IV. Occurrence and Progress of Accidents in Fukushima Nuclear Power Stations and Other Facilities

1. Outline of Fukushima Nuclear Power Stations

(1) Fukushima Daiichi Nuclear Power Station

Fukushima Daiichi Nuclear Power Station (hereinafter referred to as NPS) is located in Okuma Town and Futaba Town, Futaba County, Fukushima Prefecture, facing the Pacific Ocean on the east side. The site has a half oval shape with the long axis along the coastline and the site area is approx. 3.5 million square meters. This is the first nuclear power station constructed and operated by the Tokyo Electric Power Company, Incorporated (hereinafter referred to as TEPCO). Since the commissioning of Unit 1 in March 1971, additional reactors have been constructed in sequence and there are six reactors now. The total power generating capacity of the facilities is 4.696 million kilowatts.

Table IV-1-1 Power Generating Facilities of Fukushima Daiichi NPS

	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6
Electric output (10,000 kW)	46.0	78.4	78.4	78.4	78.4	110.0
Start of construction	Sep. 1967	May 1969	Oct. 1970	Sep. 1972	Dec. 1971	May 1973
Commissioning	Mar. 1971	Jul. 1974	Mar. 1976	Oct. 1978	Apr. 1978	Oct. 1979
Reactor type	BWR-3	BWR-4				BWR-5
Containment type	Mark I					Mark II
Number of fuel assemblies	400	548	548	548	548	764
Number of control rods	97	137	137	137	137	185

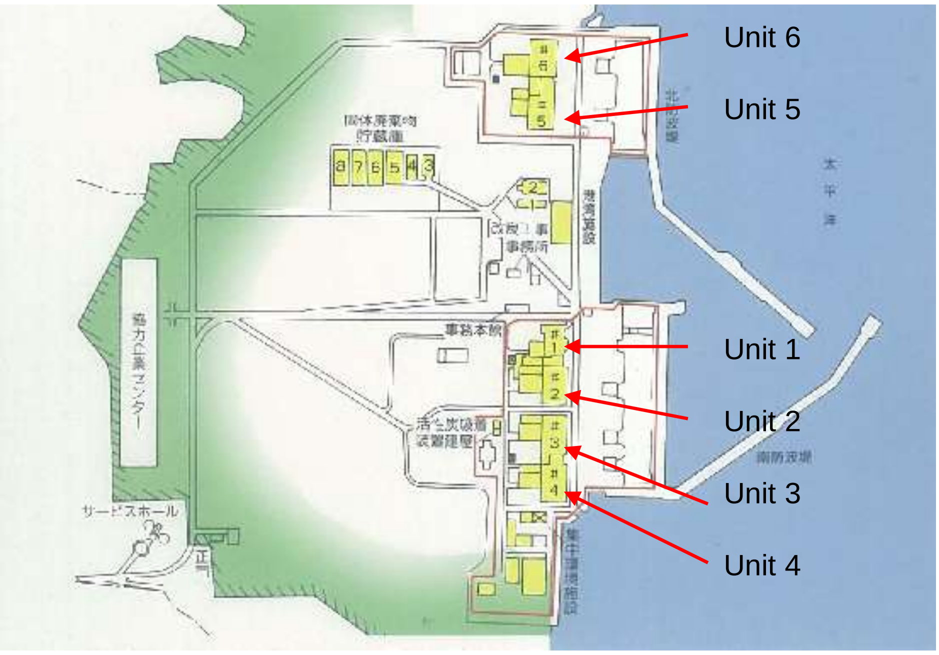

Figure IV-1-1 General Layout of Fukushima Daiichi NPS

(2) Fukushima Daini NPS

Fukushima Daini NPS is located in Tomioka Town and Naraha Town, Futaba County, Fukushima Prefecture, approx. 12 km south of Fukushima Daiichi NPS, and faces the Pacific Ocean on the east side. The site has a nearly square shape and the site area is approx. 1.47 million square meters. Since the commissioning of Unit 1 in April 1982, additional reactors have been constructed in sequence and there are four reactors now. The total power generating capacity of the facilities is 4.4 million kilowatts.

Table IV-1-2 Power Generating Facilities of Fukushima Daini NPS

	Unit 1	Unit 2	Unit 3	Unit 4
Electric output (10,000 kW)	110.0	110.0	110.0	110.0
Start of Construction	Nov. 1975	Feb. 1979	Dec. 1980	Dec. 1980
Commissioning	Apr. 1982	Feb. 1984	Jun. 1985	Aug. 1987
Reactor type	BWR-5			
Containment type	Mark II	Improved Mark II		
Number of fuel assemblies	764	764	764	764
Number of control rods	185	185	185	185

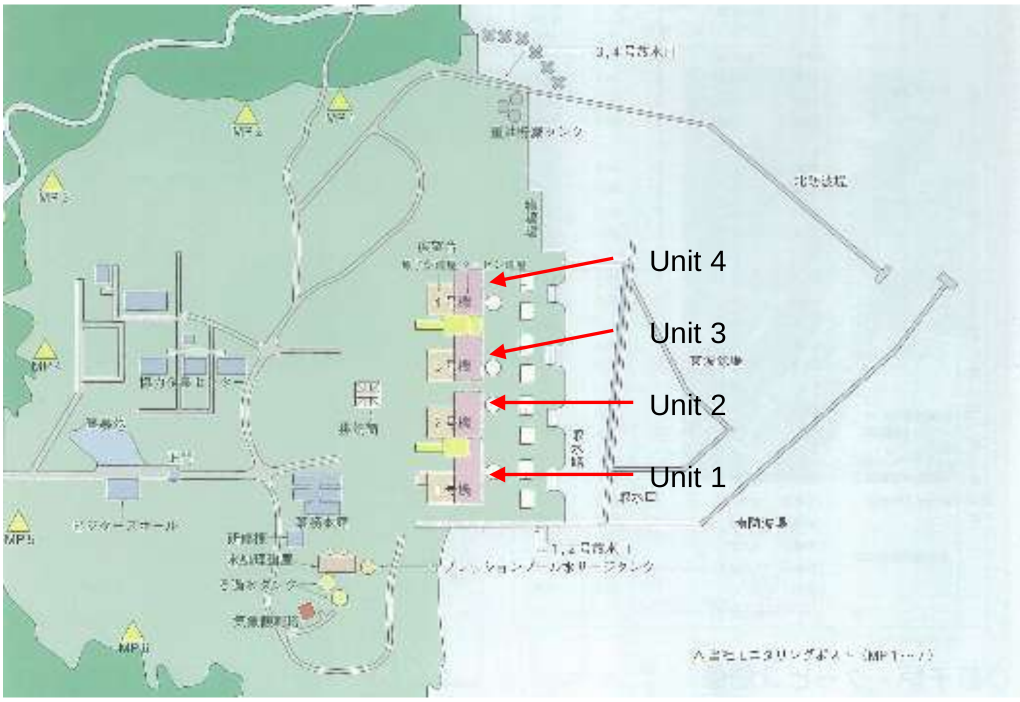

Figure IV-1-2 General Layout of Fukushima Daini NPS

2. Safety Assurance and Other Situations in Fukushima NPSs

(1) Design requirements of nuclear power stations

As described in Chapter II, nuclear power stations must satisfy legal requirements specified in the Reactor Regulation Act, the Electricity Business Act and other relevant laws and regulations.

When receiving an application for installing a nuclear power station from an applicant, Nuclear and Industrial Safety Agency (hereinafter referred to as NISA) conducts the primary safety review, should consult the Nuclear Safety Commission (hereinafter referred to as the NSC Japan) and shall receive their opinion based on the result of their secondary safety review. After NISA considers the opinions of the NSC Japan and examines the results of the safety reviews, the Minister of Economy, Trade and Industry gives the applicant permission to install individually for each reactor. In these safety reviews, NISA and the NSC Japan check that the basic design or the basic design policy of the nuclear power station conforms to the permission criteria specified in the Reactor Regulation Act, for example, in Article 24, “The location, structure, and equipment of the nuclear reactor facility shall not impair prevention of disasters caused by the nuclear reactor, its nuclear fuel material, or objects contaminated with the nuclear fuel material.” The NISA Japan conducts safety reviews based on the most recent knowledge and by referring to regulatory guides established by the NSC Japan as specific judgment criteria.

Regulatory guides are roughly divided into four types: siting, design, safety evaluation, and dose target values. One of the regulatory guides for design, the “Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities,”[IV2-1] (hereinafter referred to as Regulatory Guide for Reviewing Safety Design) specifies the basic design requirements for nuclear power stations. It contains a provision about design considerations against natural phenomena, which specifies that structures, systems, and components (SSCs) with safety functions shall be designed to sufficiently withstand appropriate design seismic forces and shall be designed such that the safety of the nuclear reactor facilities will not be impaired by postulated natural phenomena other than earthquakes, such as floods and tsunami.

It also specifies requirements for safety design against external human induced events, such as collapse of a dam, and fires and others.

Basic Judgment criteria for validation of design policies against earthquakes and tsunami are specified in the "Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities"[IV2-2] (the latest version established by the NSC Japan in September 2006, hereinafter referred as Regulatory Guide for Reviewing Seismic Design), which supplements the Regulatory Guide for Reviewing Safety Design.

The Regulatory Guide specifies the basic policy, "Those Facilities designated as important from a seismic design standpoint shall be designed to bear even those seismic forces exerted as a result of the earthquake ground motion, which could be appropriately postulated as having only a very low possibility of occurring within the service period of the Facilities and could have serious affects to the Facilities from seismological and earthquake engineering standpoints, considering the geological features, geological structures, seismicity, etc. in the vicinity of the proposed site, and such Facilities shall be designed to maintain their safety functions in the event of said seismic forces." It also specifies that uncertainties (dispersion) in formulating the Design Basis Ground Motion Ss shall be considered by appropriate methods and that the probabilities of exceedence should be referred to.

The Regulatory Guide also contains consideration of tsunami as accompanying events of earthquakes, "Safety functions of the Facilities shall not be significantly impaired by tsunami of such magnitude that they could only be reasonably postulated to have a very low probability of occurring and hitting the Facilities within the service period of the Facilities." A commentary in this Regulatory Guide describes that at the design of the Facilities, appropriate attention should be paid, to possibility of occurrence of the exceeding ground motion to the determined one and, recognizing the existence of this "residual risk", every effort should be made to minimize it as low as practically possible.
The NSC Japan requests that government agencies ask licensees to conduct backchecks of seismic safety based on specifications in this Regulatory Guide, along with quantitative assessment of "residual risks" by positively introducing the probabilistic safety assessment (hereinafter referred to as PSA), and review the results. In response to this request, NISA issued "Implementation of seismic safety assessment on existing nuclear power reactor facilities and other facilities to reflect the revisions of the 'Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities' and other safety assessment regulatory guides"[IV2-3] and requested licensees to carry out backchecks of seismic safety and assess "residual risks".

(2) Design basis events to be considered in safety assessment

1) Defining design basis events in safety assessment

As described in Chapter II, the Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities identifies events to be considered in the safety design and assessment of nuclear facilities and defines them as design basis events.

Design basis events regarding loss of external power supply, total AC power loss, and systems for transporting heat to the ultimate heat sink (hereinafter referred to as the ultimate heat sink), which occurred as part of this accident, are described below.

The Regulatory Guide for Evaluating Safety Assessment of Light Water Reactor Facilities takes loss of external power supply as an abnormal transient during operation and requires check of appropriateness of relevant safety equipment. On the contrary, the Regulatory Guide for Reviewing Safety Design does not take total AC power loss as a design basis event. This is because it requires emergency power supply systems to be designed with a high degree of reliability as AC power supplies. Specifically, the "Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities"[IV2-4] (established by the NSC Japan in August 1990, hereinafter referred as Regulatory Guide for Reviewing Classification of Importance of Safety Functions) classifies emergency power supply systems as systems with safety functions of especially high importance. The Regulatory Guide for Reviewing Safety Design specifies in its guidelines, such as Guideline 9 (Design Considerations for Reliability) and Guideline 48 (Electrical Systems), that systems with safety functions of especially high importance shall be designed with redundancy or diversity and independence and shall be designed such that adequately high reliability will be ensured. As described above, the Regulatory Guide for Reviewing Seismic Design specifies that safety functions shall be maintained in the event of an earthquake. Based on this prerequisite, the Regulatory Guide for Reviewing Safety Design specifies that the nuclear reactor facilities shall be designed such that safe shutdown and proper cooling of the reactor after shutting down can be ensured in case of a short-term total AC power loss, in Guideline 27 (Design Considerations against Loss of Power). However, the commentary for Guideline 27 states that no particular considerations are necessary against a long-term total AC power loss because the repair of interrupted power transmission lines or an emergency AC power system can be depended upon in such a case, and that the assumption of a total AC power loss is not necessary if the emergency AC power system is reliable enough by means of system arrangement or management. Accordingly, licensees are to install two independent emergency diesel generator systems (hereinafter referred to as emergency DG), which are designed such that one emergency DG is activated if the other emergency DG is failed, and that the reactor is shut down if a failure persists for a long time.

Loss of all seawater cooling system functions is not taken as a design basis event. This is because the Regulatory Guide for Reviewing Classification of Importance of Safety Functions classifies seawater pumps as systems with safety functions of especially high importance, just like emergency power supply systems. The Regulatory Guide for Reviewing Safety Design specifies that systems with safety functions of especially high importance shall be designed with redundancy or diversity and independence, in Guideline 9 (Design Considerations for Reliability), Guideline 26 (Systems for Transporting Heat to Ultimate Heat Sink) and other guidelines. Also, the Regulatory Guide for Reviewing Seismic Design specifies that safety functions shall be maintained in the event of an earthquake.

The generation of flammable gas inside the primary containment vessel (hereinafter referred to as PCV) when reactor coolant is lost is postulated in the design basis events as a cause of hydrogen explosion accidents. To prevent this event, a flammability control system (hereinafter referred to as FCS) that suppresses hydrogen combustion inside the PCV is installed in compliance with Guideline 33 of the Regulatory Guide for Reviewing Safety (the system controlling the atmosphere in the reactor containment facility). Additionally, keeping the atmosphere inside the PCV inert further reduces the possibility of hydrogen combustion. These designs are aimed at preventing hydrogen combustion in the PCV from the viewpoint of PCV integrity, and are not aimed at preventing hydrogen combustion inside the reactor building.

2) Safety design for the design standard events at Fukushima NPSs

The safety designs for the design basis events of offsite power supplies, emergency power supply systems, and reactor cooling functions related to the accidents at Fukushima NPSs are the following:

The power sources are connected to offsite power supply grids via two or more power lines. Multiple emergency diesel generators are installed independently with redundant design as the emergency power supplies for a loss of external power supply. Also, to cope with a short-period loss of all AC power sources, emergency DC power sources (batteries) are installed maintaining redundancy and independence.

Unit 1 of Fukushima Daiichi NPS is equipped with isolation condensers[1] (hereinafter referred to as IC) and a high pressure core injection system (hereinafter referred to as HPCI), and Unit 2 and Unit 3 of Fukushima Daiichi NPS are equipped with HPCI and a reactor core isolation cooling system[2] (hereinafter referred to as RCIC) to cool the reactors when they are under high pressure and the condenser does not work. Unit 1 of Fukushima Daiichi NPS is equipped with a core spray system (hereinafter referred to as CS) and a reactor shut-down cooling system (hereinafter referred to as SHC), and Unit 2 and Unit 3 of Fukushima Daiichi NPS are equipped with a residual heat removal system (hereinafter referred to as RHR) and a low pressure CS to cool the reactors when they are under low pressure.

Additionally, in the main steam line that leads to the reactor pressure vessel (hereinafter referred to as RPV) are installed main steam safety relief valves (hereinafter referred to as SRV) that discharge steam in the reactor to the suppression chamber (hereinafter referred to as S/C) and safety valves that discharge steam in the reactor to the dry well (hereinafter referred to as D/W) of the PCV. The SRV functions as an automatic decompression system. Table IV-2-1 shows a comparison between these safety systems. Their system structures are shown in Figures IV-2-1 to IV-2-7.

As shown in Figure IV-2-8 and Figure IV-2-9, the heat exchanger in the SHC for Unit 1 or RHR for Units 2 and 3 of Fukushima Daiichi NPS transfers heat using seawater supplied by the seawater cooling system to the sea, as the ultimate heat sink.

To prevent hydrogen explosion in the PCV, it is filled with nitrogen gas and a flammability control system FCS is installed.

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1 This facility condenses steam in the RPV and returns the condensed water to the RPV by natural circulation (driving pumps not needed), when the RPV is isolated due to loss of external power supplies, for example, (when the main condenser cannot work to cool the reactor). The IC cools steam that is led to a heat transfer tube with water stored in the condenser (in the shell side).

2 This system cools the reactor core when the RPV is isolated from the condensate system due to loss of external power supplies, for example. It can use water either in the condensate storage tank or in the suppression chamber. The turbine that uses part of the reactor steam drives the pump of this system.

(3) Measures against severe accidents

1) Basis of measures against severe accidents

a. Consideration of measures against severe accidents

Severe accidents [3] has drawn attention since "The Reactor Safety Study" (WASH-1400)[IV2-5], which assessed the safety of nuclear power stations by a probabilistic method, was published in the United States in 1975.

Severe accidents, which are beyond design basis events on which nuclear facilities are designed, are considered to be at defense depth level 4 in multiple protection as described in IAEA's Basic Safety Principles for Nuclear Power Plants, 75-INSAG-3, Rev.1, INSAG-12 (1999)[IV2-6]. Multiple protection generally refers to a system that comprises multi-layered safety measures through ensuring design margin at each level of defense, and these levels include: preventing occurrence of abnormalities (level 1); preventing progression of abnormalities into accidents (level 2); and mitigating impact of accidents (level 3). The design basis events are usually for setting safety measures up to level 3. Measures against severe accidents belong to actions at level 4, and they provide additional means to prevent events from progression into severe accidents and mitigate impacts of severe accidents, and also provide measures effectively using existing facilities or based on procedures. They are stipulated as actions to control severe accidents or actions to protect the function of confining radioactive materials to prevent events from worsening.

In Japan, following the 1986 Chernobyl accident in the former Soviet Union, the NSC in Japan set up the Round-table Conference for Common Problems under its Special Committee on Safety Standards of Reactors in July 1987 to study measures against severe accidents. The Round-table Conference members did research on the definition of severe accidents, PSA methods, and maintaining the functions of the PCV after a severe accident, and they put together the "Report on Study of Accident Management as a Measure against Severe Accidents—Focused on the PCV"[IV2-7] in March 1992.

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3 These events significantly exceed design basis events causing the system to become incapable of appropriately cooling the reactor core or controlling reactivity by any methods covered by the safety design, and consequently will lead to serious reactor core damage.

This report says, “Nuclear facility safety is secured through safety ensuring activities that deal with design basis events, and the risk of radioactive exposure of the general public in the vicinity is sufficiently low. Even if a severe accident or events that may lead to a severe accident occurred at a nuclear facility, appropriate accident management[4] based on the PSA would reduce the possibility of it becoming a severe accident or mitigate the impact of a severe accident on the general public, further lowering the risk of exposure.”

Following this report, the NSC Japan made a decision called “Accident Management as a Measure against Severe Accidents at Power Generating Light Water Reactors”[IV2-8] (herein after called the “Accident Management Guidelines”) in May 1992. Based on this decision, licensees have taken voluntary actions (not included in regulatory requirements), such as measures to prevent accidents from becoming severe accidents (phase I) and measures to mitigate the impact of severe accidents (phase II).

The (former) Ministry of International Trade and Industry, based on these Accident Management Guidelines, issued the “Implementation of Accident Management”[IV2-9] to request licensees to carry out PSA on each of their light water nuclear power reactor facilities, introduce accident management measures based on PSA, and submit result reports on these actions, the content of which MITI was to confirm.

After that, the Basic Safety Policy Subcommittee of the Nuclear and Industrial Safety Subcommittee studied overall safety regulations in Japan, and it put together a report “Issues on Nuclear Safety Regulations”[IV2-10] in 2010. This report says that based on moves overseas such as introducing severe accident measures as a regulatory requirement in some countries, it is appropriate to consider dealing with safety regulations on severe accidents measures in terms of their position in the regulation system and legislation. In response to this, NISA has been considering how to deal with severe accidents.

b. Utilization of risk information

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4 Appropriate severe management is measures taken to make effective use of not only safety margin allowed in the current design and original functions provided in safety design but also other functions expected to work for safety as well as newly installed components and equipment so that any situation which exceeds design basis events and may cause serious damage to core will not progress to a severe accident, and, even if the situation progresses to a severe accident, its influences will be mitigated.

The NSC Japan started a study of periodic safety reviews[5] (hereinafter referred to as PSR) in order to consider using PSA, and it worked out a basic policy on PSR including implementation of PSA in 1993.

This policy requested implementation of PSA as part of PSR activities to effectively improve the current level of safety even further, because PSA comprehensively and quantitatively assesses and helps get the whole picture of the safety of a nuclear power station by postulating a wide range of abnormal events that may occur at a nuclear power station. As a result, the (former) MITI has requested that licensees implement PSR since 1994, and has reported to the NSC Japan on licensees' assessment results including PSA.

Later in 2003, PSR was included in regulatory requirements as part of the measures for aging management, while PSA was left as voluntary measures taken by licensees. Then it was decided that PSR results would be confirmed by NISA and reports to the NSC Japan were discontinued. Meanwhile, licensees have been taking severe accidents measures using PSA.

In Japan, civil standards on PSA related to internal events are established. For external events, a civil standard on seismic PSA is also established, while study of PSA related to other external events such as flooding has only started.

The Study Group on Use of Risk Information of Nuclear and Industrial Safety Subcommittee studied utilization of risk information to put together "the basic policy of utilization of risk information in nuclear regulation"[IV2-11] in 2005. However, later the activity had been temporarily suspended. In 2010, this study group was resumed, and it has been considering measures for further utilization of risk information.

On the other hand, the safety goals associated with the use of risk information have been being examined by the Special Committee on Safety Goals of the NSC Japan since 2000, and the "Interim Report on Investigation and Examination"[IV2-12] was issued in 2003. In addition, the "Performance Goals of Commercial Light Water

5 It conducts comprehensive re-evaluation of the safety of nuclear power stations approximately once every ten years based on the latest technological knowledge in order to improve the safety of existing nuclear power plants. Specifically, it re-evaluates comprehensive evaluation of operating experience, reflection of the latest technological knowledge, conduction of technical evaluations for aging, and PSA results.

Reactor Facilities: Performance Goals Corresponding to Safety Goal Proposal"[IV2-13] was issued in 2006. However, the use of risk information based on the safety goals has not progressed because the safety goals of Japan have not been determined.

Accordingly, compared to other countries, Japan has not been sufficiently promoting the use of risk information.

c. Examination of total AC power loss and cooling functions, etc.

The following are the status of the severe accidents associated with the current accident.

According to the “Interim Report on the Conference on Common Issues”[IV2-14] issued by the NSC Japan ((the Special Committee on Nuclear Safety Standards of on February 27, 1989, hereinafter referred to as the "Common Issue Interim Report"), accident management during total AC power loss includes efforts such as core cooling by using RCIC powered by direct current (from batteries), recovery of offsite power systems or emergency DGs, bringing in portable diesel generators or batteries, and power interchange between emergency DGs in adjacent plants. The Common Issue Interim Report states that an accident has a high chance of being settled before it results in core damage if preparation has been made for such management.

In addition, if RHR lose its functionality, the inner pressure and temperature of the PCV increase with decrease in the pressure of the reactor. Accordingly, the Common Issue Interim Report additionally states that to prevent the PCV from being damaged, facilities for depressurization of the PCV to vent pressure in order to prevent PCV rupture (hereinafter referred to as “PCV vent”) should be built and that the procedures for the operation of the individual facilities should be prepared.

The accident management guidelines mention alternative coolant injection into the reactor by using a fire extinguishing line and the PCV vent as the Phase I (core damage prevention) accident management of BWR plants. The accident management guidelines also state that PCV vent facilities with a filtering function installed in combination with other measures, such as coolant injection into the PCV, may be an effective measure for Phase II (after core damage) accident management. The accident management guidelines additionally state that coolant injection into the PCV should be included in the Phase I (core damage prevention) and Phase II (after core damage) accident management of BWR plants. In the PSA that is the basis of this guideline, it was concluded that injecting an alternative coolant into the PCV would suppress increases in the temperature and pressure of the atmosphere in the PCV and prevent debris-concrete reaction[7] and melt shell attack[8].

2) Status of preparation for accident management by TEPCO

TEPCO issued the “Report on Accident Management Examination” [IV2-15] in March 1994, and has been preparing for accident management and establishing procedures, education, etc. associated with the application of the accident management based on the report. TEPCO presented the “Report on Preparation for Accident Management”[IV2-16] describing the status of the preparation for accident management to the Ministry of Economy, Trade and Industry in May 2002.

TEPCO has prepared accident management for the reactor shutdown function, coolant injection into reactors and PCVs function, heat removal from PCVs function, and support function for safety functions. The main measures of accident management are shown in Table IV-2-2. In addition, the system structures of accident management facilities of Units 1 to 3 are shown in Figs. IV-2-10 to IV-2-17.

With regard to alternative coolant injection in the Fukushima NPSs, TEPCO has built the following lines for injecting coolant into reactors: lines via condensate water makeup systems from the condensate storage tanks as the water sources; and lines via fire extinguishing systems and condensate water makeup systems from the filtrate tanks as the water sources. TEPCO has also developed “procedures for coolant injection using these lines during accidents (severe accidents)” (hereinafter referred to as “procedures for operation in severe accidents”).

In addition, TEPCO has built a switching facility in Unit 3 for injecting seawater into the reactor via the residual heat removal sea water system (hereinafter referred to as RHRS)

7 When core melt drops down through the bottom of RPV, it causes thermal decomposition of floor concrete as well as erosion with concrete constituents.

8 When core melt drops down through the bottom of RPV, it drops into and spreads over the cavity area at the bottom of RPV. Then debris spreads over the dry well floor through a pedestal opening and causes damage to walls of PCV.

as shown in Fig. IV-2-12 and has developed a procedure for switching operation of the relevant facilities. However, Units 1 and 2 are not provided with the such facility because no seawater lines lead into the reactor buildings of Units 1 and 2.

TEPCO built new vent pipes extending from the S/C and D/W to the stacks from 1999 to 2001 as PCV vent facilities during severe accidents as shown in Figs. IV-2-13 and IV-2-14. These facilities were installed to bypass the standby gas treatment system (hereinafter referred to as SGTS) so that they can vent the PCV when the pressure is high. The facilities are also provided with a rupture disk in order to prevent malfunction.

The procedures for operation in severe accidents define the PCV vent conditions and the PCV vent operation during severe accidents as follows: PCV vent from the S/C (hereinafter referred to as “wet vent”) shall be given priority operation; and when the PCV pressure reaches the maximum operating pressure before core damage, when the pressure is expected to reach about twice as high as the maximum operating pressure after core damage and if RHR is not expected to be recovered, wet vent shall be conducted if the total coolant injection from the external water source is equal to or less than the submergence level of the vent line in the S/C or PCV vent from the D/W (hereinafter referred to as “dry vent”) shall be conducted if the vent line of the S/C is submerged. The procedures for operation in severe accidents specify that the chief of emergency response headquarters shall determine whether PCV vent operation should be conducted after core damage.

For accident management associated with the function of heat removal from the PCV, alternative coolant injection to a PCV spray (D/W and S/C) (hereinafter referred to as the alternative spray function) has also been provided as shown in Figs. IV-2-15 and IV-2-16. PCV sprays (D/W and S/C) are installed to reduce the pressure and temperature generated due to energy released within the PCV if reactor coolant is lost, according to guideline 32 (containment heat removal system) of the Regulatory Guide for Reviewing Safety Design. The procedures for operation in severe accidents specify criteria such as the standard for starting and terminating coolant injection from RHR by using this modified line and the criteria for starting and terminating coolant injection from the condensate water makeup system and the fire extinguishing system.

Power interchange facilities have been installed such that the power supply of the alternating current source for power machinery (6.9 kV) and the low voltage alternating current source (480 V) can be interchanged between adjacent reactor facilities (between Units 1 and 2, between Units 3 and 4, and between Units 5 and 6) as shown in Fig IV-2-17. The procedures for operation in severe accidents specify procedures for the relevant facilities.

In order to recover emergency DGs, the procedures for operation in severe accidents specify procedures for recognition of failures, detection of the location of failures, and recovery work for faulty devices by maintenance workers.

Table IV-2-1 Comparison between Engineering Safety Equipment and Reactor Auxiliary Equipment

Fukushima-Daiichi Nuclear Power Station		Unit 1	Unit 2	Unit 3
Core spray system (CS)	No. of systems	2	2	2
	Flow (T/hr per system)	550	1020	1141
	No. of pumps (per system)	2	1	1
	Pump discharge pressure (kg/cm2g)	20	35.2	35.2
Containment cooling system (CCS)	No. of systems	2	2	2
	Design flow (T/hr per system)	705	2960	2600
	No. of pumps (per system)	2	2	2
	No. of heat exchangers (per system)	1	1	1
High pressure coolant injection system (HPCI)	No. of systems	1	1	1
	Flow (T/hr)	682	965	965
	No. of pumps	1	1	1
Low pressure coolant injection system (LPCI)	No. of systems		2	2
	Flow (T/hr per pump)		1750	1820
	No. of pumps (per system)		2	2
Residual heat removal system (RHR)	Pump			
	No. of pumps		4	4
	Flow (t/h)		1750	1820
	Total pump head (m)		128	128
	Seawater pump			
	No. of seawater pumps		4	4
	Flow (m3/h)		978	978
	Total pump head (m)		232	232
	Heat exchanger			
	No. of units		2	2
	Heat transfer capacity (kcal/h)		7.76E+06	7.76E+06
Reactor shut-down cooling system (SHC)	Pump			
	No. of pumps	2		
	Flow (m3/h per unit)	465.5		
	Pump head (m)	45.7		
	Heat exchanger			
	No. of heat exchangers	2		
	Heat exchanging capacity (kcal/h)	3.8E+06		
Reactor core isolation cooling system (RCIC)	Steam turbine			
	No. of steam turbines		1	1
	Reactor pressure (kg/cm2g)		79-10.6	79-10.6
	Output (HP)		500-80	500-80
	Speed of rotation (rpm)		5000-2000	4500-2000
	Pump			
	No. of pumps		1	1
	Flow (t/h)		95	97
	Total pump head (m)		850-160	850-160
	Speed of rotation (rpm)		Variable	Variable
Isolation condenser (IC)	No. of systems	2		
	Effective water retention capacity of the tank (m3 per tank)	106		
	Steam flow (T/hr per tank)	100.6		
Standby gas treatment system (SGTS)	No. of systems	2	2	2
	No. of fans (per system)	1	1	1
	Exhaust capacity (m3/hr per unit)	1870	2700	2700
	Iodine filtration efficiency of the system (%)	≥97	≥99.9	≥99.9
Safety valve	No. of valves	3	3	3
	Total capacity (T/hr)	900	900	900
	Blowout pressure (kg/cm2g)	86.8 (two valves) 87.9 (one valve)	87.2	87.2
	Blowoff area	Drywell	Drywell	Drywell
Main steam safety relief valve	No. of valves	4	8	8
	Total capacity (T/hr)	1090	2900	2900
	Relief valve function	74.2 kg/cm2g (1 valve)	75.9 kg/cm2g (1 valve)	75.9 kg/cm2g (1 valve)
		74.9 kg/cm2g (2 valves)	76.6 kg/cm2g (3 valves)	76.6 kg/cm2g (3 valves)
		75.6 kg/cm2g (1 valve)	77.3 kg/cm2g (4 valves)	77.3 kg/cm2g (4 valves)
	Safety valve function	78.0 kg/cm2g (2 valves)	78.0 kg/cm2g (2 valves)	
		78.7 kg/cm2g (2 valves)	78.7 kg/cm2g (3 valves)	
			79.4 kg/cm2g (3 valves)	
	Blowoff area	Suppression Chamber	Suppression Chamber	Suppression Chamber

Table IV-2-2 Accident Management Measures at Fukushima Daiichi and Daini NPSs

	Fukushima Daiichi			Fukushima Daini
	Unit 1 (BWR-3)	Units 2 to 5 (BWR-4)	Unit 6 (BWR-5)	Units 1 to 4 (BWR-5)
1. Accident Management Associated with Reactor Shutdown Function				
(1) Recirculation Pump Trip (RPT) RPT is a function inducing an automatic trip of the recirculation pump to reduce the reactor power by using an instrumentation and control system that has been installed separate from the emergency reactor shutdown system.	○	○	○	○
(2) Alternative Control Rod Insertion ARI is a function for automatically opening a newly installed valve and inserting control rods to shut down the reactor upon detecting an abnormality by using an instrumentation and control system that has been installed separate from the emergency reactor shutdown system.	○	○	○	○
2. Accident Management Associated with Coolant Injection into Reactor and PCV				
(1) Alternative Means of Coolant Injection In order to effectively utilize the existing condensate water make-up systems, fire extinguishing systems, and PCV cooling systems, the destination of the piping is modified so that coolant injection into reactors is possible from these existing systems via systems such as core spray systems, so that they can be used as alternative means of coolant injection facilities.	○	○	○	○
(2) Automatic Reactor Depressurization (Reactor depressurization is already automatic. Therefore, it should be regarded as improvement in the reliability of ADS.) In the event where only the reactor water level is decreasing due to insufficient high pressure coolant injection during a abnormal transient signals indicating high D/W pressure are not generated, and the automatic depressurization system is not automatically activated in the conventional facilities. Accordingly, the reactor has been modified to be automatically depressurized by using safety relief valves after the occurrence of a signal indicating a low reactor water level, which makes it possible for systems, such as emergency low pressure core cooling systems, to inject coolant into the reactor even in such an event.	—	○	○	○
3. Accident Management Associated with Heat Removal Functions in PCV				
(1) Alternative Heat Removal with D/W coolers and Reactor Coolant Cleanup System D/W coolers and reactor coolant cleanup systems are manually activated to remove heat from PCV. The procedure is defined in the accident operation standard.	○	○	○	○
(2) Recovery of PCV Cooling System (Residual Heat Removal System) Recognition of failures of the PCV cooling system (residual heat removal system), detection of the locations of failures, and recovery work for the failures by maintenance workers are defined in the recovery procedure guidelines as basic procedures.	○	○	○	○
(3) Compressive Strengthening Vent Reactor containment vent lines with strengthened pressure resistance are installed to be directly connected to stacks from inert gas systems without passing through standby gas treatment systems, so that the applicability of depressurization operation as a means of prevention of over-pressurization in the PCV is extended to improve the heat removal function in PCV.	○	○	○	○
4. Accident Management Associated with Support Function for Safety Functions				
(1) Interchange of Power Supplies Power supply capacity is improved by constructing tie lines of low-voltage AC power supplies between adjacent reactor facilities.	○	○	○	○
(2) Recovery of Emergency DGs Recognition of failures of emergency DGs, detection of the location of failures, and recovery work for the failures by maintenance workers are defined in the recovery procedure guidelines as basic procedures.	○	○	○	○
(3) Dedicated Use of Emergency DGs One of the two emergency DGs was commonly used between adjacent Units. However, new emergency DGs have been installed at Units 2, 4, and 5, so that each DG is used for only one Unit.	○	○	○	○

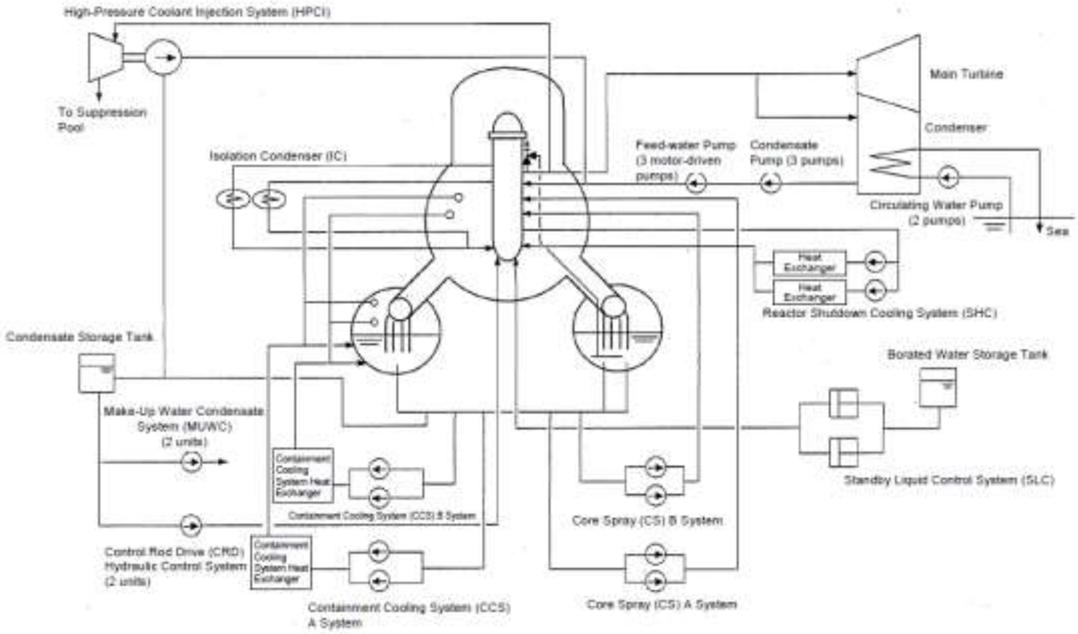


Fig. IV-2-1　System Structure Diagram of Fukushima Daiichi NPS Unit 1

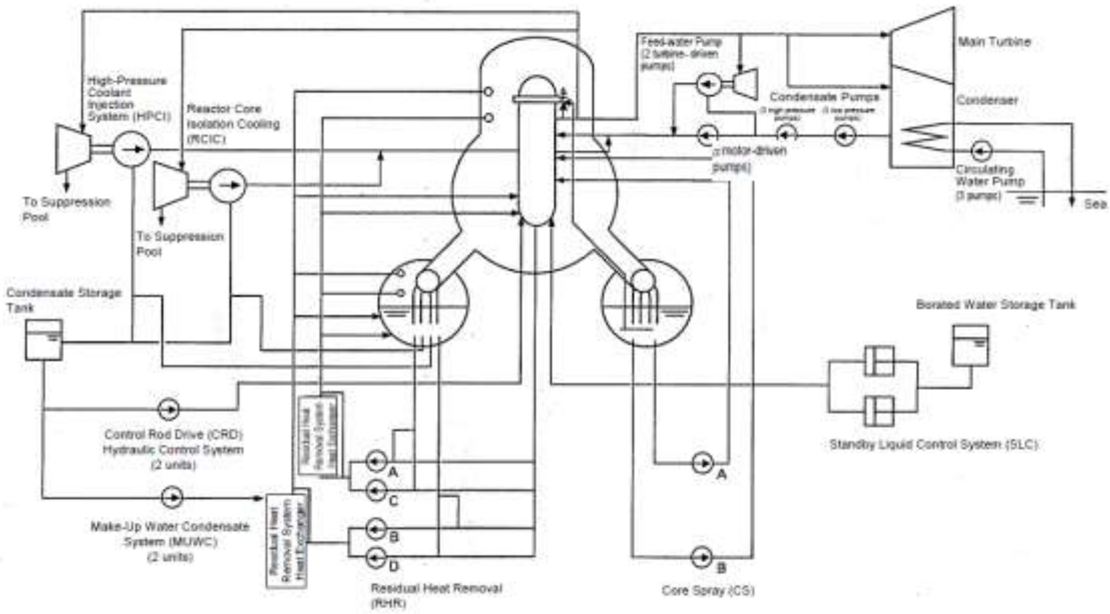
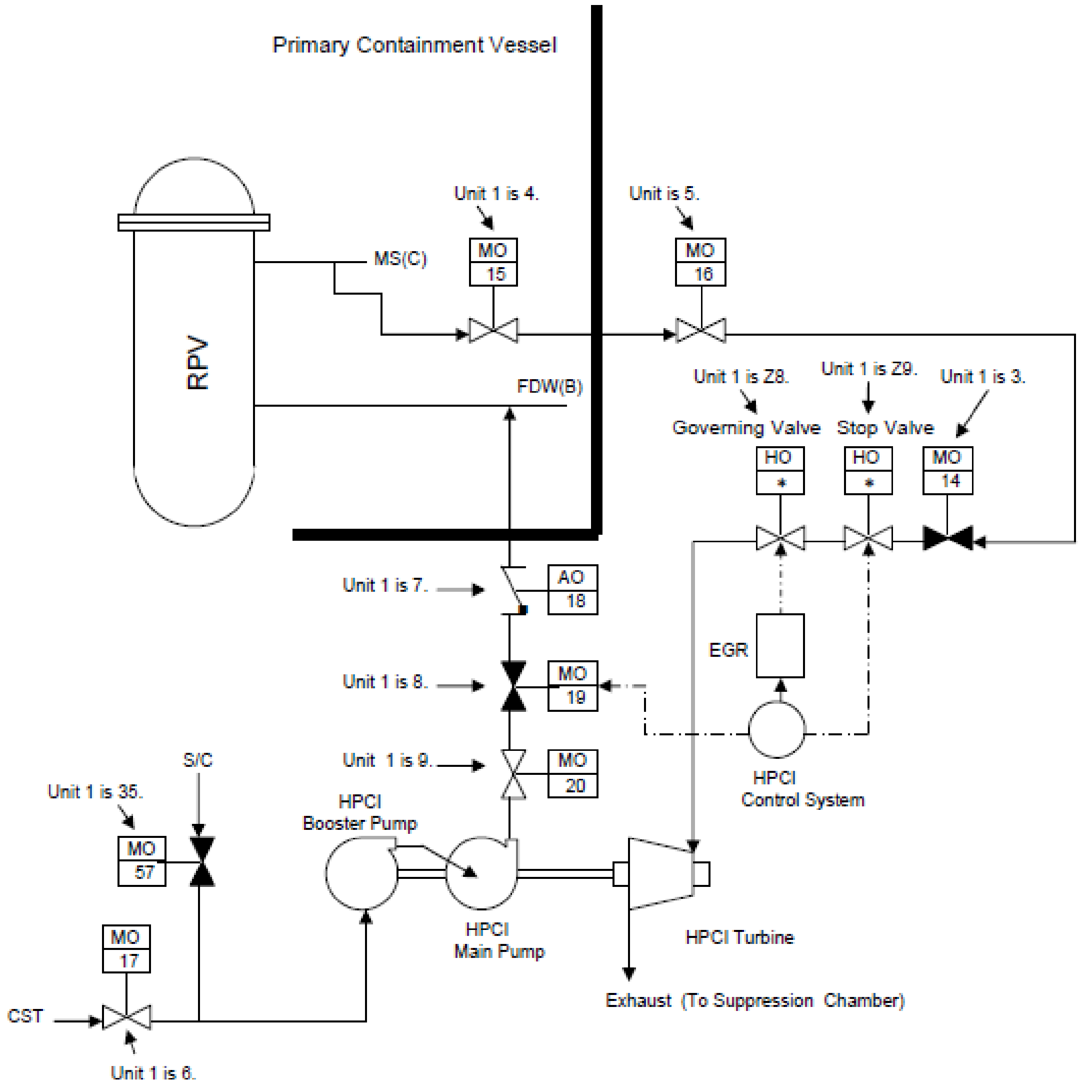
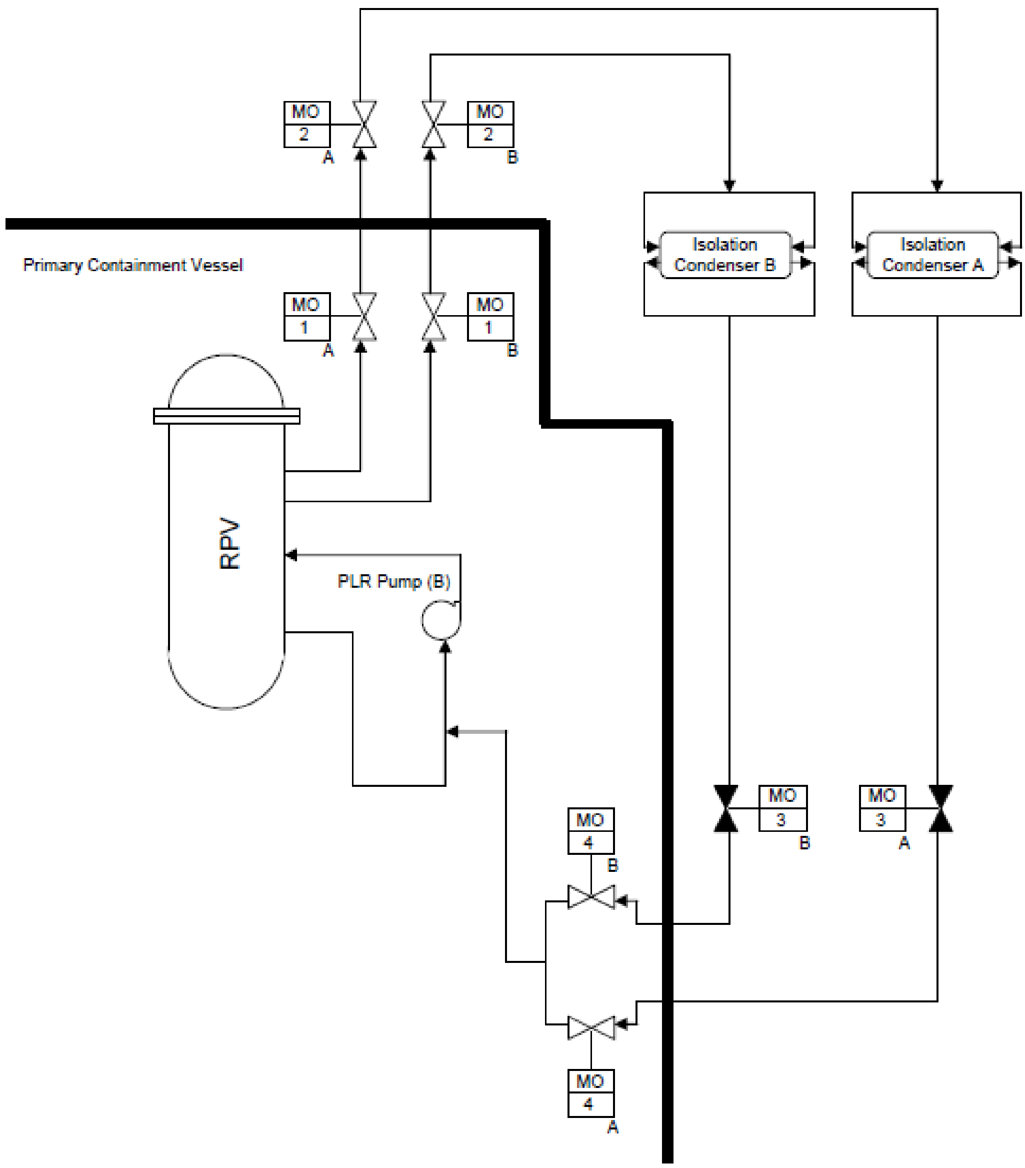


Fig. IV-2-2　System Structure Diagram of Fukushima Daiichi NPS Units 2 and 3



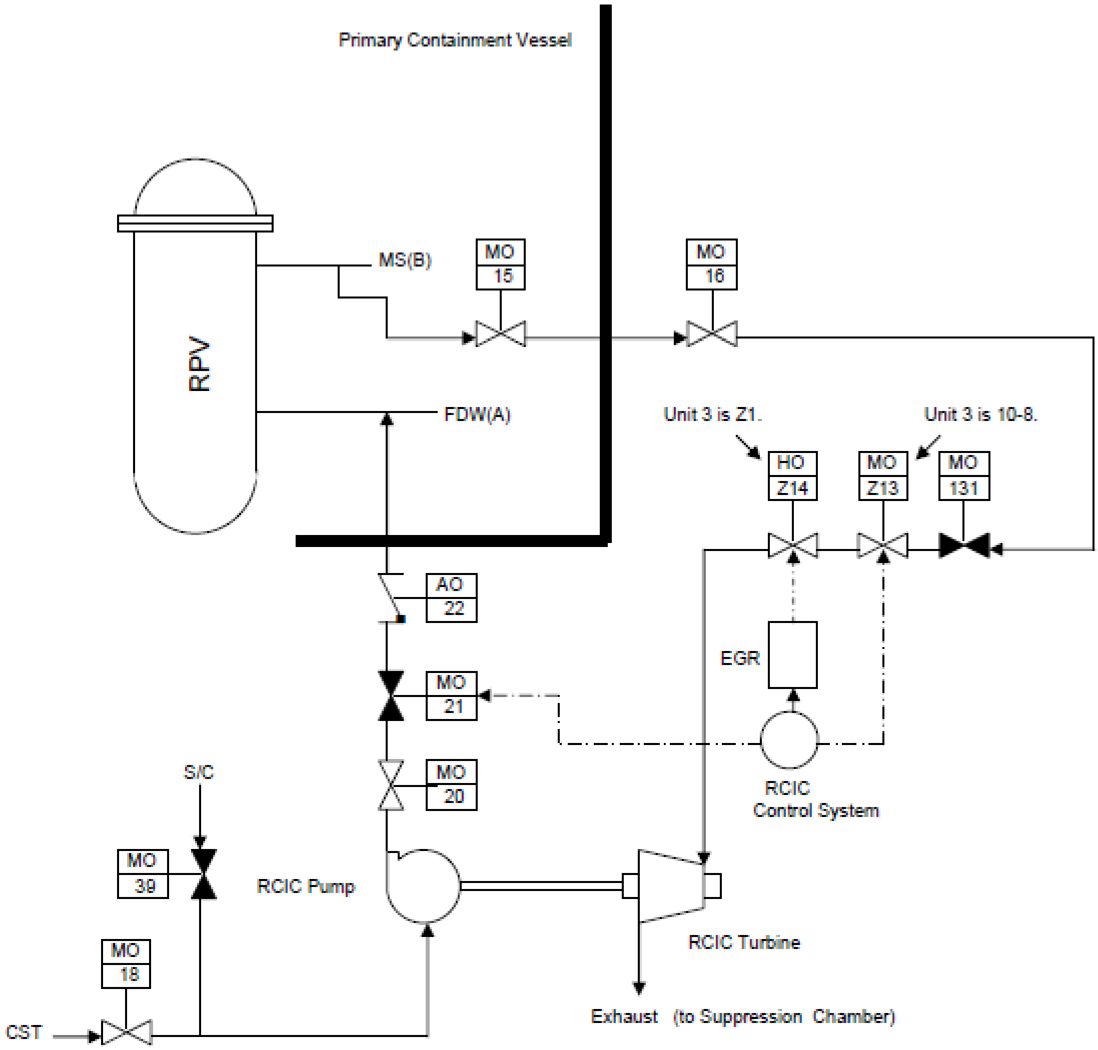
*1: During normal operation, MO-15, 16, 17, 20 and HO valves are "open" and MO-14 and 19 valves are "close".
At startup, 14 and 19 valves are "open".
*2: MO-15 valve is inoperative due to AC power loss. (as-is)
*3: MO-14, 16, 17, 19 and 20 valves are inoperative due to DC power loss (the separate power source from isolation logic circuits). (as-is)
*4: During DC power loss, isolation (close) logic circuits are operative.
At that time, if the drive power of each valve (written in *2 and *3) is activated, each valve is closed. If the drive power of each valve is already lost, the circuits are inoperative. (as-is)

Fig. IV-2-3 System Structure Diagram of High Pressure Coolant Injection System (Units 1 to 3)



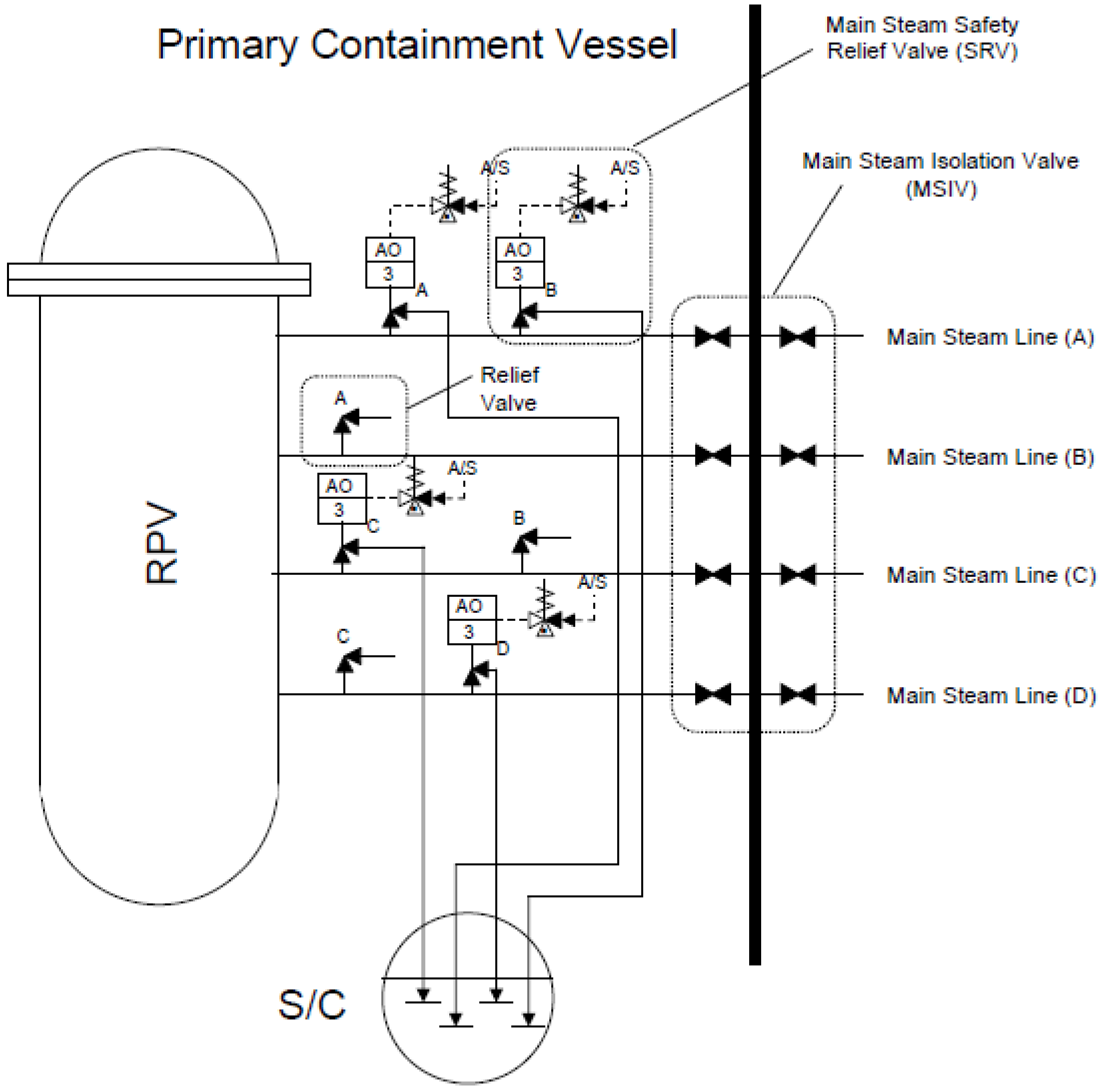
*1: During normal operation (in a standby condition), MO-1, 2 and 4 valves are "open" and MO-3 valve is "close". At startup, MO-3 valve is "open".
*2: MO-1 and 4 valves are inoperative due to AC power loss. (as-is)
*3: MO-2 and 3 valves are inoperative due to DC power loss (the same power source as isolation logic circuits). (as-is)
*4: During DC power loss, isolation (close) logic circuits are operative.
At that time, if the drive power of each valve (written in *2 and *3) is activated, each valve is closed. If the drive power of each valve is lost, the valves are inoperative. (as-is)

Fig. IV-2-4 System Structure Diagram of Isolation Condenser (Unit 1)



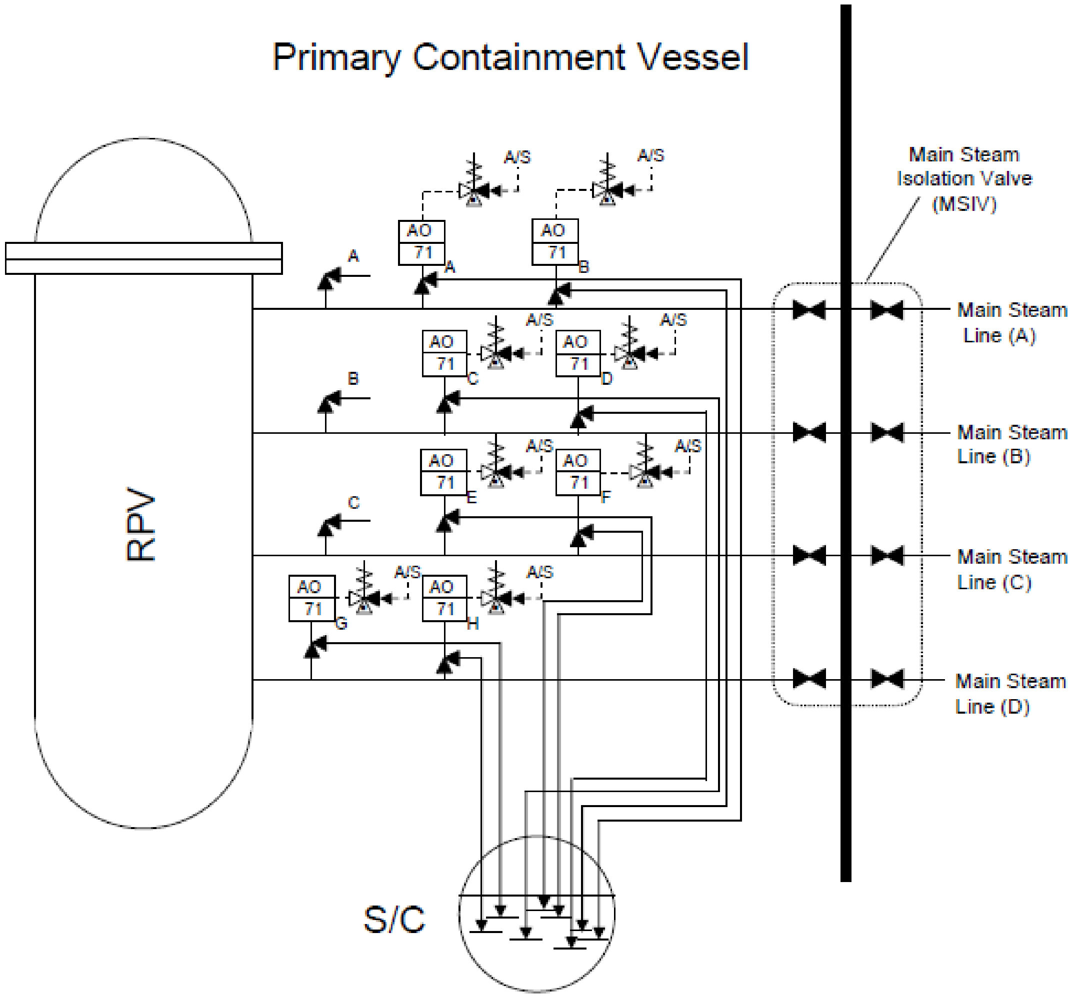
*1: During normal operation (in a standby condition), MO-15, 16, 18, 20 and Z13 valves and HO-Z14 are "open" and MO-131 and 21 valves are "close".
At startup, 131 and 21 valves are "open".
*2: MO-15 valve is inoperative due to AC power loss. (as-is)
*3: MO-16. 18, 20, 21 and 131 valves are inoperative due to DC power loss (the separate power source from isolation logic circuits). (as-is)
*4: During DC power loss, isolation (close) logic circuits are operative.
At that time, the drive power of each valve (written in *2 and *3) is activated, each valve is closed. If the drive power of each valve is already lost, the valves are inoperative. (as-is)

Fig. IV-2-5 System Structure Diagram of Reactor Core Isolation Cooling System (Units 2 and 3)



*1: The main steam safety relief valves (4 valves) are AO valves, and open drive air is supplied by the energized solenoid valves of air supply lines.
During power loss, solenoid valves become deenergized and main steam relief valves are in a closed condition.

Fig. IV-2-6 System Structure Diagram of Main Steam Safety Relief Valve (Unit 1)



*1: Main steam safety relief valves (8 valves) are AO valves, and open drive air is supplied by the energized solenoid valves of air supply lines.
During power loss, solenoid valves become deenergized and main steam relief valves are in a closed condition.

Fig. IV-2-7 System Structure Diagram of Main Steam Safety Relief Valve (Units 2 and 3)

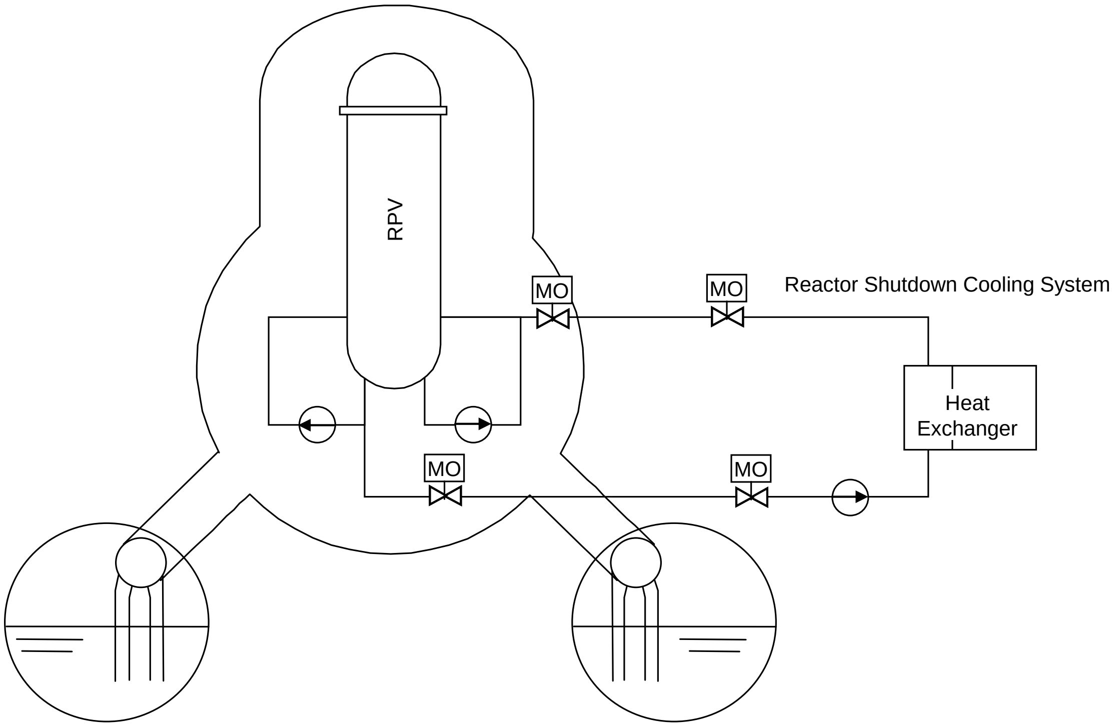


Fig. IV-2-8　System Structure Diagram of Reactor Shutdown Cooling System (Unit 1)

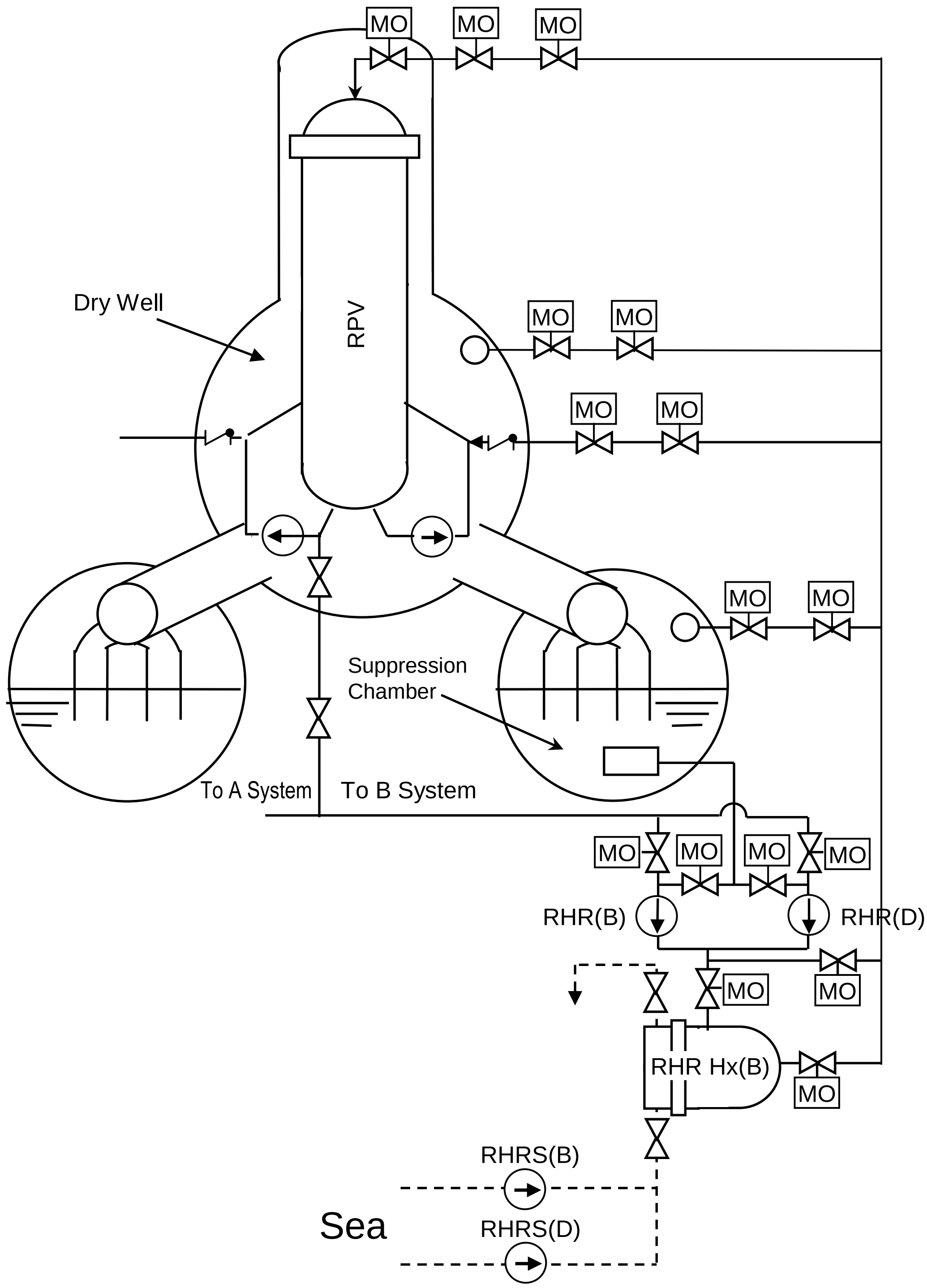


Fig. IV-2-9 System Structure Diagram of Residual Heat Removal System
(Units 2 and 3)

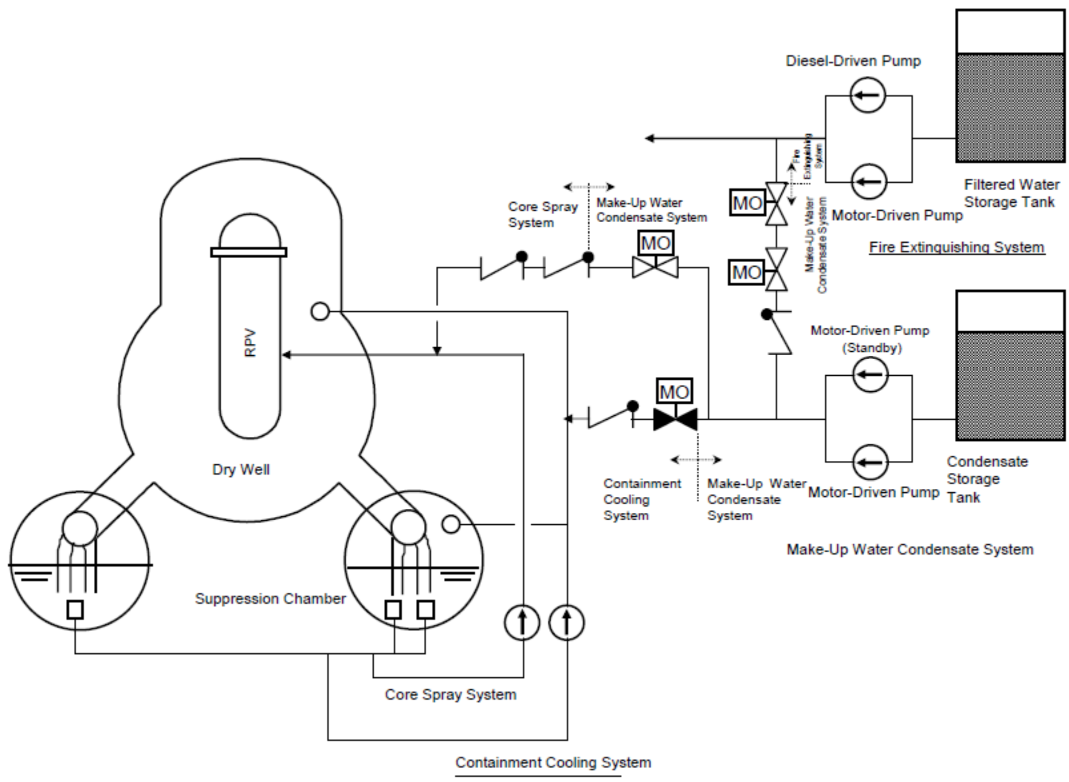


Figure IV-2-10 Overview of the Alternate Water Injection Facility for Unit 1 (by Fresh Water)

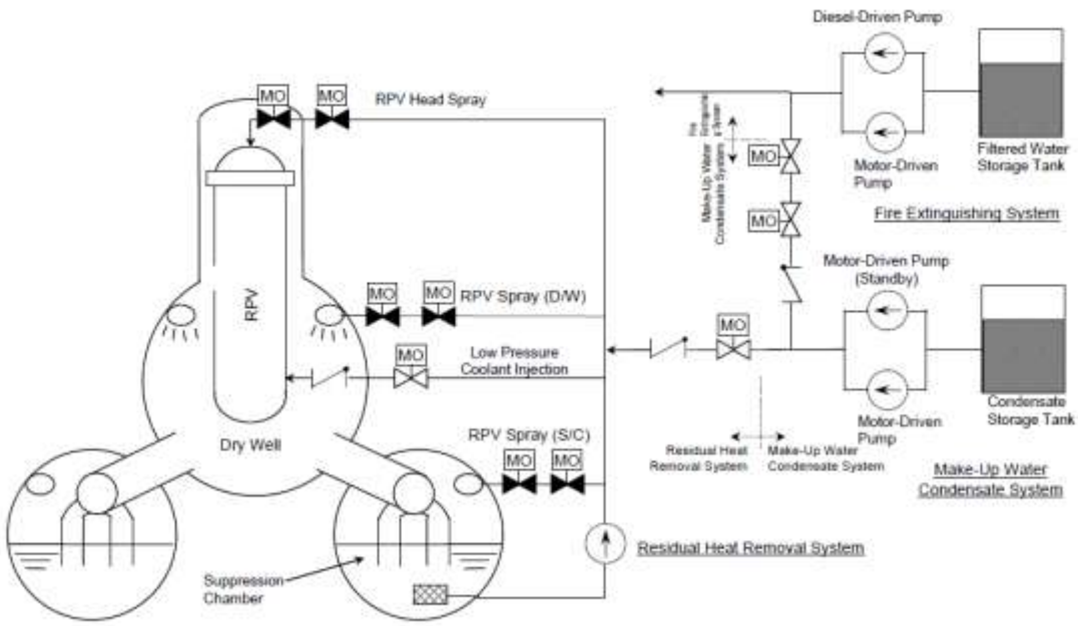


Figure IV-2-11 Overview of the Alternative Water Injection Facility for Units 2 and 3 (by Fresh Water)

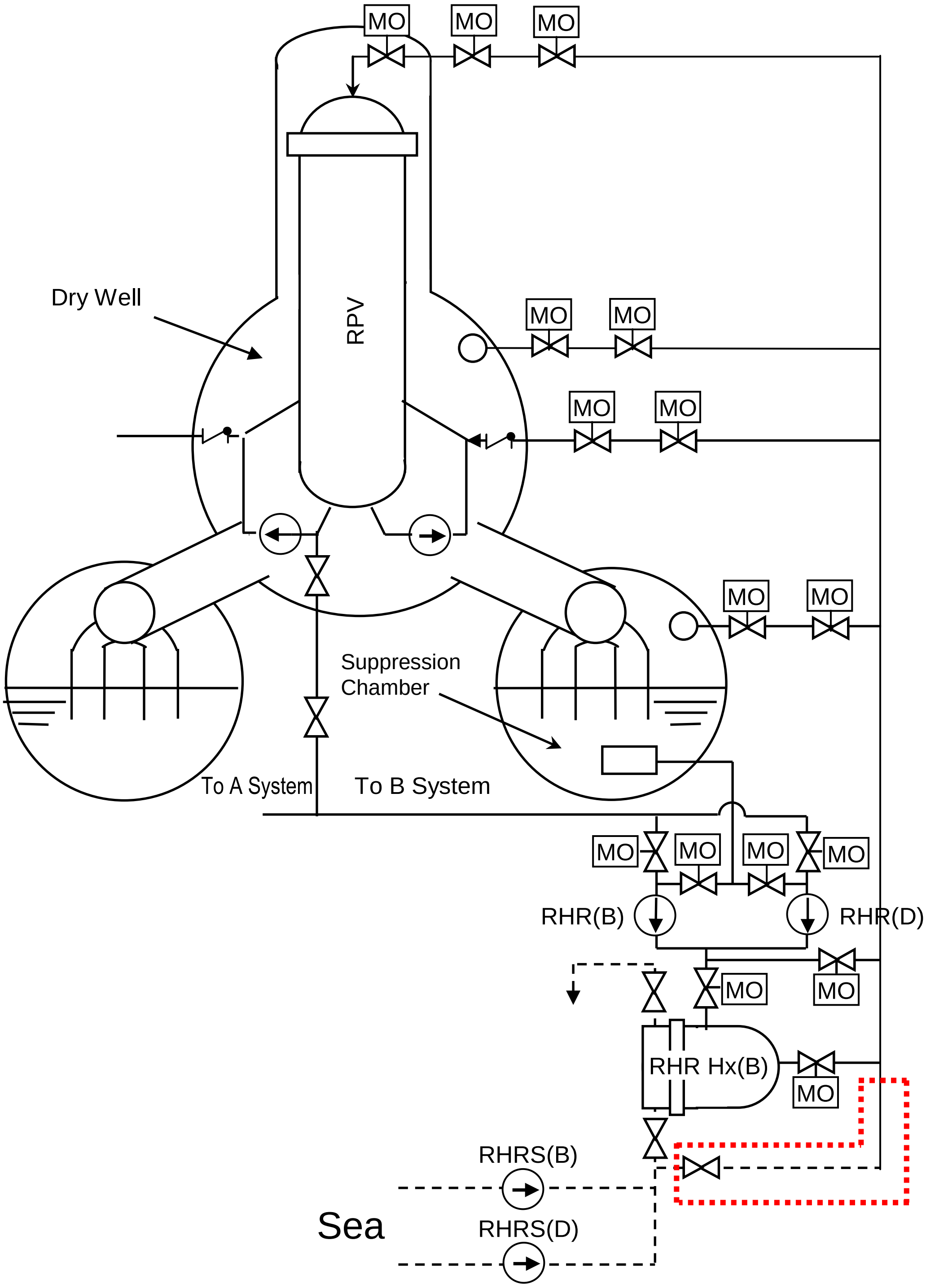


Figure IV-2-12 Overview of the Alternative Water Injection Facility for Unit 3 (by Seawater)

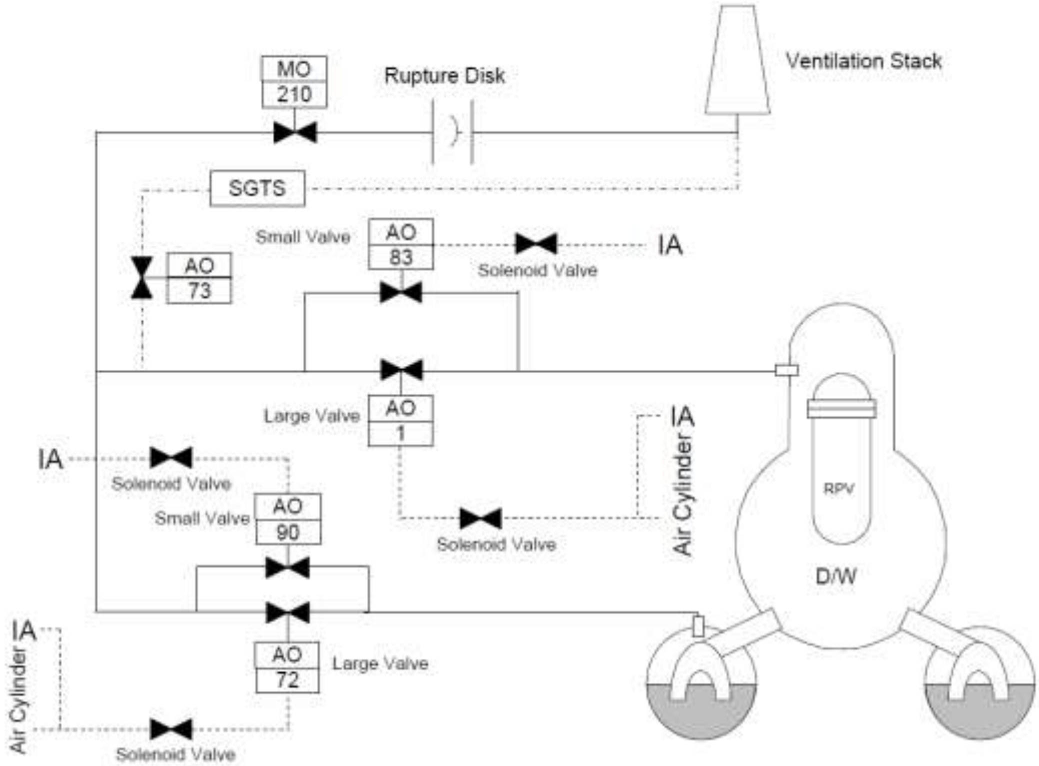


Figure IV-2-13 Overview of PCV Venting Facility (Unit 1)

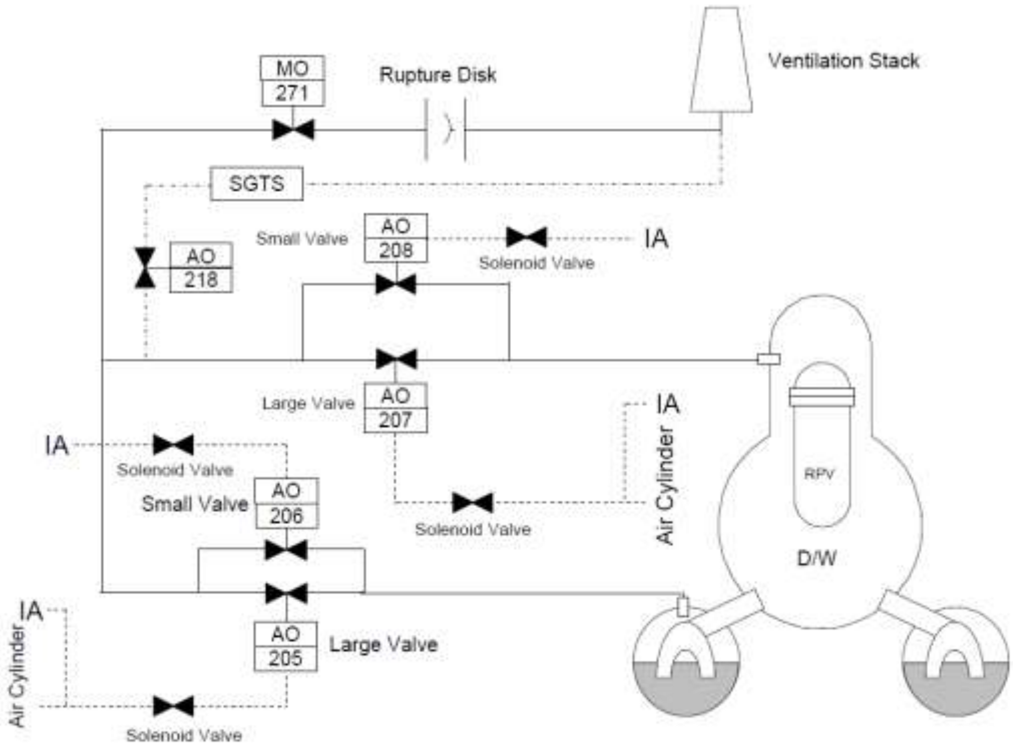


Figure IV-2-14 Overview of PCV Venting Facility (Units 2 and 3)

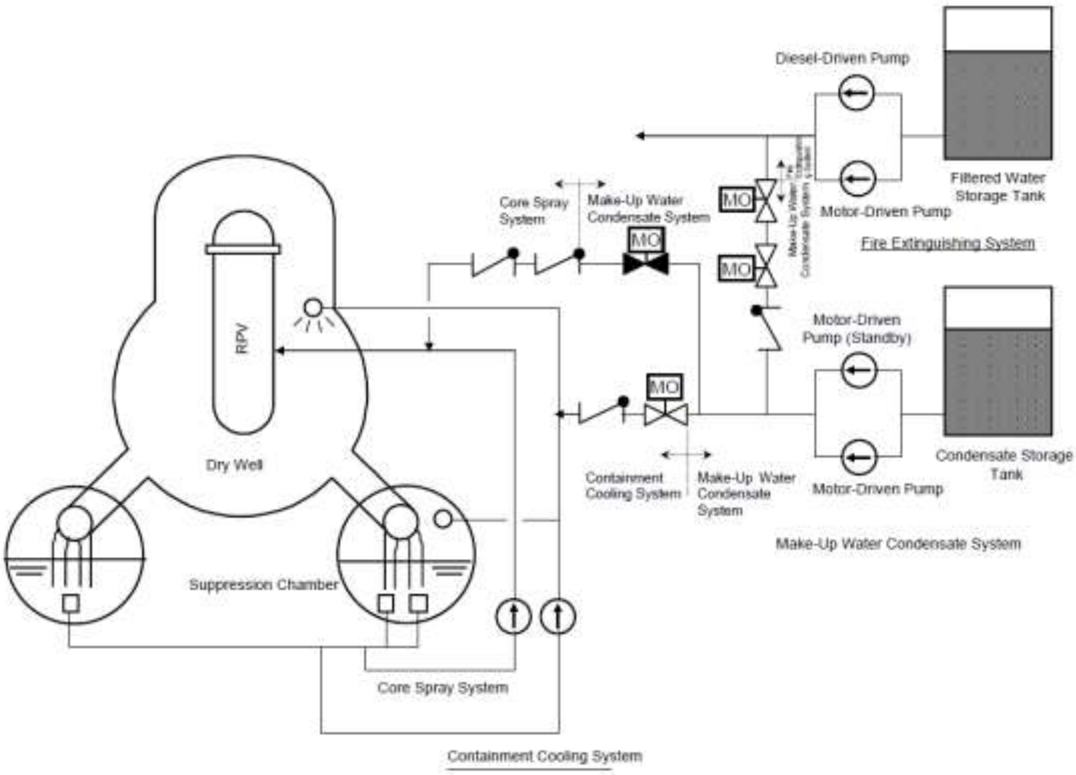


Figure IV-2-15 Overview of PCV Spray (D/W and S/C) Facility (Unit 1)

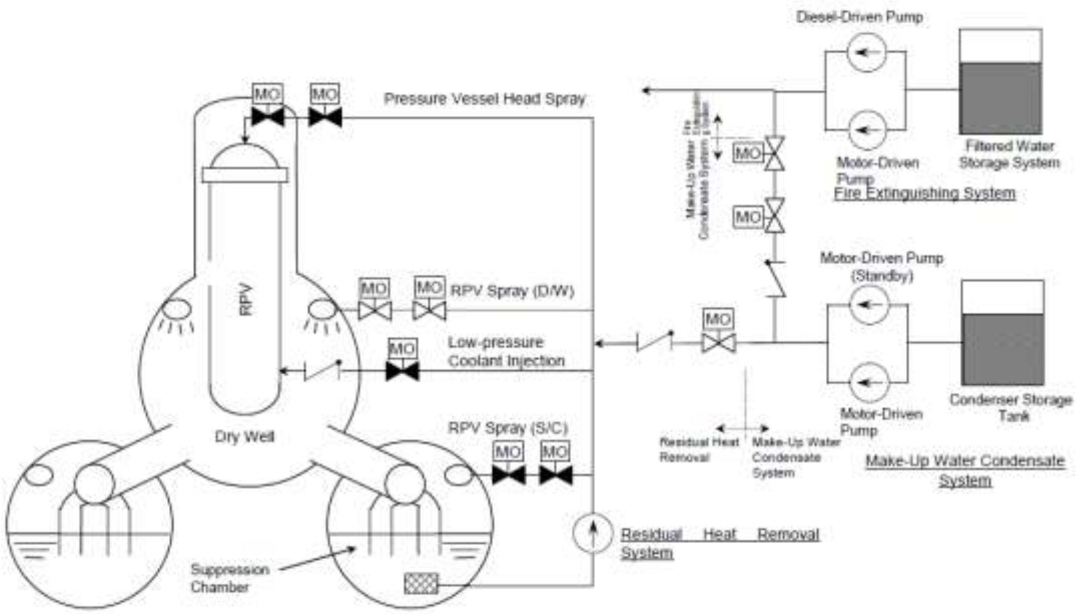


Figure IV-2-16 Overview of PCV Spray (D/W and S/C) Facility (Units 2 and 3)

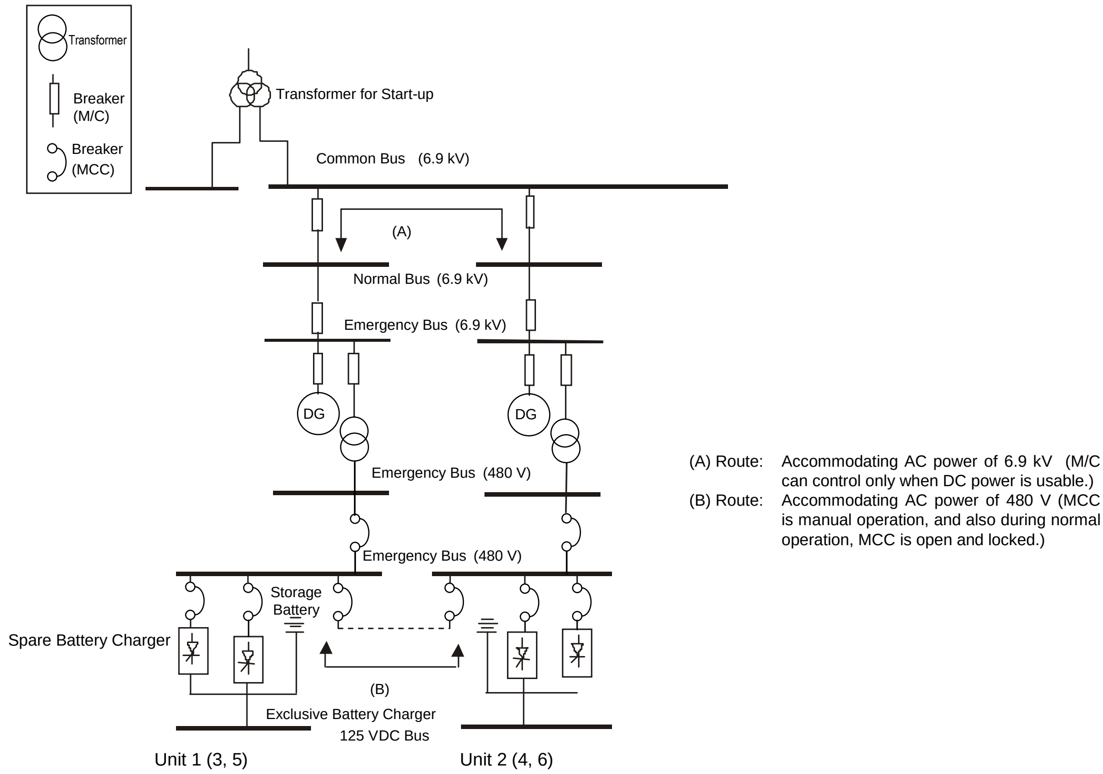


Figure IV-2-17 Conceptual Diagram of Power Supply Interchange among Units

3. Condition of the Fukushima NPSs before the earthquake

(1) Operation

On the day when the earthquake occurred, Unit 1 of the Fukushima Daiichi NPS was in operation at the constant rated electric power, and Units 2 and 3 of the Fukushima Daiichi NPS and all units of the Fukushima Daini NPS were in operation at the constant rated thermal power. The condition of the Fukushima NPSs before the occurrence of the earthquake is indicated in Table IV-3-1.

Fukushima Daiichi NPS Unit 4 was in periodic inspection outage. Large-scale repair work was under way to replace the core shroud, and all fuel assemblies had been transferred to the spent fuel pool from the reactor core with the reactor well filled with water and the pool gate closed.

Fukushima Daiichi NPS Unit 5 was in periodic inspection outage, all fuel assemblies were loaded in the reactor core and the pressure leak test for RPV was being conducted.

Fukushima Daiichi NPS Unit 6 was in periodic inspection outage, and all fuel assemblies were loaded in the reactor core that was in cold shutdown condition.

Table IV-3-1 The Condition of the Fukushima NPSs before the Earthquake

Power stations and reactor units			Condition before the occurrence of the earthquake
Fukushima Daiichi	Unit 1	Reactor	In operation (400 fuel assemblies)
		Spent fuel pool	392 fuel assemblies (including 100 new ones)
	Unit 2	Reactor	In operation (548 fuel assemblies)
		Spent fuel pool	615 fuel assemblies (including 28 new ones)
	Unit 3	Reactor	In operation (548 fuel assemblies, including 32 MOX fuel assemblies)
		Spent fuel pool	566 fuel assemblies (including 52 new ones; no MOX fuel assembly)
	Unit 4	Reactor	Undergoing a periodic inspection (disconnection from the grid on November 29, 2010; all fuel assemblies were removed; the pool gate closed; and the reactor well filled with water)
		Spent fuel pool	1,535 fuel assemblies (including 204 new ones)
	Unit 5	Reactor	Undergoing a periodic inspection (disconnection from the grid on January 2, 2011; RPV pressure tests under way; and the RPV head put in place)
		Spent fuel pool	994 fuel assemblies (including 48 new ones)
	Unit 6	Reactor	Undergoing a periodic inspection (disconnection from the grid on August 13, 2010 and the RPV head put in place)
		Spent fuel pool	940 fuel assemblies (including 64 new ones)
	Common pool		6,375 fuel assemblies (stored in each Unit's pool for 19 months or more)
Fukushima Daini	Unit 1	Reactor	In operation (764 fuel assemblies)
		Spent fuel pool	1,570 fuel assemblies (including 200 new ones)
	Unit 2	Reactor	In operation (764 fuel assemblies)
		Spent fuel pool	1,638 fuel assemblies (including 80 new ones)
	Unit 3	Reactor	In operation (764 fuel assemblies)
		Spent fuel pool	1,596 fuel assemblies (including 184 new ones)
	Unit 4	Reactor	In operation (764 fuel assemblies)
		Spent fuel pool	1,672 fuel assemblies (including 80 new ones)

(2) Connection of offsite power supply

1) Fukushima Daiichi NPS

Connection of an offsite power supply to the NPS were as follows: Okuma Lines No. 1 and No. 2 (275 kV) of the Shin-Fukushima Substation were connected to the switchyard for Units 1 and 2, Okuma Lines No. 3 and No. 4 (275 kV) were connected to the switchyard for Units 3 and 4, and Yonomori Lines No. 1 and No. 2 (66 kV) were connected to the switching yard for Units 5 and 6. In addition, the TEPCO Nuclear Line (66 kV) from Tomioka Substation of the Tohoku Electric Power was connected to Unit 1 as the spare line.

The three regular high voltage switchboards (6.6 kV) are used for Unit 1, for Unit 2, and for Units 3 and 4, respectively. The regular high voltage switchboards for Unit 1 and for Unit 2 were interconnected, and the regular high voltage switchboards for Unit 2 and for Units 3 and 4 were interconnected in a condition that enabled the electricity fed each other. When the earthquake occurred, the switching facilities for Okuma Line No. 3 in the switchyard for Units 3 and 4 were under construction, so that six lines were available for power of the NPS from offsite power supply.

2) Fukushima Daini NPS

A total of four lines of offsite power supply from the Shin-Fukushima Substation were connected to the Fukushima Daini NPS: Tomioka Lines No. 1 and No. 2 (500 kV) and Iwaido Lines No. 1 and No. 2 (66 kV).

When the earthquake occurred, Iwaido Line No. 1 was under construction, so that three lines were available for power of the NPS from offsite power supply.

4. Occurrence and progression of the accident at the Fukushima NPSs

(1) Overview of the chronology from the occurrence of the accident to the emergency measures taken

1) Fukushima Daiichi NPS

The earthquake which occurred at 14:46 on March 11, 2011 brought all of the Fukushima Daiichi NPS Units 1 through 3, which were in operation, to an automatic shutdown due to the high earthquake acceleration.

Due to the trip of the power generators that followed the automatic shutdown of the reactors, the station power supply was switched to the offsite power supply. As described in Chapter III, the NPS was unable to receive electricity from offsite power transmission lines mainly because some of the steel towers for power transmission outside the NPS site collapsed due to the earthquake. For this reason, the emergency DGs for each Unit were automatically started up to maintain the function for cooling the reactors and the spent fuel pools.

Later, all the emergency DGs except one for Unit 6 stopped because the emergency DGs, seawater systems that cooled the emergency DGs, and metal-clad switchgears were submerged due to the tsunami that followed the earthquake, and the result was that all AC power supply was lost at Units 1 to 5.

At 15:42 on March 11, TEPCO determined that this condition fell under the category of specific initial events defined in Article 10 of the Act on Special Measures Concerning Nuclear Emergency Preparedness (hereinafter referred to as Nuclear Emergency Preparedness Act) and notified the national government, local governments, and other parties concerned.

At 16:36 on the same day, TEPCO found the inability to monitor the water level in the reactors of Units 1 and 2, and determined that the conditions of Unit 1 and 2 fell under the category of an event that is “unable to inject water by the emergency core cooling system” as defined in Article 15 of the Nuclear Emergency Preparedness Act, and at 16:45 on the same day, the company notified NISA and other parties concerned of this information.

TEPCO opened the valve of the IC System A of Unit 1 IC, and in an effort to maintain the functions of the IC, it continued to operate it mainly by injecting fresh water into its shell side. Immediately after the tsunami, TEPCO could not

confirm the operation of the RCIC system of Unit 2, but confirmed about 3:00 on March 12 that it was operating properly. Unit 3 was cooled using its RCIC system, and as a result, the PCV pressure and water levels remained stable.
In order to recover the power supply, TEPCO took emergency measures such as making arrangements for power supply vehicles while working with the government, but its efforts were going rough.

Later, it was confirmed around 23:00 on March 11 that the radiation level in the turbine building of Unit 1 was increasing. In addition, at 0:49 on March 12, TEPCO confirmed that there was a possibility that the PCV pressure of the Unit 1 had exceeded the maximum operating pressure and determined that the event corresponded to the event ‘abnormal increase in the pressure in the primary containment vessel’ as defined in the provisions of Article 15 of the Nuclear Emergency Preparedness Act. For this reason, in accordance with Article 64, Paragraph 3 of the Reactor Regulation Act, the Minister of Economy, Trade and Industry ordered TEPCO to reduce the PCV pressure of Units 1 and 2.

At 5:46 on March 12, the company began alternative water injection (fresh water) for Unit 1 using fire engines. (The conceptual diagram of alternative water injection using fire engines is shown in Figure IV-4-1.) In addition, TEPCO began preparations for PCV venting because the PCV pressure was high, but the work ran into trouble because the radiation level in the reactor building was already high. It was around 14:30 on the same day that a decrease in the PCV pressure level was actually confirmed. Subsequently, at 15:36 on the same day, an explosion was considered as a hydrogen explosion occurred in the upper part of the Unit 1 reactor building.

Meanwhile, the RCIC system of Unit 3 stopped at 11:36 on March 12, but later, the HPCI system was automatically activated, which continued to maintain the water level in the reactor at a certain level. It was confirmed at 2:42 on March 13 that the HPCI system had stopped. After the HPCI system stopped, TEPCO performed wet venting to decrease the PCV pressure, and fire engines began alternative water injection (fresh water) into the reactor around 9:25 on March 13. In addition, PCV venting was performed several times. As the PCV pressure increased, PCV venting was performed several times. As a result, the PCV pressure was decreased. Subsequently, at 11:01 on March 14, an explosion that was considered as a hydrogen explosion occurred in the upper part of the reactor building.

At 13:25 on March 14, TEPCO determined that the RCIC system of Unit 2 had stopped because the reactor water level was decreasing, and began to reduce the

RPV pressure and inject seawater into the reactor using fire-extinguishing system lines. The wet venting line configuration had been completed by 11:00 on March 13, but the PCV pressure exceeded the maximum operating pressure. At 6:00 on March 15, an impulsive sound that could be attributed to a hydrogen explosion was confirmed near the suppression chamber (hereinafter referred to as S/C), and later, the S/C pressure decreased sharply.

The total AC power supply for Unit 4 was also lost due to the earthquake and tsunami, and therefore, the functions of cooling and supplying water to the spent fuel pool were lost. Around 6:00 on March 15, an explosion that was considered as a hydrogen explosion occurred in the reactor building, damaging part of the building severely.

At 22:00 on March 15, in accordance with Article 64, Paragraph 3 of the Reactor Regulation Act, the Minister of Economy, Trade and Industry ordered TEPCO to inject water into the spent fuel pool of Unit 4. On March 20 and 21, fresh water was sprayed into the spent fuel pool of Unit 4. On March 22, a concrete pump truck started to spray seawater onto the pool, followed by the spraying of fresh water instead of seawater, which began on March 30.

On March 17, a Self-Defense Forces helicopter sprayed seawater into the spent fuel pool of Unit 3 from the air. Later, seawater was sprayed into the pool using high-pressure water-cannon trucks of the National Police Agency’s riot police and the Self-Defense Forces, as well as fire engines of the Tokyo Fire Department, Osaka City Fire Bureau, and Kawasaki City Fire Bureau.

Later, the concrete pump truck started to spray seawater into the spent fuel pool of Unit 3 on March 27 and into the spent fuel pool of Unit 1 on March 31.

The total AC power supply for Unit 5 was also lost due to the earthquake and tsunami, resulting in a lost of the ultimate heat sink. As a result, the reactor pressure continued to increase, but TEPCO managed to maintain the water level and pressure by injecting water into the reactor by the reactor shutdown cooling (SHC) mode after the power was supplied from Unit 6. Later, the company activated a temporary seawater pump, bringing the reactor to a cold shutdown condition at 14:30 on March 20.

One of the emergency DGs for Unit 6 had been installed at a relative high location, and as a result, its functions were not lost even when the NPS was hit by the tsunami, but the seawater pump lost all functionality. TEPCO installed a temporary seawater pump while controlling the reactor water level and pressure by injecting water into the reactor and reducing the reactor pressure on a continuous basis. By doing this, the company recovered the cooling functions of the reactor, thus bringing the reactor to a cold shutdown condition at 19:27 on March 20.

After the accident, seawater was used for cooling the reactors and the spent fuel pools for a certain period of time, but the coolant has been switched from seawater to fresh water with consideration given to the influence of salinity.

2) Fukushima Daini NPS

Units 1 through 4 of the Fukushima Daini NPS were all in operation but automatically shutdown due to the earthquake. Even after the occurrence of the earthquake, the power supply needed for the NPS was maintained through one of the three external power transmission lines that had been connected before the disaster. (Incidentally, the restoration work for another line was completed at 13:38 on March 12, enabling the NPS to receive electricity through two external power transmission lines.) Later, the tsunami triggered by the earthquake hit the NPS, making it impossible to maintain reactor cooling functions because the seawater system pumps for Units 1, 2, and 4 could not be operated.

For this reason, at 18:33 on March 11, TEPCO determined that a condition had occurred that fell under the category of events specified in Article 10 of the Nuclear Emergency Preparedness Act and notified the national government, local governments, and other parties concerned of this information. Later, since the temperature of the suppression chamber exceeded 100°C, and the reactor lost its pressure suppression functions, the company determined that an event where "pressure suppression functions are lost" defined in Article 15 of the Nuclear Emergency Preparedness Act had occurred at Unit 1 at 5:22 on March 12, at Unit 2 at 5:32 on the same day, and at Unit 4 at 6:07 on the same day, and notified the Nuclear and Industrial Safety Agency and other parties concerned of this information.

Units 1, 2 and 4 of the Fukushima Daini NPS recovered their cooling functions due to the restoration work that followed the earthquake because the offsite power supply was maintained, and the metal-clad switchgears, DC power supply, and other facilities were not submerged. As a result, Unit 1 was brought to a cold shutdown condition, in which the temperature for reactor coolants goes down below 100°C, at 17:00 on March 14, Unit 2 at 18:00 on the same day, and Unit 4 at 7:15 on March 15. Unit 3 was brought to a cold shutdown condition at 12:15 on March 12 without losing reactor cooling functions and suffering other kinds of damage.

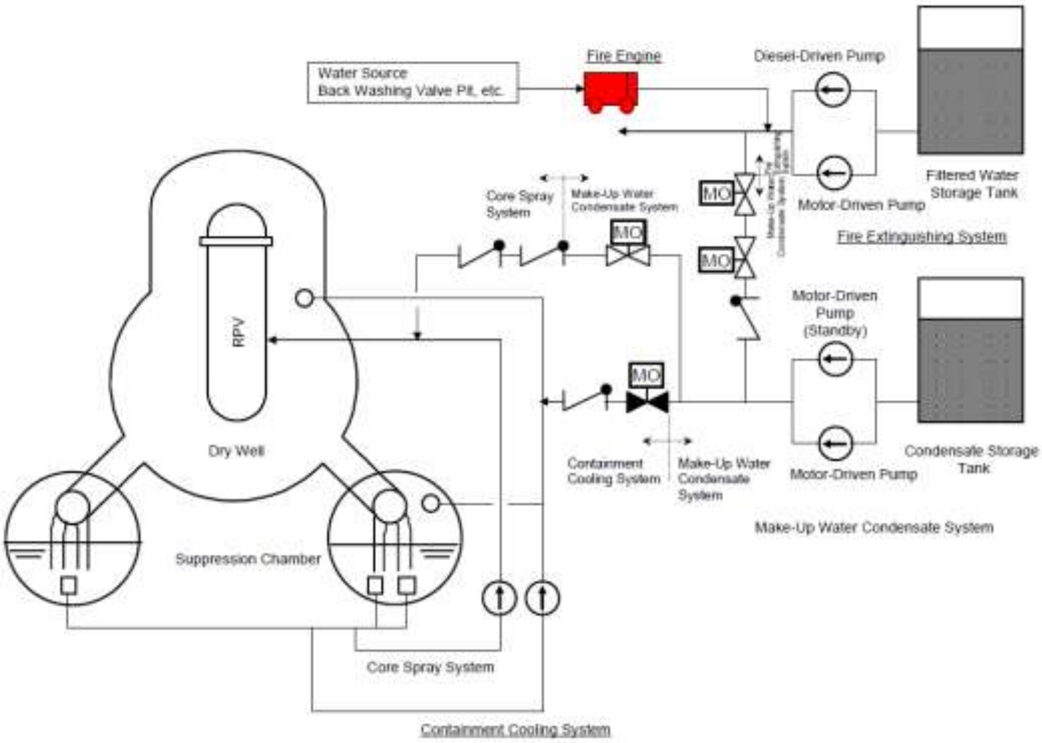


Figure IV-4-1 Conceptual Diagram of Alternative Water Injection Using Fire Engines