Overview of Safety at Fukushima Daiichi Nuclear Power Station

We at TEPCO feel deep remorse that the safety systems that we stated would operate in case of accident in the Fukushima Daiichi establishing permit lost nearly all functionality after the March 11 tsunami due to insufficient planning considerations for the prevention of accidents due to common causes resulting from external events, thus causing a core meltdown accident and subsequent release of large quantities of radioactive materials.

Furthermore, as will be shown below, from this perspective we are determined to reflect on the overall loss of functionality of the safety systems that led to this accident and the safety design based on the establishing permit application and the subsequently submitted AM reports, and to further implement safety measures.

- Measures based on the establishing permit application

(1) Design basis considerations for natural phenomena

In the safety design based on the Fukushima Daiichi Unit 1 establishing permit (approved in 1966), with respect to the facility design of important buildings, structures, and equipment piping and the like of the nuclear power station (NPS) which ought to take natural phenomena into consideration, the foundation of the reactor building is capable of withstanding 600 gals of seismic motion (seismic resistance reinforcement is being carried out), the ground level of the site is 10 meters above sea level and the site could withstand 6.1 meters height of the tsunami (pumps have since been moved to higher ground), but even though the safety functions retained functionality in this earthquake, the tsunami exceeding 15 meters in height could not be foreseen and this is what led to the loss of safety functions.

(2) Safety design

Important systems of the NPS are furnished with redundancy and diversity as well as autonomy and are designed in such a way that integrity can be maintained even when hypothesizing loss of off-site power or malfunction of equipment in any given single system, but as explained above, the safety design assumed singular failures and did not sufficiently take into account natural phenomena, so when the tsunami caused common cause failures it lead to simultaneous loss of safety functions.

In other words, TEPCO's safety design based on the establishing permit application did not include any disaster prevention measures (safety systems layout design) against natural phenomena that exceeded design considerations, and serious thoughts taking into account common cause failures such as long term station blackout (SBO) and the ultimate loss of heat-sink were insufficient.

- Response based on AM report

Also in the AM report (submitted 2002) the safety evaluation only deals with internal events such as human error and singular equipment failure accidents similar to those in the establishing permit application, and there is no safety evaluation for natural phenomena and other external events, and thus evaluations of and consideration of countermeasures against disastrous accidents were insufficient.

Although TEPCO implemented measures based on the establishing permit application and AM reports, up until the accident as expressed above, we were unable to rectify these inadequacies in the design safety.

END

（1）General Structure of Nuclear Power Reactor Facilities

Item	Contents of license for establishing a reactor facility (at 1F) Design policies	Responses to guideline revisions	During the 3.11 earthquake
Earthquake-resistant construction	[Establishment and new construction of nuclear power reactor facilities (licensed between December 1966 and December 1972)] Seismic designs of nuclear power reactor facilities are based on the following policies: ・Principally, a seismic design must ensure a rigid structure. ・Important buildings such as reactor buildings must be directly supported by bedrock. ・Reactor facilities are classified according to their importance as follows and must be provided with a seismic design according to their importance. Class As: Among Class A facilities, Class As facilities correspond to particularly important facilities in terms of safety measures, i.e. facilities such as reactor vessels and control-rod drive mechanisms. Class A: Class A facilities are those whose loss of function can cause a significant accident, i.e. facilities such as reactor building and nuclear power reactor, as well as those that are vital in the prevention of disasters that could affect the public in the surrounding area. Class B: Class B facilities are those handling high-level radioactive materials, such as waste-processing buildings and waste processing facility, excluding the Class As and A facilities described above. Class Ｃ: Class C facilities are those other than of the classes As, A and B. ・Class As and A facilities are designed to ensure safety against a seismic motion with a max. acceleration of 0.18 G at the foundation. ・In addition, Class As facilities must be checked to see whether their functions are ensured even against a seismic motion with a max. acceleration 1.5 times the above-mentioned 0.18 G.	・Atomic Energy Commission developed the Regulatory Guide for Reviewing the Seismic Design of Nuclear Power Reactor Facilities (established in 1978 and revised in 1981). → Our company developed the basic design ground motion S2 (Max. 370 Gal [shallow focus earthquake]) and carried out seismic back-checks.	○Hypocenter: off the coast of the Sanriku District (M9, 178km epicentral distance, and 24km earthquake focal depth) ○Observation records and responses to the basic design ground motion Ss obtained at 1F of each unit are as follows:
	[Establishment of auxiliary shared (common use) facilities, used fuel transportation container buildings, and DG buildings (licensed in March 1994)] These facilities and buildings must have a seismic structure based on the fallowing policies in accordance with the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (1981). ・Principally, these buildings and structures must have a rigid structure. ・Principally, important buildings and structures must be supported by bedrock. ・These facilities and buildings are classified as follows according to their importance in terms of seismic design and must be provided with a seismic design according to their importance. Class A: Class A facilities contain radioactive materials or those directly related to such facilities, and at the same time, those whose functional loss can release radioactive materials outside, as well as those required to prevent such release or to reduce the effect of radioactive materials released Class B: Class B facilities are those having relatively smaller effect or effectiveness than the above-mentioned facilities. Class C: Class C facilities are those other than Class A and B facilities and only required to ensure safety level necessary for common industrial facilities. ・Class A facilities and equipment must be designed to withstand the seismic force (maximum velocity amplitude) of basic design ground motion S1 at the free rock surface of the site. ・Part of Class A facilities and equipment are called Class As facilities and equipment, which must be designed to maintain their safety function such that it can withstand the seismic force (maximum velocity amplitude) of basic design ground motion S2 at the free rock surface of the site.	・Nuclear Safety Commission revised the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (2006 revision). Nuclear and Industrial Safety Agency has instructed our company to carry out seismic back-checks. → Based on the 2006 Regulatory Guide and the lessons learned from the Niigataken Chuetsu-oki Earthquake of 2007, our company has developed the basic design ground motion Ss (Max. 600 Gal [ocean intraplate earthquake]). In parallel with the seismic back-checks, seismic reinforcement is ongoing. We have submitted the following seismic back-check reports (interim reports): - Unit 5 (1F): March 2008 - Units 1, 2, 3, 4 and 6 (1F): June 2009 ※The observation record obtained at 1F was partially interrupted for a period of approximately 130 to 150 seconds. However, maximum acceleration was subsequently obtained from records at the base mat levels of the reactor buildings. We have confirmed that maximum acceleration caused by the 3.11 earthquake had also occurred before such interruption.	

Observation point (at the base mat level of reactor building)		Observation record Max. acceleration (Gal)			Max acceleration (Gal) in response to the basic design ground motion Ss		
		NS direction	EW direction	UD direction	NS direction	EW direction	UD direction
1F	Unit 1	460	447	258	487	489	412
	Unit 2	348	550	302	441	438	420
	Unit 3	322	507	231	449	441	429
	Unit 4	281	319	200	447	445	422
	Unit 5	311	548	256	452	452	427
	Unit 6	298	444	244	445	448	415

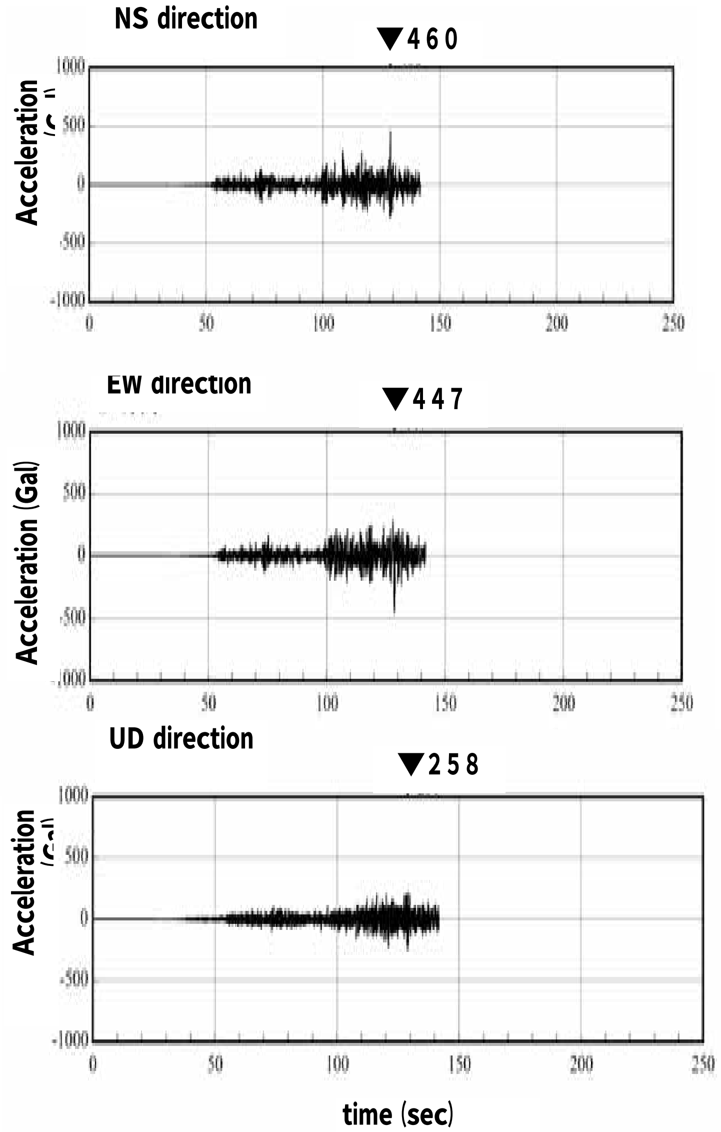

Figure 1. Acceleration time history at the base mat level of Unit 1 reactor building

<table>
<tr><th rowspan="2">Item</th><th>Contents of license for establishing a reactor facility (at 1F)</th><th rowspan="2">Responses to guideline revisions</th><th rowspan="2">During the 3.11 earthquake</th></tr>
<tr><th>Design policies</th></tr>
<tr><td rowspan="3"></td><td rowspan="3">[Unit 6 (1F) - Flammable gas control system (flammability control system) replacement work (licensed in November 2010)]
・Design must conform to the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (2006)
・Facilities important in terms of seismic design must be designed so that they will not lose safety function due to seismic forces caused by seismic motions that should be expected to occur during their service period and that may significantly affect them even if the possibility of such occurrence is minimal in terms of geology, geological structure and earthquake activities at the site area.
・Buildings and structures must be constructed on ground that can adequately withstand design loads specified according to importance in terms of seismic design.
・Facilities and equipment are classified as follows according to their importance in terms of seismic design and
Class S: Class S facilities contain radioactive materials or those directly related to such facilities, and at the same time, those whose functional loss can release radioactive materials outside, as well as those required to prevent such release or to reduce the effect of radioactive materials released outside, and that effectively.
Class B: Class B facilities are those having a comparatively smaller effect or effectiveness than the above-mentioned facilities.
Class C: Class C facilities are those other than Class S and B facilities and are only required to ensure the safety level necessary for common industrial facilities.
・Class S facilities must be designed to maintain safety functions against seismic forces corresponding to basic design ground motion Ss specified for the free rock surface of the site. Moreover, they must be designed to withstand either seismic forces corresponding to elastic design ground motion Sd or static seismic forces, whichever is greater.
・Basic design ground motion Ss shall be determined by identifying (and not identifying) the hypocenter for each site, and is determined as the horizontal and vertical seismic motions at the free rock surface of the site.
・An elastic design ground motion Sd shall be set by multiplying a basic design ground motion Ss by a coefficient (0.5 or higher) obtained through engineering judgment.</td><td></td><td></td></tr>
<tr><td colspan="2">Summary of seismic performance assessment</td></tr>
<tr><td colspan="2">・Records obtained at Unit 2, 3 and 5during the 3.11 earthquake partially exceeded the response to the basic design ground motion Ss. However, we carried out seismic assessments of major facilities and equipment having functions important to safety and confirmed that their safety margins are within the specified tolerances. Therefore, we estimate that, after the 3. 11 earthquake, the safety functions of the major facilities and equipment important to safety were ensured.

・When developing the basic design ground motion Ss for the Fukushima Dai-ichi NPP, we assumed the scale of an interplate earthquake as M 7.9, which was a scale exceeding the estimate regarding off the coast of Fukushima (M 7.4) provided by the government's Earthquake Research Promotion Headquarters. However, <u>the scale of the 3.11 earthquake was M 9, which was caused by co-movements of multiple areas, and therefore our seismic hazard assumption proved to be inadequate.</u></td></tr>
<tr><td rowspan="3">Other structures</td><td rowspan="3">・The finished grade of a site on which a reactor building is constructed is approximately 10 cm above sea level.

(Appended document 6)
・Heights of tide (tidal levels) at the Onahama Port, located about 50 km southward from the site, are as follows:
Highest tidal level: O.P. + 3.122 m (May 24, 1960 Chilean Earthquake tsunami)
Mean monthly highest water level: O.P. + 1.410 m
Mean tidal level: O.P. + 0.824 m
Mean monthly lowest water level: O.P. + 0.075 m
Lowest tidal level: O.P. - 1.918 m (May 24, 1960 Chilean earthquake tsunami)</td><td>・In 2002, the Japan Society of Civil Engineers issued the Tsunami Assessment Method for Nuclear Power Plants in Japan.
→Our company carried out its assessment based on the above-mentioned tsunami assessment method and modified the expected tidal level to O.P. + 5.4m to 5.7m, thereby elevating the height of pumps, developing relevant procedures and enhancing the water-tightness of the buildings.</td><td>○Tsunami inundation height
・Units 1 to 4 (1 F): O.P.＋11.5 to ＋15.5m approximately
・Units 5 and 6 (1F): O.P.＋13 to ＋14.5 m approximately</td></tr>
<tr><td colspan="2">Summary of tsunami assessment</td></tr>
<tr><td colspan="2">・The tidal records to which we referred in determining the appropriate height of the finished grade of the site (1F) were insufficient, and therefore our tsunami design considerations were also insufficient (underestimated).

・Based on subsequent results from research institutes and in accordance with regulatory guide revisions, we have reviewed the tidal level expected at 1F and taken measures such as thte elevation of the height of pumps.

・Although we were aware that a severe accident due to the cliff-edge effect might occur if a tsunami beyond expectations should occur, we did not take measures against tsunami exceeding the site elevation.</td></tr>
</table>

(2) Safety design

System/facility	Function (Segment on guidelines for important degree classification)	1F1 contents description for installation permit: Main specifications	Design policy	Aseismic design class	After March 11th earthquake (○Sound function ×Function lost): Before tsunami attacks — Operation status	After tsunami attacks — Operation status	(Function lost factors)	
Control rod (CR)	・Emergency stop function of the reactor (MS-1) ・Subcriticality maintenance function (MS-1)	97 pcs	These are stainless steel U-shaped sheaths containing a neutron-absorbing material or stainless steel plates containing a neutron-absorbing material, combined in a cross shape; they are arranged uniformly throughout the entire core with an interval of approximately 305 mm from the center of the 4 fuel bundle that make up each group, and they are capable of providing sufficient control of the maximum excess reactivity of the core. They are designed so that even in the event of an accident in which the control rods with the maximum value permitted by the control rod value minimizer drop for some reason, the drop speed will be controlled by the control rod drop speed limiter and the designed limit for the maximum enthalpy of the fuel will not be exceeded through the sudden introduction of reactivity.	A s	○ SCRAM succeeded	○ Maintain the scram status	Power supply Cooling system Main body	－ － ○
Control rod drive facility (CRD)	・Emergency stop function of the reactor (MS-1) ・Subcriticality maintenance function (MS-1)	97 pcs	This is a latched hydraulic main drive piston type and has a structure in which each control rod is set up independently and the drive mechanism and CR cannot be separated easily through coupling. During normal drive operation, this is carried out using drive water that is pressurized by a pump and during scram, it is carried out by drive water that is pressurized by the high-pressure nitrogen in the accumulators of the hydraulic control units that each drive mechanism is provided with.	A s	○ SCRAM succeeded	○ Maintain the scram status	Power supply Cooling system Main body	－ － ○
Liquid toxicant injection system (SLC system)	・Subcriticality maintenance function (MS-1)	1st system 2 units of pumps (1 unit for spare)	Its purpose is to stop the nuclear reactor by applying negative reactivity through the injection of a liquid poison from the base of the core when an inability to insert the control rods makes it impossible to carry out cold shutdown of the reactor, ensuring that cold shutdown of the reactor remains possible even when it is impossible to operate any of the control rods. Remote manual operation is carried out from the central control room. There are two parallel valves in front of the entrance to the reactor pressure vessel, and ensure that injection takes place when necessary.	A s	○ Standby status	× Mutiplexed by 100% capacity of 2 units of pumps, but function was lost by AC power loss	Power supply Cooling system Main body	AC power : X (M/C poured water) － ○
Safety valve and relief safety valve (SRV)	・Function to prevent over-pressurization of Reactor Coolant Pressure Boundary (MS-1) ・Heat removal function after stopping the reactor (MS-1) ・Core cooling function (MS-1) ・Mitigation function for the increase of reactor pressure (MS-2)	3 pcs of safety valves 4 pcs of relief safety valves	The safety valve is set up in the dry well to vent steam, while the relief valve is set up in the suppression chamber for the same purpose. The safety valve operates at set pressure using a screw system. The relief safety valve has a safety valve function (screw type) and a relief valve function (air type), and the relief valve function has a manual and an automatic relief function (ADS).	A	○ Standby status	Safety valve function: ○ Relief valve function: X Relief function of manual (Operating switch)/auto was lost by DC power loss.	Power supply Cooling system Main body	DC power: X (DC facility poured water) － ○
Cooling system in reactor stopping (SHC system)	・Heat removal function after stopping the reactor (MS-1)	2 system (1 unit for spare) Pump 1 unit/system Heat exchanger 1 unit/system	Cools the reactor after shutdown by removing the core decay heat and the water heat that is retained in the reactor pressure vessel, pipes and coolant. The core is cooled by the condenser immediately after shutdown and when the temperature of the reactor water reaches approximately 135°C, it is cooled by the SHC system. The SHC system can cool the reactor water from approximately 135°C to approximately 52°C in 20 hours or less.	A	○ Standby status	× Multiplexed by 100% 2 systems, but function was lost due to AC power loss and cooling system loss.	Power supply Cooling system Main body	AC power: X (M/C poured water) × ○
Emergency condenser (IC system)	・Heat removal function after stopping the reactor (MS-1)	2 system Tank 1 unit/system	This removes the decay heat of the nuclear reactor when the condenser cannot be used, such as when the bypass valve is not operating or the condenser vacuum is decreased when the turbine trips or in situations such as the closure of the main steam isolation valve due to a main steam line break accident. Its conditions of operation are automatic opening of the drain pipe when the pressure of the nuclear reactor is high and that high pressure continues for a certain length of time. When the drain pipe valve opens, the reactor core is cooled automatically by natural circulation resulting from the difference in weight between the steam in the steam pipe and the condensate in the drain pipe. Even if the cooling water in the condenser tank is not refilled, the two tanks are capable of cooling the reactor for 8 hours.	A	○ Automatic startup	× After close operating the outer side valve (3A), heat removal function was lost due to tsunami attacks and power loss.	Power supply Cooling system Main body	DC power : X (DC equipment poured water) AC power : X (M/C poured water) － ○
		1 system	By itself, this works with the reactor core spray system to prevent meltdown of the fuel in the event of a small break in the primary piping of the reactor, such as in the recirculation circuit. The high pressure injection system does not require any external power source.				Power supply	DC power: X (DC equipment poured water)

System/facility	Function (Segment on guidelines for important degree classification)	1F1 contents description for installation permit: Main specifications	Design policy	Aseismic design class	After March 11th earthquake (○Sound function ×Function lost): Before tsunami attacks: Operation status	After tsunami attacks: Operation status	(Function lost factors)	
High pressure coolant injection system (HPCI system)	・Heat removal function after stopping the reactor (MS-1) ・Core cooling function (MS-1)	1 system Vapor turbine 1 unit Pump 1 unit	Its condition for activation is the reactor abnormally low water level signal or the dry well high pressure signal, and injection is stopped by the reactor high water level signal, such as when the water level has recovered. As a backup when the high pressure injection system is not activated, the automatic pressure relief valve is activated by the simultaneous reactor abnormally low water level signal, dry well high pressure signal and high pressure injection system non-operation signal, lowering the pressure of the reactor and activating the core spray system at an early stage.	A	○ Standby status	× Function was lost due to DC power loss	Cooling system	—
							Main body	○
Core spray system (CS system)	・Core cooling function (MS-1)	2 system (1 unit for spare) Pump 2 units/system	This prevents failure of the fuel and cladding due to overheating of the fuel when the core is exposed by a loss of coolant accident, such as the breakage of the recirculation circuit, preventing reactions between zirconium and water that accompany such failure. Its activation is automatic, with the 2 systems being started by the reactor abnormally low water level signal or the dry well high pressure signal. The 4 pumps can be started by the emergency DG even when external power is lost.	A	○ Standby status	× Multiplexed by 100% 2 systems, but function was lost due to AC power loss and cooling system loss.	Power supply	AC power : X (M/C poured water)
							Cooling system	—
							Main body	○
Container (PCV)	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	Design pressure ・D/W (Inner side)4.35kg/cm^{2}g (Outer side)0.14kg/cm^{2}g ・S/C (Inner side)4.35kg/cm^{2}g (Outer side)0.07kg/cm^{2}g Design temperature ・138℃ both D/W and S/C	The pressure suppression containment vessel consists of the dry well, suppression chamber and the related vent pipes, headers and downcomers that surround the reactor pressure vessel and the recirculation circuit. When a loss of coolant accident occurs, the mixture of steam and water that is released into the dry well is led into the pool water in the suppression chamber through the vent pipes. Here, the steam is cooled by the pool water and the increase in the internal pressure of the dry well is controlled by condensation, while the radioactive substances that are released are retained within the containment vessel.	A s	○	× Container pressure exceeds design pressure, containment function was lost.	Power supply	—
							Cooling system	—
							Main body	×
PCV isolation valve	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	2 pcs /piping (PCV In/Out)	This is fundamentally part of the containment vessel, and as a general principle, it is installed according to the following standards. - It is connected to the reactor steam generation system, or 2 isolation valves are set up inside and outside of the dry well in the dry well penetration pipe that is opened in the spaces inside and outside of the dry well. - In the other penetration pipes, those penetration pipes for which there is a risk of external release of radioactive substances due to the rupture of piping within the dry well are provided with at least one isolation valve and these isolation valves are closed automatically by appropriate signals such as reactor low water level, dry well high pressure or high radioactivity, preventing the release of radioactive substances from the containment vessel.	A s	○	○ AC power was lost, but isolation function is maintained.	Power supply	AC power: X (M/C poured water)
							Cooling system	—
							Main body	○
Container cooling system (CCS)	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	2 system (1 unit for spare) Pump 2 units/system Heat exchanger 1 unit/system	This sprays pool water from the suppression chamber within the DW and the suppression chamber after a loss of coolant accident, decreasing temperature and pressure within the containment vessel and suppressing the leakage of airborne radioactive substances. Any 2 of the 4 pumps can eliminate the energy of the coolant release caused by the breakage of the reactor recirculation circuit and the reaction heat and decay heat caused by the zirconium-water reaction that accompanies the total meltdown of the fuel, preventing the DW internal pressure from exceeding the design pressure and temperature. Startup occurs automatically as a result of simultaneous high DW pressure and low reactor water level signals. Even if external power is lost, it is possible for the 2 CCS pumps and 2 CCSW pumps to start up on emergency power.	A s	○ Manual startup (S/C cooling)	× Multiplexed by 100% 2 systems, but function was lost due to AC power loss and cooling system loss.	Power supply	AC power : × (M/C poured water)
							Cooling system	×
							Main body	○
Reactor building (R/B)	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	Reinforced concrete build	This is an airtight building that completely surrounds the containment vessel, and because the SGTS system maintains negative pressure, even if radioactive substances are leaked from the containment vessel, they will not be released directly without passing through the filters surrounding the plant. The lock for the introduction of equipment to the building and the airlock for staff have double doors that are mechanically interlocked and the other passages are sufficiently sealed, meaning that the reactor building is airtight to a high degree.	A	○	× Building damaged due to hydrogen explosion, containment function was lost.	Power supply	—
							Cooling system	—
							Main body	×

System/facility	Function (Segment on guidelines for important degree classification)	1F1 contents description for installation permit: Main specifications	1F1 contents description for installation permit: Design policy	1F1 contents description for installation permit: Aseismic design class	After March 11th earthquake (○Sound function ×Function lost): Before tsunami attacks — Operation status	After tsunami attacks — Operation status	After tsunami attacks — (Function lost factors)
Gas treatment system for emergency (SGTS)	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	2 system (1 unit for spare) Exhauster 1 unit/ system Iodine removal rate 97% or more	When the levels of radioactivity inside of the reactor building increase due to a accident or other cause, the routine ventilation system is automatically shut off and the SGTS starts, maintaining negative pressure within the reactor building and eliminating radioactive material that is leaked from the containment vessel with filters. The filter efficiency is designed so that at least 97% of the iodine and solid fission products in the air that is emitted from the exhaust pipes leading from the reactor building is eliminated. This can be started using the emergency DG even in the event that external power is lost.	A	○ Automatic startup	× Multiplexed by 100% 2 systems, but function was lost due to AC power loss and cooling system loss.	Power supply: AC power: × (M/C poured water) Cooling system: — Main body: ○
Combustible gas concentration Control system (FCS)	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	2 system (1 unit for spare)	Usually, with filling up a container with nitrogen gas according to an atmospheric control system during operation conjointly, In order to make hydrogen or oxygen concentration in the reactor container after a coolant loss accident not reach a limit of inflammability, it designs to maintain less than hydrogen concentration 4vol% or oxygen gas concentration to less than 5vol%. It consists of 2 independent systems, which are supplied with electricity by separate external power sources as well as separate diesel emergency generators for each series.	A	○ Standby status	× Multiplexed by 100% 2 systems, but function was lost due to AC power loss and cooling system loss.	Power supply: AC power: × (M/C poured water) Cooling system: × Main body: ○
Shield equipment (Primary, secondary shield walls)	・Radioactive materials confinement function, radiation shield and release reduction function (MS-1)	Concrete walls Primary shield: RPV and D/W shell Secondary shield: Reactor building side	The primary shielding consists primarily of the concrete walls that surround the reactor pressure vessel and the concrete that surrounds the exterior of the dry well shell, with the thickness of the latter being approximately 1.7-2.0 meters. The secondary shielding consists of the concrete walls of the sides of the reactor building, which also serve as structural materials. Their height is approximately 48 meters above ground level and their thickness ranges from approximately 0.3 to approximately 1.2 meters.	A	○	× Shield function was lost.	Power supply: — Cooling system: — Main body: ×
Reactor protection system (RPS)	・Engineered safety features and generation function of actuation signal to reactor stopping system (MS-1)	—	In the event that a situation that threatens to harm the safety of the reactor occurs or is expected to occur, this has functions that include emergency insertion of the control rods (scramming) in order to control the situation and protect the reactor and power plant, interlocking of withdrawal prevention, sounding of the alarm or closure of the main stem isolation valve and startup of the ECCS. <Scram Conditions> DW high pressure, reactor low water level, reactor high pressure, high neutron flux, MSIV closure, loss of plant power, seismic acceleration high, etc.	A	○ SCRAM succeeded ECCS startup	× Function was lost due to all power loss of DC and AC.	Power supply: DC power: × (DC equipment poured water) AC power: × (M/C poured water) Cooling system: — Main body: ○
Reactor auxiliary cooling system (RCW)	・Particularly important related function on safety (MS-1)	3 system 3 pump units Heat exchanger 3 units	The reactor component is cooled by the reactor component cooling system. Even if radioactive materials are leaked from the reactor component, they are confined within the closed circuit of this system, and because the circuit includes a continuous monitor for radioactivity, leaks can be detected. This system has 1 surge tank, which absorbs expansions and contractions in the volume of the closed loop system and also is the location where injection of the make-up water takes place. During normal operation, the heat exchangers and pumps of 2 systems are in operation and 1 is in reserve. The heat exchangers and pumps of 3 systems are needed during normal startup and shutdown.	A	○	× Function was lost due to AC power loss and cooling system loss.	Power supply: AC power : × (M/C poured water) Cooling system: × Main body: ○
Emergency DG	・Particularly important related function on safety (MS-1)	2 units	Emergency DG works when charges two 6.9 kV buses in the plant and two 480 V buses to shut down the plant safely during power outages of the 275 kV system as well as the 66 kV system. It has enough capacity to operate the auxiliary necessary for shutdown. <Main Loads> CS system, CCS system, SHC system, SLC system, SGTS system, CRD pumps and rectifiers for DC power supply	A	○ Automatic startup due to loss of external power	× Function was lost due to DC power loss and cooling system loss and main body as well as related equipment poured water	Power supply: DC power: × (DC equipment poured water) Cooling system: × Main body: × (Poured water)

System/facility	Function (Segment on guidelines for important degree classification)	1F1 contents description for installation permit			After March 11th earthquake (○Sound function ×Function lost)			
		Main specifications	Design policy	Aseismic design class	Before tsunami attacks Operation status	After tsunami attacks Operation status	(Function lost factors)	
Central control room Ventilation system for emergency (MCR)	・Particularly important related function on safety (MS-1)	1 system	Conditions the air through the recycling method of taking in some external air during normal operation. In the event of an accident, the system is designed to be able to block off the opening to external air and switch to the loop circulation method, where the air goes through a charcoal air filter.	A	○ Standby status	× Function was lost due to AC power loss.	Power supply	AC power: × (M/C poured water)
							Cooling system	—
							Main body	○
Battery	・Particularly important related function on safety (MS-1)	2 sets for house (Floating method) 2 sets for neutron monitor (Floating method)	Storage batteries are equipped for equipment that needs steady power source on a regular basis. Storage batteries for inside the plant are charged by the floating method using two sets of static rectifiers connected to 480 V buses. DC loads include control load, emergency power load, and emergency lighting. Storage batteries for 24 V neutron monitors are used for the reactor protection system instrumentation. <Main Loads> vibration alternating current MMG set, high pressure coolant injection system auxiliary, emergency oil pump, signals, emergency lights, and neutron monitors	A	○	× Function was lost due to battery and power distribution panel being drenched with water. Temporarily it was restored, but thereafter function was lost again.	Power supply	DC power: X (DC equipment poured water)
							Cooling system	—
							Main body	× (Poured water)
RPV boundary (Reactor pressure vessel, reactor recirculation pump, RPV boundary piping)	・Reactor coolant pressure boundary function (PS-1)	1 set	Materials for the RPV boundary are selected so that brittle behavior and rapid propagative fractures do not occur. RPV boundary components are not used in lower-temperature areas, where brittle behavior occurs. During the heating and cooling operations of reactors, proper heating or cooling rates (55℃/h or lower) are set up to control the operations. Changes in the pressure and temperature of the RPV boundary during normal or abnormal operations can be controlled to stay within the allowable range by the functions of the reactor cooling system, engineered safety features, and instrumentation and control system equipment. Leaks from the RPV boundary can be detected early. The boundary can be isolated when an abnormal leak occurs in a piping system by establishing an isolation valve between the boundary and the piping system to which it is connected.	A s	○	× Core was melted, and RPV boundary function was lost due to melt-through.	Power supply	—
							Cooling system	—
							Main body	×
Core Support Structure (CSS)	・Core geometry maintenance function (PS-1)	1 set	Consists of a core shroud, upper core support grid plate, lower core support plate, fuel support plate, and control rod guide tube, supporting the fuel assembly. During normal operations, abnormal operational transients, and accidents, these structures can shut down the reactors safely, at the same time they ensure the cooling of the cores.	A	○	× Water injection to RPV was stopped, water levels reduced, fuel was melted.	Power supply	—
							Cooling system	—
							Main body	×
Main stack	・Function to prevent of the discharge of radioactive materials (MS-2)	1 pce (Common use in 1/2 units) Approx. 70m from reactor Main stack height approx. 120m	Exhaust gas from the main air off-take system first goes through the gas decay tank. Then the activated carbon hold-up device decays and filters the radiation. Exhaust gas from the vapor sealed in the turbine shaft goes through the gas decay pipe to be decayed. These exhaust gases are discharged into the atmosphere from the stack approximately 120 m tall.	A	○	○	Power supply	—
							Cooling system	—
							Main body	○
Containment atmospheric radiation monitor (CAMS)	・Grasp function of plant status in accident (MS-2)	—	(None listed in 1F Installation Permission)	A	○	× Function was lost due to AC power loss, thereafter, it was restored by the temporary power supply.	Power supply	AC power: × (M/C poured water)
							Cooling system	—
							Main body	○
Spent fuel pool (SFP)	・The one not directly connected to reactor coolant pressure boundary, and the function to store radioactive materials (PS-2)	1	The Spent Fuel Pool (SFP) can store all reactor cores, fuel equal to more than one replacement (approximately 225% of a reactor core), and control rods. It can also handle and store activated devices. The wall thickness and depth are designed thick enough to shield radiation. The internal wall is lined with stainless steel to prevent leakage.	A s	○	Pool main body：○ FPC system: x Function was lost due to AC power loss and cooling system loss.	Power supply	AC power: × (M/C poured water)
							Cooling system	×
							Main body	○

System/facility	Function (Segment on guidelines for important degree classification)	1F1 contents description for installation permit: Main specifications	1F1 contents description for installation permit: Design policy	1F1 contents description for installation permit: Aseismic design class	After March 11th earthquake (○Sound function ×Function lost): Before tsunami attacks: Operation status	After tsunami attacks: Operation status	(Function lost factors)	
Fire extinguishing system (FPS)	・Important one on measures in emergency and grasping function of abnormal status (MS-3)	MDFP 1 unit DDFP 1 unit	Branch pipes are taken out of the underground circular piping for fire protection of individual buildings. Hose yards and mobile CO_2 fire extinguishers are installed inside main buildings. Diesel driven extinguishing pumps are also installed as a backup.	C	○	× DDFP pump was able to temporarily start up, thereafter, function was lost due to poured water.	Power supply	DC power: × (DC equipment poured water) AC power : × (M/C poured
							Cooling system	—
							Main body	× (Poured water)
Communications equipment	・Important one on measures in emergency and grasping function of abnormal status (MS-3)	—	(None listed in 1F Installation Permission)	C	○	× Function was lost due to AC power loss.	Power supply	Paging/PHS AC power: × (M/C poured water) Hot line ○
							Cooling system	—
							Main body	○
On site emergency center	・Important one on measures in emergency and grasping function of abnormal status (MS-3)	—	(None listed in 1F Installation Permission)	C	○	○	Power supply	○
							Cooling system	—
							Main body	○
Emergency lighting	・Important one on measures in emergency and grasping function of abnormal status (MS-3)	—	(None listed in 1F Installation Permission)	C	○	× Function was lost due to DC power loss.	Power supply	DC power: × (DC equipment poured water)
							Cooling system	—
							Main body	○
Transmission line	・Power supply (Except for emergency) (PS-3)	Receiving power system 275kV 1 circuit 66kV 1 circuit	During normal times, power is transmitted using 275 kV one circuit to the Shin-Fukushima Power Substation, and 500kV four circuits beyond that substation. When all have power outage, the in-plant power needed to shut down the power plant safely can be supplied from the standby transformer connected to the 66 kV (the Tohoku region system). If the power of the 66 kV circuit is also out, the emergency DG supplies the power needed to shut down the power plant safely.	C	× Damage by earthquake	—	Power supply	—
							Cooling system	—
							Main body	× (Earthquake)
Switch yard	・Power supply (Except for emergency) (PS-3)	275kv circuit	Power generated by the main generator is transmitted to an extra-high-voltage switching station through the main power transformer, and it is then sent out to the Shin-Fukushima Power Substation 10 km away. Although in-plant power is supplied by the main generator through the in-plant power transformer, it can also be supplied by the 275 kV circuit at the extra-high-voltage switching station through a startup transformer.	C	× Damage by earthquake	—	Power supply	—
							Cooling system	—
							Main body	× (Earthquake)
Transformer	・Power supply (Except for emergency) (PS-3)	Main transformer 1 Starting transformer 1 House transformer 1 Spare transformer 1	The main transformer raises the generator voltage (18 kV) to the voltage of th extra-high-voltage switching station (275 kV). The in-plant transformer lowers the generator voltage (18 kV) to the voltage of the in-plant high-voltage bus (6.9 kV). The startup transformer lowers the voltage of the extra-high-voltage switching station (275 kV) to the voltage of the in-plant high-voltage bus (6.9 kV). The standby transformer lowers the 66kV transmission voltage to the voltage of the in-plant high-voltage bus (6.9 kV).	C	× Damage by earthquake	—	Power supply	—
							Cooling system	—
							Main body	× (Earthquake)

(3) Accident Analysis （１Ｆ１ Installation Permit Request, Attachment 10）

･The accident analysis in the installation permit request (attachment 10) is to verify the appropriateness of the design related to equipment critical to safety by confirming that under various postulated accidents as described below, the judgment criteria are satisfied even if a conservative element of a severe single equipment failure is added.

･<u>In the installation permit request and the AM report, the accidents due to failures having the prolonged loss of all electric power, and loss of final heat-sink are not postulated (anticipated), and therefore by design it could not deal with situations arising from the tsunami of 3.11.</u>

Item	Cause	Accident Prevention Measures	Accident Escalation Prevention Measures	Single Failure	Key safety functions considered for mitigating impacts under various accidents.
１．Loss of coolant for nuclear reactor or extreme changes in cooling					
ａ．Loss of coolant to nuclear reactor （ＬＯＣＡ）	The coolant for the reactor will leak out if the RPV boundary structure pipes or devices attached to them are damaged due to some reason while the reactor is operating. If the coolant is not supplied, the cooling ability for the core will decrease. In the worst case, the temperature of the fuel will rise extremely due to decay heat and the fission products may be released from the fuel. Furthermore, combustible gas may be emitted by a zirconium - water reaction and the radiolytic decomposition of water. Moreover, if cooling of a primary containment vessel cannot be performed, primary containment vessel internal pressure and temperature may rise too much.	・ In designing plumbing (pipes and tubes), strict conditions shall be applied giving sufficient considerations to various pressures existing throughout the life cycle of the reactor. ・Adequate quality control shall be implemented in manufacturing processes such as selecting materials, fabricating and plumbing. ・While in use, inspect important areas and verify soundness. ・Plumbing configuring the RPV boundary shall be designed incorporating nonductile fracture prevention. ・The leak detection system shall monitor and detect fractures early before progressing to breakage, and appropriate corrective measures shall be implemented.	・ECCS shall be implemented in order to prevent severe damage to the fuel coating tube, to restrain the Zr-water reaction to a low level, and to remove the decay heat for a long period of time. ・The containment vessel (PCV, R/B) shall be installed in order to contain coolant and radioactive materials released from the pressure container. In PCV, CCS shall be installed in order to prevent exceeding the maximum usage pressure and design temperature, and FCS shall be installed to prevent combustible gasses from reaching the combustion limit. In R/B, SGTS shall be installed so that the load pressure can be maintained even at the time of accident and to perform element removal before releasing into the stack.	・Safety protection system failure (reactor low water level signal SCRAM) ・High pressure coolant injection system failure (small to medium fracture accident) ・System 1 failure in the core spray system (large fracture accident)	･Safety protection system (reactor low water level SCRAM, MSIV closure) ･High pressure coolant injection system (in case of large fracture) ･Automatic depressurization system (in case of small to medium fracture) ･Core spray system（a medium fracture･a large fracture） ･Release the safety valve (safety valve function) ･Emergency in-plant electric power system ･Reactor auxiliary cooling system
ｂ．Loss of flow volume of coolant to nuclear reactor (APTA)	If the two recirculation pumps are tripped for some reason while the reactor is operating, the coolant flow volume to the core will significantly be reduced down to the natural circulation flow volume from the normal flow volume of rated output, and the cooling ability for the core will be reduced.	・Each of two recirculation pumps shall be separately connected to a separate high-voltage bus-line so that a failure of one high voltage bus-line would not disable both pumps simultaneously. ・While in use, inspect important areas and verify soundness.	・Since it is contained by the turbine trip and the reactor SCRAM, there is no danger of the accident expanding.	・Safety protection system failure (turbine main steam stop valve closure SCRAM)	・Safety protection system (turbine main steam stop valve closure SCRAM) ・Release the safety valve (safety valve function) ・Emergency in-plant electric power system ・Reactor auxiliary cooling system

Item		Cause	Accident Prevention Measures	Accident Escalation Prevention Measures	Single Failure	Key safety functions considered for mitigating impacts under various accidents.
	ｃ． Sticking of the shaft of the pump for nuclear reactor coolant	If the rotation shaft of one recirculation pump freezes for some reason while the reactor is operating, the flow volume to the core will be rapidly reduced and the cooling ability for the core will be reduced.	･In designing devices, strict conditions shall be applied giving sufficient considerations to various pressures existing throughout the life cycle of the reactor. ･For PLR-pumps, material selection, manufacturing and installation shall conform to various specifications and standards. Quality control and process control shall be implemented adequately. Especially the bearing shall be designed to withstand any wear-and-tear from a prolonged operation of the recirculation pump, and the probability of the pump shaft to freeze up shall be minimized as much as possible. ･Any anomaly such as an alarm with high bearing temperature, large vibration,in the bearing lubrication system shall be notified to the central control, and the operator shall stop the operation of the recirculation pump to prevent sticking of the shaft. ･While in use, inspect important areas and verify soundness.	・Since it is contained by the turbine trip and the reactor SCRAM, there is no danger of the accident expanding.	・Safety protection system failure (turbine main steam stop valve closure SCRAM)	・Safety protection system (turbine main steam stop valve closure SCRAM) ・Release the safety valve (safety valve function) ・Emergency in-plant electric power system ・Reactor auxiliary cooling system
２． Abnormal injection of reactivity or extreme changes in reactor						
	ａ． Dropping control rod (CRDA)	If the control rod is separated from the control rod drive shaft and falls to the core while the reactor is at or near criticality, the output distribution change of the reactor will arises due to sudden injection of reactivity.	･The connection section between the control rod and the drive shaft shall be configured to have sufficient reliability. It shall be designed such that the control rod will not remain in the core if by any chance they are separated. ･When the reactor is at or near criticality, the movement of the control rod shall be able to be verified by responses from nuclear instrumentation. ･The operation procedures shall be defined to verify that the control rod is moving surely based on responses of nuclear instrumentation especially at the start up or when the control rod is being moved a large amount. ･While the reactor is in operation, try to pull the control rod further out from the completely-pulled-out-position, and verify that it cannot be pulled any further. ･Define the CR pull-out sequence, and draws out according to this sequence.	・It shall be designed with the fall-speed limiter so that the free-fall speed will not exceed 0.95m/s. ・Prevent abnormal pull out by installing RWM and monitoring CR pull-out sequence. ・Automatically close MSIV by the signal indicating high radiation level in the main steam pipe, and minimize releasing of the fission products into the outside of the plant. ・Initiate reactor SCRAM using signals indicating high neutron flux amount, high radiation level in the main steam pipe, etc.	・Conservatively assume that the safety protection system (high neutron flux amount SCRAM)) is inoperative. ・Safety protection system failure （high neutron flux amount SCRAM (APRM))	・Safety protection system (high neutron flux amount SCRAM (APRM), MSIV closure for high radiation in main steam pipe) ・Release the safety valve (safety valve function) ・Emergency in-plant electric power facility ・Reactor auxiliary cooling system

Item	Cause	Accident Prevention Measures	Accident Escalation Prevention Measures	Single Failure	Key safety functions considered for mitigating impacts under various accidents.
３． Anomaly or releasing of radioactive material to environment					
ａ． Damage to Off-Gas System (OGR)	If a part of the off-gas system is damaged for some reason while the reactor is operating, the noble gas contained in the off-gas system may be released to the environment.	・ In designing plumbing (pipes and tubes), strict conditions shall be applied giving sufficient considerations to various pressures existing throughout the life cycle of the reactor. ・Adequate quality control shall be implemented in the manufacturing process such as selecting materials, fabricating and plumbing. ・The entire system shall be designed for near atmospheric pressure. ・In order to make the hydrogen gas and oxygen gas in the exhaust gas that are extracted from the condensate device, to be below the combustion limit, they shall be diluted by the drive steam of the Off-Air-Take system, and shall be combined at the recombining device.	・Detect fractures by monitoring the stack, and take countermeasures such as isolating the Off-Air-Take system. ・In front and behind every major equipment, isolation valves will be installed that can be remotely operated from the central control.	OG isolation valve failure	・Radiation monitor equipment (the stack monitor, etc.) ・ＯＧ isolation valve ・Main stack ・Emergency in-plant electric power facility
ｂ． Main steam line breakage accident (MSLBA)	If the main steam pipe outside of the containment vessel is damaged for some reason while the reactor is operating, the coolant will be leaked from the damaged opening, and radioactive materials may be released to the environment.	・ In designing plumbing (pipes and tubes), strict conditions shall be applied giving sufficient consideration to various pressures existing throughout the life cycle of the reactor. ・Adequate quality control shall be implemented in manufacturing processes such as selecting materials, fabricating and plumbing. ・Fractures shall be detected early before progressing to breakage by monitoring atmospheric temperature within the MS tunnel, and appropriate corrective measures shall be implemented.	･The flow-limiter will be installed at the up-stream side of the DW through-section of the MS pipe in order to limit coolant output flow in case of accident. ･MSIV installed at both sides of the DW through-section of the MS pipe will be automatically closed by signals of MS pipe flow size, the atmosphere temperature quantity in MS pipe tunnel, MS pipe radioactivity quantity, and MS pipe pressure low, etc, and the discharge of the coolant will be restrained.	Safety protection system failure （MSIV closure SCRAM due to main steam pipe large flow signal）	・Safety protection system (MSIV closure for high radiation in main steam pipe, MSIV closure SCRAM) ・Emergency condensate equipment ・Reactor cooling system when shutdown ・Reactor containment vessel ・Main steam isolation valve ・Main steam flow limiter ・Emergency in-plant electric power system ・Reactor auxiliary cooling system
ｃ． Fuel assembly handling accident (FHA)	At the time of fuel exchange of a nuclear reactor, a fuel bundle falls by failure of fuel handling equipment, breakage, etc., it damages, and a radioactive material may be emitted to environment.	･The fuel handling machine shall be designed with strength for adequately handling the total weight of the fuel assembly. ･The fuel handling machine shall be designed with doubled wire. ･The fuel handling machine shall be designed with the fail-safe feature that the fuel assembly cannot be removed when the compressed air is lost. ･When the fuel handling machine is not holding the fuel bundle certainly, interlock whose lifting is impossible is prepared. ･The operation control system shall be implemented so that the fuel handling is carried out under the direct command from the supervisor who is adequately knowledgeable in operation essentials and well trained.	・If an accident happens in the reactor building, the air conditioning and ventilating system will be monitored. SGTS will be automatically started if necessary to reduce releasing of radioactive gasses into the atmosphere.	ＳＧＴＳ system failure	・Safety protection system (SGTS operation for high radiation in the reactor building) ・Reactor building ・Emergency gas processing system ・Main stack ・Emergency in-plant electric power system ・Reactor auxiliary cooling system ・Shielding installation (primary and secondary shielding walls)

Item		Cause	Accident Prevention Measures	Accident Escalation Prevention Measures	Single Failure	Key safety functions considered for mitigating impacts under various accidents.
	ｄ． Loss of coolant for nuclear reactor	The same as 1. a.	The same as 1. a.	The same as 1. a.	ＳＧＴＳ system failure	・Safety protection system (reactor low water level SCRAM, MSIV closure, SGTS operation for high DW pressure or high radiation in reactor building) ・Reactor containment vessel ・Main steam isolation valve ・Emergency gas processing system ・Reactor building ・Containment vessel cooling system ・Main stack ・Emergency in-plant electric power system ・Reactor auxiliary cooling system ・Shielding installation (primary and secondary shielding walls)
	ｃ． Control rod driving accident	The same as 2. a.	The same as 2. a.	The same as 2. a.	MSIV failure	・Safety protection system (high neutron flux amount SCRAM (APRM), MSIV closure for high radiation in main steam pipe) ・Release the safety valve (safety valve function) ・Reactor containment vessel ・Main steam isolation valve ・Main stack ・Emergency in-plant electric power system
４． Abnormal change in pressure or atmosphere of the reactor						
	a. Loss of coolant for the reactor	The same as 1. a.	The same as 1. a.	The same as 1. a.	CCS1 system failure	・Reactor containment vessel ・Core spray system ・Containment vessel cooling system ・Main stack ・Emergency in-plant electric power system ・Reactor auxiliary cooling system
	ｂ. Generation of combustible gas	The same as 1. a.	The same as 1. a.	The same as 1. a.	FCS failure	･Primary containment vessel ･FCS ･Primary containment vessel cooling system ･Emergency in-plant electrical power system

(4) Accident Management Measures

AM measures	Function	1 F1AM report (2002)		After 311 earthquake (○ Functioned well, X Function lost)	
		System and facility	Nature	Operation condition Before the tsunami arrival	Operation condition After the tsunami arrival
Alternative reactivity control	Reactor shutoff function	RPT	The reactor shutoff function is improved by detecting any abnormality (high reactor pressure or low reactor water level) with the use of a measurement and control system that differs from the reactor emergency shutoff system and by automatically activating the recirculation pump trip (RPT) and alternative control rod insertion (ARI).	—	—
		ARI (Scrum air header exhaust valve)		—	—
Alternative water-supply measure	Water-supply function for reactor and containment vessel	Fire-extinguishing system	Water is injected into the reactor and containment vessel by the pumps in the fire-extinguishing system and make-up water (condensate) (MUWC) system.	○	× Although it was possible to activate the DDFP pump temporarily, its function was subsequently lost due to water damage.
		Make-up water system (condensate)		○	Pumps: X Due to external power loss as a normal usage system, it lost its function
		Containment vessel cooling system	Water is injected into the reactor from the containment vessel cooling system through the shutdown cooling system.	○ Manual activation (S/C cooling)	× Although it was multiplexed 100% with two systems, due to the loss of AC power supply and the cooling system, it lost its
		Reactor shut down cooling system		○ Stand-by status	× Although it was multiplexed 100% with two systems, due to loss of AC power supply and the cooling system, it lost its function
Measure for removal of heat from containment vessel	Heat-removal function from containment vessel	gas cooling device (D/W)	Alternatively, heat is removed from the containment vessel through means of the existing facilities (D/W cooler, reactor water clean-up system (CUW)) when it is not possible to remove heat from the vessel.	○ Stand-by status	× Due to the loss of the AC power supply and cooling system, it lost its function.
		Reactor water clean-up system		○ Stand-by status	× Due to the loss of the AC power supply and cooling system, it lost its function.
		Pressure-resistance vent facility	When it is not possible to remove heat from the containment vessel, the containment vessel vent line (with reinforced pressure-resistance) prevents the pressure increase of the containment vessel due to steam.	○	× With the loss of the AC power supply, it lost its function. (then compulsive opening)
		Containment vessel cooling system	When it is not possible to remove heat from the containment vessel by means of the CSS system, the failure of the CCS system is recovered with an allowance for time.	○ Manual starting (S/C cooling)	× Although it was multiplexed 100% with two systems, due to loss of the AC power supply and cooling system, it lost its function. Restoration of failure cannot be
Power supply measure	The support function of a safety feature (Power-supply function)	AC power source	When the supply of power from any emergency DG or other source fails, the power-supply capability is increased by accommodating the AC power supply on 6.9 kV or 480V among the reactor facilities (1F 1~2, 1F3~4, 1F5~6).	○ With the loss of external power supply, DG activated automatically	× Loss of AC power supply (1F1~4)
		Emergency DG	When the power supply from any emergency DG or other source has failed, the failure of the emergency DG is recovered with an allowance for time.	○ With the loss of external power supply, DG activated automatically	× Function was lost due to DC power loss and cooling system loss and main body as well as related equipment poured water. Restoration of failure cannot be performed, either.