evaluated, confirmed, and then reflected to the subsequent works.

(4) Radiation Measurement and Monitoring

(a) Routine Measurements

Measurement and monitoring of radiation conducted regularly in order to confirm that there is no abnormal radiation condition within the plant site. Items to be measured and the measuring frequencies are as listed in Table Ⅳ. 3-3.

Radiation level and concentration of radioactive substances have been surveyed extensively during initial operation of the plant, and local measuring points representative of the facility's radiation level have been selected and fixed. Measurements and monitoring are made on the field and the fixed points using the routine measurement methods. Some measurements are made using portable equipment, and some are through continuous monitoring

of the radiological monitoring equipment installed as a process monitor (e.g. area monitors, dust monitors, etc.).

In order to make data acquisition/processing faster and more accurate, efforts have been made to rationalize and to upgrade works related to data acquisition/processing and document preparation by use of computer.

(b) Non-routine Measurements

Typical measurements include the following.

① Measurement of Working Environment

Items listed in the Special Work Permit (SWP) are measured at the specified frequencies during works. The measured values are compared with anticipated values to confirm validity and effectiveness of the planned protection measures.

② Measurement at Fixed Points in Reactor Cooling System, etc.

For every reactor shutdown, measurements are made at representative points to survey the upward trend of dose equivalent rate as operating hours accumulate, to make presumptions of radiological environment for various inspection works and to utilize them to work out practical radiological protection plans.

(5) Management of Measuring Instruments

Radiological instruments are maintained to ensure correct operation for daily and periodic inspections and for emergency use. Instruments which are regularly kept are listed in Table Ⅳ. 3-4.

(6) Management of Protective Materials and Equipment

Maintenance of protective materials and equipment is to make their inventory management.

Their minimum quantities in regular inventory are listed in Table Ⅳ. 3-5.

(7) Control of Transfer of Various Articles

On the occasion of the transfer of articles in the controlled area, thier contamination are surveyed that the radiation level is below the relevant reference values. The criteria for transfer of articles across area boundaries are listed in Table Ⅳ. 3-6 and Table Ⅳ. 3-7. In practice, 1/10 of the values in these tables are used as reference values for extra safety margins.

(8) Laundry Management

Contamination of protective clothes used in controlled areas are checked, and classified according to the contamination level. Then they are laundered by using two large washing machines of 90 kg/h and one dry cleaning machine of 90

kg/h. After being cleaned, the clothes are checked for contamination again and reused.

3.2 Personal Exposure Control

Classification of people who enter the controlled areas are shown in Table Ⅳ. 3-8. All those who are to be assigned to radiation occupational worker are required of the medical and radiological examination and the education/training specified.

Following the Labor Ministry's official notice, company-wide education guidelines for designation of radiation occupational personnel has been developed for radiological safety education/training. The contents of the education/training programs are outlined in Table Ⅳ. 3-9.

Details of exposure does control are described below.

(1) External Exposure Control

① Daily External Dose Equivalent Control

For daily dose equivalent control, daily dose and per-work dose are checked and reflected to the work plan every day. The maximum planned dose equivalent in Fugen is 1 mSv/day and 20 mSv/year in principle.

② Monthly External Dose Equivalent Control

For monthly external dose equivalent control, monthly dose equivalent is evaluated and ensure not to exceed the dose equivalent limit provided by the relevant ordinance.

③ Measuring Instruments Used for External Exposure Control

(ⅰ) Alarm pocket dosemeter for daily control

(ⅱ) TLD badges, film badges (FB) for monthly evaluation

They are listed and described in Table Ⅳ. 3-10.

(2) Internal Exposure Control

Internal exposure resulted from intake of γ ray emitting nuclides such as ^{60}Co, ^{59}Fe, and ^{54}Mn, as well as ^{3}H leakage from heavy water/helium system are controlled.

A liquid scintillation counter is used for measuring urinary ^{3}H concentration, and bed type whole-body counters is used to monitor γ ray emitting nuclides.

These measurements are made to workers at the times of assigned to the work and disengaged, when necessary, and also once every 3 months in case of long-term assignment. The analyses of measurement indicate that no abnormal internal exposure has occurred up to now.

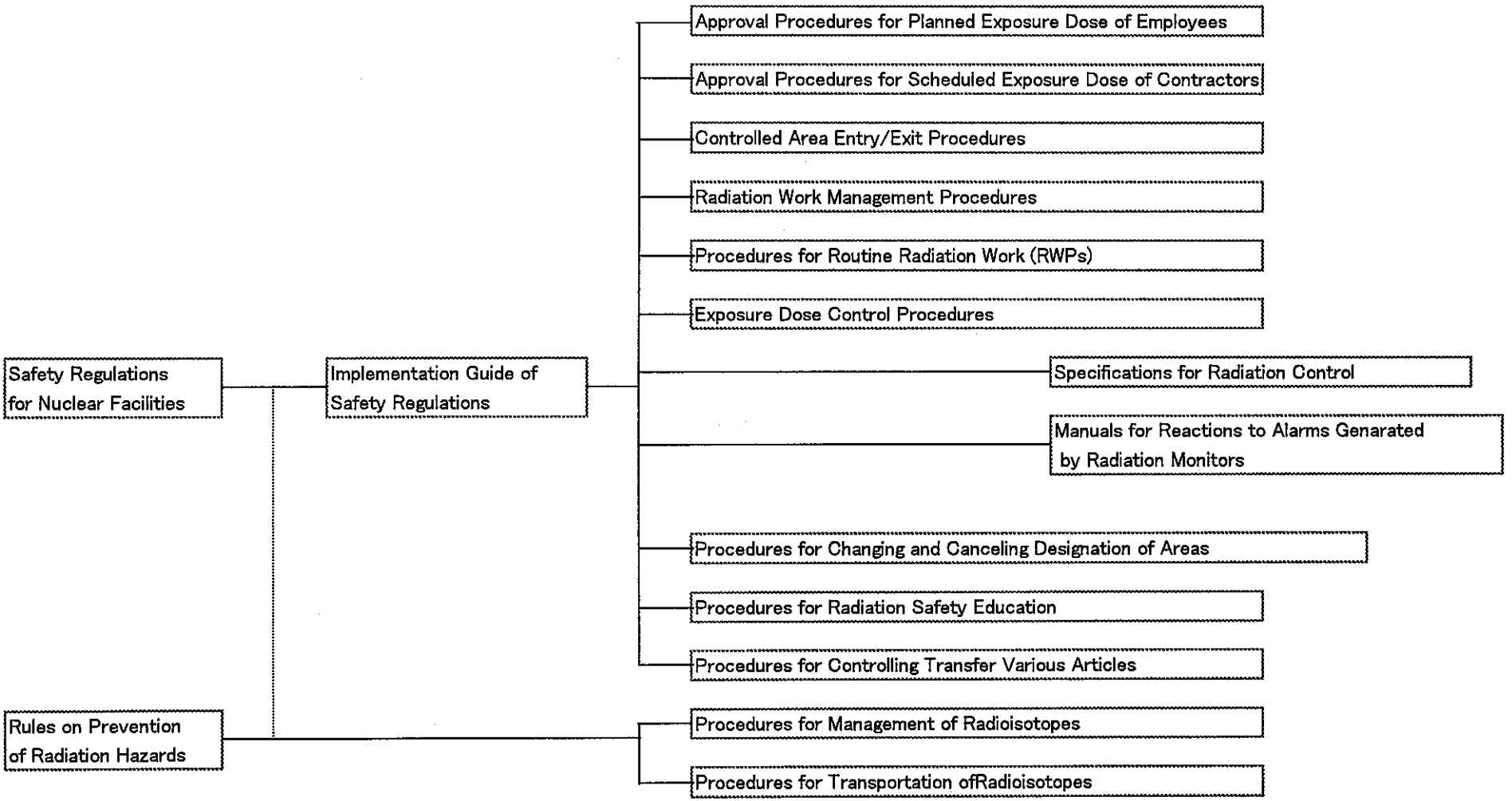


Fig. IV.3-1 System of Rules and Regulations for Radiation Control

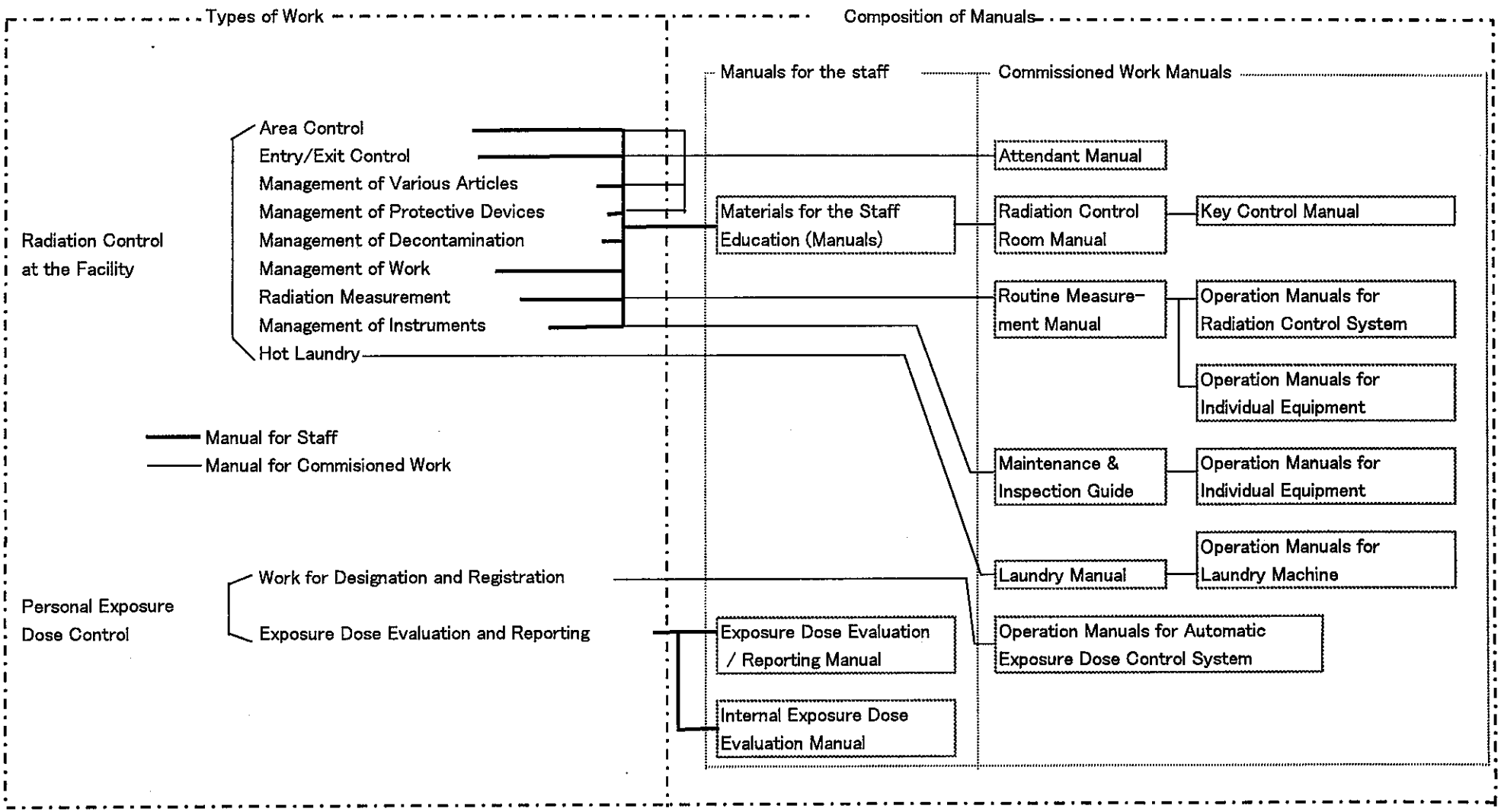


Fig. IV. 3-2 System of Manuals for Radiation Control

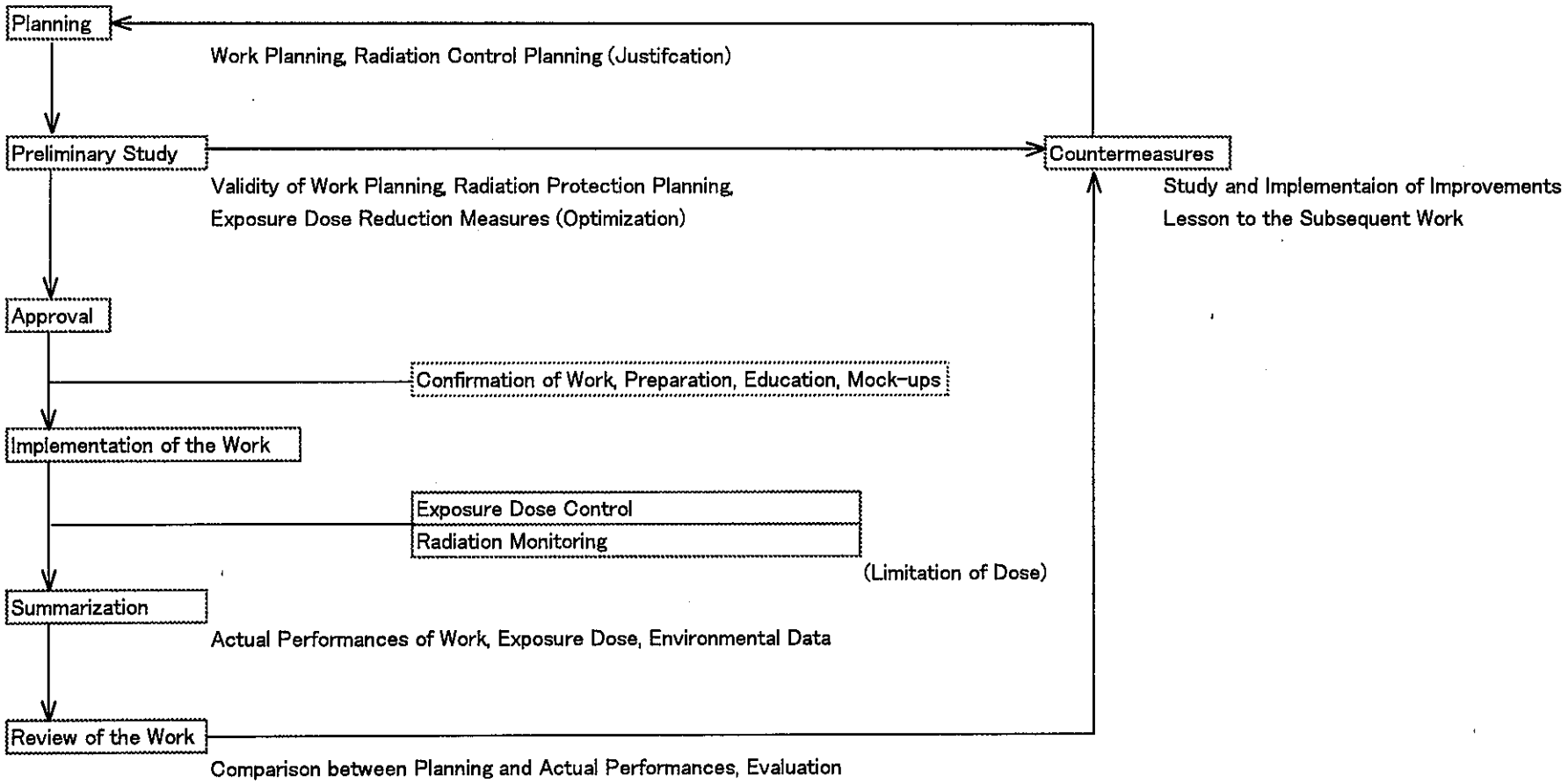


Fig. IV.3-3 Basic Approach to Radiation Work Management

Table IV.3-1 Criteria Concerning Classification of Controlled Area

Classsification of Dose Equivalent Rate \ Contamination Area		Area-A Areas may not occur contami-nations	Area-B Areas may not exceed Ms or Ma	Area-C Areas may not exceed 10 times of Ms or 10 times of Ma	Area-D Areas may exceed 10 times of Ms or 10 times of Ma
1	Areas exceed or may exceed 6.25 μ Sv/h, and may not exceed 0.1 mSv/h	Area 1-A Dose - 1 No Contami-nation	Area 1-B Dose - 1 Contamination - B	Area 1-C Dose - 1 Contamination - C	Area 1-D Dose - 1 Contamination - D
2	Areas may not exceed 1.00 mSv/h	Area 2-A Dose - 2 No Contami-nation	Area 2-B Dose - 2 Contamination - B	Area 2-C Dose - 2 Contamination - C	Area 2-D Dose - 2 Contamination - D
3	Areas may exceed 1.00 mSv/h	Area 3-A Dose - 3 No Contami-nation	Area 3-B Dose - 3 Contamination - B	Area 3-C Dose - 3 Contamination - C	Area 3-D Dose - 3 Contamination - D

Notes: ・Areas -A of contamination classification are called Radiation Controlled Area
・Areas -B, -C, -D of contamination classification are called Contamination Controlled Area as a whole
・Ms : Surface density of radioactivities in the controlled area
・Ma : Aerial concentration of radioactive materials in the controlled area

Table IV.3-2 Criteria for Use of Breathing Protection Equipment

Non-Tritium	Aerial Concentration Bq/cm^3	Below 4×10^{-5}	4×10^{-5} to 4×10^{-4}	4×10^{-4} to 4×10^{-3}	4×10^{-5} to 4×10^{-4}
	Breathing Protection Equipment	Half Masks as necessary	Half of Full Masks	Full Masks or Air-Line/Self-Air Sets	Air Lines or Self-Air Sets
Tritium	Aerial Concentration Bq/cm^3	Below 7×10^{-1}		7×10^{-1} or more	
	Breathing Protection Equipment	Air Lines or Self -Air Sets		Tritium Protection Clothes	

Table IV.3-3 Routine Measurements and Their Frequencies

Area	Item	Subjects of Measurement	Frequencies	Number of Measuring Points
Within Controlled Areas	Dose Equivalent Rate related to External Radiation	γ ray	Continuous	44
	Aerial Concentration of Corpuscular Radioactive Materials	Total β ray	Continuous	23
	Surface Density	Total β ray	Continuous	50
	Aerial Tritium Concentration	^3H	Continuous	11
	Aerial Tritium Concentration	^3H	Weekly	12
Boundary of Controlled Areas	Dose Equivalent Rate Related to External Radiation	γ ray	Weekly	12

Table IV.3-4 Regular Inventory of Radiological Instruments (Minimum quantity regularly prepared)

Classification	Items	Quantity	Inspection and Calibration Frequncy
Fixed Instruments for Radiation Monitoring	Monitoring Posts	1	Once Per Year
	Area Monitors	47	
	Main Stack Monitor	1	
	Stack Monitor for Radwaste Building	1	
	Water Discharge Tank Monitor	1	
Instruments for Radiation Control	Background Dose Equivalent Rate Survey Meters	2	Once Every 6 Months
	Dose Equivalent Rate Survey Meters	6	
	Neutron Survey Meters	1	
	Surface Radiation Density Survey Meters	8	
	Dust Samplers	7	
	Body-surface Monitors	2	
	Alarm Meters	50	
	Laundry Monitors	1	
	Radiation Measuring Instruments for Samples	3 *	
	TLD Badge Measuring Units for Personal Dose Control	1	
	Whole-body Counters	1	
	Tritium Monitors	1	

* comprises the three instrumens as follows;

Radiation Measuring Instruments for Samples	GM Radiation Measuring (Instrument	1
	2π Gas Flow Radiation Measuring Instrument	1
	Low-Energy Beta Radiation Measuring Instrument	1

Table IV.3-5 Quantities of Regularly Prepared Protective Equipment (Minimum quantity regularly prepared)

Items	Quantity	Inspection Frequency
Tritium Protection Clothes	10	Once Every 6 Months
Self-air Sets	10	
Air-line Masks	10	
Dust Masks	100	

Table IV.3-6 Values Concerning Transfer of Articles to/from Controlled Area

Items	Dose Equivalent Rate (mSv/h)
Moving Articles to Dose Equivalent Rate Area-2 from Area-3	1.0
Moving Articles to Dose Equivalent Rate Area-1 from Area-2 or -3	0.1
Moving Articles to the Out of Controlled Area from Dose Equivalent Rate Area-1, -2 or -3	0.005

Table IV.3-7 Values Concerning Transfer of Articles to/from Contamination Controlled Area

Items	Surface Radiation Density (Bq/cm^2)		Dose Equivalent Rate (mSv/h)
	α ray	No α ray	
Moving Articles to Contamination Area -C from Area -D	4	40	—
Moving Articles to Contamination Area -B from Area -D	0.4	4	—
Moving Articles to Contamination Area -B from Area -C	0.4	4	—
Moving Articles to Contamination Area -A from Area -D, -C, -B	0.4	4	—
Moving Articles to the Out of Controlled Area from Contamination Area -D, -C, -B	0.4	4	0.005

Table IV.3-8 Classification of Entrants to Controlled Area

Access Classification	Description
Radiation Occupational personnel	People who are engaged in operation or use of nuclear reactors, maintenance of nuclear facilities, transfer, disposal, or decontamination of nuclear fuel materials or articles contaminated by nuclear fuel materials (hereinafter called "nuclear fuel materials, etc."), who also enter controlled areas
Temporary Entrants	People who are not radiation occupational personnel and enter controlled areas temporarily

Table IV.3-9 Education and Training Programs Concerning Designation on Radiation Occupational Personnel

Program Name	Subject		Scope	Generality	Hours
Initial Education	Basic Knowledge of Radiation Protection	A	- Nuclear energy in general - Basic knowledge of radiation - Effect of radiation to the human body - Exposure dose limitation - Radiation protection - Radiological measurement - Personal exposure control - Rules to be observed in controlled areas (general rules)	Things which are common to all nuclear power plant	2 hours
	Practical Knowledge of Radiation Protection (Lectures)	B	- Area management - Procedures before entering controlled areas, etc. - Procedures to enter controlled areas - Rules to be observed in controlled areas (specific rules)	Things which are unique to each nuclear power plant	1.5 hours each
	Practical Knowledge of Radiation Protection (Exercises)	C	- Types of protective clothes and equipment, and directions for use - Measures to be taken in an emergency - Procedures for withdrawal from nuclear facility after completion of intended work		
Subsequential Education	Practices for Entry/Exit to/from Controlled Areas	D	- Layout of buildings within the site - Procedures to enter controlled areas - Procedures to leave controlled areas		2 hours
	Comprehensive Field Training	E	- Outlines of work - Working environment - Protective clothes - Measures to prevent propagation of contamination - Exposure reduction measures - Planned exposure dose - Organization at emergency - Exercises with mock-up	Things which are common to all the work carried out in the nuclear power plant	approx. 10 days

Table IV.3-10 Measuring Instruments to be Worn by Entrants in Controlled Areas

Access Classification	Measuring Instruments	Assumptions	Criteria for Use	Measuring Frequency
Occupational Personnel (PNC and Permanentry Stationed Contractors)	APD for Daily Control	Evaluation of X, γ rays Presence of n, β rays	Personnel entering controlled areas	at every entry
	TLD Badges for Evaluation	X, β, γ ┐ n_{th}, n_f ┘ measurement/evaluation	Personnel entering controlled areas	once per month or as necessary
	Alarm Meters	ANN (Announciation) depending on the set value	Personnel entering controlled areas	at every entry
Occupational Personnel (Other than the Above)	APD for Daily Control	Evaluation of X, γ rays Presence of n, β rays	Personnel entering areas of high dose equiv. rate	at every entry
	Film Badges	Evaluation of n, β rays	Personnel entering controlled areas	once per month or as necessary
	Alarm Meters	ANN depending on the set value	Personnel entering areas of high dose equiv. Rate	at every entry
Temporary Entrants	TLD Badges for Daily Control	Evaluation of X, γ rays Presence of n, β rays	Personnel entering controlled areas	at every entry

4. Experience of Radiological Control [IV-5] [IV-6] [IV-7] [IV-8] [IV-9]

14 periodic inspections have been made since the beginning of commercial operation in March 1979, and radiation control has been fairly conducted without any particular problem.

4.1 Field Radiation Control

(1) Area Control

In line with "Safety Regulations for Nuclear Facilities", controlled areas were designated at the time of initial fuel loading in the core. Subsequently, some areas were designated as controlled areas necessary for dose equivalent rate increase during startup, and some parts of the controlled areas were canceled as dose equivalent rates were reviewed. The designation of controlled areas following the relevant laws and ordinances is shown in Table Ⅳ. 4-1.

(2) Entry / Exit Control

When the number of people entering the controlled areas increases substantially during special events such as periodic inspections when maintenance and inspection works are intensively performed, the organization and facilities for entry / exit control are particularly important.

The annual total number of entrants to controlled areas in "Fugen" is 40,000 to 100,000 man·day/year as indicated in Fig. Ⅳ. 4-2. The number of entrants depends on how large the scope of each work is. Particularly large increase was seen in '81, '85, and '89 when large-scale SCC prevention works took place. At the opportunity of these works, a computerized exposure dose data acquisition and processing system, including TLD badge read-out gates for daily control and an automatic alarm meter check-out units, was introduced, and as well, the body surface monitor was renewed. Thus, entry / exit control has been improved with introduction of advanced access control devices. The history of introduction of these devices is outlined in Table Ⅳ. 4-1.

(3) Work Management

The history of the numbers of radiological works taken place in controlled areas is shown in Fig. Ⅳ. 4-3.

During '79 ～ '86 period, approximately 1000 works were carried out annually, and about 30% of them were SWPs and 70% were RWPs. After '87, the average number of works has decreased. This is attributable to standardization of maintenance and inspection works by introduction of the maintenance management system.

SWP and RWP management systems were also introduced, and form / slip preparation procedures and data retrieval works were modernized and improved.

It is important for implementation of effective exposure reduction measures to analyze exposure dose data of works incurring high dose equivalent and to learn the lessons of them results to the future works. For this purpose, Work Dose Analysis Program was developed to accumulate various work dosage data. The collected data have been used as the base data for exposure reduction measures. Fig. Ⅳ. 4-4 shows data flow of the work dose analysis program.

(4) Radiation Measurement and Monitoring

(a) Routine Measurements

Gamma / neutron dose rates, radioactive dust levels, etc., in the controlled area are continuously monitored by means of process monitors such as area monitors, ventilation dust monitor, etc.

Area monitor's indications fluctuate due to not only abnormal conditions but also the following factors:

① Effect of a nuclide of short half-life (^{16}N) in the main steam which increases the values during reactor operation, and decreases the values during reactor shutdown.

② Transfer of radioactive wastes and other substances generated during works, etc., which fluctuate the values inrelatedly to the reactor operating conditions.

Tritium concentration in the air of compartments in the heavy water system has been leveling off as a result of leakage prevention measures such as improvementation of leak-tight joints.

(b) Non-Routine Measurements

The most typical non-routine measurements are of working environmental radiation condition and of γ dose levels at the fixed points around the reactor cooling system, etc.

① Measurement of Working Environment

Radiation levels of areas surrounding workers are measured at each step of work processes. The measured values are compared with those of anticipated during the work planning stage, validity and effectiveness of protection measures are confirmed at all times, and the works are safely carried out.

Contaminant nuclides found on the internal surface of equipment

dismantled for inspections during the periodic inspection were corrosion products such as ^{60}Co, ^{54}Mn, and ^{51}Cr, and no fission product was detected. This fact confirms that no failure of the fuel has occurred in the past.

② Measurement at Fixed Points around Reactor Cooling System, etc.

The dose equivalent rate of reactor recirculation piping, the major radiation source within controlled areas, increases with the operating days, which corresponds to the accumulation of corrosion products.

Fig. Ⅳ. 4-5 shows the surface dose equivalent rate of the RCP outlet pipe (D).

(5) Management of Measuring Instruments

Approximately 1,000 units of 60 types of instruments are maintained. GM counter type survey meters, ionization chamber type survey meters, and alarm meters are popularly used and maintained in the large number of units. The history of their possessions is as shown in Fig. Ⅳ. 4-6. In addition, from the exposure reduction point of view and for advancement of measuring techniques, the following systems and equipment have been developed and installed:

① "Film-Separation Tritium Monitors which has a detectable limit one order lower than ordinary monitors", capable of monitoring tritium concentration in working atmosphere continuously

② "Radio Area Monitoring System" which collectively and digitally displays aerial- zone dose equivalent rate in the reactor building.

(6) Management of Protective Materials and Devices

Protective materials are indispensable for prevention of in and out of body contamination. The stock of paper suits and vinyl suits, which are used in large quantities, are well controlled based on the experience of the correlation between the amount of contaminating works and protective materials used.

(7) Control of Transfer of Various Articles

The number of articles transferred out of controlled areas peaks at around 1,500 articles/month during annual inspection. Measurements of surface contamination related to transfer of articles out of the controlled areas require time and labor consuming works, so that they were automated by installation of transfer monitors for large articles and hand-carried articles.

(8) Laundry Management

The amount of laundered items peaks at around 50,000 pieces/month during annual inspection.

In order to reduce radioactive substances released from laundry waste water, an

emulsion type dry cleaning machine was installed. Released radioactivities of laundry waste are become below the detectable limit (4×10^{-3} Bq/cm^3) by improving the laundry technique and abandoning used socks etc. since the 12th periodic inspection.

4.2 Personal Exposure Dose Control

Because of that personal exposure dose control works require fast and accurate processing of extensive amount of data, Personal Dosimetry Data Processing System using a large-scale computer has been introduced. In addition, for more strict and effective control personal dose, the system was improved using automatically readable TLD badges (Fig. Ⅳ. 4-7).

(1) External Exposure Control

The total number of radiation occupational personnel is up to 2,000 people/year. Fig. Ⅳ. 4-8 shows the annual changes of the number of occupational personnel, about 90% of their annual dose equivalents are concentrated during periodic inspection period. The total annual dose equivalent is affected particularly by the scope and details of modification/reconstruction works, and has been fluctuating between 2.4 to 15 man・Sv. Fig. Ⅳ. 4-9 shows the total dose equivalent of each fiscal year and each periodic inspection.

The dose equivalent related to regular periodic inspection works has been around 3.0 man・Sv despite yearly increase of the dose equivalent rate from the piping of the reactor cooling system. This is attributable to rationalization and various countermeasures for reduction of exposure dose including improvement of procedures and/or method of works, chemical decontamination, etc. Table Ⅳ. 4-2 shows the exposure dose equivalent in each fiscal year.

(2) Internal Exposure Control

Results of evaluation of internal intake of γ nuclides using whole-body counters (WBCs) have shown no occurrence of abnormal internal exposure until today. In addition, internal exposure due to tritium evaluated by bioassay (Bio) have been kept sufficiently low without any abnormal exposure until today despite increase of tritium concentration in heavy water. This is attributable to preventive measures in work management program and improvement / development of protective devices and work methods.

4.3 Experience of Exposure Reduction

As stated above, various radiation protection and exposure reduction measures

have been actively implemented in Fugen. In practices, as outlined in Fig. Ⅳ. 4-10, exposure reduction measures are targeted at three elements of exposure i.e. radiation sources, works, and workers.

(1) Reduction of Radiation Sources

(a) Minimization of Deposit of Radioactivities

① Hot Drain-off of Feed/Condensate Water Systems

The method of lay-up management during periodic inspections has been changed from wet lay-up to dry lay-up by hot drain-off. In order to control corrosion during dry lay-up, introduction of Fe to the reactor cooling system during reactor startup is restricted.

② Water Chemistry Control During Operation

Oxygen injection to feed/condensate water system and usage of the condensate demineralizer have been improved.

(b) Removal of Deposited Radioactivities (Chemical Cleaning) [IV-10][IV-11] [IV-12]

Parts and components which have high dose equivalent rates chemically cleaned, and their deposited crud is dissolved and removed before inspection. These parts and components include the lower shielding plug latches, fuel handling machines, recirculation pump internals, regenerative/non-regenerative heat exchangers of the reactor clean-up system, heat exchangers of the residual heat removal system, evaporative condensers, etc. The reactor cooling system has been chemically cleaned, and dose equivalent rate and the total dose equivalent have decreased substantially.

(2) Rationalization of Works

(a) Introduction of Automatic Systems and Remote Operation

For works which should be carried out under the circumstances of high dose equivalent rate, manual works are to be eliminated as much as possible. Typical examples of such rationalization include use of automatic ultrasonic testing equipment for inlet/outlet piping of the primary system, pressure tube monitoring equipment automatic welding machine and automatic beveling machines.

(b) Optimization of Shield

Lead wool blankets are used as local shields for pipes and equipment with high dose equivalent rates to reduce the aerial-zone dose equivalent rate in the working space.

(c) Notification of Field Dose Equivalent Data

Some parts of radiation measuring works have been automated by installing automatic dose indicators (digital are monitors) in various parts of major pathways in the reactor building. Radio area monitoring systems have been installed to ensure to let workers know the on-going radiological conditions at real time without delay. These monitoring systems automatically measure aerial-zone dose equivalent rates and display the data on their display panels collectively.

(3) Training and Education

Since the initial stage of operation, much effort has been made to promote workers' awareness of the importance of exposure reduction through the education programs necessary for designation of radioactive occupational personnel. Since 1990, for further thoroughgoing promotion, "ALARA Campaigns" were carried out. Dose equivalent values measured during regular periodic inspections have been stable at 3.0 to 3.4 man·Sv, despite year-by-year increase of dose equivalent rates of major radiation sources in Fugen, i.e. equipment and piping of the reactor cooling system. This may be attributable to the above mentioned exposure reduction measures.

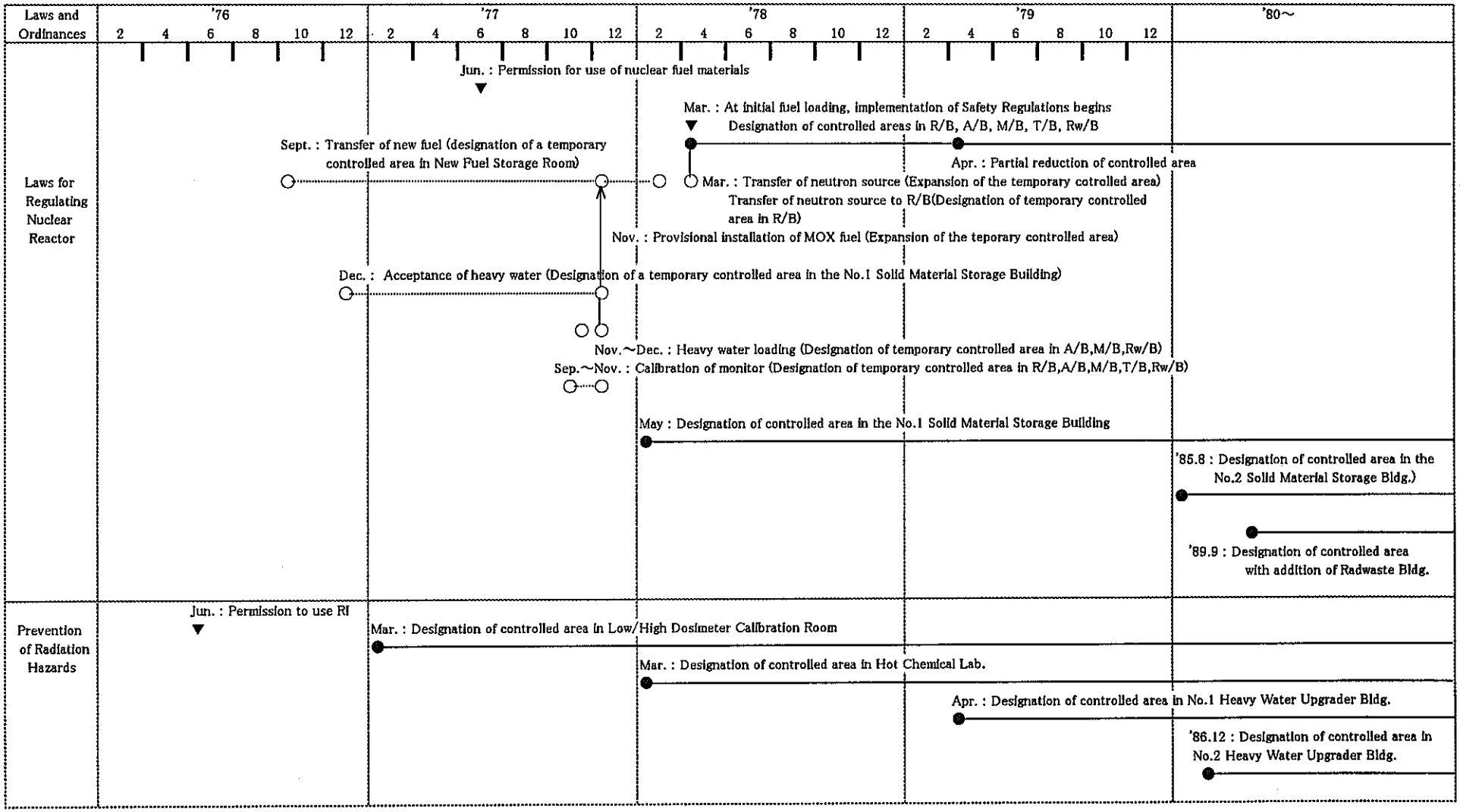


Fig.IV.4-1 Designation ofControlled Area Following Relevant Laws and Ordinances

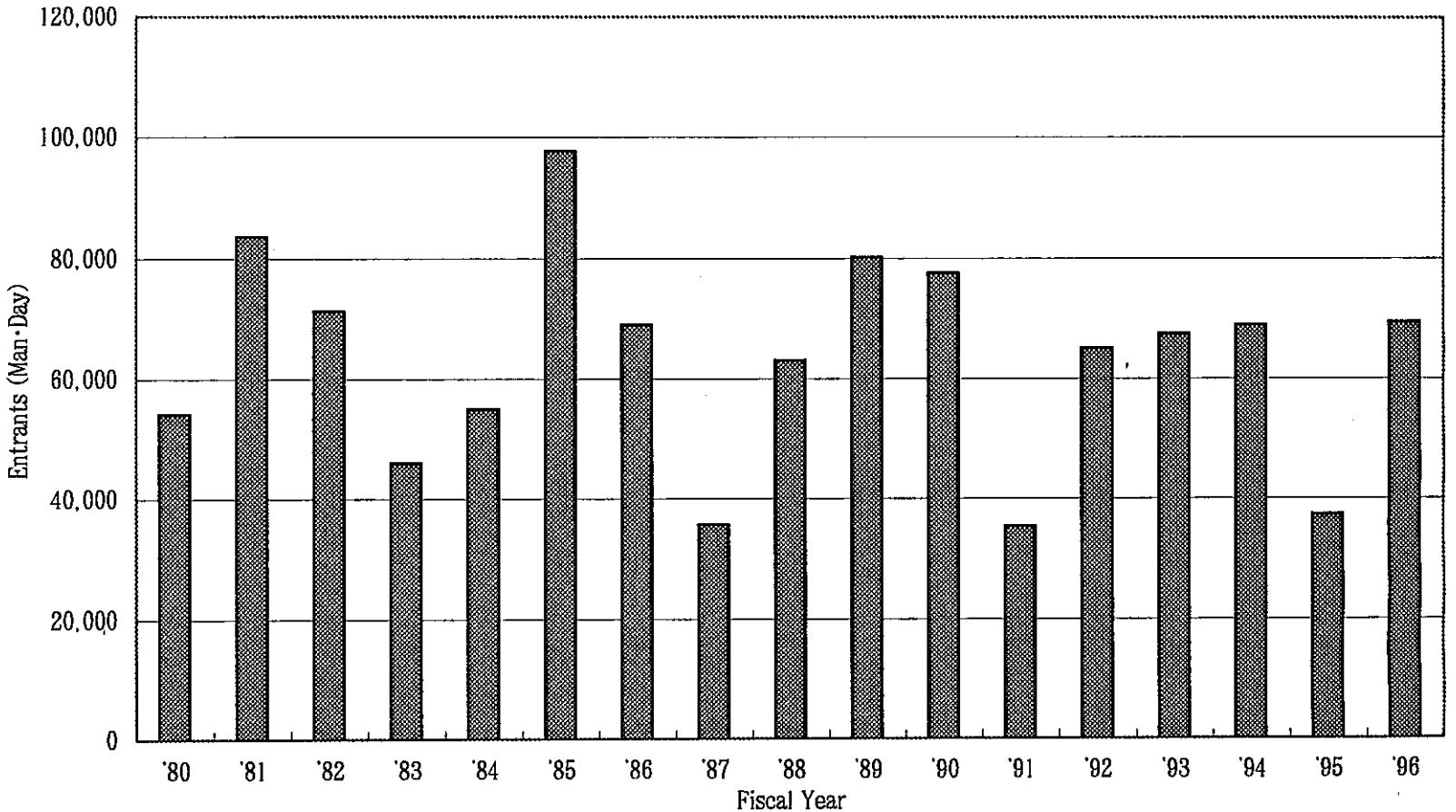


Fig. IV.4-2 History of Annual Total Entarants to Controlled Area

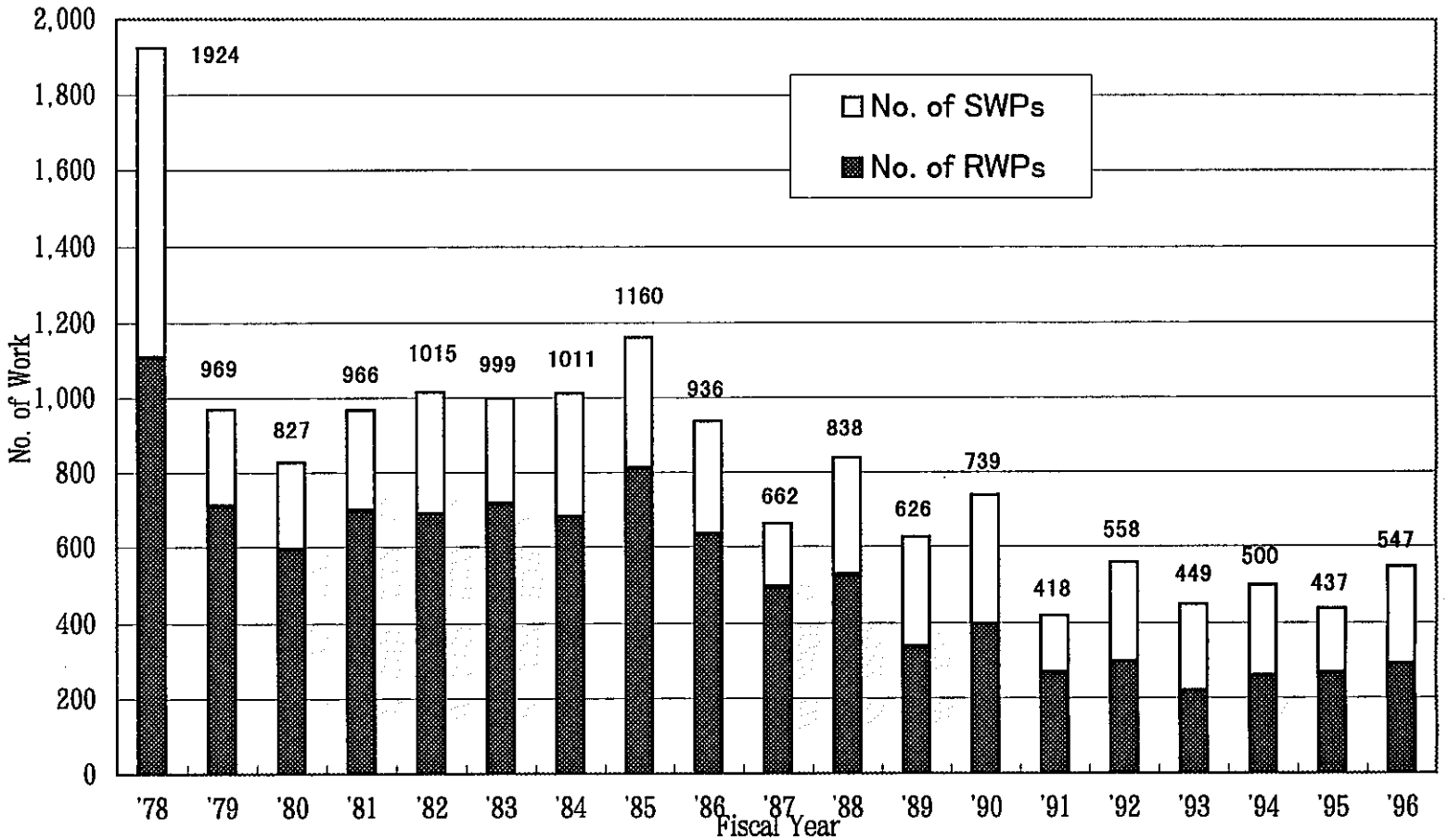


Fig.IV.4-3 History of Annual Total No. of Radiation Work

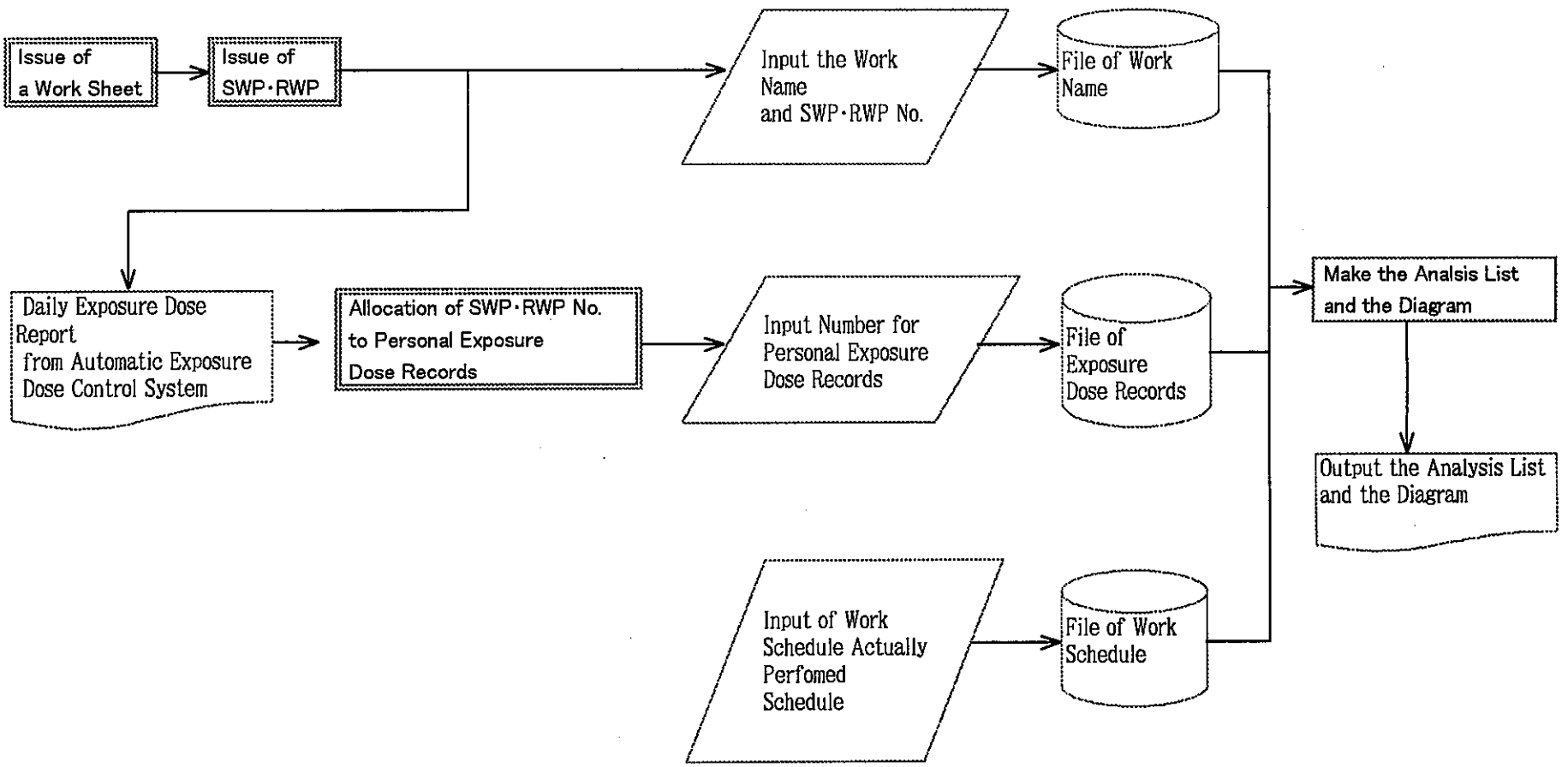


Fig. IV.4-4 Data Flow of Work Analysis Program

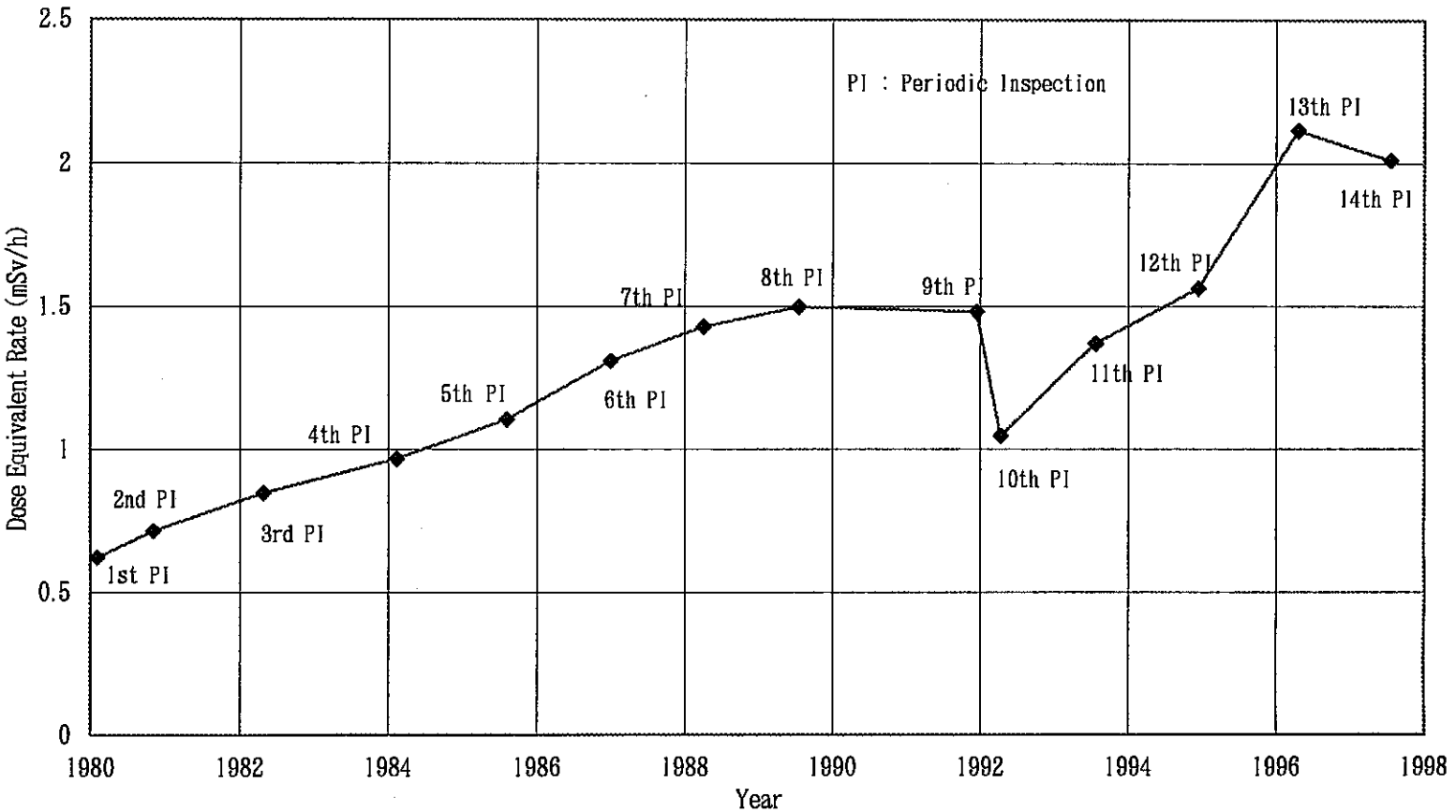


Fig. IV.4-5 Dose Equivalent Rate of Outlet Pipe of Reactor Coolant Pump (D)

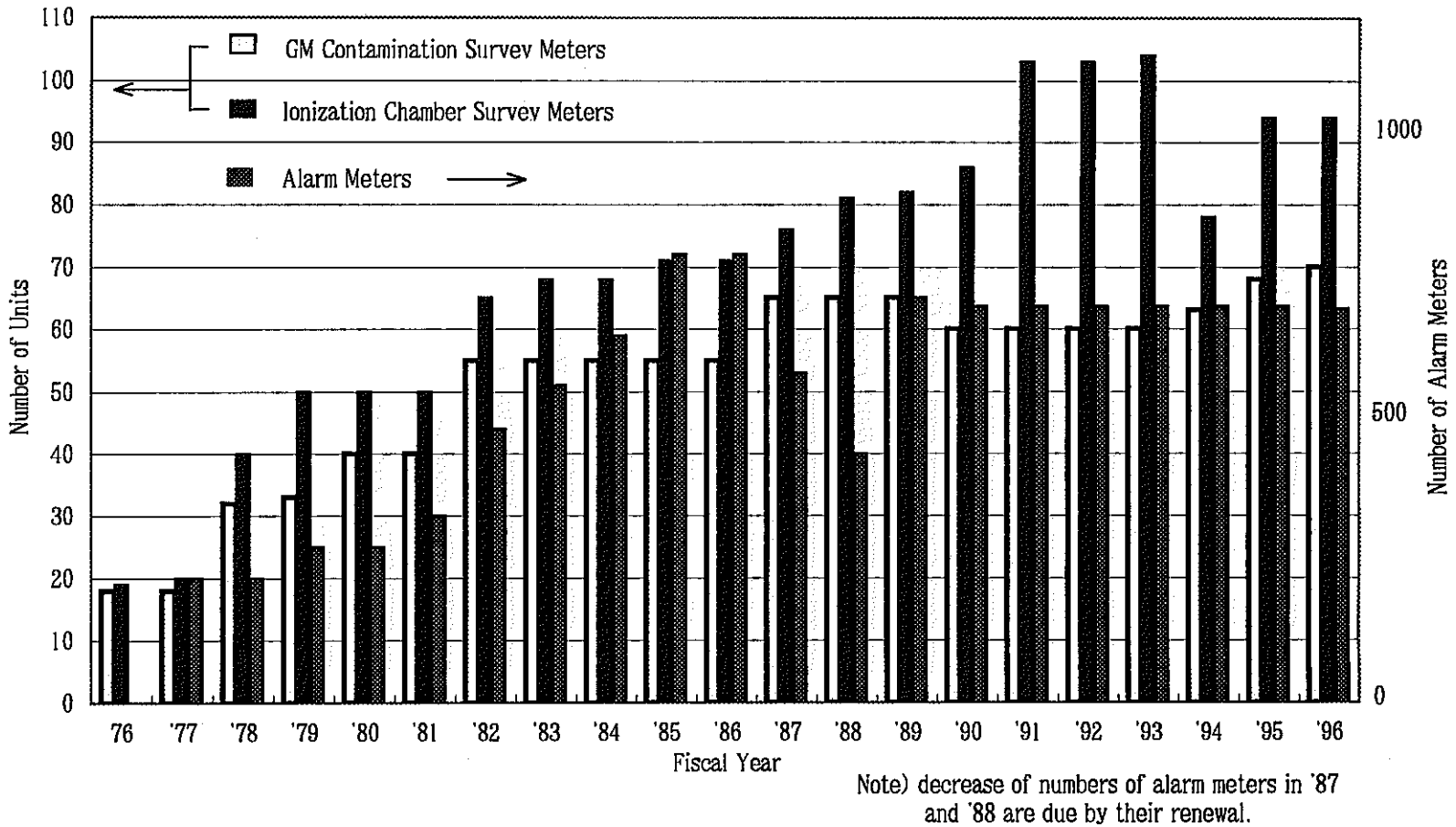


Fig.IV.4-6 History of Inventory of Measuring Devices

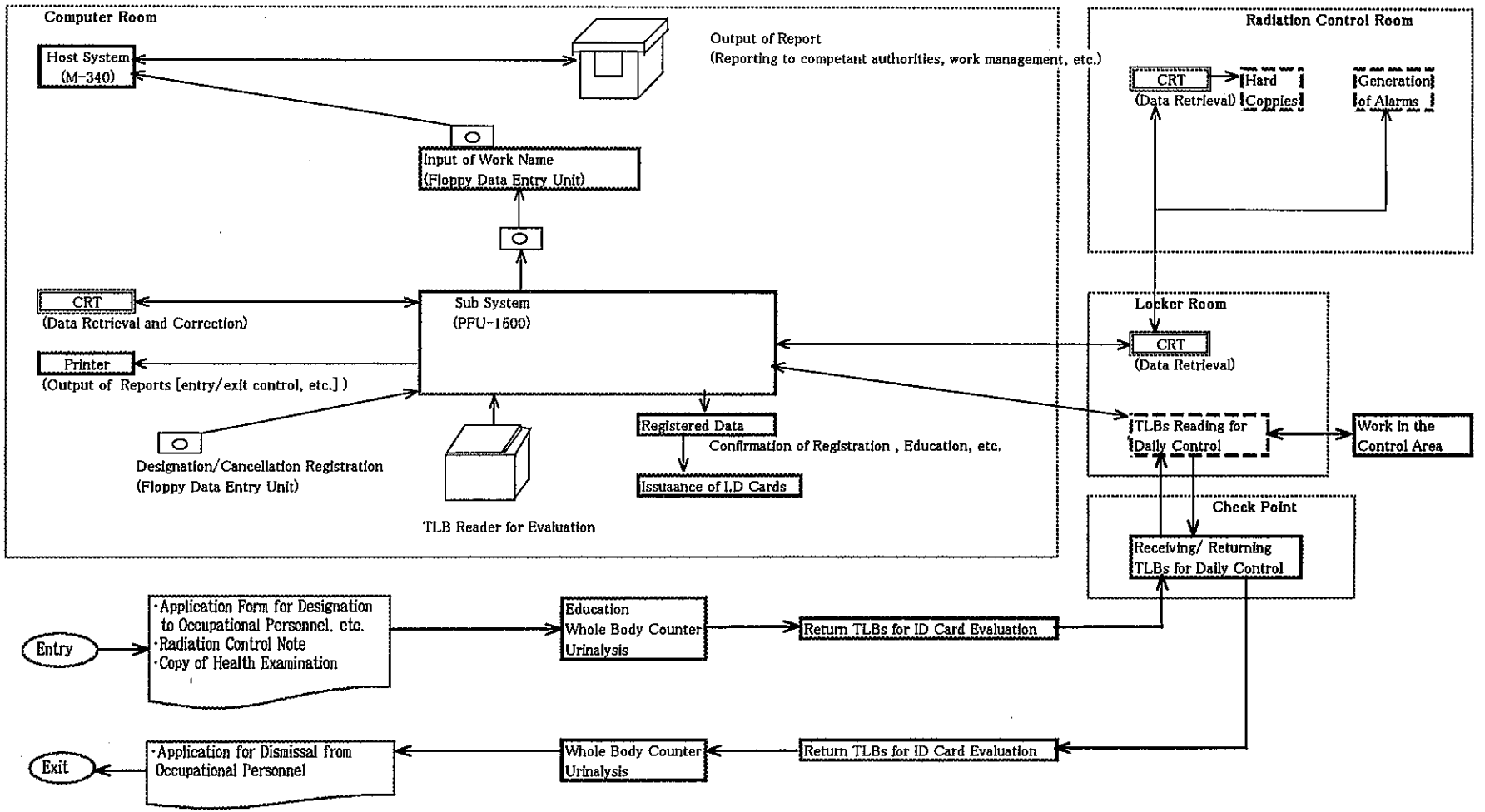


Fig.IV.4-7 Outline of Automatic Exposure Dose Control System

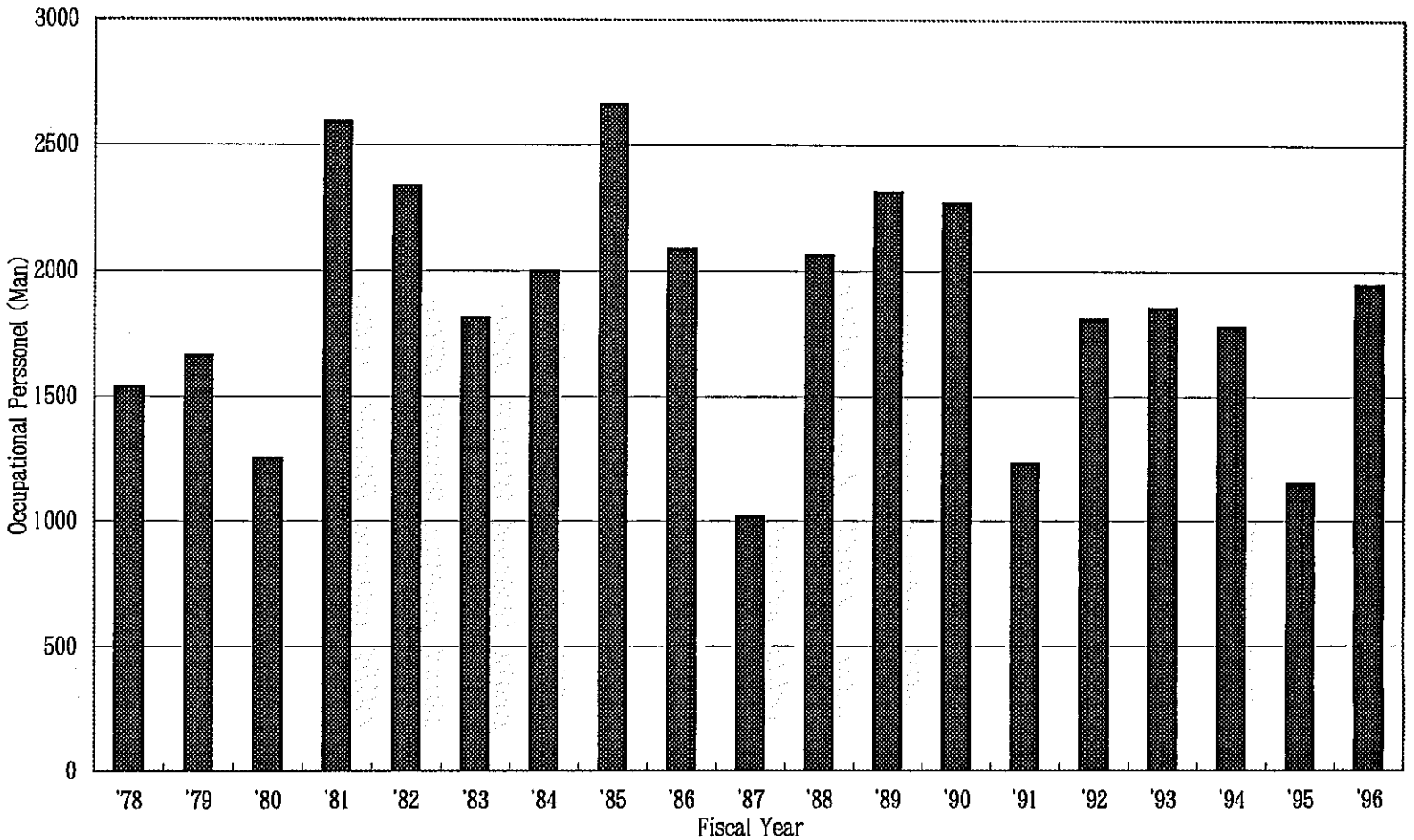


Fig. IV. 4-8 History of Annual Total Number of Radiation Occupational Personnel

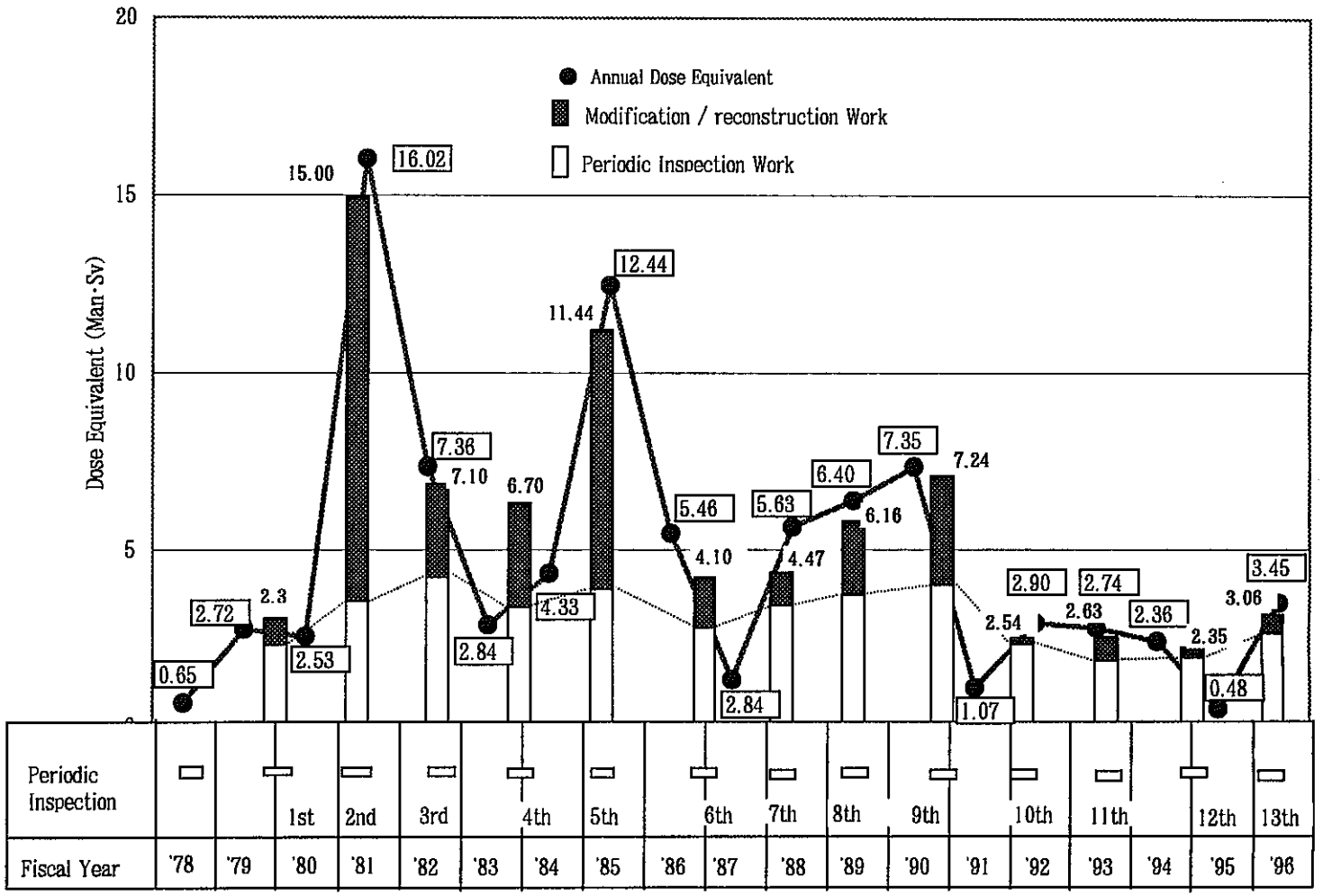


Fig. IV.4-9 Changes of Total Dose Equivalent

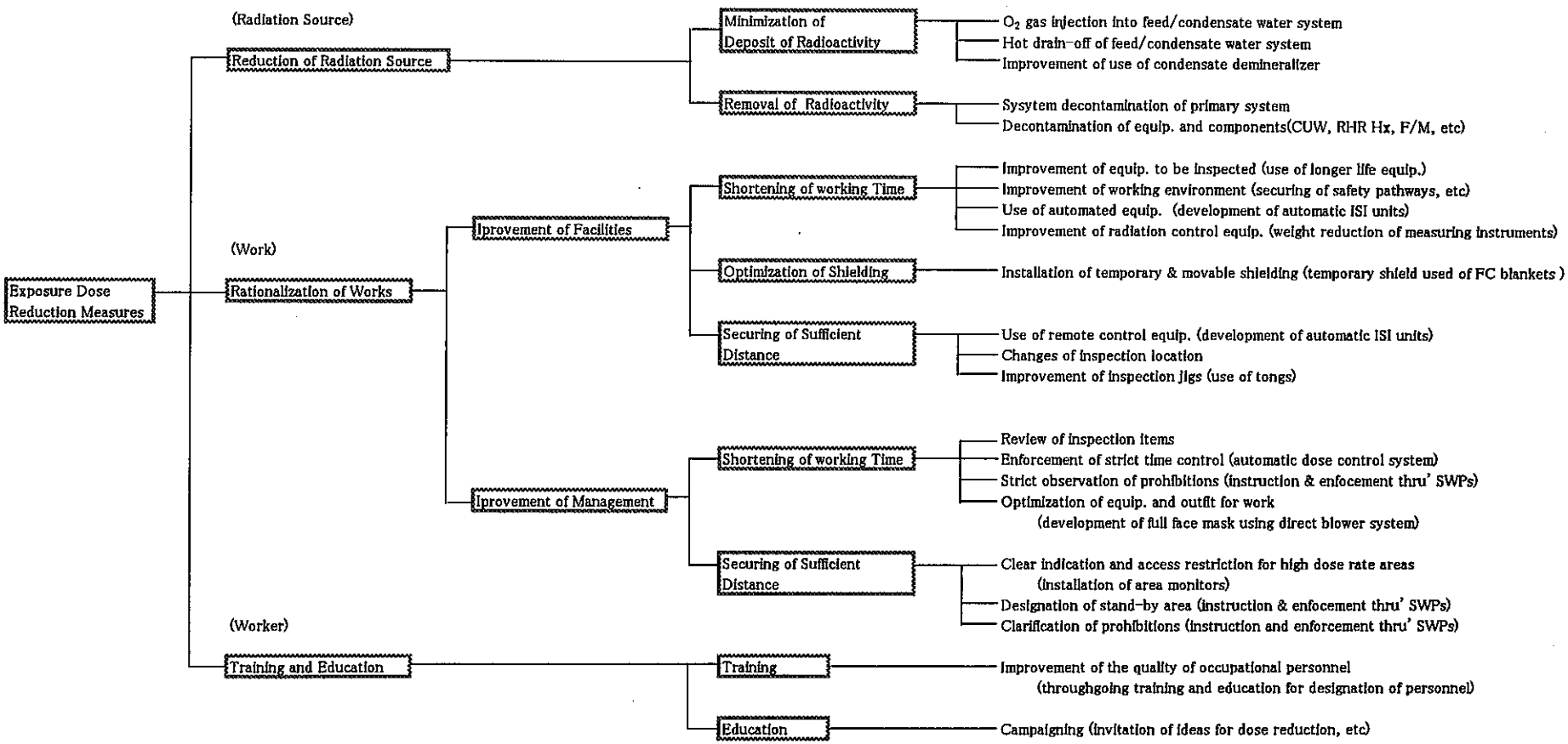


Fig.IV.4-10 Exposure Dose Reduction Measures in "Fugen"

Table IV.4-1 Introducing History of Entry/Exit Controlled Devices

Devices	Date of Update /Introduction	Effects of Introduction of Devices
Introduction of an Automatic Exposure Dose Control System (TLD Badge Read-out Gates)	Apr., 1985	The following work were rationalized: - Confirmation of designation and permission - Confirmation of the remaining valid term of registration - Confirmation of wearing TLD badges for daily control - Daily exposure dose control - Presence of n, β ray exposure
Introduction of an Automatic Alarm Meter Check-out Units	Jul., 1989	The following work were rationalized: - Confirmation of operation of alarm meters before lending out - Confirmation of set values for alarms - Elimination of the lending legder - Record-keeping of remaining dose at the time of checking out - Confirmation of personnel permitted to conduct high dose work
Updated of Body-surface Monitors	Aug., 1990	The following effects were achieved by updating: - Reduction of time required for measurement - Available the gate control - Automatic recording of conditions of contaminated sections

Table IV 4-2 Annual Changes of Dose Equivalent from Fiscal '78 through '89

		'78	'79	'80	'81	'82	'83	'84	'85	'86	'87	'88	'89	'90	'91	'92	'93	'94	'95	'96
No. of Staff	PNC Staff	180	219	219	240	228	205	214	211	197	193	199	203	200	200	195	208	198	193	194
(man)	Contracted Co.	1,340	1,440	1,031	2,316	2,095	1,610	1,789	2,440	1,885	817	1,860	2,119	2,070	1,031	1,615	1,644	1,580	955	1,750
	Total	1,520	1,659	1,250	2,556	2,323	1,815	2,003	2,651	2,082	1,010	2,059	2,322	2,270	1,231	1,810	1,852	1,778	1,148	1,944
Total Dose Equivalent	PNC Staff	70	220	230	363	303	165	240	360	350	290	380	337	334	170	257	198	210	116	225
(man·mSv)	Contracted Co.	570	2,500	2,290	15,654	7,059	2,673	4,090	12,080	5,100	1,010	5,250	6,063	7,016	902	2,639	2,540	2,151	366	3,223
	Total	650	2,720	2,530	16,016	7,362	2,838	4,330	12,440	5,460	1,300	5,630	6,400	7,350	1,072	2,896	2,738	2,361	482	3,448
Average Dose Equivalent	PNC Staff	0.4	1.0	1.1	1.5	1.3	0.8	1.1	1.7	1.8	1.5	1.9	1.7	1.7	0.9	1.3	1.0	1.1	0.6	1.2
(mSv)	Contracted Co.	0.4	1.7	2.2	6.8	3.4	1.7	2.3	5.0	2.7	1.2	2.8	2.9	3.4	0.9	1.6	1.5	1.4	0.4	1.8
	Total	0.4	1.6	2.0	6.3	3.2	1.6	2.2	4.7	2.6	1.3	2.7	2.8	3.2	0.9	1.6	1.5	1.3	0.4	1.8
Max. Dose Equivalent	PNC Staff	8.6	15.5	13.1	20.8	16.0	8.3	10.4	12.4	14.8	15.1	20.0	13.3	13.8	6.6	9.3	9.4	9.0	5.4	12.9
(mSv)	Contracted Co.	10.2	22.8	27.3	39.6	28.4	12.9	19.8	34.5	24.7	25.2	19.5	29.2	19.6	14.4	14.8	13.7	14.0	11.4	16.1
Major Programs		Jul.	2/1 ~4/30		4/20 ~11/5	9/24 ~2/17	2/15 ~7/19		7/3 ~1/10	1/7 ~4/17		4/16 ~7/20	7/19 ~12/1	12/14 ~5/23		4/26 ~8/20	8/18 ~1/7	1/6 ~5/18		4/22 ~8/8
(PI: Periodic Inspection)		First Synch-ronizing	1st PI		2nd PI	3rd PI	4th PI		5th PI	6th PI		7th PI	8th PI	9th PI		10th PI	11th PI	12th PI		13th PI

Appendix IV. 1 : Outline of Computerized Exposure Dose Data Acquisition and Processing System (Automatic Exposure Control System)

1. Outline of System

The automatic exposure dose control system in the "Fugen" Nuclear Power Station (hereafter called "Fugen") consists of the "Exposure dose control host system" and a "Rapid check system for entry/exit qualification" built on the Fugen operation management computer, and a exposure dose control sub-system using a sub computer.

The exposure dose control sub-system was updated in February 1998. New system is comprised of a sub computer, Alarm Pocket Dosimeters (hereafter called "APDs"), a APD data processing system and APD carrying check gates at entrance.

The sub-system performs the automatization of the entrance control into the controlled area of "Fugen" and the individual exposure dose control of workers. Fig. APP. IV.1 shows the configuration of the automatic exposure control system.

2. Main Function of System

(1) APD(100 Dosimeters)

The APD can displays not only the dose equivalent but also the lapsed time in the controlled area. It actuates the alarm when it gets to the alarm set point. It also receives the signals from the APD leading machine or the work information register, and memorizes the identifying the name of the work involved. Moreover, it outputs the signal enabling the identification of APD so that the ADP carrying check gate can recognize it at the entrance.

(2) APD Lending Machine (11 Machines)

4 lending machine can keep up to 100 APDs charging and has the function for worker identification, APD data reading, checking and locking, work name registration, output of the dose equivalent record, instruction for users, etc.

(3) APD Carrying Check Gate (8 Gates)

① For Changing Room

Two gates for entering and two for leaving are installed at the entrance of the main building changing room. Those workers who are not carrying APDs or not fully qualified for entering are prevented from entering, and the alarm is actuated in a watch room.

② For Reactor Building

One gate for entering and one for leaving are installed at the entrance of the reactor building to check if a APD is carried with and at the same time, to transfer the work

name involved to the main computer. The list of the entering/leaving workers are also displayed on the CRT (Cathode Ray Tube) installed in the building.

③ For Heavy Water Upgrader Building

One gate for entering and one for leaving are also installed at the entrance of the heavy water upgrader building.

3. Sketch drawing of main equipment and layout drawing

The sketch drawings of APD, APD lending machine and APD checking gate (gate for checking if APD is carried), as well as the layout drawing, are illustrated in Fig. APP. IV. 2 while the layout drawing of the main building changing room is in Fig. APP. IV. 3.

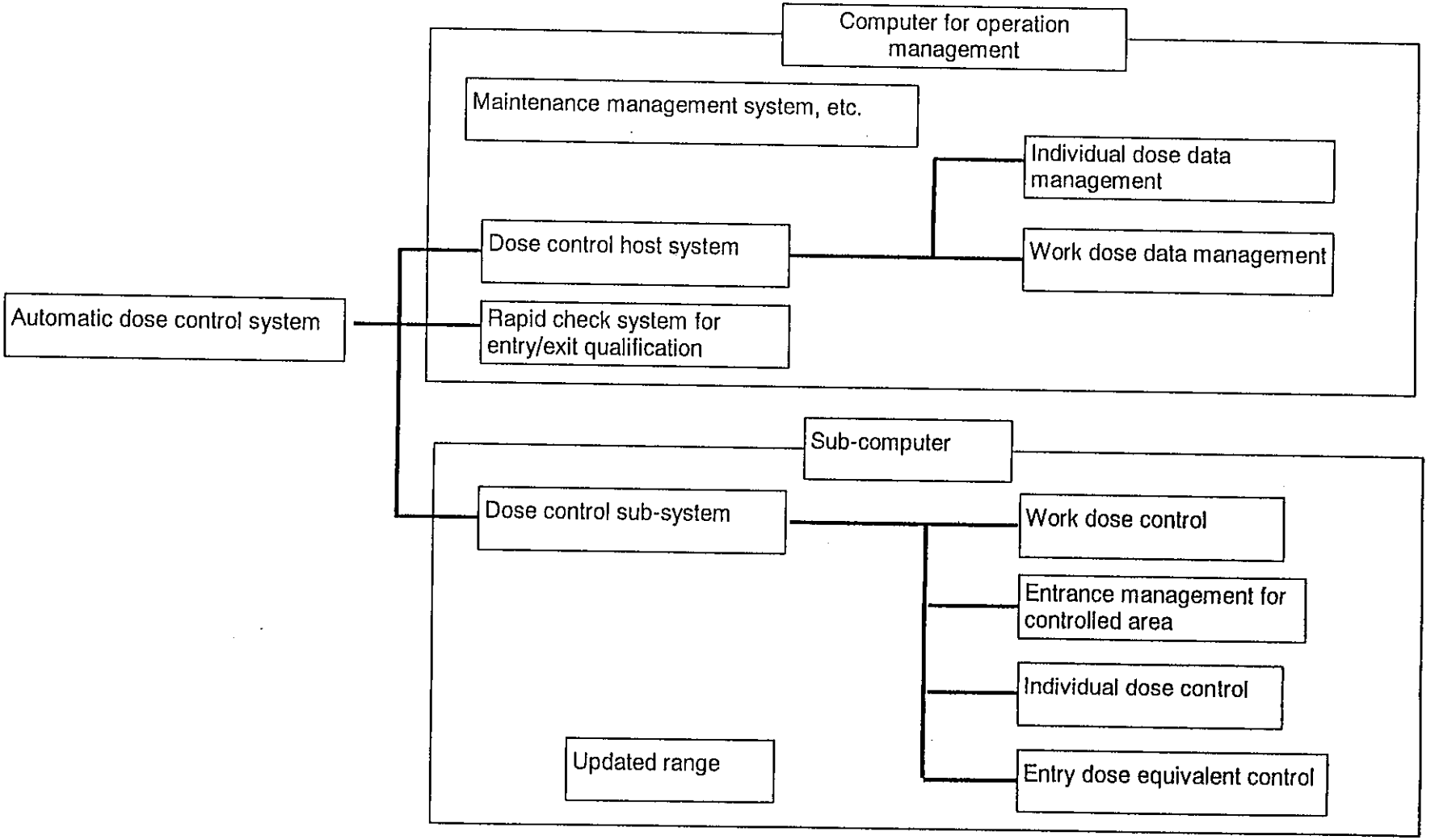
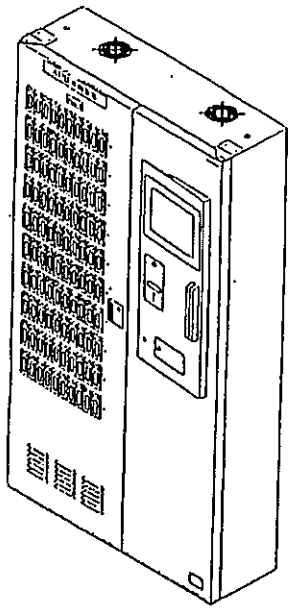


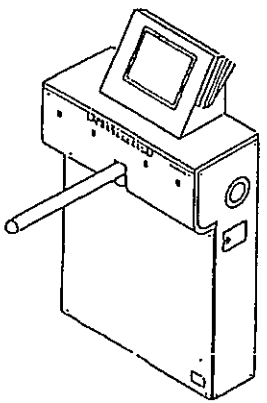
Fig.APP.IV.1 Automatic Dose Control System Configuration



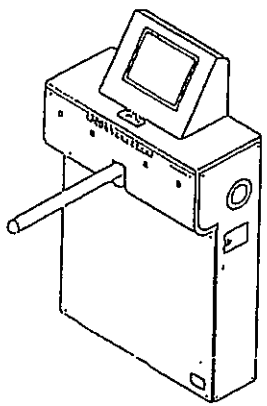
Sketch Drawing of APD



Sketch Drawing of APD Lending Machine

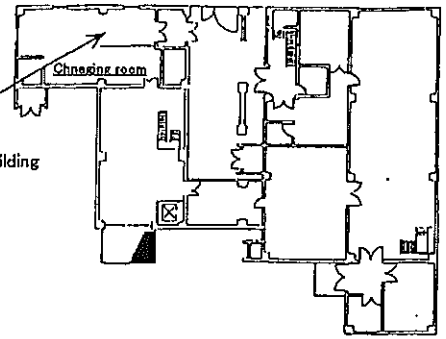


Installed at exit of main building changing room

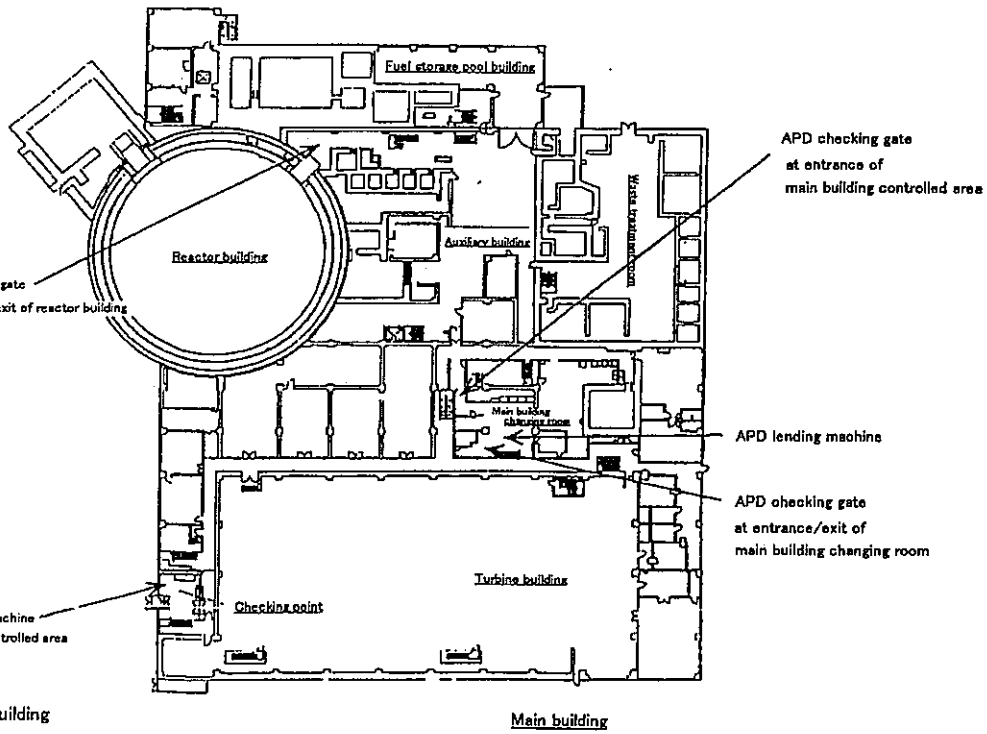


Installed at exit of reactor building,
main building and havey water upgrader building

Sketch Drawing of APD Checking Gate



Heavy Water Upgrader Building



Main building

Locations of APD Lending Machine and APD Checking Gate

Fig.APP.IV.2 Sketch Drawing and Locations of APD Lending Machine and APD Checking Gate

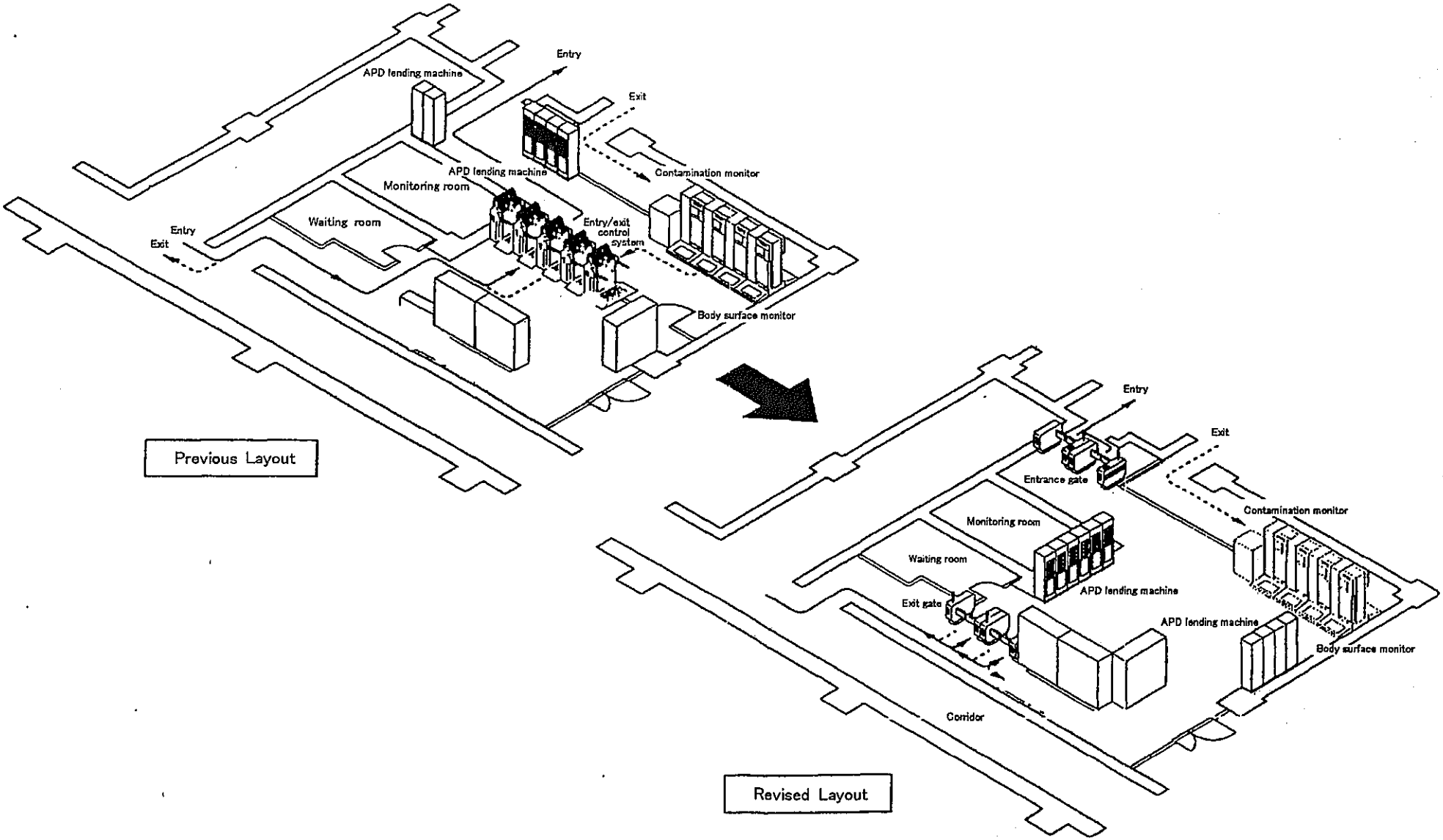


Fig.APP.IV.3 Layout Drawing of Main Building Changing Room

Reference

[IV-1] IAEA Safety Series No.50-SG-D9, Design aspects of radiation protection for nuclear power plants, 1985

[IV-2] IAEA Safety Series No.50-C-O(Rev. 1), Code on the safety of nuclear power plants: Operation, 1988

[IV-3] IAEA Safety Series No.50-SG-O5, Radiation protection during operation of nuclear power plants, 1983

[IV-4] IAEA Safety Series No.50-C/SG-Q Q13, Quality assurance in operation, 1996

[IV-5] H. Higuchi, "Dose Reduction Measures in Fugen", OECD NEA Specialist Meeting on Dose Impact on Work Management, July 1994

[IV-6] PNC Technical Review No.53, 1985. 5, P18～35, "Radiological Safety Management in Fugen" ; N. Sakurai, et al.

[IV-7] PNC Technical Review No.69, 1989. 3, P58～64,"Reduction of Radiation Exposure with Improvement of Radiological Safety Management in Fugen" ;

[IV-8] PNC Technical Review No.81, 1992. 3, P54～58, "Development of ^{3}H Control" ; H. Nakajima, et al

[IV-9] T. Kitabata, et al.; "Heavy Water Management and Radiation Protection against ^{3}H in Fugen", (PNC TN3413 97-002), 19th Japan-Korea Nuclear Seminor, Souel, Korea, Sept. 1997

[IV-10] H. Higuchi,; "Full System Chemical Decontamination in Fugen", (PNC TN3410 95-012), OECD NEA ISOE Topical Meeting, Paris, France, Nov. 1995

[IV-11] M. Koike, et al.; "Development and Experience of Chemical Decontamination in Fugen (I)", P382～392, Vol.38, No.5, 1996. 5, J. of Atomic Energy Society of Japan

[IV-12] Y. Naoi, et al.; "Development and Experience of Chemical Decontamination in Fugen (II)", P511～520, Vol.38, No.6, 1996. 6, J. of Atomic Energy Society of Japan

V. Chemistry Control [V-1][V-2] [V-3] [V-4] [V-5] [V-6]

1. Outline of Chemistry Control

Chemistry control in "Fugen" mainly consists of the quality control of the reactor cooling water and the heavy water moderator. By considering the characteristics of "Fugen" based on experiences obtained through operation of BWRs, efforts have been made to improve various water chemistry control methods in order to ensure stable plant operation and reduced radiation dose equivalent.

To achieve stable plant operation, the conditions of water quality were continuously measured and monitored and, at the same time, environment of water quality was attempted to be improved by the deaeration operation before startup and the hydrogen injection to the reactor water for the first time in Japan to reduce the dissolved oxygen concentration, in pursuit of prevention of SCC of the reactor cooling piping and others.

For reduction of radiation exposure dose, the following actions are being taken to decrease the flow of corrosion products including Co, the main factor of exposure, and its carrier Fe into the core from the feed water / condensate system:

(i) prevention of corrosion of the feed water / condensate system piping by injection of oxygen into the feed water system

(ii) improvement of crud Fe removal performance through enhanced operation of condensate demineralizer (improvement of the feed water quality)

(iii) decrease in the crud Fe intake to the core during startup by hot drain off of the feed/condensate water system

(iv) decrease in the crud Fe intake to the core during startup by clean-up operation of the feed water / condensate system before startup

Experiences of the preceding heavy water reactors were applied to the heavy water quality control in "Fugen". However, we faced many new challenges during operation of "Fugen". Through about ten years operating experiences and various demonstration tests, we have established the heavy water quality control.

Actions to improve the heavy water quality control include:

(i) control of oxidation and degradation of resins by decreasing the circulation heavy water temperature in the heavy water purification resin bed

(ii) decrease of impure chlorine sources by adoption of the extremely low chlorinated anion exchange resin for cleaning up heavy water

(iii) removal of impurities by enhanced helium gas purification operation

All of these measures contributed to achieve the improved water quality.

2. Policy of Chemistry Control

As the chemistry control work, we have been actively involved in maintenance and monitoring of integrity of structural materials, such as component piping, fuel, purification system demineralizer and others, as well as reduction of exposure and technological development. To ensure proper execution of the work, the control criteria including control methods and goals were prescribed in the regulations or guidance of the Safety Regulation of the reactor facility and the chemistry control follows this regulation.

Fig. V. 2-1 shows the system of chemistry control regulations in "Fugen". Table V. 2-1 describes history of R&D programs to resolve problems encountered during operation as well as regular routine tasks for operation management.

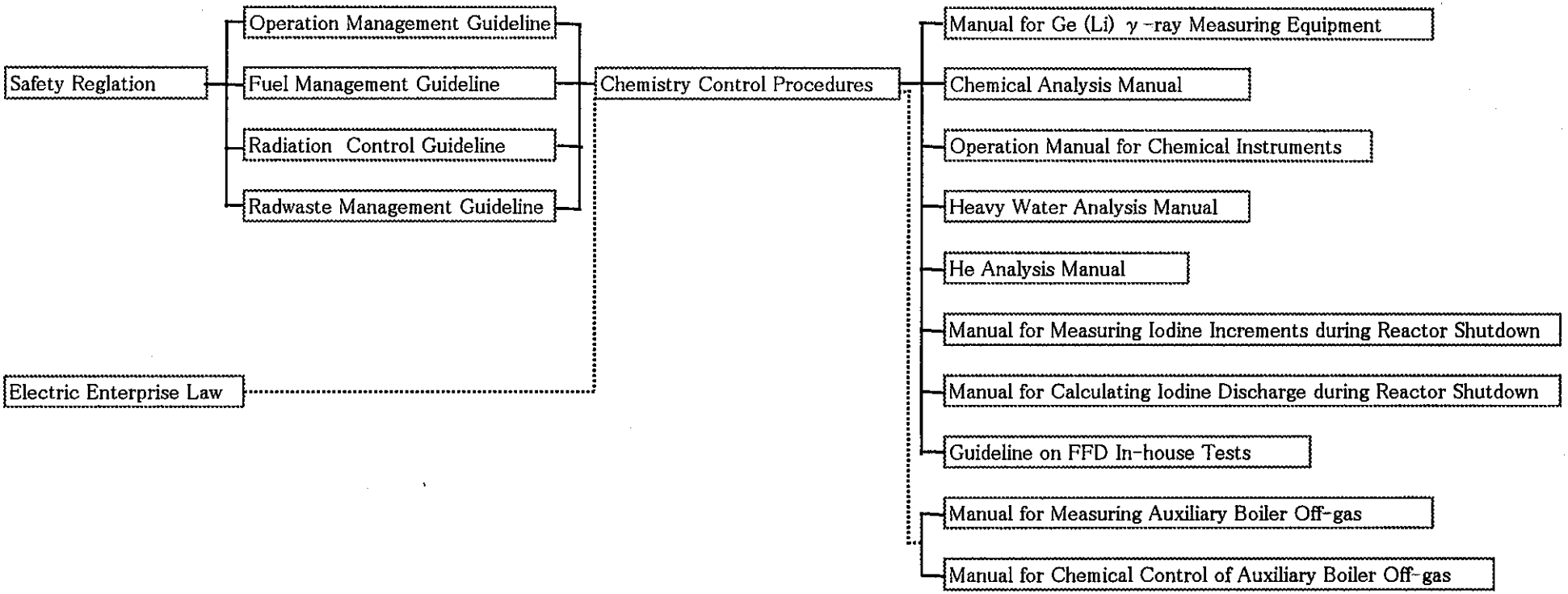
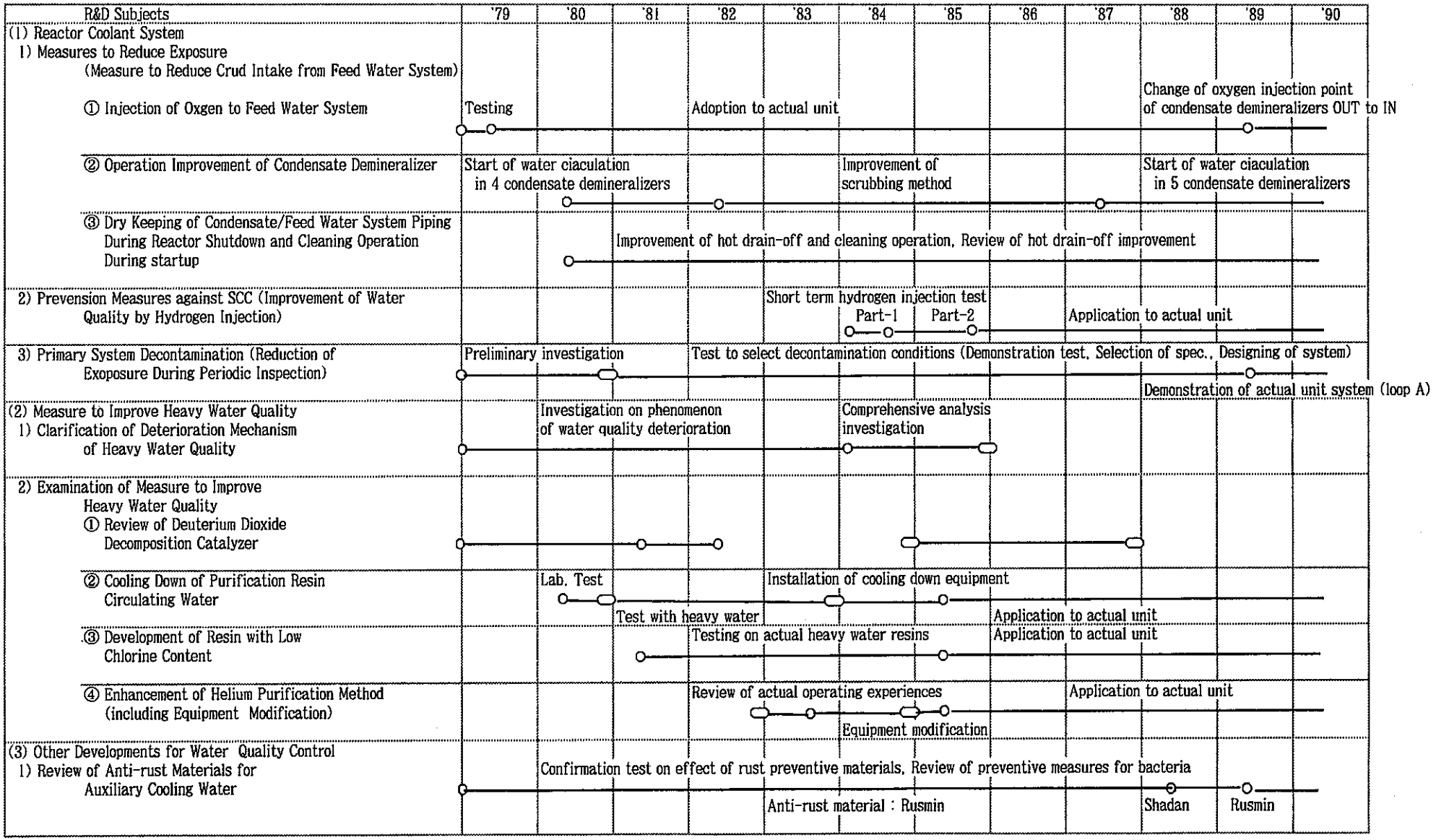


Fig.V.2-1 Systematic Chart of Chemistry Control Regulations

Table V.2-1 History of R&D Related with Chemistry Control



3. Organization

In April, 1975, the Chemistry Control Group was organized to establish the laboratory (chemical laboratory) and to make the control of the system flushing water. Since then, in accordance with the full power operation of the plant, the Chemistry Control Group of the Safety and Chemistry Section were assigned to the chemistry control of liquid and gas including the reactor coolant and heavy water, material testing, reduction of exposure, system decontamination and others, with modifying the work organization according to the situations.

As of 1998, the staff totals 16 persons; 8 PNC personnel and 8 contractors. The work organization is shown in Fig. V. 3-1.

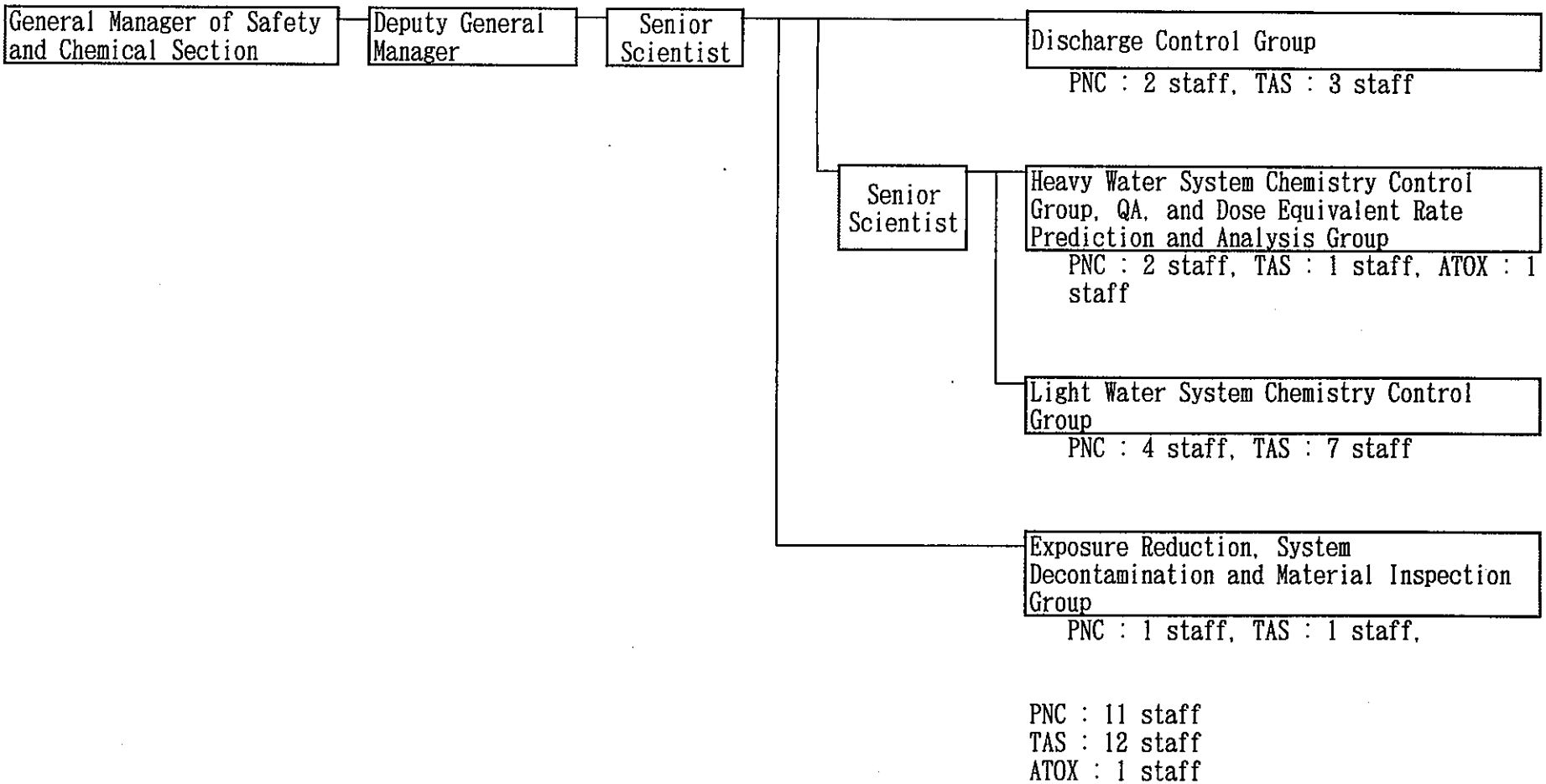


Fig.V.3-1 Organization of Chemistry Control (as of March, 1998)

4. Results and Evaluation of Chemistry Control of Light Water System

4.1 Introduction

The chemistry control of the light water system mainly consists of the water control.

It covers the following various systems:

(i) reactor cooling system
(ii) thermal shield cooling water system
(iii) fuel pool cooling system
(iv) reactor auxiliary cooling water system
(v) turbine cooling water system
(vi) auxiliary boiler water
(vii) sea water system
(viii) demineralized water produced by the demineralizer

Through the above mentioned water quality control, the chemistry control aims at

(i) maintenance and control of integrity of the system components including fuel assemblies,
(ii) control of deposit of radioactive nuclides on the reactor cooling system piping and others,
(iii) monitoring of integrity of the clean-up system demineralizers, and
(iv) maintenance of purity of demineralized water, service water, gases and others.

To realized the above, the control criteria were established and whether they were satisfied or not was checked by patrol inspections of process instruments, periodic sampling and analysis. Particularly, in applying the hydrogen injection method to reduce the stress corrosion cracking of the reactor cooling system SUS304 materials, we established the unique criteria on the dissolved oxygen concentration based on the in-plant tests, taking the leadership in this field in Japan.

4.2 Criteria and Method of Water Quality Control

Table V. 4-1 shows the water quality control criteria, their rationales, the measuring methods of water quality and the measurement frequency. Water quality is monitored by daily patrol checking and periodic sampling/analysis of process instruments such as pH meters, electric conductivity meters and dissolved oxygen meters. When any anomaly is detected, proper actions to improve water quality are taken to ensure good water quality.

(1) Dissolved Oxygen Concentration in Reactor Cooling Water

With a view to reducing stress corrosion cracking (SCC) of the reactor coolant system SUS304 materials, hydrogen is injected to the feed water to reduce the dissolved oxygen in the reactor water. Results of the slow strain rate test in the hydrogen injection test showed 1% or below of the fractured surface ratio due to SCC (cross section area where SCC occurred to the total fractured areas) when the dissolved oxygen was 0.03 ppm or below (Fig. V. 4-1), which indicated a considerable improvement in the sensitivity to SCC. As the slow strain rate test is considered to be a severe material test, a possibility of SCC is low in the actual plant when the fractured surface ratio is 1% or below.

Fig. V. 4-2 showed the dissolved oxygen concentrations in the reactor cooling system with hydrogen injection rate. Reducing the dissolved oxygen concentration to 0.03 ppm or below in the inlet of the clean-up system demineralizer where reduction effect is the smallest can prevent SCC from occurring in the other reactor cooling system regions.

(2) Chlorine Ion, Electric Conductivity and pH in Reactor Cooling Water

Chlorine concentration is controlled to prevent SCC in the austenitic stainless steel which may be accelerated by the coexisting dissolved oxygen. The dissolved oxygen concentration in the reactor water during normal operation is about 0.2 ppm before hydrogen injection, and the allowable chlorine ion concentration under this condition is found to be 0.6 ppm; thus the control target is determined to be 0.2 ppm, with considering variations of the data. The limit value of 1 ppm is allowable value in short-term operation.

As the demineralized water with a high purity is used for the reactor water, electric conductivity and pH are determined by the dissociation constant of the chloride in the reactor water.

① Electric Conductivity

When 0.2 ppm of chlorine ions exist, the highest electric conductivity achieved is 2.5 μ S/cm in the form of HCl, as shown in Fig. V 4-3 , while in the form of NaCl it is 0.8 μ S/cm. Accordingly, the control target of electric conductivity is determined to be 1.0 μ S/cm.

② pH

The control target is determined to be 5.5~8.5 . It should be noted that the limit value corresponding to 1% chlorine concentration is 10 μ S/cm for electric conductivity and 4~10 for pH.

(3) Silica in Reactor Cooling Water

Silica will be carried over to the steam when its concentration in the reactor water is high, and precipitated and deposited due to a decrease in steam pressure in the turbine. Precipitation will not occur if its concentration in the steam is lower than the solubility.

The solubility at the outlet of the low pressure turbine is found to be 0.0155 ppm and the corresponding silica concentration in the reactor water should be 9 ppm. With adding a margin to this value, the control target was determined to be 2 ppm.

(4) Insoluble Impurities in Reactor Cooling Water

Insoluble impurities in the reactor water are determined by the amount of intake of metallic impurities and others from feed water. Concentration of metallic impurities in the feed water system was assumed to be 12 ppb (actually ~5 ppb), plus amount of corrosion products generated in the core and the deposit rate on fuel rods was assumed to be 0.5 . By adding a margin to the resulting value of 0.72 ppm, the impurities were decided to be 1 ppm.

(5) I-131 Concentration in Reactor Cooling Water

I-131 concentration is determined by the rate of release from a failed fuel, capacity of the reactor clean-up water system, carry-over rate to the main steam side and others. The limit of I-131 concentration during operation is 0.8 μ Ci/g (2.7×10^4 Bq/g), which corresponds to 54 μ Ci/g (2.0×10^6 Bq/g) of the maximum allowable concentration of fission products used in the evaluation of the exposure dose to the local public in the case of the main steam pipe rupture accident.

In this evaluation, the additional discharge of I-131 to the reactor water was assumed to be 4,000 Ci (1.5×10^{14} Bq). The operation limit uses the 50% of this value, 2,000 Ci (7.4×10^{13} Bq).

(6) Electric Conductivity of Condensate Water

This is a set value to monitor leakage of sea water from the condenser. The upper limit of normal control value of the condensate water electric conductivty is determined to be 0.2 μ S/cm.

4.3 Actual Results of Water Quality Control

(1) Actual Results of Reactor Cooling Water Quality

(a) Actual Results of Reactor Cooling Water Quality

The reactor water in "Fugen" is recirculated at 8,800 t/h; mixed bed demineralizers (two units: amount of cation resin : anion resin = 1:1, 1.35 m^3/unit) are used in the Clean-up Water System (CUW) to maintain the water

quality, with cleaning up at the rate of about 60 m^3/h. The condensate/ feed water systems are provided with four demineralizers with the cation/anion ratio of 2:1 and 3.3 m^3/unit. The whole volume of the condensate water is cleaned up at 910 t/h. The capacity of the clean-up water system is about 7% of the feed water flow rate, which shows good efficiency for cleaning up the reactor water compared with 1~2% obtained in the BWRs in Japan. The reactor water is controlled to remain neutral without adding chemicals. The dissolved oxygen concentration in the reactor water is 10 ppb, with a pH of 7 and 0.1 μ S/cm of electric conductivity, showing good water quality environment due to hydrogen injection.

Insoluble impurities (SS) and boron have been below the measurement limits (1 ppm and 5 ppb or below, respectively) and chlorine ion is scarcely detected during operation since startup. Electric conductivity at the demineralizer outlet of the clean-up water system remains $\sim$ 0.08 μ S/cm and the DF (Decontamination Factor) of radioactivity is 1,000 or more.

When the reactor stops operation due to periodic inspections or other reasons, the reactor water quality slightly deteriorates because of temporary stop of the clean-up water system, dissolution of carbon dioxide caused by opening of the steam drum and release of crud from piping followed by temperature changes of the reactor water. Chlorine ions may increase to 20~50 ppb, which may be attributed to the ingress of materials containing chlorine deposited to rubber gloves during overhauling.

During annual inspections, it is Co-60 that contributes the dose equivalent rate in the work areas of the reactor cooling system, by over 80%. Increase of ionic Co-60 concentration raised the dose equivalent rate of the reactor cooling system piping by 15%, as shown in Fig. V. 4-4, compared with the predictive value without hydrogen injection. However, its effect on the total dose equivalent during annual inspections cannot be found.

During reactor operation I-131 concentration in the reactor water constantly remains 0.05 Bq/cm^3, and additional discharge of I-131 during reactor shutdown dose not exceed $4x10^7$ Bq. No indication of fuel failure has been identified so far.

Tritium concentration in the reactor water, as shown in Fig. V. 4-5 , was about 10 Bq/cm^3 at the time of startup. It gradually increased to 70 Bq/cm^3 in 1985 and rapidly increased to 300 Bq/cm^3 in 1986-87. This may be caused by activation of natural deuterium contained in light water, as well as ingress of

drain including tritium generated in the heavy water system such as the emergency gas process system drain and the drain generated from resin exchange of the heavy water system via the liquid waste treatment system to the light water system.

Increased tritium concentration can be lowered by decreasing the recovery rate from the liquid waste treatment system as the condensate water or by blowing the condensate storage tank water. Such an extent of the tritium concentration increase will not pose any problem for operation management, but considerations are given not to allow tritium in the heavy water to enter the light water system.

(b) Improvement of Reactor Cooling Water Quality by Hydrogen Injection [V-7] [V-8]

① Application of Hydrogen Injection Method

In "Fugen", the control criteria on the dissolved oxygen and chlorine concentration during operation are determined for prevented measures against SCC of the reactor cooling system piping and other components. In addition, deaeration operation is performed during startup and the dissolved oxygen concentration is reduced to 0.2 ppm or less over 100℃ not to make sensitive environment to SCC. However, SCC was detected during the scheduled shutdown in November, 1957, and implementation of systematic SCC countermeasures was made, such as change of piping materials of the reactor cooling system, clean-up water system and feed water system to SUS316L and Induction Heating Stress Improvement (IHSI) process for the piping welds.

On the other hand, the above mentioned measures are difficult to apply to small diameter piping. As appropriate measures for inlet pipes and outlet riser pipes in the pipe bundle structures, hydrogen injection to the feed water system to reduce the dissolved oxygen concentration in the reactor water were reviewed in order to prevent occurrence of SCC by improving the water quality.

Results of the short-term hydrogen injection test for about a year revealed that injection of about 1.3 ppm hydrogen to the feed water system led to a reduction of the dissolved oxygen concentration in the inlet header to 20 ppb (100～180 ppb before injection).

The slow strain rate test and the corrosion potential test of the sensitized SUS304 stainless steel also gave insights that SCC of SUS304 stainless steel

could be almost prevented if oxygen concentration be 30 ppb or below and the corrosion potential -230~-180 mV(SHE).

Hydrogen injection caused change of the reactor water from the oxidizing atmosphere to the reducing atmosphere, and increase in carry over of ^{16}N in vapor and high dose equivalent of the main steam by a factor of about four. However, it was confirmed that these phenomena did not pose significant problems to plant operation management.

According to these test results, the hydrogen/oxygen generator using bipolar filter press type alkali water electrolytic method has been installed since December, 1985, hydrogen is continuously injected to the feed water pump inlet to maintain the dissolved oxygen concentration in the reactor water 20 ppb or less.

Currently, an amount of hydrogen injection is increased to keep the oxygen concentration in the reactor water 10 ppb or less.

② Change of Radioactive Nuclide Concentrations in Reactor Water Due to Hydrogen Injection

As the reactor water environment was changed to the reducing atmosphere due to hydrogen injection, dissolution of crud deposited on fuel surfaces was induced and concentrations of Co-60, Co-58, Mn-54, iron ions and nickel ion tend to increase.

There is a good correlation between Mn-54, Co-58 and Co-60 in crud form and iron concentration in the feed water. Cr-51 concentration in the reactor water shows different behavior from the above mentioned nuclides. Chromium oxides are reduced from hexavalent chromium (in ionic form) to trivalent (in crud form) by oxidation-reduction equilibrium, resulting in an increase of the crud form Cr-51 concentration. Following an increase deposit of the crud form Cr-51 with the operating time elapsed, the oxidation-reduction equilibrium also proceeds reversely. As a result, both the ions and the cruds gradually increase to reach the equilibrium condition for one cycle operation.

Total chromium concentration combined with those of ions and cruds slightly increased after hydrogen injection.

$$4CrO_4^{2-} + 8H^+ = 2Cr_2O_3 + 3O_2 + 4H_2O$$

$$2Cr_2O_7^{2-} + 4H^+ = 2Cr_2 + 3O_2 + 2H_2O$$

Changes of radioactive nuclide concentration in the reactor water due to hydrogen injection slightly raised the dose equivalent rate of the reactor coolant system piping; however, thorough implementation of measures to

reduce exposure and advanced radiation control technique have contributed to avoiding any effects on total dose equivalent during the periodic inspections. Therefore, such increases may not cause problems to plant operation management.

③ Change of Dose Equivalent Rate of Main Steam System Due to Hydrogen Injection

Indications of the monitor of the main steam piping increased by a factor of 3 ~5 after hydrogen injection.

It is ^{16}N which is formed by nuclear reaction ^{16}O (n, p) ^{16}N that controls the dose equivalent rate of the main steam system. As shown in Fig. V. 4-6, investigation on causes of increase in the transfer rate of ^{16}N to the main steam with using ^{13}N as a tracer revealed quantitatively that the chemical form of ^{16}N in the reactor water would shift from the anion form (NO_2^- , NO_3^-), which would be difficult to transfer to the main steam, to the cation form (NH_4^+), easy to transfer, and that this resulted in the increased transfer rate to the main steam system.

(2) Results of Condensate/Feed Water Quality [V-9]

The condensate/feed water system and steam bleeding system are composed of carbon steel piping, which generates iron rusts due to corrosion and elution. The rusts include the ionic form and the crud form, both of which are carried to the core by the feed water, deposit on the fuel cladding and are activated.

Then, they will elute into the reactor water again, deposit on the internal surface of components and piping and increase the dose equivalent rate, causing an increase in dose equivalent rate during maintenance and inspection work executed in the periodic inspections and others.

In pursuit of reduction of the dose equivalent, measures to reduce iron intake are being implemented in terms of water quality control as shown in Fig. V. 4-7 .

Fig. V. 4-8 shows contents of such measures, their implementation schedule and changes of the amount of iron intake.

(a) Injection of Oxygen to Condensate/Feed Water System

Experiences with the leading BWRs indicated that injection of the oxygen gas to the feed water system would result in formation of oxidized protective films on the surface of the carbon steel, which would be effective to prevent iron from eluting.

It has been determined that, based on results of various confirmatory tests, the oxygen gas should be injected from the condensate demineralizer inlet to control

ionic concentration to 1 ppb at condensate demineralizer inlet and iron ionic concentration at the No. 5 feed water heater outlet to remain 0.08 ppb, respectively.

(b) Improvement of Condensate Demineralizer Operation

"Fugen" is equipped with 4 condensate demineralizer columns are flow rate of condensate/feed water is 930m^3/h. The volume of ionic iron which was considerably brought into the core at the beginning of operation could be controlled by oxygen injection to the feed water system. Crud iron was determined to be controlled by improving the removal efficiency of the condensate demineralizer, and operation method for condensate demineralizer was changed as shown in Fig. V. 4-9. Operations of condensate demineralizer are those; water circulated through 3 beds from 1978 (at the beginning of startup test) to 1980, 4 beds from 1980 to 1987, and 4 beds circulation and 5 beds in use from 1987 up to now. And the period of water circulation is; 2 week circulation from 1987 to March. 1981, 4 week circulation from 1981 to 1987, 8 week circulation from 1987 to March. 1997, and 4 week circulation after March. 1997. Especially, after hydrogen injection was started in December. 1985, difficulties have been encountered to remove crud by the condensate demineralizers.

For aiming the reduction of Fe in feed water, high removal efficiency resin for crud is examined in real plant from the 10th periodic inspection held in July 1992 to the 12th periodic inspection in December 1994, and this test results showed that although the removal efficiencies of cruds had been increased, the total organic carbons (TOC) had been so dissolved out more than the usual resin used that electric conductivity of reactor water at the time of reactor startup became high. Then we returned to use the usual one.

On the other hand, at the 13th periodic inspection in 1996, side-blow of drain water was done for getting high efficiency of iron drainage in condensate demineralizer scrubbing drainage. However, none of improvement effects was obtained. Testing was made on high removal functional resins for crud, which has equal quantity of dissolved TOC as the usual resin and has high crud removal efficiency.

(c) Hot Drain off of Condensate/ Feed Water System

A considerable amount of metallic impurities generated as iron rusts during long-term shutdown, such as periodic inspections, is brought into the core in the reactor startup. To control this, the hot drain off measure is taken for dry

keeping of the condensate/feed water systems during the reactor shutdown to prevent formation of rust on the surface of carbon steel. Application of the hot drain off method contributed to reduction of cleaning operation of the water system before startup to about 2 weeks; it also resulted in the decreased iron intake during startup by a factor of about 10, compared with the first periodic inspection, as shown in Fig. V. 4-8.

(3) Quality Control of Thermal Shield Cooling Water

The thermal shield cooling water system is to cool the iron water thermal shield heated by the γ ray, neutron beam and heat conduction from the reactor. Its components and piping are made of carbon steel so that K_2Cr_4 (chromate) is added to prevent corrosion.

The following various tests were initially conducted on several types of anti-rust materials:

① Increases of radiation activity of the cooling water and dose equivalent rate of the components;

② Influence on the waste treatment system;

③ Formation of combustible gas (H_2);

④ Make-up due to decreased concentration of the anti-rust materials,

⑤ Anti-rust effects.

Based on these test results, chromate was selected as the best anti-rust material.

In the course of formation of the passive state films on the inner surface of the system, oxidation action of chromate will cause iron to elute to the cooling water, which will in turn reduce chromate to change to Cr_2O_3, chromate and others. They are absorbed to the passive state films so that chromate may sharply decrease and iron concentration may increase in the cooling water.

Since make-up of chromate and completion of the formation of the passive state films, hexavalent concentration has been stabilized. Concentration of ^{51}Cr (half life of about 28 days) formed by activation gradually increases as operation continues to reach the equilibrium condition at $\sim 7 \times 10^3$ Bq/cm^3.

(4) Quality Control of Spent Fuel Storage Pool Water

The demineralizer pre-coated with powder resin is used for the spent fuel storage pool water. When the iron exchange capacity decreases, a new resin is pre-coated to maintain the water quality. On the other hand, clean-up of the fuel exchange pool starts about a month before initiation of the refueling to recover transparency of the pool, and finishes simultaneously with the completion of the refueling.

(5) Quality Control of Auxiliary Cooling Water

(a) Quality Control of Reactor Auxiliary Cooling Water

Quality control of the auxiliary cooling water aims to maintain anti-rust effects on materials such as carbon steel and brass. After changes of several types of amti-rust materials, Rusmin MK9BT containing copper guard materials was selected. However, a problem of secondary rust protective capability caused by multiplication of bacteria in the cooling water remained to be solved. The bacteria problem was settled, but the rust protective effect is still being reviewed.

(b) Quality Control of Turbine Auxiliary Cooling Water

As the materials used are the same as those for the reactor auxiliary cooling water, Rusmin MK9BT was selected.

(6) Water Quality Control of Auxiliary Boiler

The auxiliary boiler is the facility to supply steam to be used for the hot laundry, the evaporative condenser of the waste treatment system, the gland seal of the steam turbine, heat-up of chemicals, heating of building and others. The oxygen scavenger (hydrazine: for corrosion protection of the water piping) and the anti-scales (phosphate: for protection of formation of hard scales due to concentration of minor impurities in the boiler drum) are added to the boiler water to control the water quality.

At the beginning of operation, unique water quality control targets were established without corresponding with the JIS standard; however, JIS B-8233 standard values were applied later. Following revision of JIS in June, 1989, target values of water quality control were changed and remained effective up to now. Changes of water quality targets are shown in Fig. V. 4-2.

(7) Chemical Control of Sea Water System

To monitor pollution conditions of the effluent, pH, extracted materials of n-hexane, COD and SS as inhibitors of the living environment are measured at intake and discharge ports once/2 months.

Ferrous sulfate as a corrosion inhibitor is added to the sea water system (after completion of the periodic inspection, it is added every day for about a month to accelerate formation of an initial iron coating film and then added once a week).

Chlorine generated in the sea water electrolyte unit is added to the intake port to prevent marine lives from depositing on the sea water system piping, the heat exchanger, the condenser and others. It should be noted that residual chlorine concentration in the discharge port is controlled to be within a range of 5～20 ppb not to give effects on the living environment of the marine lives.

(8) Purity Control of Demineralized Water

The water is controlled to maintain a high purity, as it is used for the feed water to the reactor facility, the turbine facility and auxiliary boiler.

(9) Quality Control of Diesel Generator Primary Cooling Water

After application of several types of anti-rust materials, Krilex L-111 has been adopted.

(10)Quality Control of Ventilation System Cooling Water

The cooling tower is open to air so that it is easy to foul due to chlorine, sandy dust and others entering from the atmosphere. In addition, contents dissolved in the cooling water are condensed in the cooling tower, causing various faults.

As anti-rust material, Tower Clean S-512 is used in a combination with the concentration control by the intermittent blow to ensure good water quality.

4.4 Exchange Experiences of Ion Exchange Resin

(1) Exchange Experiences of Reactor Clean-up Water System Resins

There are two mixed bed demineralizers which consist of the strong acid cation exchange resin (SKN-1: 1.35 m^3/2 columns) and the strong basic anion exchange resin (SAN-1: 1.35 m^3/2 columns).

An exchange frequency of the ion exchange resins is determined to be once in every two years based on the leakage of silica. Table. V. 4-3 shows experiences of the resin exchange.

(2) Exchange Experiences of Condensate Demineralizer Resins

There are four mixed bed demineralizers which consist of the cation exchange resin Amberlite IR-120B of 2,200 liter and the anion exchange resin IRA-400T of 1,000 liter. With considering the stabilization of water quality of the whole feed water system and economy, resin exchange is performed on for one column a year. Table. V. 4-4 shows experiences of the resin exchange.

(3) Exchange Experiences of Fuel Pool Purification System

Two filter demineralizers with the solid cylindrical pre-coat type filter are installed. They consist of the powder type styrene series strong acid cation ion exchange resin (Diaion DFK-10: pre-coating of 5 kg dry-resin/application) and the styrene series strong basic anion exchange resin (Diaion DFA-10: pre-coating of 3 kg dry-resin/application), coupled with the swelling material (Krisoap: pre-coating of 2 kg/application) to accelerate volume filtration.

The spent fuel storage pool water operates in a clean-up mode all the time during operation to prevent corrosion products and crud from accumulating, while the

fuel exchange pool water is cleaned up in every usage.

The resins are to be exchanged in the cases of:

① Decrease of DF of ^{60}Co to 3

② Electric conductivity of the demineralizer outlet $\geqq$ 1.5 μ S/cm

③ Differential pressure of the demineralizer resin strainer > 0.38 kg/cm^2

④ Differential pressure of the filter demineralizer > 1.5 kg/cm^2

⑤ Total radioactivity concentration > 4 Bq/cm^3

Table. V. 4-5 shows experiences of the resin exchange.

(4) Exchange Experiences of Raw Water Demineralizer Resins

The unit consists of the cation exchange resin bed, the anion exchange resin bed and polisher column, installed in two lines.

① Cation exchange resin bed: Diaion SK-1B 400 l/bed

② Anion exchange resin bed: Diaion SA-20 600 l/bed

③ Polisher column: Diaion SK-1B 250 l/column and Diaion SA-10A 250 l/column

These resins are entirely or partly renewed during the periodic inspections. Exchange experiences of the purifying water unit resins are shown in Table V. 4-6.

4.5 Chemistry Control of Hydrogen/ Oxygen Generator

This unit uses a bipolar filter press type alkali electrolysis method which generates H_2 gas to inject to the feed water and O_2 gas to recombine H_2 gas extracted from the condenser to the gaseous waste treatment system.

Purity of the generated O_2 and H_2 gases is monitored twice a month for every electrolysis tank and for the injection rack which supplies gas. The control target of the gas purity and actual results of purity control of hydrogen gas are shown in Table V. 4-7 and Fig. V. 4-10 , respectively.

4.6 Management of Charcoal Filter

To prevent radioactive materials from scattering during the loss of coolant accident which will cause radiation exposure of personnel and local residents, tray type charcoal filters are installed in the containment air recirculation system, the containment annulus air exhaust system and the control room ventilating air system.

During the periodic inspection, sample charcoal filters installed in the tray type charcoal filter unit of every system are taken out to perform iodine removal efficiency tests using radioactive inorganic iodine and radioactive organic iodine so that the single charcoal (Tsurumicoal HCA-S) can be checked for degradation. In

addition, Freon (R-112) is injected to upstream of the filter and leak rate of Freon which leaks to downstream is calculated in order to verify normal installation conditions of filter and no damages or faults.

Based on these test results, the comprehensive efficiency of the charcoal filter is calculated to ensure the integrity of its performance. Table V. 4-8 shows the criteria on the iodine removal efficiency of the charcoal filter.

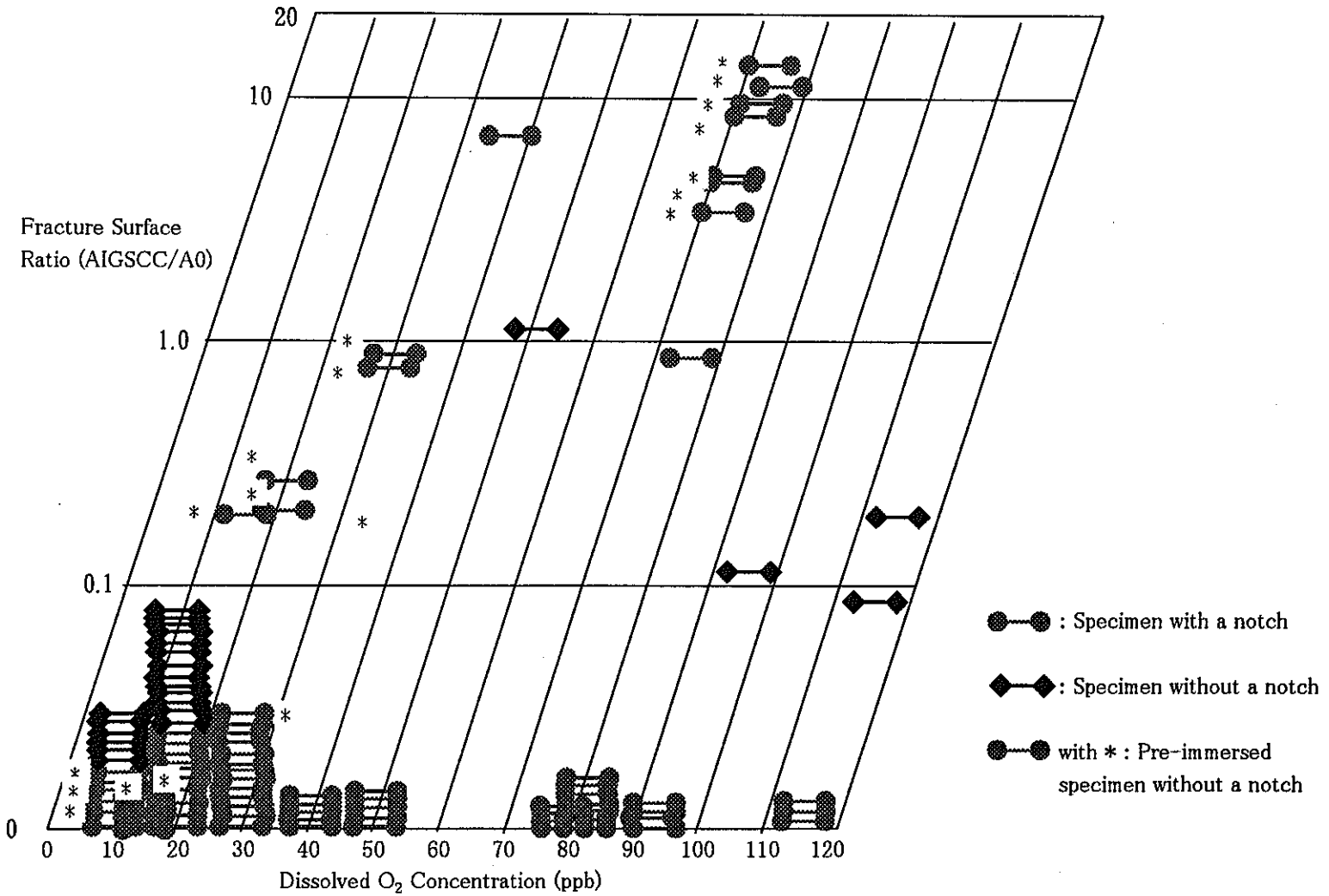


Fig.V.4-1 Relations between Dissolved Oxygen Concentration and Fracture Surface Ratio by SSRT

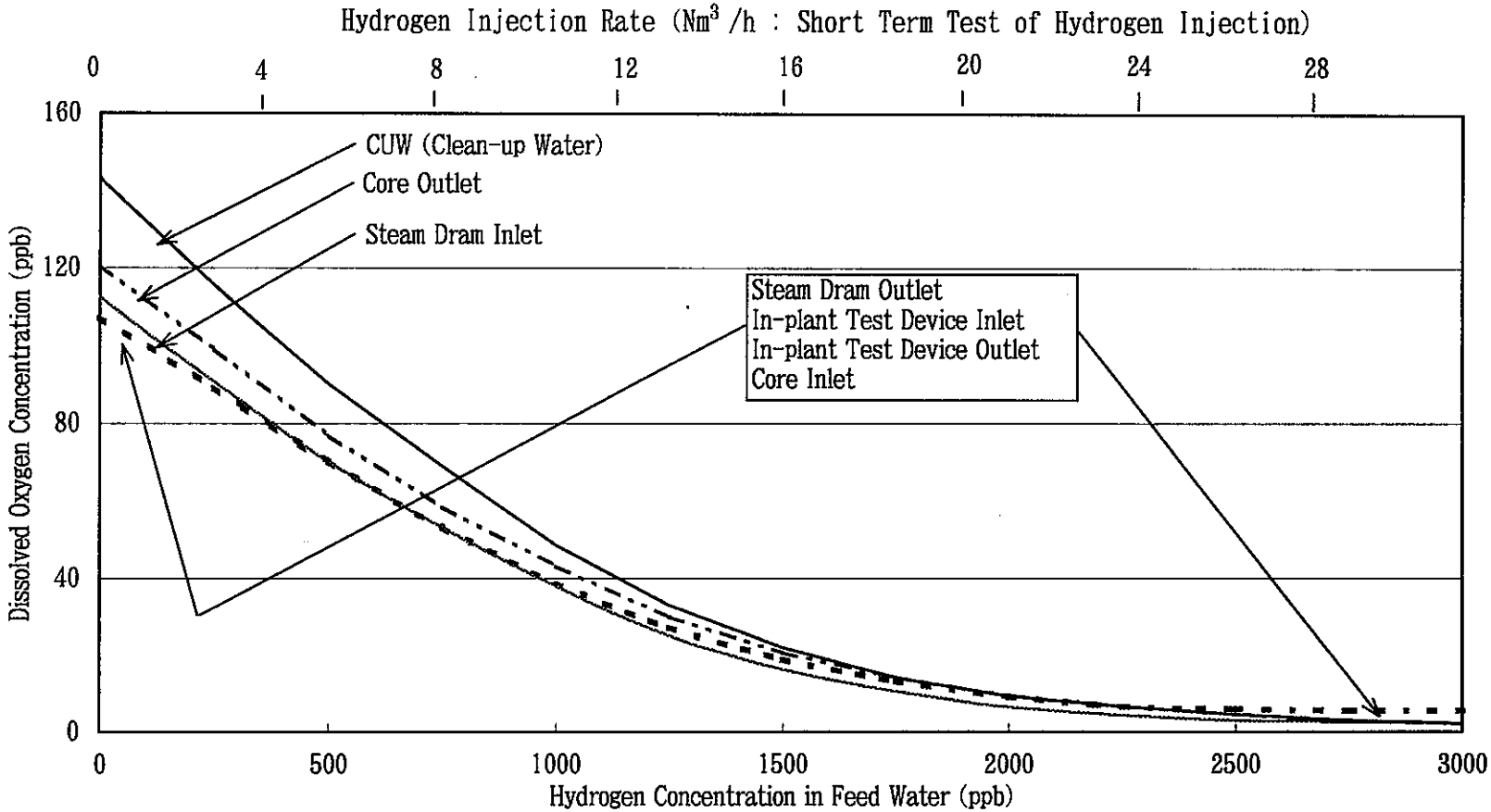


Fig.V.4-2 Analysis of Dissolved Oxygen Concentration of Reactor Coolant System with Hydrogen Injection Rate

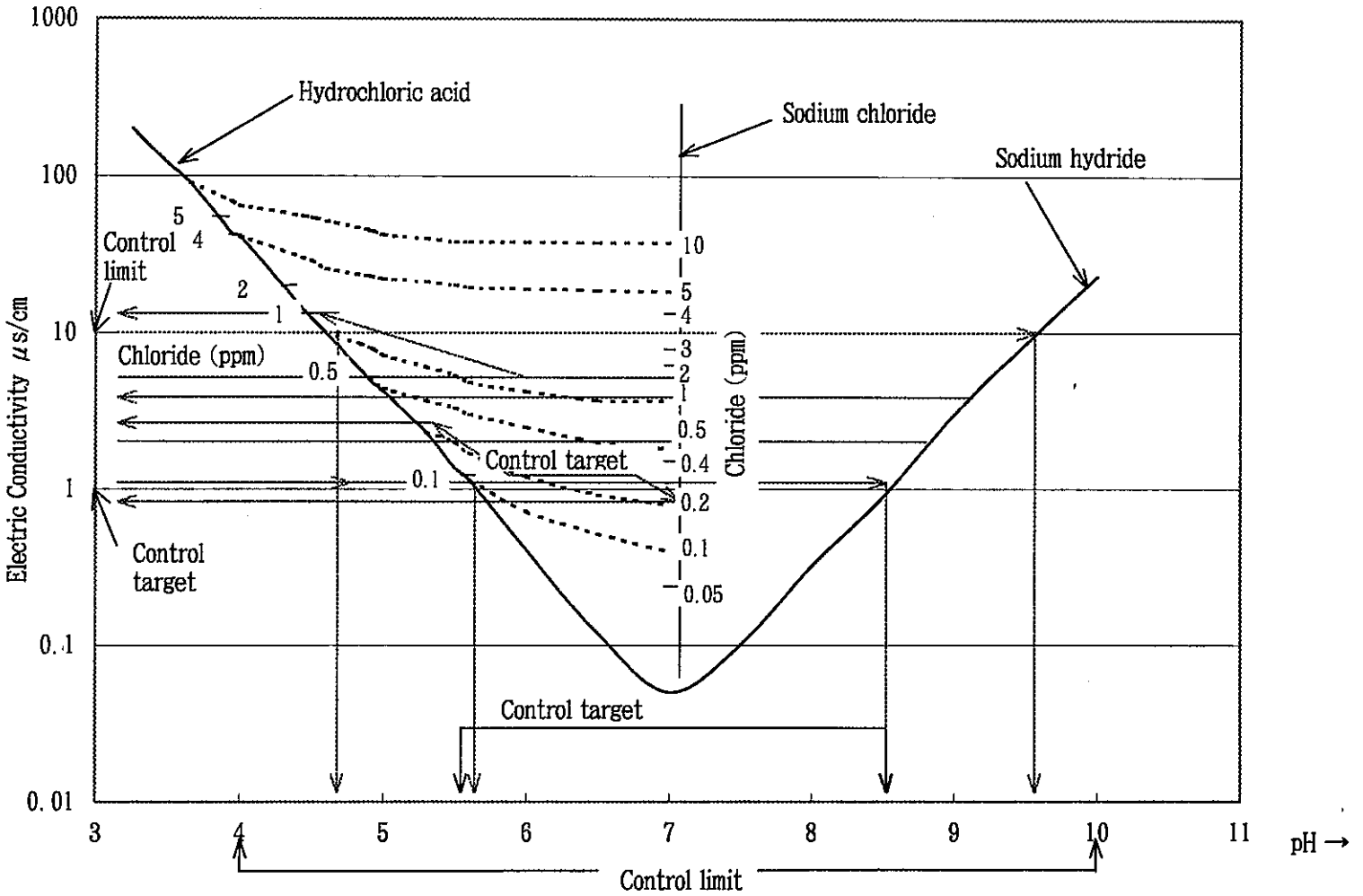


Fig.V.4-3 Impurity Concentration, Electric Conductivity snd pH in Demineralized Water

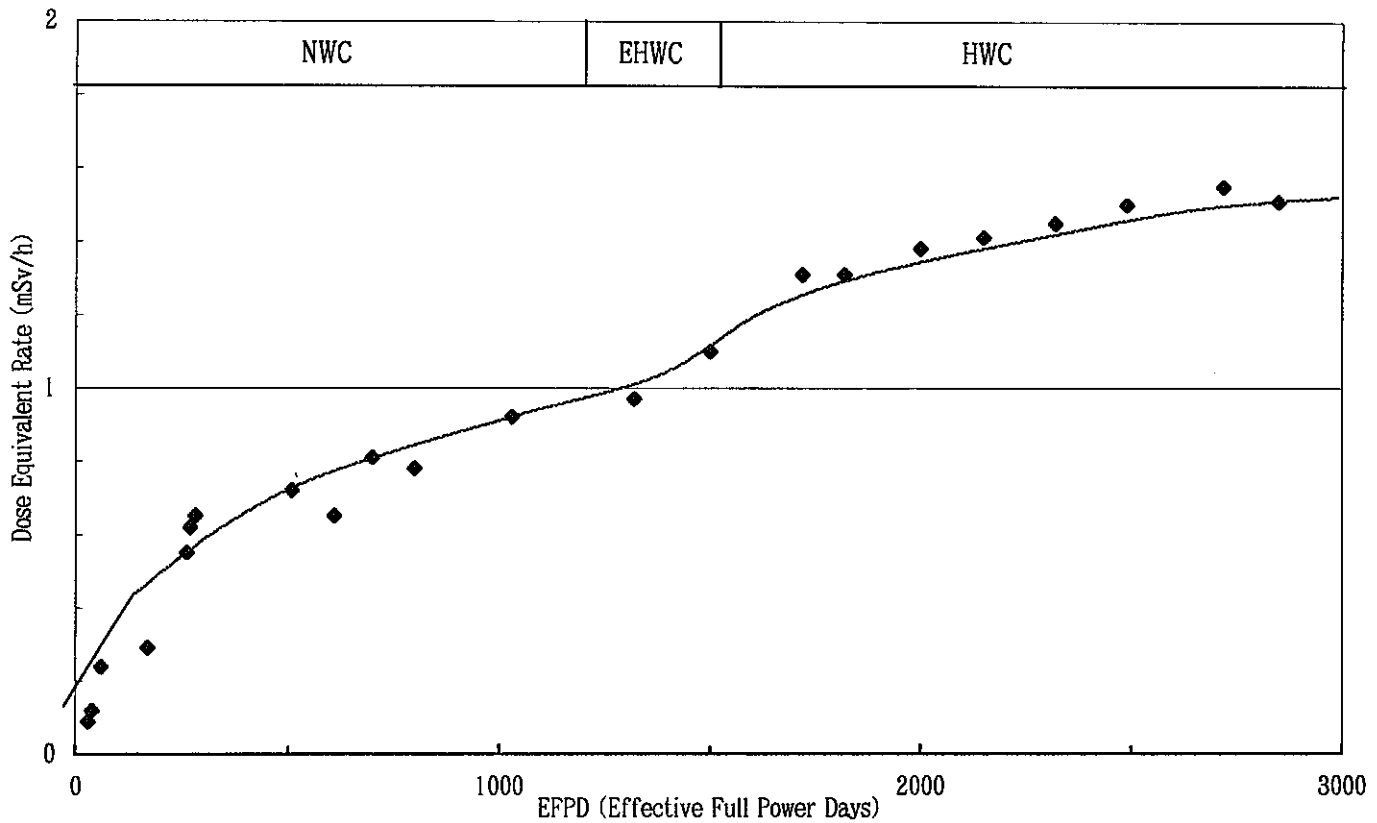


Fig.V.4-4 Change of Dose Equivalent Rate at RCP Outlet Pipe

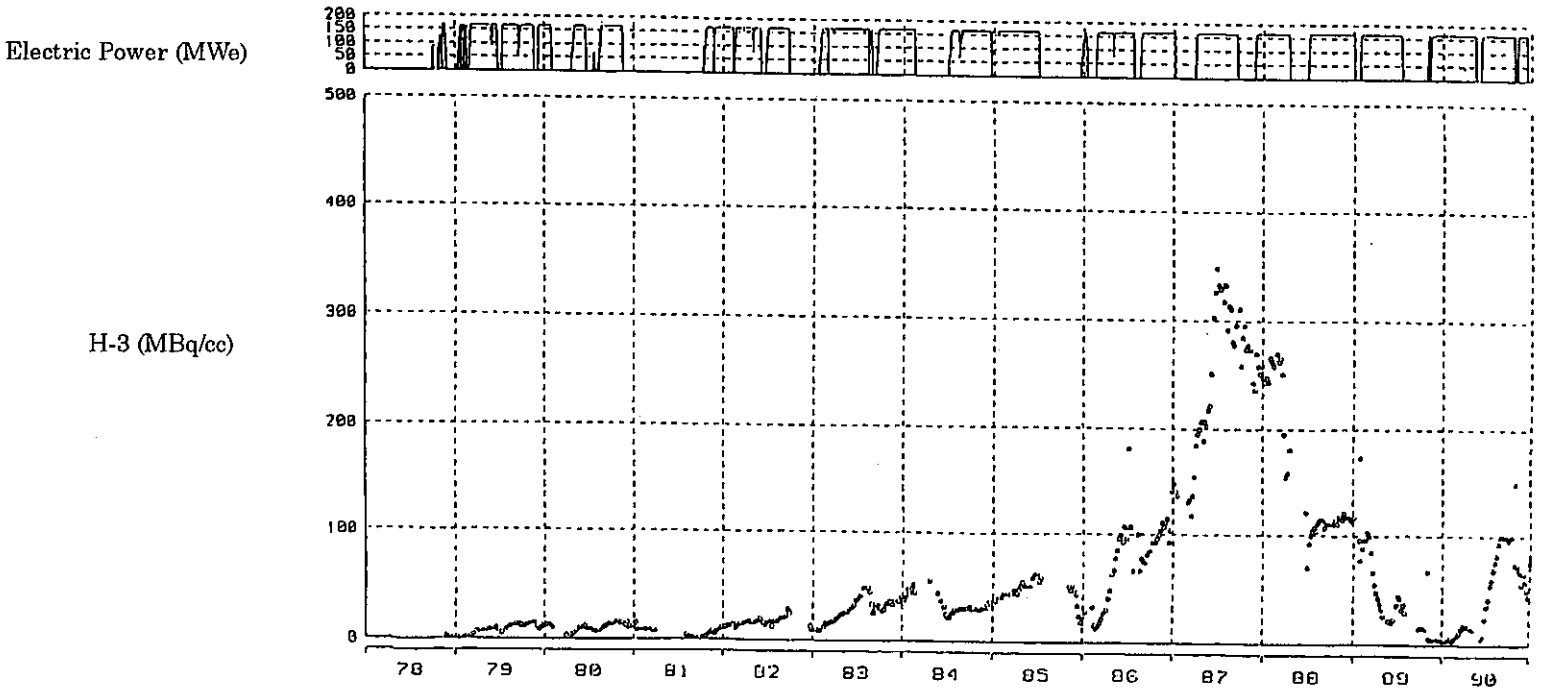


Fig.V.4-5 Change of Tritium Concentration in Reactor Cooling Water

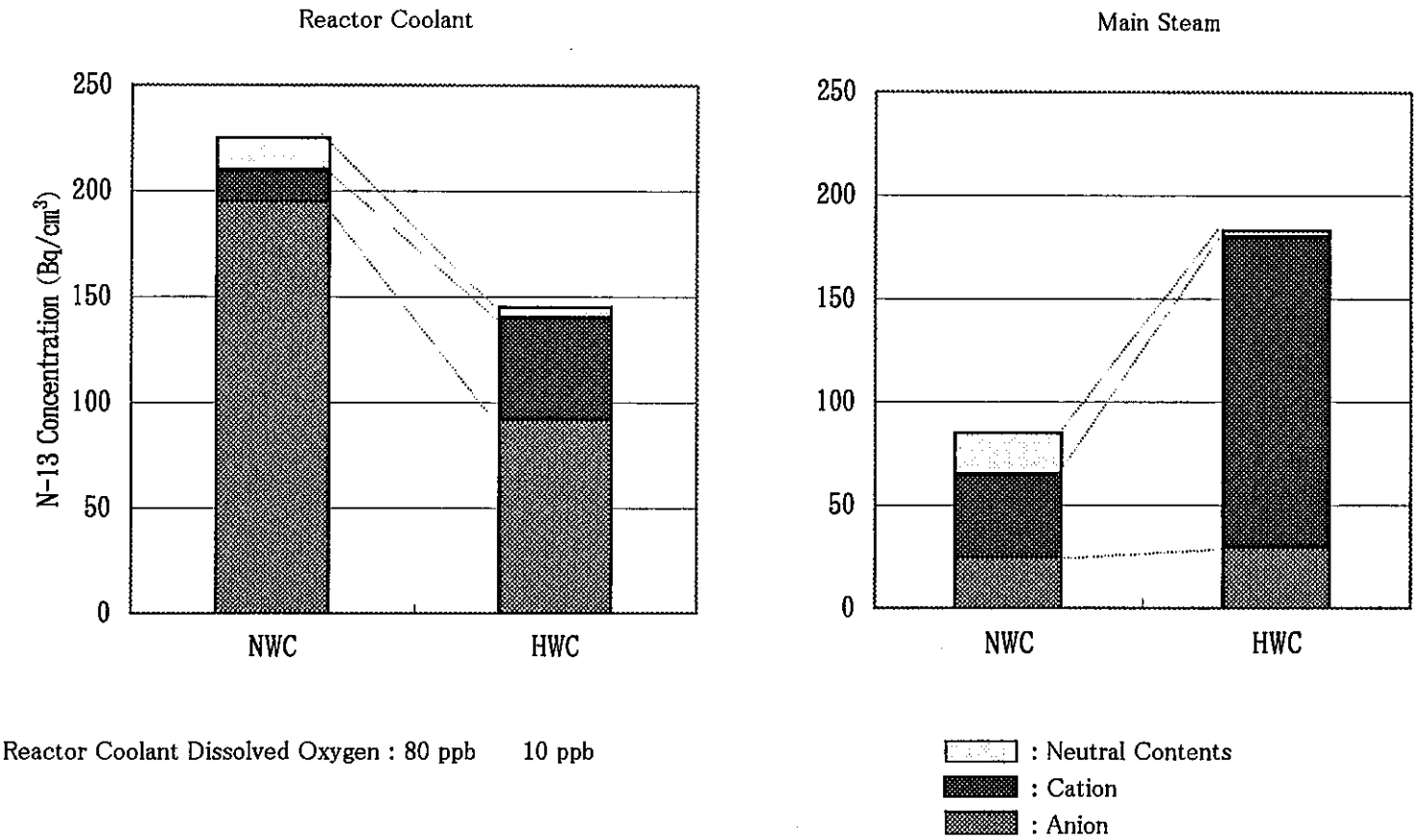


Fig.V.4-6 Change of Chemical Form of N-13 in Reactor Coolant and Main Steam before and after Hydrogen Injection (at 20% power)

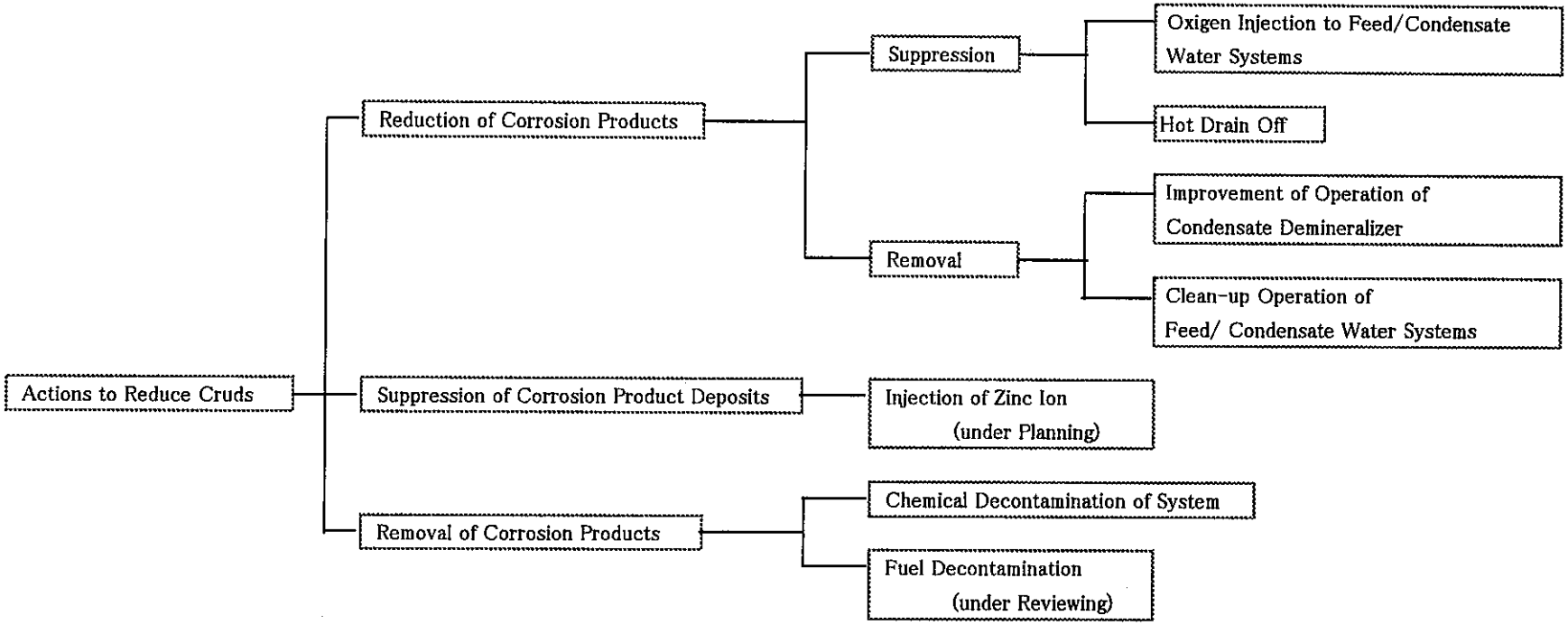


Fig.V.4-7 Measures to Reduce Iron Intake

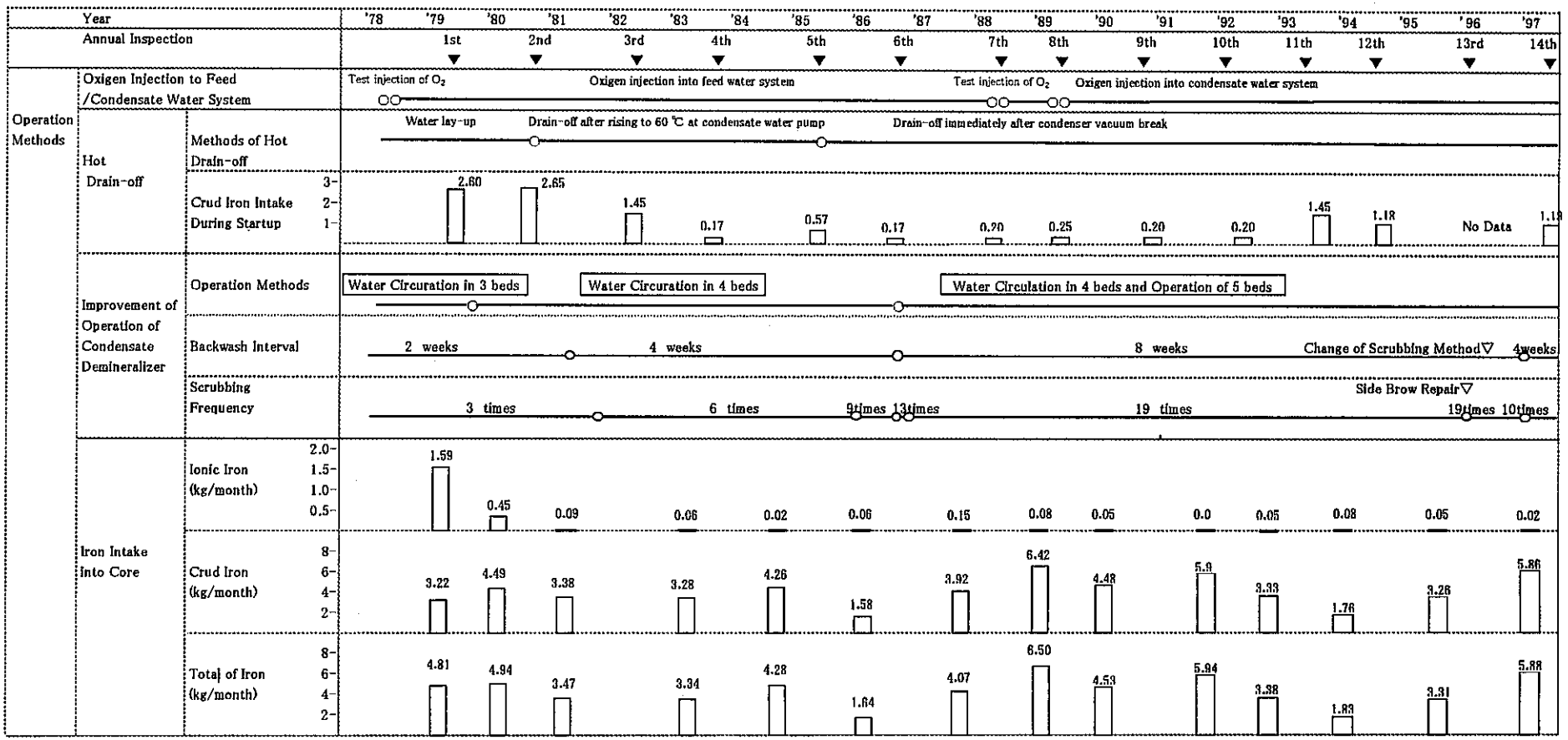


Fig.V.4-8 Measures to Reduce Iron Intake and Actual Results of Iron Intake

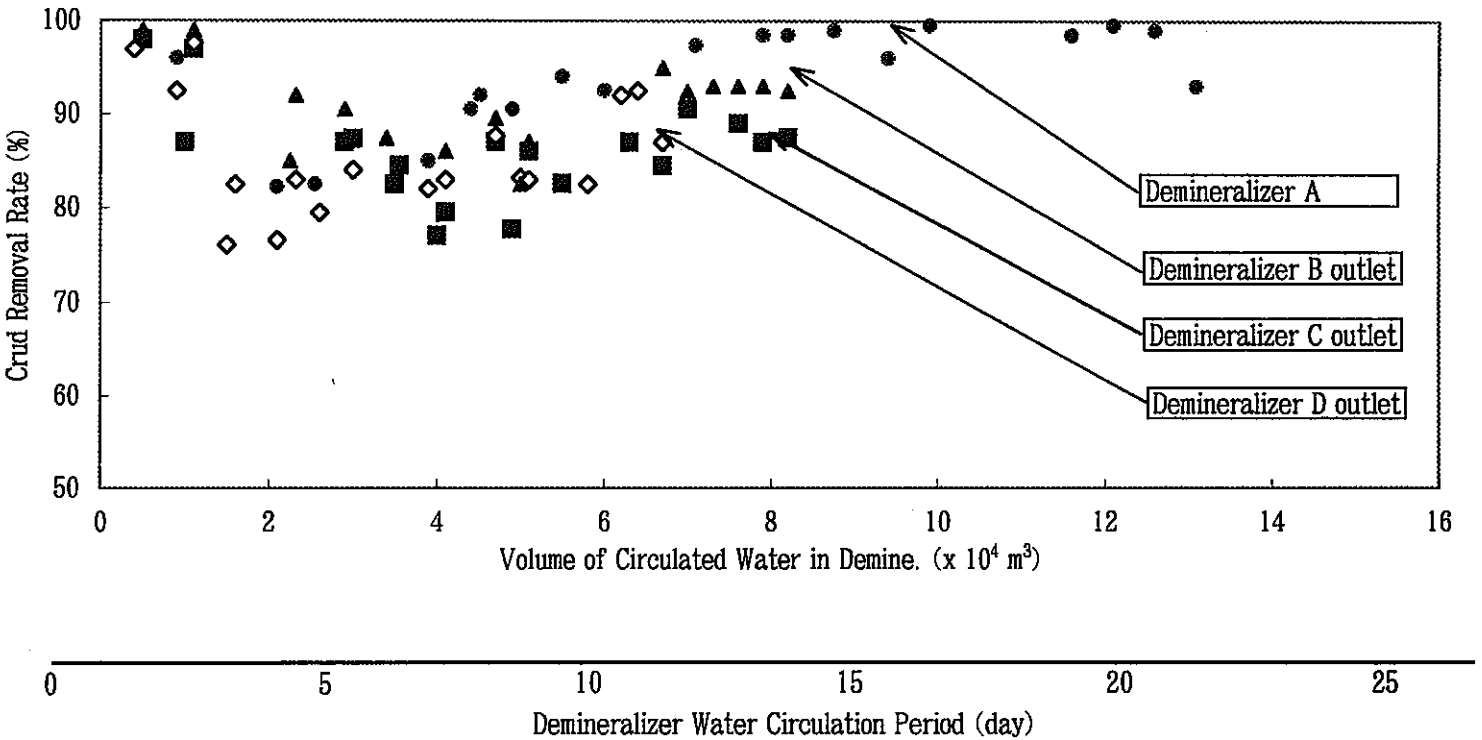


Fig.V.4-9 Change of Crud Removal Rate According to Water Circulation in Condensate Demineralizer

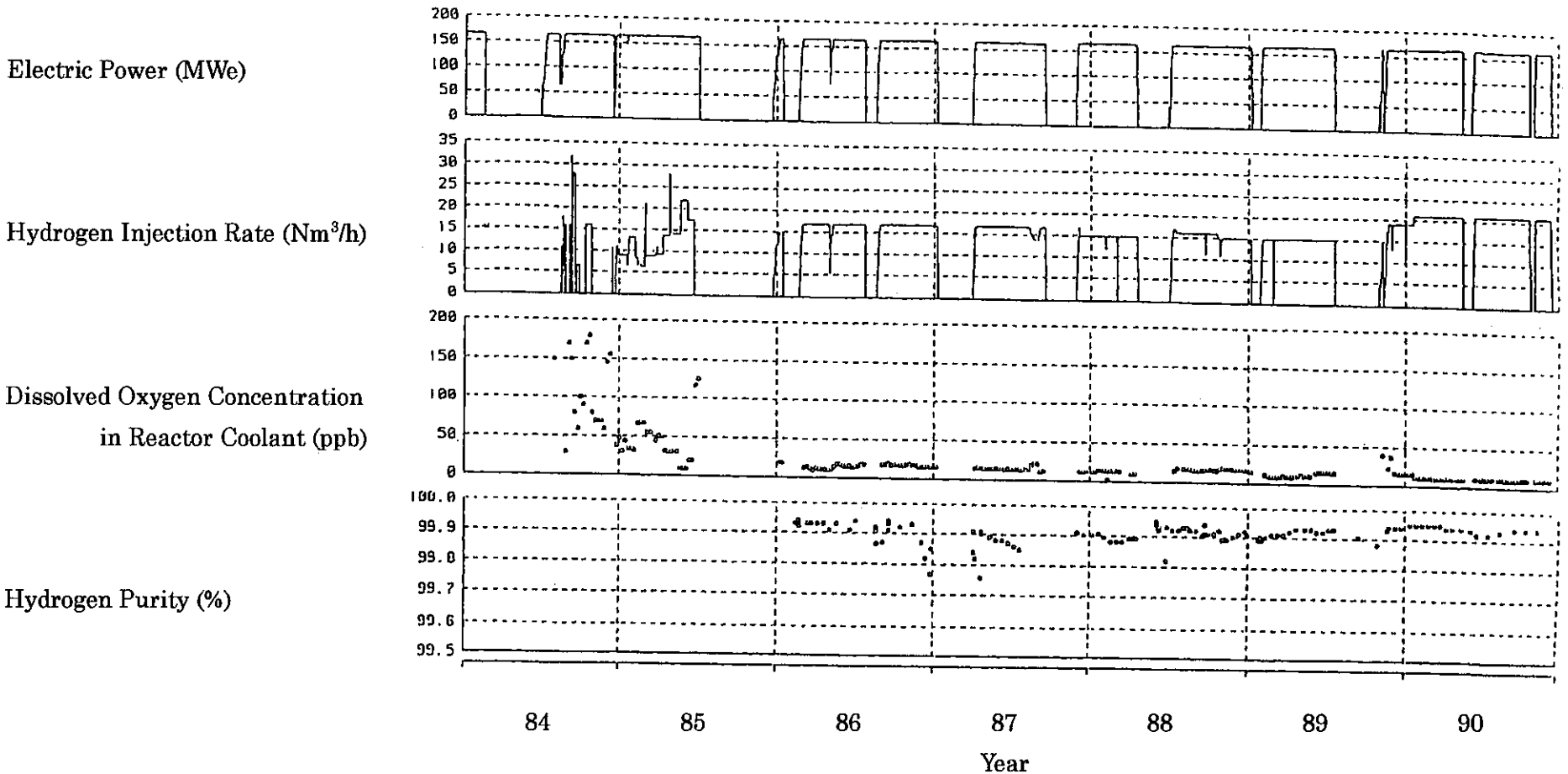


Fig.V.4-10 Results of Hydrogen Gas Purity Control of Hydrogen/Oxygen Generator

Table V.4-1 Control Criteria of Light Water System, their Rationales and Measuring Methods

System/sampling pt.		Item	Limit	Control Target	Rationales	Measuring Method	Frequency
Inlet of RCW Demine.		pH	4～6	5.5～8.5	Calculated based on lower limit of chlorinate	To measure sample with flow cell type pH meter by circulating water	Once /week
		Electric conductivity	≦10 μS/cm	≦1.0 μS/cm	concentration to generate SCC	To measure sample with flow cell type μS/cm meter by circulating water	
		Chlorine	1 ppm	0.2 ppm		To measure sample with ion chromatography	
		Silica		≦2 ppm	Calculated from precipitation limit of silica at turbine outlet	To measure sample with absorptiometer (molybdate blue meted)	
		Insoluble impurities		≦1 ppm	Calculated from condensation effects of insoluble impurity conc. in feed water	To percolate with 0.4 μm filter and calculate conc. from weight of entrapped materials	
		Dissolved oxygen conc.		≦0.03 ppm	Lower dissolved oxygen conc. limit in feed water to generate SCC	To measure sample by circulating water using in-line DO meter	
		Boron		≦2 ppm	Corresponding to BWR value	To measure sample with absorptiometer (methyleneblue absorptiometer)	
		γ material	I^{131} 2.9x10^{3} Bq/g	I^{131} 0.4 Bq/g	Reactor water conc. used in evaluation of main steam pipe rupture acc.	To measure and quantify using Ge(Li) semiconductor detector	Twice/week
Outlet of A. B Demine		Electric conductivity		≦0.1 μS/cm	Criteria on demineralizer break-through	To measure sample with flow cell type μS/cm meter by circulating water	Once /week
		Electric conductivity		≦0.2 μS/cm	Monitoring of sea water leak	To measure sample with flow cell type μS/cm meter by circulating water	
		Dissolved oxygen conc.		≦0.06 ppm	Criteria on deaeration performance of condenser	To measure sample by circulating water using in-line DO meter	
Inlet of Cond. Demine		Electric conductivity		≦0.2 μS/cm	Criteria on demineralizer break-through	To measure sample with flow cell type μS/cm meter by circulating water	
Outlet of Cond. Demine		Dissolved oxygen conc.		0.02～0.06ppm	Iron corrosion rate (upper limit : pitting, lower : generating O_2 film)	To measure sample by circulating water using in-line DO meter	
Shield Cooling Water		Chlorine		≦1 ppm	Corresponding to chlorine ion conc. In heavy water	To measure sample with ion chromatography	
		Hexavalent chromium		≧300 ppm	Conc. Necessary to prevent corrosion of carbon steel	To color sample with reagent to measure absorptiometer (biphenyl curbside method)	
Spent Fuel Pool Water		Electric conductivity		≦1.5 μS/cm	Established to prevent corrosion and ensure transparency	To measure sample with flow cell type μS/cm meter by circulating water	
Fuel Exchange Pool Water		γ material		≦4 Bq/g	According to requirement for contamination control	To measure and quantify using Ge(Li) semiconductor detector	
Reactor Auxiliary Cooling Water		pH		7～10	Applied anti-corrosion criteria of general cooling water	To measure sample with laboratory pH meter	
		Electric conductivity		≦1000 μS/cm	Corresponding to appropriate Rusmin conc.	To measure sample with laboratory μS/cm meter	
		Chlorine		≦0.1 ppm	Prevention of SCC in SUS-304	To measure sample with ion chromatography	
Turbine Auxiliary Cooling Water		Nitrous acid ion		≧50 ppm	Corresponding to Rusmin conc.	To measure sample with ion chromatography	
		Nitric acid ion		≦200 ppm	Anti-corrosion conc. determined by test(check for generation of bacteria)	To measure sample with ion chromatography	
		γ material		Not detectable	Ensuring no tube leak	To measure and quantify using Ge(Li) semiconductor detector	
		Rusmin conc.		≧800 ppm	Effective anti-corrosion conc. Determined by test	To measure sample with absorptiometer	
		Total carbonic acid		≦250 ppm	Anti-corrosion conc. Determined by test(check for generation of bacteria)	To measure sample with strontium chloride-hydrochloric acid method	
DEG Primary Cooling Water		pH		7～10	Anti-corrosion criteria of general cooling water	To measure sample with laboratory pH meter	Once/month
		Nitrous acid ion		380～460ppm	Established by corrosion prevention test of rilex L-111	To measure sample with ion chromatography	
		Chlorine		≦50 ppm	Established by corrosion test	To measure sample with ion chromatography	
		Nitric acid ion				To measure sample with ion chromatography	
Bldg. Ventilating Syst.		Tower clean S-512		150～300ppm	Established by corrosion test of Fugen system	To reduce and color sample by ascorbic acid, measure with absorptiometer(molybdate blue)	Once /week
		Silica		≦130 ppm	Established by silica recitation test of FUGEN system	To measure sample with absorptiometer(molybdate yellow)	
Water Make-up System		pH		5.5～7.5	Review with considering actual results of raw water quality	To measure sample with laboratory pH meter	Once/week for demine. water tank whenever sampling for polisher
		Electric conductivity		≦1.5 μS/cm	based on water make-up system design value	To measure sample with laboratory μS/cm meter	
		Chlorine		≦0.06 ppm		To measure sample with ion chromatography	
		Silica		≦0.03 ppm		To measure sample with absorptiometer(molybdate blue method)	
		Insoluble impurities		≦1.5 ppm		To percolate with 0.4 μm filter and calculate conc. from weight of entrapped materials	
		Iron		≦0.2 ppm		To measure sample with atomic absorption frame method	
Auxiliary Boiler System	Boiler Water	pH		9.8～10.8	Corresponding to JIS B-8223	To measure sample with laboratory pH meter	Twice/month
		Electric conductivity		≦1200 μS/cm		To measure sample with laboratory μS/cm meter	
		Chlorine		≦150 ppm		To measure sample with ion chromatography	
		Phosphoric acid ion		10～30ppm		To measure sample with absorptiometer(molybdate blue method)	
		Silica		≦50 ppm		To measure sample with absorptiometer(molybdate yellow method)	
		Hydrazine		≧0.1 ppm		To measure with p-dmethylamino-enzaldehyde method according to conc. of sample	
		Acid consumption (pH 4.8)		≦130 ppm		To add mixed solution of methyl red-bromcresolgrin as indicator to sample and measure with caustic soda	
		Acid consumption (pH 8.3)		≦100 ppm		To add phenolphthalein solution as indicator to sample and measure with nitric acid	
		γ material		Not detectable		To measure and quantify using Ge(Li) semiconductor detector	
		^{3}H		Not detectable		To measure with liquid scintillation counter	
	Feed Water	pH		8.8～9.5		To measure sample with laboratory pH meter	
		Dissolved oxygen conc.		≦0.5 ppm		To measure with simplified DO meter by local sampling rack	
		Iron		≦0.1 ppm		To measure sample with atomic absorption frame method	
Steam Water Syst.		Residual chlorine		≦0.02 ppm	Based on criteria on water quality for fishery environment	To measure with absorptiometer by adding O-tolidine	Once/week
Drainage(General Drain, Discharge Outlet)		pH	5.8～8.6	5.8～8.6	Established by ordinance of FUKUI Pref.	To measure sample with laboratory pH meter	Once/month
		Insoluble impurities	≦120 ppm	≦120 ppm		To percolate with 0.4 μm filter and calculate conc. from weight of entrapped materials	
		COD	≦100 ppm	≦100 ppm		To make sample anaerbic nitric acid before adding potassium permanganate solution as oxideant, and calculate conc. from amount of permaganate consumed after reaction in boiling water for 30 min.(OODM method)	
		n-hexane extractred	≦5.0 ppm	≦5.0 ppm		To measure weight of residues after extraction, dehydration and vaporization of sample by n-hexane to calculate conc. of extracted material	

Table V.4-2 Changes of Target for Water Quality Control of Feed Water and Boiler Water of Auxiliary Boiler

			Start-up	'78/11/22	'79/11/9	'81/7/14	'81/11/6	'84/9/17	'86/7/4	'90/4/1
Feed Water	pH	(25℃)	---	---	---	---	---	8.0～9.5	8.0～9.5	8.0～9.5
	Hardness	($CaCO_3$mg/l)	---	---	---	---	---	0	0	---
	Fat and Oil	(mg/l)	---	---	---	---	---	0	0	---
	Dissolved oxigen	(mg/l)	---	---	---	---	---	≦0.03	≦0.03	≦0.5
	Iron	(mg/l)	---	---	---	---	---	≦0.1	≦0.1	≦0.1
	Copper	(mg/l)	---	---	---	---	---	≦0.05	≦0.05	---
	Hydrazine	(mg/l)	---	---	---	---	---	≧0.06	≧0.06	---
Boiler Water	pH	(25℃)	10.0～11.0	10.0～11.0	10.0～11.0	10.0～11.0	10.0～11.0	9.4～11.0	9.4～11.0	9.8～10.8
	Electric conductivity	(μS/cm)	---	---	---	---	---	≦800	≦800	≦1200
	M alkalinity	($CaCO_3$mg/l)	---	---	---	---	---			≦130
	P alkalinity	($CaCO_3$mg/l)	---	---	---	---	---			≦100
	Total evaporation residues	(mg/l)	---	---	---	---	---			
	Chlorine ion	(mg/l)	---	---	---	---	---	≦80	≦80	≦150
	Phosphoric acid ion	(mg/l)	20～40	30～90	80～140	40～80	---	5～40	5～15	10～30
	Hydrazine	(mg/l)	20	10～30	10～30	10～30	10～100	---	---	0.1 <
	Insoluble impurities	(mg/l)	≦50	≦50	≦50	≦50	≦50	≦20	≦20	---
	Silica	(mg/l)	≦50	≦50	≦50	≦50	≦50	≦50	≦50	≦50
Conc. of Chemical Tank	Dilute hydrazine tank	(%)		5	1.9	0.96	0.96	0.96	0.64	0.64
	Cndensed hydrzine tank	(%)		12.5	9.6	9.6	9.6	9.6	9.6	9.6
	No 3 sodium phosphate tank	(%)	4	10	10	10	10	10	10	10
	Chemical adjustment tank	(%)						Hidorazine 12.3 phosphoric acid 0.03	Hidorazine 12.3 phosphoric acid 0.04	
Reason to Change Control Target				Values of pH cannot satisfy control target	Values of pH cannot satisfy control target	Values of pH cannot satisfy control target	Residual hydrazine cannot follow load changes to satisfy the control target	The MITI directed to get to near the value of JIS standard at annual inspection on auxiliary boiler	The MITI directed to get the phosphoric acid conc. near to the JIS standard at annual inspection on boiler (1985)	Changed following the revision of JIS B-8223 in 1989
Remarks								Change of injection point of dilute hydrazine to FW pump inlet	Automatic blow system was installed and in full operation in Oct./85	To follow the JIS B-8223, 210μS/cm of electric conductirity was set up for automatic blow contorol

Table V.4-3 Experiences of Exchange of Reactor Clean-up Water System Resins

	A Loop		B Loop		Remarks
	Cation resin	Anion resin	Cation resin	Anion resin	
'75	---	---	---	---	
'76	---	---	---	---	
'77	900	450	900	450	First charging
'78	---	---	---	---	
'79	900	450	900	450	
'80	900	450	900	450	
'81	---	---	---	---	
'82	900	450	900	450	
'83	---	---	---	---	
'84	675	675	675	675	Change the ratio of cation/anion exchange resin from 2:1 to 1:1
'85	---	---	---	---	
'86	675	675	675	675	
'87	---	---	---	---	
'88	675	675	675	675	
'89	2535	3015	1915	2235	Implementation of chemical decontamination on A loop

Table V.4-4 Experiences of Exchange of Condensate Demineralizer Resins

(unit:liter)

	No. 1		No. 2		No. 3		No. 4		No. 5	
	Cation resin	Anion resin	Cation resin	Anion resin	Cation resin	Anion resin	Cation resin	Anion resin	Cation resin	Anion resin
'75	---	---	---	---	---	---	---	---	---	---
'76	---	---	---	---	---	---	---	---	---	---
'77	---	---	2200	1100	2200	1100	2200	1100	2200	1100
'78	---	---	---	---	---	---	---	---	---	---
'79	---	---	---	---	---	---	---	---	---	---
'80	---	---	---	---	---	---	---	---	---	---
'81	---	---	2200	1100	---	---	---	---	---	---
'82	---	---	---	---	---	---	---	---	2200	1100
'83	---	---	---	---	---	---	2200	1100	---	---
'84	---	---	---	---	2200	1100	---	---	---	---
'85	---	---	2200	1100	---	---	---	---	---	---
'86	2200	1100	---	---	---	---	---	---	---	---
'87	---	---	---	---	---	---	---	---	---	---
'88	---	---	---	---	---	---	---	---	2200	1100
'89	---	---	---	---	---	---	2200	1100	---	---
'90	---	---	---	---	---	---	---	---	---	---
'91	---	---	---	---	2200	1100	---	---	---	---
'92	---	---	2200 2200	1100 1100	---	---	---	---	---	---
'93	2200 2200	1100 1100	---	---	---	---	---	---	---	---
'94	---	---	---	---	---	---	---	---	---	---
'95	---	---	2200	1100	---	---	---	---	2200	1100
'96	---	---	---	---	---	---	2200	1100	---	---
'97	---	---	---	---	2200	1100	---	---	---	---

Table V.4-5 Experiences of Exchange of Fuel Pool Purification System Resins

'75	'76	'77	'78	'79	'80	'81	'82	'83	'84	'85	'86	'87	'88	'89
			330	1380	900	750	450	180	180	1800	840	270	750	750

Table V.4-6 Experiences of Exchange of Water Make-up System Resins

Input of new resin is shown

	A Line				B Line			
	H Bed	OH Bed	Po bed		H Bed	OH Bed	Po bed	
	Cation resin	Anion resin	Cation resin	Anion resin	Cation resin	Anion resin	Cation resin	Anion resin
'75	10	25			5	25		
'76								
'77		600	250	250		600	250	250
'78								
'79	50			50				50
'80	20	150	5	100	10	50		100
'81	50	100		40	50	120		10
'82	8	20	3	10	15	5	10	10
'83	50	300	75	400	50	612	100	400
'84		50						
'85		600	25	100		240	25	100
'86	150	20	25	25	270	20	25	25
'87								
'88	400	600	250	250		600	250	250
'89	100	100	100	100	100	100	100	100

Table V.4-7 Control Target of Gas Purity of Hydrogen/Oxygen Generator

Hydrogen/ Oxygen Generator (at Outlet of A and B Separators)	O_2 concentration in H_2 N_2 concentration in H_2 H_2 concentration in O_2	≦ 0.15 % --- ≦ 1.7 %
Hydrogen/ Oxygen Generator (at Injection Rack Outlet)	O_2 concentration in H_2 N_2 concentration in H_2 H_2 concentration in O_2	≦ 0.1 % --- ≦ 1.5 %

Table V.4-8 Control Criteria on Charcoal Filter

	Unit Test of Activated Charcoal Filter		Removal Efficiency of Iodine in System	Remarks
	Inorganic Iodine	Organic Iodine		
Annulus Air Exhaust System	≧ 99 %	≧ 97 %	≧ 95 %	Specified value in safety regulations
Containment Air Recirculation System	≧ 99 %	≧ 97 %	≧ 95 %	Specified value in safety regulations
Main Control Room Ventilation System	≧ 99 %	≧ 97 %	≧ 95 %	

5. Experiences and Evaluation of Chemistry Control of Heavy Water and Helium Systems

5.1 General

The moderator system in "Fugen" consists of the heavy water system and the helium system. To absorb excess reactivity, boron (^{10}B) is injected to the heavy water, but chemicals for adjustment of the water quality are not added specially.

The heavy water system is the closed cycle; its purification is conducted by the mixed bed demineralizer using the weak basic anion exchange resin and the strong acid cation exchange resin.

With fuel burning, the liquid poison removal resin charged by the strong basic anion exchange resin reduces ^{10}B concentration, which is 5~9 ppm at the beginning of fuel cycle to decrease to nearly zero at the end of cycle.

Part of the heavy water is decomposed to deuterium and oxygen due to radiolysis in the core, transferred to the cover gas (helium) side, introduced to the recombiner with helium gas and recovered again as heavy water. As deuterium is combustible, its concentration in helium is measured all the time to maintain below the inflamability limit. Impurities such as nitrogen in helium are cleaned up by the activated charcoal absorption device, if necessary.

At the beginning of operation, we faced various challenges to be solved in the field of chemistry control, which were resulted from the electric conductivity of heavy water higher than predicted. This increase in the electric conductivity is mainly caused by deuterium dioxide generated when heavy water received γ ray and neutron beam in the core, oxidizing and decomposing the weak basic anion exchange resin used for purification of heavy water, the resultant organic materials were further subject to radiolysis in the reactor to form nitric acid ions and others with high electric conductivity. It was also identified that chlorine contained in the ion exchange resins or nitrogen in helium of the cover gas for heavy water accelerated the increase of electric conductivity of heavy water.

To decrease the electric conductivity of heavy water, the following measures were taken so that the water quality could be improved:

① Development of a catalyzer to decompose deuterium dioxide

② Review of application of the strong basic anion exchange resin to the heavy water purification system

③ Cooling down of temperature of circulating heavy water in the heavy water purification resin beds

④ Development and adoption of the extremely low chlorinated resins

⑤ Enhancement of helium purity control

5.2 Measures for Chemistry Control of Heavy Water and Helium [V-5] [V-10]

Heavy water should maintain the "purity", or isotopic purity not to deteriorate the neutron economy as the neutron moderator, while keep the "water quality", or chemical purity within a certain range to secure integrity of the core structural materials.

Considerations in terms of chemistry control should be also given to the concentration of radioactive corrosion products such as Co-60, which must remain low to reduce total dose equivalent of workers during periodic inspections.

(1) Control Targets of Heavy Water Quality and Their Basis

Table V. 5-1 shows target values of heavy water quality control.

As refueling was determined to be conducted during reactor shutdown, B-10 was selected to compensate the excess reactivity due to burning. Therefore, in establishing the Operation Control Guideline, the reference value of the electric conductivity of boric acid was determined to be 1.5 μ S/cm and the reference value of chlorine ion concentration 1.0 ppm or less, with considering chlorine contained in boric acid as impurities. Later, the electric conductivity increased due to outflow of organic impurities which was resulted from oxidation of the ion exchange resin charged in the heavy water purification resin bed by deuterium dioxide in heavy water. Accordingly, the reference value of electric conductivity was increased to 5.0 μ S/cm and that of pD to 4.5 ~8.5.

(a) pD and Electric Conductivity

① Electric conductivity

A value of 5.0 μ S/cm was determined by referring to the control target of SGHWR, and, at the same time, by calculating the nitric acid concentration corresponding to pD of 4.5 in heavy water and the electric conductivity of 5.0 μ S/cm, followed by evaluation of entire surface corrosion of SUS304 and Zircaloy 2 materials in the solution of the above nitric acid concentration.

② PD

Nitric acid concentration corresponding to pD of 4.5 is 3.24×10^{-5} mol/l, and that corresponding to the electric conductivity of 5.0 μ S/cm is 1.24×10^{-5} mol/l. Under these conditions, the corrosion rate of SUS304 retains 0.0025 mm/yr or less and that of Zircaloy 2 material 0~0.075 mm/yr. It was confirmed that they showed complete anti-corrosion characteristics.

The value of pD also corresponds to the optimum anti-corrosive value (room temperature: pH 4~8.5; 70~95℃ : pH 4.5~7) of aluminum anti-vibration plate installed in the calandria tank.

(b) Chlorine Ion

It should be controlled within a range not to increase sensitivity of SUS304 materials

to stress corrosion cracking (SCC). As the conditions of occurrence of TGSCC (Trans Granular Stress Corrosion Cracking) caused by chlorine ion were assumed to be in a range of more than about 10 ppm below 100 ℃, the reference value was determined 1.0 ppm, 1/10 of the above value. According to the recent studies, however, the limit of IGSCC (Inter Granular Stress Corrosion Cracking) occurrence for the sensitized stainless steel showed a tendency to spread toward lower concentrations of about 2～3 ppm even under conditions of the low temperature as in the heavy water system. Therefore, it is actually controlled not to excess 10 ppb within the system now.

(c) Silica and Insoluble Impurities

Silica (SiO_2) and insoluble impurities (SS) can be neglected to maintain the integrity of materials, but it is preferable to control them as low as possible. The control values were determined based on those of the primary system of BWRs.

(2) Control Targets of Impurity Gases in Helium and Their Basis

Table V. 5-2 shows the control targets of helium.

(a) Deuterium

The control target of deuterium concentration has been determined 4.5% or below, based on the inflamability limit of deuterium with oxygen present in helium in a stoichiometric proportion plus margin.

The inflamability limit concentration of deuterium is shown in Fig. V. 5-1. As the helium temperature in the calandria tank is 70 ℃ and the helium purity of the inflamability limit concentration with 0.01% risk probability under the temperature is 91.5%, the inflamability limit concentration of deuterium is evaluated 5.6%, considering the stoichiometric proportion of deuterium and oxygen.

The control target was decided 4.5% or below by adding a margin to the evaluated value.

(b) Nitrogen

Based on operating experiences, the increase rate of the electric conductivity of heavy water per day was correlated with nitrogen concentration. It was found that the increase of the electric conductivity accelerated when nitrogen concentration in helium exceeded 0.18%. Therefore, the control target is determined 0.1% or less.

(3) Measurement Methods of Heavy Water Quality and Impurity Gases in Helium

(a) Measurement Methods of Heavy Water Quality

The heavy water quality is continuously monitored by process electric conductivity meter and the water is sampled once in two weeks by the analytical methods shown in Table. V. 5-3.

It should be noted that the method to measure heavy water concentration extending a

broad range (10~99.9%) with utilizing the absorption characteristics of heavy water in the near infrared range and the anion analytical method for the chlorine ion, nitric acid ion, sulfuric acid ion and others by ion chromatography were developed and established in "Fugen".

(b) Measarement Methods of Impurity Gases in Helium

Deutrium, oxygen, nitrogen and carbon dioxide, impurity gases in helium, are automatically measured every 4 hours by the in-line gas chromatography.

5.3 History of Chemistry Control of Heavy Water and Helium Systems

The major history of chemistry control of the heavy water and helium systems are described below:

① Change of water circulation of the heavy water purification resin beds from continuous to intermittent circulation and development of a decomposition catalyzer to remove deuterium dioxide (adoption has not been msde, but applicable to the actual unit)

② Lowering the circulating heavy water temperature in resin beds (installation of the cooling equipment)

③ Improvement and development of the resins with lower chlorine contents

④ Change of the control target of nitrogen concentration in helium (improvement of helium purification operation method)

⑤ Clarification of the deterioration mechanism of heavy water quality, and review and adoption of solutions

⑥ Commencement of using the extremely low chlorinated resins

⑦ Facility modification to enhance helium purification operation

As a result of these measures, the heavy water quality was significantly improved to achieve the original design value of 1 μ S/cm or less.

5.4 Operating Experiences of Heavy Water Quality [V-5]

(1) Subjects of Chemistry Control of Heavy Water System and History of Countermeasures

During the startup test in "Fugen", the electric conductivity of heavy water exceeded the control target of 5.0 μ S/cm, reaching 7.0 μ S/cm at the maximum. To keep it within the control target, continuous water circulation was changed to intermittent circulation for one to four hours per day, in order to limit the contact of the ion exchange resin in the heavy water purification resin beds with heavy water.

Maintaining of this water quality required to increase frequencies of resin exchanges, leading to increase the fresh heavy water usage and the degraded heavy water. The history of the life extension of the heavy water purification resins and measures to

improve the water quality can be divided into following four phases, according to the major subjects:

① Phase Ⅰ: Till the control method of electric conductivity of heavy water was established

② Phase Ⅱ: Till the through-year operation with two heavy water purification resin beds could be expected

③ Phase Ⅲ: Till the control of the dose equivalent rate of the heavy water piping could be expected

④ Phase Ⅳ: Period to improve the heavy water quality

These processes are indicated in Fig. V. 5-2.

(a) Phase Ⅰ: Startup Test to Commencement to Commercial Operation

The electric conductivity of heavy water reached 7.0 μS/cm at the maximum. This was because deuterium dioxide formed by radiolysis of heavy water oxidized and decomposed the weak basic anion exchange resin which was charged in the heavy water purification resin bed, releasing organic materials such as amine and oxime into heavy water. As a result of radiation oxidative degradation of these materials in the core, nitric acid ions or carbonic acid ions with rather good electric conductivity were formed. To address with this problem, intermittent operation was adopted.

(b) Phase Ⅱ: Commencement of Commercial Operation to 3rd Periodic Inspection

It was found that the heavy water electric conductivity could keep around 3.5 μS/cm even if the electric conductivity of the purification resin bed outlet exceeded to 2 μS/cm, a standard limit value of the resin exchanges. Thus we judged that it would be possible to loosen the resin exchange criteria to extend the usage period to about 250 EFPD and to make through-year continuous operation with the two purification resin beds.

(c) Phase Ⅲ: 3rd to 4th Periodic Inspection

Fig. V. 5-3 shows the change of surface dose equivalent rate of heavy water dump tank. During the third periodic inspection, the dose equivalent rate of the heavy water system components was considerably rising. Thus, further life extension of the heavy water purification resin beds and more frequent clean-up were required. As a result of the periodic water circulation with a 4% circulation rate (1 hour per day) in the heavy water purification resin bed by using the low chlorinated resin WA21L, the increase of the dose equivalent rate could be suppressed after the fourth periodic inspection.

(d) Phase Ⅳ: 4th to 7th Annual Inspections

In the gaseous waste treatment system condenser which was subject to the similar temperature and water quality conditions of the heavy water system, SCC was

detected on the SUS304 material. Measures including enhancement of helium purity control, adoption of the extremely low chlorinated resins and initiation of cooling down heavy water were taken so that the control of the heavy water electric conductivity could be achieved below 1.0 μ S/cm.

(2) Actual Results of Heavy Water Quality

The average water quality during rated power operation in every fuel cycle is shown in Table V. 5-4～6.

The water quality indexes which represent the heavy water quality consist of the electric conductivity, pD, chlorine ion, nitric acid ion (Table V. 5-4～6). The heavy water quality has been significantly improved since the fifth periodic inspection.

(3) Detorioration Mechanism of Heavy Water Quality

Outline of the deterioration mechanism of the heavy water quality and the measures for water quality improvement are shown in Fig. V. 5-4.

Main impurities and their sources are:

① Nitric Acid Ion, Nitrous Acid Ion

Formed by nitrogen (N) component of the ion exchange resins and nitrogen gas (N_2) as an impurity of helium gas due to the radiation oxidation reaction in the core.

② Carbonic Acid Ion, Bicarbonate Ion

Formed by carbon (C) component of the ion exchange resins due to oxidative degradation in the core.

③ Sulfate Ion

Formed by sulfur (S) component of the ion exchange resins due to oxidative degradation in the core.

④ Chlorine Ion

Exist in the ion exchange resin as an impurity and elutes to heavy water during water circulation in the resins.

⑤ Lithium Ion

Formed by the ^{10}B neutron absorption reaction $^{10}B(n, \alpha)\,^{7}Li$.

⑥ ^{10}B

Added to heavy water to control the excess reactivity of the core.

Under these mechanisms, impurity ions are formed in heavy water to increase its electric conductivity.

(4) Measures for Heavy Water Quality Improvement and Their Results

The history of measures taken to improve the heavy water quality is shown in Fig. V. 5-5. Those reflected to the actual unit consisted of following

① Shortening contact period of the resin and deuterium dioxide (adoption of intermittent water circulation)

② Development of a decomposition catalyzer of deuterium dioxide

③ Lowering of circulation water temperature in the resin beds (control of oxidized degradation)

④ Review of adoption of the strong basic anion exchange resin with anti-oxidation

⑤ Adoption of the low chlorinated resin WA-21L

⑥ Adoption of the extremely low chlorinated resin WA-21X

⑦ High purity control of helium by enhancement of helium purification

⑧ Extension of service period of the purification resin bed by changing a proportion of the anion resin to cation resin

⑨ Removal of impurities by increasing of the water circulation rate

The heavy water quality has been gradually improved by the above improvements. This history is shown in Fig. V. 5-6.

These various measures taken to improve the heavy water quality contribute to the improvement of economy, as they extended service period of the resins, thereby reducing exchange frequencies of the ion exchange resins and resulting in decrease of generation of the degraded heavy water. Table V. 5-6 shows the results of water circulation in the heavy water purification resin beds.

(5) Findings and Special Mention Obtained from Operating Experiences

(a) Change of System Heavy Water Concentration

Heavy water with the concentration of 99.85% was loaded in the heavy water system as the initial load. It decreased to 99.74% at the commencement of commercial operation, as shown in Fig. V. 5-7, but no major decrease in the concentration has been found since then.

The heavy water concentration should be kept at a high level to improve the neutron economy of the core and burn fuel economically. Therefore, control of heavy water concentration has ben made during inspections of the heavy water and helium systems. In particular, during the deuteration work conducted in the exchange of the resins, the criteria on completion of the deuteration were determined to be higher than the heavy water concentration in the system, aiming at prevention of the heavy water concentration in the system from decreasing.

In measurement of the heavy water concentration, the method developed in "Fugen" was applied which used the absorption of the near infrared beam (wave length of 1,660 nm, etc.).

This method is to measure the absorbency of the sample heavy water in the quartz glass cell with a 5～20 mm thickness by using the near infrared radiation absorptiometer and has simple and highly accurate features. The accuracy of measurement is within ±0.005% in the case of heavy water of about 99.73%.

(b) Behavior of Increase of Tritium in Heavy Water System

① Formation of Tritium

The formation reaction of tritium is described below:

$^{2}H(n,\gamma)^{3}H$

$^{3}He(n,p)^{3}H$

$^{7}Li(n,n\cdot\alpha)^{3}H$

$^{10}B(n,2\alpha)^{3}H$

In "Fugen", the source nuclides of the above reactions exist in heavy water in the following forms:

^{2}H : composes the molecular of heavy water

^{3}He : formed by β decay of tritium and deposits

^{7}Li : formed by $^{10}B(n,\alpha)^{7}Li$ reaction

^{10}B : added to heavy water as a control material of reactivity

TableV. 5-7 shows the calculation results of tritium formed.

② Tritium Concentration in Heavy Water

Based on the result of tritium increase in heavy water as shown in Table V. 5-8, the saturated concentration of tritium was estimated:

(i) Basic equation

$$dn = N\Phi\sigma\cdot[\tau_i/(\tau_o+\tau_i)]\,dt - \lambda n\,dt$$

where, N : the number of atoms of deuterium ($1/cm^3$)

n : the number of atoms of tritium ($1/cm^3$)

λ : tritium decay constant (1/sec)

Φ : effective neutron flux in calandria ($1/cm^2/sec$)

σ : nuclear reaction cross section (cm^2)

τ_i : residence time of the heavy water in calandria (sec)

τ_o : residence time of the heavy water out of calandria (sec)

(ii) Estimation of saturated concentration of tritium

After the above equation is solved, the number of atoms is transformed to the tritium concentration and the constant terms are simplified to obtain the equation for a day.

Then,

$$T_i = PK[1-\exp(-\lambda t)] + T_{i-1}\exp(-\lambda t)$$

where, T_i : tritium concentration on October 30, 1978

T_{i-1} : tritium concentration at i-1 days from October 30, 1978

P : specific power (thermal output 557 MW)

K : saturation value of tritium concentration

The value of K was calculated to be 16.6 mCi/cm^3 (615 Bq/cm^3)

(c) Distribution of Radiation Dose Rate of Heavy Water System and Purification of Heavy Water During Reactor Shutdown

① Surface Radiation Dose Rate of Components and Piping of Heavy Water System

The surface dose equivalent rate of the heavy water dump tank is shown in Fig. V. 5-9. The surface dose equivalent rate of the piping and components of the heavy water system during the seventh periodic inspection is shown in Fig. V. 5-10. Where heavy water flows, the dose equivalent rate was found to be as low as 10～20 mR/h, while where the flow rate was low and crud was easy to deposit, a local increase of the dose equivalent rate was observed.

② Heavy Water Purification During Reactor Shutdown

During normal operation, Co-60 concentration of heavy water is in the order of $1x10^{-2}$ μ Ci/cm^3, which is higher than that of the primary system by a factor of 100 ～200. Therefore, heavy water should be purified during the reactor shutdown period.

(i) Heavy water purification work

The heavy water purification resin bed B which was in service during reactor operation has been used for continuous water circulation for 24 hours (flow rate of 20 m^3/h) to remove impurities in heavy water.

(ii) Effect of heavy water purification on reduction of dose equivalent rate during reactor shutdown

Fig. V. 5-10 shows measurements of the surface dose equivalent rate of the components and piping of the heavy water system before and after the heavy water purification operation.

(iii) Estimation of Co-60 deposited on inner surface of heavy water dump tank

Based on the surface dose equivalent rate of the heavy water dump tank and measurements of Co-60 concentration in heavy water, the density of Co-60 deposited on the inner surface of the heavy water dump tank was evaluated. These results were indicated in Table V. 5-8.

Table V. 5-9 shows effects of the heavy water purification operation on reduction of dose equivalent rate from the scheduled shutdown in 1986 to the seventh periodic inspection.

5.5 Experiences of Chemistry Control in Helium System

(1) Operating Experiences of Helium System

Deuterium and oxygen formed by radiolysis of heavy water generally reach their maximum values after about two weeks from the begining of core and then decrease with approaching the end of core. This results from the effects of lithium ion, carbonic

acid ion and bicarbonate ion in heavy water on the radiolysis reaction of heavy water.

Even if the nitrogen gas concentration is decreased to 0.005 vol% or below before reactor operation, it gradually rises with reactor operation, often reaching about 0.05 vol% at the end of core.

Carbonic gas is the final product of the purification bed resins oxidized and decomposed. Carbon monoxide and methane have not been detected by the measuring method with 0.01% lower limit of detection capability.

The catalyzer for the recombination uses 0.5% palladium content alumina pellet (Al_2O_2-Pd cylindrical alumina coated with palladium). The heavy water decomposition gases generated have been fully recombined at the recombiner outlet without degradation of the catalyzer.

(2) Operating Experiences of Helium Purification

Helium purification is made for removing impurity gases, such as air, which enter the system during the periodic inspection of the heavy water and the helium systems and for reducing formation of radioactive noble gases including Ar-41 in heavy water, as well as for removing nitrogen in helium.

(a) Increase of Nitrogen Concentration Due to Air Entering Heavy Water and Helium Systems During Periodic Inspections

Nitrogen concentration will increase to 3～5%, the main cause of which lies in exchange and inspection works of the control rod drive mechanisms and the local output meters.

(b) Helium Purification Work

① Selection of Helium Purification Operation Mode

The helium purification operation mode consists of two modes – the dehumidifying operation mode and the dehumidifying and decarboxylation operation mode – as shown in Table V. 5-10. When carbonic acid concentration in helium is found to be 0.001 vol% or more before helium purification work, the dehumidifying and decarboxylation operation mode is selected.

② Injection of Oxygen into Helium System Accompanying with Helium Purification Work

To reduce release of tritium gas (DT, T_2) in helium into the atmosphere, oxygen is injected to the helium system, thereby deuterium and tritium gas are recovered as heavy water.

③ Impurity Absorption Performance of Helium Purification Unit

70 l of the activated carbon is charged as the absorbent. It is cooled to –190℃ by liquid nitrogen to use in the helium purification operation after cooling. The absorption distribution, as seen in Fig. V. 5-11, is expected to occur inside of the

helium purification unit.

(3) Experiences of Oxygen Injection into Helium System

Table V. 5-11 shows the excess deuterium concentration. Oxygen is determined to be injected a day before the reactor shutdown.

(4) Deuterium Decomposition [V-5]

The deuterium decomposition is designed to be 12.5 g-D_2O/MWh, but now it is lower than this value, resulting from the improvement of heavy water quality.

The value of G of radiolysis of heavy water differs from types of radiation, as shown in Table V. 5-12. By incorporating the reaction equation which considered the effects of impurities shown in Fig. V. 5-14 into the reaction equation in Table V. 5-13, radiolysis reaction was analyzed based on "Fugen" data.

The analysis revealed the following behaviors:

(i) With concentration changes of L_i^+ formed by the reaction of $^{10}B(n, \alpha)\ ^7L_i$, changes of proportion of CO_2/DCO_3^- vary the amount of deuterium decomposition. DCO_3^- accelerates the heavy water decomposition , while CO_2 control it.

(ii) D_2 in the helium system depresses the heavy water decomposition.

(iii) The heavy water decomposition is accelerated with increasing ^{10}B concentration.

5.6 Exchange Experiences of Heavy Water Purification System Resins

(1) Operating Experiences of Heavy Water Purification System

(a) Operating Experiences of Heavy Water Purification Resin Column

The heavy water purification resin column is charged with the mixed bed consisting of the strong acid cation exchange resin (SKN1) and the weak basic anion exchange resin (WA21X) and so that boron in heavy water may not be removed.

The ratio of resins (WA21X : SKN1) per bed is 2 : 1, with a 1000 liter of resin in total. After the measures had been taken to improve the water quality, accumulated water circulation period amounted to 800～1200 hours, which extended the service life of the resin column at least five times longer than before.

(b) Operating Experiences of Liquid Poison Removal Resin Column

The removal resin column is charged with a 800 liter of strong basic anion exchange resin (SAN-1). Exchange of resin should be conducted during the reactor shutdown, the same as the case of the purification resins. Therefore, the poison removal resin will be exchanged when the equilibrium boric acid concentration (obtained from the absorption equilibrium equation mentioned below with using the absorption amount of boric acid in the removal resin column just before reactor shutdown) exceeds the boric acid concentration required at the end of the core. The absorption equilibrium relation of boric acid in heavy water with the strong basic anion exchange resin is:

$$Q = SY / (1 + Y) + C \cdot \exp (2000 / T - 2.90)$$

where, $Y = C \cdot \exp (4230 / T - 3.65)$

Q : amount of equilibrium absorption of boric acid (mol-B/l-Resin)

S : Langmuir's absorption constant (1.07 mol-B/l-Resin)

C : equilibrium boric acid concentration (mol-B/l-D_2O)

T : equilibrium temperature (K)

(2) Exchange Experiences of Heavy Water Purification System Resins

The ion exchange resins are exchanged when the performance degrades. Procedures to exchange the heavy water purification system resins and the standard work schedule are shown in Fig. V. 5-12 and Table V. 5-15, respectively. As the used resins contain large amount of heavy water and tritium, the loss of heavy water and the tritium discharge will be increased if they are disposed without treatment. Dedeuteration in the exchange process, aims at recovering heavy water from the used resins before disposal and completely taking tritium from them, by replacing heavy water in the resin with light water.

(a) Exchange of Heavy Water Purification System Reins

Most important matters of handling technique of the heavy water purification system resins are how to make dedeuteration and deuteration efficiently in the resin exchange and how to reduce the amount of degraded heavy water as low as possible with using small amount of new heavy water.

① Improvements of Dedeuteration Work

To dedeuterate the resins of the heavy water purification resins or the poison removal resin, the resins are transferred from the column to the deuteration column, by downward supply of light water from the upper part of the column to replace heavy water in the resins with light water. The resultant degraded heavy water, a mixture of heavy water and light water, with a heavy water concentration of about 95%～1% is transferred to the degraded heavy water tank B. When the heavy water concentration decreases to 1% or less, dedeuteration will be stopped.

Changes of the heavy water concentration during dedeuteration is shown in Fig. V. 5-13.

The degraded heavy water (converted to 100% concentration) generated during the dedeuteration amounts to about 0.6 m^3.

② Improvements of Deuteration Work

After transferring the new resins cleaned in the resin clean-up column to the deuteration column, heavy water is supplied upward from the bottom of the deuteration tower to replace light water in the resins. When the heavy water concentration at the deuteration column outlet exceeds the system heavy water

concentration (99.7%), deuteration work is completed and the new resins deuterated are transferred to the heavy water purification column or the poison removal column.

The heavy water used during deuteration is about 3 m^3.

Improvements taken to efficiently deuterate the resins and to lower the generation of the degraded heavy water are described below:

(i) The linear velocity of heavy water supplied during deuteration process was determined to be about 1.2 cm/min based on the operating experiences of the actual unit.

(ii) Light water in the resin should be blown with dry air before deuteration.

(iii) Light water remained in the upper piping of the heavy water purification column should be blown with dry air.

(iv) After light water in the lower piping of the heavy water purification column is blown, flushing is done with new heavy water to raise the heavy water concentration in the lower header of the heavy water purification column and increase efficiency of deuteration.

(v) The new resins in the heavy water purification column are in a uniform condition by scrubbing with dry air to uniformize the heavy water flow and increase efficiency of deuteration.

The heavy water concentration during deuteration is shown in Fig. V. 5-14. The the degraded heavy water generated during deuteration is about 1.5 m^3.

(b) Experiences of Resin Exchange

The degraded heavy water generated during dedeuteration and deuteration of the resins of the heavy water purification system averages about 2.5 m^3 per resin exchange work. It takes about 7 days to exchange the resins. As a result of life extension measures for the resin (measures to improve the heavy water quality), the resin exchange has been recently reduced to about 2 beds a year, while it had to be performed on four beds a year in early stage of the commercial operation. The degraded heavy water has been also decreased to below the design value. These results apparently show the effects of improvements and development.

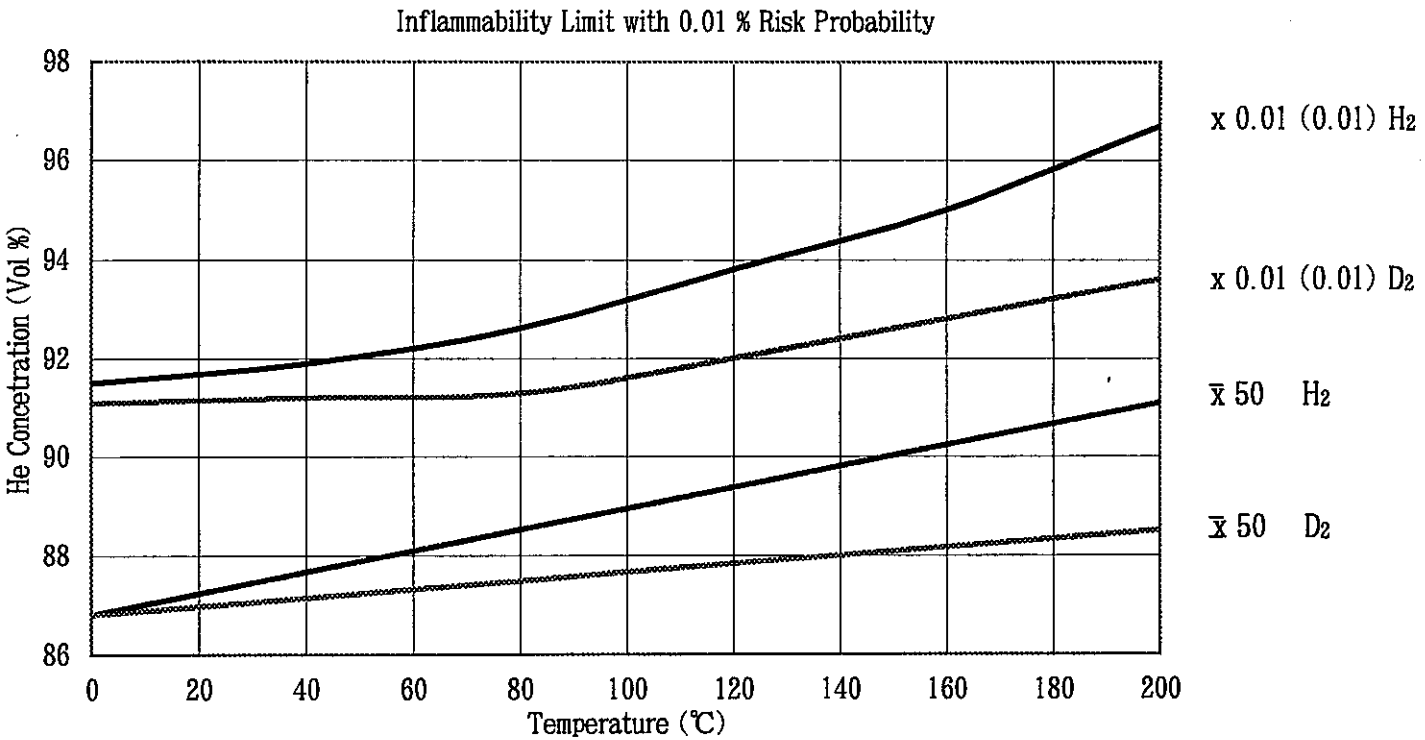


Fig.V.5-1 Inflammability Limit Test of Heavy Water Decomposition Gas in Helium

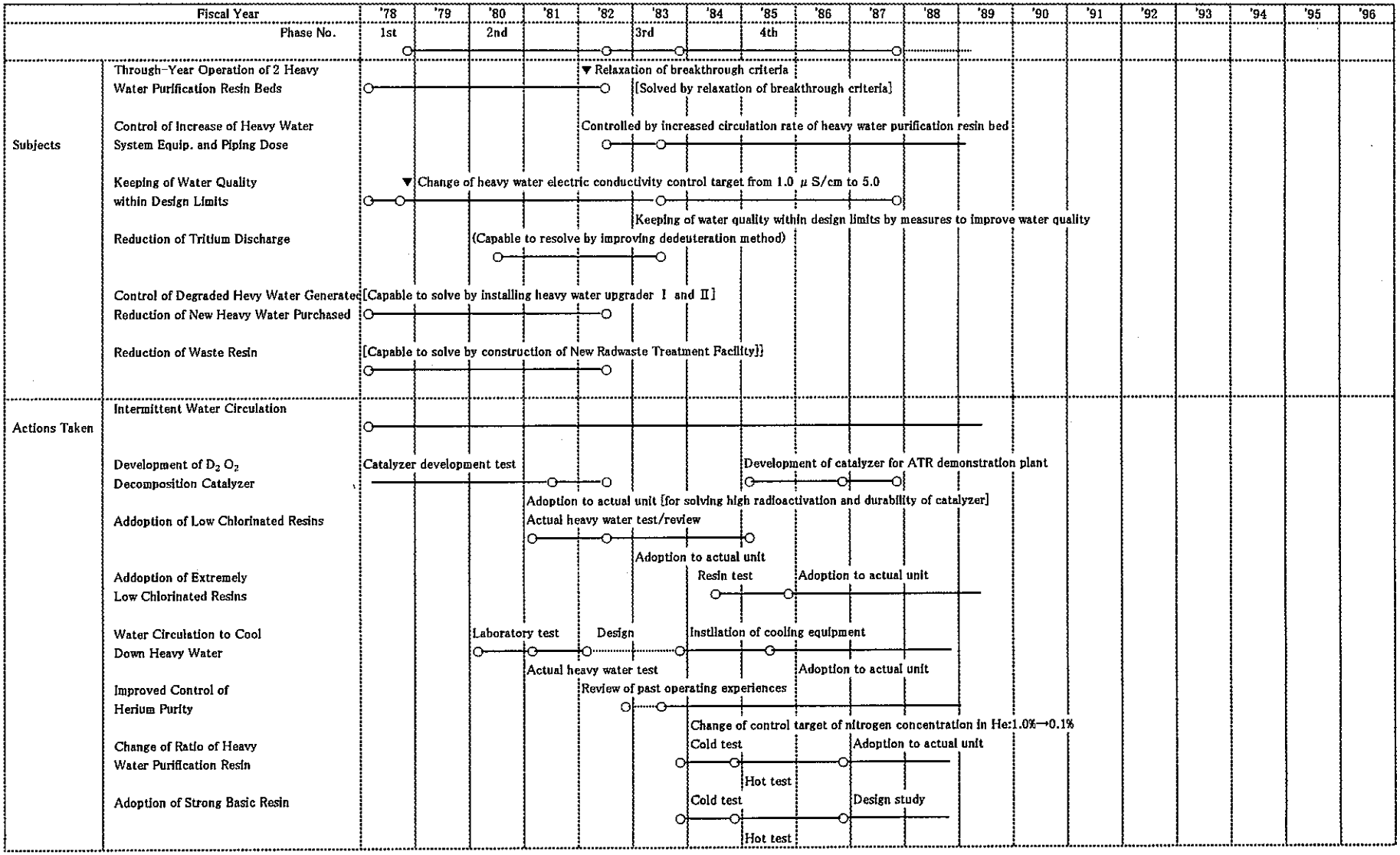


Fig. V. 5-2 Subjects and Actions Taken for Life Extension of Resin Used in Heavy Water Purification Resin Bed

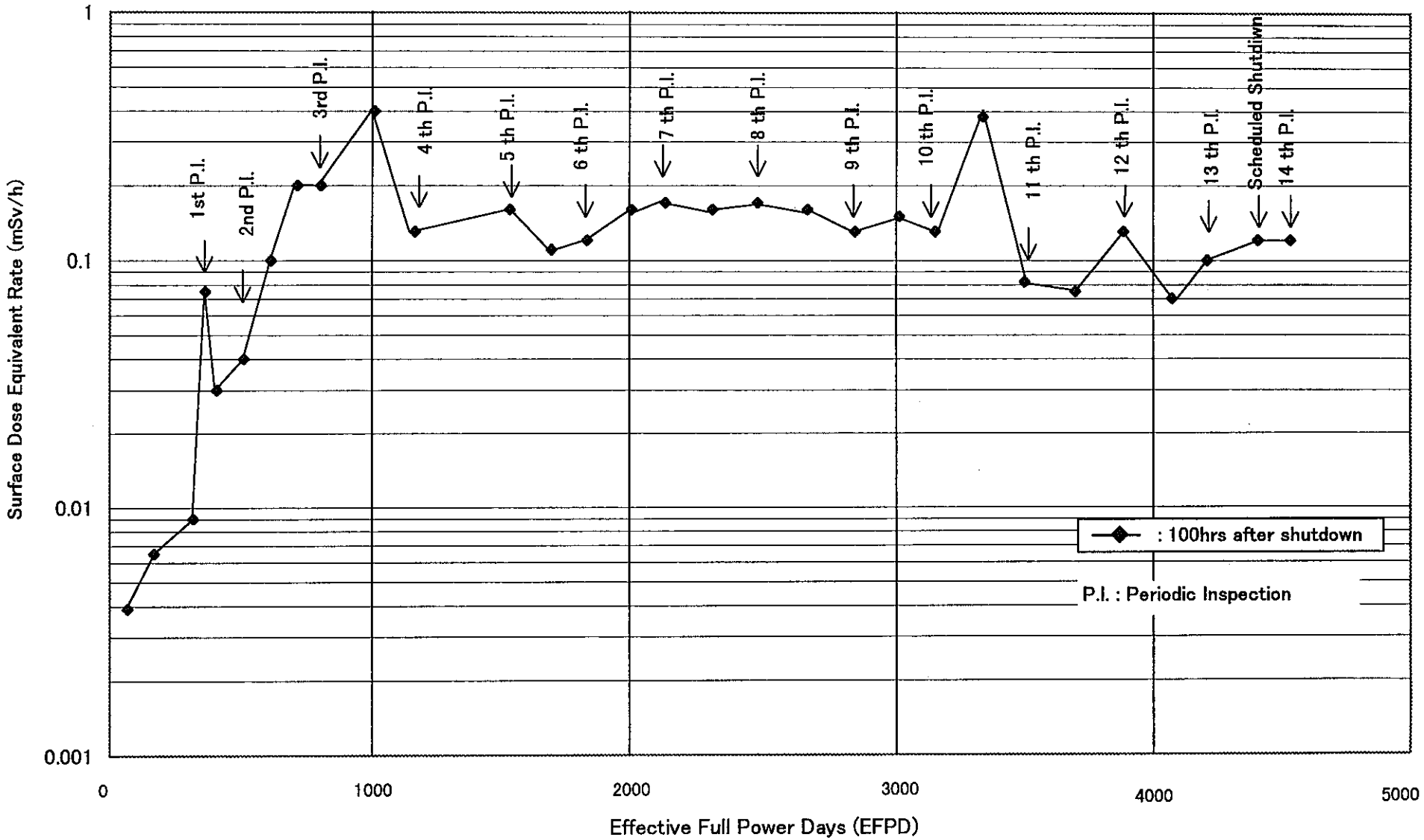


Fig.V.5-3 Chenge of Surface Dose Equivalent Rate of Heavy Water Dump Tank

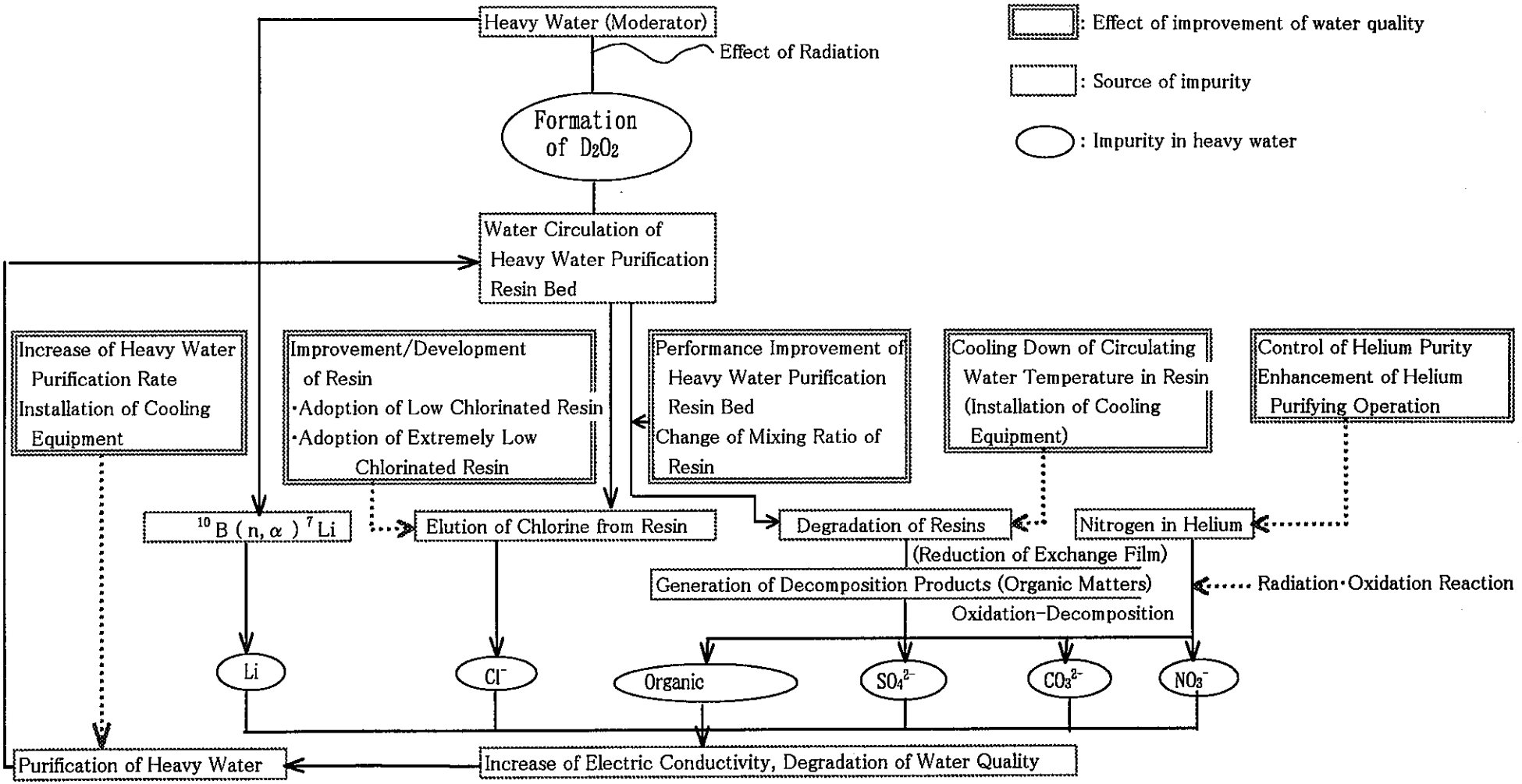


Fig.V.5-4 Deterioration Mechanism of Heavy Water Quality and Outline of Measures to Improve Water Quality

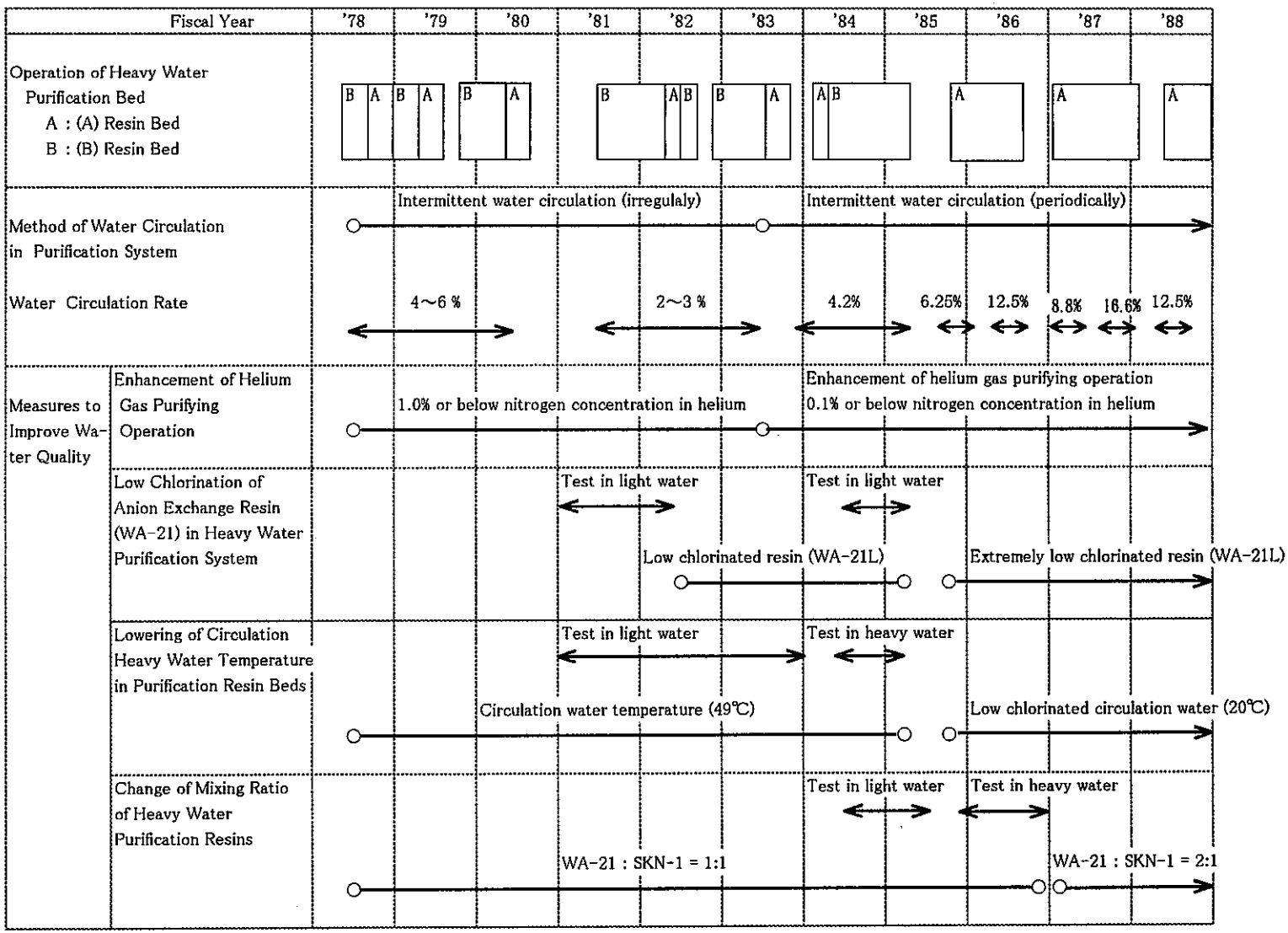


Fig. V. 5-5 History of Measures Taken to Improve Heavy Water Quality

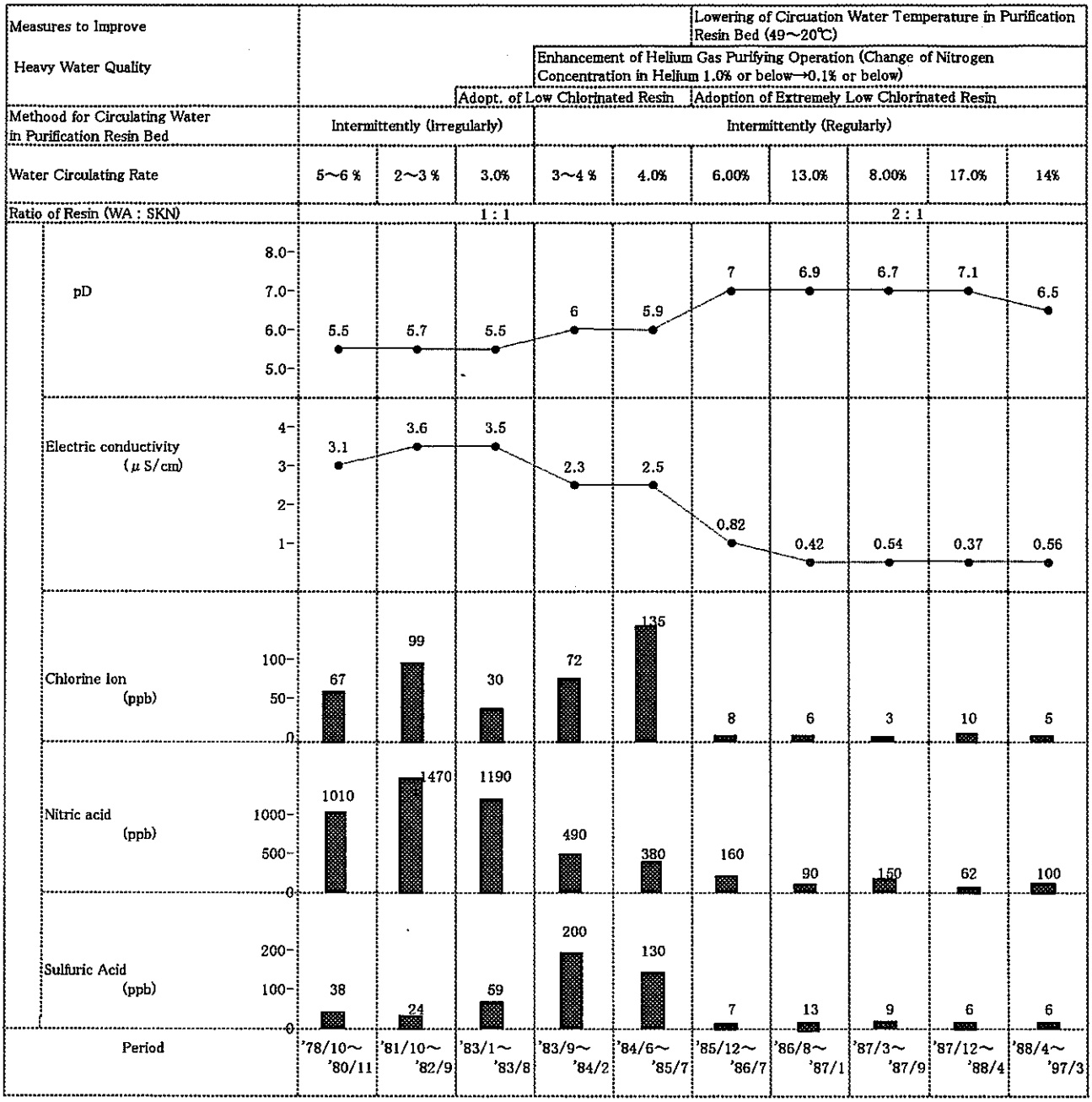


Fig. V. 5-6 Measures to Improve Heavy Water Quality and Their Actual Effects

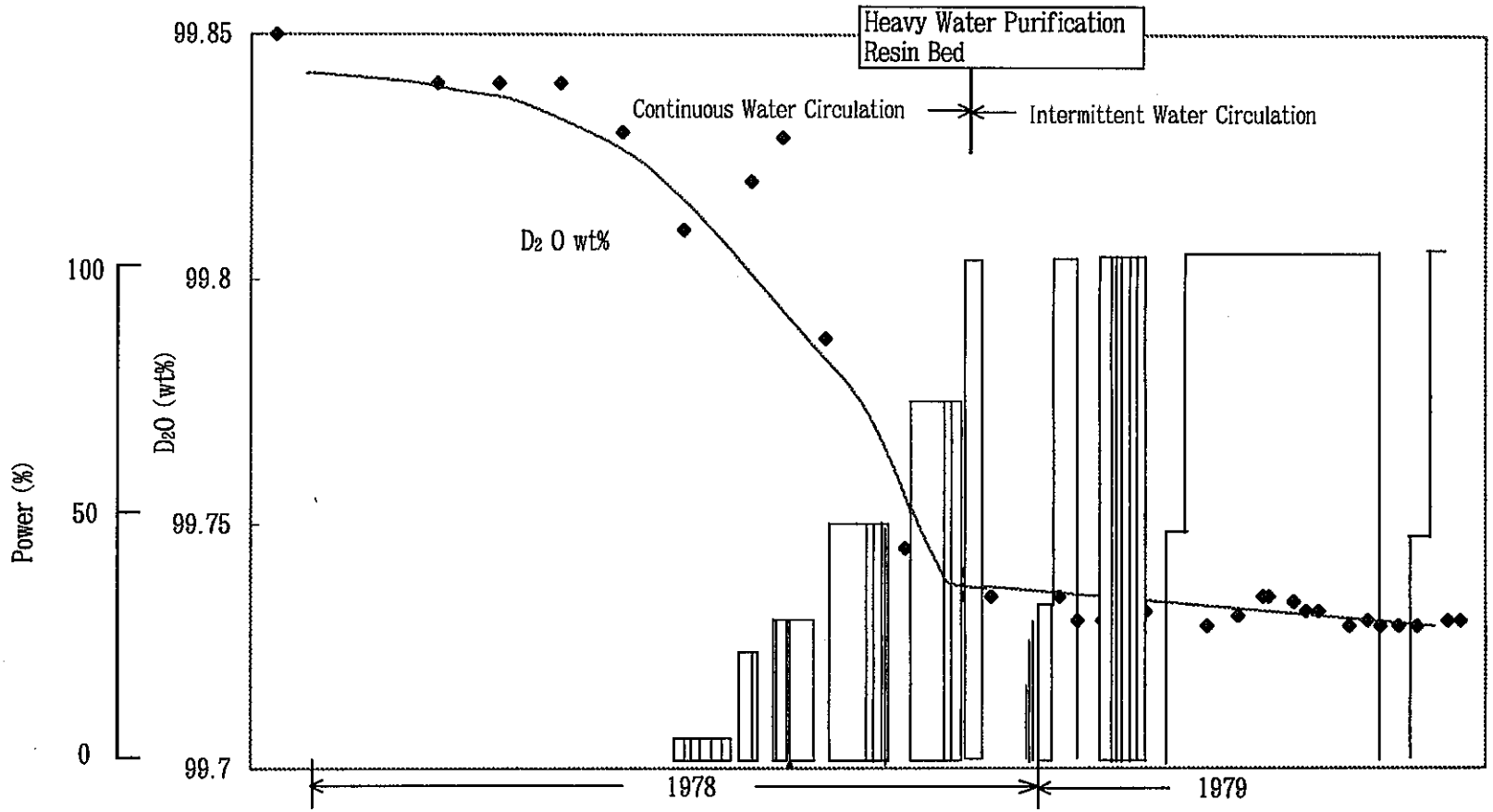


Fig.V.5-7 Change of Heavy Water Concentration from Initial Fuel Loading to Startup Test

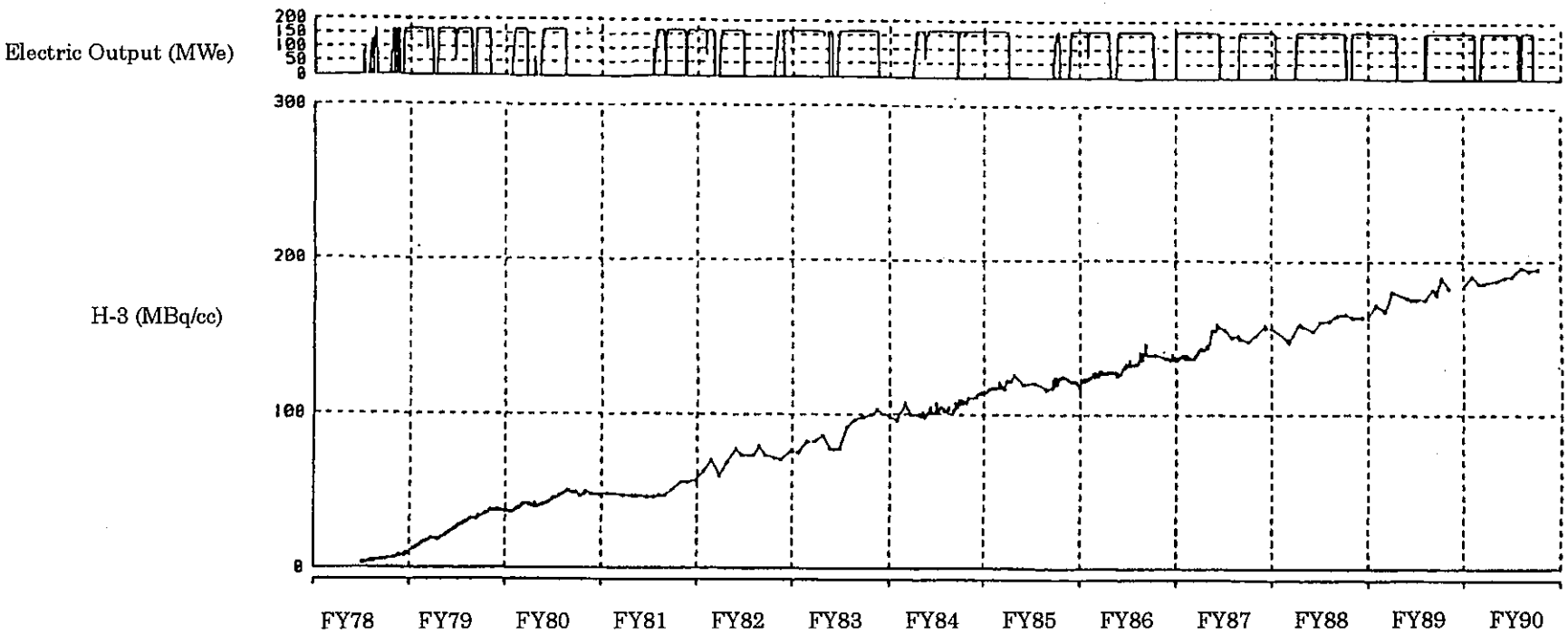


Fig.V.5-8 Change of Tritium Concentration in Heavy Water

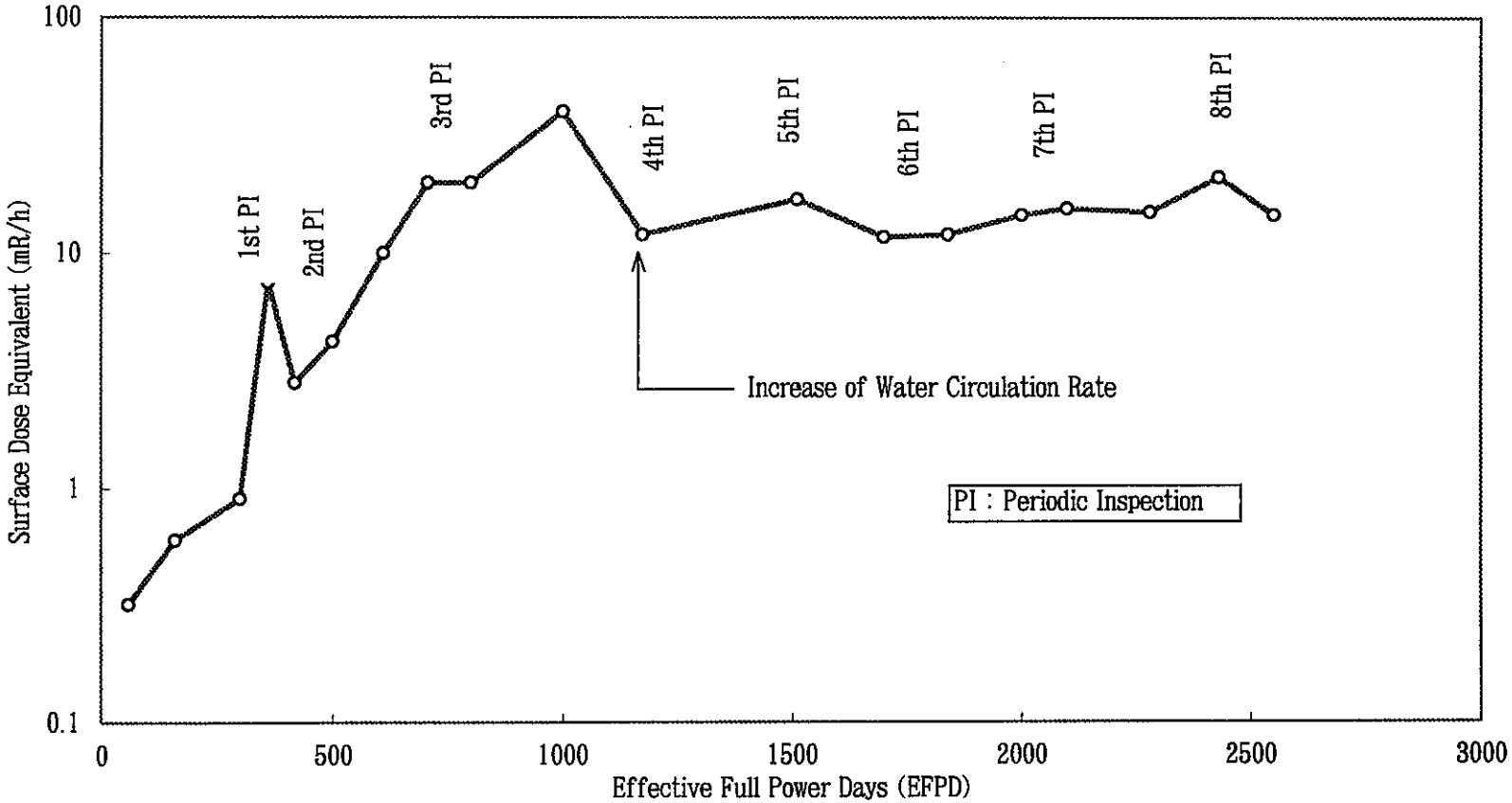
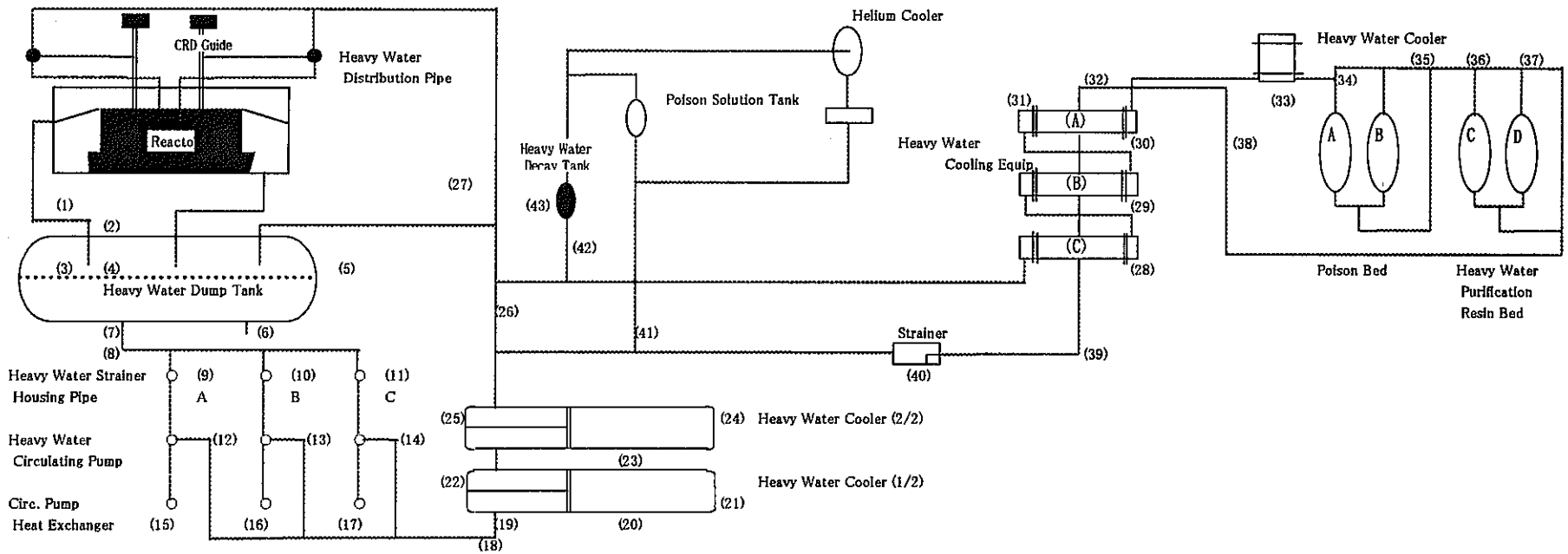


Fig.V.5-9 Change of Surface Dose Equivalent of Heavy Water Dump Tank



Measurement of Dose Equivalent Rate of Component and Piping of Heavy Water System (mSv/h)

Measured Point	(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)
Before Purification	0.92	0.41	0.34	0.26	0.11	17.8	0.50	0.24	8.00	10.0	0.55	0.17	0.20	0.19	5.70	5.20	5.00	0.17
After Purification	0.83	0.36	0.20	0.16	0.04	22.0	0.33	0.07	0.08	0.08	0.20	0.07	0.08	0.08	3.00	3.40	3.30	0.08
Measured Point	(19)	(20)	(21)	(22)	(23)	(24)	(25)	(26)	(27)	(28)	(29)	(30)	(31)	(32)	(33)	(34)	(35)	(36)
Before Purification	0.23	0.08	0.07	0.32	0.11	0.05	0.58	0.45	0.65	4.50	5.50	4.00	4.50	0.50	1.50	0.40	0.35	0.10
After Purification	0.12	0.05	0.04	0.20	0.08	0.02	0.43	0.27	0.53	4.00	5.00	3.50	4.00	0.45	1.35	0.35	0.30	0.09

Measured Point	(37)	(38)	(39)	(40)	(41)	(42)	(43)	Remarks
Before Purification	0.19	0.36	0.30	7.50	0.15	30.0	200	Dose equivalent rate measured 1 day after reactor shutdown and before purification of heavy water
After Purification	0.08	0.20	0.25	7.50	0.12	30.0	200	Dose equivalent rate measured after heavy water purification operation 24 hrs ('88/4/17 10:00~4/18 10:00)

Fig. V. 5-10 Measurement of Dose Equivalent Rate of Component and Piping of Heavy Water System

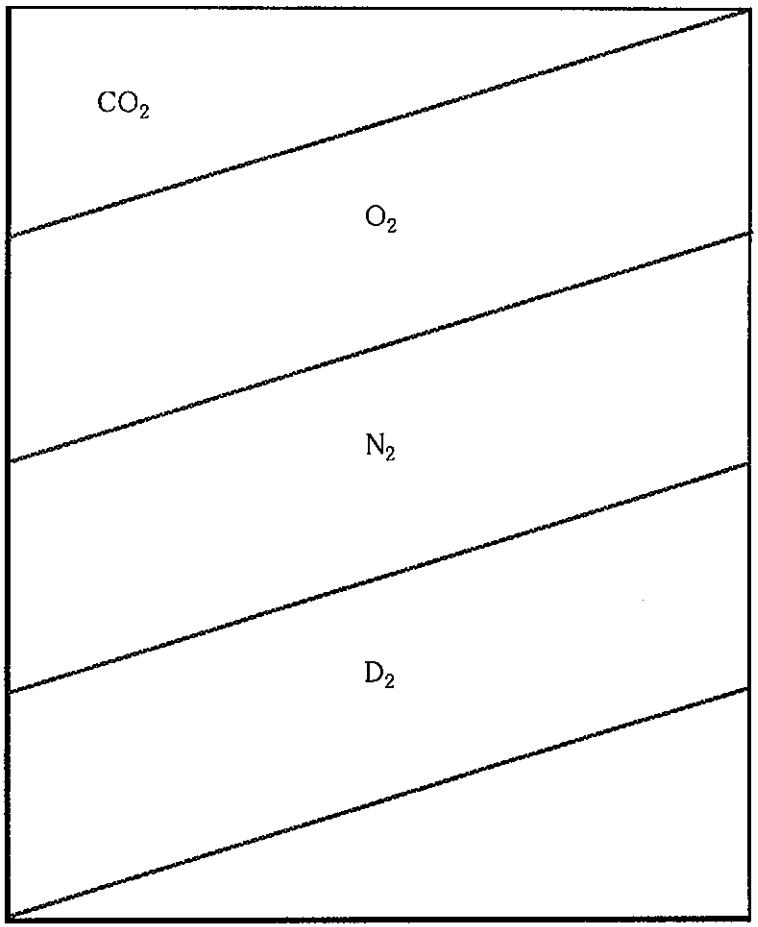


Fig.V.5-11 Adsorption Characteristics of Impurity Gases in Activated Carbon

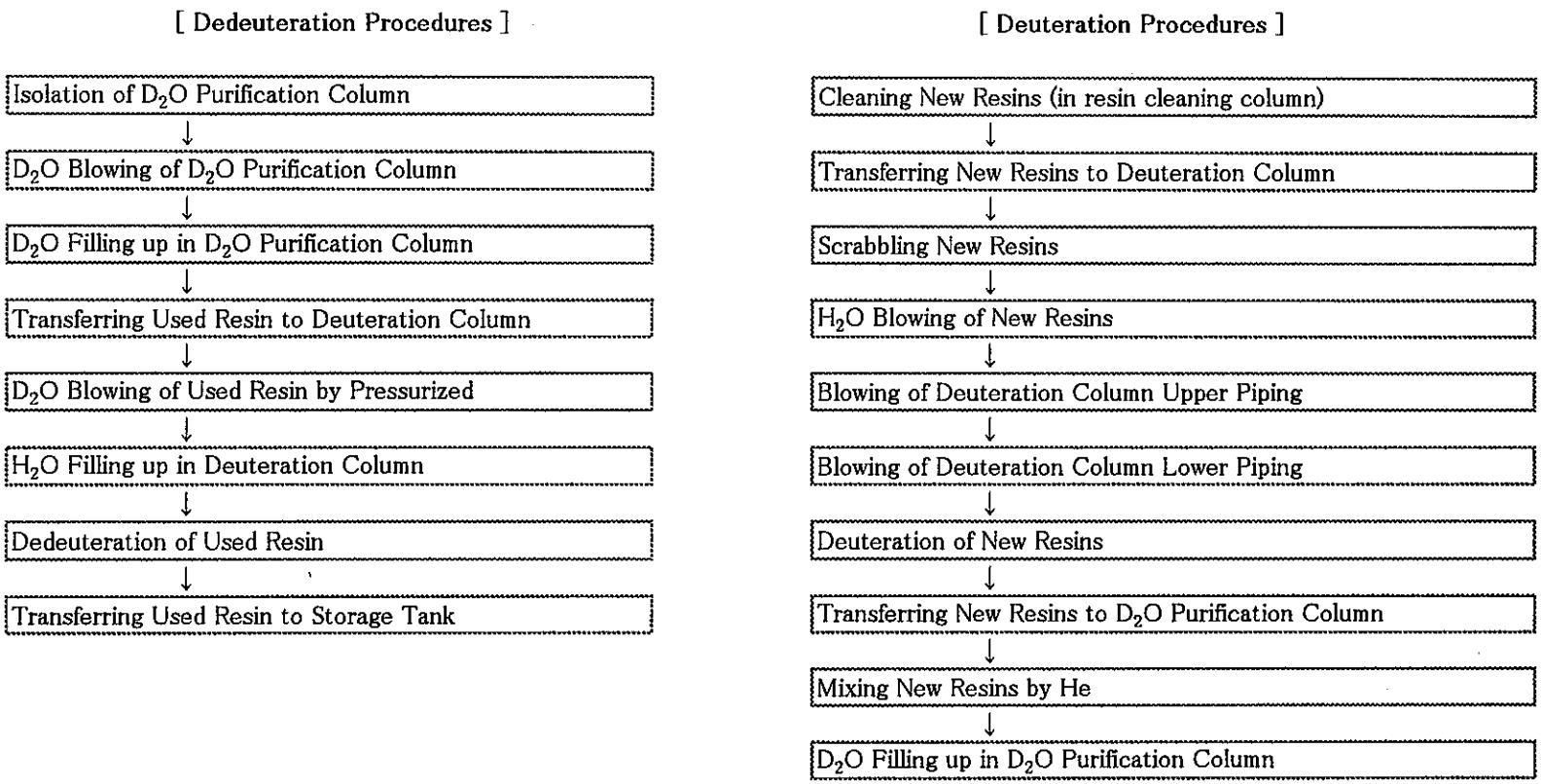


Fig.V.5-12 Resin Exchange Procedures in Heavy Water Purification System

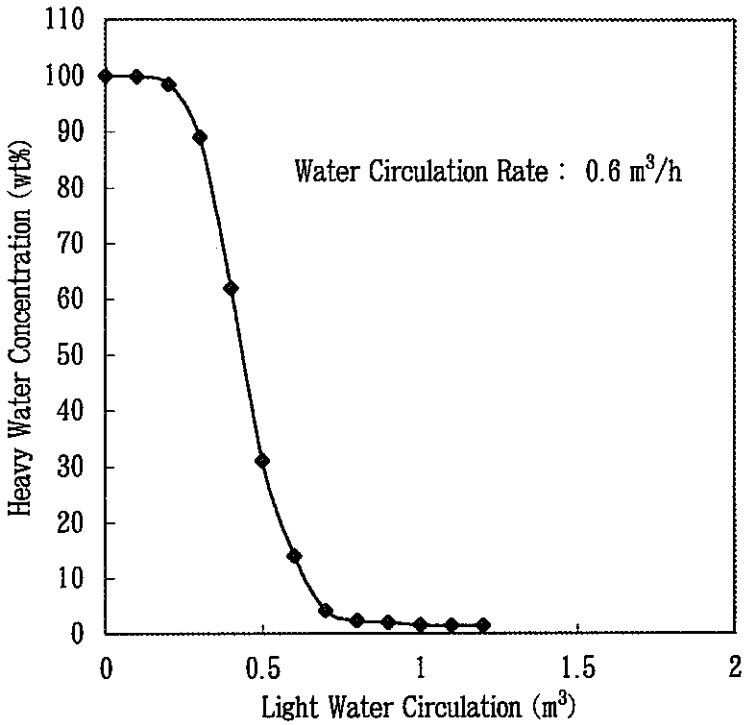


Fig. V. 5-13 Dedeuteration Curve

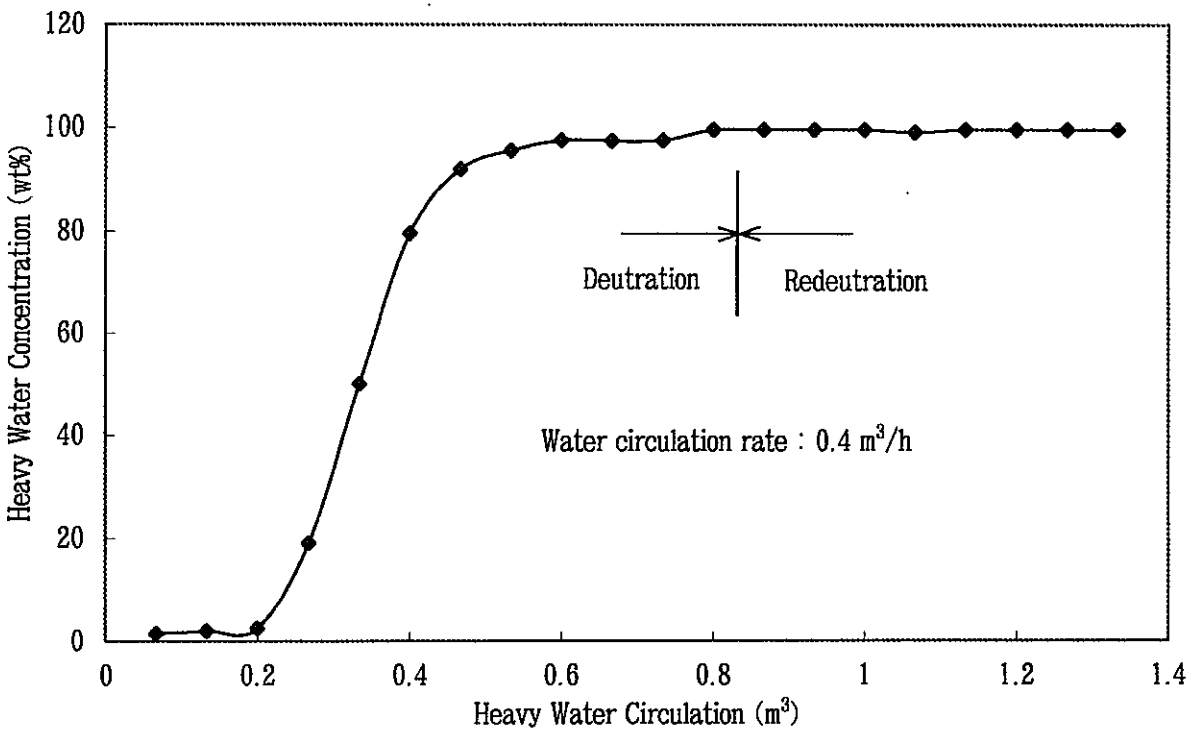


Fig. V. 5-14 Deuteration Curve

Table V.5-1 Heavy Water Quality Control Target

Item	Design Value	Value in Operation Control Guideline	
		Original Value	Revised in '79/3
pD (at 25 ℃)	5.5～8.5	5.5～8.5	5.5～8.5
Electric Conductivity (μS/cm)	≦ 1.0	≦ 1.0	≦ 1.0
Chlorine Ion (ppm)	≦ 0.1	≦ 0.1	≦ 0.1
Insoluble Impurities (ppm)	≦ 0.5	≦ 0.5	≦ 0.5
Silica (ppm)	---	≦ 1.0	≦ 1.0

Table V.5-2 Helium Control Target

Item	Control Target
Deuterium (%)	4.5
Nitrogen (%)	0.1

Table V.5-3 Measurement Methods of Heavy Water Quality

Item to Be Measured	Control Target	Measurement Method
Conc. of Heavy Water	---	To measure heavy water percolated by 0.1 μ membrane filter using near infrared spectrophotometer
Electric Conductivity	5.0 μS/cm	To measure flowing sample water with cell type electric conductivity meter
pD	4.5~8.5	To measure samle water with flow cell type pD meter
Insoluble Impurioties	≦0.5 ppm	To percolate sample by 0.4 μ Nucli Pore Filter and to measure quantitatively weight of entrapped material
Silica	≦1.0 ppm	To color with JIS-K010 molybdate blue method and to measure with spctrophotometer
Chlorne Ion	≦1.0 ppm	To measure sample water with ion chromatography
Nitrous Acid Ion	---	〃
Nitric Acid Ion	---	〃
Sulfuric Acid Ion	---	〃
Lithium	---	To measure sample water with flameless atomic absorption phtometer
Boron	---	To measure sample water with plasma emmission nalyzer
Deuterium Peroxide	---	To color sample water with Ti-EDTA method and to measure with spectrophotometer
Organic Carbonic Acid (TOC)	---	To measure sample water with TOC meter. to measure total carbon TC to calculate TOC. (TC-IC=TOC)
Inorganic Carbonic Acid (IC)	---	To directly measure sample water with TOC meter
Dissolved Oxigen	---	To measure sample water with flow cell type DO meter by circulating water
Dissolved Hydrogen	---	〃
Metal (Fe, Ni, Co, Cr, Al)	---	The filtrate obtained by sample water percolated by 0.4 μ Nucli Pore Filter and the sample water as-taken are measured with frameless atomic absorption photometer. The filtrate is determined for iron, and the sample water as a toal for (total - ion =) crud.
γ Nuclide		The sample water is passed through a 0.4 μ Nucli Pore Filter, two cation exchange filters and two anion exchange filters, sequentially. The radioactive nuclides entrapped by each filter are measured quantitatively with Ge(Li) detector. The nuclides entrapped by 0.4 μ Nucli Pore Filter are deteamined as crud and those entrapped by ion exchange filter are determined as ion.

Table V.5-4 Results of Heavy Water Quality (1/2) (Average Water Quaity During Rated Power Operation)

Fuel / Operating Cycle			Startup	1st cycle	2nd	3rd	4th	5th	6th	7th	8th	9th	10th	11th	12th
Periodic Inspection				1 st		2 nd		3 rd		4 th	5 th		6 th		7 th
Electric Conductivity		[μS/cm]	2.33	2.98	3.79	3.64	3.45	3.92	3.51	2.34	2.49	0.82	0.42	0.54	0.37
pD		[-]	5.85	5.55	5.10	5.34	5.83	5.34	5.53	5.96	5.89	7.00	6.91	6.71	7.10
Chlorine		[ppb]	63	65	104	62	87	141	30	72	135	8	6	3	10
Nitrous Acid Ion		[ppb]	54	52	48	73	45	---	74	29	14	13	7	7	5
Nitric Acid Ion		[ppb]	692	1010	823	1230	1550	1260	1190	486	380	160	90	153	62
Sulfuric Acid Ion		[ppb]	47	44	9	27	16	40	59	203	133	7	13	9	6
Lithium Ion		[ppb]	---	---	---	---	---	---	---	---	66	62	23	30	21
B-10		[ppm]	6.40	2.56	1.20	2.86	2.89	2.54	319	1.74	4.93	1.88	1.72	2.14	2.93
SiO_2		[ppb]	31	77	46	36	37	57	38	32	84	15	8	20	16
SS		[ppm]	0.06	0.2	< 0.1	< 0.1	0.2	< 0.1	< 0.1	< 0.1	0.1	< 0.1	< 0.1	< 0.1	< 0.1
IC		[ppm]	---	---	---	---	---	---	---	---	0.89	0.178	0.088	0.159	0.081
TOC		[ppm]	---	---	---	---	---	---	---	---	0.54	0.244	0.244	0.210	0.128
D_2O_2		[ppm]	12.0	7.72	4.96	7.13	8.01	9.67	9.72	6.72	8.51	5.38	6.39	6.03	7.08
Fe	Ion	[ppb]	---	---	---	---	---	---	---	---	2.3	2.0	< 1	< 1	< 1
	Crud	[ppb]	---	---	---	---	---	---	---	---	< 1	1.5	< 1	< 1	< 1
Ni	Ion	[ppb]	---	---	---	---	---	---	---	---	8.2	5.9	5.8	8.5	3.0
	Crud	[ppb]	---	---	---	---	---	---	---	---	5.9	2.1	1.0	4.8	8.0
Co	Ion	[ppb]	---	---	---	---	---	---	---	---	1.0	2.4	1.3	1.0	< 1
	Crud	[ppb]	---	---	---	---	---	---	---	---	0.5	1.3	< 1	2.0	< 1
Cr	Ion	[ppb]	---	---	---	---	---	---	---	---	2.4	3.1	2.9	2.2	2.0
	Crud	[ppb]	---	---	---	---	---	---	---	---	0.7	1.0	1.2	1.5	4.0
Cu	Ion	[ppb]	---	---	---	---	---	---	---	---	< 1	< 1	3.6	3.7	< 1
	Crud	[ppb]	---	---	---	---	---	---	---	---	< 1	< 1	2.5	1.8	< 1
Al	Ion	[ppb]	---	---	---	---	---	---	---	---	1.3	2.1	< 1	4.0	< 1
	Crud	[ppb]	---	---	---	---	---	---	---	---	0.9	4.2	1.0	8.0	< 1
Tritium	Ion	[MBq/cm^3]	7.03	22.9	39.6	45.1	55.9	73.3	77.70	98.1	110	125	135	143	153
Na-24	Crud	[Bq/cm^3]	2.50E+02	2.00E+02	1.20E+02	2.10E+02	2.90E+02	1.70E+02	1.00E+02	1.40E+02	9.90E+02	1.40E+02	1.40E+02	1.90E+02	1.60E+02
Cr-51	Ion	[Bq/cm^3]	1.80E+02	2.30E+02	1.40E+02	3.90E+02	5.60E+02	1.50E+03	3.20E+02	1.40E+02	1.40E+03	1.60E+02	1.10E+02	8.50E+01	4.60E+01
	Crud	[Bq/cm^3]	---	3.30E+01	1.60E+01	5.70E+01	7.00E+01	4.70E+01	4.40E+01	1.80E+02	1.50E+01	2.20E+01	1.40E+01	7.50E+00	6.50E+00
Co-60	Ion	[Bq/cm^3]	1.90E+01	7.30E+01	7.40E+01	2.90E+02	7.90E+02	7.40E+02	4.50E+02	6.10E+02	5.40E+02	7.00E+02	7.30E+02	6.20E+02	4.40E+02
	Crud	[Bq/cm^3]	---	1.90E+00	4.10E+00	1.70E+01	5.80E+02	3.80E+02	1.50E+02	3.20E+02	3.10E+01	6.20E+01	7.10E+01	2.40E+01	3.60E+01

Table V.5-5 Results of Heavy Water Quality (2/2) (Average Water Quaity During Rated Power Operation)

Fuel/Operating Cycle			13th	14th	15th	16th	17th	18th	19th	20th	21st	22nd	23rd	24th	25th	26th
Periodic Inspection			7 th		8 th		9 th		10 th		11 th		12 th		13 th	14 th
Electric Conductivity		[μS/cm]	0.48	0.31	0.60	0.45	0.51	0.76	0.67	0.68	0.87	0.59	0.54	0.43	0.47	0.52
pD		[-]	6.97	7.35	6.09	6.88	6.68	5.88	6.18	6.14	5.95	6.58	6.56	6.41	6.61	6.58
Chlorine		[ppb]	2	2	5	5	3	6	5	7	5	8	6	4	4	2
Nitrous Acid Ion		[ppb]	7	7	3	3	11	20	18	15	29	14	14	17	12	11
Nitric Acid Ion		[ppb]	85	67	152	87	90	197	135	98	171	77	70	68	56	53
Sulfuric Acid Ion		[ppb]	5	17	8	4	1	3	3	9	4	3	3	3	6	3
Lithium Ion		[ppb]	26	24	23	29	30	33	36	36	39	36	24	23	28	29
B-10		[ppm]	2.97	2.42	2.48	2.45	2.57	2.16	2.28	2.38	2.74	3.17	2.61	2.38	2.86	3.48
SiO_2		[ppb]	38	38	53	42	40	5	5	8	< 1	8	34	26	25	14
SS		[ppm]	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1	< 0.1
IC		[ppm]	0.142	0.065	0.107	0.069	0.127	0.121	0.138	0.184	0.184	0.186	0.175	0.113	0.153	0.199
TOC		[ppm]	0.090	0.146	0.101	0.073	0.064	0.081	0.068	0.049	0.086	0.076	0.069	0.082	0.059	0.06
D_2O_2		[ppm]	6.05	6.08	3.57	3.90	3	5.6	7.2	8.2	8.6	9	7.9	8.6	9.4	9.4
Fe	Ion	[ppb]	1.5	< 1	1.7	1.1	3	< 1	< 1	< 1	< 1	< 1	< 1	< 1	2	< 1
	Crud	[ppb]	1.0	< 1	4.4	1.4	1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	2	< 1
Ni	Ion	[ppb]	9.9	1.3	2	3.1	1.3	3.5	2.3	2.9	2.7	3.7	2.7	2	< 1	< 1
	Crud	[ppb]	7.0	1.3	1	1.1	2	1	< 1	< 1	1	3	< 1	< 1	< 1	< 1
Co	Ion	[ppb]	2.3	< 1	1	1	< 1	< 1	1	< 1	< 1	1	4	1.5	3	< 1
	Crud	[ppb]	< 1	1	1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	1	< 1
Cr	Ion	[ppb]	3.4	1.3	1.1	1.1	1.3	1.5	2	1.7	2.5	1.3	2.5	1.5	2.8	< 1
	Crud	[ppb]	< 1	1	1	1	1	< 1	< 1	< 1	< 1	< 1	< 1	1	< 1	< 1
Cu	Ion	[ppb]	4.3	< 1	< 1	< 1										
	Crud	[ppb]	< 1	< 1	1.8	< 1										
Al	Ion	[ppb]	1.0	< 1	< 1	< 1	1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1
	Crud	[ppb]	1.0	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1	< 1
Tritium	Ion	[MBq/cm^3]	160	169	179	190	194	202	207	215	216	229	233	237	248	252
Na-24	Crud	[Bq/cm^3]	1.90E+02	1.80E+02	2.00E+02	2.20E+02	7.70E+02	1.60E+02	1.50E+02	1.50E+02	1.60E+02	1.90E+02	2.00E+02	2.20E+02	2.50E+02	2.50E+02
Cr-51	Ion	[Bq/cm^3]	7.80E+01	6.10E+01	4.60E+01	6.00E+01	3.10E+01	4.60E+00	6.80E+01	5.20E+01	4.40E+01	4.00E+01	4.70E+01	4.90E+01	4.30E+01	4.90E+01
	Crud	[Bq/cm^3]	6.30E+00	<	7.70E+00	9.01E+00	1.70E+00	3.90E+00	2.00E+01	8.50E-01	2.40E+00	3.70E+00	3.00E+00	3.50E+00	1.60E+00	2.80E+00
Co-60	Ion	[Bq/cm^3]	4.30E+02	3.50E+02	2.30E+02	2.20E+02	8.10E+01	2.00E+02	2.60E+02	2.60E+02	2.30E+02	2.40E+02	3.00E+02	3.30E+00	2.90E+02	2.90E+02
	Crud	[Bq/cm^3]	4.10E+01	2.50E+01	2.50E+01	1.90E+01	1.30E+00	1.40E+01	1.40E+00	1.80E+00	1.30E+01	1.20E+01	3.70E+00	5.70E+00	4.00E+00	6.40E+00

Table V.5-6 Water Circulation of Heavy Water Purification Resin Bed

No of Bed	Name of Bed	Service Period	Total Heavy Water Circulation Time (h)	Remarks
1	B	'78.10~' 79.2	159	·Intermittent water circulation
2	A	'79.3~' 79.5	77	
3	B	'79.3~' 79.6	42	
4	A	'79.7~' 79.11	83	
5	B	'79.12~' 80.5	91	
6	A	'80.5~' 80.11	81	
7	B	'81.10~' 82.9	126	
8	B	'83.1~' 84.2	209	·Low chlorinatd resin
9	A	'81.7~' 84.2	75	
10	A	'84.6~' 85.1	210	·Low chlorinatd resin
11	B	'84.12~' 85.7	169	·Extremely low chlorinatd resin, lowering heavy water temperature
12	A	'86.1~' 87.1	640	·Extremely low chlorinatd resin, lowering heavy water temperature,resin ratio
13	A	'87.4~' 88.4	873	·Extremely low chlorinatd resin, lowering heavy water temperature,resin ratio
14	A	'88.7~' 89.7	1215	·Extremely low chlorinatd resin, lowering heavy water temperature,resin ratio
15	B	'86.1~' 89.7	401	·Used to purify heavy water during reactor shutdown
16	A	'89.11~' 90.5	900	·Extremely low chlorinated resin, lowering heavy water temperature
17	A	'90.6~' 90.12	563	·Extremely low chlorinated resin, lowering heavy water temperature

Table V.5-7 Calculation of Tritium Formation

Reaction of Formation	Formation of Tritium (kCi/y)
$^{2}H(n, \gamma)^{3}H$	200
$^{3}He(n, p)^{3}H$	$\leqq$ 1
$^{7}Li(n.n, \alpha)^{3}H$	$\leqq$ 0.01
$^{10}B(n, 2\alpha)^{3}H$	$\leqq$ 0.01

Calculation Conditions:

Thermal Neutron Flux : 5.6×10^{13} ($1/cm^2/s$)

Fast Neutron Flux : 1.1×10^{13} ($1/cm^2/s$)

Operatin Hours : Rated Power Operaton in 365 days

Table V.5-8 Evaluated Co-60 Deposited on Inner Surface of Heavy Water Dump Tank

		Before clean-up	After clean-up
Measured Value	① Surface dose rate of heavy water dump tank (mR/h)	26.0	15.8
	② Co-60 concentration in heavy water (μCi/cc)	1.9×10^{-2}	9.7×10^{-4}
Calculated Value	③ Dose contributed by Co-60 in heavy water [②x566 mR/h/μCi/cc]	10.8	0.5
	④ Dose contributed by Co-60 deposited inner surface of heavy water dump tank (mR/h) [①-③]	15.2	15.3
	⑤ Co-60 deposited on inner surface of heavy water dump tank [④/64.6 mR/h/μCi/cm^2] (μCi/cm^2)	0.235	0.237

Table V.5-9 Change of Co-60 Deposited on Inner Surface of Heavy Water Dump Tank

	'86 Scheduled Shutdown	6th Periodic Inspection	'87 Scheduled Shutdown	7th Periodic Inspection
Surface Dose Rate of Heavy Water dump tank [mR/h]	21.0 12.5	23.0 12.0	29.0 14.0	26.0 15.8
Co-60 Concentration in Heavy Water (total) [μCi/cc]	2.4×10^{-2} 5.6×10^{-4}	2.3×10^{-2} 5.3×10^{-4}	2.0×10^{-2} 1.1×10^{-4}	1.9×10^{-2} 9.7×10^{-4}
Calculated Co-60 Deposited on Inner Surface of Heavy Water Dump Tank [μCi/cm^2]	0.11 0.19	0.15 0.18	0.27 0.21	0.24 0.24

upper row : before clean-up
lower row : after clean-up

Table V.5-10 Helium Purification Operation Mode

Purification System Operation Mode	Processing Rate (m^3/h)	Features
Dehumidifying Operation Mode	30	Dehumidify helium flowing to helium purifier
Dehumidifying and Decarbonating Operation Mode	10	Dehumidify and decarbonate helium flowing to helium purifier

Table V.5-11 Maximum Excess Deuterium Concentration in Operating Cycles

Operating Cycle	Excess Deuterium Concentration (%)	Operating Cycle	Excess Deuterium Concentration (%)
1	2.18	8	3.49
2	2.72	9	1.45
3	1.74	10	1.48
4	1.44	11	2.44
5	0.80	12	1.5
6	1.73	13	1.25
7	2.94	14	1.59

Table V.5-12 G Value of Radical Formed by Radiolysis of Water

Radiation	LET (eV/Å)	G Value (number/100eV)						
		e_{aq}	H^+	H_2O_2	OH	HO_2	H	H_2
γ	0.1	2.7	2.7	0.61	2.86	0.03	0.61	0.43
Fast Neutron	4.0	0.93	0.93	0.99	1.09	0.04	0.50	0.88
$^{10}B(n, \alpha)^7Li$	24	0.04	0.04	1.3	0.10	0.30	0.16	1.7

Table V.5-13 Radiolysis Reaction of Water

No	Reaction Equation	Rate Constant ($dm^3\ mole^{-1}\ s^{-1}$ at 25℃)	Activation Energy (k Cal/mol)
1	$e^- + O_2 = O_2^-$	2.0×10^{10}	3.0
2	$e^- + H_2O_2 = OH + OH^-$	1.2×10^{10}	3.0
3	$H + O_2 = HO_2$	2.0×10^{10}	3.0
4	$OH + H_2 = H + H_2O$	1.6×10^{7}	4.6
5	$OH + H_2O_2 = HO_2 + H_2O$	1.14×10^{7}	4.5
6	$HO_2 + HO_2 = H_2O_2 + O_2$	2.7×10^{6}	4.5
7	$O_2^- + O_2^- = HO_2^- + O_2$	1.7×10^{7}	3.0
8	$OH + HO_2 = H_2O + O_2$	1.2×10^{10}	3.0
9	$OH + OH = H_2O_2$	5.5×10^{9}	4.5
10	$H + H_2O_2 = OH + H_2O$	9.0×10^{7}	3.0
11	$e^- + H^+ = H$	2.2×10^{10}	3.0
12	$e^- + HO_2 = HO_2^-$	2.0×10^{10}	3.0
13	$H + HO_2 = H_2O_2$	2.0×10^{10}	3.0
14	$e^- + e^- = H_2 + 2\ OH^-$	5.0×10^{9}	3.0
15	$e^- + OH = O_2$	3.0×10^{10}	3.0
16	$H + H = H_2$	1.0×10^{10}	3.0
17	$H + OH = H_2O$	2.0×10^{10}	3.0
18	$e^- + H = H_2 + OH^-$	2.5×10^{10}	3.0
19	$OH + O_2^- = O_2 + OH^-$	1.2×10^{10}	3.0
20	$H + OH^- = e^- + H_2O$	2.1×10^{7}	4.5
21	$e^- + HO_2^- = OH + 2\ OH^-$	3.5×10^{9}	3.0
22	$H^+ + OH^- = H_2O$	1.44×10^{11}	3.0
23	$H^+ + O_2^- = HO_2$	5.0×10^{10}	3.0
24	$HO_2 + O_2^- = O_2 + HO_2^-$	1.5×10^{7}	4.5
25	$H + O_2^- = HO_2$	2.0×10^{10}	3.0
26	$e^- + O_2^- = HO_2^- + H_2O$	1.0×10^{10}	4.5
27	$OH^- + H_2O_2 = HO_2^- + H_2O$	1.8×10^{8}	4.5
28	$e^- + H_2O = H + OH^-$	1.6×10^{1}	3.0
29	$HO_2^- + H_2O = H_2O_2 + OH^-$	1.0×10^{4}	4.5
30	$HO_2 = H^+ + O_2^-$	8.0×10^{5}	3.0
31	$H_2O = H^+ + OH^-$	* $K_{22} \cdot Kw/[H_2O]$	
32	$OH + HO_2^- = OH^- + HO_2$	8.0×10^{9}	3.0

* : Equilibrium with reaction 22

Table V.5-14 Radiation Chemical Reaction in Impurities in Heavy Water

No	Reaction Equation	Rate Constant ($dm^3\ mole^{-1}\ s^{-1}$ at 25℃)	Activation Energy (k Cal/mol)
1	$NH_3 + OH = NH_2 + H_2O$	1.0×10^8	3.0
2	$NH_2 + O_2 = NO + H_2O$	3.0×10^8	3.0
3	$NO + OH = NO_2^- + H^+$	2.0×10^{10}	3.0
4	$NO_2^- + OH = NO_2 + OH^-$	1.0×10^{10}	3.0
5	$NO_2^- + H = NO + OH^-$	1.5×10^9	3.0
6	$NO_2^- + e^- = NO + 2\ OH^-$	4.3×10^9	3.0
7	$2\ NO_2 = 2\ H^+ + NO_3^- + NO_2^-$	1.0×10^{10}	3.0
8	$NO_3^- + e^- = NO_3^{2-}$	1.1×10^{10}	3.0
9	$NO_3^{2-} + H_2O = NO_2 + 2\ OH^-$	1.0×10^{10}	3.0
10	$NO_3^- + H = NO_2 + OH^-$	2.4×10^7	4.5
11	$CO_2 + e^- = COO^-$	7.7×10^9	3.0
12	$COO^- + O_2 = CO_2 + O_2^-$	2.0×10^9	3.0
13	$COO^- + H_2O_2 = CO_2 + OH + OH^-$	7.0×10^5	4.5
14	$HCO_3^- + OH = CO_3^- + H_2O$	1.1×10^8	(at 60℃)
15	$CO_3^- + O_2^- = CO_3^{2-} + O_2$	1.3×10^8	3.0
16	$CO_3^- + H_2O_2 = CO_3^{2-} + HO_2^- + H^+$	8.0×10^5	4.5
17	$CO_3^- + HO_2^- = CO_3^{2-} + O_2^- + H^+$	5.6×10^7	4.5
18	$CO_3^- + CO_3^- = 2\ CO_2 + HO_2^- + OH^-$	1.25×10^7	4.5
19	$COO^- + HCO_3^- = HCOO + CO_3^-$	2.0×10^3	4.5
20	$COO^- + CO_3^- = CO_2 + CO_3^{2-}$	5.0×10^7	4.5
21	$HCOO + CO_3^- = HCO_3^- + COO^-$	1.5×10^5	4.5
22	$HCOO + OH = COO^- + H_2O$	2.9×10^9	3.0
23	$HCOO + H = COO^- + H_2$	4.5×10^8	3.0
24	$CO_3^{2-} + OH = CO_3^- + OH^-$	4.2×10^8	3.0
25	$CO_2 + H_2O = HCO_3^- + H^+$	2.3×10^1	(at 60℃)
26	$HCO_3^- + H^+ = CO_2 + H_2O$	1.0×10^{10}	(at 60℃)
27	$HCO_3^- + OH^- = CO_3^{2-} + H_2O$	1.0×10^9	(at 60℃)
28	$CO_3^{2-} + H^+ = CO_2 + OH^-$	2.86×10^4	(at 60℃)
29	$LiOH = Li^+ + OH^-$	1.0×10^2	
30	$HNO_3 = H^+ + NO_3^-$	1.0×10^2	

Table V.5-15 Standard Schedule of Resin Exchange of Heavy Water Purification System

	Work Procedure	1st day	2nd day	3rd day	4th day	5th day	6th day	7th day	8th day
Preparation	1. Work preparation								
	Isolation of D_2O purification column								
	Measurement of water quality of each tank								
	2. D_2O blowing of D_2O purification column								
	3. D_2O filling up in D_2O purification column								
Dedeuteration	4. Transferring used resins to deuteration column								
	5. D_2O blowing of used resins by pressurized He								
	6. H_2O filling up in deuteration column								
	7. Dedeuteration of used resins								
	8. Transferring used resins to storage tank (disposal)								
	9. Measurement of water quality in each tank								
	10. Cleaning new resins (in resin cleaning column)								
	11. Transferring new resins to deuteration column								
	12. Scrabbling new resins								
	13. H_2O blowing of new resins								
	14. Blowing of deuteration column upper piping								
	15. Blowing of deuteration column lower piping								
Deuteration	16. Deuteration of new resins								
	17. Transferring new resins to D_2O purification column								
	18. Mixing new resins by He								
	19. D_2O filling up in D_2O purification column								
	20. Return to system								
	21. Measurement of water quality in each tank								
Sampling Data		Each Tank			Dedeute-ration	Each Tank	Deute-ration	Each Tank	

Reference

[V-1] IAEA Safety Series No.50-C-O(Rev. 1), Code on the safety of nuclear power plants: Operation, 1988

[V-2] IAEA Safety Series No.50-SG-O5, Radiation protection during operation of nuclear power prants, 1983

[V-3] IAEA Safety Series No.50-SG-O9, Management of nuclear power plants for safe operation, 1984

[V-4] IAEA Safety Series No.50-C/SG-Q Q13, Quality assurence in operation, 1996

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VI. Waste Management [VI-1] [VI-2] [VI-3] [VI-4] [VI-5]

1. Outline of Waste Management

The radioactive waste management consists of a series of activities to treat radioactive gaseous, liquid and solid wastes generated at nuclear power plants in an appropriate manner according to their physical forms.

Radionuclides generated at the reactor facility are roughly divided into 2 types; fission products directly generated from nuclear fission of fuels and activation products generated in reactor structures. Most of the radionuclides are retained inside the fuel elements or the reactor facility while a small amount is discharged into the environment in gaseous or liquid form through respective treatment facilities.

For radioactive gaseous and liquid wastes generated with operation of "Fugen", it is essential to control the concentration below the discharge control target value prescribed in the reactor safety regulations and the limits stipulated by the laws and regulations by taking actions to reduce the concentration and the volume of the radioactive materials according to its form. In addition, it is required to control the concentration as low as reasonably achievable. For the radioactive solid wastes, it is also necessary to take actions to reduce the amount of radioactive materials generated and stored.

Therefore, in "Fugen", the collection of gaseous tritium, the evaporation and concentration treatment of the equipment drain waste liquid and floor drain waste liquid, the introduction of the filter pre-coated with powdered resins, the introduction of hollow yarn membrane filters, the introduction of dry-cleaning system, etc. have been implemented in order to reduce the discharged radioactivity. In FY 1989, the miscellaneous solid waste incineration system was installed to reduce the volume of combustible miscellaneous solid wastes, resulting in reduction of the volume for storage. At the same time, development of the techniques for treatment and disposal of solid wastes, for example, the spent ion exchange resin, is being made for the stabilization and volume reduction of the solid wastes.

2. Waste Management Policy

The disposal of radioactive wastes generated in the plant is performed based on the laws, including the "Regulations Concerning the Installations, Operation, etc. of Nuclear Reactors for Purposes of Testing and Research", and the "Safety Review Guideline Stipulated by the Nuclear Safety Commission".

The radioactive wastes are classified into gaseous wastes, liquid wastes and solid wastes based on its physical form; they are stored or disposed after the appropriate treatment.

2.1 Gaseous Wastes

The gaseous wastes are discharged into the atmosphere through the main stack or stack of the waste treatment building after being treated in the gaseous waste treatment system. In this case, the radioactive material concentration in the off-gas is to be reduced as much as possible by the filtering, radioactivity decay due to time, dilution by large amount of air, etc. At the same time, the radioactive material concentration in the air outside the supervised area shall be controlled not to exceed the concentration limits stipulated by the law by monitoring the radioactive material concentration in the off-gas using the main stack monitor and the waste treatment building stack monitor.

2.2 Liquid Wastes

The liquid wastes shall be, in principle, reused after treated in the liquid waste treatment system. When the liquid wastes is discharged into the environment, the radioactive material concentration in the waste water shall be reduced as much as possible by the filtering, evaporating, removing with ion exchange resins, etc., radioactivity decay due to time elapsed, diluting with a large amount of water, etc. When they are discharged, the radioactive material concentration in the water in the outer boundary of the supervised area shall be controlled not to exceed the concentration limit stipulated by the law by means, such as monitoring the radioactive material concentration in the waste water using the discharge pond monitor in the condenser cooling water discharge canal.

2.3 Solid Wastes

The solid wastes shall be stored in containers including drum cans after their volume is reduced by incineration, compression, etc., or shall be immobilized to be stored and maintained in the solid waste storage facility at the site. The solid

wastes with the residual decay heat, such as spent control rods, etc. shall be stored in the fuel storage pool until their radioactivity is fully decayed.

3. Organizational System

3.1 Waste Treatment

The waste treatment work is mainly performed by Power Generating Section.

As for the gaseous waste treatment, the air ejector off-gas, the turbine gland steam system off-gas, the carbon dioxide gas system off-gas, the helium system off-gas, etc. are processed in the noble gas hold-up system, decay tube filter, decay tube, etc. As the generation amount of such wastes changes according to the reactor operation status, the operation and treatment status monitoring of the treatment system are performed by the operation shift of Power Generating Section.

As for the liquid waste treatment, the equipment drain waste liquid, the floor drain waste liquid, regenerant waste liquid and the laundry drain are processed with the filtering equipment, the demineralizer, the evaporation / concentration equipment, etc. During the liquid waste treatment, the waste management personnel in Power Generating Section confirms the liquid waste generation status, asks for the measuring of the liquid waste properties, determines the treatment method, evaluates the treatment results and prepares the treatment facility operation programs. The operation of the liquid waste treatment system is consigned to Tsuruga Nuclear Service Co.

The solid waste treatment includes the asphalt solidification of the concentrated waste liquids, the storage and maintenance of spent ion exchange resins and filter sludge, and the dry active waste volume reduction by incineration or by compression. During the solid waste treatment, the waste management personnel in Power Generating Section confirms the solid waste generation status, asks for the measuring of the solid waste properties, determines the treatment method, evaluates and reports on the treatment results, performs the stored waste management, and prepares the treatment facility operation programs. The operation of the treatment system and the charging, storage, transportation of wastes are consigned to Tsuruga Nuclear Service Co.

The waste management personnel in Power Generating Section consists of two employees, and 24 employees of Tsuruga Nuclear Service Co. are engaged in the waste management.

3.2 Discharge Control

The discharge control work is performed mainly by Safety and Chemistry Section (one administrator and two management / development personnel).

As for the discharge control of gaseous wastes, the radioactive material

concentration of the air ejector off-gas, the turbine gland steam system off-gas, the carbon dioxide gas system off-gas, the helium system off-gas, etc. are periodically measured at their discharging sources, and the radioactive material concentration at the stacks are also periodically measured in order to calculate the discharged radioactivity and to evaluate the radioactive material concentration at the boundary of the supervised area. The evaluation results are summed up and reported to the agencies or authorities concerned.

As for the discharge control of liquid wastes, when the water processed in the liquid waste treatment system is discharged, the water is stored in the liquid waste sample tank of the equipment drain treatment system, the floor drain sample tank of the floor drain treatment system, the laundry drain sample tank of the laundry drain treatment system, etc.; the radioactive material concentration in the processed water is measured per batch to calculate the discharged radioactivity and to evaluate the radioactive material concentration at the boundary of the supervised area. The evaluation results are summed up and reported to the agencies or authorities concerned.

As for the solid wastes, the types, amount, etc. of the waste disposed in the waste facility are notified by Power Generating Section, and their physical properties and radioactive material concentration are evaluated.

Among the works related to the discharge control of the gaseous and liquid wastes and the control of the solid wastes, the sampling measurement, periodical calibration of the dosage meters and chemical analyzer, etc. are consigned to Tsuruga Nuclear Service Co. (Two employees assigned).

In addition, the maintenance section is in charge of the maintenance and control of the radiation monitoring systems such as main stack monitors and discharge pond monitors, making the periodical checks and calibration in order to keep the integrity of the system.

4. Waste Management Method [VI-6]

The radioactive wastes generated in "Fugen" is managed after classified into gaseous wastes, liquid wastes and solid wastes.

4.1 Gaseous Wastes

(1) Type and Releasing Path

Fig. VI.4-1 shows the types and releasing paths of the gaseous wastes, and each monitoring point. The gaseous wastes are classified as described in the following according to its generation sources; it is diluted and discharged into the atmosphere through the main stack or the waste treatment building stack (or through the heavy water purification building stack in the heavy water upgrader building) after they are filtered or their radioactivity is decayed due to time delay depending on its property. Continuous monitoring is performed at the generation point of each waste and at the stacks.

(a) Air Ejector Off-gas

The radioactive materials potentially contained in the air ejector off-gas are the fission products which leak if the fuel cladding should fail. The radioactive noble gases are discharged into the atmosphere through the main stack after they are adequately hold up (usually for about 27 days for Xenon, and about 40 hours or more for Krypton) in the activated charcoal noble gas hold-up system to fully decay the radioactivity. The actual values of the radioactive noble gases discharging ratio (equivalent to the value after 30-minute decay) are kept at the sufficiently low level, for the fuel cladding's integrity is maintained.

(b) Turbine Gland Steam System Off-gas

The amount of the radioactive noble gases contained in the turbine gland steam system off-gas are about 1/1000 of that contained in the air ejector system off-gas. The radioactive noble gases are discharged into the atmosphere through the main stack after they go through the decay pipe filter and kept in the decay pipe for about 15 minutes or more.

(c) Startup Vacuum Pump Off-gas

Thestartup vacuum pump off-gas is discharged into the atmosphere through the main stack after going through the decay pipe or the activated charcoal noble gas hold-up system.

(d) Ventilation System Off-gas

The off-gas of the reactor building ventilation system, the reactor auxiliary

building ventilation system, the waste treatment facility ventilation system, the storage room ventilation system, the turbine building ventilation system and the fuel building ventilation system is discharged into the atmosphere though the main stack after going through the filter.

As for the ventilation system off-gas of the waste treatment building, the off-gas from the ventilation system for the dry active waste incineration facility room, etc. is discharged from the waste treatment building stack after gong through the filter; on the other hand, the off-gas from the ventilation system for the spent ion exchange resin storage tank room, etc. is discharged into the atmosphere after going through the filter, for the radioactive materials contained are similar to those in the ventilation system off-gas from the storage tank room of the main building.

The heavy water upgrader ventilation system off-gas is discharged into the atmosphere through the heavy water upgrader building stack after going through the filter.

(e) Carbon Dioxide Gas System Off-gas

In the carbon dioxide gas system off-gas, when the reactor is shut down for a long time, such as the periodic inspection, etc., the small amount of Argon-41, etc. is contained, for the air mixed into the carbon dioxide gas system is activated during the reactor startup.

Such radioactive materials are discharged into the atmosphere through the main stack after going through the decay pipe filter and the decay pipe.

(f) Helium System Off-gas

The helium system off-gas is exhausted as the helium volume in the calandria tank is expanded during the reactor startup. The radioactive materials including tritium in the off-gas are processed in the dehumidifier in the helium purification system, the decay pipe filter and the decay pipe, and then discharged into the atmosphere from the main stack.

(g) Miscellaneous Solid Waste Incineration System Off-gas

The miscellaneous solid waste incineration system off-gas is discharged into the atmosphere from the main stack of the waste treatment building after it goes through the primary and the secondary filters and the high-performance filter of the off-gas treatment equipment of the miscellaneous solid waste incineration system to reduce the radioactive material concentration in the off-gas to $1/10^5$ or less.

(2) Discharge Control Method

When the gaseous wastes are discharged into the atmosphere, the radioactive material concentration is measured at the main stack to ensure that the annual dose equivalent shall not exceed the value specified in the "Specifications on the Permissible Exposure Dose, etc. Based on the Regulations on the Rules for the Reactor Installation, Operation, etc.". At the same time, the radioactive material concentration in the off-gas has been continuously measured using the main stack monitor, and it has been monitored in the control room, etc. to ensure that there is no anomaly.

Meanwhile, the discharge control target values of the radioactive material from the main stack were specified, based on the "Guidelines on the Dosage Target Value Around the Light Water Reactor Facilities for Power Generation" (1975, by the Atomic Energy Commission), as follows: for the radioactive noble gases, 5.1 x 10^{14} Bq or less per year; for iodine-131, 2.7 x 10^{10} Bq or less per year; for tritium, 1.8 x 10^{13} Bq or less. Each of the radioactive material concentrations has been controlled not to exceed these values. Furthermore, as for the discharge control target values from the waste treatment building stack, they were specified as 7.4 x 10^{6} Bq or less per year for iodine-131, and as 3.7 x 10^{11} Bq or less per year for tritium. They have been controlled not to exceed these values.

The radioactive material discharge from the heavy water upgrader building stack has been controlled based on the "Law on the Radiation Damage Prevention".

Table VI.4-1 shows the outline of the radiation monitors concerning the discharge contorol. Table VI.4-2 shows the nuclides to be measured, detection low limits, detection frequencies and detectors concerning the discharge radioactive materials.

4.2 Liquid Wastes

(1) Types and Releasing Paths

Fig.VI.4-2 shows the types and releasing paths of the liquid wastes, and each monitoring point. The liquid wastes are classified, according to its generation source and physical properties, into the equipment drain waste liquid, the floor drain waste liquid, the regenerant waste liquid, the laundry drain, the heavy water drain waste liquid and various drain waste liquids in the heavy water upgrader building.

Continuous monitoring is performed at the generation sources of each waste (tank outlets) and at the cooling water discharge tail ends when each waste is

discharged.

(a) Equipment Drain Waste Liquid

The equipment drain waste liquid consists of the drain water from each system, the waste water of the sampling lines, etc. Its chemical purity is relatively high, and the actual value of the radioactive material concentration is 2×10^{1} Bq / cm^3 (design value: 1.5×10^{3} Bq / cm^3).

The waste liquids were collected in the liquid waste collection tank of the liquid waste treatment system, and then filtered and demineralized until about FY 1981. From FY 1982 to 1989, in order to reduce the discharged radioactivity, the operation was changed: more than 90% of the generated waste liquid was evaporated and concentrated, and then demineralized (also filtered as necessary). After FY 1989, in order to enhance the liquid waste treatment capacity of the liquid waste treatment system and to reduce the generation amount of the filter sludge, the hollow yarn membrane filter was adopted in the liquid waste filtering system (in August, 1989) in addition to the original filtering and demineralization.

The processed water, whose radioactive material concentration and chemical purity are measured in the waste liquid sampling tank, is collected in the condensate storage tank or discharged if it satisfies the water chemistry standard of the condensate storage tank water; if it does not, it is again evaporated and concentrated, filtered and demineralized.

(b) Floor Drain Waste Liquid

The floor drain waste liquid is the collected package cooler water drain, etc. of each building; its chemical purity is low, and its radioactive material concentration is relatively low. The generated amount of the floor drain waste liquids are not so large; however, once it is generated, it is collected in the floor drain collection tank, and processed in the same way as the regenerant waste liquid or the equipment drain waste liquid according to the waste liquid physical properties.

(c) Regenerant Waste Liquid

The regenerant waste liquid mainly consists of the regenerant waste liquid of the ion exchange resin of the demineralizer of the condensate demineralizer system and the demineralizer of the liquid waste treatment system, the drain waste liquid of the chemical analysis room, etc. As it usually contains salts, and its chemical purity is low, it is processed in the equipment drain treatment system after it is evaporated and concentrated. The actual concentration of

the radioactive materials is about 2 x 10^0 Bq / cm^3 (deign value: abut 2 x 10^3 Bq/ cm^3). Before FY 1989, besides such waste liquid, the equipment drain waste liquid and the floor drain waste liquid were also processed in the regenerant waste liquid system; however, as the hollow yarn membrane filter was introduced, they can be processed in the equipment drain treatment system.

(d) Laundry Drain Waste Liquid

The laundry drain is the waste liquid generated when the protective clothes, etc. used in the controlled area are washed, etc. The chemical purity of the waste liquid is extremely low (the actual value of the insoluble impurity concentration: about 100 ppm), and the actual value of the radioactive material concentration is about 2 x 10^{-2} Bq /cm^3 (design value: about 4 x 10^0 Bq /cm^3).

Such waste liquid is collected to the laundry drain tank, and filtered when necessary (the filtering is not normally performed, for the insoluble impurity concentration is high). After it is confirmed that the specified value on discharging is fully satisfied, the waste liquid is diluted by the condenser cooling water and discharged into the sea.

(e) Heavy Water Drain Waste Liquid

The heavy water drain waste liquid includes the waste liquid drained from the heavy water system equipment during the periodic inspection, the waste liquid generated when the heavy water purification system ion exchange resins is deuterated or dedeuterated. Among such waste liquids, the one with relatively high heavy water concentration is stored in the degraded heavy water storage tank and reused after being upgraded by the heavy water upgrader, while the heavy water with extremely low concentration collected in the emergency gas treatment system * is discharged into the environment.

* The system which captures the steam containing tritium transferred to the air by the dehumidifier and collects it as the liquid

(f) Heavy Water Upgrader Building Drain Waste Liquid

The waste liquid generated in the heavy water upgrader building is collected in the acceptance tank in the building. After its radioactive material concentration is measured, it is diluted in the dilution tank in the building so that the radioactive isotope concentration limit concerning the waste water shall not be exceeded, and then discharged into the sea from the condenser cooling water discharge canal of the reactor facility. No other radioactive

nuclides than tritium have been detected in the waste liquid.

(2) Discharge Control Method

The liquid wastes are, in principle, not to be discharged except for the wastes with extremely low radioactive material concentration.

When the liquid wastes are discharged from the rector facility to the environment, they are stored in the waste liquid sampling tank to measure the radioactive material concentration. The radioactive material concentration in the waste water discharged from the condenser cooling water discharge canal into the sea has been controlled not to exceed the concentration limit outside the supervised area specified in the "Specifications on the Permissible Exposure Dose, etc. based on the Regulations on the Rules for Reactor Installation, Operation, etc." At the same time, the radioactive material concentration in the waste water has been monitored by the liquid waste treatment system waste water monitor and the condenser cooling water discharge canal discharge pond monitor, and it is confirmed that the concentration is low enough.

Meanwhile, the discharge contorl target values of the radioactive materials (excluding tritium) in the liquid wastes were specified, based on the "Guidelines on the Dosage Target Value Around the Light-Water Reactor Facilities for Power Generation" (1975, by the Atomic Energy Commission), to be 7.4×10^{9} Bq or less per year (3.7×10^{10} Bq or less per year until FY 1988); for tritium, the target value was specifies to be 1.1×10^{13} Bq or less. They have been controlled not to exceed these target values. Furthermore, as for the liquid waste discharge from the heavy water upgrader building is controlled based on the Law of Radiation Damage Prevention. Table VI.4-1 shows the outline of the radiation monitors concerning the discharge contorol. Table VI.4-3 shows the nuclides of discharge radioactive material to be measured, detection low limits, detection frequencies and detectors.

4.3 Solid Wastes

(1) Types and Treatment Methods

Fig. VI. 4-3 shows the types, treatment methods, etc. of the solid wastes.

The solid wastes are classified as follows, based on their generation sources and physical properties. As to the solid wastes, except for the concentrated waste liquids, considering the radioactivity decay, the improvement of the future stabilization and volume reduction technology, etc., they have been stored and maintained in the storage tank or the solid waste storage facility of the solid

waste treatment system, without making them the final form.

(a) Spent Ion Exchange Resins

The spent ion exchange resins are divided into the granular waste resins and the powdered waste resins. The granular waste resins include the reactor coolant purification system demineralized waste resins, the heavy water system demineralized waste resins, the condenser demineralizer system demineralizer waste resins and the liquid waste demineralizer system demineralizer waste resins. These waste resins have been separately stored in the granular waste resin storage tank, etc. so that they can be stabilized or reduced in volume in the future, becaue their generation sources and their internal radioactive concentration are different.

The powdered waste resins are the pool water cooling and circulation system demineralizer waste resins, and they have been stored in the powdered waste resin storage tank.

In October, 1989, granular waste resin storage tanks (tank capacity: 50 m^3 x 5 tanks (conventional tank capacity: 55 m^3 x 2 tanks)) and a powdered waste resin storage tank (tank capacity: 50 m^3 x 1 (conventional tank capacity: 20 m^3 x 1)) were newly constructed.

(b) Filter Sludge

The filter sludge is generated from the liquid waste treatment system filtering equipment. The sludge is made of diatom earth until FY 1985, the powdered resins from FY 1985 to 1989, and the hollow yarn membrane filter back washing sludge after FY 1989. The filter sludge of the diatom earth and the powdered resins (including back washing sludge) are stored in two storage tanks separately, for their properties are different.

(c) Concentrated Waste Liquid

The concentrated waste liquid is generated as the evaporation residue of the regenerant waste liquid treatment system evaporation and concentration equipment of the liquid waste treatment system. It is temporarily stored in the concentrated waste liquid storage tanks (tank capacity: 30 m^3 x 3 tanks). The concentrated waste liquid in the concentrated waste liquid storage tanks is stored in the solid waste storage facility after it is asphalt solidified.

(d) Miscellaneous Solid Wastes (active)

The miscellaneous solid wastes have been charged in the 200 l drum cans or iron boxes (box pallets) equivalent to four drum cans to be stored in the solid waste storage facility, classifying into combustible waste and incombustible

waste assuming the incineration and volume reduction treatment. The incineration and volume reduction treatment have been performed for the combustible active solid waste since October, 1989. The miscellaneous solid wastes generated in the heavy water upgrader building has been treated in the same way as the ones in the reactor facility.

(2) Management method

The solid wastes are stored in various storage tanks in the reactor auxiliary building and the waste treatment building, the first and the second solid waste storage facilities, the fuel pools, etc., considering the generation sources, physical properties, etc. Meanwhile, from the viewpoint of managing the shielding classification of the storage facility shielding deign and the stored radioactivity, the internal radioactivity of the spent ion exchange resins and the filter sludge is calculated from the water chemistry at each demineralizer inlet and the results of the demineralizer operation: the surface dosage equivalent rate of the drum cans, etc. is measured for the miscellaneous solid wastes to calculate the internal radioactivity.

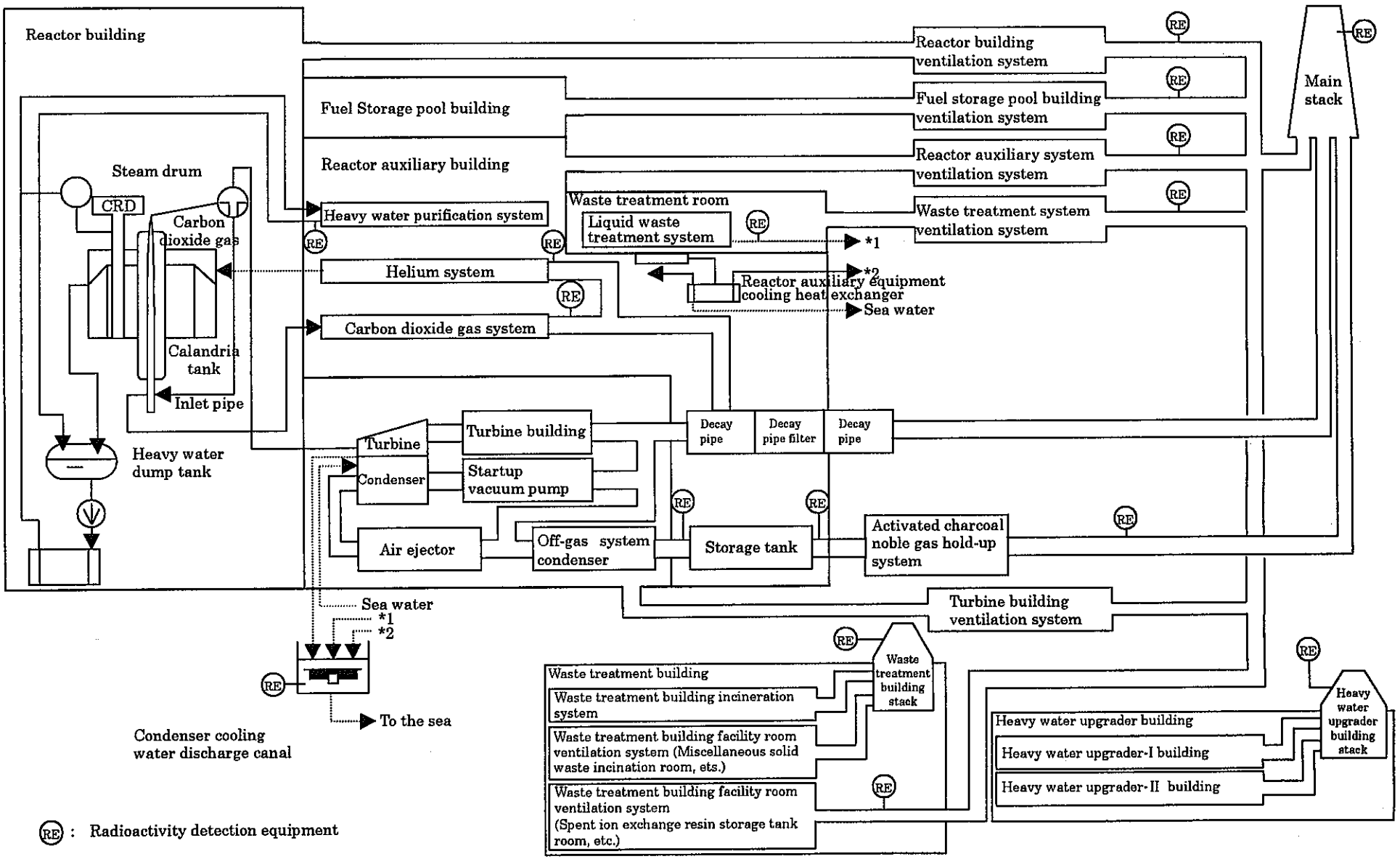


Fig. VI 4-1 Types and Release Paths of Gaseous Waste, and Monitoring Points

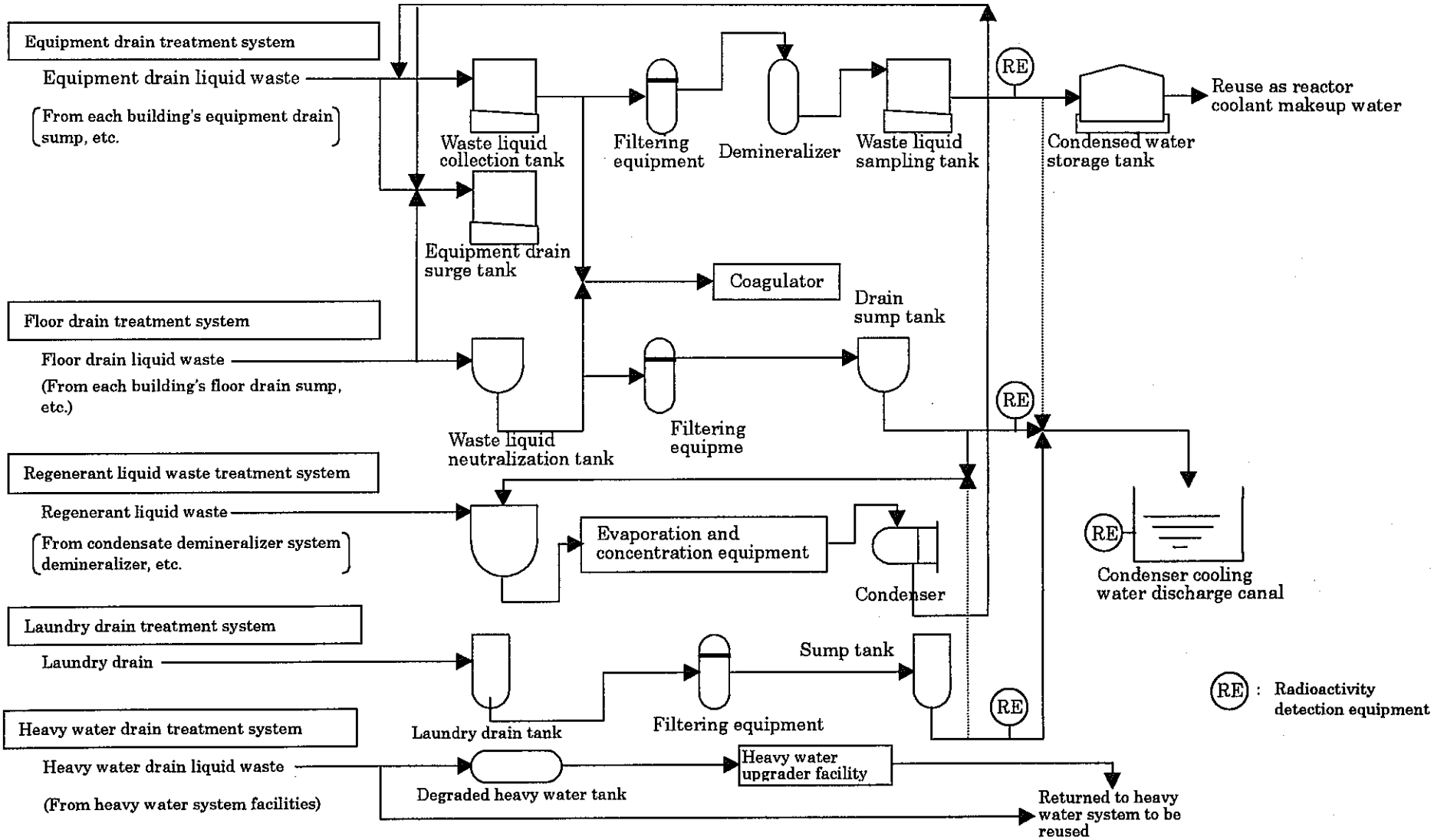


Fig. VI 4-2 Types and Release Paths of Liquid Wastes and Monitoring

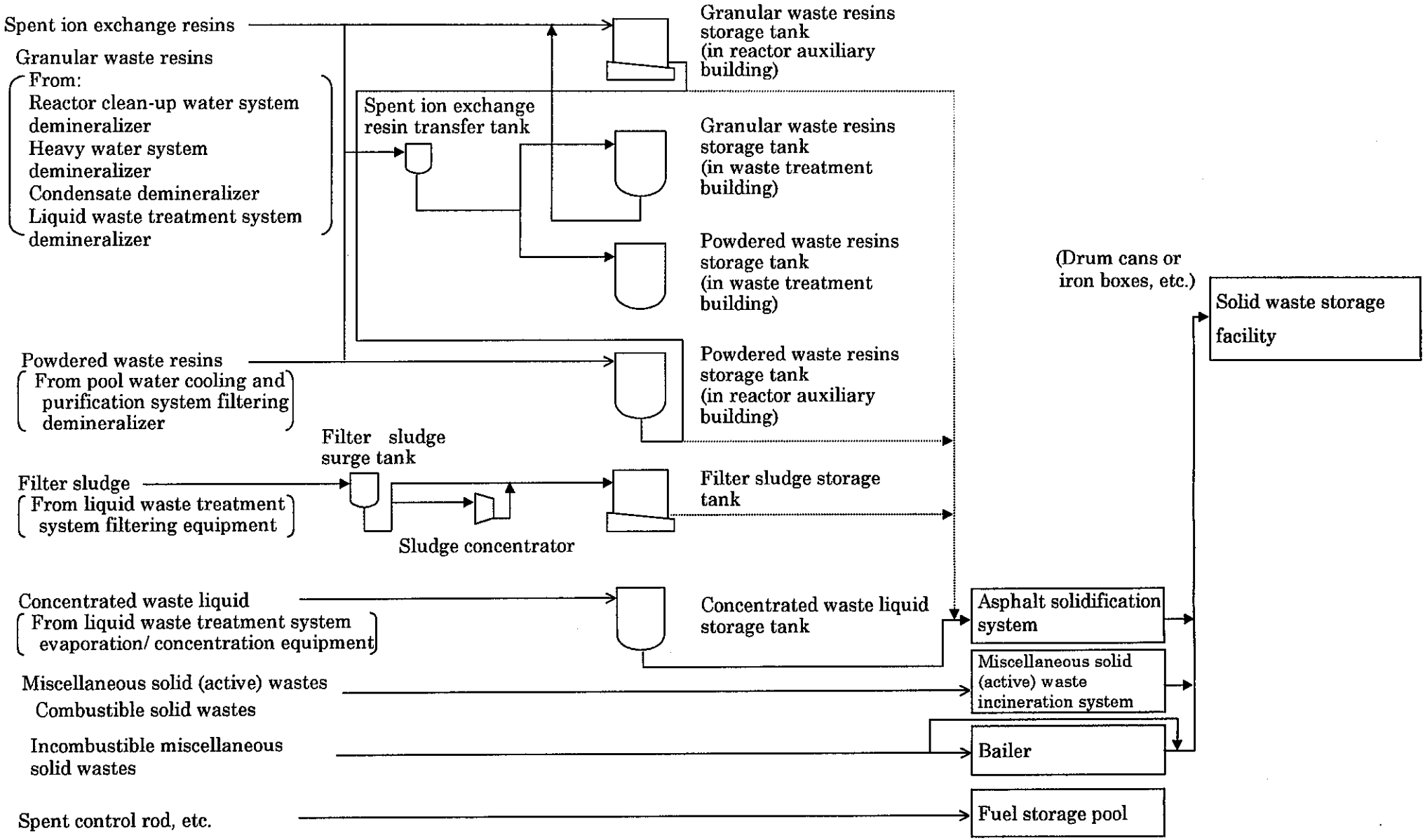


Fig. VI.4-3 Solid Waste Types and Treatment Methods, etc.

Table VI.4-1 Outline of Radiation Monitoring System

	Monitor	Installed Position	Quan-tity	Detector	Scope of Measurement	Objective	Basis for Alarm Setup
1	Main steam pipe monitor	Immediately after isolation valves inside/outside main steam tunnel	4	Ionization chamber	1×10^{-5} ~1×10^{0} (Sv/h)	To monitor the mass of fission products leaking from fuels.	At 3 times and 6 times normal operation value, "high" and "high high" alarms are set respectively. On "high high" alarm actuation, the main steam isolation valve is "closed".
2	Gaseous waste treatment system off-gas monitor	・Attenuation tube filter inlet piping ・Noble gas holdup inlet/outlet piping	3	Scintillation	1×10^{-2} ~1×10^{-6} (Bq/cm^3)	To monitor the concentration of radioactive materials in the condenser air ejector and turbine gland steam off-gas.	The release rate at individual positions during the worst failure mode under normal operation(described in reference material issued by 105 th subcommittee of the Safety Examination Committee) is determined from the evaluation results.
3	Carbon dioxide gas system monitor	Discharge piping of carbon dioxide gas system	1	Scintillation	1×10^{0} ~1×10^{6} (Bq/cm^3)	To monitor the concentration of radioactive materials in carbon dioxide gas of the carbon dioxide gas system.	"High" alarm is actuated on 3 times B. G.
4	Helium system monitor	Discharge piping of helium system	1	Scintillation	1×10^{0} ~1×10^{6} (Bq/cm^3)	To monitor the concentration of radioactive materials in helium of the helium system.	"High" alarm is actuated on 3 times B. G.
5	Main stack monitor	Sampling piping from main stack	2	Ionization chamber	1×10^{-2} ~1×10^{2} 1×10^{3}~1×10^{9} (Bq/cm^3)	To monitor concentrations of radioactive materials in off-gas discharged into air from the main stack.	"High" alarm is actuated on 10 times the detection limit
			2	Scintillation	1×10^{-3} ~1×10^{3} (Bq/cm^3)		"High high" alarm is actuated on 1/50 release of the safety regulation value per hour, "high" alarm on 1/2 release of "high high" alarm
			2	Geiger-Muller counter	4×10^{-8} ~4×10^{-2} (Bq/cm^3)		"High" alarm is actuated on the value lower than the lowest concentration (^{90}Sr) of anticipated nuclides at peripheral area.

6	Waste treatment building stack monitor	Sampling piping from waste treatment building stack	1	Ionization chamber		To monitor the concentration of radioactive materials in off-gas released into air from the waste treatment building stack, including incineration off-gas from the miscellaneous solid waste incinerator.	"High" alarm is actuated on the concentration when 1/10 of the annual release control target for tritium is not released per hour
			1	Geiger-Muller counter			Similar to the main stack monitor
7	Condenser cooling water tank monitor	Condenser cooling water discharge tank	1	Scintillation	2×10^{-3} $\sim 2 \times 10^{3}$ (Bq/cm^3)	To monitor the concentration of radioactive materials in drain discharged from the condenser cooling water discharge tank into the sea.	"High" alarm is actuated on10 times or less the lowest limit of the concentration set forth in the test reactor regulations

Table VI. 4-2 Discharge Control of Radioactive Gaseous Wastes

Radionuclides to be Measured		Low Limit of Concentration for Measurement (Bq/cm^3)	Minimum Measurement Frequency	Instrument System
Physical Form	Type of radioactive material			
Gaseous Material	Radioactive noble gas	approx. 10^{-3}	Continuous	Stack gas monitor
Volatile Material	Iodine-131	4×10^{-9}	Once/week	Ge(Li) semiconductor spectrometer
	Iodine-133	3×10^{-8}	Once/week	
	Tritium	4×10^{-5}	Once/week	Liquid scintillation counter
Particulate Material	γ-ray emitting nuclides Chrome-51, Mangan-54, Iron-59, Cobalt-58, Cobalt-60, Iodine-131, Cesium-137, etc.	4×10^{-2} *1	Once/week	Ge(Li) semiconductor spectrometer
	Strontium-89, Strontium-90	4×10^{-10} *2	Once/quarter	2π thin window gas flow counter
	All β radioactivity	1×10^{-13}	Once/month	2π thin window gas flow counter
	All α radioactivity	1×10^{-14}	Once/month	2nS(Ag) scintillation counter

*1 Value for Cobalt-60 as representative
*2 Value for Strontium-90 as representative

Table VI.4-3 Discharge Control of Radioactive Liquid Wastes

Radionuclides to Be Measured	Lower Limit of Concentration for Measurement (Bq/cm^3)	Measurement Frequency	Measurement System
γ-ray emitting nuclides Chrome-51, Mangan-54, Iron-59, Cobalt-58, Cobalt-60, Iodine-131, Cesium-137, etc.	2×10^{-3} *1	Every release case	Ge(Li) semiconductor spectrometer
Strontium-89 Strontium-90	2×10^{-4} *2	Once/quarter	2π thin window gas flow counter
Tritium	4×10^{-2}	Every release case	Scintillation counter
All β radioactivity	4×10^{-2}	Once/month	2π thin window gas flow counter
All α radioactivity	4×10^{-3}	Once/month	ZnS(Ag) scintillation counter

*1 Value for Cobalt-60 as representative
*2 Value for Strontium-90 as representative

5. Waste Discharge Experience, Storage Experience and Evaluation

5.1 Release Experience of Gaseous Wastes

Table VI.5-2 shows the annual discharged gaseous waste radioactivity and the discharge control target values. Fig. VI.5-1 indicates the trend of annually discharged tritium radioactivity.

The discharged radioactivity of gaseous wastes have been well below the discharged control target values, except for tritium.

The discharged radioactivity of tritium has also been below the discharge cotorl target value; however, the reduction measure for the discharged radioactivity has been taken since FY 1988 because the discharged amount was increasing as the tritium concentration in the heavy water increased.

The detailed discharge experience is described in the following.

(1) Radioactive Noble Gases

Radioactive noble gas were discharged at an order of about 10^{11} Bq/year from 1979 to 1983 during the early stage of operation as illustrated in Table 5-2. The radioactive noble gas, whose nuclide is Argon-41, was originated from the helium gas released to control the pressure increase inside the system due to expansion of helium volume during reactor startup and the carbon dioxide gas released from the carbon dioxide gas system. The Argon-41 is generated by activation of Argon-40 contained in the air which was introduced into the system during inspection of the helium system and the carbon dioxide gas system components while the plant was shutdown along with operation of the reactor.

Since 1984, the discharge of the radioactive noble gases have been controlled under the detection limit owing to the following actions; for the helium system, enhancement of helium purification performed as a part of the measures to improve the heavy water chemistry resulted in reduction of air remained inside the helium system, leading to reduction of Argon-41: for the carbon dioxide gas system, sufficient purging was performed before reactor startup. The once-through type carbon dioxide gas system was modified to the recirculation type to reduce the consumption of carbon dioxide gas in 1985. This has also contributed to the reduction of the carbon dioxide gas discharge. Since then, the radioactive noble gas discharge has been controlled under the detection limit.

A small amount of radioactive noble gas was discharged in 1989 and 1991. This was caused by the helium release when the ventilation path of the sampling hood was changed from the general ventilation system to the tritium removal system as a measure to reduce the tritium release from the heavy water system

sampling hood (in 1989) and by activation of Argon-40 remained in the carbon dioxide gas system during reactor startup (1991).

The discharged radioactive noble gases at the main stack have been evaluated since 1986 by taking "the integrating evaluation with the gas monitor" based on the "Guidelines on the Radioactive Material Measurement in the Light-water Reactor Facility for Power Generation" stipulated the Atomic Energy Commission, in order to improve the detection sensitivity.

(2) Iodine-131

The Iodine-131 discharge has been consistently controlled under the detection limit owing to no fuel failure experience as illustrated in Table VI.5-1. Iodine-131 was discharge at an order of 10^7 Bq in 1986. This was caused by the Chernobyl accident when increased iodine-131 in surrounding environment (in the air) was taken into the ventilation system and then exhausted from the main stack.

(3) Tritium

(i) The tritium discharge was kept down to the detection limit from 1979 to 1981 as shown in Table VI.5-1 and Fig. VI.5-1. This was because the discharge amount of tritium was evaluated using the "ion chamber monitor".

Since 1982, the "pressurized cooling condensation method sampling system" was introduced for the evaluation of the tritium discharge according to the "Guidelines on the Released Radioactive Material Measurement in the Light Water Reactor Facility for Power Generation". This improved the detection sensitivity, resulting in the discharge amount at an order of $10^{11}\sim10^{12}$ Bq/year.

(ii) The discharged amount of tritium increased from 1982 to 1984 as the tritium concentration in heavy water increased, in spite of the measures taken to reduce the amount of tritium transported into air (heavy water draining and drying operation during heavy water system components inspection) and to reduce the tritium concentration in the working environment (installation of the exclusive house for tritium release and ventilation through the tritium removal system).

(iii) From 1985 to 1987, the tritium discharge showed a significant increase. This was because a series of heavy water sampling were performed in addition to the normal sampling (twice/month) as an investigation and analysis method for improvement of heavy water chemistry, resulting in an

increased frequency of sampling, that is, once or twice a week.

(iv) The tritium concentration has decreased since 1988 because the above mentioned investigation/analysis of heavy water chemistry was completed and the frequency of heavy water sampling was returned to the normal frequency. In addition, it is also because the ventilation of the sampling hood was performed through the tritium removal system. As a result, the annual discharge of tritium halved to about 2×10^{12} Bq and then has shown little changes.

(4) Particulate Materials

About 3×10^{5} Bq of particulate materials were detected in 1986 due to the effects of the Chernobyl accident similar to iodine-131. Besides this year, the released amount of particulate materials have been controlled under the detection limit as shown in Table VI.5-1.

5.2 Discharge Experience of Liquid Wastes

Table VI.5-2 shows the annual discharged liquid wastes radioactivity and the discharge control target values.

Although the discharged radioactivity of the liquid waste γ nuclides (Co-60, Mn-54, etc.) was well below the discharge contorol target value {7.4×10^{9} Bq (except for ^{3}H) / year}, in order to gain the general public's consensus, numerous discharge reduction measures have been taken. First, for the floor drain / equipment drain waste liquids, after FY 1982, the evaporation and concentration treatment was taken, and there is almost no discharge from such wastes. On the other hand, for the laundry drain, which dominates the discharged amount after FY 1982, the dry-cleaning equipment was introduced and its operation was modified. As a result, the discharged radioactivity was reduced to the value below the discharge control target value after FY 1994.

The discharged amount of tritium is increasing as the tritium concentration in the heavy water increases, like the case with the gaseous tritium. Although the discharged amount has been below the discharge contorol target value {1.1×10^{13} Bq (^{3}H) / year}, it reached about 1/2 of the discharge control target value in FY 1989. The equipment to collect the tritium as the reusable degraded heavy water is being examined for the generation sources.

The detailed discharge experience is described in the following:

(1) γ Nuclides (See Table VI.5-2 column excluding those for tritium)

Most of the liquid γ nuclide discharge occurs during the periodic inspection.

Up to FY 1982, the contribution of the floor drain and the equipment drain waste liquids, besides the laundry drain, was large. Therefore, after FY 1982, the operation was changed for most of the floor drain and equipment drain waste liquids from filtering to evaporation and concentration, and then little discharge occurred from such waste liquids.

After FY 1982, the laundry drain has been dominant. The largest discharge amount from the laundry drain in FY 1981 and the second largest amount in FY 1985 was due to the constraction work for SCC countermeasures which resulted in the long reactor shutdown bringing about a lot of contaminated clothes.

(i) The annual discharge has been reduced since 1990. This is owing to introduction of the large dry cleaning system (capacity: 90 kg/h). Working clothes (red and yellow clothes and socks) used inside the area with high possibilities of contamination were washed by the dry cleaning system, resulting in reduction of the liquid γ nuclides discharged from laundry drain. In 1991, the discharged amount of liquid wastes are rather low because few annual inspection work was conducted during this year.

(ii) Since 1994, the amount of liquid wastes other than tritium has been consistently controlled under the detection limit. This is because all the working clothes used inside the controlled area during annual inspection period were washed by the dry cleaning system while socks which had been washed by the dry cleaning system were disposed of after use.

The amount of γ nuclides dischargered from those other than laundry drain has been reduced owing to evaporation/concentration treatment of equipment drain and floor drain implemented since 1982 and adoption of the hollow yarn membrane filter as the equipment drain filter for efficient liquid waste treatment and improved efficiency of crud collection in 1989. As a result, the discharged amount has been controlled under the detection limit since 1992.

(2) Tritium (See Table VI.5-2, column for Tritium)

(i) Liquid tritium waste is roughly divided into liquid waste generated from resin exchange of the heavy water purification system and concentrated water generated from the tritium removal system. It accounts for most of the sources of tritium release.

The liquid tritium discharge showed an increasing trend in the early stage of operation as the tritium concentration in heavy water increased as shown in Table VI.5-2. The procedures and method of resin exchange of the heavy

water purification system were improved incorporating operational experience and the inspection procedures were fully reviewed before start of inspection so that the liquid waste generated would be minimized to reduce the amount of liquid tritium waste.

As a result, the liquid tritium release has been kept at around 3~5 x 10^{12} Bq/year since 1984 in spite of increases of the tritium concentration in heavy water.

(ii) The increase of the discharge in 1985 is due to the increased load on the tritium removal system since intensive inspection was conducted for the heavy water system components during the 5th annual inspection.

(iii) The liquid tritium discharge has slightly increased since 1988. This is because the ventilation path of the sampling hood which handles heavy water was changed to the tritium removal system as mentioned in the section of "Gaseous waste discharge results". In 1991, the tritium discharge was rather low because few annual inspection works were conducted during this year.

5.3 Solid Waste Generation and Storage Experience

(1) Spent Powdered resins

Spent powdered resins are generated from the exchange of powdered ion exchange resins which are used for the demineralizer of the fuel storage pool cooling water purification system. The generated amount of spent powdered resins depends on the cleanliness of fuel storage pool water; it usually amounts to around 0.5 m^3/year though it changes with the number of fuel assemblies handled.

One unit of spent powdered resin storage tank was additionally installed in 1989 to enhance the capacity to store resins.

Table VI.5-3 and Fig. VI.5-2 illustrate the actual results of the powered resin wastes generation.

(i) The generated amount of spent powdered resins were 1.4 m^3/year in 1979 when "Fugen" started operation. This is due to pool water cleaning along with modification of the spent fuel storage racks (increased the number of fuel assemblies stored from 270 to 730).

(ii) The generated amount of spent powdered resins were 1 m^3/year from 1980 to 1981. This is because all the fuel assemblies in the entire core (224 assemblies) were unloaded and transferred into the fuel storage pool for the

modification work of the reactor cooling system piping to prevent SCC which was detected in November 1980. They were reloaded in the core in FY 1981.

(iii) In FY1985, 1.7 m^3 of spent powdered resins were generated. This was due to unloading and reloading of 112 fuel assemblies of A loop for replacement work of the A loop piping to prevent SCC conducted during the 5th annual inspection.

(iv) In FY 1989, spent powdered resins showed a slight increase at 0.8 m^3/year. This was due to unloading and reloading of 112 fuel assemblies of A loop for chemical cleaning of the reactor cooling system along with material replacement work of A loop lower header central area etc. to prevent SCC.

(v) FY 1990, though 112 fuel assemblies of B loop were unloaded and reloaded during the chemical cleaning performed following the previous year, spent powdered resin generation was low at 0.4 m^3/year. The reason was as follows: the unloaded fuel assemblies were cleaned with ultrasonic wave and the crud attached on the fuel surface was removed with the filter; thus, fuel storage pool water was maintained under a relatively clean condition, leading to less spent powdered resin generation.

As few annual inspection work was conducted in FY 1991, the spent powdered resin generation was low.

(2) Spent Particulate Resins [VI-7]

Spent particulate resins are generated from exchange of particulate resins used for the demineralizers of the reactor coolant purification system, heavy water purification system, feed water system (condensate demineralizer) and the equipment drain system (liquid waste demineralizer). As resin exchange is performed depending on the treatment water chemistry and the resin exchange capacity of the demineralizers, the generation differs by year as shown in Table VI.5-3 and Fig. VI.5-3. For the demineralizers of the reactor coolant purification system and the heavy water purification system for which resin exchange is performed every year, about 3～5 m^3 of spent particulate resins are generated annually. Similar to spent powdered resins, 5 units of spent particulate resin storage tank were additionally installed in FY 1989 to enhance the storage capacity.

(i) The spent particulate resin generation was rather high, about 15～16 m^3/year in FY 1989 and 1990. This was because the cleaning of decontamination agent used for chemical decontamination of the reactor

cooling system was performed using the demineralizer of the reactor coolant purification system.

(ii) The spent particulate resin generation showed a slight increase, about 7~9 m^3/year from FY 1992 to 1994. This was because spent resin generated from the condensate demineralizer and the liquid waste demineralizer was stored in the spent particulate resin storage tank, considering the capacity of the additionally installed spent particulate resin storage tanks.

(3) Miscellaneous Solid Wastes

The number of drum cans containing solid wastes changes to a large extent as ahown in Table VI.5-4 and Fig. VI.5-4 depending on the status of the modification work such as piping replacement. Without any modification work which generates a lot of wastes, about 1,500 drum cans of wastes are generated annually. Among them, combustible waste accounts for about 60 % while non-combustible waste remaining 40 %.

The second solid waste storage building with a storage capacity of 13,000 drum cans was constructed in FY1984 to enhance the storage capacity.

(i) About 2,200~2,500 drum cans of solid wastes were annually generated in FY 1981, 1982 and 1985 respectively. Such a large amount of generation was due to piping replacement work for the reactor recirculation system, etc. conducted as a measure to prevent SCC.

(ii) In FY 1983, only the drum cans containing combustible waste were stored. This is because ventilation system filters which had been stored in the miscellaneous solid waste storage building (equivalent to 472 drum cans) were compressed to reduce waste volume, resulting in 228 drum cans.

(iii) Since FY 1990, the number of waste drum cans has shown a significant decrease, to about 500 drum cans/year. This is owing to completion of SCC prevention work in FY 1990 which reduce the number of non-combustible solid waste drum cans: in addition, incineration and volume reduction of combustible solid wastes were conducted making use of the "miscellaneous solid waste incineration system" installed in October 1989. This incineration system is used for incineration and volume reduction not only for combustible waste generated at the reactor facility but also for those stored at the solid waste storage building; thus it has contributed to the reduction of the volume of combustible miscellaneous solid waste stored in "Fugen".

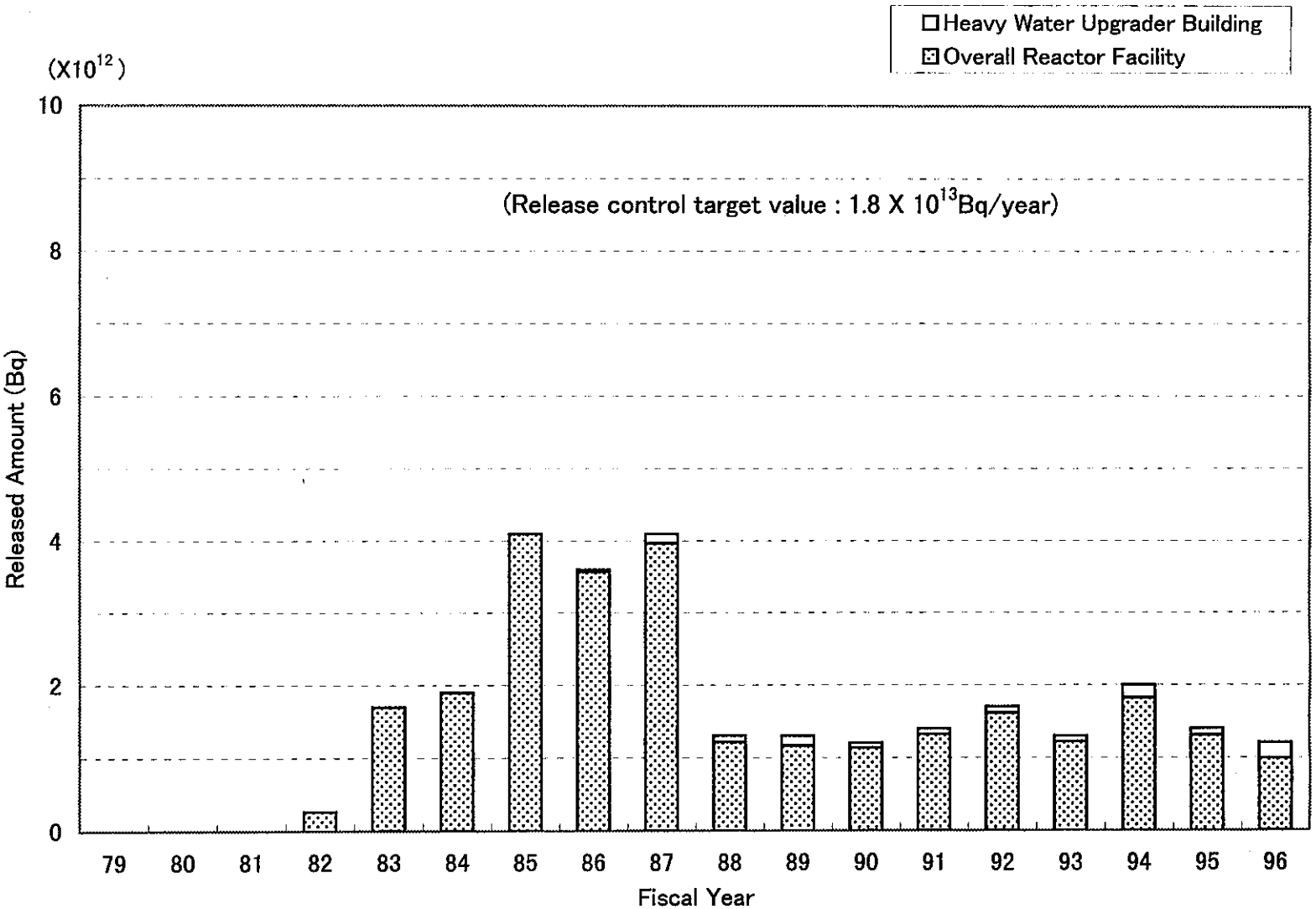


Fig. VI.5-1 Trend of Annually Discharged Amount of Gaseous Waste (Tritium)

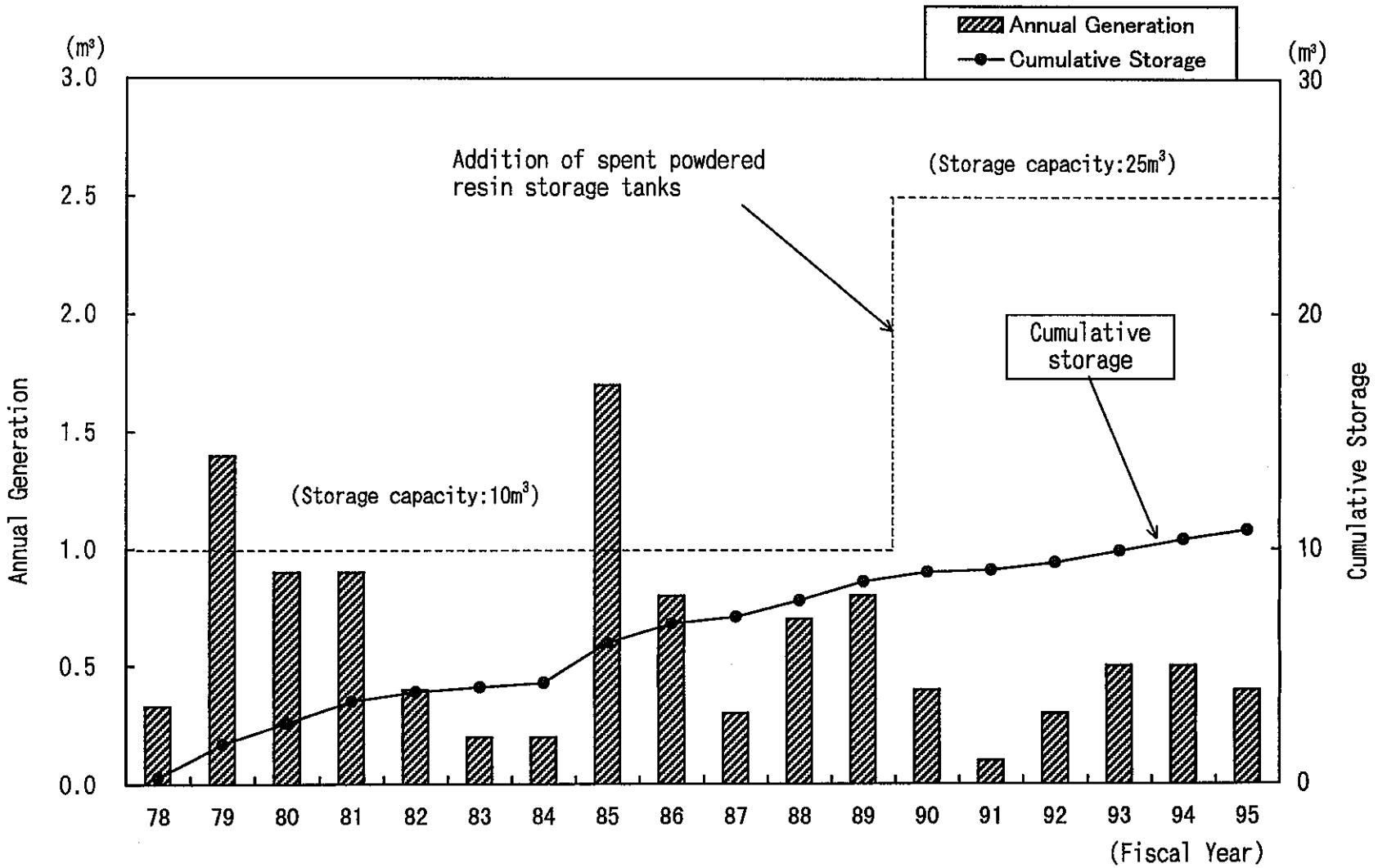


Fig.VI.5-2 Results of Spent Powdered Resin Generation

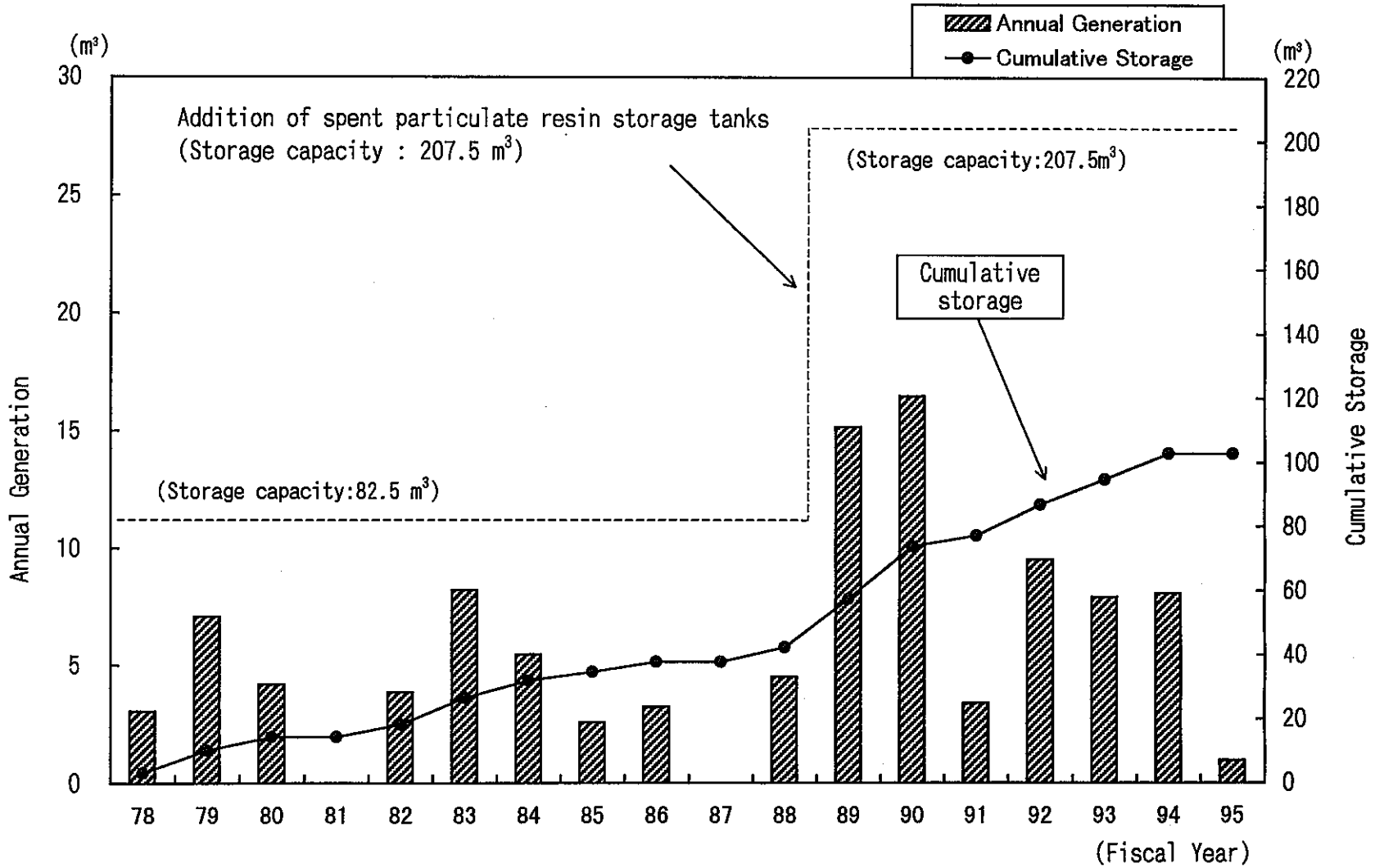


Fig.VI.5-3 Results of Spent Particulate Resin Generation

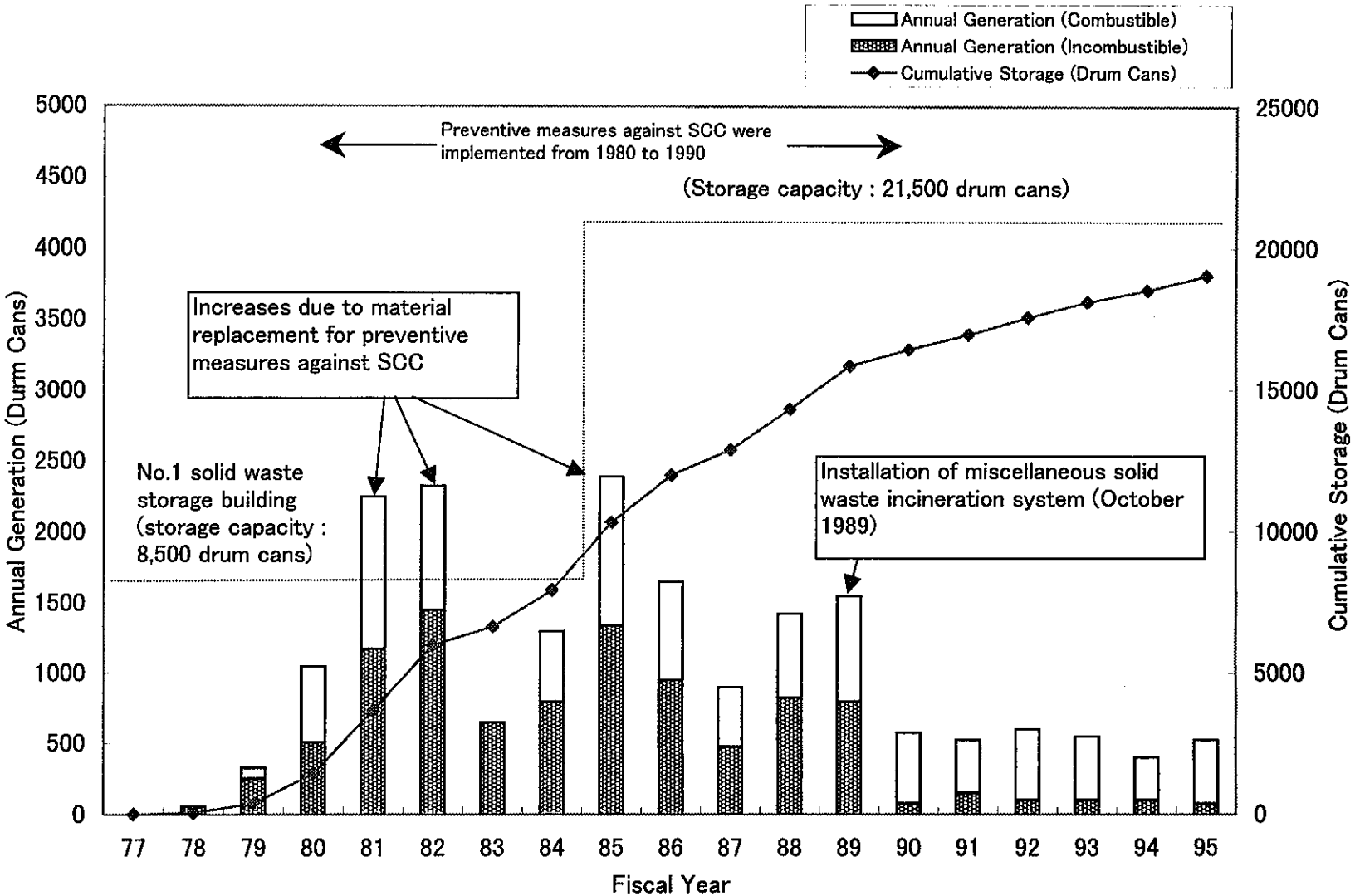


Fig. VI. 5-4 Trend of Stored Drum Cans Containing Radioactive Solid Wastes

Table VI.5-1 Radioactivity of Discharged Gaseous Wastes per Year

(Unit :Bq)

Fiscal Year	Radioactive Noble Gas	Iodine-131	Tritium	Particulate Material
78	3.1×10^{10}	ND	ND	ND
79	8.9×10^{10}	ND	ND	ND
80	4.1×10^{11}	ND	ND	ND
81	2.2×10^{10}	ND	ND	ND
82	9.6×10^{10}	ND	2.6×10^{11}	ND
83	2.4×10^{10}	ND	1.7×10^{12}	ND
84	ND	ND	1.9×10^{12}	ND
85	ND	ND	4.1×10^{12}	ND
86	ND	*5.6×10^{7}	3.6×10^{12}	*3.0×10^{5}
87	ND	ND	4.1×10^{12}	ND
88	ND	ND	1.3×10^{12}	ND
89	1.2×10^{9}	ND	1.3×10^{12}	ND
90	ND	ND	1.2×10^{12}	ND
91	2.2×10^{10}	ND	1.4×10^{12}	ND
92	ND	ND	1.7×10^{12}	ND
93	ND	ND	1.3×10^{12}	ND
94	ND	ND	2.0×10^{12}	ND
95	ND	ND	1.4×10^{12}	ND
96	ND	ND	1.2×10^{12}	ND
Discharge Control Target Value	5.1×10^{14}	2.7×10^{10}	1.8×10^{13}	———

* Effect of Chernobyl Nuclear Power Station Accident

Table VI.5-2 Radioactivity of Discharged Liquid Wastes per Year

(Unit :Bq)

Fiscal Year	Radioactive Materials Other than Tritium *	Tritium *
78	3.3×10^{7}	2.6×10^{11}
79	5.3×10^{7}	2.7×10^{11}
80	3.7×10^{7}	7.7×10^{11}
81	2.9×10^{8}	8.5×10^{11}
82	3.1×10^{7}	1.2×10^{12}
83	4.8×10^{7}	1.3×10^{12}
84	1.9×10^{7}	2.6×10^{12}
85	1.0×10^{8}	3.6×10^{12}
86	4.8×10^{7}	2.2×10^{12}
87	1.9×10^{7}	1.9×10^{12}
88	4.8×10^{7}	4.4×10^{12}
89	5.8×10^{7}	7.0×10^{12}
90	1.4×10^{7}	3.3×10^{12}
91	4.7×10^{6}	1.8×10^{12}
92	1.1×10^{7}	3.9×10^{12}
93	1.6×10^{6}	3.5×10^{12}
94	ND	4.7×10^{12}
95	ND	4.1×10^{12}
96	ND	5.9×10^{12}
Discharge Control Target Value	7.4×10^{9}	1.1×10^{13}

* Values before 1988 fiscal year were converted to Bqs from the reported value in curies.

Table VI.5-3 Generation of Spent Ion Exchange Resins per Year

(Unit :m^3)

Fiscal Year	Spent Powder Resin	Cumulation	Spent Granular Resin	Cumulation
78	0.3	0.3	3.0	3.0
79	1.4	1.7	7.0	10.0
80	0.9	2.6	4.2	14.2
81	0.9	3.5	0.0	14.2
82	0.4	3.9	3.7	17.9
83	0.2	4.1	8.1	26.0
84	0.2	4.3	5.5	31.5
85	1.7	6.0	2.5	34.0
86	0.8	6.8	3.2	37.2
87	0.3	7.1	0.0	37.2
88	0.7	7.8	4.5	41.7
89	0.8	8.6	15.0	56.7
90	0.4	9.0	16.4	73.1
91	0.1	9.1	3.3	76.4
92	0.3	9.4	9.4	85.8
93	0.5	9.9	7.8	93.6
94	0.5	10.4	8.0	101.6
95	0.4	10.8	1.0	102.6

Table VI.5-4 Generation of Drum Cans Containing Miscellaneous Radioactive Solid Wastes

(Unit : Drum Can Equivalent)

Fiscal Year	Generation per Year		Cumulative Stock
	Incombustible	Combustible	
77	6	0	6
78	8	60	74
79	95	225	394
80	538	512	1,444
81	1,089	1,163	3,696
82	887	1,446	6,029
83	-43	640	6,626
84	518	792	7,916
85	1,066	1,333	10,335
86	691	967	11,993
87	443	480	12,916
88	610	820	14,346
89	778	774	15,898
90	526	49	16,405
91	374	164	16,362
92	503	95	16,715
93	464	100	16,993
94	320	62	16,966
95	450	81	17,064

6. Gaseous Waste and Liquid Waste Discharge Reduction Measures and Solid Waste Generation Reduction Measures

6.1 History and Outline of Individual Gaseous Waste Discharge Reduction Measures

In "Fugen", the past gaseous waste discharge of both noble gas and iodine-131 has been well below the discharge control target value because of no fuel failure experience. For tritium, the gaseous discharge has been maintained at an constant level of spite of the increase of the tritium concentration in heavy water.

On the other hand, measures to further reduce the gaseous waste discharge have been taken as shown in Table VI.6-1 since a slight amount of noble gas was discharged from the helium system and the carbon dioxide gas system which were specific to "Fugen" in the early stage of operation and tritium was discharged during equipment inspection.

(1) Radioactive Noble Gas

(a) Reduction of Residual Air by Enhancement of Helium Purification (reduction of Argon-41 generation)

A trace amount of air is contained in the helium system as an impurity, resulting in the presence of argon-41 which is generated by activation of Argon-40 in the air. During reactor startup, it is necessary to discharge helium properly, to control the system pressure rise resulting from helium volume expansion.

Therefore, some argon-41 was discharged along with helium discharge during reactor startup from the start of operation to FY 1983. However, enhancement of helium purification (lower the target value of nitrogen concentration which remains in the helium system from 1 % to 0.1 %) has been conducted as a part of heavy water chemistry improvement measures since 1983, and this resulted in the reduction of residual air inside the helium system. In addition, the helium discharge procedures was changed from continuous operation to intermittent and short period discharge .

These measures have contributed to the reduction of the noble gas discharge along with helium discharge operation during reactor startup since 1984.

(b) Reduction of Noble Gas Discharge by Enhancement of Carbon Dioxide System Purging

The carbon dioxide gas system is filled with air during plant shutdown; therefore, carbon dioxide purging is performed before reactor startup to minimize residual air. However, the complicated structure of the carbon dioxide gas system mainly consisting of the pipes with small diameters is

likely to prevent sufficient purging. This has resulted in a discharge of argon-41 from the main stack which was generated from activation of a trace amount of argon-40 remained in the system. Therefore, the start timing, the flow rate, etc. of the purging operation were reviewed and then improved purging has been conducted since 1984 to reduce the noble gas discharge caused by insufficient purging of the carbon dioxide gas system. In addition, for reduction of the carbon dioxide gas consumption, the conventional "once-through type" system was modified to the "recirculation type" system in FY 1985; combined with the effects of this new system to reduce the carbon dioxide gas discharge, the noble gas discharge has been reduced since then.

(2) Tritium

(a) Change of Heavy Water Handling Hood Ventilation Path

Since heavy water is used as the moderator in "Fugen", the tritium concentration in the heavy water is increasing year after year with the reactor operation, due to the reaction of $^{2}D\ (n, \gamma)\ ^{3}H$, etc.; the tritium discharge showed an increasing trend as the tritium concentration increased in the early stage of operation. Most of the tritium discharge is caused by overhaul of the heavy water system components. Investigation of the tritium discharge path revealed that there might be possibilities of tritium discharge from the heavy water sampling hood and the heavy water handling hood in the hot chemical laboratory. As a result, in FY 1988 the ventilation path from these heavy water handling hoods were switched from the general ventilation system to the tritium removal system to reduce the tritium discharge from the heavy water handling hoods. In the tritium removal system, heavy water steam transferred into air was adsorbed by drying agent (molecular sieve).

(b) Heavy Water Draining and Drying Reflected by Maintenance Experience

If heavy water remains inside the equipment during overhaul, the tritium concentration in the working environment increases, causing not only increases in the tritium discharge but also reduction of work efficiency. Therefore, improvements for more efficient heavy water draining and drying have been made since the commencement of commercial operation; e.g. heavy water remained in the heavy water system components was recovered as much as possible by draining before overhaul, inspection experience of the heavy water system components has been incorporated into the following heavy water draining and drying operations, and modification work of the pipes with small diameters connecting to the heavy water circulation pump was

implemented for smoother heavy water draining from the pump.

6.2 History and Outline of Individual Liquid Waste Discharge Reduction Measures

For radioactive material other than tritium, the liquid waste discharge in Fugen is well below the discharge control target value. The tritium discharge has shown no marked changes in spite of increases of the tritium concentration in heavy water. However, to further reduce the liquid waste discharge, measures have been taken as illustrated in Table VI.6-1.

(1) Radioactive Material Other than Tritium

(a) Evaporation/Concentration Equipment Drain and Floor Drain Wastes

The equipment drain system is so designed that it can treat liquid waste with relatively high chemical purity and low crud content. In "Fugen", a large amount of liquid wastes containing a lot of crud elements is generated including liquid wastes recovered from the condensate delimineralizer and drain water from the refueling machine. The existing equipment drain system is not able to sufficiently remove the crud; this resulted in re-treatment or dilution of liquid wastes with condensate cooling water to discharge after confirming that the radioactive material concentration was low enough, causing extension of the treatment time or increases in the liquid waste discharge.

Therefore, for efficient treatment of liquid wastes containing much crud and reduction of the liquid waste discharge, improvements were made in FY 1982 so that the evaporation/concentration equipment could also treat equipment drain waste and floor drain waste and this treated water could be treated by the equipment drain system. In FY 1989, the "hollow yarn membrane filter" was adopted as equipment drain filter to improve the efficiency of equipment drain waste treatment and crud collection. This contributed to efficient operation of the equipment drain system and floor drain system, resulting in the liquid waste discharge not more than the detection limit since 1992.

(b) Introduction of Dry Cleaning System

For reduction of the liquid waste discharge from the laundry drain system, the large dry cleaning system (capacity: 90 kg/h) was introduced in FY 1989 so that all working clothes (red and yellow clothes and socks) used in the area with high possibilities of contamination can be washed. This reduced the liquid waste discharge from the laundry drain system to about 1/5.

(c) Introduction of Shoes Washing Machine

The dry cleaning system, which was used for washing working clothes, was not applicable to washing working shoes. Therefore, working shoes were hand-washed and generated liquid wastes was discharged as laundry drain; this contributed to the discharge from the laundry drain system.

Therefore, in FY 1991, the washing machine exclusively used for working shoes was introduced and generated liquid waste was treated by the liquid waste treatment system to reduce the liquid waste discharge from laundry drain and the working load.

(d) Enhancement of Laundry System Operation

The liquid waste discharge from the laundry drain system showed a decreasing trend owing to introduction of the dry cleaning system and the shoes washing machine. For further improvement, all working socks, which had been reused after dry or water cleaning, were disposed of after usage during annual inspection since 1994. On the other hand, working clothes (blue clothes) which had been washed with water were washed by the dry cleaning system. At the same time, other improvements for enhancement of laundry system operation were made including extension of dry cleaning system operation time. These improvements have contributed to make the liquid waste discharge activity from the laundry drain system less than the detection limit since 1994.

(2) Tritium

(a) Tritium Discharge Reduction by Improvement of Heavy Water Purification System Resin Exchange Method

The heavy water purification system has 2 types of demineralizers; the "heavy water purification cloumn" for removal of impurities in heavy water and the "poison removal cloumn" for removal of poison in heavy water.

The resins used for these demineralizers are exchanged according to the results of its performance. To minimize the tritium discharge, it is important to reduce liquid tritium generated from resin exchange. Therefore, previous experience in resin exchange was analyzed and investigated to incorporate the results into the following working procedures.

(i) Improvement in dedeuteration of spent resins in heavy water purification system

When transporting spent resins from the heavy water purification column and poison removal column, heavy water with a low tritium concentration (heavy water purified by the heavy water upgrader) is used instead of

system heavy water with a high tritium concentration to lower the tritium concentration in resins.

In addition, it is necessary to dedeuterate the spent resins for disposal. The heavy water with concentration not exceeding 1 % has been transported to the degraded heavy water storage tank for effective recycle of heavy water. In the past, however, the heavy water with concentration not exceeding 1 % was once stored in the light water storage tank and then discharged after being processed by the liquid waste treatment system.

Improvement was so made to reduce the tritium discharge that water containing tritium in the light water storage tank should be reused to dedeuterate the spent resins for the next resin exchange.

6.3 History and Outline of Solid Waste Reduction Measures

To reduce solid waste generated on-site, it is necessary to restrict articles brought into the controlled area and to make materials and goods to be reused. For reduction of wastes to be stored at the solid waste storage building, volume reduction of wastes is required.

To achieve these goals, various measures have been taken as shown in Table VI.6-2.

(i) To restrict unnecessary articles brought into the controlled area, a monitoring persons perform monitoring of articles and give some instruction and advice when necessary to do so.

(ii) When removing thermal insulating material during ISI or other inspection activities, conventional insulating material is liable to be broken, therefore, it would be difficult to reuse it.

To reduce solid waste generation, the piping insulating material has been changed to metalic one, starting with the RCS (Reactor Coolant System) piping.

(iii) To reduce combustible miscellaneous solid wastes to be stored, the miscellaneous solid waste incinerator was introduced in FY 1989 for incineration and volume reduction of miscellaneous solid wastes. As a result, combustible miscellaneous solid waste generated on-site and waste stored at the solid waste storage building have been significantly reduced to about 1/45; thus, the load on the solid waste storage building capacity has been reduced.

Table VI.6-1 History of Liquid/Gaseous Waste Discharge Reduction Measures

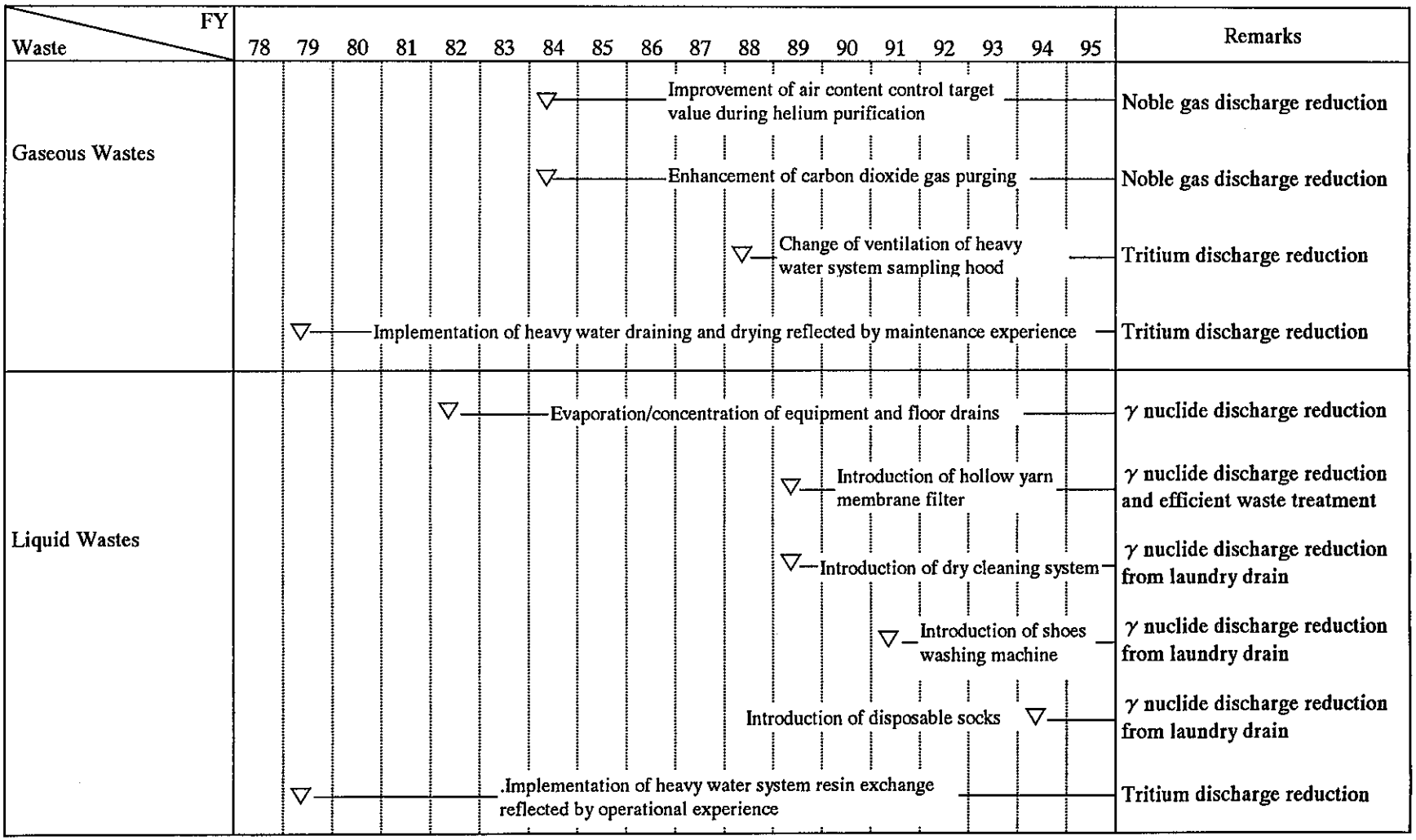
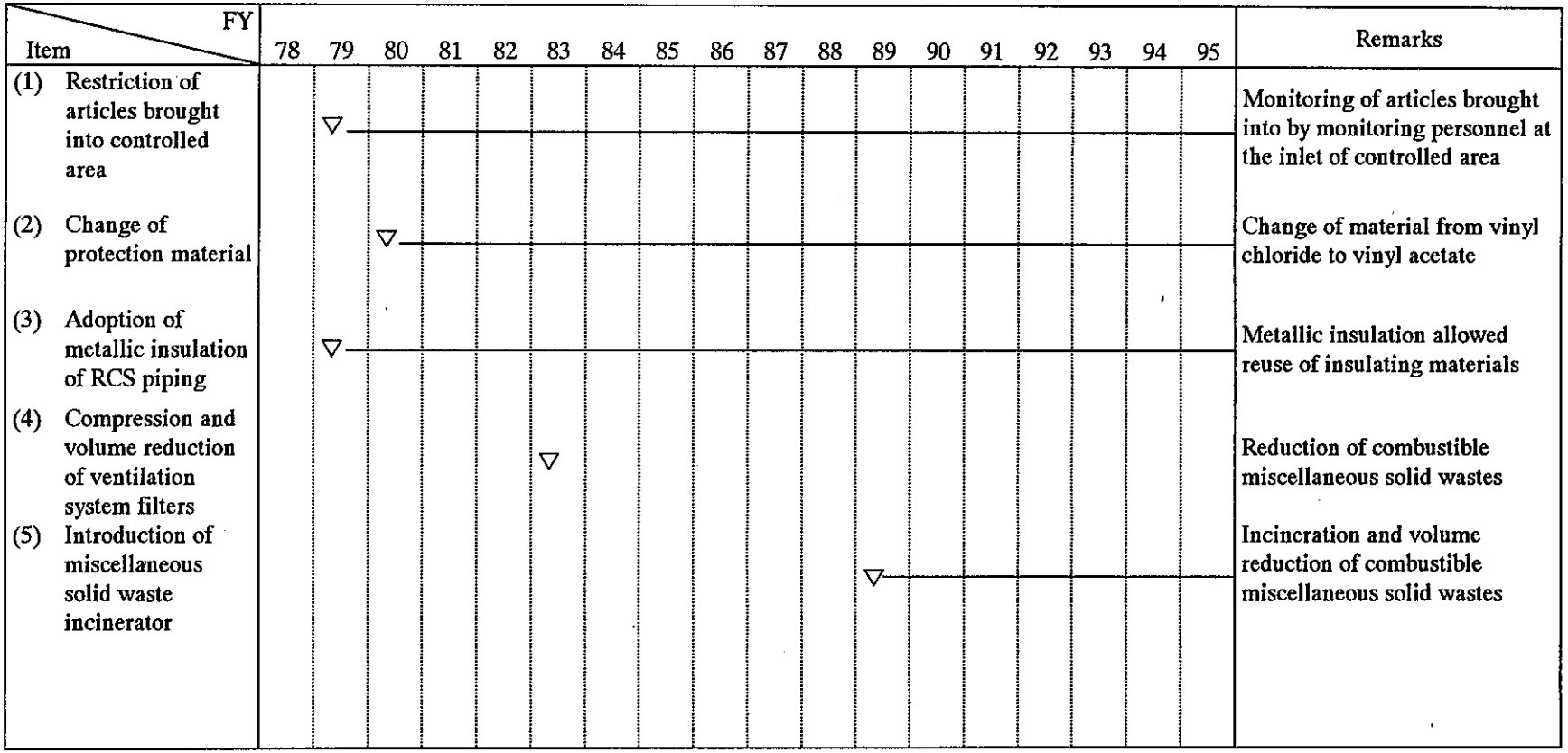


Table VI.6-2 History of Solid Waste Reduction Measures



Item \ FY	78	79	80	81	82	83	84	85	86	87	88	89	90	91	92	93	94	95	Remarks
(1) Restriction of articles brought into controlled area		▽	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	Monitoring of articles brought into by monitoring personnel at the inlet of controlled area
(2) Change of protection material			▽	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	Change of material from vinyl chloride to vinyl acetate
(3) Adoption of metallic insulation of RCS piping		▽	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	Metallic insulation allowed reuse of insulating materials
(4) Compression and volume reduction of ventilation system filters						▽													Reduction of combustible miscellaneous solid wastes
(5) Introduction of miscellaneous solid waste incinerator												▽	—	—	—	—	—	—	Incineration and volume reduction of combustible miscellaneous solid wastes

7. Conclusion and Future Plan

7.1 Conclusion

(1) Radioactive Gaseous Wastes

Various measures have been taken to reduce the gaseous waste discharge. As a result, the noble gas discharge has decreased year after year while the tritium discharge has shown little changes in spite of continuous increases of the tritium concentration.

At present, comparing thr actual discharge results with the discharge control target values set forth in the reactor facility safety regulations, the noble gas discharge is $1/10^4$ or less and the tritium discharge is about 1/10.

(2) Radioactive Liquid Wastes

Various measures have also been taken to reduce the liquid waste discharge. As a result, the discharged amount of radioactive materials other than tritium has decreased year after year; the liquid tritium discharge has shown little changes similar to the gaseous tritium discharge. At present, comparing the actual discharge results with the discharge control target values set forth in the reactor facility safety regulation, the radioactive material discharge other than tritium is $1/10^3$ or less and the tritium discharge is about 1/3.

Based on the actual results of the radioactive gaseous and liquid waste discharge, the dose equivalent received by the public population around "Fugen" is evaluated to be under $1\,\mu$Sv/year, which is well below the target dose ($50\,\mu$ Sv/year).

(3) Radioactive Solid Wastes

Various measures have also been taken to reduce solid wastes. Incineration and volume reduction of combustible waste by the miscellaneous solid waste incinerator has made significant effects. Combustible wastes stored at the solid waste storage building is also subject to incineration and volume reduction by the incinerator; as a result, the increase of annual waste generation (to be stored at the solid waste storage building) is controlled at about 150 drum cans/year.

7.2 Future Plan

(1) Reduction of Radioactive Gaseous Waste Discharge

The gaseous waste discharge is controlled at a low level at present. It is necessary to keep this level in the future by maintaining the reliability of fuels and the integrity of reactor facilities.

(2) Reduction of Radioactive Liquid Waste Discharge

The liquid waste discharge is controlled at a low level at present; therefore, it is planned to continue existing operation and control method. For further reduction of the tritium discharge generated during resin exchange of the heavy water system and during equipment inspection, various efforts including investigation and review of the waste treatment method will be continued.

(3) Reduction of Radioactive Solid Waste Discharge

In addition to various measures taken to reduce radioactive solid waste generated and stored, following investigation and review will be conducted to continue efforts for reduction of solid wastes:

(i) Investigation of volume reduction and treatment method for spent ion exchange resins

(ii) Investigation of treatment and disposal method for in-combustible miscellaneous solid wastes

8. Outline and Experience of Environmental Radiation Monitoring [VI-8]

8.1 Outline

The major objectives of the off-site environmental monitoring of "Fugen" are the estimation and evaluation on the public exposure dose in the adjacent area based on the safety regulations, etc., and the understanding of the long-term radioactivity change in the surrounding environment. The off-site environmental monitoring includes the continuous measurement of the space dose rate, the measurement of the integrating dose and the periodical measurement of the radioactive concentration in the land environmental samples such as the floating dust, inland water, land soil, index plants and precipitates, and in the marine environmental samples such as the sea water, marine ground soil, index sea creatures and seafood.

Before the full-scale monitoring around "Fugen" was started, the pre-investigation had been commenced in October, 1974 to identify the environmental radioactivity standard before operation. During the pre-investigation, the sampling, pre-treatment and analysis evaluation were performed with support of Tokai Works.

The full-scale monitoring in "Fugen" was started in November, 1977 when the safety regulations and the guidelines on radiation management were established. As for the items to be measured, etc. the measurement of various environmental samples were shared between "Fugen" and Japan Atomic Power Co.'s Tsuruga Nuclear Power Station based on the "Agreement on the Environmental Monitoring Items" concluded between these two power stations, for the management of the supervised area is jointly performed by "Fugen" and Tsuruga Nuclear Power Stations.

Meanwhile, in Fukui Prefecture, the purely technical review of the environmental monitoring results has been performed by the Fukui Prefecture Technical Committee on the Environmental Radiation Measurement (hereafter called "the Measurement Technical Committee") consisting of Fukui Nuclear Safety Management Office, Fukui Nuclear Environment Surveillance Center, Fukui Fisheries Experimental Station, Japan Atomic Power Co., the Kansai Electric Power Co., Inc. and PNC.

The Measurement Technical Committee reviews and evaluates the investigation plans and results of each measurement of the agencies in Fukui Prefecture purely from the technical point of view, and provides the investigation plan reports, quarterly interim reports (for 1st to 4th quarters) and annual reports. These data are reported to the "Fukui Nuclear Environmental Safety Management Association" and widely published after the discussion and confirmation.

A special case was made mention of the environmental monitoring: the case influenced by the nuclear testing in China. Because of the influence of the nuclear weapon testings in China which had been conducted for about 10 years since 1970, the nuclides related to the nuclear weapon testings including Cs-137, I-131, Ru-103 and Ce-141 were detected until around 1982 as the measurement result of the environmental samples such as the floating dust, precipitates, plants and land soil. The significant amount of Cs-137 is still detected in the land soil as of 1997.

During the accident of Chernobyl Nuclear Power Station occurred on April 26, 1986, the monitoring was performed under the special system; the typical FPs (Fission Products) including I-131, Cs-137, Cs-134 and Ru-103 were detected from the airborn dust, precipitates, mugworts, etc.

In the preliminary pre-operation investigation for the surrounding environmental monitoring concerning Fast Breeder Prototype Reactor Monju construction, the sampling, pre-treatment, measurement and report to the Measurement Technical Committee, etc. were conducted from February, 1987 based on the request of support from the Monju Construction Office and completed at the end of March, 1992.

8.2 Organization

The full-scale environmental monitoring was started in November, 1977 under the 6 members organization consisting of three PNC staff and three consigned personnel. This organization was maintained until around 1984. Since April, 1990, nine members, namely four PNC staffs (including one manager) and five consigned personnel (including four consigned personnel for Monju pre-operational investigation) had been engaged in the monitoring work. With the start of Environmental Safety Group (the present Environmental Safety Section in Tsuruga Office) in April, 1991, however, environmental monitoring work for the surrounding area of "Fugen" and "Monju" was unified, resulting in transfer of the environmental monitoring for "Fugen" to this Group.

8.3 Monitoring Methods

In "Fugen", the radioactive materials discharged from the stack and the cooling water discharge tail ends to the environment are strictly checked and controlled per discharge so that the discharged amount shall be sufficiently below the control target value specified in the safety regulations. The environmental monitoring is performed in order to directly check the influence on the environment by

measuring the radiation and radioactivity in the environment. At the same time, the monitoring is planned to pay attention to the standard targets (creatures, etc.) and nuclides which are important to the human exposure calculation and whose behavior in the environment is measurable, focusing on the major decision paths, as well as to obtain the information on the transfer status of the radioactive materials in the environment, the accumulation status from the long-term point of view, etc.

The environmental monitoring is roughly classified into three categories the space dose, the land and the ocean monitorings.

(1) Space Dose Monitoring

In order to measure and evaluate the dose rate change in a short time, one monitoring post[*1] was installed near the supervised area boundary, and another monitoring post and one monitoring station[*2] in Higashiura Area on the opposite coast.

The measured results which are continuously observed in these monitoring posts and station are constantly gathered in the control room computer and their measurement status is constantly watched as well as being automatically processed and recorded. Those continuous measurement data are transmitted to the Nuclear Environmental Safety Telemeter System installed in July, 1995 in the Environment Management Facility of "Monju" Construction Office, and are collected and exhibited as part of whole prefectural data in Fukui Nuclear Environment Surveillance Center through the Fukui Nuclear Environment Information Network. Then, the data are exhibited in the Sub-Surveillance Bureaus of city, town and village offices eleven in number located around the Reinan district as the center, to widely open to the public including the prefecture residents.

On the other hand, to monitor the space dose in the longer period, ThermoLuminescence Dosimeters (TLDs) are installed on 14 points around the site to measure the 3-month integrated dose.

The specific monitoring objectives and methods are shown in the following:

＊1: a fixed observation station made of reinforced conctrte, equipped with a NaI (Tl) scintillation dosimeter of energy compensation (DBM) system (low range monitor) and an ionization chamber type dosimeter (high range moniter), etc., installed at about 3.5m above the ground.

＊2: in addition to the above-mentioned dosimeters, equipped with such apparatus for meteorological observation as an anemoscope-meter, a

hyetometer and a rain-snow gauge.

① Space Integrated Dose

Objective: To estimate the external exposure integrated dose due to the external radiation

Method: Integrated dose is measured every three months using the thermoluminescence dosimeters (TLD) installed on 22 points (including 8 measurement points of Japan Atomic Power Co.) around the site.

② Space Dose Rate

Objective: To estimate the external exposure due to the external radiation and identify the short-term change of the space radiation by the continuous measurement

Method: Continuous monitoring is provided by the monitoring posts installed at four points around the site (including three points for Japan Atomic Power Co.'s measurement) and in Higashiura Area on the opposite coast. In the Area the continuous monitoring is also provided by the monitoring station installed.

(2) Land Monitoring

In order to monitor the accumulation in the environment from the radioactive materials discharged into the atmosphere, the environmental samples including the floating dust, land soil, precipitates and inland water are periodically collected, to check and evaluate that no radioactive nuclides originated form the nuclear power station are detected by γ ray nuclide analysis, etc. following the pre-treatment.

The specific monitoring objectives and methods of various samples are described in the following:

① Airborne Dust

Objective: To identify the internal exposure due to the air inhalation

Method: A dust sampler is installed around the site, and the air is continuously sucked to HE-40T filter paper for about a month (sucked volume : approx. 3,000 m^3). Then the γ ray nuclide analysis is performed with a Ge (Li) detector.

② Precipitates

Objective: To identify the precipitated amount of radioactive materials

Method: A large water basin (0.5 m^3) is installed around the site to

collect the dust and rainwater fallen into the basin per month. The collected dust and rainwater are adsorbed to resin, and γ ray nuclide analysis is performed with a Ge (Li) detector; the tritium analysis is performed with the liquid scintillation measurement equipment, after they are distilled.

③ Inland Water

Objective: To identify the internal exposure due to the intake of drink

Method: The inland water is collected regularly (every 3 months). The collected inland water is adsorbed to resin, and γ ray nuclide analysis is performed with a Ge (Li) detector; the tritium analysis is performed with the liquid scintillation measurement equipment, after it is distilled.

④ Land Soil

Objective: To identify the radioactive material accumulation trend in the environment

Method: The land soil is collected at two points around the site (including one measurement point of Japan Atomic Power Co.) regularly (every 3 months) to be dried and sifted for the γ ray nuclide analysis with a Ge (Li) detector.

⑤ Index Plant (pine needles)

Objective: To identify the radiation level and to supplement the agricultural product investigation

Method: Pine needles are gathered at two points around the site (including one measurement point of Japan Atomic Power Co.) regularly (every 3 months) to be dried, crushed and sifted for the γ ray nuclide analysis with a Ge (Li) detector.

⑥ General Waste Water

Objective: To check whether radioactive materials are contained in the general waste water or not

Method: The waste water in the general drain canal is collected regularly (every month) to be adsorbed to resin; the γ ray nuclide analysis is performed with a Ge (Li) detector; the tritium analysis is performed with the liquid scintillation measurement equipment, after the water is distilled.

⑦ Deposited Soil in General Waste Water

Objective: To identify the radioactive material accumulation trend in the

general drain canal

Method: The deposited soil in the general drain canal is collected regularly (every month) to be dried and sifted for the γ ray nuclide analysis with a Ge (Li) detector.

⑧ Groundwater

Objective: To identify the radioactive material changing trend in the environment

Method: The groundwater is collected regularly (every week) and distilled for the tritium analysis with the liquid scintillation measurement equipment.

⑨ Moisture in the Air

Object: To identify the radioactive material accumulation trend in the environment

Method: The water sample in the dehumidifiers installed in the monitoring posts and the monitoring station is collected regularly (every month) and accumulated for the period of three months. After this sample (collective sample) is distilled, the tritium analysis is performed with the liquid scintillation measurement equipment.

(3) Ocean monitoring

In order to monitor the accumulation of the radioactive waste discharged from the cooling water discharge tail ends of "Fugen" to the ocean, the environmental samples including the sea water, marine ground soil, and sea creatures are periodically collected, to check and evaluate that no radioactive nuclides originated from the nuclear power station are detected by γ ray nuclide analysis, etc. after the pre-treatment. Fig.VI.8-1 shows the environmental sampling points and measuring points around "Fugen" based on the safety regulations, etc.

The specific monitoring objectives and methods of various samples are described in the following:

① Sea Water

Objective: To identify the radiation level and to estimate the potential internal exposure of the seafood through the concentration

Method: The sea water is collected at the cooling water discharge tail ends and cooling water intakes regularly (every three months) and the γ ray nuclide analysis is performed with a Ge (Li)

detector following the chemical treatment; the tritium analysis with the liquid scintillation measurement equipment is performed after the distillation.

② Marine Ground Soil

Objective: To identify the radioactive material accumulation trend in the environment

Method: The marine ground soil is collected at the cooling water discharge tail ends, cooling water intakes, and Tateishi district regularly (every three months) to be dried and sifted for the γ ray nuclide analysis with a Ge (Li) detector.

③ Seafood

Objective: To identify the internal exposure from the food and drink intake

Method: The seafood is purchased during the fishing season to be dried and ashed for the γ ray nuclide analysis with a Ge detector.

④ Index Sea Creature (Gulfweed)

Objective: To identify the radiation level and to supplement the seafood investigation

Method: Gulfweed is collected at the cooling water discharge tail ends regularly (every three months) to be dried and crushed for the γ ray nuclide analysis with a Ge (Li) detector.

8.4 Environmental Monitoring Experience and Its Evaluation

No influence caused by the operation of "Fugen" on the public around the site was identified in the measurement results of the environmental monitoring around "Fugen" conducted from 1977.

Although the measured values are different from region to region depending on the geological properties ,etc. and the dose decrease due to the shielding effect of the snowfall (in winter) is shown in the space dose measurement result, no greater decrease than the one caused by such natural changes was observed. No significant dose increase due to the operation of Japan Atomic Power Co.'s Tsuruga Nuclear Power Station and "Fugen" Nuclear Power Station was recognized.

No nuclide generated from the nuclear power plant was detected in the measurement result of various environmental samples collectd from the land and sea, except for Cesium-137, Iodine-131, etc. detected during the nuclear weapon testing in China, radioactive material leakage by accident at Japan Atomic Power

Co.'s Tsuruga Nuclear Power Station in 1981 and Chernobyl accident in 1986.
As for the tritium, the measured value was within the fluctuation range due to the past nuclear weapon testings.

8.5 Measurement Method Related to Environmental Monitoring

The following describes the outline of the measurement method related to the environmental monitoring around "Fugen"

(1) Spatial Dose Measurement Method

① Spatial Integrated Dose Measurement Method

3 pieces (6 elements) of thermoluminescence dosimeters (TLD) are positioned at each point to measure the integrated dose every 3 months.

② Spatial Dose Rate (continuous measurement)

Central control by means of the telemeter system is performed making use of the energy compensating (DEM) type 2" ϕ X 2" NaI (Tl) scintillation type dosimeter (eliminates 3MeV or more) and the 14.5 l spherical pressurizer type ionization chamber dosimeter which are installed at about 3.5m height from the ground on the roof of the iron reinforced concrete fixed observation station.

(2) Nuclide analysis and measurement method with Ge(Li) detector

① Nuclide analysis and measurement method with Ge(Li) detector

The measurement time is 500 minutes (30,000 seconds) or more and the target period from sampling to measurement is : 3 days or more for airborne dust; within 10 days for precipitates, sea weed, and gulfweed; within 30 days for sea foods. In addition, plastic containers with following specifications are used for the measurement container.

The attached Table VI.8-1 shows the nuclide analysis and measurement method with the Ge(Li) detector.

(i) v-1 (diameter : 60mm, height :30mm)
(ii) v-2 (diameter : 80mm, height :40mm)
(iii) v-3 (diameter : 95mm, height :50mm)
(iv) v-4 (diameter : 130mm, height :77mm)

② Target Detection Value of Nuclide Analysis with Ge(Li) Detector

The attached Table VI.8-2 shows the target detection values of the nuclide analysis with the Ge(Li) detector which is used during the environmental monitoring.

(3) Instrument

The attached Table VI.8-3 shows the types and specifications of the instruments used during the environmental monitoring.

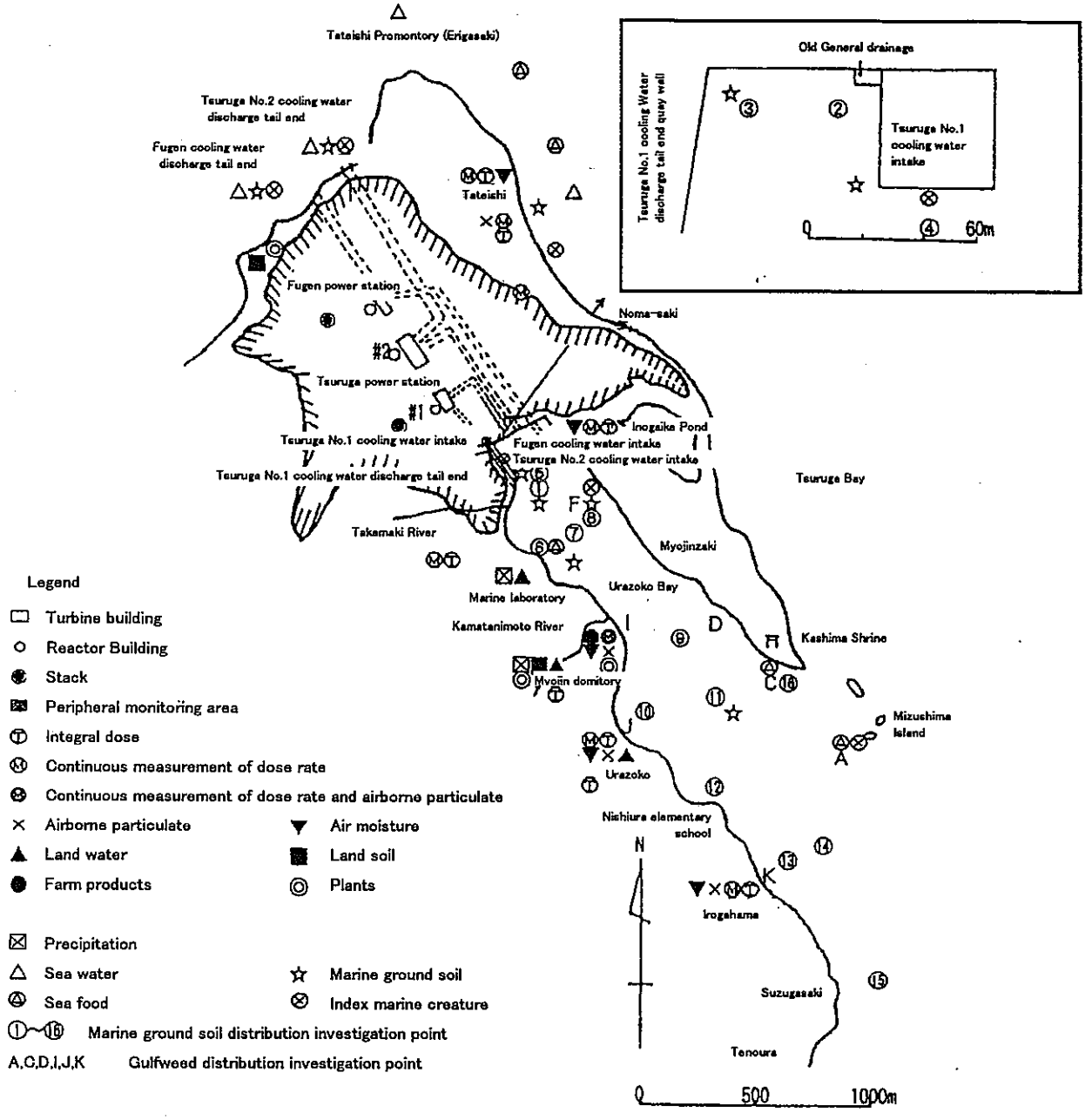


Fig. VI.8–1 Sampling Point around Tsuruga and Fugen Power Stations

Table VI.8-1 γ Ray Nuclide Analysis Method with Ge (Li) Detector

	Sample	Sample Form	Sample Quantity	Nuclide to Be Measured	Reference Nuclide
Land Monitoring	Airborne dust	HE-40T	Approx. 2,000m^3 (Contimuous gathering)	Mn-54,Co-58,Co-60, I-131, Cs-137	Ru-106, Ce-144
	Inland water	Resin adsorpted	5 l	Mn-54,Co-58,Co-60, I-131, Cs-137	Ru-106, Ce-144
	Land soil	Dried and sifted (less than 2mm)	Dry soil more than 300g	Mn-54,Co-58,Co-60, Cs-137	Ru-106, Ce-144
	Plant	Dried substance (crushed)	Row more than 500g	Mn-54,Co-58,Co-60, Cs-137	Ru-106, Ce-144,Ba-140
	Precipitates	Resin adsorpted	Approx. 0.5m^2	Na-22,Mn-54,Co-58, Co-60,I-131, Cs-137	Ru-106, Ce-144,Ba-140
Ocean Monitoring	Sea water	MnO_2 method AMP method	20 l	Mn-54,Co-58,Co-60, Cs-137	Fe-59, Cs-134
	Marine ground soil	Dried and sifted (less than 2mm)	Dry soil more than 300g	Mn-54,Co-58,Co-60, Cs-137	Ru-106, Ce-144
	Fishes	Ashes	Raw more than 1kg	Mn-54,Co-58,Co-60, Cs-137	Ru-106, Ce-144
	Shellfish	Ashes	Raw (deshelled) more than 200g	Mn-54,Co-58,Co-60, Cs-137	Ru-106, Ce-144
	Algae	Dried substance (crushed)	Index creature : Raw more than 1kg Food : Raw more than 500g	Na-22,Mn-54,Co-58, Co-60,I-131, Cs-137	Ru-106, Ce-144,Ba-140 (Gulfweed alone)

Table VI.8-2 Detection Target Value of γ Ray Nuclide Analysis with Ge Detector

	Sample	Na-22	Mn-54	Co-58	Co-60	Cs-137	I-131	Unit
Land	Airborne dust	0.08	0.08	0.08	0.08	0.08	0.20	mBq/m^3
	Inland water	200	200	200	200	200	400	mBq/l
	Land soil	1.5	2	2	1.5	1.5	----	Bq/Kg dry soil
	Index plant	0.4	0.4	0.4	0.4	0.4	0.8	Bq/Kg raw
	Pine needles	0.8	0.8	0.8	0.8	0.8	2	Bq/Kg raw
	Precipitation	0.8	0.8	0.8	0.8	0.8	2	Bq/Kg raw
Ocean	Sea water	----	8	8	8	8	----	mBq/l
	Marine ground soil	2	2	2	2	2	----	mBq/kg dry soil
	Fishes	0.4	0.4	0.4	0.4	0.4	----	Bq/kg raw
	Shellfish	0.4	0.4	0.4	0.4	0.4	----	Bq/Kg raw
	Edible algae	0.4	0.4	0.4	0.4	0.4	0.8	Bq/Kg raw
	Index marine creatures	0.4	0.4	0.4	0.4	0.4	0.8	Bq/Kg raw

Table VI.8-3 Types and Specifications of Measurement Instruments

Space Integrated Dose	・National UD-512 and US-200S
Space dose rate (continuous measurement)	・Fuji NaI(Tl) 2"ϕ X 2" DBM type of temperature compensation ・Fuji 14.5L spherical Type ionization chamber Ar gas of 8atm. [both detectors; with 1mm thick aluminum cover, air sample in extracted to detector at constant temperature]
Tritium	・Aloca LSC-LB2
γ ray spectrometry	・Ortec GEM-30185 HPGe30% : 3 sets ・Seiko EG&G 7800series pulse height analyzer : 2 sets ・NEC PC-9801FA1 : 2 sets
α activity (Pu239 + Pu240)	・Ortec surface barrier type silicon semi-conductor [Detector : BR-024-450-100 Pulse height analyzer : Seiko EG&G 7800series]

Reference

[VI-1] IAEA Safety Series No.50-SG-O7(Rev. 1), Maintenance of nuclear power plants, 1990

[VI-2] IAEA Safety Series No.50-SG-O9, Management of nuclear power plants for safe operation, 1984

[VI-3] IAEA Safety Series No.50-SG-O11, Operational management of radioactive effluents and wastes arising in nuclear power plants, 1986

[VI-4] IAEA Safety Series No.50-C/SG-Q Q13, Quality assurence in operation, 1996

[VI-5] PNC Technical Review No.77, 1991. 3, P23～35, "Radioactive Waste Treatment Technique in Fugen" ; Y. Tsukamoto, et al

[VI-6] PNC TN1410 91-052, "An Outline of Achievements of Fugen Project", Aug. 1991, P38～41 (In Japanese)

[VI-7] N. Higashiura, et al.; "Spent Bead Pyrolysis for volume Reduction and Disposal", (PNC TN3411 97-001), ICEM, 97, Singapole, Oct. 1997

[VI-8] PNC Technical Review, No.69, 1989. 3, P63～64

VII. Core Management/Fuel Management [VII-1] [VII-2] [VII-3] [VII-4] [VII-5]

1. Outline of Core Management

Core management in "Fugen" makes up an essential part of ATR development. Therefore, PNC has consistently carried out the core management and accumulated related know-how.

The objective of the core management is to assure safety of the reload core and fuel integrity.

For each reload core, prediction analysis of reactor physics characteristics is performed. In addition, reactor physics tests are conducted to verify the safety for the maximum excess reactivity and reactor shutdown margin.

The power-up rate of the fuel is limited by applying PCIOMR (Preconditioning Interim Operating Management Recommendations) during reactor startup, while fuel integrity is assured by periodically monitoring the thermal limits during rated power operation. The power-up control on such a basis has proved to be effective, so that no fuel showed any indication of failure in the past.

The core management is implemented in accordance with the flow chart shown in Figure Ⅶ-1-1. Since "Fugen" is a prototype reactor and also has the role of a fuel irradiation reactor, the evaluation and improvement processes have an essential role. Their results are incorporated into the operational methods and future refuel planning.

During the planning process, a basic refueling plan is prepared based on a long-term operational plan and a concrete startup program is made before start of each operating cycle.

Operational data are evaluated and reviewed using core analysis codes. Then modifications are made on the codes to improve the accuracy of prediction, when necessary to do so.

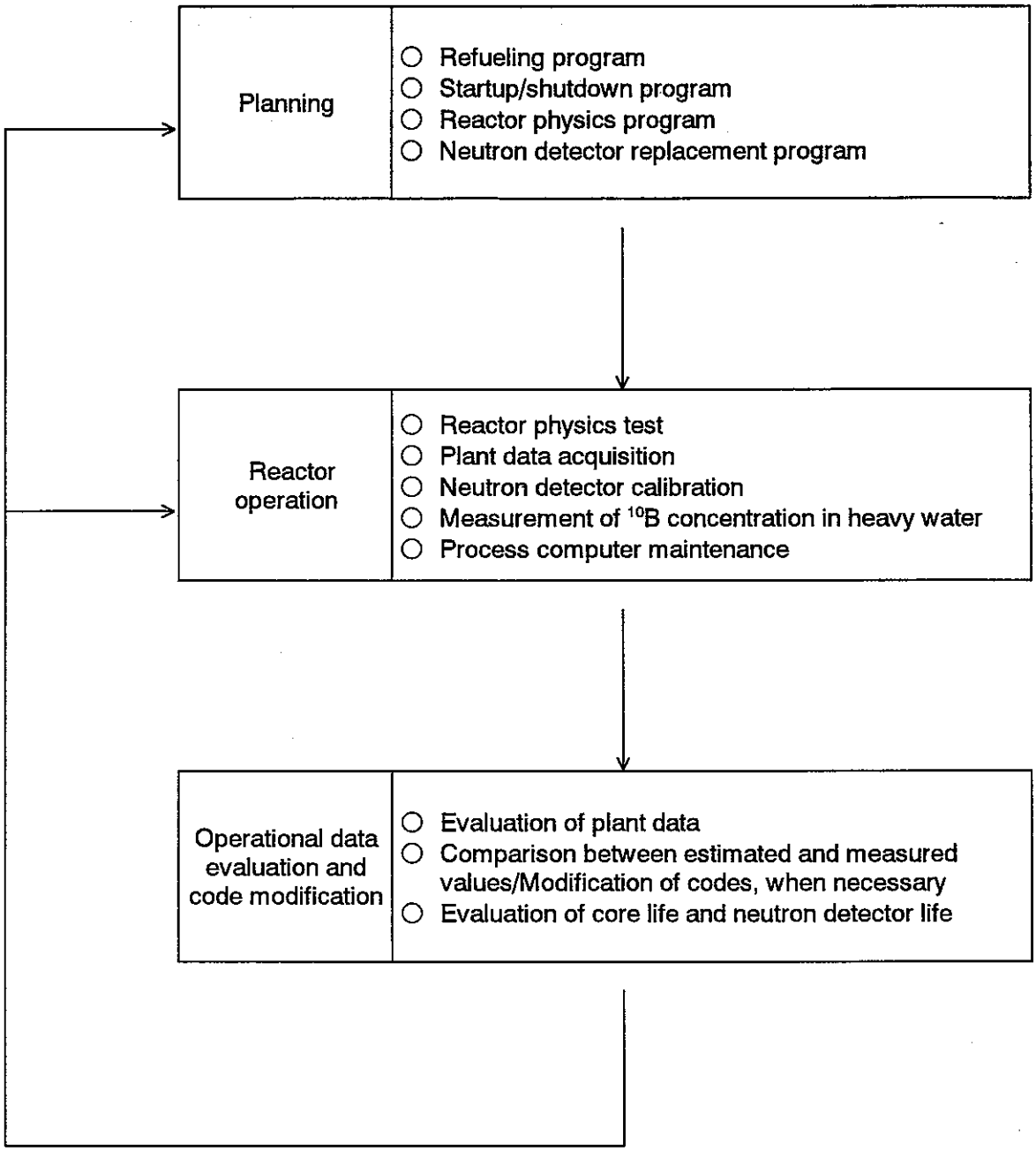


Fig.VII.1-1 Core Management Flowchart

2. Core Management Organizational System

All the activities related to the core management are performed by individual divisions of PNC in collaboration:

Headquarters :	Planning
	Corresponding and Communicating with Governmental Authorities
"Fugen" :	Planning and Programing
	Implementation
	Evaluation
	Corresponding and Communicating with Governmental Authorities
Tokai Works :	Part of Refueling Programing
	(Done by Reactor Core and Fuel Design Section)

PNC owns calculation codes required for the core management. SEs (System Engineers) are positioned in the Technical Section (3 or 4 members for the core management) to perform calculation.

Manpower required for the core management is estimated to be 8.8 men・year. Calculation are performed with support of the SEs and the contractor.

The core management organizational system is as shown below:

"Fugen", Technical Section



3. Core Management Method [VII-5] [VII-6]

3.1 Core Management Method in "Fugen"

The core management is performed in accordance with the safety regulations, the operation management guidelines, the fuel management guidelines and the core management procedures. For strict implementation, core management manuals are also used. The core management activities mainly consist of those for a long-term plan and the others for each operating cycle.

(1) Establishment of Refueling Plan

(a) Refueling and Control Rod Pattern

For each operating cycle, a refueling programs (determines the number, position and fuel type of reload fuel assemblies and the control rod pattern) and a fuel procurement programs based on the refueling programs are established.

The fuel procurement programs is worked out based on a long term refueling plan.

The refueling program is made based on the past operational data for each operating cycle (burnup, incore fuel stay days, etc.). The refueling program for "Fugen" is made based on the scattered 90 degree rotation symmetrical refueling pattern in accordance with following conditions:

(i) Core life required by the plant operation plan

(ii) Thermal limits in view of fuel integrity

(iii) Reactor shutdown margin

(iv) Discharged fuel burnup

The control rod pattern is determined so that fuel thermal limits can be assured through an operating cycle. There is little necessity for use of the control rods for power flattening due to characteristics specific to the heavy water reactor.

(b) Reactor Physics Test During Startup

The maximum excess reactivity of the reload core is measured and its shutdown margin is confirmed during reactor startup. The maximum excess reactivity is evaluated based on the measurement of the critical ^{10}B concentration obtained under cold condition with almost all the control rods withdrawn.

(c) Startup and Shutdown Programs

The power-up control criteria is established based on operational experience of LWRs as well as operational experience and results of fuel irradiation tests in

"Fugen". And power-up operation control is performed under a low power-up rate. Such fuel preconditioning has contributed to preventing indication of fuel failure caused by pellet/cladding interaction (PCI).

To raise the reactor power, ^{10}B in heavy water is removed; this approach leads to a small fluctuation of power peaking factors. This allows easy prediction of thermal parameters such as the maximum linear heat generation rate and the minimum critical heat flux ratio during rated power operation after establishing the control rod pattern during rated power operation at about 40 % of the reactor thermal power. As a result, it enables prompt response to operational control.

Change in the control rod position is limited within the range in which fuels have already been preconditioned.

The power-up contorol limits in accordance with the power-up control criteria are confirmed by analysis in advance because they differ depending on the fuel assemblies. In addition, the core management notice is sent to the Power Generating Section, and notes, etc. are explained at the meeting of assistand shift superrisors are informed to everyone.

The core management notice is also issued before reactor shutdown to inform the Power Generating Section of the reactor power-down rate, etc.

(d) Neutron Detector Replacement Plan

The nuclear life of the neutron detectors loaded in-pile is evaluated to determine the number and position of the detectors (mainly LPM) to be replaced during periodic inspection.

(2) Core Management During Operation

(a) Implementation of Reactor Physics Tests

Criticality test, shutdown margin test, reactor performance test, etc. are performed based on the programs.

(b) Monitoring of Fuel Thermal Limit

During operation, reactor analysis is performed every hour with the process computer making use of plant data to confirm that fuel thermal limits are met for each fuel assembly. The process computer is provided with the on-line core performance evaluation code (ATROPOS). ATROPOS has been developed based on the 3-dimensional nuclear thermal-hydraulic core analysis code LAYMON.

For nuclear calculation, the nuclear constants set up by the lattice calculation codes (METHUSELAH-II, WIMS-ATR) whose accuracy has been evaluated by

Deuterium Critical Assembly (DCA) experiments, etc. are used. For thermal-hydraulic calculation, thermal-hydraulic analysis codes and critical heat flux evaluation equation whose accuracy has been evaluated by HTL (Heat Transfer Loop) experiments, etc. are built in.

Power distribution during operation is determined by correcting analytical values with readings obtained by 64 units of neutron detectors which are located at 16 positions inside the core.

(c) Process Computer Maintenance

Changes of integrated values (burnup, etc.) and nuclear constants for the process computer required for fuel replacement are performed before the start of an operating cycle.

(d) Core Life Management

The ^{10}B concentration in heavy water is periodically (once/month) evaluated by means of the mass analysis, etc. to confirm that the core life is assured as planned.

(e) Neutron Detector Management

The life of the Local Power Monitor (LPM) is evaluated based on the results of LPM calibration performed by the Maintenance Section once a month. In addition, the results of Range Power Monitor (RPM) and LPM calibration are confirmed.

(3) Accuracy of Core Analysis Codes

Core management codes are improved to obtain a constant accuracy of prediction independent of changes in the burnup and the core configuration.

(a) Core Life

The accuracy of core life prediction can be evaluated in accordance with the accuracy to predict the ^{10}B concentration in heavy water. For the core loaded with MOX fuels, it is necessary to quantitatively evaluate parameters shown below because of reactivity changes due to ^{241}Pu disintegration.

(i) ^{241}Am content in fresh fuels contained at fabrication

(ii) Period from fabrication to core loading of fresh fuels

(iii) ^{241}Am generation/depletion of fuels in service

(iv) Reactor shutdown period for periodic inspection, etc.

The accuracy of the core analysis codes was evaluated and improved based on the operational results in "Fugen" and experimental results of DCA. A focus was placed on nuclear characteristics associated with Pu isotopes, ^{241}Am and uranium isotopes to improve the prediction accuracy. As a result, the

accuracy to predict the core life is about ±7 days for 12 month operation (about ±0.2% △k/k in terms of the reactivity), showing good agreement with actual results.

(b) Power Distribution

The accuracy to predict the power distribution showed changes with fuel burning and the increasing number of highly enriched fuels (Type B fuel) loaded after the 4th operating cycle. The following improvements have enabled to get about 5 % of the accuracy:

(i) Review and modification of dependency of the nuclear constants on the coolant void fraction

(ii) Review and modification of the power/neutron flux conversion constants

(iii) Establishment of optimum boundary constants

(iv) Review and modification of a part of nuclear calculation models

Investigation will be continued to further improve the prediction accuracy for high burnup core.

(c) Core Characteristics

The coolant void coefficient and power coefficient are evaluated from the test data of plant response and power coefficient at recirculation pump speed switching, and the accuracy of analysis is confirmed by comparing the predicted values with the above data.

(d) Xe Reactivity Evaluation

The power coefficient in "Fugen" is small and almost all the control rods are withdrawn during rated power operation. The accuracy to predict the reactivity control during reactor shutdown depends on the evaluation of the Xe reactivity. Therefore, the Xe reactivity is evaluated during every reactor shutdown to reflect on making an appropriate reactor shutdown program.

(4) Technical Documents

Investigation and evaluation results related to the core management are described in the FCM documents (Fugen Core Management Memorandum) and then registered as an internal materials.

3.2 Core Management in "Fugen" and Comparison with Other Utilities and Manufacturers

The ability to freely handle the core analysis codes is indispensable for the development of the advanced reactor. As an essential part of the development of

ATR, PNC has been making evaluation and improvement of the core analysis codes by comparing the analytical results with experimental results. As a result, the Technical Section can perform core management activities including planning, reactor operation monitoring, operational data evaluation and analytical code evaluation by themselves. The current status and future trend of LWR core management as well as the current status of "Fugen" core management are shown below:

	1. Planning/ Programing	2. Operation Monitoring	3. Evaluation/Review
Current Status of LWR	Manufactures	Utilities	Manufactures
Future Trend of LWR	Utilities	Utilities	Utilities
Current Status of "Fugen"	Technical Section	Technical Section	Technical Section

4. Core Management Results and Evaluation [VII-6] [VII-7] [VII-8]

4.1 Refueling Plan

(1) Establishment of Refueling Plan

Refueling plans provide a fundamental part of core management/fuel management activities. Being a prototype reactor, "Fugen" has another role as a fuel irradiation reactor. This results in a number of matters to be considered for core arrangement.

When establishing refueling programs for individual operating cycles, engineers with a several years of experience in the core management prepare several draft plans, considering operational data. They investigate various aspects including the control rod withdrawal procedures during reactor startup, power-up method after switching the recirculation pump speed and control rod operation during rated operation, in addition to the conditions shown below; thereby determiming the optimum refueling draft programs. Figure Ⅶ-4-1 illustrates procedures of refueling programs establishment.

The 2-group 3-dimensional nuclear thermal-hydraulic code (POLESTAR) and the 1-gourp 3-dimensional nuclear thermal-hydraulic code (LAYMON) have verified that the draft programs poses no operational problem.

Conditions for establishment of the refueling programs are as follows:

(a) Limiting Values for Operation

① Meet the fuel thermal limits

② Assure reactor shutdown margin

(b) Fuel Management

③ Keep the limit of incore fuel stay days for each fuel

④ Keep the maximum fuel burnup limit

(c) Operation Program and Operation Control

⑤ Meet the core life required by the plant operation plan

⑥ Adopt the 90° rotation symmetrical refueling system

(d) Role as Irradiation Reactor

⑦ Achieve the irradiation target of irradiation test fuels

(e) Fuel Economy

⑧ Plan effective use of fuels (improve discharged fuel burnup)

⑨ Reduce the number of fresh fuel assemblies required

(f) Improvement of Work Efficiency During Reactor Shutdown and Reduction of Shutdown Period

⑩ Reduce the number of unloading fuel assemblies and shuffling one as low as

possible

It would be impossible to realize all the above conditions simultaneously. The priority of the conditions depends on the operation program and irradiation program. For example, fuel shuffling was started from the 7th operating cycle and loading irradiation test fuels from the 8th operating cycle.

Past experience indicates that some of the above conditions cannot realize in following cases:

(i) When planning ⑤ to extend the core life as long as possible with ⑨ reducing the number of fresh fuel assemblies, considering the fuel economy, it is inevitable to raise the fresh fuel power. This may in turn result in some difficulty to meet ① giving little margin against fuel thermal limits and increasing the time of shuffling fuel assemblies.

(ii) When realizing the role of an irradiation reactor ⑦, irradiation test fuels are loaded at the non- rotation symmetrical position. Then it is impossible to keep ⑥, the 90° rotational symmetrical refueling.

(iii) When realizing ① and ⑨, to mitigate the fuel thermal limits without increasing the number of fresh fuel assemblies, ⑩ will not be satisfied by increasing the number of shuffling assemblies.

(iv) Type B fuels loaded at the outermost periphery of the core have to be unloaded before the fuels are fully burnt because of ③, the limit of the incore fuel stay days.

(v) Sufficient reactor shutdown margin is assured for "Fugen" and it will poses few problems on establishing refueling programs. There is no problem for the maximum excess reactivity until operation period exceeds one year.

(2) Automation of Rreparing Refueling Program Document

In order to automatically prepare the refueling program document, an expert system is being under development by accumulating quantitatively evaluated above items, by setting up knowledge and know-how obtained from operational experience and by combining them with the genetical algorithm which is one of the newest knowledge processing techniques.

It is planned to develop a highly reliable expert system by combining Artificial Intelligence (AI) and core analysis codes.

(3) Fuel Thermal Limits

(a) Maximum Linear Heat Generation Rate

The Maximum Linear Heat Generation Rate (MLHGR) depends on the fuel

type. As the fuel burnup increases, the MLHGR decreases. The MLHGR for each fuel assembly is as shown below:

[Maximum linear heat generation rate]

①	Standard fuel assembly (Type A)	17.5 kW/ft
②	Standard fuel assembly (Type B)	17.5 kW/ft
③	Special fuel assembly	15.0 kW/ft
④	36-rod fuel assembly for irradiation test	15.0 kW/ft
⑤	Segmented fuel assembly for irradiation test	12.0 kW/ft
⑥	Gadlinia fuel assembly for irradiation test (Type I)	15.0 kW/ft
⑦	Gadlinia fuel assembly for irradiation test (Type II)	13.0 kW/ft

(b) Minimum Critical Heat Flux Ratio

The Minimum Critical Heat Flux Ratio (MCHFR) is designed to be 1.9 or more for all types of fuels.

Knowledge related to the crud adhesion rate has been obtained from operational data. Considering this effect, monitoring criteria, specific to "Fugen", have been established, so that they are incorporated into the refueling programs for the 9th operating cycle and later.

Since the amount of crud adhered depends on the Effective Full Power Days (EFPD), MCHFR is represented by a following equation in terms of the EFPD:

Operation monitoring criteria in terms of crud adhesion:

$$MCHFR = 1.9 / (1 - 2.64 \times 10^{-4} \times EFPD)$$

(c) Fuel Power and Power Difference During Rated Power

In "Fugen", the own limiting values are provided for the fuel power (Pmax) and power difference (ΔP) in addition to MLHGR and MCHFR for prevention of fuel failure (Fig. Ⅶ-4-2).

The fuel power during rated power operation (Pmax) represents the linear heat generation rate after the reload core reached the rated power. The power difference ($\triangle P$) represents the difference between the linear heat generation rate at the end of the previous operating cycle and that after the reloaded core reached the rated power. These limiting values have been established because findings indicated that fuel failure would possibly occur when both the Pmax and $\triangle P$ exceed certain values.

The limiting values pose few problems for normal fresh fuel loading or fuel shuffling from the inner core to the periphery. They become an issue when fuel shuffling is made from the core periphery to the inner core.

It has been confirmed that the limiting values are met in "Fugen" for all the

fuels because fuels loaded at the outermost periphery may be relocated to the inner core for effective use of fuels.

(d) Maximum Fuel Assembly Burnup

The maximum fuel assembly burnup depends on the fuel type. Predictive calculation is performed in advance to confirm that the maximum fuel assembly burnup would not be exceeded during operation period.

The maximum burnup for each type fuel assembly is as follows:

[Maximum fuel assembly burnup]

①	Standard fuel assembly (Type A)	20000 MWd/t
②	Standard fuel assembly (Type B)	20000 MWd/t
③	Special fuel assembly	30000 MWd/t
④	36-rod fuel assembly for irradiation test	35000 MWd/t
⑤	Segmented fuel assembly for irradiation test	30000 MWd/t
⑥	Gadlinia fuel assembly for irradiation test (Type I)	40000 MWd/t
⑦	Gadlinia fuel assembly for irradiation test (Type II)	30000 MWd/t

(e) Incore Fuel Stay Days

The limit of incore fuel stay days differs depending on the fuel type and it is confirmed in advance that the loaded fuel would not exceed the limits during operation period. The incore fuel stay days do not include the cold shutdown period.

The incore fuel stay days for each type of fuel assemblies are as follows:

[Incore fuel stay days]

①	Standard fuel assembly (Type A)	1500 days
②	Standard fuel assembly (Type B)	1500 days
③	Special fuel assembly	1500 days
④	36-rod fuel assembly for irradiation test	2375 days
⑤	Segmented fuel assembly for irradiation test	1980 days
⑥	Gadlinia fuel assembly for irradiation test (Type I)	2600 days
⑦	Gadlinia fuel assembly for irradiation test (Type II)	2200 days

(f) Other Design Conditions

Other design conditions on establishing refueling programs in addition to the above limiting values of fuel performance are as follows:

① Maximum excess reactivity 0.16 △k/k or less

② Reactor shutdown margin (with one rod stuck)
0.01 △k/k or more

③ Reactivity required during reactor shutdown

0.001 △k/k or more

(E = approx. 65%, C = approx. 75%)

The maximum excess reactivity would be less than 0.16 △k/k.

Past operational results indicate that the excess reactivity after 6 months operation is usually about 0.08 △k/k, so that there may be no possibility that the reactivity exceeds the above limits.

The reactor shutdown margin with one rod stuck needs over 0.01 △k/k. Past operational results show that it is always kept about 0.07～0.08 △k/k except for the beginning of the 1st operating cycle (0.02 △k/k), so that there is no problem.

The reactivity required during reactor shutdown represents the reactivity required to keep the reactor power decreasing rate during shutdown constant. The reactivity of 0.001 △k/k equals to about 0.1 ppm of the poison concentration in heavy water.

Therefore, the time when the poison concentration at the end of cycle reaches 0.3 ppm is defined as the end of the core life, considering the prediction accuracy (±0.2 ppm) of the core analysis code (POLESTAR).

(4) Long-Term Refueling Plan

Fuel procurement plans are prepared in accordance with a long term refueling plan. Accordingly, when establishing the long-term refueling plan, considerations are taken on the amount of Pu supply and the MOX fuel fabrication capability as well as the fact that fuel procurement needs about 2 years period including the licensing procedures.

If the number of reload fuel assemblies and operation period are changed in accordance with changes of the operation plan, the refueling programs for individual operating cycles are reviewed, considering past operational results. In such a case, the long-term refueling plan (for about 6 years, 12 operating cycles) is also reviewed accordingly.

Establishment of the long-term refueling plan requires several years of experience and knowledge related to the core management, especially knowledge about the predictive analysis codes as well as fuel economy, core characteristics and optimum fuel loading pattern. It takes 3 or 4 months to prepare the programs every year.

(5) Refueling Results and Evaluation

"Fugen" was loaded with 469 MOX fuel assemblies (including irradiation test fuels) and 339 UO_2 fuel assemblies (including special fuels) as of June 1990

(after the 15th refueling). At that time, 308 MOX fuel assemblies and 336 UO_2 fuel assemblies were unloaded. This demonstrates use of MOX fuels in "Fugen" (Fig. Ⅶ-4-3).

The ratio of MOX fuels in the core, which was about 34 % at the minimum for the 4th operating cycle core, gradually increased to as much as about 72 % for the 16th operating cycle core.

Even if the ratio of MOX fuels loaded in the core be much changed, the core characteristics would not be significantly affected in "Fugen". This is due to little change in neutron spectrum in heavy water by the fuel type (fuel composition), because neutrons are sufficiently moderated in heavy water.

The results of refueling (up to the 13th periodic inspection) showed that there is no significant difference in use between MOX fuels and UO_2 fuels because both types of fuels have almost the same average discharged fuel burnup.

While the burnup of Type A fuel is about 9～10 GWd/t, it reaches 16～17 GWd/t for Type B fuel. This is not only because of use of Type B fuel but also because of fuel shuffling performed since the 7th operating cycle for effective use of fuels.

4.2 Core Management During Reactor Shutdown

Core management is so performed that the reactor would not reach the criticality during or after fuel loading.

(1) Core Management During Fuel Loading

Predictive calculation is performed in advance to confirm the subcriticality before fuel loading. An inverse multiplication factor curve is made by plotting out indications of the Start Up range Monitor (SUM) to confirm the subcriticality of the reactor during fuel loading. In addition, for preparedness against unexpected trouble, I control rod is withdrawn up to the predetermined position (65 %) in advance.

That is why the reactivity worth of I control rod insertion (from 65 % position) is larger than that of a fresh fuel loaded into the core; thus, if the reactor should reach the criticality, it can be brought into a subcritical condition without fail.

If the excess reactivity is high like the 1st and 8th operation cycles, as can be seen in Table Ⅶ-4-1, poison is so injected n in heavy water as to keep subcirticality. There is no need to adjust the poison concentration for the normal equilibrium core (in case of 6 month operation).

(2) Core Management after Fuel Loading

Criticality tests and shutdown margin tests are performed to confirm that the

maximum excess reactivity is controlled under the limit (0.16 △k/k in Type B core) and the shutdown margin is assured.

To determine the maximum excess reactivity, the poison concentration is measured with all the control rods withdrawn. As shown in Table Ⅶ-4-1, measured and predicted values show good agreement. The reactivity after 6 month operation is about 0.08 △k/k.

The shutdown margin test is performed to confirm that subcriticality of the reactor is assured when one control rod with maximum reactivity is stuck. Therefore, this test is performed under the poison concentration during rated power operation.

In the heavy water reactor, the reactivity required for burning fuels (burnup reactivity) can be compensated by poison concentration and the power coefficient is little affected by the core configuration. As a result, the shutdown margin hardly changes through an operating cycle. Past results show that sufficient shutdown margin, 0.08 △k/k, was assured for each operating cycle.

4.3 Core Management During Reactor Startup

During reactor startup, control rod management to assure the reactor safety, power-up rate monitoring to prevent fuel failure and process computer maintenance for accurate understanding of the core conditions are performed.

(1) Reactor Startup Process

Figure Ⅶ-4-4 indicates reactor startup time. It takes about 140 hours to bring the core into the rated power (for the 10th operating cycle and later). Improved power-up method reduced the time by about 68 hours. This results in an increase of electricity generated for 1 day while assuring the fuel integrity.

Core management involves various kinds of activities during reactor startup. These activities have to be performed in accordance with the reactor power condition in an accurate and timely manner. The activities mainly consist of reactor physics tests, inspection by the governmental authorities, LPM calibration, poison sampling, core data acquisition, and power increase rate monitoring.

A poison control prediction map and a Xe and Sm reactivity prediction map are prepared to accurately perform reactor power control during startup. For that purpose, predictive calculation is performed.

Management items differ depending on the reactor power condition during reactor startup; therefore, the reactor startup management is done in

accordance with individual reactor power conditions.

(2) Core Management Under Low Power Level

(a) Control Rod Control

When withdrawing control rods during reactor startup, the reactivity worth per control rod is determined not to exceed 0.015 △k/k to assure the fuel integrity during a startup accident. To warrant this limit, the control rod withdrawal sequence is adopted for less than about 35% reactor power (less than 30 % feedwater flow rate), which prevents operational errors. In addition, calculations are conducted every operating cycle to confirm that the reactivity worth per control rod is less than 0.015 △k/k since the reactivity worth of the control rod changes with refueling.

If the above limit can not be met, it becomes necessary to change the control rod withdrawal sequence. However, past results show few cases that the control rod withdrawal sequence was changed due to unconformity with the limit. In many cases the sequence is determined so that the most reasonable withdrawal order can be achieved, considering the control rod pattern during rated power operation. That is why there is few possibilities that the reactivity worth exceeds 0.015 △k/k due to relatively flattened overall core power distribution because fresh fuels are loaded in a scattered refueling pattern.

(b) Monitoring During Nuclear Heating

While power control during nuclear heating is performed by control rods, the power-up rate is limited by the reactor coolant temperature change rate (55℃ /h or less). This limit is provided to control temperature changes which affect the piping around the core and to maintain the integrity of the facilities. To meet the above limit, nuclear heating in "Fugen" is performed with aiming at the target of 30℃/h. When the reactor coolant temperature is 250℃ or more, the rate of change is reduced to 15 ℃/h to mitigate the reactivity change inserted into the core and to keep the reactor as stable as possible. The actual results show that the reactor has been controlled under stable condition with a constant rate of rector coolant temperature increase of about 30℃/h.

(c) Evaluation of Coolant Temperature Coefficient

The past results show that the control rod was withdrawn or inserted during nuclear heating. Figure Ⅶ-4-5 shows actual data for nuclear heating during the 9th operating cycle. When comparing with those of the 5th operating cycle,

there are little difference in the coolant temperature changes and heavy water temperature changes. On the other hand, the control rods were temporarily inserted during the 9th operating cycle, while control rod withdrawal operation was dominant during the 5th operating cycle. This is accounted for by the changes of the coolant temperature coefficient, as shown in Fig. Ⅶ-4-6. It is clear that the coolant temperature coefficient depends on the core configuration (types of fuel, burnup). The coolant temperature coefficient changed from negative values during the 5th operating cycle to positive ones during the 9th operating cycle because of a high average core burnup and a large number of MOX fuels loaded. As can be seen from this figure, the coolant temperature coefficient remained at around zero during the 10th and 11th operating cycles.

(3) Core Management at 40 % Power

(a) Measurement of Control Rod Reactivity Worth

The control rod reactivity worth is measured, making use of the control rod pattern formation in accordance with poison injection at 40 % power.

Normally, reactivity worth of only one control rod can be measured due to a restricted time. Figure Ⅶ-4-7 shows comparison between J control rod reactivity measured in the 6th operating cycle and the calculation results obtained by the core analysis code LAYMON. Both values show good agreement within a tolerance, as illustrated in the figure. The control rod reactivity worth curve in "Fugen" looks like a straight line within a power range from 20 to 70% and does not always show a clear S-letter curve. This indicates that the axial neutron flux distribution is rather flat, which may be partially supported by effects of the vibration isolating plate in the calandria tank.

(b) Recirculation Pump Speed Switching

It is set to keep 40 % power for about 40 hours to obtain Xe saturation following reactor startup, to assure stable reactor conditions, to form the control rod pattern for rated power operation and then to switch over the Reactor Coolant Pump (RCP) speed.

RCP speed switching was performed at 45 % power operation until the 8th operating cycle. Reactor scram occurred when the neutron flux exceeded 60 % at RCP speed switching. After that, RCP speed switching has been performed at 40 % power to allow over 5 % margin.

In many cases, RCP speed switching is performed with the control rod pattern

for rated power, in order to reduce the time necessary for control rod position adjustment as short as possible, because power-up operation is performed immediately after RCP speed switching.

(c) Evaluation of Coolant Void Reactivity Coefficient

The coolant void reactivity coefficient is evaluated by making use of changes of the core condition at RCP speed switching. Figure Ⅶ-4-8 indicates the changes of automatic control rods at RCP switching (A loop, low speed to high speed) at the beginning of the 9th operating cycle. If the void reactivity coefficient is negative, positive reactivity is inserted into the core due to abrupt decrease of voids in the core, which results in temporary increase of TPM. However, the automatic control rods (E) are inserted to return the reactor power to the original value and makes the core stable about 2 minutes later. The void reactivity coefficient is evaluated in accordance with changes of E control rods.

The void reactivity coefficient during the 9th operating cycle was evaluated to be $-5 \times 10^{-5} \triangle k/k$. The void reactivity coefficient has been moved to negative from the 4th operating cycle. That is why Type B fuels were loaded and the ratio of MOX fuels in the core increased from the 4th operating cycle.

(4) Management During Power-Up

(a) Core Management of Power-Up Rate

Table Ⅶ-4-2 shows the power-up control limit in "Fugen". The table indicates the limit applied in "Fugen" since the 10th operating cycle as well as the previously adopted limit. "Fugen" adopts the revised power-up control limit and no indication of fuel failure has been observed. Evaluation on operational experience in "Fugen", fuel irradiation data obtained in SGHWR, Studsvic R-2 and Halden, as well as open data of power ramping tests, showed that the fuel integrity could be maintained even if the conventional limit was mitigated. As a result, the power-up limit was revised from the 10th operating cycle.

Figure Ⅶ-4-4 shows comparison between the past startup curve and that with the revised limit.

Up to the 9th operating cycle, the PCIOMR (Per-Conditioned Interim Operating Management Recommendation) was applied over 45 % power and the power increased with 0.4 %/h of power-up rate. In the 10th operating cycle, the PCIOMR was applied over 58 % with 0.58 %/h of power-up rate. Then since the 11th operating cycle, the PCIOMR applied power has been fixed

over 58 % with 0.6 %/h of power-up rate.

As a result, efficient operation can be achieved with proper margin against the limit from the 11th operation cycle.

(b) Monitoring of Thermal Limits

It has been confirmed that the thermal limits are always met during power-up.

MLHGR at rated power can be rather accurately predicted with operational data at an early stage of power-up due to its linearity. On the other hand, MCHFR does not always show linearity, so that operational data alone can not allow easy prediction.

4.4 Core Management During Rated Power Operation

(1) Formation of Envelope

It is believed that if fuels are exposed under a certain power for a predetermined period of time(18 hours), fuel failure can be prevented until its power level against sudden changes of power. To enable automatic operation at rated power, the control rod position is adjusted, based on the evaluation, so that envelopes are formed and overlapped.

Normally in "Fugen", immediately after reaching the rated power, the reactor is automatically operated with E control rods in the range of 60 %～70 % after maintaining the power for about 18 hours with E control rod at 60 % or 70 % position. Since the 11th operating cycle, control rod operation is slowly performed at rated power (until the 11th operating cycle, power is reduced to 85 %) to keep the power change rate (which affects fuels) within the limits. Operational data show that the fuel integrity can be fully assured if the control rod is inserted at a rate of 0.5 %/h and withdrawn at 0.2 %/h. The difference occurs, as shown in Fig. Ⅶ-4-9, due to a large power increase of the fuels around the control rod in case of withdrawal of the control rod.

(2) Monitoring of Thermal Limits

7 types of fuels which have been used require individual management due to different their thermal limits. Figure Ⅶ-4-10 shows an example of linear heat generation rate history (for Type B standard fuel) during the 11th operating cycle. For the fuel under the severest condition in each refueling time, as shown in this figure, the linear heat generation rate during operation is periodically monitored for each type of fuels to assure the fuel integrity.

In addition, the Minimum Critical Heat Flux Ratio (MCHFR) whose limit is 1.9

and operation monitoring criteria related to CRUD deposition are also monitored. Figure Ⅶ-4-11 shows actual values during the 11th operating cycle. For the fuel having experienced MCHFR, its power decreases as burnup progresses and it results in increase in MCHFR values. On the other hand, the MCHFR of the fuels at the core periphery decreases as the power distribution all over the core changes.

(3) Core Life Management

The poison concentrations in heavy water are measured to confirm the prediction accuracy of the core analysis codes and the possibility of operation until the expected date. "Fugen" adopts 2 kinds of poison concentration measurement methods; isotope mass spectrometry and plasma spectrochemical analysis. In addition, a calculation program to make easy determination of the poison concentrations was developed. Boron is used in the from of B_2O_3 with 90 % enriched ^{10}B.

(a) Isotope Mass Spectrometry

Isotope mass spectrometry measures the concentrations of ^{10}B and ^{11}B by adding known specimen (natural boron) to unknown specimen. This method requires pretreatment and removal of radioactive materials. The measurement accuracy has been confirmed to be within ± 0.1 ppm and it is dealt with as a correct value for evaluation.

(b) Plasma Spectrochemical Analysis

This method determines the boron concentration in specimens based on boron specific optical absorption. The measurement is rather easily made within about 30 minutes. To obtain the ^{10}B concentration, it is necessary to multiply the measured value by the isotopic ratio of ^{10}B. The isotopic ratio changes as ^{10}B is burnt during operation. A calculation methodology to obtain the isotopic ratio has been developed and is utilized in "Fugen". And its accuracy is about ± 0.1ppm. These data are dealt with as reference values.

(c) Incore Poison Concentration Calculation Program

This program calculates the incore poison concentrations based on the thermal power and operating status of the heavy water purification column, as well as boron injection and removal conditions. The Technical Section has developed and operated the program. Figure Ⅶ-4-12 illustrates a basic calculation model.

The poison concentrations in heavy water change depending on poison burning in the core, poison injection and removal and poison absorption by

water circulation through the heavy water purification column. These items are programmed to calculate the poison concentrations in heavy water based on their records. The benefits of the program include the ability to calculate minor poison injection/removal operations in addition to reduction of time and cost compared with field measurement of the poison concentrations. It also enables detailed calculations during reactor startup, reactor power changing and control rod pattern changing, so that the incore poison concentrations can be obtained with a high accuracy. This program has been confirmed to have a ± 0.1 ppm accuracy by periodically comparing the calculated results with the measured values during each operating cycle. (It can be utilized online. It has been used for routine management since the 8th operating cycle.)

(4) Process Computer Maintenance

Core performance evaluation is performed at the beginning of each cycle to confirm that the process computer operates normally and all the fuels meet the thermal limits during reactor startup, operation and shutdown. The core performance evaluation is intended to confirm that the calculated thermal power with the process computer and the results calculated from the readings of indicators and charts agrees within $\pm 2\%$.

Both the maximum linear heat generation rate and minimum critical heat flux ratio of the fuels obtained during the core performance evaluation approached the thermal limits at the beginning of operating cycle since the 9th operating cycle. This has been resulted from emphasizing on effective utilization of fuels and on performing fuel shuffling.

(5) Control Rod Pattern Changing

For a normal operation plan, E control rods are automatically controlled within a range between 60 %～70 % while the other control rods except for C control rods located at the outermost periphery of the core are fully withdrawn. Such a control rod pattern is determined when establishing refueling plans. The control rods which are not used for the purpose of power control should be withdrawn as many as possible so that power distribution in the core can be flattened and control rod pattern changing would not be necessary in the operating cycle.

On the other hand, if all the fuels are fresh as in the 1st operating cycle or if irradiated fuels are loaded as in the 9th or 16th operating cycles, the control rods are temporarily inserted for adjusting the power distribution, in addition to the automatic control rods.

In these cases, the control rod pattern is so changed as to withdraw control rods getting unnecessary for adjusting the power distribution in the operating cycle, thereby extending the core life as long as possible.

In "Fugen", the control rod pattern is changed usually once during a operating cycle; during the 16 operating cycle, however, the control rod pattern was changed twice because gadlinia fuels for irradation test are loaded in the center and the 1st region of the core.

When changing the control rod pattern, the reactor power was once reduced to 40 % power until the 9th operating cycle. However, during the 16th operating cycle, the control rods were slowly operated maintaining the rated power.

Such an operation method is derived by the investigation result that slow withdrawal of the control rod (M) within the limit of the fuel power-up rate could assure the fuel integrity. As a result of investigation, during the 16th operating cycle, the M control rods were withdrawn at a rate of 0.2%/h. This has demonstrated the feasibility of control rod pattern changing without lowering the reactor power and resulted in a minimum loss of the revenue by electricity sales.

Fig. Ⅶ-4-13 shows comparison between the control rod pattern changing with/without lowering the reactor power.

(6) Neutron Detector Maintenance [VII-9]

(a) LPM Calibration

The Local Power Monitor (LPM) degrades gradually because of neutron irradiation. Therefore, it requires periodic (once a month) calibration. For calibration of the LPM, LPM indications are compared with those of PCM inserted into the core. Calibration constants are automatically calculated by the process computer. The long life type LPM (LLPM) is used and the degradation of indications per month is below about 5 % at its begining of life. However the degradation rate is getting lower with reactor operating days due to the effect of conversion of ^{234}U to ^{235}U.

(b) Service History of LPM

Total 102 (including 16 currently in service) units of 3 types of LPM with different compositions have been used: 32 units of normal type LPM with only ^{235}U, 40 units of long life type with ^{234}U /^{235}U at a ratio of 3, and 30 (including the above 16) units of long life type with different compositions with ^{234}U /^{235}U at a ratio of 4. Since the 7th periodic inspection, only the long life type LPM has been used. Use of the long life LPM with different compositions ($^{234}U/^{235}U$=

4) was started after 8th annual inspection. Since 14th annual inspection all the LPMs were changed to this type of long life LPM.

(c) LPM Spike Noise Generation Status

The LPM in service may generate spike noise. Mainly many of new LMPs temporarily show such a phenomenon. Analyses of the spike noise generation status, experienced from April 1984 to April 1990, showed that about 19 % of the LPM elements used generated noise, 85% of which occurred in B and C elements with high neutron flux. Noise generation mainly occurs within 3 months after loading the LPM; it accounts for about 77% of the total number of occurrences. Noise generation tends to decrease suddenly after 3 months.

It is believed that spike noise is caused by surface roughness of the LPM element cathode, the condition of UO_2 attachment, etc. Most of spike noise has recovered to the normal condition after a while.

Based on the past operation experience, the spike noise generation rate of an element per day is evaluated as follows:

(Spike noise generation rate)

$=77\text{times}/[(144\text{ elements}) \times (1496\text{days})] \fallingdotseq 3.6 \times 10^{-4}$ (times/element/day)

(d) History of Sensitivity Degradation Characteristics

Histories of the sensitivity degradation characteristics of the normal type LPM (with only ^{235}U), the long life type LPM($^{234}U/^{235}U = 3$) and the long life type LPM with different compositions ($^{234}U/^{235}U = 4$) are shown as below.

The normal type LPM reached its end of life at about 3.7×10^{21}nvt (14 % of the initial sensitivity), showing exponential sensitivity degradation with a little dispersion.

The sensitivity degradation of long life type LPM scatters widely and its life was evaluated by extrapolation to be $8 \sim 16 \times 10^{21}$nvt(12×10^{21}nvt on an average).

Although the long life type LPM with different compositions has different inclination of the sensitivity degradation at the beginning of irradiation largely depending on the loading position, etc., the life of the LPM was certainly elongated to the extent of 14×10^{21} nvt on an average.

(e) LPM Life Evaluation

Fig. Ⅶ-4-14 shows the sensitivity degradation characteristics of 3 types of LPMs. As it can be seen from the figure, the life of the long life type LPM exceeds about $9 \sim 17 \times 10^{21}$nvt. The life of each LPM was evaluated in

accordance with irradiation results as follows:

Normal type about 4×10^{21}nvt

Long life type about $(12\pm4)\times10^{21}$nvt

Long life type with different compositions

over about 17×10^{21}nvt (estimated)

Fig. Ⅶ-4-15 shows average thermal neutron exposure per month. Based on these values, the operating years of LPMs would be as follows:

Normal type 1.0～2.2 years (1.3 years on an average)

Long life type 1.9～8.9 years (4.0 years on an average)

Long life type with different compositions

4.0～9.4 years or more (5.7 years on an average)

(However, the position of 34 - 56 is excluded.)

(7) Management of Neutron Exposure of Pressure Tube and Tube Material Specimens

(a) Fast Neutron Exposure of Pressure Tube

Fast neutron exposure of the pressure tube is controlled by making use of the process computer. In "Fugen", fast neutron exposure of the pressure tube is calculated every periodic inspection to provide data for pressure tube monitoring.

Table Ⅶ-4-3 illustrates fast neutron exposure of the pressure tube at the completion of each operating cycle.

(b) Fast Neutron Exposure of Pressure Tube Material Specimens

Pressure tube material specimens in special fuels are exposed to make lead irradiation. Neutron exposure of these specimens is also controlled by the process computer. The calculated values are output from the computer to evaluate the neutron exposure.

4.5 Core Management During Reactor Shutdown

(1) Measurement of Power Coefficient

It is desirable to control the time required for reactor shutdown as short as possible; it takes 4 hours at present to shut down the reactor. The power coefficient is measured just before reactor shutdown. Change of TPM is measured by inserting one of E control rod by about 5 %, during rated power. The power coefficient can be obtained based on the reactivity inserted at that time and power decrease; for example, $-6\times10^{-5}\Delta k/k/\%$ power.

(2) Power-down Rate

There are no special limits on the power-down rate both for the core side and for the fuel side; however it is limited by the temperature change rate of the metal in the 1st stage steam room in the turbine.

The reactor power was lowered at a rate of 0.6 %/min from the rated power to 0% power until the 8th operating cycle. Since the 9th operating cycle, power decrease has been performed at a higher rate of 0.9 %/min from the rated power to 45 % power so as to make the time for switching the feedwater valve as long as possible. This is due to the fact that the reactor power may not be lowered in accordance with thw predetermined curve if E control rods used for reactor power control were withdrawn out to compensate Xe buildup during reactor shutdown.

4.6 Maintenance of Calculation Codes

In "Fugen", many kinds of calculation codes to support routine core management activities as well as major codes used for core design have been developed so that the Technical Section can make use of them.

It is essential to prepare the organizational system under which the above codes can be used at any time and to let personnel/ to get familiar with using these codes so that calculations required for core management can be performed.

Support of system engineers (SE) who have skills in using a main computer is indispensable for utilization of the computers. In addition, education must be continued to keep and improve technique.

At present, as shown in Table Ⅶ-4-4, the system has been so established that major core design codes for "Fugen" can be used by ourselves.

In addition to the above core design codes, we are fully familiar with relatively minor programs which have been developed to support core management activities and to make use them for core management by ourselves.

These codes which support our core management activities include REBECCA (calculates incore poison), ARENA(calculates core noise), LIRAS(calculates LPM exposure), fuel assembly burnup calculation code, power-up rate analysis code, fuel assembly price calculation code, FULOS (calculates poison injection colculation), PCMREAD (normalizes PCM values), accountancy code, fuel assembly manufacture data management code, etc. ; 16 codes in total.

For all the codes, operation manuals are fully prepared. Computer calculation activities are performed with support of 2 or 3 SEs.

4.7 Evaluation of Core Analysis Code Accuracy and Improvements

(1) Evaluation of ATROPOS Calculation Accuracy and Improvements

ATROPOS and LAYMON are used for calculation of the power distribution in the core. Both codes individually have 3 nuclear calculation models, FLARE, ECOH and mixed one. The accuracy of these codes is evaluated based on the difference between expected neutron flux and actual neutron flux measured by PCM. Evaluation consist of nuclear constants and nuclear calculation models. Nuclear constant evaluation has been mainly performed with the FLARE model. The nuclear constants are obtained by fitting the calculation results of METHUSELAH-II and following corrections are made based on the evaluation:

(i) Corrected the dependency of the control rod effect ($\Delta K_{C/R}$) on the coolant void fraction based on the calculation results of the code WIMS.

(ii) Corrected the dependency of the infinite multiplication factor (K_∞) of fresh fuels on the coolant void fraction based on the calculation results of the code WIMS.

(iii) The dependency of the infinite multiplication factor change rate (ΔK_∞) with fuel burnup on void fraction was determined as 1/2 of the calculation results of METHUSEAH-II.

(iv) The Conversion Factor of fuel segment Power/LPM Neutron Flux (FACT) was obtained by fitting the calculation results of WIMS-multi-cell model.

The above corrections were also incorporated into the constants for the process computer.

Improvements shown below were made on ATROPOS-1(FLARE model). As a result, less than 5 % accuracy was obtained in terms of RMS (root mean square error) to the actual values of PCM (Power Calibration Monitor : thermal neutron flux) distribution measured during 1st to 11th operating cycles:

(i) Correction of two-group spectrum.

(ii) Modification of axial average differential Kernel model.

(iii) Detailed correction of heavy water FACT at the location of PCM adjacent to heavy water.

(iv) Correction of control rod SUS follower.

(v) Correction of 1, 16 nodes spectrum.

(vi) Correction of spectrum of fuels adjacent to heavy water.

(vii) Usage of the optimum albedo obtained with the above models from BOC to EOC of the 8th operating cycle.

Evaluation of the nuclear calculation models was made on the FLARE, ECOH and mixed models.

Followings show the results of burnup calculation of the 1st to 7th operating cycle with the boundary conditions which were optimized based on the data measured in the 1st operating cycle.

(i) For the root mean square error (RMS) of the predicted PCM neutron flux, ECOH and mixed model showed 5～7 % while FLARE model showed higher errors over all the cycles by 6～9 %.

(ii) FLARE model underestimates power mismatch in the scattered core compared with ECOH and mixed model. This effect remains on the power distribution which was corrected by measured PCM neutron flux distribution.

The accuracy of ATROPOS-2 was evaluated from the 1st to the 8th operating cycle with the nuclear constants based on WIMS-ATR. As a result, the accuracy was confirmed to be almost the same as that based on METHUSELAH-II.

(2) Evaluation of POLESTAR Accuracy and Its Improvement [VII-8]

POLESTAR is a two-group three dimensional nuclear thermal-hydraulic code. PNC developed the code and have performed various evaluation and improvements of the code. It is used for prediction and evaluation of the core life, thermal power distribution, and reactivity coefficients. Its calculation accuracy is confirmed based on a number of operational data obtained since the 1st operating cycle. Following improvements have been added to the code:

(i) Detailed evaluation of control rod constants.

(ii) Introduction of ^{241}Am accumulation effect due to decay of ^{241}Pu.

(iii) Introduction of spatial dependency of the Doppler effect.

(iv) Introduction of dependency of Xe and Sm number density, space and time.

(v) Introduction of modern nodal analysis method.

Core management codes should have a high prediction accuracy. However, such a single code would be the best as to have all the functions required for core management. This is important for making core management efficiently with less manpower.

Furthermore, it is necessary to get skilled in using the codes and to improve them when necessary, in order to ordinarily evaluate and to upgrade analysis accuracy with using operational data.

Analysis accuracy of POLESTAR obtained so far is as follows:

(i) Core life	±7	(EFPD)
(ii) Thermal neutron flux distribution	±5 %	(rms)
(iii) Void reactivity coefficient	$\pm 2 \times 10^{-5}$	(△k/k/% void)
(iv) Power reactivity coefficient	$\pm 1 \times 10^{-5}$	(△k/k/% power)
Reactivity change when changing reactor power	$\pm 2 \times 10^{-4}$	(△k/k)

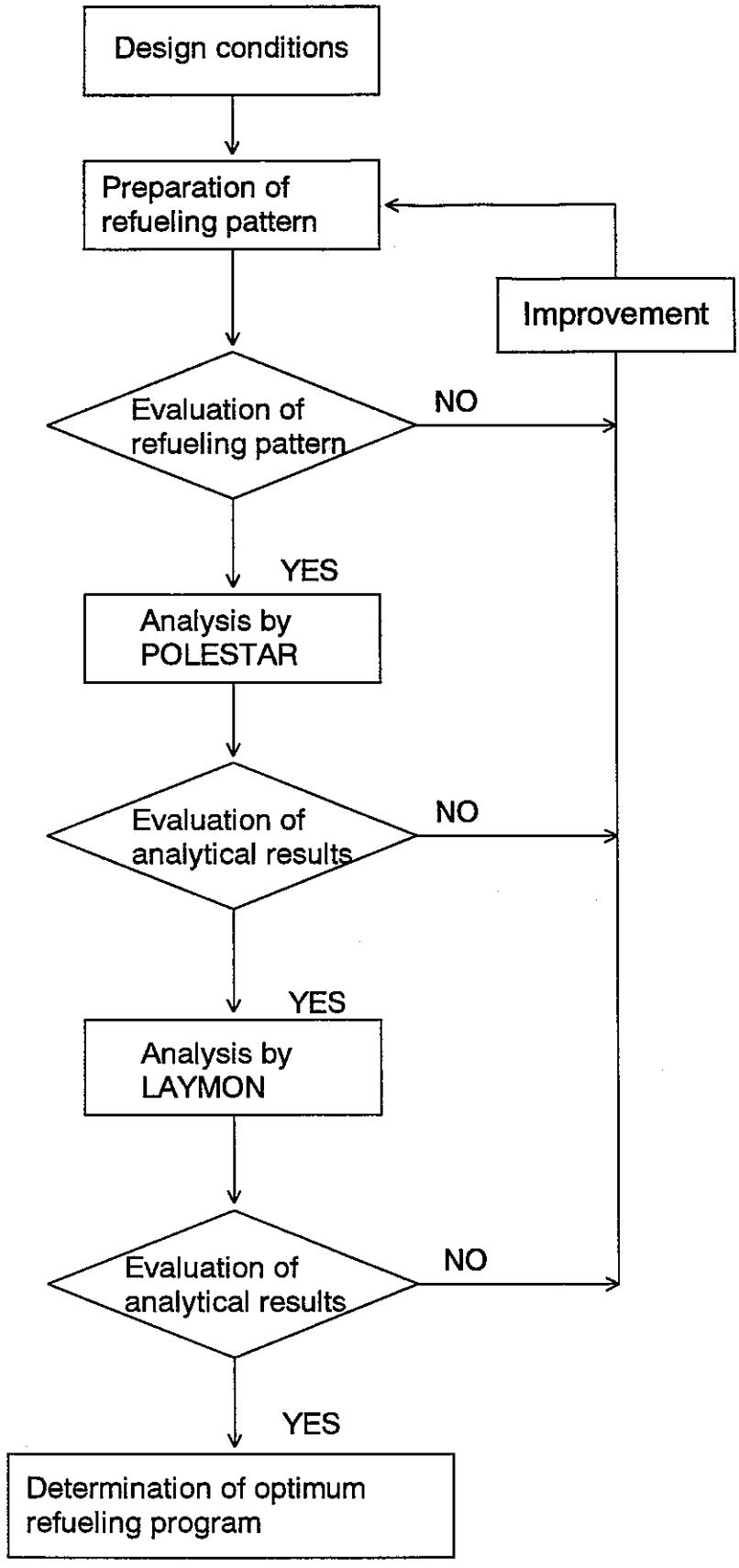


Fig.VII.4-1 Flowchart of Refueling Program Establishment

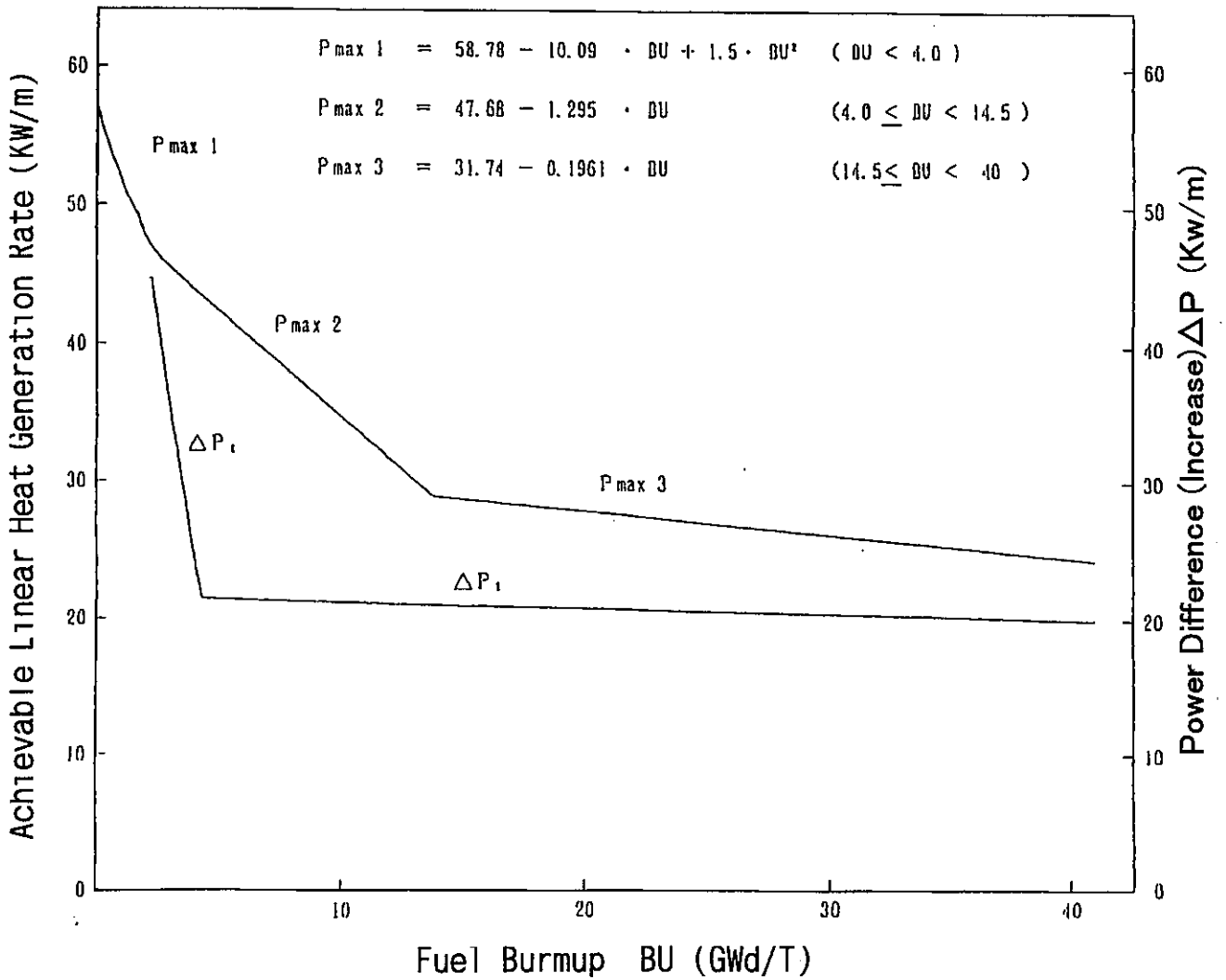
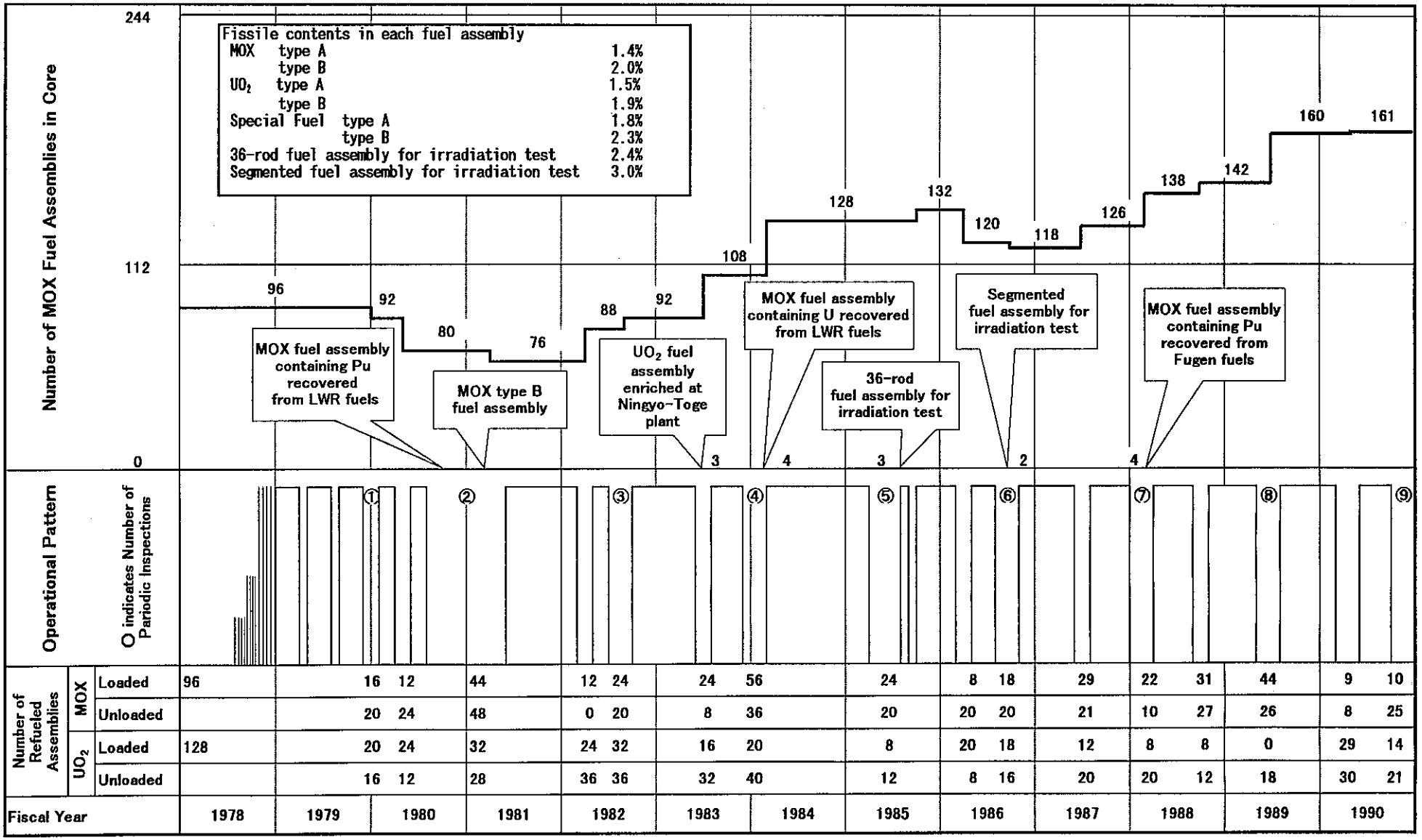


Fig.VII.4-2 Fuel Power and Power Difference During Rated Power Operation



Number of Refueled Assemblies		1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990
MOX	Loaded	96	16 12		44	12 24	24 56		24	8 18	29	22 31	44	9 10
	Unloaded		20 24		48	0 20	8 36		20	20 20	21	10 27	26	8 25
UO_2	Loaded	128	20 24		32	24 32	16 20		8	20 18	12	8 8	0	29 14
	Unloaded		16 12		28	36 36	32 40		12	8 16	20	20 12	18	30 21
Fiscal Year		1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990

Fig.VII.4-3 Fuel Loading History in "Fugen"

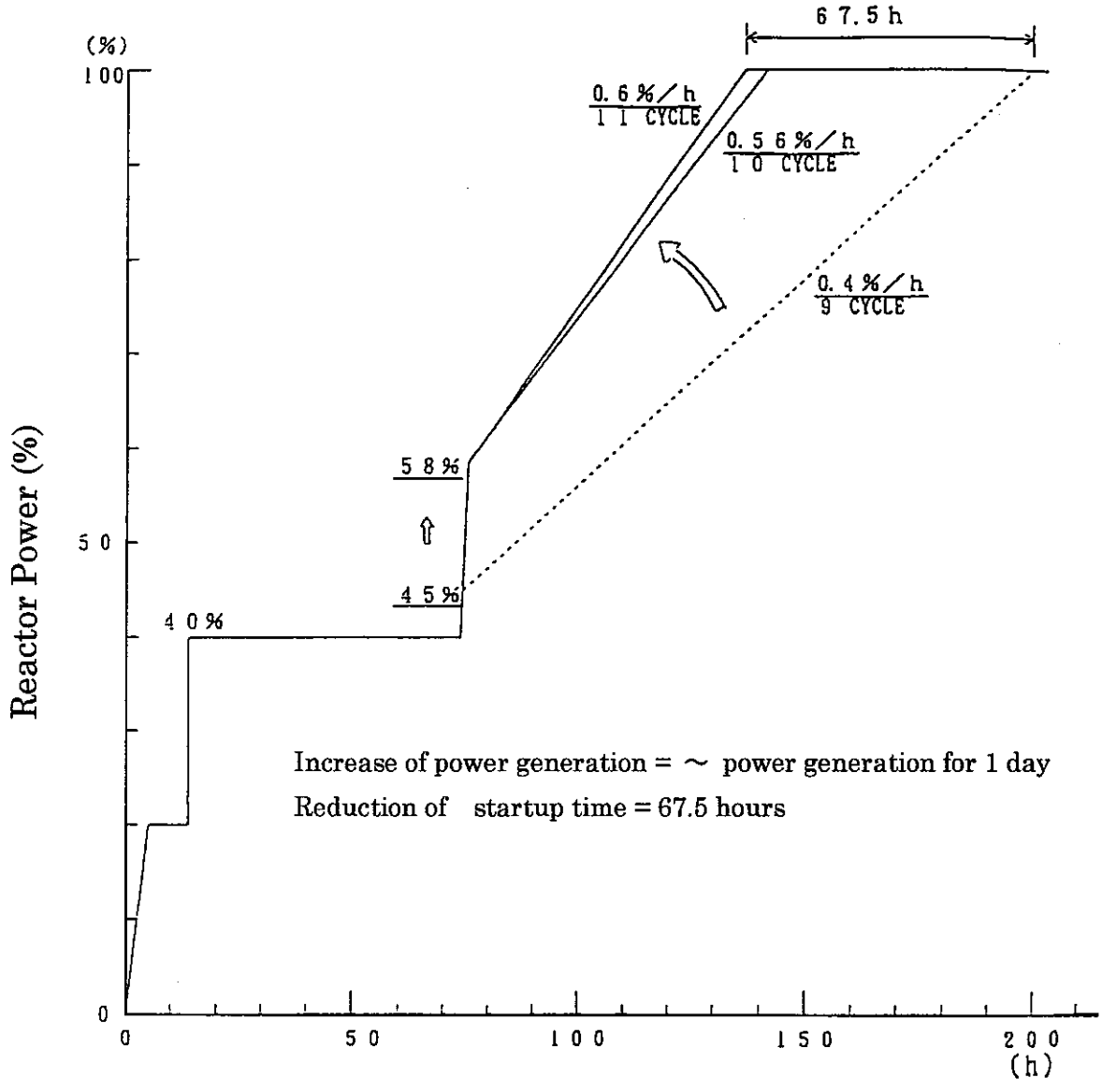


Fig.VII.4-4 Improvement of Power-up Method in "Fugen"

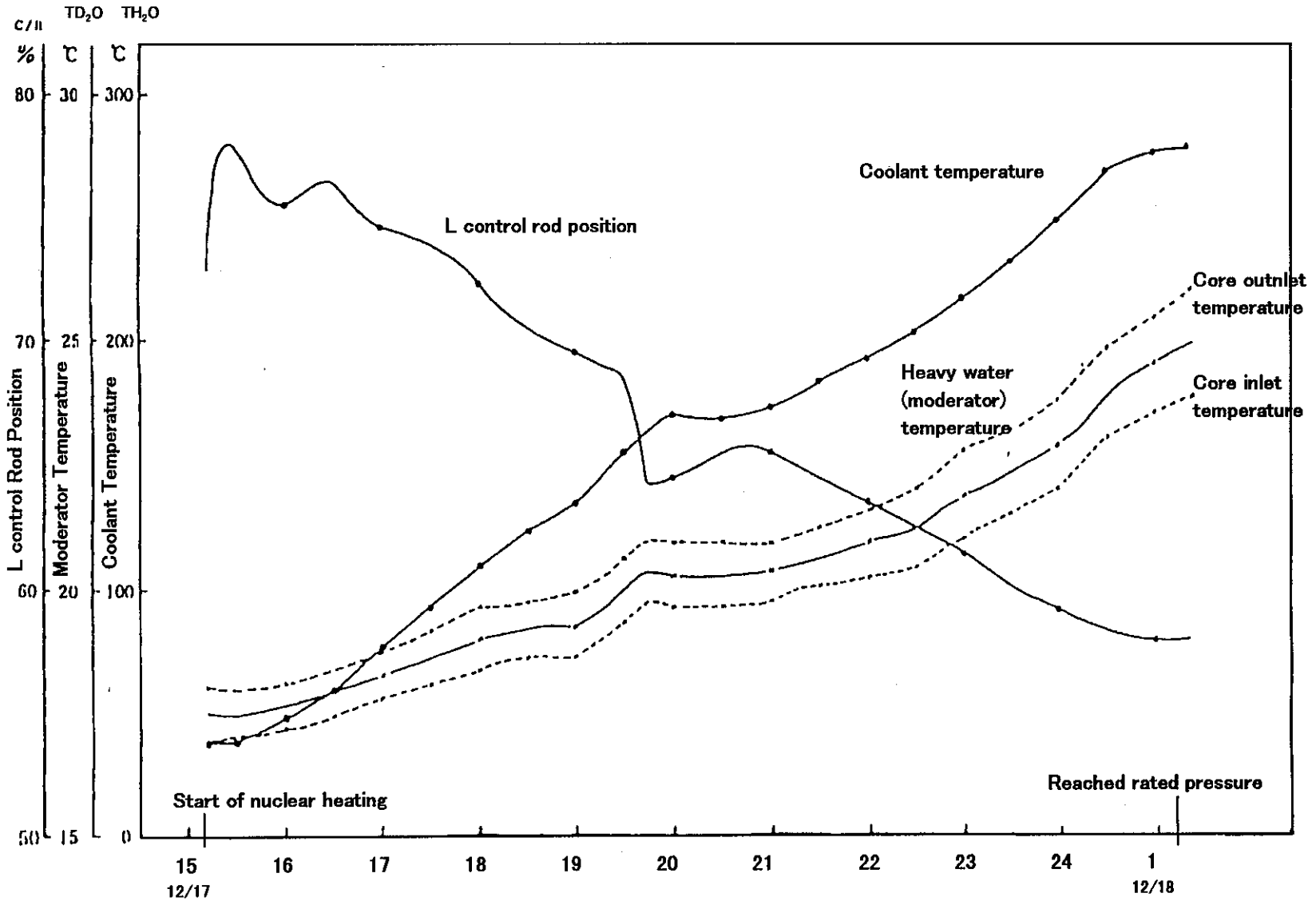


Fig.VII.4-5 Changes of Reactor Coolant Temperature, Heavy Water Temperature, and Control Rod Position at Beginning of 9th Operating Cycle

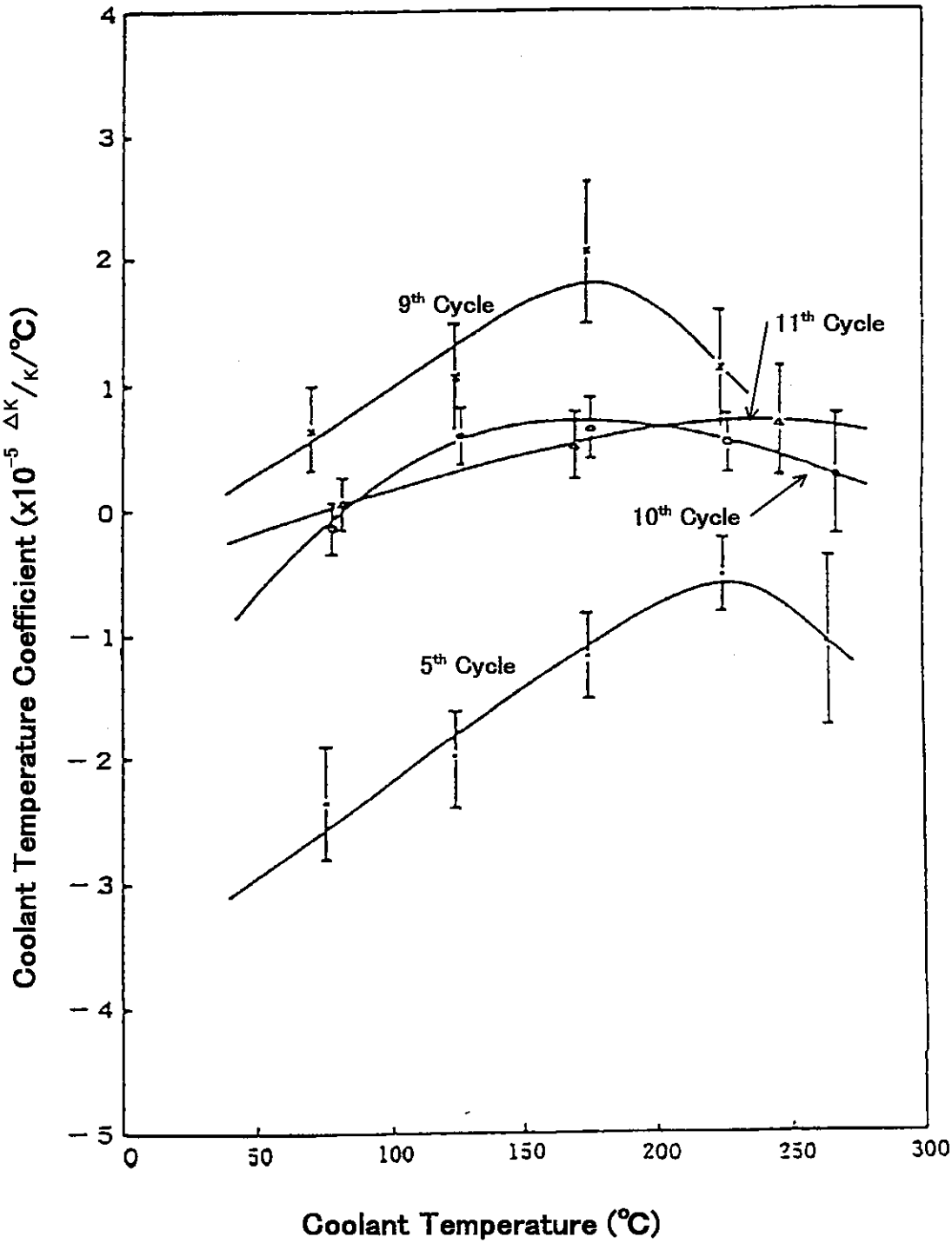


Fig.VII.4-6 Coolant Temperature Coefficient at Beginning of 5th , 9th, 10th, and 11th Operating Cycles

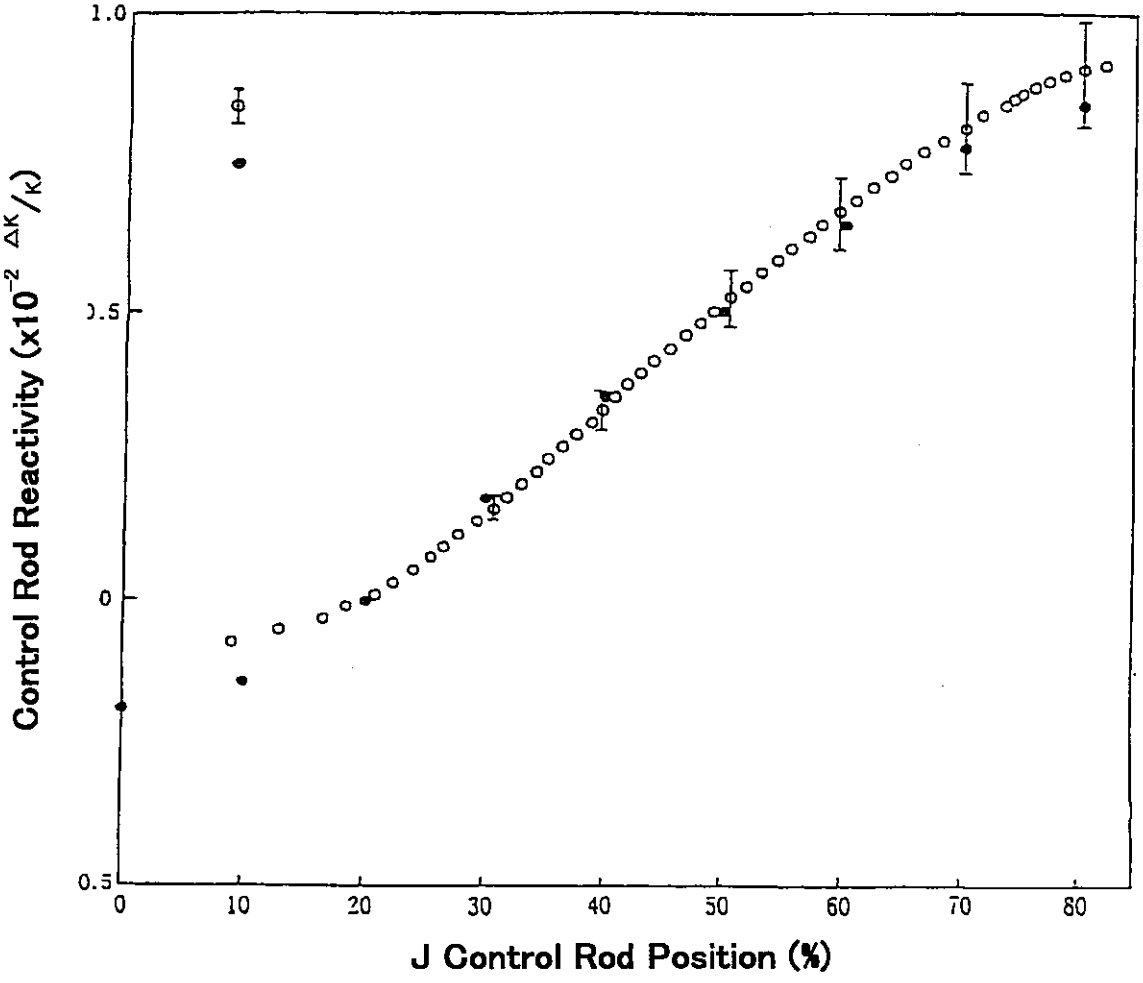


Fig.VII.4-7 Hot J Control Rod Reactivity Curve During 6th Operating Cycle

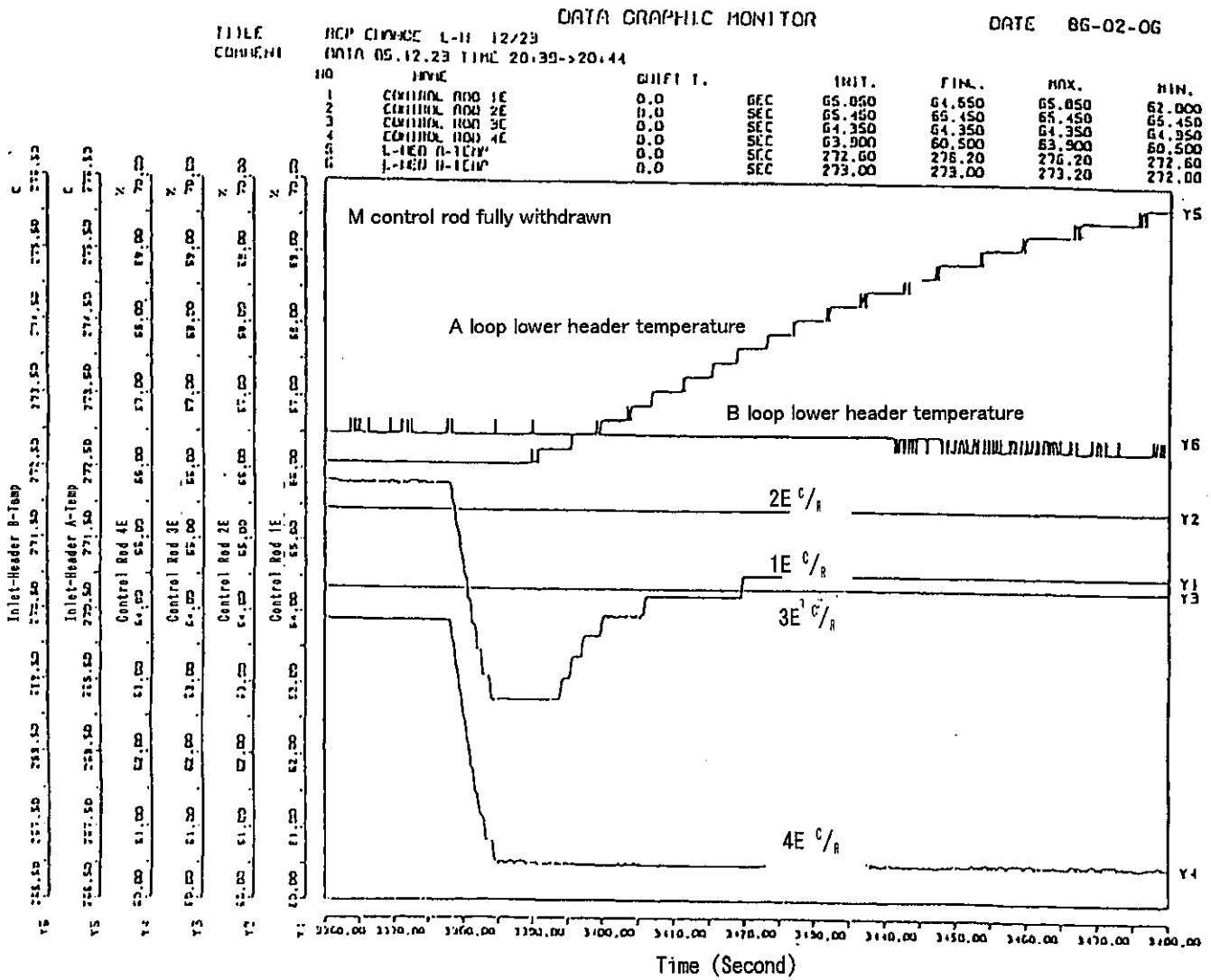


Fig.VII.4-8 Change At Loop RCCP Speed Switching (L→H)

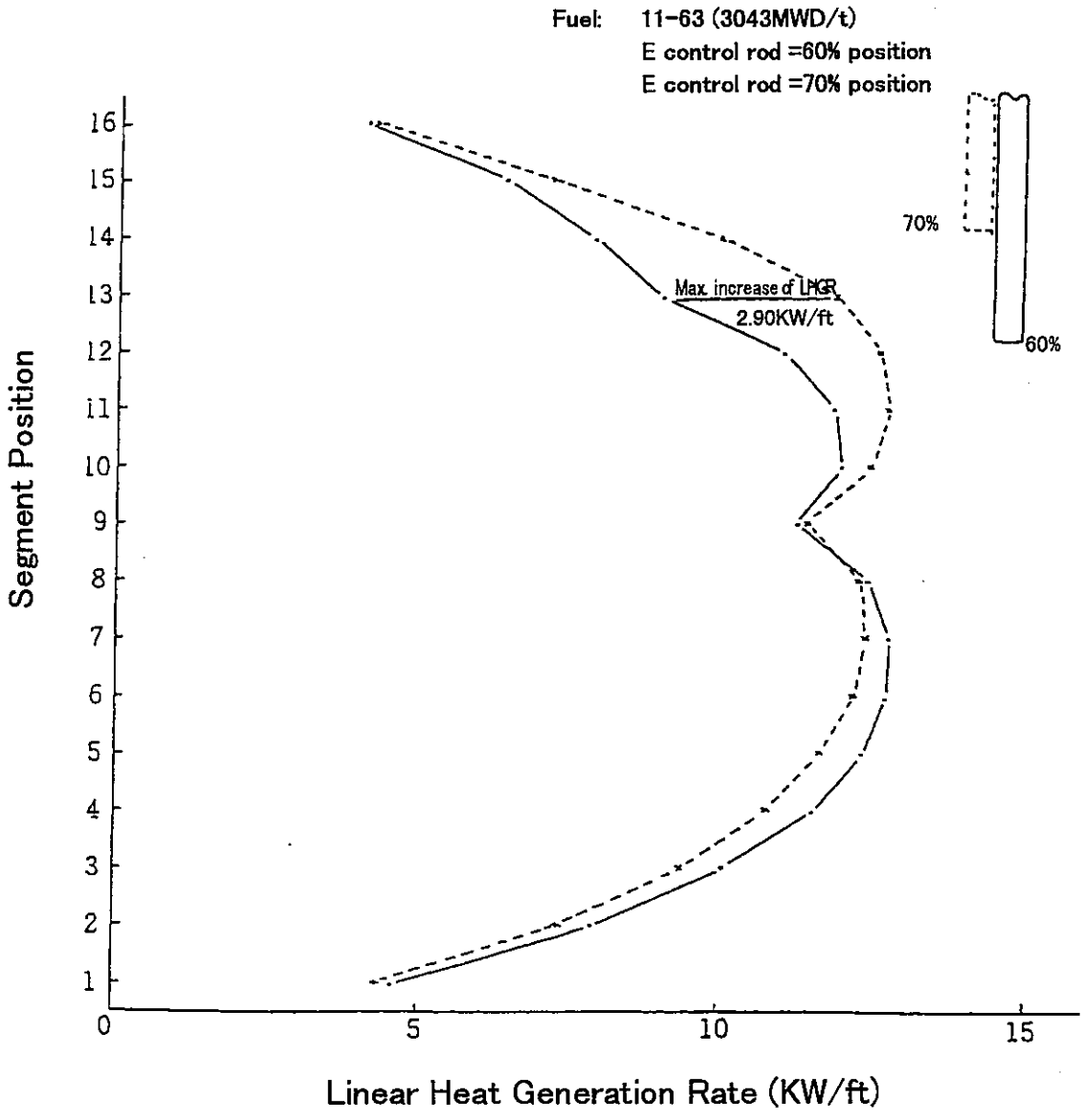


Fig.VII.4-9 Axial Linear Heat Generation Rate Distribution (at Beginning of 11th Operating Cycle)

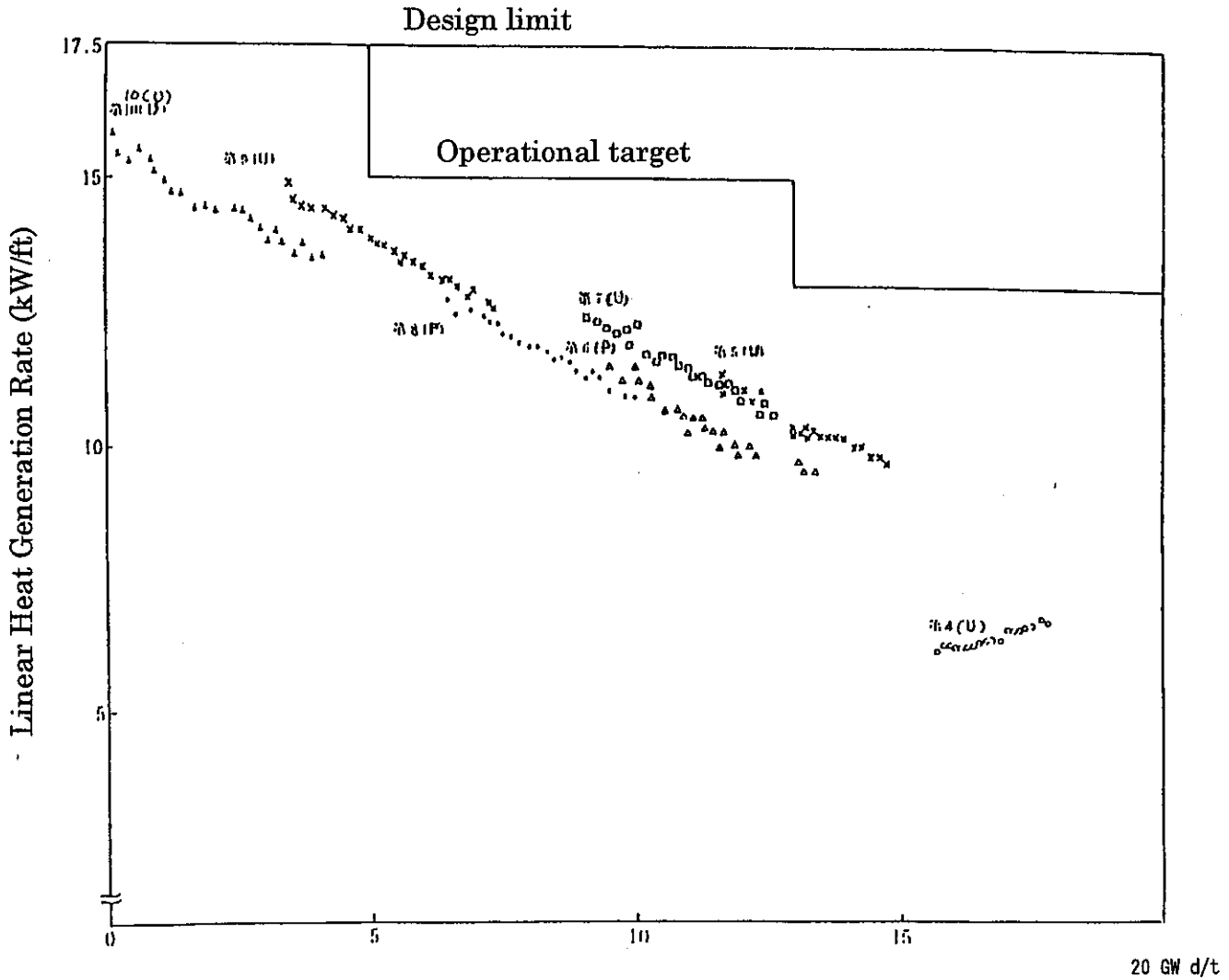


Fig.VII.4-10 Linear Heat Generation Rate History (Type B Standard Fuel Assembly)

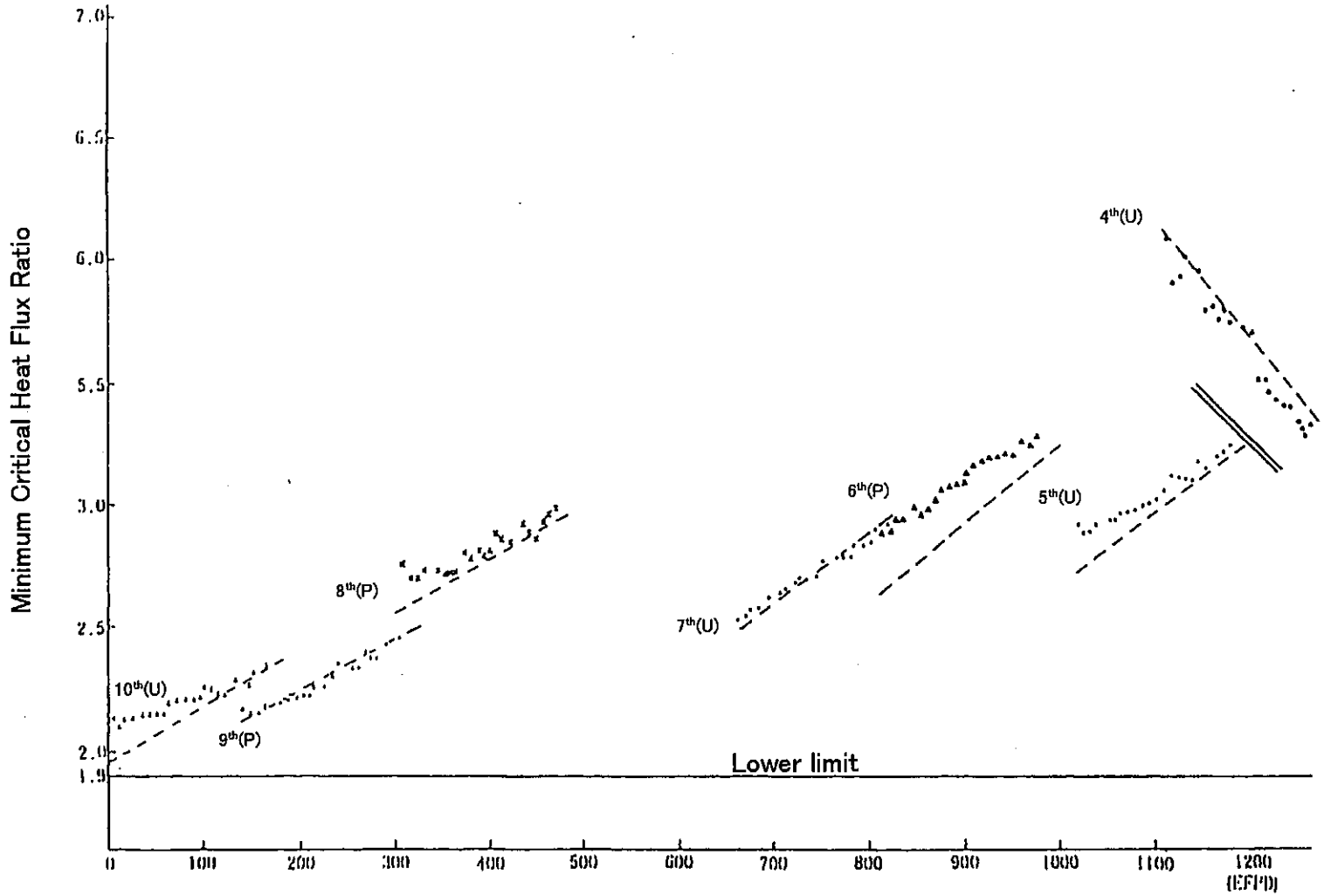


Fig.VII.4-11 History of Critical Heat Flux Ratio (During 11th Operating Cycle)

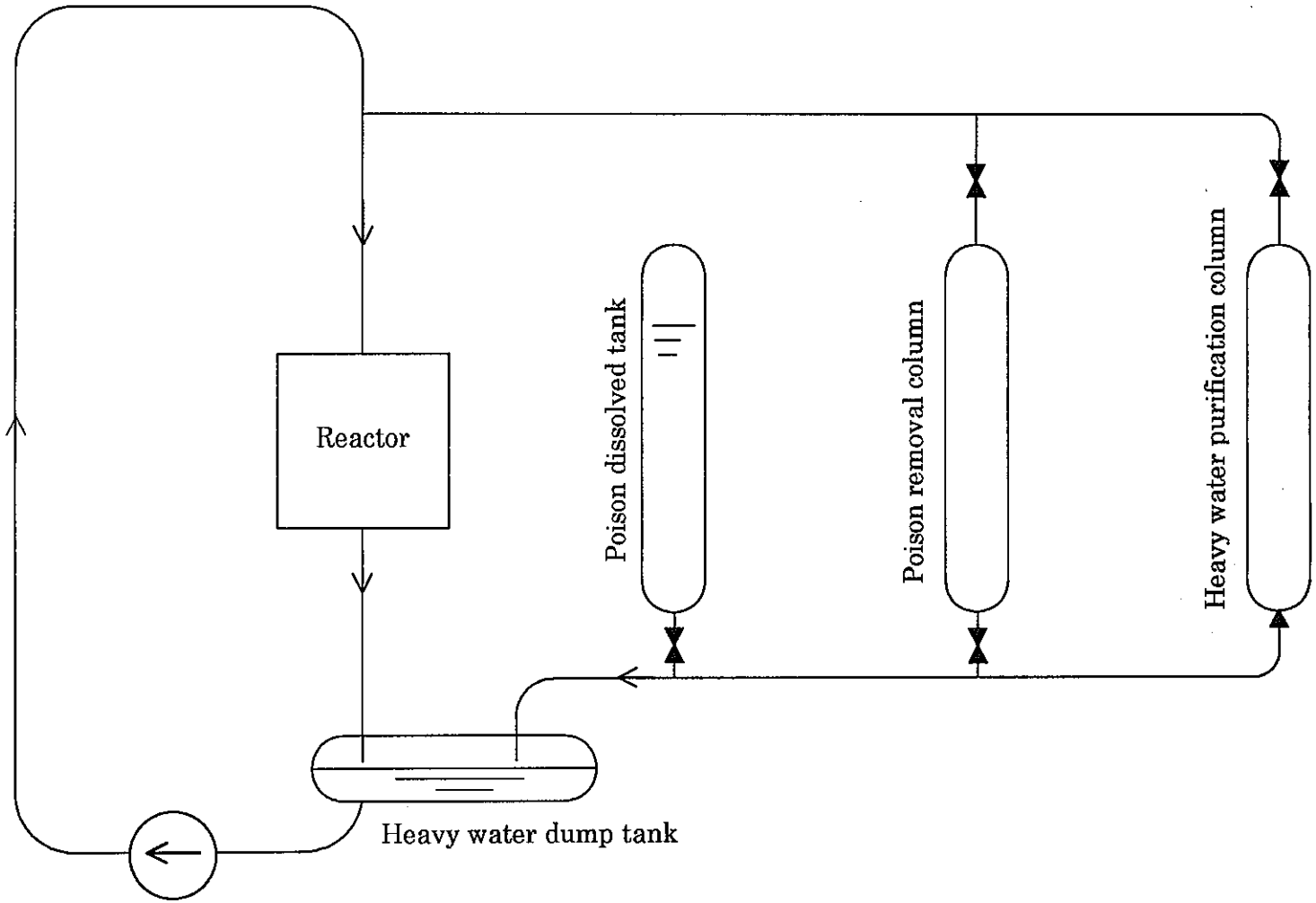


Fig.VII.4-12 Poison Injection/Poison Removal/Heavy Water Purification System Model

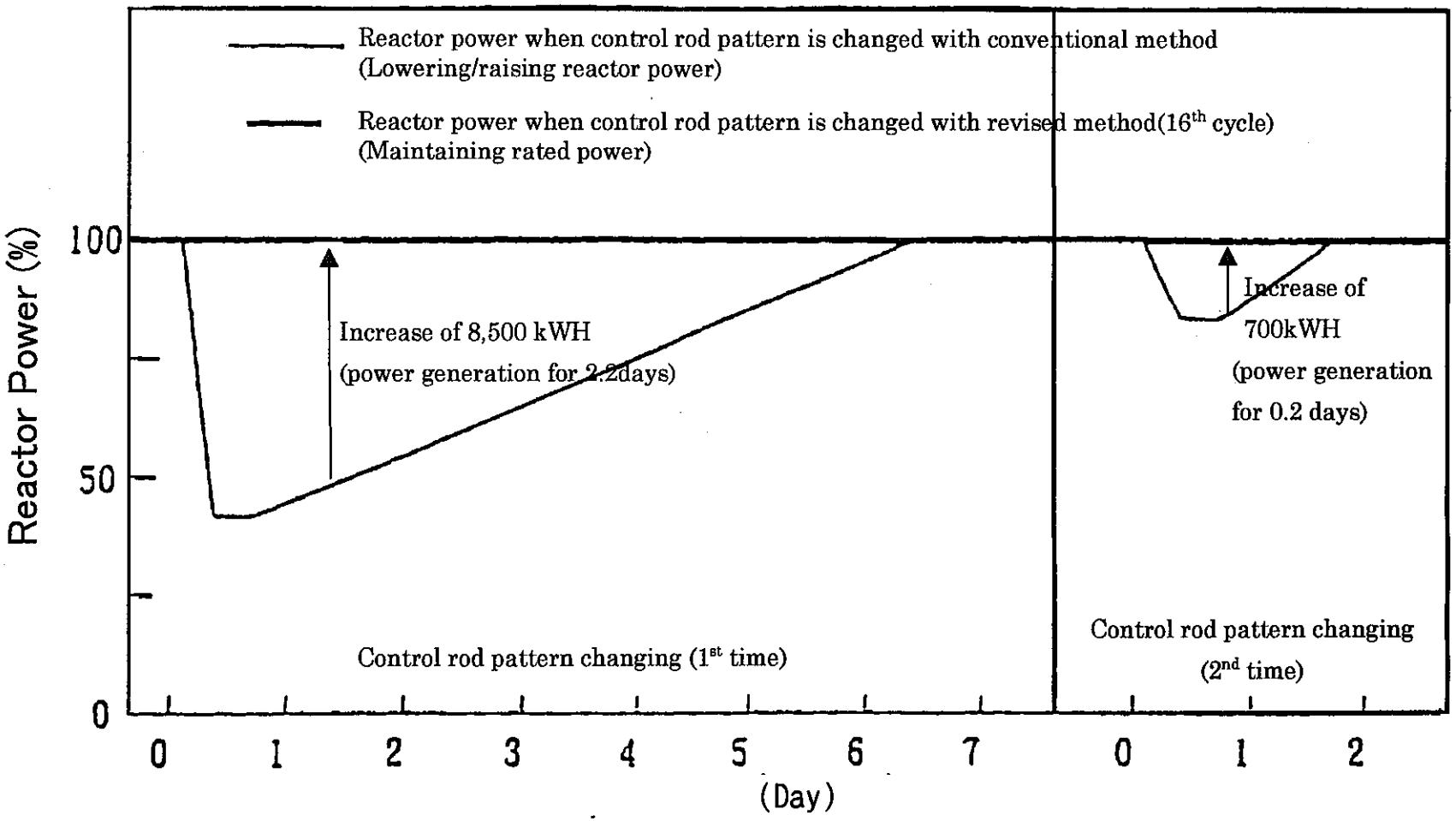


Fig.VII.4-13 Comparison of Control Rod Pattern Changing Methods

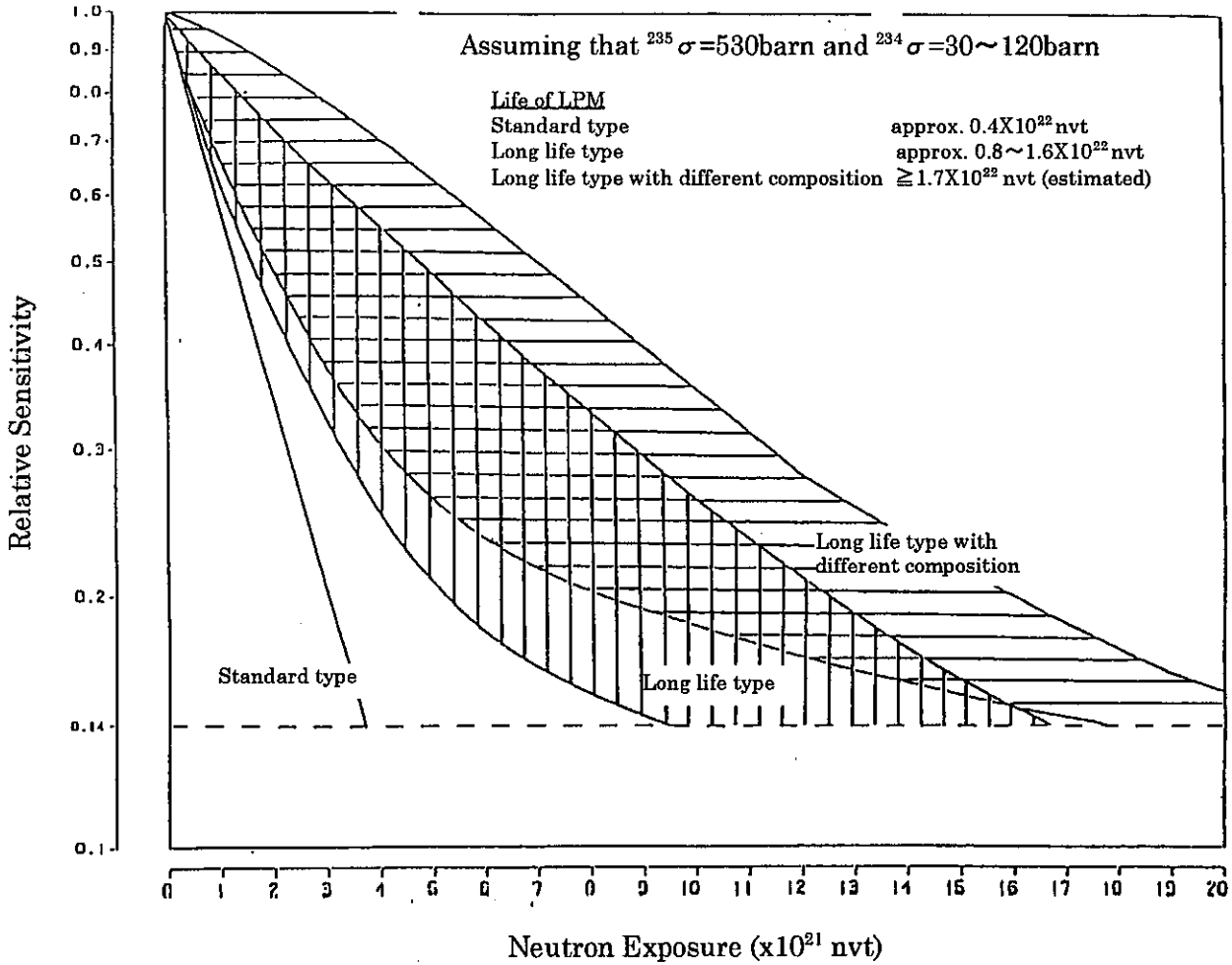


Fig.VII.4-14 Evaluation of LPM Sensitivity Degradation Characteristics

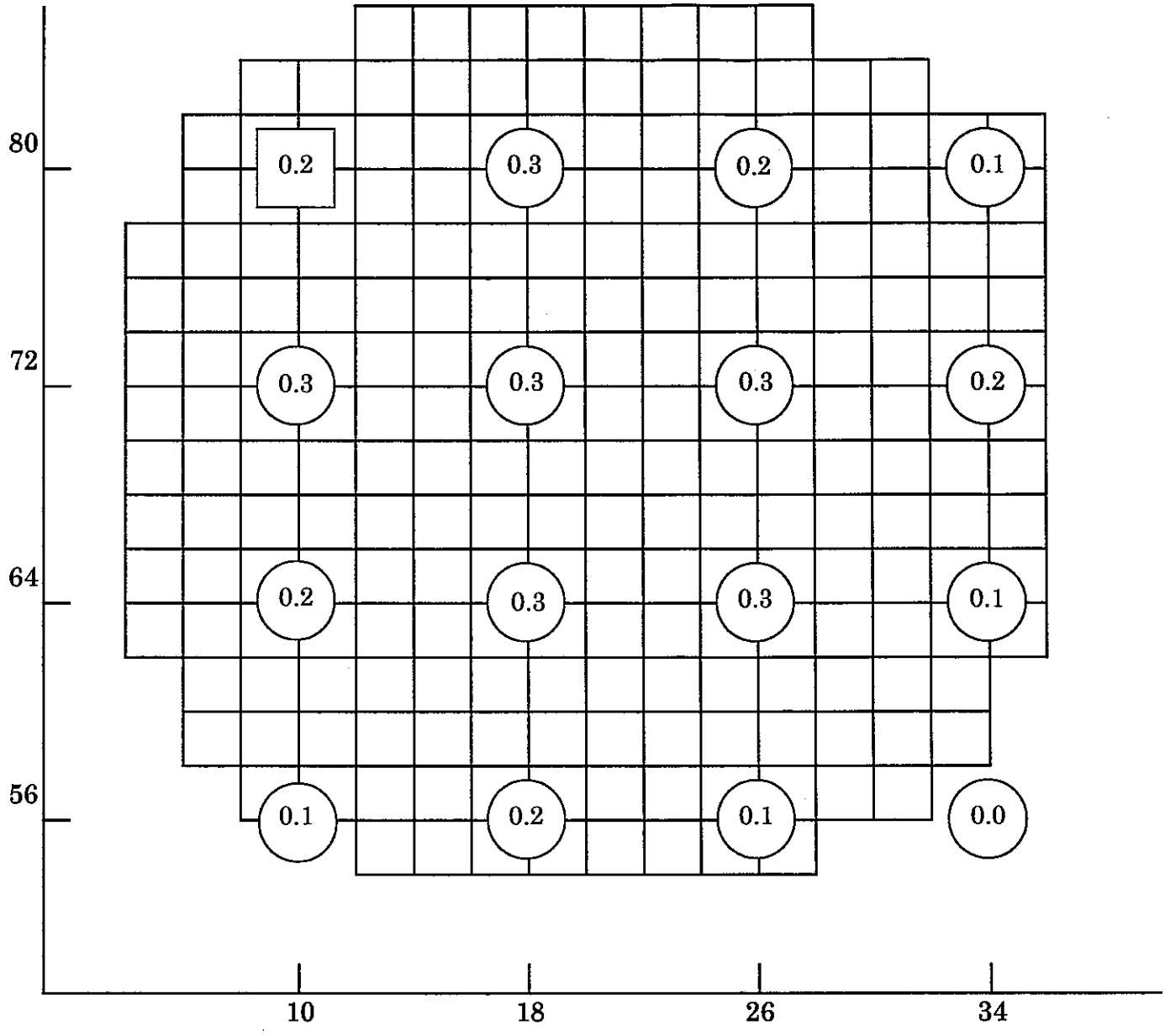


Fig.VII.4-15 LPM Position and Thermal Neutron Exposure per Month

Table VII.4-1 Maximum Excess Reactivity During Cold Criticality

Fuel Cycle	Core Average Burnup (GWd/t)		B-10 Concentration (ppm)	Excess Reactivity Δ K/K	
	Beginning of Cycle	End of Cycle		Estimated Value	Measured Value
1ST	0	5.9	12.9	0.17	0.16
2nd	4.9	5.8	3.0	0.06	0.05
3rd	4.6	6.1	4.3	0.07	0.06
4th	3.0	6.4	9.9	0.13	0.13
5th	4.8	6.3	5.6	0.08	0.08
6th	4.1	7.4	7.9	0.11	0.11
7th	5.7	8.2	6.4	0.09	0.09
8th	4.0	9.8	13.2	0.15	0.15
9th	7.5	10.3	7.5	0.09	0.09
10th	8.2	10.5	6.1	0.08	0.07
11th	7.9	10.7	6.3	0.08	0.08
12th	7.8	10.0	7.2	0.08	0.09
13th	7.8	———	7.7	0.09	0.09

All control rods withdrawn

Table VII.4-2 Revised Limits and Applied Values in "Fugen"

(Associated with P_1 and $\dot{P}$)

	Revised Limit		Applied Value in "Fugen"	
PCIOMR Applied Linear Heat Generation Rate P_1	Fuel rod burnup (MWd/t)	P_1(kW/m)	Fuel assembly burnup (MWd/t)	P_1(kW/m)
	0 ~ 5,000	40 (12.2kW/ft)	0 ~ 4,166	40 (12.2kW/ft)
	5,000 ~ 13,000	32 (9.7kW/ft)	4,166 ~ 10,833	32 (9.7kW/ft)
	13,000 ~ 38,000	26.2 (8.0kW/ft)	10,833 ~ 31,666	26.2 (8.0kW/ft)
Power-up Rate $\dot{P}$	Power-up rate	Fuel assembly burnup	Power-up rate	Fuel assembly burnup
	$\dot{P}$ (kW/ft/h)	BU(MWd/t)	$\dot{P}$ (kW/ft/h)	BU(MWd/t)
	0.1	(BU≦3,000)	0.1	(BU≦2,500)
	$0.1038-1.27\times10^{-6}$	BU(3,000<BU<21,500)	$0.1035-1.524\times10^{-6}$	BU(2,500<BU<17,916)
	$0.0889-0.577\times10^{-6}$	BU(21,500≦BU)	$0.0889-0.692\times10^{-6}$	BU(17,916≦BU)

Table VII.4-3 Pressure Tube Fast Neutron Exposure

(Exposure unit ; x 10^{21} nvt)

End of Cycle	Fuel Segment with Maximum Exposure		Channel with Maximum Exposure	
	Position	Exposure	Position	Exposure
1	19-79-07	0.7840	19-79	0.6229
2	19-79-07	0.9149	19-79	0.7351
3	19-79-07	1.1178	27-73	0.9098
4	13-63-06	1.8293	11-73	1.4505
5	13-63-06	1.9903	11-73	1.5982
6	11-73-07	2.4061	11-73	1.9459
7	13-63-06	2.7044	11-73	2.1870
8	27-63-07	3.3646	17-63	2.7303
9	17-63-11	3.7000	17-63	3.0222
10	17-63-11	4.0326	17-63	3.2886
11	17-63-11	4.4470	27-69	3.6385
12	26-69-11	4.7900	29-69	3.9118
13	29-69-11	5.2704	29-69	4.3046
14	29-69-11	5.6441	27-73	4.6188
15	29-69-11	6.1124	11-71	5.0063
16	29-69-11	6.5605	29-69	5.3635

Table VII.4-4 Major Calculation Codes for Core Management

No.	Calculation Code	Outline
1	METHUSELAH-II	Unit cell nuclear constants calculation code
2	WIMS-ATR-D	Unit cell nuclear constants calculation code
3	CALDUM	LAYMON input constants preparation code based on calculation results with METHUSELAH-II and WIMS
4	LAYMON	Three dimensional nuclear thermo-hydraulic united code/neutron one group FLARE model: used for prediction and analysis of process computer data
5	LAYMON-2A	Three dimensional nuclear thermo-hydraulic united code: has options of neutron one group FLARE model and ECOH model
6	LAYMON-2C	Three dimensional xenon dynamic characteristics analysis code
7	POLESTAR	Two group 3 dimensional nuclear thermal-hydraulic united code; used for making refueling program
8	ATROPOS	Core performance evaluation code for process computer: function of power distribution correction based on neutron flux distribution data is added to LAYMON code
9	ATROPOS-2	Core performance evaluation code for process computer: function of power distribution correction function based on neutron flux distribution data is added to LAYMON-2A code
10	ATLAC	Code for converting process computer data so as to be read by main computer
11	CALTA	Code for converting main computer data so as to be read by process computer
12	CITATION	Neutron diffusion calculation code
13	HAPI	Thermal-hydraulic analysis code (used for calculating channel flow by LAYMON, etc.)
14	FATRAC	Plant dynamic analysis code (1 for "Fugen", 2 for ATR demonstration reactor)
15	DINAGAP	Fuel rod space evaluation code
16	FODES	Fuel operational history evaluation code (Evaluates linear heat generation rate histories, etc. for individual fuel rods using operational data obtained by "Fugen" process computer)

5. Outline of Fuel Management [VII-4]

"Fugen" is characterized by the usage of nuclear fuel as shown below:

① "Fugen" has first and mainly used uranium-plutonium mixed oxide fuel assemblies (MOX fuel) in thermal neutron power plants in the world. 676 MOX fuel assemblies and 587 UO_2 fuel assemblies used in "Fugen" did not show any failure or abnormal deformation as of the end of 1997.

② Favorable neutron economy owing to usage of heavy water as the moderator allowed combined use of slightly enriched uranium UO_2 fuel and MOX fuel with natural uranium. Both types of fuels have been used since the first core loading.

③ "Fugen" has used various kinds of nuclear materials as shown below which are provided from PNC's nuclear fuel cycle facilities:

(i) Plutonium extracted from LWR spent fuels at Tokai reprocessing plant

(ii) Uranium enriched at Ningyo Toge uranium enrichment plant

(iii) Uranium recovered from LWR spent fuels

④ The own nuclear fuel cycle of "Fugen" has been established by loading MOX fuel containing plutonium which was extracted from the "Fugen" spent MOX fuel at Tokai reprocessing plant.

To confirm the irradiation characteristics of MOX fuel and to evaluate the safety margin of fuel design, irradiation tests were performed at the commercial fuel test facility of Japan Atomic Energy Research Institute, the irradiation fuel test facility of PNC O-arai engineering center in addition to fuel inspection with underwater fuel inspection system in "Fugen". As a result, it was confirmed that the irradiation characteristics of MOX fuel met the design value.

In addition, fuel assemblies for the development of ATR demonstration reactor fuels and of the high burnup and high performance are loaded in "Fugen" for irradiation tests.

6. Fuel Management System

The fuel/heavy water group of the Technical Section leads fuel management activities including MOX and UO_2 fuel transfer for delivery, inspection and loading in accordance with the safety Regulations and subordinative procedures.

Usually, 2 staff members of the Technical Section are in charge of fuel management activities including safeguards actions and accountancy. For routine field work, orders are placed on resident contractors on demand. The Accounting Section deals with office work for accountancy.

7. Fuel Development Results and Evaluation [VII-10] [VII-11] [VII-12] [VII-13]

7.1 Initial Loading Fuel and Reloading Fuel

Fuels consist of MOX and UO_2 fuels as the standard fuel and the special fuel for irradiation of pressure tube monitoring specimens.

Fuel development for "Fugen" was performed based on the design basis and methodology complied with those of the BWR, because the fuel service conditions in "Fugen" was almost same as the BWR. Irradiation tests for fuel assemblies were conducted at overseas reactors to clarity the fuel behavior characteristics and to confirm the fuel assembly integrity, following the development of the spacer, flow tests, and heat-transfer and hydraulic characteristics tests. Finally, fuel assemblies for initial loading were manufactured.

Fuels for "Fugen" are divided into 2 types, Type A and Type B according to the fissile material content. Type A fuel, whose fissile material content is low, was used for initial loading and reloading in the early stages. Since the 2nd periodic inspection, Type B fuel with a high fissile material content has been used. Both types of fuels have the same structure. Table Ⅶ-7-1 shows parameters of the standard fuel, special fuel and irradiation test fuel (to be referred later).

7.2 Irradiation Test Fuels

Irradiation tests are performed in "Fugen" loading the following fuel assemblies as well as driver fuels:

(1) 36-rod Fuel Assembly

The 36-rod cluster MOX fuel assembly which was developed for the ATR demonstration reactor. Three assemblies were loaded during the 5th periodic inspection (December 1985). Its target of maximum burnup is 23,000 MWd/t and 35,000 MWd/t. One assembly was removed to make the post-irradiation test when its burnup exceeded 24,400 MWd/t (May 1990), while one of the two others was removed when its burnup exceeded 33,000 MWd/t (February 1993) to perform post-irradiation test.

The test results show that they have similar irradiation behavior as the standard fuel and do not show any abnormal behavior. Fuel elongation is smaller than the standard fuel, showing good characteristics.

(2) Segmented Fuel Assembly

The segmented fuel assembly was made for deveropin a high performance fuel aimed at improvement of the reactor operability. Segmented fuel rods for irradiation tests consisting of zirconium-liner cladding and hollow pellets are

installed in a 36-rod cluster MOX fuel assembly.

Two assemblies were loaded during the 6th periodic inspection (March 1987). Its target of maximum burnup is 20,000 MWd/t and 30,000 MWd/t. One was removed when its burnup exceeded 18,000 MWd/t (October 1991) and the other when exceeded 30,000 MWd/t (August 1993).

Post-irradiation tests were performed for the removed fuels while power ramp tests for the segmented fuel rods were conducted in JMTR to clarity fuel performance during load change.

(3) Gadlinia Fuel Assembly

The gadlinia fuel assembly was planned as the reload fuel for the ATR demonstration reactor intended for burnup extension. Type I is used to confirm the fuel integrity and Type II is for clarification of gadlinia contained fuel characteristics. The target of maximum burnup of both types of fuels are about 40,000 MWd/t (Type I, 2 assemblies, irradiated until around 1995 and 1997) and about 30,000 MWd/t (Type II, 4 assemblies, irradiated until around 1995), respectively.

7.3 Post-irradiation Tests

(1) Test Fuels

Test fuels consist of each one of initial loading fuel assemblies(MOX Type A fuel, 13.6 GWd/t)and reload fuel assemblies(MOX Type B fuel, 18.1GWd/t), showing the highest burnup.

The test fuels were sent to Tokai Research Institute, JAERI for post-irradiation tests to confirm the validity of fuel design codes, design margin and manufacturing method.

(2) Results of Post-irradiation Test and Evaluation

(a) Fuel Assembly

① The integrity was confirmed, as no abnormal deformation nor failure were observed.

② The CRUD deposit depth was 240 μm at the maximum. It was brownish red soft type CRUD mainly consisting of Fe_2O_3 and had little thermal effect on the cladding tube.

③ Elongation was within a range of design; no problem in spite of the tendency to increase with fuel burnup. The mean gap between fuel rods got narrower to a extent of CRUD deposition.

(b) Fuel Rod

① The integrity was confirmed though nodular corrosion and signs of contact with the spacer were observed on the surface.

② Fuel elongation showed the tendency of increasing with fuel burnup; it is almost same as the results of lead irradiation and BWR fuel data. In the axially central part of the fuel rod with a high power, the outer diameter showed external pressure creep phenomena due to coolant pressure.

③ The dependence of the FP gas release rate on the burnup and power for the MOX fuel showed similar behavior of the UO_2 fuel for the LWR.

④ Typical crack was observed on pellets. Equi-axed grain growth was found at the center of pellets of the outer fuel rods while columnar grain growth at the center of pellets with higher burnup (18.1 GWd/t).

⑤ The length of a pellet stuck decreased (as expected) due to densification with fuel burnup.

⑥ The depth of nodular corrosion on the rod surface was about 60μm at the maximum which was within a range of BWR rod data.

⑦ Mechanical properties of the cladding including the yield strength and tensile strength fully satisfy the design values though they changed with irradiation. There was no significant decreasing tendency in rupture elongation with neutron irradiation 343 ℃.

(c) Fuel Assembly Components

① The integrity of the rod spring and plenum spring was confirmed, as no anomaly, such as failure and so on, was observed.

② Visual inspection confirmed the integrity of the spacer, tie plate and spacer tie rod. Though the rod pulling out force of the spacer decreased to less than that at the time of manufacture, fretting corrosion on the rod surface was minor.

Table VII.7-1 Major Specifications of Fuel Assembly

Item \ Fuel Assembly	Standard Fuel Assembly				Special Fuel Assembly		36-Rod Fuel Assembly for Irradiation Test (all rods)	Segmented Fuel Assembly for Irradiation Test (5 rods)	Gadlinia Fuel Assembly for Irradiation Test	
	UO_2 Fuel Assembly		MOX Fuel Assembly		Type A special fuel assembly	Type B special fuel assembly			Type I	Type II
	Type A UO_2 fuel assembly	Type B UO_2 fuel assembly	Type A MOX fuel assembly	Type B MOX fuel assembly					4 rods	3 rods
(1) Pellet										
Material	Uranium dioxide sintered pellet		Uranium/plutonium mixed oxide sintered pellet		Uranium dioxide sintered pellet		Uranium/plutonium mixed oxide sintered pellet		Uranium/plutonium mixed oxide sintered pellet Middle layer $UO_2 - Gd_2O_3$	
Outer diameter	approx. 14.4 mm		approx. 14.4 mm		approx. 8.3 mm (inner layer) approx. 12.8 mm (outer layer)		approx. 12.4 mm		approx. 12.4 mm	
Initial density	approx. 95 % of theoretical density		approx. 95 % of theoretical density		approx. 95 % of theoretical density		approx. 95 % of theoretical density		approx. 95 % of theoretical density	
Content of fissile material										
(inner layer, middle layer)	approx.1.5wt%	approx.1.9wt%	approx.1.5wt%	approx.2.3wt%	approx.2.5wt%	approx.3.2wt%	approx.3.2wt%	approx.3.7wt%	approx.2.6~5.0 wt%	approx.2.1~4.8 wt%
(outer layer)	approx.1.5wt%	approx.1.9wt%	approx.1.3wt%	approx.1.8wt%	approx.1.5wt%	approx.1.9wt%	approx.1.7wt%	approx.2.2wt%		
(2) Cladding										
Material	Zircaloy-2		Zircaloy-2		Zircaloy-2		Zircaloy-2	Zircaloy-2 and Zircaloy-2 with zirconium liner	Zircaloy-2 and Zircaloy-2 with zirconium liner	
Outer diameter	approx. 16.46mm		approx. 16.48mm		approx. 9.7 mm approx. 14.72 mm		approx. 14.5 mm		approx. 14.5 mm	
Thickness	approx. 0.88mm		approx. 0.88mm		approx. 0.62 mm approx. 0.84 mm		approx. 0.90 mm		approx. 0.90 mm	
(3) Fuel Rod										
Effective fuel length	approx. 3700 mm		approx. 3700 mm		approx. 3540 mm		approx. 3647 mm		approx. 3650 mm	
Plenum volume ratio	approx. 0.054		approx. 0.054		approx. 0.098		approx. 0.1	approx 0.1~0.44		
Helium sealing pressure	approx. 1kg/cm^{3}a		approx. 1kg/cm^{3}a		approx. 1kg/cm^{3}a		approx. 3kg/cm^{3}a			
(4) Fuel Assembly										
Shape	Three-layer concentric 28-rod cluster		Three-layer concentric 28-rod cluster		Two-layer concentric 36-rod cluster		Three-layer concentric 36-rod cluster		Two-layer concentric 36-rod cluster	
Total length	approx. 4,388 mm		approx. 4,388 mm		approx. 4,388 mm		approx. 4,393 mm		approx. 4,393 mm	
Outer diameter	approx. 111.6 mm		approx. 111.6 mm		approx. 111.6 mm		approx. 111.6 mm		approx. 111.6 mm	
Number of fuel rods	28		28		36		36		36	
(5) Design Conditions										
Maximum fuel assembly burnup	approx. 20,000MWd/t		approx. 20,000MWd/t		approx. 30,000MWd/t		approx. 35,000MWd/t	approx. 30,000MWd/t	approx. 40,000MWd/t	approx. 30,000MWd/t
Maximum linear heat generation rate	574W/cm(17.5kW/ft)		574W/cm(17.5kW/ft)		492W/cm(15kW/ft)		492W/cm (15kW/ft)	394/cm (12kW/ft)	492W/cm (15kW/ft)	427/cm (13kW/ft)

1) In the case of special fuel assembly values are those of inner layer.

8. Fuel Management History and Evaluation

8.1 Fuel Acceptance History

A total of 71 times of fresh fuel transportation had completed until December 1997; a total of 1305 assemblies, consisting of 676 MOX fuel assemblies (including various irradiation test fuels) and 629 UO_2 fuel assemblies (including special fuels) were accepted. The MOX fuels were delivered from PNC's Tokai Works and the UO_2 fuels were from Nuclear Fuel Industries' Tokai Works (those for initial loading were from Takeyama Research Center).

For the MOX fuels, α check was conducted prior to unpacking the cask (BU type fuel transportation cask) to confirm that there was no failure of fuel elements.

8.2 Inspection Results of Fresh Fuel

Visual inspection (4 items including surface appearance and intrusion of foreign materials) and dimensional inspection (4 items including measurement of the gap between outer fuel elements and measurement of bend) are performed to confirm that the integrity of the delivered fresh fuels has not been losed by manufacturers' handling and transportation.

The inspection method for MOX fuel were changed to manual measurement like UO_2 fuels since exposure was less than originally expected (0.1 mSV or less for one day work).

As a result of fresh fuel inspection, the integrity of all the fuels was confirmed and the measured gap was 1.87~2.43 mm, meeting the design basis of 1.7 mm or more. The standard deviation of measured values was as small as 0.05~0.07, showing no dependency on the manufacture time.

The above results demonstrated that fuel handling and land transportation had no effect on the fuel assembly.

8.3 Inspection Results of Irradiated Fuel

Leakage inspection, visual inspection and dimensional inspection are performed with the underwater inspection system, etc. to confirm the structural integrity of the fuel assembly and to find whether the fuels were failed or not.

One thousand and several hundreds assemblies of irradiated fuels including shaffled fuels and those removed temporarily have been inspected so far.

At present, the inspection is being conducted with the inspection system (non-contact type) which is newly installed in the fuel delivery pool. Additionally, γ scanning measurement was performed for about 10 other fuel assemblies to

confirm the power distribution and the burnup distribution.

(1) Leakage Inspection

The iodine (I-131) concentration in reactor coolant (normally 0.1 Bq/g or less) is periodically measured to detect fuel failure. No sign of fuel failure has been observed.

(2) Visual Inspection

Surface appearance of the outer rods, the spacer, etc. is observed with the periscope of the underwater fuel inspection system. Though the overall surface of the fuel rods were covered with crud (thin deposition to a extent that luster of zircaloy can be confirmed), no abnormal conditions such as broken failure, hit failure or deformation was found.

Investigation showed that the crud chemically consisted of 80~90 % α-Fe_2O_3 (hematite). The brownish red crud deposited on components other than the fuel rod; however, no anomaly was observed on the attachment condition of the guide spring, the spacer, etc.

(3) Dimensional Inspection

Over sixty thousands points of the gap between fuel rods have been measured so far. The gap is getting narrower as the number of operating days increases, which was due to the effect of deposited crud and had little thermal effects on the cladding, as seen in previously mentioned post-irradiation tests (7.3(2)(a)①). Elongation of the fuel assembly can be determined according to the difference in the position connecting with the gripper of the fuel transfer machine between for the fresh fuel and for the spent fuel. The elongation tends to increase proportional to the burnup. The elongation rate was about 0.6~0.9 mm/GWd/t and even the maximum value was within a range of the design. No significant difference in the elongation rate was observed between the MOX fuel and the UO_2 fuel.

8.4 Transportation of Irradiated Fuel

15 times of irradiated fuel transportation have been performed including spent fuels for reprocessing sent to PNC's Tokai Works and irradiation test fuels sent to JAERI's Tokai Research Center. HZ-75T-ATR type cask (containing 17 assemblies) and NH-25 type cask (containing one assembly) were used respectively.

It is difficult to load the fuels into the cask in the spent fuel storage pool in "Fugen", as its fuel loading is performed under about 8 m deep. Therefore, the

positioning device was newly developed to set it on the upper part of the cask thereby improving the fuel loading work.

8.5 Reprocessing of "Fugen's" spent fuel

It was in 1986 when the first reprocessing of "Fugen's" spent fuels was conducted at Tokai Reprocessing Plant to recover plutonium.

As a result, the shape (bundle type) and the composition (PuO_2-UO_2) of the fuel assembly did not affect reprocessing work. In addition, there was no difference in fuel handling, extraction sratus of uranium and plutonium, and major FP and TRU extracting behavior compared with reprocessing of LWR fuels. The recovery of plutonium was 99.9 %, showing no difference with that of LWR fuels.

The recovered plutonium was loaded into "Fugen" during the 7th periodic inspection and used; thereby showing to have established the "Fugen's" independent nuclear fuel cycle.

9. Irradiation of Pressure Tube [VII-14] [VII-15]

9.1 Surveillance Test

The pressure tube which makes up the pressure boundary of reactor coolant is subject to high neutron irradiation. Therefore, it is important to investigate the changes in the pressure tube material characteristics and to confirm its integrity. In this context, post-irradiation tests of the pressure tube surveillance specimens are performed in accordance with "the Ministry ordinance specifying the technical standards for nuclear power plant".

Fig. Ⅶ-9-1 illustrates the structure of the pressure tube used in "Fugen". The pressure tube is roll-jointed with the pressure tube extension as shown in Fig. Ⅶ-9-2. Therefore, the pressure tube extension material surveillance test specimens are also irradiated.

A capsule containing the pressure tube material surveillance test specimens, consisting of a. tensile test specimen, b. burst test specimen and c. corrosion test specimen, is installed in a special fuel as shown in Fig. Ⅶ-9-3 and it is irradiated in-pile. Cu and Fe flux monitors for measurement of fast neutron exposure are also contained in the capsule.

In addition, the material specimens made in Japan for the pressure tube material of the ATR demonstration reactor are also subject to irradiation tests. These specimens include the pressure tube material surveillance test specimens similar to those for "Fugen", as well as the small diameter irradiation creep test specimens sealed with high pressure gas.

The post-irradiation tests were performed in Fuels Monitoring Facility of PNC, O-arai Engineering Center. Both irradiation and pot-irradiation tests for the pressure tube extension material of the ATR demonstration reactor were conducted at JMTR, JAERI.

9.2 Irradiation Plan

Four special fuel assemblies, each of which contains 9 capsules, are loaded in the middle of the core in the rotation symmetry position. It is so designed that the center capsule is subject to cumulative fast neutron flux irradiation equal to or more than that of the pressure tube (acceleration factor 1.1～1.2).

Two capsules are removed from individual special fuel assembly every time and then the capsules positioned above or under the removed ones are moved. Thus, the surveillance specimens installed in 4 special fuel assemblies can serve for 8 times of surveillance tests.

9.3 Irradiation Result

Exposure of the surveillance test specimens removed at the 1st time was 3×10^{21}n/cm^2 and its acceleration factor was 1.09 against the maximum pressure tube exposure. The secondly removed specimens showed irradiation of 6.5×10^{21}n/cm^2 and the acceleration factor of 1.16, as evaluated in the irradiation plan.

9.4 Capsule Handling Work

It is necessary to operate 10 m or longer tools (3 types of special tools) underwater to handle the capsules. This results in difficulties in positioning of the tools. Therefore, improvements including attachment of the balancer were made.

Since CRUD adhered on the surface of the capsules may give some difficulty on irradiation tests, decontamination is made with a special screw brush in the decontamination facility inside the pool.

The capsules containing the surveillance test specimens are activated (25~460 TBq); therefore, they are treated as B type package during transportation. A total of 22 capsules have been transported in 6 times of shipment.

9.5 Results of Post-Irradiation Test and Evaluation

The results of PIE on the pressure tube specimens in the 1st surveillance test are shown below:

① Fast neutron exposure of the specimens was $(2.7\pm0.5)\times10^{21}$n/cm^2, showing good agreement with that evaluated in "Fugen", 2.9×10^{21}n/cm^2.

② The tensile strength increased by about 20~30 % while the elongation showed slight decrease due to irradiation.

③ The fracture toughness showed decrease due to irradiation opposite to the tendency of the tensile strength. However, the decrease rate was large at the early stages of irradiation, while at 10^{21}n/cm^2 of exposure, the decrease rate was small. At about 300℃ of operating temperature in "Fugen", reduction of the fracture toughness due to irradiation is small.

④ Hydrogen absorption reached about 10ppm during 5 years incore stay after start of operation. It is a good result because it is less than the design estimation of 7 ppm/year.

As described in the above, the integrity of the pressure tube has been confirmed because material characteristics of the pressure tube due to irradiation meet the design values. Irradiation tests of surveillance specimens are to be continued.

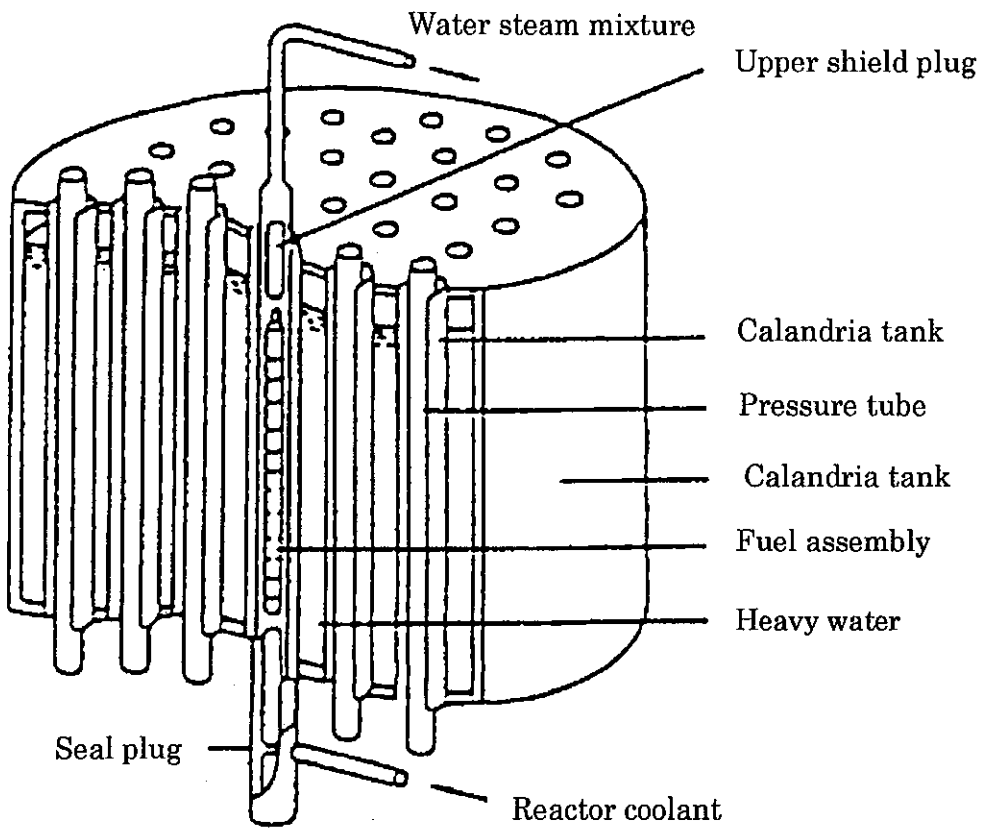


Fig.VII.9-1 Core Cross Section

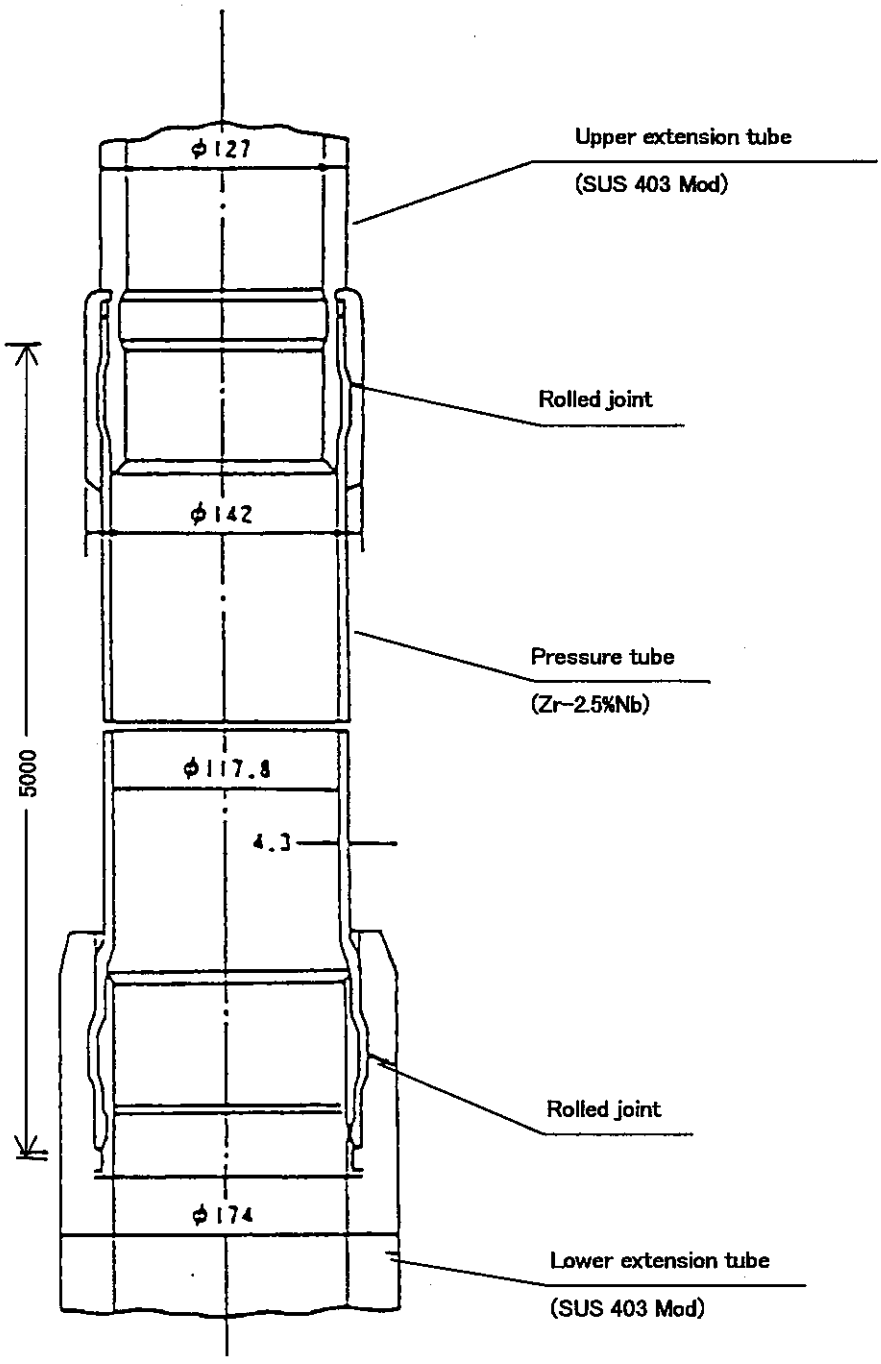


Fig.VII.9-2 Pressure Tube Assembly

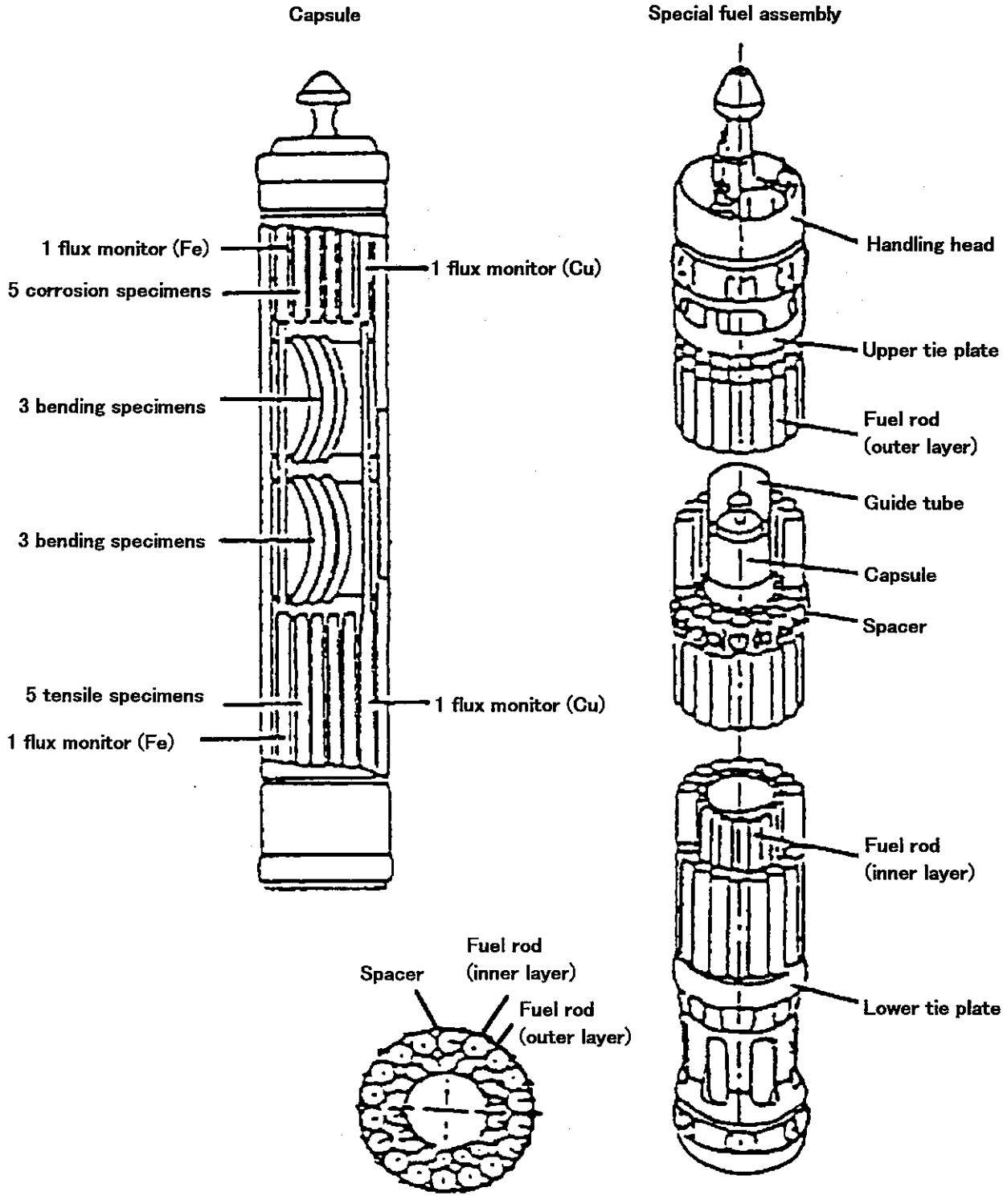


Fig.VII.9-3 Special Fuel Assembly with Monitoring Specimen

10. Safeguards [VII-16] [VII-17]

10.1 Outline of Safeguards

Nuclear non-proliferation treaty has been internationally agreed for the purpose of peaceful use of nuclear fuel materials. In this context, safeguards actions have been taken in Japan for the nuclear material handling facilities.

Especially, handling plutonium and making R&D for peaceful use of atomic energy with an emphasis on the nuclear fuel cycle, PNC is in an important position in view of the safeguards.

"Fugen" is required to impose more stringent safeguards than LWRs since it handles MOX fuels. The MOX fuels are subject to the safeguards for the Item Facility because they are those fuel assemblies sealed with cladding.

Since it is impossible to verify the fuel assemblies loaded in the core because they are contained inside the pressure tube, strict management for transportation is required by making radiation measurement (γ , n) and by the documents.

In addition, "Fugen" has an inventory of 195 tons (100 % conversion) of heavy water, including about 160 tons loaded in the heavy water system as moderator. There is no need to manage heavy water as material subject to the international control since the revised atomic energy agreement between the U. S. and Japan came into effect in July 1988. However, PNC voluntarily performs strict accountancy for heavy water similar to those subject to the international control.

10.2 Accountancy

The Material Balance Area (MBA) related to the fuel assembly is a single for accountancy, consisting of 3 Key Measuring Points (KMP) of fuel flow and 5 KMP of fuel inventory. There is another MBA which is related to permission of nuclear material usage; it addresses small amount of nuclear material (neutron detectors).

Management of fuel transfer within a facility is performed based on transfer management slips. When unloading spent fuels from the core (KPM2), generation and consumption or loss of nuclear fuel materials are calculated for individual fuel assembly to make Inventory Change Reports (ICR).

When shipping spent fuels out of the facility, Pu-241 decay during cooling period is calculated and reported.

10.3 Inspection

The Japanese government and IAEA perform routine inspection once a month in "Fugen". During the inspection, book audit including check of the conformity

between "Fugen's" data (ICR, etc.) and the record and check of the source data. At site, ID (Identity Data) confirmation, item counting, NDA (Non-Destructive Assay), etc. are conducted while video tapes of the monitoring cameras are replaced (camera service). The monitoring cameras are installed at 7 points in "Fugen".

In addition, Physical Inventory Verification (PIV) is conducted according to Physical Inventory Taking (PIT). The contents of the PIV is similar to the routine inspection.

11. Heavy Water Management [VII-18] [VII-19]

11.1 Outline of Heavy Water Management

Use of heavy water allows the reactor to have excellent fuel utilization characteristics since it has a greater neutron moderating ratio than light water, leading to reactor operation with use of less fissile material and uranium-plutonium mixed oxide (MOX) fuels can be used in a similar way to uranium fuels. "Fugen" has about 195 tons inventory of heavy water, including about 160 tons loaded in the heavy water system.

Exchange of ion exchange resins in the heavy water purification system is made and degraded heavy water is generated with reactor operation. Therefore, makeup heavy water was imported in the early stages. In 1979, the heavy water upgrader-I started operation and then in 1987 the heavy water upgrader-II came into use. As a result, heavy water recycling was established in "Fugen", so that there was no need to import makeup heavy water since 1983.

11.2 Accountancy of Heavy Water

Accountancy of heavy water is performed by dividing heavy water into 3 types; loaded heavy water in the heavy water sysytem, fresh heavy water (including upgraded heavy water) and recovered heavy water. As mentioned in the above 10. (1), "Fugen" performs heavy water management in a way similar to safeguards.

11.3 Heavy Water Management System

The heavy water group of the Technical Section is in charge of overall management of heavy water. Major management activities include heavy water operation planning, heavy water delivery, storage management and accountancy. On the other hand, the Safety and Chemistry Section is in charge of heavy water analysis during exchange work of ion exchange resins, heavy water charging and degraded heavy water draining, as well as during rated power operation. For the heavy water upgraders, the Power Generating Section has been responsible for operation and maintenance since start of commercial operation.

11.4 Heavy Water Purchase History

About 160 tons heavy water of the U. S. nationality was purchased from Karlsruhe Nuclear Research Center of ex-West Germany as initial inventory of heavy water. It was transported to Japan in two shipments with drum cans in 1976 and then temporarily stored at the solid waste storage facility. After that, about 17 tons of

makeup heavy water was imported from the U. S., 16 tons from Norway and about 5 tons from China. Since 1979, no heavy water has been imported due to reason mentioned in the above 11. (1).

11.5 Degraded Heavy Water Generation History

Through the operation of "Fugen", degraded heavy water is generated during resin exchange work in the heavy water purification column and the poison removal column and associated sampling to measure the heavy water concentration, sampling for routine chemistry control and overhauling for periodic inspection of the heavy water system components.

Degraded heavy water generated during exchange of ion exchange resins and heavy water sampling consists of highly concentrated heavy water (approx. 97 %) and low concentrated heavy water (approx. 30 %). Usually, a total of 2.0～2.5 tons heavy water is degraded during resin exchange of a single column.

The degraded heavy water is stored in 2 types of tanks according to the concentration; degraded heavy water tank A (high concentration) and B (low concentration).

Highly concentrated heavy water stored in the tank A is drained down to drum cans to supply it to the heavy water upgrader when it is appropriate to do so.

For chemistry control of the heavy water contained in the system, around 10 ～30 kg heavy water is sampled every month. After resin exchange, 100～150 kg of heavy water is sampled. As a result, sampled heavy water amounts to some 0.5 ton annually. The sampled heavy water is recovered into drum cans to supply it to the heavy water upgrader; thereby giving almost no loss of heavy water.

11.6 Heavy Water Upgrader Operation History [VII-20] [VII-21] [VII-22]

In 1979, the heavy water upgrader I set about operation to treat degraded heavy water with high concentration. It has an ability to treat 5 tons of 95 % degraded heavy water annually to produce 4.4 tons of upgraded heavy water (99.8%). In 1979, 97 % degraded heavy water was treated to produce a large amount of upgraded heavy water. The resultant about 90 % degraded heavy water was treated with a reduced treatment ability and decreased amount of upgraded heavy water. In 1987, the heavy water upgrader II started operation and enabled treatment of degraded heavy water with low concentration. As a result, with operation of 2 types of the upgraders, it became possible to set up an efficient

heavy water upgrading and management plan. Table Ⅶ-11-1 shows heavy water management history in "Fugen".

Table VII.11-1 Heavy Water Management History

(unit : t)

Fiscal Year	Procurement History			Degraded D_2O Generated		Upgraded D_2O	
	U.S.A.	Norway	China	High concentration	Low concentration	I	II
1976	159.2 (W.Germany)						
1977		6.6					
1978		5.0					
1979	8.0	2.2		17.6	2.4	6.6	
1980	7.0		2.0	3.4	0.9	2.5	
1981	2.0	2.0	1.0	0.2	0	2.7	
1982			2.0	2.5	0.4	4.0	
1983				0.4	0	5.7	
1984				4.3	9.1	3.5	
1985				3.9	2.5	4.4	
1986				2.5	1.9	3.7	
1987				4.2	0.8	3.8	1.9
1988				5.3	2.8	2.8	4.9
1989				3.7	2.3	2.9	1.8
1990				1.6	0.8	0.2	2.5

* Procurement history is shown in net weight (approx. 99.8% D_2O).

The others are shown in D_2O (100%) equivalent.

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VIII. Periodic Safety Review and Probabilistic Safety Assessments

1. Periodic Safety Review [VIII-1] [VIII-2]

1.1 Introduction

The ATR "Fugen" is positioned as a prototype reactor. Its operation has been continued after the basic design of the safety function was developed based on the defense in depth concept like the case in commercial light-water reactors. Its construction was performed according to the specified technical standard and it was confirmed by the commissioning that each of the components and systems has the specified performance. Even during the operational phase, it is confirmed that the reliability of the safety function is appropriately maintained through the periodic inspections. Meanwhile, the latest knowledge obtained through the safety-related experience in the world has been integrated into the appropriate knowledge to "Fugen". For example, the prevention measures were taken against the recurring of accidents and failures happened in Japan and in foreign countries.

However, as the technical knowledge is accumulated with experiencing the operation, it is profitable to evaluate the plant safety from the comprehensive point of view based on the operational experience and the trend of knowledge both gained in Japan and in foreign countries to provide improvement measures.

In addition, the Nuclear Safety Commission in Japan recommends that the effective accident management shall be developed voluntarily in order to further reduce the reactor facility risk though the risk has been sufficiently diminished. In response to this recommendation, in Fugen Nuclear Power Station, the Probabilistic Safety Assessments (PSA) of the individual plant is performed with the latest knowledge taken into consideration, and the accident management will be developed based on the PSA result.

Accordingly, the Periodic Safety Review (PSR) of "Fugen" is to evaluate comprehensively the safety of "Fugen" from the point of view similar to PSR in commercial light-water reactors, based on the operational experience and the latest technical knowledge, and to perform the appropriate improvement for the facilities and methods of operation management based on the results including the PSA result.

The following three items were conducted in "Fugen" PSR:

① Comprehensive Evaluation of Operational Experience

② Integration of Latest Technical Knowledge

③ Evaluation Using PSA

1.2 Operational Experience [VIII-3] [VIII-4] [VIII-5] [VIII-6]

(1) Operational History

The construction of "Fugen" was started in December, 1970 as a prototype heavy-water-moderated boiling-light-water-cooled pressure-tube-type reactor, and its full-power operation was started on March 20, 1979. Since then, it has been playing the role of demonstrating the technical development of the ATR. In particular, as for using the MOX (Mixed Oxide Fuel), its experience is unique and outstanding as a reactor: among the total number of 1,119 fuel assemblies loaded up to August, 1996, 638 of them are MOX fuel containing about 1,500 kg of plutonium with about 1,170 kg of fissile plutonium.

"Fugen" has been making every effort for safety, including the maintenance and improvement of the facilities and enhancement of operation management, which resulted in about 19 years of safe and stable operation after the commencement of commercial operation. The gross generated power up to now reaches about 18.05 billion kWh, the gross generation hour about 113,000 hours, the average capacity factor about 66%, and the average periodic inspection period about 112 days. The capacity factor changes since the commencement of commercial operation are shown in Fig. VIII.1-1.

(2) Number of Accidents and Failures

Fig. VIII.1-2 shows the changes of numbers of reported accidents and failures experienced from the commencement of commercial operation to March 1995. Number of accidents and failures related to the reactor cooling system facilities are the most dominant, occupying about 1/4. The accumulated total number is 39, and the accumulated total number of unscheduled shutdown is 20.

1.3 Operation Management

The objectives of "Fugen" are, while maintaining the stable operation, to perform various property tests for R&D and to confirm the fuel integrity and operational reliability. Therefore, appropriate actions are required in order to assure safety not only during the normal operation but also during the accident. As a matter of course, it is required that the appropriate operation system is maintained, the operator's work is appropriately conducted and the operating

procedures for implementing such work is provided, and the operator education and training are performed to assure the accurate performance of work.
Therefore, the investigation and comprehensive evaluation were performed on the organizations concerning operation, the members of the operation shift, the working style of operators, the improvement status of the operation system, the operator's work, the operation procedures, the improvement status of the operation procedures based on the "Operator's Work and Operating Procedures", the operator education and training, and the improvement status of the education and training based on the "Operator's Education and Training". The results are shown in the following:

① Operation System

The system was changed from "four crews working on three shift- system" to "five crews working on three shifts plus a day shift system", from the viewpoint of enhancing the operator's education and training. As a result, the participation of all the shift members to BTC (BWR Training Center) and the implementation of "Fugen" simulator training per shift became possible.

② Operator's Work

Various operating procedures were renewed. The facilities were improved so that the operability and monitoring capability could be enhanced.

③ Operating Procedures

The operating procedures were reviewed from time to time based on the operation experience, accident or failure experience, facility improvement, etc. Especially, as in October 1990, the automatic reactor trip resulted from the human error occurred, a thorough countermeasure was taken to prevent the recurring of human errors.

④ Education and Training

The education consisting of the in-site on the job training (OJT), the "Fugen" compact-type simulator training and BTC simulator training were implemented to maintain and improve the knowledge and technique to help the full understanding of the operating procedures and the appropriate operation. At the same time, the operation experience in "Fugen" and other nuclear power stations in Japan and foreign countries were integrated into the operator education and training plan as appropriate, and the education and training have been conducted

regularly based on this plan.

1.4 Maintenance Management

In order to assure the safe and stable operation, it is important to shut down the plant periodically

(i) to perform the periodic inspection based on the Law for the Regulations of Reactor, etc., and to conduct the equipment overhauling and maintenance as a part of the independent maintenance

(ii) to implement the modification or replacement work as necessary to integrate the latest technical knowledge, the lessons learned from the domestic and international nuclear power station operation experiences in order to maintain and enhance the facility and equipment integrity and reliability.

It is also important, during the plant operation, to confirm that the integrity and reliability of each facility and equipment are maintained. The facilities and equipment are checked by the periodic tests and patrol by the operators and the daily inspection by the maintenance personnel. If any anomaly should be found, the required maintenance work is performed to recover the plant to the normal state.

Therefore, the evaluation was performed targeting the following items:

① The investigation was conducted no the implementation status of the periodic inspection and periodic overhauling and their results, the ISI program and experience, the test result of the pressure tube monitoring test specimen, the evaluation result of the fatigue accumulation coefficient of the class 1 pressure vessel and the operation management of the periodic inspection to check if the integrity of the system and equipment was maintained and if the operation management of the periodic inspection was appropriately performed.

② The investigation was conducted on the modification and replacement experiences of major facilities, the SCC measures taken in the past, and the operation management of the facility modification and replacement work. From the viewpoint of the equipment reliability enhancement, the maintenance of function against aging and the prevention of accident or failure recurring, it was checked if the facility modification and replacement were performed carefully and positively and if the operation management for the modification or replacement work was accurately

performed.

③ The investigation was conducted on the overhauling system during the periodic inspection, the management system for the facility modification and replacement work, the quality assurance for the maintenance management to check if the equipment overhauling, modification or replacement work were accurately conducted under the appropriate maintenance management system based on the appropriate quality assurance activities.

④ The investigation was conducted on the history and improvement status of the education and training for the maintenance personnel to check if the education and training for the maintenance personnel were adequately performed and if the education and training contents and their methods were enhanced as necessary.

(1) Implementation Status of Periodic Inspection and Periodic Inspection

"Fugen" has the operation history of about 19 years after it started commercial operation in March 1979, and 14 periodic inspections have been performed. These inspections including ISI and the class 1 pressure vessel inspection were implemented based on the guideline as is mentioned in Chapter III, Maintenance Management. The pressure tube material integrity is checked not only by the inspection using the pressure tube monitoring equipment but also using the monitoring test specimens irradiated in the reactor. Fig. VIII.1-3 shows an example of the irradiation test of the pressure tube monitoring test specimens.

(2) Modification or Replacement of Major Facilities, Countermeasures Against SCC and Operation Management [VIII-7]

The facilities are modified or replaced for the purpose of the preventive maintenance, operability improvement and conforming to the revised guidelines.

"Fugen" is a plant constructed before the knowledge on SCC was obtained, and SCC was detected in the residual heat removal system piping in November 1980. Therefore, it was confirmed that the following measures were taken for the systems where the three factors, such as the material (austenitic stainless steel with high carbon content) that was the SCC generation factor, the residual stress (high tensile stress) and the environment (temperature: 100°C or more with high dissolved oxygen) were present:

① Replacement to SCC-Resistant Material

The material with high corrosion resistance (ex. SUS316L) was used for the replacement of the replaceable parts of the equipment including the piping with small diameters, the nozzle safe end parts, and the inlet headers.

② Implementation of IHSI

IHSI (Induction Heating Stress Improvement) was performed in the parts for which IHSI could be implemented.

③ Adoption of Water Quality Improvement Method Using Hydrogen Injection

The water quality improvement method in which the hydrogen was injected into the feed water to reduce the dissolved oxygen in the reactor coolant was adopted to prevent the SCC generation. For more details, see Chapter III, Maintenance Management.

(3) Overhauling System During Periodic Inspection and Management System of Facility Modification or Replacement

As for the operational management in the maintenance management section concerning the periodic inspection, it was confirmed that the periodic inspections were appropriately managed based on the "Quality Assurance Plan" provided in 1981 and the "Quality Assurance Plan for Fugen Nuclear Power Station Facility" prepared in 1986 through the planning phase, implementation phase, evaluation phase and reflection phase, and that the efforts had been made for the system or equipment integrity maintenance. Since 1985, the standardization of the guideline on the periodic inspection work for equipment has been performed.

(4) Status of Operator Education and Training and Their Improvement

In order to check if the appropriate education or training is conducted for the maintenance personnel and if the efforts are made for the enhancement of the education or training contents and its methods, the education or training history for the maintenance personnel and the improvement of the education or training are investigated.

The investigation results show that education and training of the maintenance personnel are conducted based on the education and training plan, and they are classified into technical education, general education and other education.

① Technical Education

② General Education

③ Other Education

Education and training of the maintenance personnel are conducted under the careful management so that education and training of the maintenance personnel can be performed through the planning phase, implementation phase, evaluation phase and reflection phase.

1.5 Fuel Management

It is important for the fuel management to specify the operation limit, observe the limit to assure the fuel integrity, and to try to enhance the fuel reliability by integrating the operation experiences at other plants, etc. into the management. Accordingly, the investigation is conducted for the items concerning the fuel and the core which have been used in "Fugen" to check whether or not the fuel integrity is assured.

(1) Operating Limit [VIII-8]

(a) Basis of Specification of Thermal Limit and Relating Issue

① Minimum Critical Heat Flux Ratio (MCHFR)

The minimum critical heat flux ratio (MCHFR: the in-pile minimum heat flux ratio of the critical heat flux where the transition boiling occurs to the local heat flux of actual fuel) is specified to prevent the fuel cladding failure by the super heating due to the transition boiling even during the abnormal transient during operation; in "Fugen", the limit is "1.9 or more", which was the same value as the light-water reactors at that time.

The MCHFR design curve is correlated from the lowest line of the relationship between the CHF data and the average thermal-equilibrium steam quality at the fuel assembly horizontal cross-section, which were obtained from the full-scale dry-out experiment performed using the 14 MW heat transfer loop (HTL) at O-arai Engineering Center of PNC.

② Maximum Linear Heat Generation Rate

The maximum linear heat generation rate is specified as the limitation during the normal operation to prevent the cladding deformation resulting in the cladding failure due to the relative expansion difference between the fuel cladding and the pellets even during the abnormal transient. It is the maximum in-pile value of the generating power per fuel unit length (kW/ft).

In "Fugen", both MOX fuel and the uranium oxide fuel are used as the

standard fuel, but there is no difference in their thermal limits: 17.5 kW/ft for the standard fuel, and 15 kW/ft for the special fuel.

③ Reactor Shutdown Margin

The reactor shutdown margin is the value marking the reactor shutdown capability. This is specified in order to maintain the reactor sub-critical at the low temperature even if one control rod with the maximum reactivity worth is not inserted for some reasons when the reactor scram signal is actuated. Its aim is to keep the effective multiplication factor less than 0.99.

(b) Observation Status of Thermal Limits, etc.

For the core design for each operating cycle, the operation pattern, the fuel loading pattern, the control rod pattern and the poison concentration are examined, and it is confirmed through analyses that the maximum linear heat generation rate, MCHFR and the highest burn-up at the end of the operating cycle satisfy the limits throughout the cycle. At the same time, it is confirmed by the analyses, for the reactor shutdown margin, that the effective multiplication factor when, among the operable control rods, one control rod with the highest reactivity value is not inserted is less than 0.99. After the commencement of commercial operation, it is confirmed that the thermal limits satisfy the limits by the operation monitoring using the process calculator. As for the system, the neutron flux is continuously measured by the local power monitor (LPM), and if the excessive power is detected, the "High LPM" alarm is actuated before the maximum linear heat generation rate is reached. At the same time, the function confirmation and the calibration are performed periodically (once a month) to check and maintain the functions.

(c) Changes of Iodine-131 Concentration in Reactor Coolant

The measured concentration of Iodine-131 is stable at about 2 - 5×10^{-2} Bq/g, which is well below the operation limit of 2.9×10^{4} Bq/g.

The additional released iodine amount measured per reactor shutdown is 5.2×10^{7} Bq at most, which is well below the limit specified in the safety regulation of reactor installation (7.4×10^{13} Bq). This is because no fuel is failed and the fuel cladding integrity is maintained.

The detected Iodine-131 is assumed to be generated from the fission of the uranium contained as impurity in the fuel cladding materials and pressure tube materials.

(2) Measures for Enhancement of Fuel Reliability

The operation management to enhance the fuel reliability and the fuel design improvement are investigated.

(a) Operation Management for Fuel Failure Prevention and Fuel Design Improvement [VIII-9] [VIII-10]

① Operation Management for Fuel Failure Prevention

In order to reduce the interaction between the fuel pellets and the fuel cladding, and to prevent the fuel failure, the "Pre-Conditioning Interim Operating Management Recommendation (PCIOMR)" is adopted in "Fugen" as one of operation management measures.

PCIOMR is a method obtained from integrating the knowledge into the operation management. The knowledge is gained in the experiments at GETR, etc. The knowledge is that when the fuel rod linear heat generation rate is increased to a certain level or more, if the power is increased at slow speed, the fuel is kept intact when it is maintained within the linear heat generation rate range (envelope) once experienced even after the rapid power changes are repeated, for some time after the power increase.

Specifically, before the rated power is reached, the power-up ratio is 0.6 %/h, and the ratio is maintained for a specified period at the upper and lower limits of the automatic control rod control range at the rated power. Then the "automatic" operation of the automatic control rods is available.

The total 1,164 fuel assemblies consisting of 627 MOX fuel assemblies and 537 uranium oxide fuel assemblies have been loaded in "Fugen"; no fuel failure has occurred, and the fuel reliability has been demonstrated.

② Fuel Design Change

In "Fugen", the type A fuel [fissile content: 1.4% (for MOX fuel), uranium enrichment: 1.5% (for uranium oxide fuel)] used in the initial core, to enhance the economy by increasing the burn-up, was switched to the type B fuel [fissile content: 2.0% (for MOX oxide fuel), uranium enrichment: 1.9% (for uranium oxide fuel)] in 1981.

In addition, to evaluate the MOX fuel integrity, the fuel inspection using the underwater fuel inspection device is performed in the site, and the PIE (Post-Irradiation Examination) has been conducted for two MOX fuel assemblies with the different burn-ups.

(3) Use of Experiences from Other Plants

The investigation is conducted as to how the experiences of other plants concerning the fuel/core management should be used.

In June, 1990, during the 11th periodic inspection of Hamaoka Unit No. 1, the leakage was detected for five fuel assemblies. It was found out that this was because the organic substances eluted from the resin of the condensate demineralizer flowed into the core from the feedwater system, etc. As a countermeasure against this event, in "Fugen", the monitoring of the water quality of the reactor coolant was strengthened (measurement items were added) and the cladding tubes with less accumulated annealing hours were to be used for the cladding for the fresh fuel.

(4) Fresh Fuel and Spent Fuel Management

The fresh fuel, fuel to be continuously used and spent fuel management status is investigated. As a result, the following are confirmed. Regarding new fuels, they are handled with prescribed equipment and a facility and stored at a designated place in order to prevent the damage and criticality in each step of transportation, storage and loading. The confirmation and inspection of fuels are conducted periodically. Regarding fuels to be continuously used, they are handled with the similar manner as the new fuels, and their out-looking are observed and their size are measured in order to confirm the intact of the fuel. Spent fuels are also handled with the similar manner. They are inspected carefully when they are transported to the outside of the station.

(5) Fuel Operational Management

The investigation is conducted as to the management methods for each stage of the fuel management, including the refueling planning, the fuel loading, the core management (burn-up control) and fuel integrity evaluation.

For the refueling plan, the 90-degree rotation fuel dispersion replacement method is adopted as the basic method because the power distribution can be flattened, and the burn-up control and the operation control are easy. The plan is prepared based on:

① Specified Core Cycle Lifetime According to Plant Operation Plan
② Thermal Limits Set Based on Fuel Integrity
③ Reactor Shutdown Margin
④ Fuel Assembly Discharge Burn-up, etc.

1.6 Chemistry Control

As "Fugen" is a prototype ATR reactor, its chemistry control are not only the

water quality control of the reactor coolant similar to that of BWR plant but also the water quality control of the moderator specific to the heavy water reactor (heavy water). Therefore, as for the heavy water, along with the maintenance of the integrity of the heavy water system component materials, the appropriate water quality control is required in order to restrain the crud (radioactive corrosive products) accumulation on the equipment or piping and to reduce the dose equivalent rate. Accordingly, it is necessary that the management target values for maintaining the reactor coolant and heavy water quality at high purity are appropriately specified and observed during the plant operation, and that the effective dose equivalent reduction measures are taken in terms of the water quality control.

Therefore, the investigation is conducted as to the specification bases for the reactor coolant and heavy water quality control target values, the water quality history during the plant operation, the implementation status of the water quality improvement measures for the reactor coolant and heavy water, and the water quality control status of the reactor coolant and heavy water. The aim of this investigation is to check if the water quality is controlled to the values satisfying the control target values and if the actions to improve the water quality are appropriately taken.

(1) History of Water Quality

(a) Reactor Coolant Quality Control [VIII-11] [VIII-12]

When the water quality control target values are specified, the component materials, operation conditions, operation methods, etc. of the reactor coolant system are compared to those of BWR plant, and it is determined that the component material integrity would not be affected if the same water quality target values as BWR are used because the component materials of "Fugen" are almost the same as those of BWR plant, the operation conditions (the steam drum pressure and temperature) are similar, the plant operation methods (the rated power operation was the basic method) are the same, etc. The adopted control target values are the same as those used for the BWR plants of that time as shown in the following:

[Water Quality Control Target Values for Reactor Coolant]

Item	Unit	Control Target Value
Conductivity	μS/cm	1.0 or less
pH		5.5 - 8.5
Chloride Ion	ppm	0.2 or less
Silica	ppm	2 or less
Boron	ppm	2 or less
Dissolved Oxygen	ppm	0.4 or less

The description on the specification bases for each control target value is not given here, for it has already been mentioned in the previous chapter V, Chenistiry Control.

(b) Heavy Water Quality Control [VIII-13]

Since "Fugen" is the first heavy water reactor for power generation in Japan, when the water quality control target values for the heavy water were determined, the management items and the control target values were specified based on the heavy water reactor experiences in foreign countries and the water quality control criteria for BWR plants, as well as JRR-2. The control target values are shown in the following:

[Heavy Water Quality Control Target Values]

Item	Unit	Control Target Value	Reference plant
Conductivity	μS/cm	5.0 or less	SGHWR
pD		4.5 - 8.5	
Chloride Ion	ppm	1.0 or less	
Silica	ppm	1.0 or less	BWR
Dissolved Oxygen	ppm	0.5 or less	BWR

(2) Water Quality Improvement Measures

In order to confirm that the system component material integrity of the rector coolant system and the heavy water system is maintained and the countermeasures against the crud accumulation on the piping and equipment are appropriately taken, the water quality improvement measures were investigated. The investigation result showed that the following measures are taken:

(a) Dissolved Oxygen Reduction in Reactor Coolant by Hydrogen Injection

This was done to prevent the Inter-Granular Stress Corrosion Cracking (IGSCC) of sensitive SUS304 stainless steel without any effect on plant operation.

(b) Reduction of Crud Delivery to Reactor by Injecting Oxygen into Feed Water System

Because of carbon steel pipes in the feed water and the condensate systems and bleeding system, corrosion of the pipes results in formation of crud. The crud is carried in the core by feed water and deposited on claddings and pressure tubes. The irradiated crud is dissolved in the coolant again and deposits on the inner surfaces of components and piping. This results in increasing of radiation.

(c) Reduction of Silica Concentration by Changing Ion Exchange Resin Ratio in Reactor Coolant Clean-up System

Since silica leak from the ion exchange resin occurs faster than the other ions, concentration of silica in the reactor coolant is apt to increase. The objective of the measure is to improve this fault.

(d) Heavy Water Quality Improvement Measures

① Reduction of the temperature of the heavy water flowing through the ion exchange resins which restrain the resin degradation by oxidation (D_2O_2 generated by the irradiation of the heavy water flows into the heavy water purification column and results in the damage of the resins. The objective of the measure is to improve the above point.)

② Reduction of chloride ion concentration in heavy water by adopting the low-chlorinated ion exchange resins.

③ Heavy water quality improvement by reducing the nitrogen concentration in helium (Nitrogen gas in the helium system is changed to nitric acid ion by the oxidation reaction in the core.)

1.7 Radiation Control [VIII-14]

In "Fugen", there are the dose equivalent controls for the corrosive products including Cobalt-60 and Manganese-54 generated in the rector coolant system, etc. which is the same as the light-water reactors, and the dose equivalent control of tritium which is specific to the heavy water reactor. It is important to perform these controls appropriately and to reduce the dose equivalent that the workers would receive. Therefore, the investigation was conducted to evaluate the dose equivalent change during the periodic inspection, the changes of the dose equivalent per major work, the changes of the measures for dose equivalent reduction and the outline of each measure, the dose equivalent control status, the dose equivalent control system and the changes of the

radiation environmental monitoring in the controlled areas. It was checked if the measures for the dose equivalent reduction were promoted as further as reasonably achievable to reduce the dose equivalent.

(1) Trend of Dose Equivalent

Among the dose equivalents during the periodic inspection, the dose equivalent which the workers receive during the normal periodic inspection (the dose equivalent during the normal periodic inspection) has been leveling off at about 2.0 - 3.0 man•Sv recently though the dose equivalent of the equipment, piping, etc. is increasing every year.

(2) Reduction Measures for Dose Equivalent per Major Work

(a) Overhauling of Refueling Machine

As the reactor structure of "Fugen" only allows the refueling from the lower part of the pressure tube, the crud (radioactive corrosive products) attached to the fuel assemblies is precipitated and deposited especially in the refueling machine whenever the refueling is performed, and the removal of the deposited crud is difficult because of the complex internal structure. As a result, the surface dose equivalent becomes high. Therefore, the water in the refueling machine was replaced, and during the 5th periodic inspection, the chemical decontamination of the refueling machine and the crud removal by the high-pressure jet water cleaning were conducted.

(b) Overhauling and Inspection of Reactor Recirculation Pumps

The dose equivalent caused by the pump inspection becomes high because the inspection is performed in the area with high dose equivalent rate due to the effect from the radiation of the recirculation system piping, etc. in the containment vessel where the pumps are located. Therefore, the chemical cleaning was performed for the pump main body as a dose equivalent reduction measure.

(c) In-Service Inspection (ISI) [VIII-15]

As the reduction measures for the dose equivalent during ISI, the temporary shielding using the lead mat, etc. was installed, and from the 2nd periodic inspection, the automatization of the ISI devices for inlet pipes and riser pipes has been promoted.

(3) Other Measures for Dose Equivalent Reduction

The dose equivalent reduction measures that have been implemented may be classified into the automatization of work, the reduction of dose equivalent in the working environment, the workload reduction, the notification of radiation

control information and the removal and diffusion prevention of tritium.

(a) Automatization of Work

① Development of the ISI machine for the inlet and riser pipes, and automatization of the chemical decontamination device for the lower shielding plug

(b) Reduction of Dose Equivalent in Working Environment

① Installation of the temporary shielding using the lead mat for the reactor coolant system piping in the containment vessel

② Chemical decontamination of the equipment

(c) Workload Reduction (Extension of Overhauling Cycles)

① Extension of overhauling cycles for the heat exchangers, pumps, etc. and the lifetime extension of the Local Power Monitor (LPM) to reduce the workload and shorten the working hours in the controlled areas

(d) Notification of Radiation Control Information

In order to reduce the dose equivalent, not only the measures including the shielding but also the encouragement for the workers' own initiative for the dose equivalent reduction and the careful and detailed radiation control are important. In "Fugen", "the Special Committee on Exposure Reduction" is organaged and the measures including the installation of the digital area monitors at site are taken.

(e) Tritium Removal and Diffusion Prevention Measures

In "Fugen", the prevention of internal exposure due to tritium, and of tritium diffusion is also essential. Therefore, the overhauling parts of the heavy water system equipment are dried in advance by the tritium removal system, and the dedicated working area (a house) which is ventilated by the tritium removal system is installed in order to remove tritium and prevent tritium from diffusing. In the meantime, the efforts are made to prevent the internal exposure caused by tritium. For example, the tritium protection clothes are developed and adopted.

(4) Dose Equivalent Control

In "Fugen, during operation and the periodic inspection, the planning is performed in advance, the work is implemented based on the plan, the dose equivalent is evaluated and examined, and the discussion on the item to be integrated for the next time is conducted per each work for the dose equivalent control.

① Measurement and evaluation by TLD badges and TLD badges for daily

control

② Alarm meters shall be carried with for the work under the high dose equivalent rate.

③ Inspection by whole body counters and bioassays when entering and leaving the plant, regularly (every three months) or every time for specific work.

④ Evaluation is made on the required radiation protection actions and the dose equivalent reduction actions, by evaluating the dose equivalent based on the work contents and the radiation environment in the working area with reference to the past working experience.

1.8 Radioactive Waste Management [VIII-14]

The radioactive nuclides generated in nuclear facilities are largely classified into the fission products directly generated from the fuel fission and the activated products generated from the activation of reactor structural materials, etc. Most of such radioactive nuclides are maintained in the fuel elements or in the facilities, whereas a very small part of them is released into the environment through its processing facilities in the form of gas or liquid.

It is extremely important for the gaseous and liquid radioactive wastes released into the environment accompanied by the operation of "Fugen" to undergo the actions for the reduction of radioactive material concentration and amount according to their physical properties, and to be controlled so that their concentration is kept below the discharge contorol target values specified in the reactor facility safety regulations and the law. At the same time, the efforts to keep the concentration sufficiently as low as reasonably achievable are also required. In addition, the actions for reducing the generated amount and the storage amount are also necessary for the radioactive solid waste.

Therefore, the investigation is conducted on the experiences in the each radioactive waste release and storage, the implementation status of the released and generated amount reduction measures for each waste and the operation management status of each waste. Based on the investigation, it is checked if the actions for reducing the released amount or the generated amount of the radioactive wastes are appropriately taken, and if the reduction of the released amount or the generated amount of the wastes is being promoted.

(1) Experience in Gaseous and Liquid Radioactive Waste Discharge and Solid Waste Discharge and Storage

(a) Experience in Gaseous Waste Discharge

① Radioactive Noble Gases

In the beginning of the commercial operation, the radioactive noble gases were discharged at about 10^{11}Bq/year.

② Iodine-131

Since no fuel failure has occurred, the discharged amount of Iodine-131 has been kept below the detection limits. The radioactivity of 10^{7}Bq detected on-site in 1986 was resulted from the accident of Chernobyl Nuclear Power Station.

③ Tritium

Fig. VIII. 1-4 shows the discharged amount of tritium.

④ Granular Materials

Except in 1986 when about 3×10^{6} Bq was detected on-site due to the influence of the Chernobyl accident, no granular materials have exceeded the detection limits.

(b) Experience in Liquid Waste Discharge

① Radioactive Materials Excluding Tritium

Most of the nuclides excluding tritium contained in the liquid wastes are γ nuclides. Most of them are generated from the laundry drain.

② Tritium [VIII-16]

The liquid tritium waste liquid is broadly classified into the waste liquid accompanied by the resin exchange of the heavy water purification system and the condensed water generated from the tritium removal system. Most of the tritium is discharged from the liquid tritium waste liquid. The discharged amount has been leveling off at about 3 - 5 x 10^{12}Bq / year.

(c) Experience in Solid Waste Generation and Storage

① Powdered Waste Resins

The powdered ion exchange resins are used in the demineralizer of the fuel storage pool cooling and purification system, and the powdered waste resins are generated when the resins of this demineralizer are exchanged. Though the generated amount of the powdered resins changes according to the number of handled fuel assemblies, for it depends on the purity of fuel storage pool water, it is usually about 0.5m^{3} / year.

② Granular Waste Resins

The granular resins are used in the demineralizers of the reactor coolant clean-up system, the heavy water purification system, the feed water

system (the condensate demineralizer) and the equipment drain system (the waste liquid demineralizer). The granular waste resins are generated when the resins in these demineralizers are exchanged. The generated amount is about 3 - 5cm^3 / year.

③ Miscellaneous Solid Wastes

The generated number of waste drum cans differs significantly depending on the status of the modification work including the piping replacement. The generated number of the waste drum cans is about 1,500 / year without the modification work from which much waste is generated. About 60% of the drum cans contain burnable wastes and the rest of them non-burnable wastes, judging from the breakdown of the content of the cans stored.

(2) Release Reduction Measures for Gaseous and Liquid Wastes and Generation Reduction Measures for Solid Wastes

(a) Gaseous Waste Reduction

① Reduction of Residual Air Amount by Strengthening Helium Purification (Reduction of Argon-41 Generation Amount)

A small amount of air is contained in the helium system, and Argon-41 is generated from the activation of Argon-40 in the air. The strengthening of the helium purification is performed (reduction of the target value of the residual nitrogen concentration in the helium system from 1% to 0.1% or less).

② Reduction of Noble Gas Discharged Amount by Strengthening Carbon Dioxide Gas System Purging

Since the atmosphere in the carbon dioxide gas system is the air atmosphere during the plant shutdown, the system is purged using the carbon dioxide prior to the reactor startup so that there would be as small amount of the residual air as possible.

(b) Liquid Waste Reduction

① Reduction of Discharged Amount of Tritium

Following measures were taken.

(i) Change of heavy water handling hood ventilation

The ventilation of the hood was changed to be always conducted in the tritium removal system, and the improvement was made for the heavy water steam transferred into the air to be adsorbed with the drying agent (molecular sieve).

(ii) Tritium discharge reduction by the improvement of the heavy water purification system resin exchange method

(iii) Implementation of the heavy water drain and drying

The piping with small diameters around the pumps are improved so that the heavy water in the equipment is drained and collected prior to the heavy water system equipment overhauling and that the amount of the residual heavy water amount is reduced.

(c) Radioactive materials excluding tritium

① Evaporation and concentration processing, etc. of the equipment drain and floor drain waste liquid

② Introduction of dry-cleaning system

③ Introduction of a shoes washing machine

(d) Reduction of Radioactive Materials Excluding Tritium

Following measures were taken.

① Evaporation and concentration processing, etc. of the equipment drain and floor drain waste liquid

② Introduction of dry-cleaning system

③ Introduction of a shoes washing machine

(e) Solid Waste Reduction

Following measures were taken.

① Restriction on carrying of goods into the controlled area

② Change of curing material

③ Thermal insulation for the metal used for the reactor coolant system piping

④ Compression and volume reduction of the ventilation system high-efficiency particulate air filter (HEPA filter)

⑤ Reduction of stored burnable dry active waste amount

(3) Operation Management of Radioactive Wastes

(a) Gaseous and Liquid Wastes

For the gaseous and liquid wastes, the appropriate management is performed through each of the pre-dischageing stage, the implementation stage and the evaluation stage. At the same time, the reduction measures for the discharged radioactivity are steadily conducted.

(b) Solid Wastes

For the solid wastes, the appropriate management is performed through each of the pre- discharging stage, the implementation stage and the evaluation stage. At the same time, the reduction measures for the

discharged radioactivity are steadily conducted.

1.9 Emergency Actions

For the reactor facilities, the sufficient safety measures are taken based on the Law for the Regulations of Reactors, etc. for the accident prevention, the prevention of accident expansion and the prevention of disasters.

However, regardless of such safety measures, the national and local governments are preparing disaster-relief measures against an accident which might occur and influence the peripheral regions. It is important that the nuclear power station shall provide the organization, resources and materials and shall implement the training assuming the emergency conditions so that the appropriate actions can be taken during emergency.

Therefore, the investigation is conducted on the establishment of the organization for the emergency, the communication and notification, the prevention of contamination from spreading, the restoration measures and the implementation status of the training for emergency actions. It is checked based on the investigation result if the required actions are to be performed properly.

The following items were investigated to check if the emergency actions or operating procedures are to be performed properly:

① Establishment of Organization for Emergency
② Communication and Notification
③ Leading of Evacuation
④ Medical Service During Emergency
⑤ Estimation on Radiation Affected Range
⑥ Prevention of Contamination from Spreading
⑦ Restoration Measures
⑧ Providing Status of Resources and Materials for Emergency
⑨ Implementation Status of the Training of Emergency Actions or Operating Procedures

The investigation result revealed that these items were appropriately carried out.

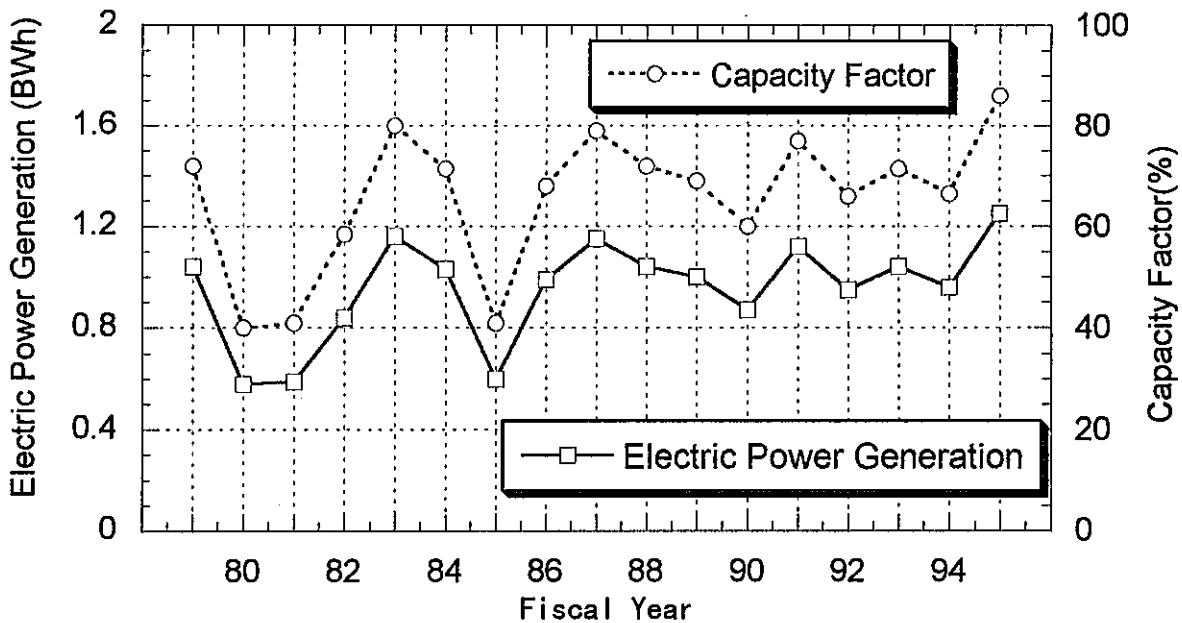


Fig.VIII.1-1 History of Electric Power Generation and Capacity Factor

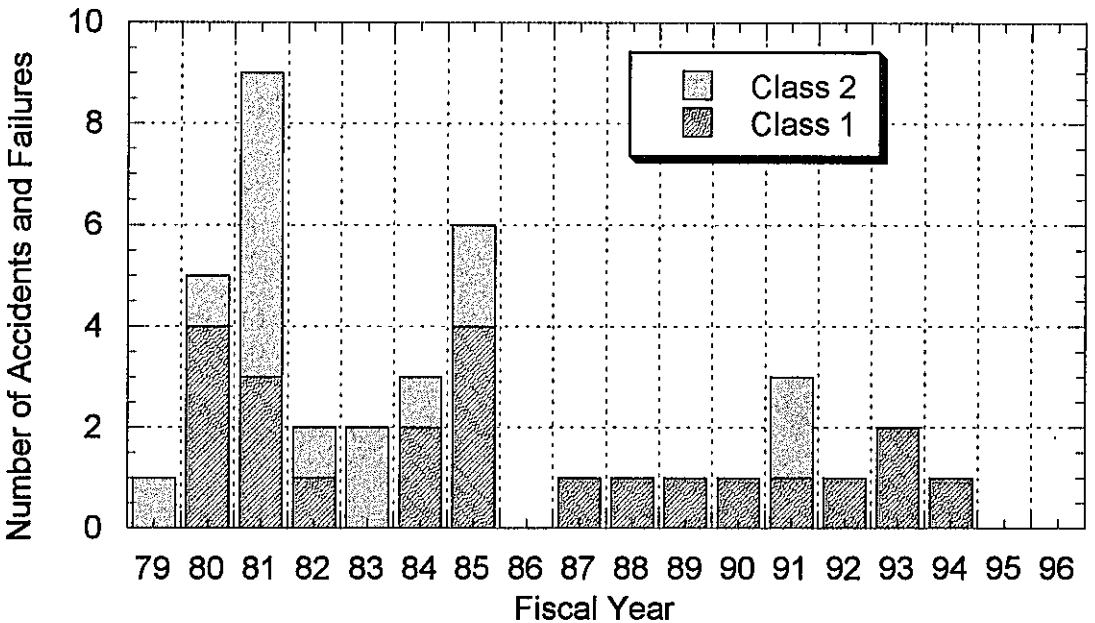
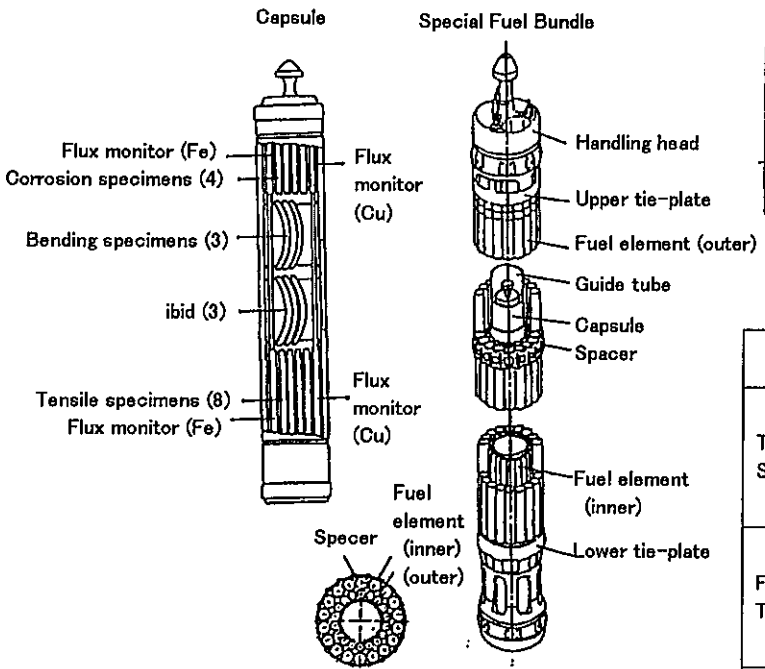
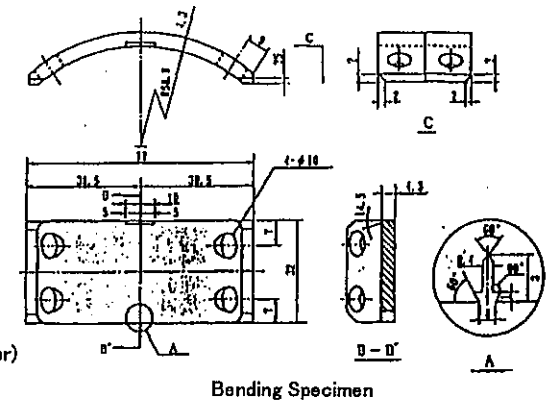
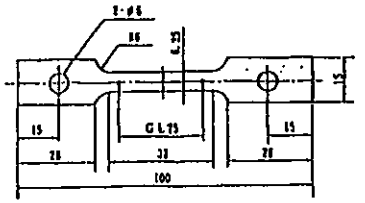


Fig. VIII.1-2 History of Reported Accidents and Failures

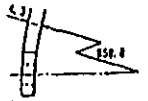




Bending Specimen



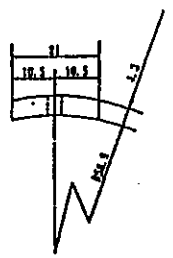
Tensile Specimen



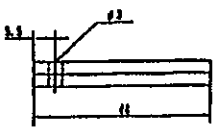
Axial Direction Specimen



Circumferential Direction Specimen



Corrosion Specimen



Item		Design Value	Measured Value
Tensile Stress	Room temp.	>77.9	98.2
	300 ℃	>54.8	73.3
Fracture Toughness	Room temp.	>35.0	136.9
	300 ℃	>40.6	215.9
Corroded Thickness (mm)		<0.31	0.0147
Hydrogen ConcentrStion (ppm)		<211	44.4

Fig.VIII.1.-3 PIE Result of Pressure Tube Monitoring Specimens

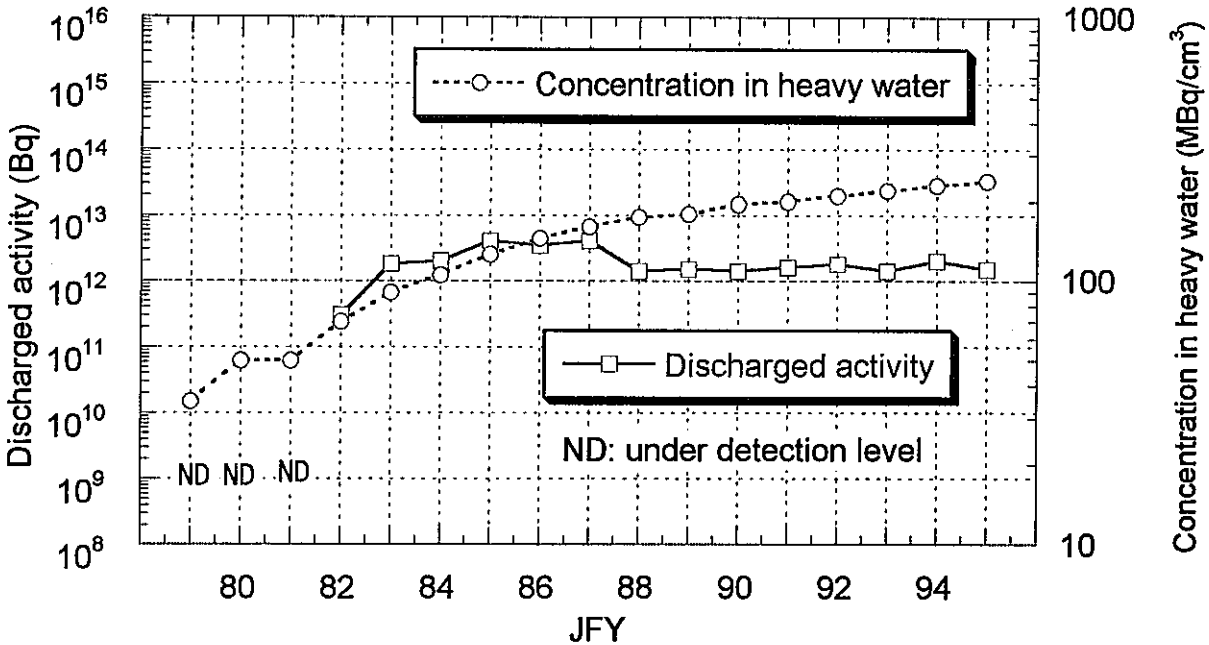


Fig. VIII.1-4 History of tritium discharge

2. PSA in "Fugen" [VIII-17] [VIII-18] [VIII-19] [VIII-20] [VIII-21]

2.1 Introduction

The PSA (Probabilistic Safety Assessments) of "Fugen" has been performed based on the studies [VIII-22] [VIII-23], etc. conducted in light-water reactors in the past, and the method suitable for the heavy water reactor has been developed taking the studies on severe accidents into consideration [VIII-24]. In the PSA in "Fugen", the core damage probability and the probability of radiation release to the environment have been calculated tentatively. The development objectives are to enable the analyses of Level 1 and Level 2 and to use the method to accident management:

Level 1: To make fault trees with considering common mode failure for systems which carry out important safety functions, and to identify the accident sequence leading to the core damage in order to evaluate the occurrence frequency of the sequence. Consequences of earthquakes and fires are not considered in the present evaluation.

Level 2: To evaluate the probability of the sequence of radioactive material releasing into the environment and the released amount of radioactive materials by integrating the sequence during the accident and the thermal-hydraulic response analyses in the containment vessel into the result of Level 1.

The most different point of heavy-water reactors from the light-water reactors is the existence of the heavy water which serves as a heat sink of the decay heat. Therefore, the severe accident studies conducted so far have indicated that generally the temperature of the fuel inside the ATR pressure tubes can be maintained at a level which never causes the fuel melting due to radiation / heat transfer even in such an accident scenario that leads to the core damage with no ECCS actuation during LOCA in light-water reactors. [VIII-25], [VIII-26] and [VIII-27] Reference [VIII-28] shows the analysis results concerning the operation status of "Fugen". Fig. VIII. 2-1 shows the results. Meanwhile, it was investigated as an experiment that when the reactor protection system was not actuated during LOCA which led to the fuel failure due to the positive reactivity insertion, first the fuel with high power failed, the light water was released from the failed fuel channel to the heavy water moderator to produce large steam bubble in the heavy water, the negative reactivity was inserted and the accident was brought to its end. [VIII-29] and [VIII-30]. The shape of measured bubble in the heavy water is indicated in Fig. VIII. 2-2. The basic data concerning the

calandria tank integrity were acquired in the experiment using the 1/6 scale model so that the deformation could be evaluated using the analysis code. [VIII-31] The outline of the experiment device is shown in Fig. VIII. 2-3. At the same time, the basic experiment [VIII-32] using the device shown in Fig. VIII. 2-4 was performed for the melted fuel behavior which led to the failure of the pressure tubes and calandria tubes; the more detailed experiment is being conducted. The MAAP-FUGEN code is being prepared by taking such experiment results into consideration.

The revision of the CONTAIN code has made the analysis of the detailed behavior in the containment vessel available. As a part of the code revision, the participation in the benchmark analysis on the international standard problems under the auspices of OECD / NEA was performed to implement the deposit behavior analysis of the thermal-hydraulic radioactive materials in the containment vessel. [VIII-33] In the meantime, the thermal-hydraulic analysis on the containment vessel of "Fugen" was also implemented. [VIII-34]
Such revision of analysis codes is performed in order to obtain the success criteria of the reactor cooling at PSA through the analyses.

2.2 Method Development

In order to perform the Level 1 analysis, the calculation of the core damage probability by quantifying fault trees (FTs) which consist of causes of failures of the components and event trees (ETs) which express passes of success and failure is required. While the evaluation of the released amount of radioactive materials, taking into account the thermal-hydraulic behavior in the containment vessel and situations of the containment vessel, is required for the Level 2 analysis. The review results on the method of applying PSA to ATR and its characteristics are described in the following:

(1) Compilation of Plant Data and Plant Behavior Data:

The design data on plant facilities including the related systems and equipment are acquired and compiled. At the same time, the behavior during the abnormal transients and accidents are collected as the data. Such behavior can be evaluated using the group of ATR safety assessment codes shown in Fig. VIII. 2-5. The validity of the analysis code for abnormal transients and accidents has been confirmed through the startup test of "Fugen", various piping rupture tests using the full-scale test loop in O-arai Engineering Center and the ECCS

performance test. [VIII-35] - [VIII-40] The success criteria can be calculated using these codes.

(2) Selection of Component Failure Data:

For the component failure data and the human error data required for the quantification of FT, WASH-1400, IEEEstd500, NUREG/CR-2815, etc. are used, like for the light-water reactors. The equipment function tests and operation experience of "Fugen" are also taken into consideration.

(3) Selection and Classification of Common Mode Initiating Events:

The same method as that is used in NUREG-1150 evaluation for the events that may cause the core damage resulting in the radioactive material release into the environment is used in order to identify the initiating events. When MLD (Mean Lethal Dose) is prepared, the characteristics of ATR shall be considered.

(4) Preparation of ET Concerning Core Damage:

An ET (Event Tree) is prepared for the classified initiating event based on the plant behavior data. When the ET is prepared, it shall be considered that the reactor coolant system has two loops.

(5) Preparation of FT:

A FT (Fault Tree) concerning each of engineered safety features and safety-related facilities is provided. The FT is developed as far as the equipment and part level in the failure data of WASH-1400 so that the data base for the U. S. light-water reactors can be applied to ATR.

(6) Quantification of Accident Sequence:

The occurrence frequency for the initiating event is calculated, like for the light-water reactors, based on the experiences of domestic light-water reactors and "Fugen". SETS code developed in the U. S. is used for the FT quantification.

(7) PCV Response Analysis:

In the early PSA, MARCH-ATR code that had been widely used for the light-water reactor PSA was modified so that it could conform to ATR, and used. The modified model is as follows:

① The Model of Heat Removal from Core by Heavy Water and Heavy Water Cooling System

② Erosion Model of Shielding Plug at Bottom of Pressure Tube Assembly

③ ATR Decay Heat Model Based on ANSI/ANS 5.1-1979 Model

④ Steam Discharge Pool Cooling System Model

⑤ Safety Valve and Steam Discharge Pool Model

At the same time, the CONTAIN code was applied in order to perform the detailed thermal-hydraulics and source term analyses in the containment vessel. Recently, the revision of the MAAP-FUGEN code based on the latest version has been ordered to FAI Corporation so that the series of behavior in the core and the containment vessel can be analyzed. [VIII-41] In the modification, correlations obtained from the blowdown experiments, models obtained by the severe accident experiments are incorporated into the code.

(8) Quantification of PCV-ET:

Based on the PCV response analysis, PCV-ET is set and the success criteria of the PCV-ET are specified. According to this, the FT evaluation is performed for the PCV mitigation system, and the PCV-ET is quantified using SETS code.

(9) Evaluation of Released Amount of Radioactive Materials:

The released amount is evaluated using the CONTAIN code and MAAP-FUGEN code. The CONTAIN code is used when the chain reactions of released materials should be considered in the evaluation.

2.3 Application Results

(1) Initiating Event

The frequencies of initiating events are shown in Fig. VIII.2-6. The initiating events are summarized into the seven items as shown in this figure.

(2) Calculation of Core Damage Frequency

Based on the approximate calculation, the initiating events with the occurrence frequency of 10^{-10}/(reactor・year) or more were selected as the major accident sequence. The core damage frequency can be calculated by quantifying the major accident sequences for each of these selected initiating events. Since little influence of the accident sequence with the frequency of 10^{-10}/(reactor・year) on all of the core damage frequency is expected, the accident sequences with the frequency of 10^{-10}/(reactor・year) or more were used for the quantification by the SETS code. The point estimation result shows the core damage frequency of approximately 1×10^{-6}/(reactor・year). Fig. VIII.2-7 indicates the breakdown of the core damage frequency for each initiating event. As is shown in the figure, about 50% is the frequency of core damage due to loss of external power. Regarding the core damage frequency for each accident sequence, Fig. VIII.2-8 illustrates the consequence.

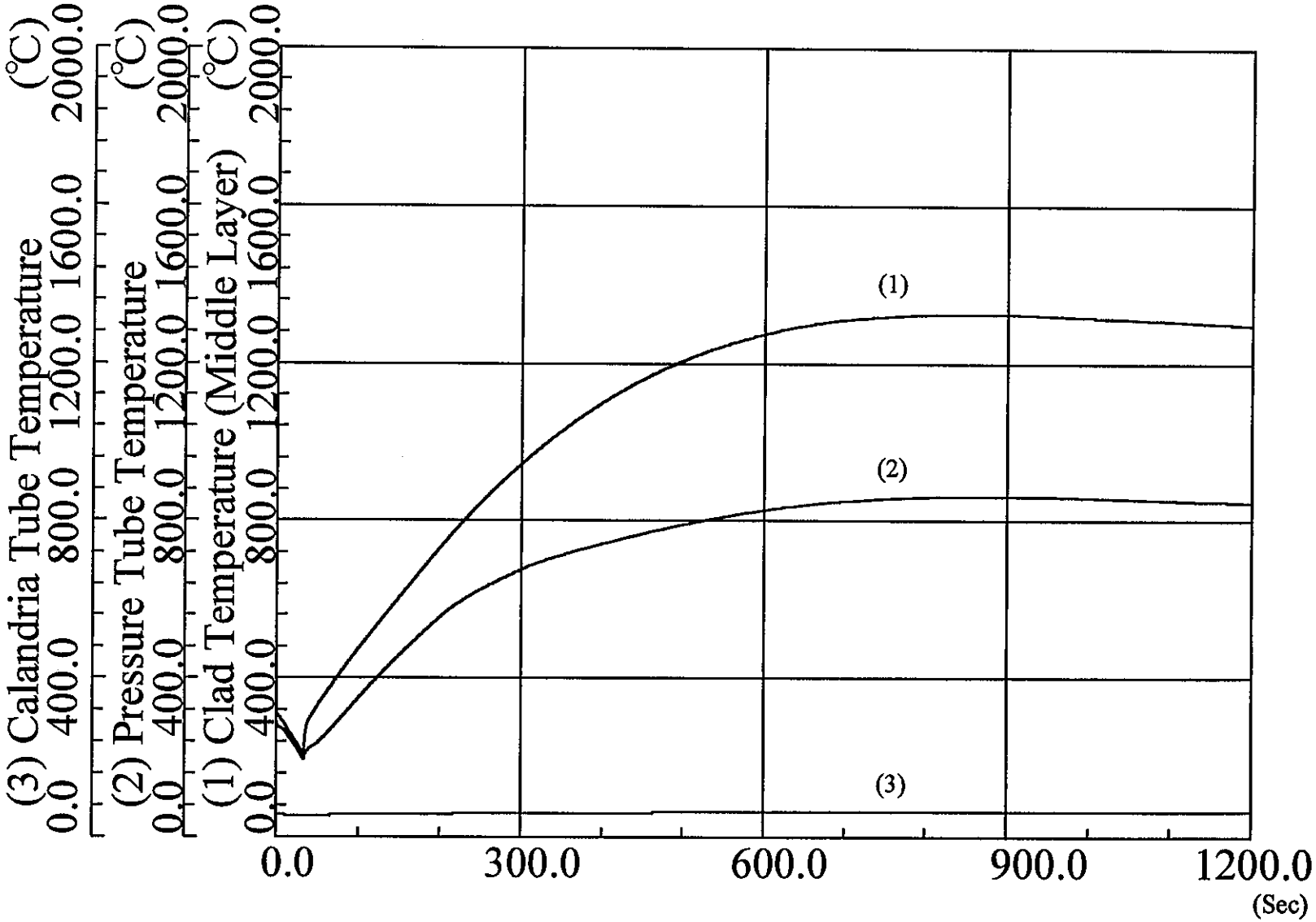


Fig.VIII.2-1 Heat-up Characteristics on Guillotine Break of Downcomer with ECCS Unavailable by HESTIA Code

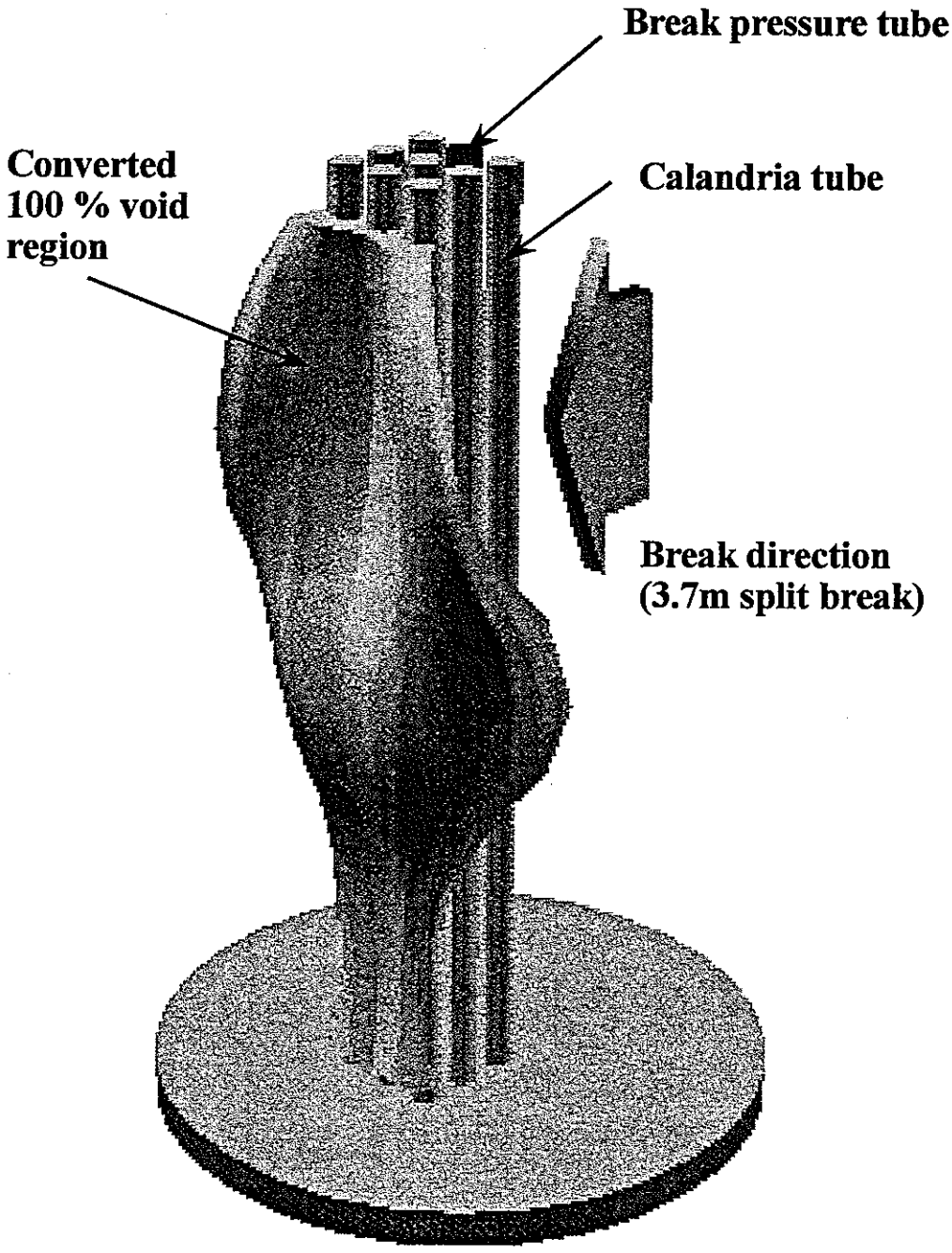


Fig. VIII.2-2 Void Configuration 2.5 sec after 3.7 m Split Break

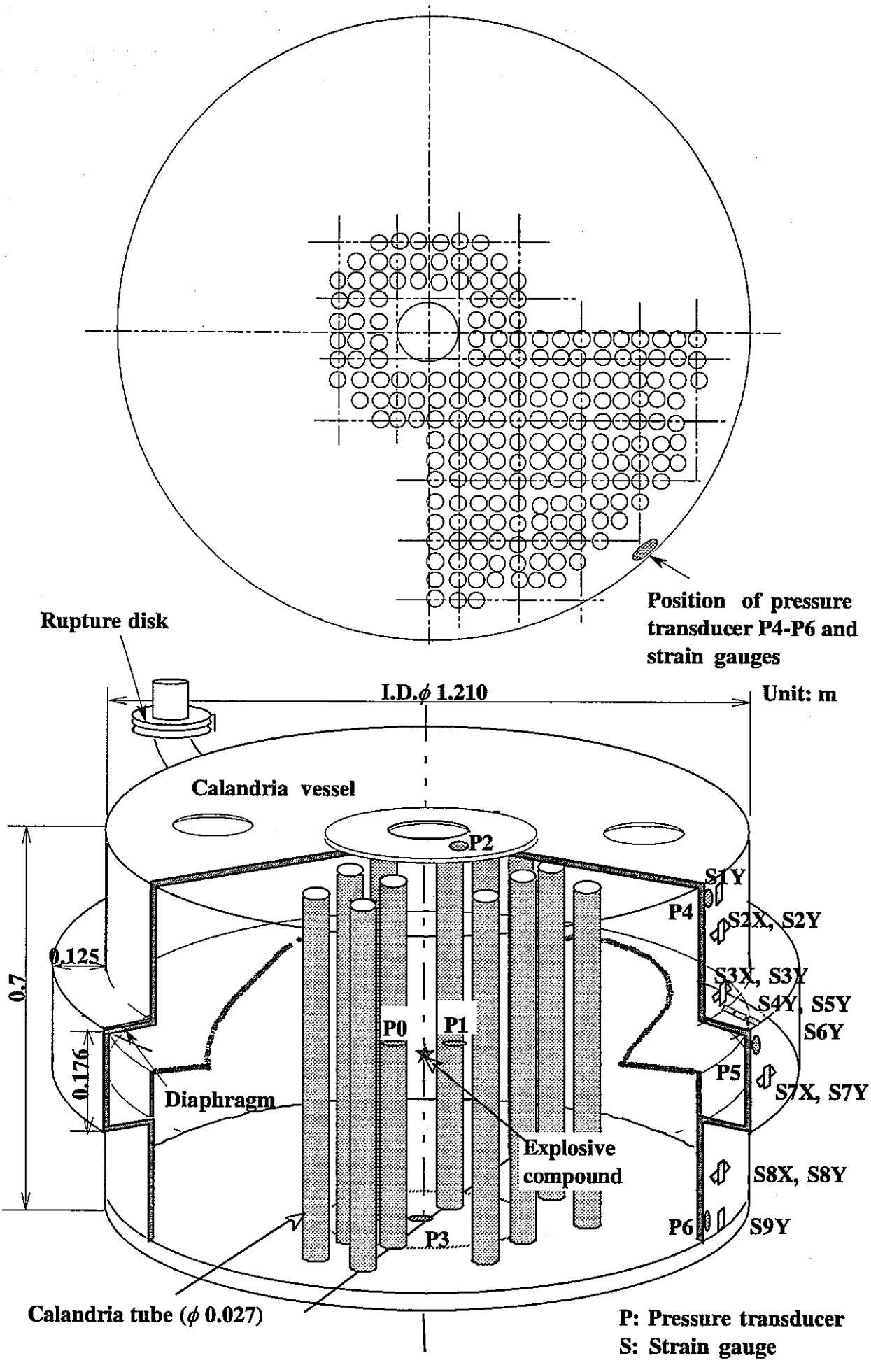


Fig. VIII.2-3 1/6-Scale Impact Experiment Set-up

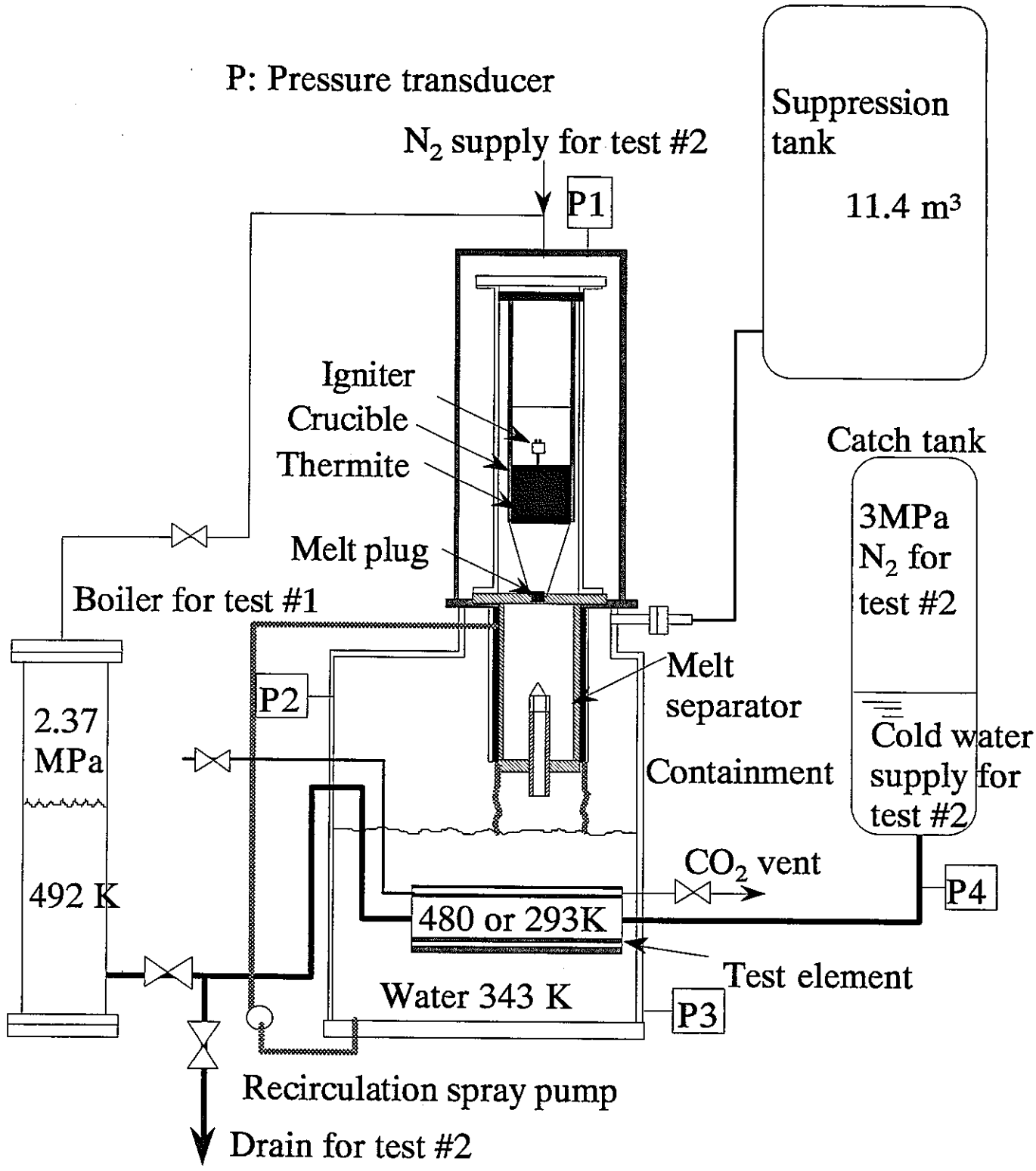


Fig. VIII.2-4 Facility Configuration of Melt Attack Test

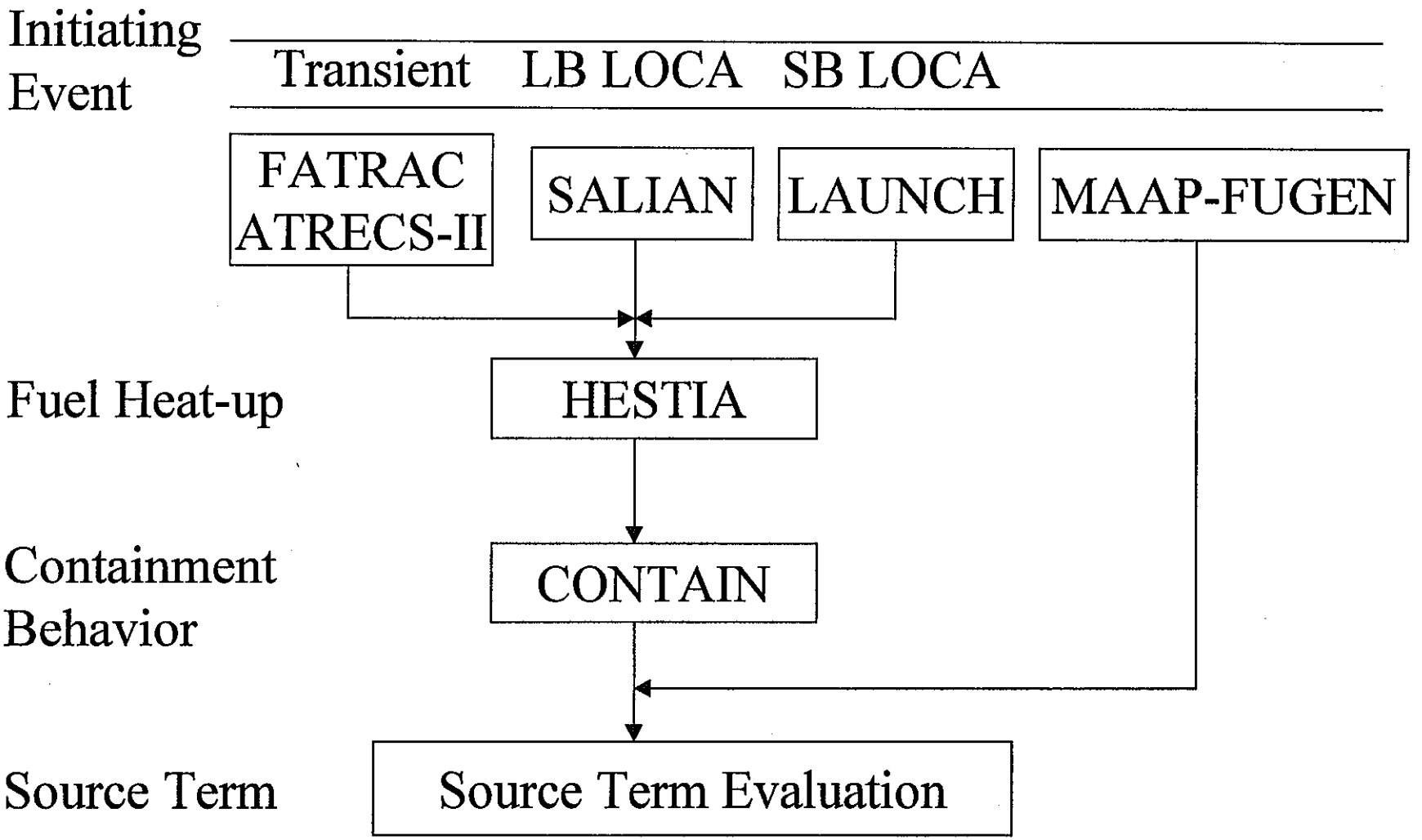


Fig. VIII.2-5 Success Criteria Evaluation Code for ATR

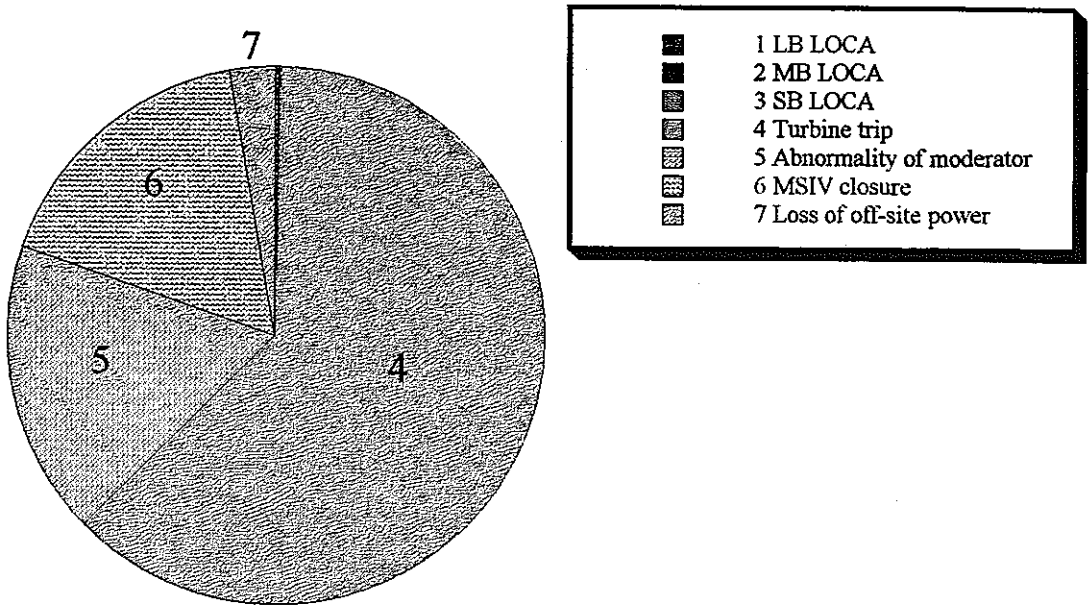


Fig. VIII.2-6 Probability of Initiating Event

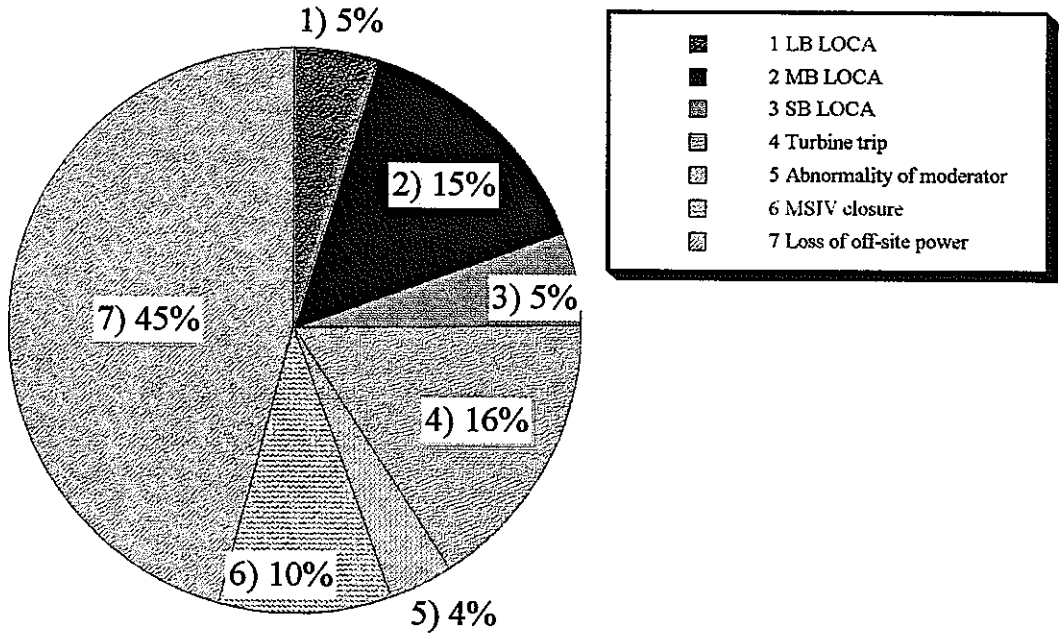


Fig. VIII.2-7 Probability of Core Damage for Each Initiating Event

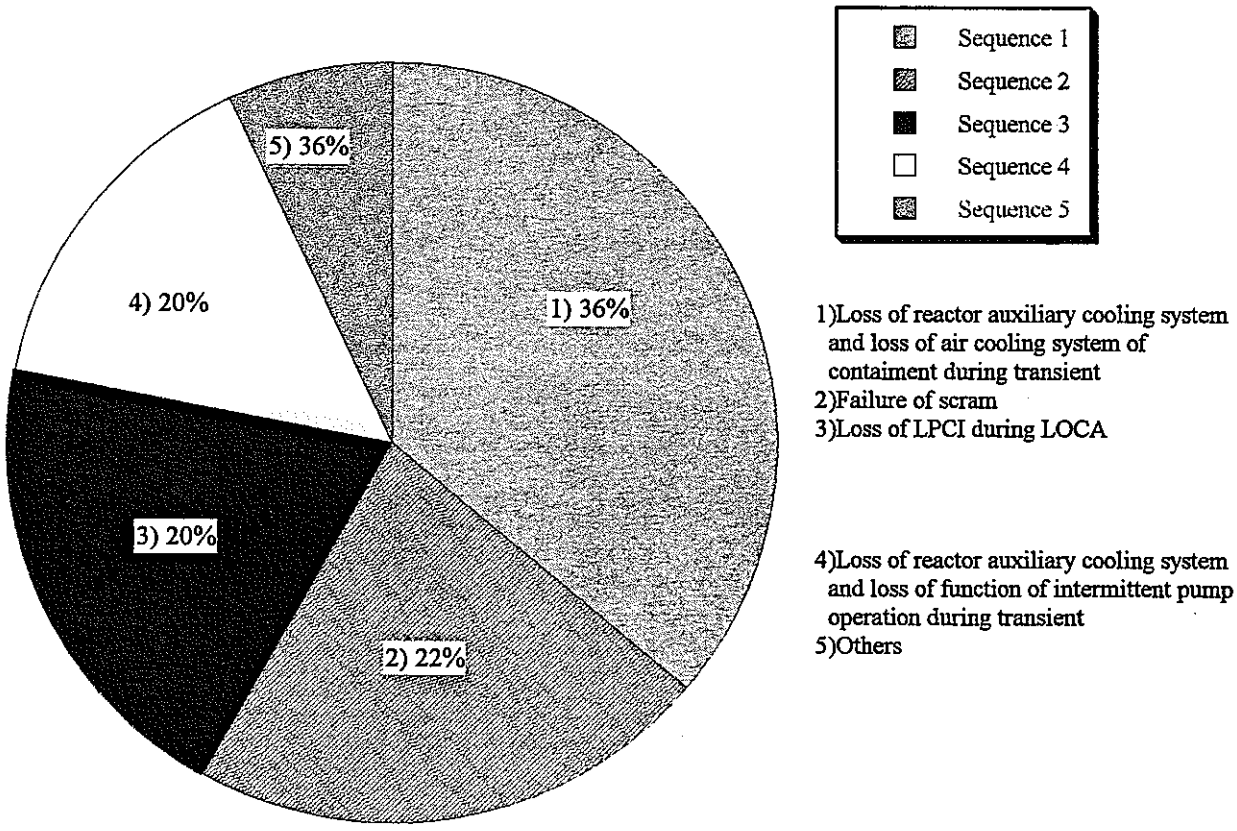


Fig. VIII.2-8 Probability of Core Damage for Each Sequence

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