The Commission closely investigated the damage caused by the earthquake and tsunami and their effects as well as the development of the accident at the Fukushima Daiichi Nuclear Power Plant, and reviewed and evaluated related issues. We also looked into the risk of accidents at other nuclear power plants hit by the earthquake and tsunami, and through comprehensive study of nuclear power generation, extracted issues and lessons for the future. We also conducted focused analysis and inquiries into some of the unresolved issues regarding the development of the accident at the Fukushima Daiichi Nuclear Power Plant.
2.1 How the accident developed and an overall review

As verified in the previous chapter, the management of TEPCO seems to have been aware that the anti-earthquake measures and measures to prevent flooding from tsunami that were in place at the Fukushima Daiichi Nuclear Power Plant were insufficient. Prior to the accident, measures against severe accidents were, in effect, limited.

The power supply system was especially weak from a defensive perspective, suffering from a lack of redundancy, diversity and independence. Multiple equipment and facilities relating to the plant auxiliary power supply system were in the same location. For Unit 1, both the emergency and normal metal clad switchgears (M/C) and normal power center (P/C) were located on the first floor of the turbine building. All equipment and facilities located upstream and downstream of the power system were located in the same or adjacent locations. All emergency and normal M/C, emergency and normal P/C, emergency diesel generator for Unit 3 were located on the basement floors of adjacent buildings, the turbine building and the control building. There were seven transmission lines that were consolidated into only three transmission towers. Yet, they were configured in such ways that all units would lose off-site power if the transmission function were to fail at the Shin-Fukushima Electrical Substation or the Shin-Iwaki Switchyard of TEPCO, and the Tomioka Electrical Substation of Tohoku Electric Power. The assumption of a normal station blackout (SBO) did not include the loss of DC power, yet this was exactly what occurred.

In the chaos following the destruction wrought by the tsunami, workers were hindered greatly in their response efforts. The problems from the loss of control room functions, lighting and communications, and the struggle to deliver equipment and materials through the debris-strewn and damaged roads in the plant and continuous aftershocks were, all in all, far beyond what the workers had foreseen. The response manuals, with detailed measures against severe accidents, were not up to date, and manuals including that of the isolation condenser (IC) were not sufficiently prepared in advance to cover circumstances such as this accident. Emergency drills and the training of operators and workers had not been sufficiently prioritized. Documents outlining the venting procedures were incomplete. These were all symptom of TEPCO’s institutional problems.

Hydrogen explosions occurred at Units 1, 3 and 4, and it is believed that the containment vessel was damaged in Unit 2. Core damage was avoided in Units 5 and 6, on the other hand. NAIIC discovered that, in reality, an even worse situation could have developed at Units 2 and 3, and the situations at Unit 5 and other nuclear power plants could also have easily worsened by minor incidents. Damage to the spent fuel of Unit 4 could also have occurred, with a worse effect on the surrounding environment. NAIIC found this accident as a mas-
sive accident that could have evolved into one with even greater damage. At the time we are composing this report, the current state of the reactor cores is still unknown, even through the analysis of nuclear reactor parameters. Special attention must be given to the situation at the Fukushima Daiichi plant because the accident is not over.

This accident revealed a number of issues relating to measures against severe accidents that had previously not been seriously considered; this should include redundancy, diversity and independence in measures against a massive disaster, the interaction of multiple units or adjacent nuclear power plants, and preparation against simultaneous multiple accidents.

### 2.1.1 Further understanding of the accident

The following information is important in understanding the nuclear reactor accident at the Fukushima Daiichi plant. It should also enable better comprehension of the study and evaluation of the accident that will be covered in following sections.

1. **Five barriers of nuclear reactor**
   a. Nuclear reactor and nuclear fuel

   A typical one-million megawatt electrical boiling water reactor (BWR) generates and sends approximately 5,600 tons of steam to turbine every hour. The turbine requires an amount of steam equal to emptying the water inside the nuclear reactor pressure vessel in a matter of few minutes. Nuclear fuel is the source of the energy. Approximately 2.2 million cubic meters of liquefied natural gas per year would be needed to equal the same amount of energy that can be generated at a nuclear power plant using only approximately 20 tons of low-enriched uranium. Four fuel assemblies are configured as a cell and loaded within a cylindrical space, which is 3.7 meters high and 4.5 meters in diameter, comprising the “Reactor” of the BWR in a strict definition. It is submerged and placed near the center of the reactor pressure vessel, and therefore often known as the reactor core. A control rod, explained later, is installed at the center of each cell to control nuclear reactions. When reactors are in operation, the reactor core is maintained in criticality. Although the state of criticality is often interpreted as a dangerous condition, criticality of nuclear reactors is a normal state under normal operation, and does not suggest abnormal operation.

   Nuclear fuels are “burned” through the plant operation. The most combusted fuels, which are approximately 25 percent of the entire burned fuels, are discharged and replaced with new fuels after about a year-long cycle of operation the fuels of the entire reactor are recomposed. The BWR plants must be shut down for refueling the reactor for the next reactor operation cycle. Meanwhile, a series of scheduled inspections, maintenance work and plant modification are performed as required. The fuel assemblies are designed in a shape that is suited for removal and installation. A typical BWR reactor contains fresh fuel that is newly installed, as well as fuel in its second cycle, third cycle and fourth cycle. The “most irradiated fuel,” which is removed permanently at the end of the fourth reactor cycle, is called “spent fuel.”

   The kinetic energy of nuclear fission fragments and energy of radiation emitted from such fragments are the main fission energies released from the new fuel. A nuclear fission fragment is a byproduct of the atomic fission of U-235, an uranium isotope artificially enriched. Its kinetic energy is immediately converted into heat energy. On the other hand, most of the radiation energy is released continuously according to different half-lives of fission products, some shorter and others longer, depending on species of radioactive nuclides. Between nuclear energy released with fission and that released as radiation, nuclear energy released in forms of radiation accounts for more than 5 percent of the total heat energy produced during the reactor operation. While nuclear fission stops instantly when the reactor is shut down, radiation and heat continues. This heat is known as decay heat.

   Some of U-238, which is the most common isotope of uranium found in nature, becomes plutonium (Pu-239) by absorbing neutrons that exist in the reactor from the nuclear fission. This plutonium behaves similar to U-235 and releases energy from nuclear fission. The amount of plutonium increases according to the number of cycles the fuel has experienced, and the proportion of energy generated by the plutonium
increases. Through reprocessing of the nuclear fuel, plutonium is extracted from the spent fuel and mixed with uranium (U-238) to create a fuel known as MOX fuel. In terms of the composition of plutonium, MOX fuel is considerably different in composition from normal uranium fuel when the fuel is new, but they become more similar through the operation cycles; the plutonium in the MOX fuel decreases whereas it increases in uranium fuel.

**b. Containment (first and second barriers)**

Uranium fuel is a sintered uranium dioxide powder made into a small cylindrical fuel pellet approximately 1cm tall and 1cm in diameter. On a microscopic level, there are some void spaces between the particles, where the fission products regardless of solid or gas are contained. The first barrier of the “containment” is the void space within the pellet. The pellet is quite dense, more than 95 percent of the theoretical density of uranium oxide, but it does not block volatile elements completely.

The cylindrical pellet is put in a fuel cladding tube, which is the second barrier of the “containment.” The tube is approximately 0.9 millimeters thick and is filled with Helium gas to prevent a significant temperature difference during reactor operation across the gaps between the fuel cladding tube and pellets. Materials for the fuel cladding tube are selected based on due consideration of mechanical, chemical, and nuclear requirements as well as processability (machinability, weldability, etc.). Very few options satisfy all aspects. In its early days in the 1950’s, stainless steel was tested, but later it was deemed not suitable because of the stress corrosion cracking. Today, it is typical to use an alloy that mainly contains zirconium (zirconium alloy, or zircaloy). Zircaloy was not an impeccable solution; it had several disadvantages, including some that will be explained later.

A fuel rod is a fuel cladding tube filled with fuel pellets. The fuel rods are arranged in a square cross-section (8-by-8 or 9-by-9), which are respectively housed in hollow square tubes made of zircaloy called channel box. With the handle on the top, it is collectively known as the fuel assembly. The reactor core is an arrangement of the fuel assembly, and the upper and the bottom parts are kept in a lateral position. The upper part has a grid to store the fuel assemblies in a 2-by-2 formation, and a cross-shaped control rod in the center. The control rod is filled with boron carbide, which is an absorbent of neutrons. These constitute one cell. The reactor core is comprised of a few hundred fuel assemblies and a control rod for every four fuel assemblies.

**c. Containment (third barrier)**

The third barrier is the reactor coolant pressure boundary. The barrier is comprised of a reactor pressure vessel and numerous piping connected to the vessel. The boundary of the barrier therefore extends to the secondary valve of each pipe. The secondary valve needs to be taken into account in addition to the primary valve because designers are required to assume a single failure\(^1\) in principle.\(^2\)

Leakage of water and steam from the pressure boundary may occur only under explicitly specified conditions.

The reactor pressure vessel has a silver plated metallic O-ring on the bolted reactor head. The O-ring seal feature may be undermined if tension is decreased from the bolt creep and relaxation as a result of high temperature exceeding the design bases. The gasket and packing, mainly made of rock wool, are used for bonnet flange and the gland of the valve. Their seal feature would deteriorate in an extremely high temperature environment. The mechanical seal is applied to the shaft of a primary loop recirculating pump. The sealing water and the cooling water are supplied from the off-site systems in normal times to keep its performance. When the supply is discontinued, the water in the coolant system leaks.

When an abnormal increase in the pressure in the reactor pressure vessel is observed, the pressure needs to be reduced actively by operating the main steam safety-relief valve (SRV). Steam released from the SRV is delivered to pool water in the pressure suppression chamber and condensed. The pressure suppression chamber

\[^{1}\text{Single failure means an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure.}\]

\[^{2}\text{The assumption of the principle is a failure of the primary valve.}\]
constitutes the fourth barrier, and will be covered in detail in the later sections.

If an SRV fails and becomes stuck open, the cooling water will be rapidly lost from the reactor coolant pressure boundary. Such a leak is known as a loss-of-coolant-accident, LOCA, and is also caused by pipe ruptures in the same boundary as a result of deterioration such as the stress corrosion crack and the flow acceleration corrosion and external load like earthquakes.

The LOCA is an extremely serious state for a reactor and no time can be lost depending on the scale. Cooling water needs to be injected before the reactor core becomes exposed and fuel damage begins. The emergency core cooling system, the ECCS, is an extremely important system, and must be designed with redundancy and diversity. Because the system is driven mostly by electrical power, redundancy and diversity are also required in the power supply. If a massive earthquake causes pipes to rupture, the possibility of losing off-site power must be considered. The plant needs to have an on-site emergency power source system, which usually is a diesel generator. When the LOCA occurs, it takes approximately 30 seconds for the makeup to start after activating a diesel power generator and the ECCS. It takes approximately 5 minutes to fully cover the top of the reactor core once it becomes exposed according to the design basis. In light of this, it is obvious that LOCA is a threat and the ECCS is an important feature to keep the reactor core under control.

d. Containment (fourth barrier)
The containment vessel is the fourth barrier (sometimes referred to as the “primary containment vessel” for reasons that will be explained later). Minor leakage is allowable through the containment vessel when the leak test verifies the leak rate is acceptable. If high temperatures and high pressure water/steam gushes out due to LOCA and in turn breaches the containment vessel, it can no longer function as a barrier. Therefore, the design of the containment vessel should be based on the temperature and pressure under an extreme case of the LOCA (represented by double-ended guillotine break of the largest diameter pipe in the containment vessel).[3] Such a “worst case” scenario postulated in the design basis assumes that the core becomes submerged as a result of automatically activating ECCS before the core damage begins, but after 1/3 of the reactor core once becomes uncovered. Logically, there is even a more severe situation where the reactor core cannot be reflooded to the top, such as a breach of the reactor pressure vessel and the LOCA of long duration resulting from events such as an SBO. If the LOCA remains uncontrolled, the nuclear accident escalates and the situation is exacerbated. The design of the containment vessel is based on a grand assumption that core damage would not occur. However, under certain conditions beyond design basis, damage is inevitable.

Inside the BWR reactor building is a containment suppression chamber, which stores a large body of cooling water used to suppress rising temperatures and pressure during the LOCA by condensing the high temperature steam filling the containment vessel. It allows the size of the BWR containment to be compact. The space where the reactor vessel and other equipment belonging to the third barrier are placed is the drywell, whereas the space with the pool storing a large body of water, namely the containment suppression pool mentioned above, is also known as the “wet well.”

If the ECCS is not started promptly after the reactor core becomes uncovered due to the LOCA, the nuclear fuel will be damaged. Zircaloy used in the fuel cladding and the channel box becomes a problem. Zirconium-water reaction \((\text{Zr} + 2\text{H}_2\text{O} \rightarrow \text{ZrO}_2 + 2\text{H}_2)\) rapidly progresses in a high temperature steam atmosphere—1000 degrees (Celsius) or above—to release hydrogen gas. The reactor stores abundant zircaloy, an ingredient of the exothermic reaction. The reaction is self-accelerating. Hydrogen leaks from a breach because of the LOCA and fulfills the containment vessel. Hydrogen becomes flammable once its atmospheric volume concentration exceeds the 4 percent level of oxygen, and detonates once it exceeds the 10 percent level. This indicates that failure of the third barrier may have a knock-on effect on the fourth. To break this connection, a containment vessel in operation is filled with nitrogen.

[3] Mark-I type containment vessel design basis allowable pressure calculated based on the same assumption is 430kPa.
A large hatch is located on the lower part of the drywell to load and unload large equipment. A larger drywell head is located on the top of the drywell to provide necessary access to open the reactor pressure vessel for replacing reactor fuel assemblies. The flanges of the hatch and the drywell head use rubber seals and bolt connections. The containment vessel contains different equipment, and many electrical cables for power supply and signal transmission penetrate the containment. Epoxy resin is a representative material used for electrical cable penetration sealing. This is endurable to the design condition defined as the “worst-case LOCA,” but not under the heat and pressure of more severe conditions.

In such a hypothetically intolerable environment, a decision must be made: whether to leave the containment vessel uncontrolled to the point of explosion, or actively release internal pressure to the external environment by abandoning the original function of the barrier to prevent the gross destruction. In fact, the latter is considered to be the only option, because the former bears a risk of an “uncontrolled release” of radiation whereas the latter is a “controlled release.” For this purpose, the hardened vent system is installed. The rupture disk is the ultimate boundary before the external environment. Where the various relevant systems are configured in one line, when the rupture disk fails, gas that filled the containment vessel is released from the top of the plant stack and dispersed according to meteorological conditions (such as wind direction and velocity and atmospheric stability). The external environment will be impacted by the amount of radioactive materials contained in the gas.

e. Containment (fifth barrier)

The nuclear reactor building is also known as the secondary containment vessel, and constitutes the fifth barrier of defence. The allowable leakage per day (i.e. 0.5 percent of the internal air volume) under design basis pressure is provided for the primary containment vessel assuming the worst-case LOCA, while the parameters on external leakage from the secondary containment, or the reactor building, are calculated based on the volume of air released from the standby gas treatment system (SGTS) in a day from the reactor building (i.e. 50 percent) and filter efficiency (i.e. 99 percent) based on an assumption that the airtightness of the reactor building as a boundary is ensured by SGTS and negative pressure by its operation (i.e. -38mmH2O). If the reactor building is damaged in events such as a hydrogen explosion, airtightness is lost and unfiltered air is released directly to the external environment.

If a steam pipe inside the building is ruptured, the building may be destroyed due to a rapid increase of its internal pressure. To prevent this, a blow-out panel is installed, just like the rupture disk of the hardened vent.

2. Nuclear reactor accident, spent fuel pool accident

a. Excursion of the reactor, possibility of nuclear explosion

There are two types of neutrons which moderate nuclear chain reactions. The prompt neutron is emitted immediately after atomic fission occurs, and the delayed neutron is emitted slowly anytime later. The criticality of an operating reactor is normally self-sustained by these two types of neutrons and any change of reaction, upward or downward, takes place relatively slow. Even if the reactivity suddenly increases for some reason, the increase is sufficiently slowed due to the delayed neutron, which helps to naturally stabilize the reaction rate as represented by the rise of water temperature and the development of steam bubbles. Thereby the reactor should not excurse. This is known as negative feedback.

However, under special circumstances—such as during the cold shutdown and the initial stage of reactor start up—reactivity is impressed excessively and rapidly beyond the delayed neutron fraction. The reactor can become critical with the prompt neutron alone as it dominates the chain reaction in so-called prompt criticality, when negative feedback is overridden. Under this condition, nuclear excursion may not be avoided. However, it has been experimentally proven in the USA during an early phase of the development of the light water reactors that a commercial nuclear reactor, which uses low-enriched (instead of high-enriched) uranium of around 4 percent of U-235 and is loaded with design and control to prevent excessive reactivity, would not explode like an
atomic weapon that uses fast neutron emitted from highly-enriched metal uranium to intentionally create prompt criticality when multiple pieces are forced to join together.

b. Decay heat and radiation

Even 5 percent of the nuclear energy in the reactor is immense, as the reactor sends approximately 5,600 tons of steam every hour to the turbine, and can empty the water in the reactor containment vessel in a matter of minutes. After the nuclear reactor successfully shuts down in an emergency (SCRAM), the decay heat continues inside the reactor. The decay heat decreases over time to 2 percent after 10 minutes, 1 percent after 100 minutes, 0.7 percent after 10 hours, 0.5 percent after a day, 0.3 percent after 10 days, and 0.1 percent after 100 days. Yet, 0.1 percent of the entire nuclear energy is still considerable because of the immense amount. Unless this decay heat is dissipated, the fuel pellet and the fuel cladding tube continue to heat up and may cause damage, decay, and meltdown. Stainless steel structure which supports high-melting-point core would experience similar sequence of events known as the fuel damage, core damage, core meltdown, and melt-through, depending on a situation and a stage. It is critically important to remove heat immediately after the reactor shutdown. A timely response in achieving submerging by the ECCS is also crucial in case of LOCA. Failure of initial cooling would make subsequent recovery an extremely difficult and complicated task because the event may transcend multiple barriers one by one and the plant may release radiation.

<table>
<thead>
<tr>
<th>Elapsed time</th>
<th>Amount of radioactivity (TBq)</th>
<th>Decay heat (W)</th>
<th>Radiotoxicity (water kl)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 year</td>
<td>110,000</td>
<td>&gt;10,000</td>
<td>1,000,000,000,000</td>
</tr>
<tr>
<td>10 years</td>
<td>22,000</td>
<td>2,000</td>
<td>400,000,000,000</td>
</tr>
<tr>
<td>100 years</td>
<td>2,600</td>
<td>500</td>
<td>150,000,000,000</td>
</tr>
<tr>
<td>1,000 years</td>
<td>800</td>
<td>100</td>
<td>30,000,000,000</td>
</tr>
<tr>
<td>10,000 years</td>
<td>26</td>
<td>20</td>
<td>10,000,000,000</td>
</tr>
<tr>
<td>100,000 years</td>
<td>4</td>
<td>2</td>
<td>800,000,000</td>
</tr>
<tr>
<td>1,000,000 years</td>
<td>1</td>
<td>0.6</td>
<td>200,000,000</td>
</tr>
</tbody>
</table>

Table 2.1.1-1 provides a collection of values (MIT, The future of Nuclear Power, 2003) that indicate the need for having a long-term view when dealing with spent fuel. These are approximate figures based on 1 ton of PWR fuel initially enriched to 4.5 percent and burnup of 50GW/t, but it can be contracted and applied to BWR, because an operating BWR reactor has a mixture of fuels from the new to fourth cycles.

As mentioned earlier, decay heat remains for a long time. Radiotoxicity is the amount of toxin dilution needed to make water drinkable without posing concerns over health. This table shows the amount of water needed to dilute all radioactive materials contained in 1t of spent fuel. In other words, radioactive materials contained in 1ton of spent fuel would still be undrinkable even if it was diluted with water from Lake Biwa 1,000 years from now.

This information is helpful in quantitatively grasping the impact of radiation released into the external environment by breaking through barriers. The reason for having the “five-layer” barrier can be understood from the table. It is crucial to prevent nuclear accidents, which would result in the effects mentioned above, and to mitigate the effects if accidents do happen. The potential sequence of events must be understood through the following sections.

c. Nuclear reactor accident and SBO

Accidents at nuclear facilities are categorized into the design basis accident (DBA) and the beyond design basis accident (B-DBA). The DBA is a postulated accident at a nuclear facility that has relevant automatic functions designed to withstand an accident,
including LOCA. When LOCA occurs, the pressure inside the containment vessel rises. Immediately after receiving the signal on hyper pressure, the reactor shuts down and ECCS is automatically started. In order to assure the process, instrument systems and the ECCS systems and their power supply systems must be designed with redundancy and diversity. B-DBA is an accident that exceeds the basis of design assumptions, creating a situation in which the automated functions are insufficient for controlling the accident. This is also known as a severe accident. Manual intervention must occur once the situation goes beyond the scope of automatic functions. SBO is the most typical of the severe accidents, and warnings have been made for many years because of its high likelihood of causing core damage. Numerous studies on SBO exist globally.

According to the Station Blackout at Browns Ferry Unit One—Accident Sequence Analysis (1981) by Oak Ridge National Laboratory commissioned by the Nuclear Regulatory Commission (NRC), a nuclear power plant accident hypothetically progresses after SBO (T=0) following several key events over time as shown on Table 2.1.1-2. A nuclear reactor that has been in full-power operation immediately prior to the accident will continue to be cooled by HPCI for four hours until its battery runs down. From then on, the accident evolves into core damage, core melt (meltdown), breach of reactor pressure vessel (melt-through), failure of reactor containment vessel (electrical penetration blow out), melt-through of the primary reactor containment vessel bottom head, and melt-through of the reactor building basement. Based on the assumption that no mitigation measures are taken to address the accident, a seven-meter thick concrete basement of the reactor building will be penetrated in a matter of approximately 14 hours. After 17 hours, the amount of water decreases and steam leak from the failed area will be 1/10 of the peak amount. 20 percent of zirconium contained in fuel cladding reacts with water and creates approximately 250 kilogram of hydrogen gas. According to the report, the chemical reaction from corium-concrete interactions (CCI) between corium from the reactor pressure vessel and the concrete floor becomes active after approximately two hours and creates several kilograms of hydrogen and steam.

### Table 2.1.1-2: Browns Ferry Nuclear Plant Unit 1: SBO Sequence of Events

<table>
<thead>
<tr>
<th>Time (minutes)</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>SBO</td>
</tr>
<tr>
<td>240</td>
<td>HPCI stops when the batteries run out.</td>
</tr>
<tr>
<td>260</td>
<td>Water level inside the reactor drops to “low” (default HPCI starting setpoint) level. Drywell and wetwell temperatures are 85°C and 87°C respectively.</td>
</tr>
<tr>
<td>280</td>
<td>Core uncovers.</td>
</tr>
<tr>
<td>320</td>
<td>Gas temperature at top of core is 485°C.</td>
</tr>
<tr>
<td>340</td>
<td>Gas temperature at top of core is 821°C. Drywell and wetwell temperatures and pressures are 103°C and 0.23MPa.</td>
</tr>
<tr>
<td>355</td>
<td>Core melt starts.</td>
</tr>
<tr>
<td>389</td>
<td>Water level in vessel drops below core support plate.</td>
</tr>
<tr>
<td>390</td>
<td>Core support plate fails.</td>
</tr>
<tr>
<td>392</td>
<td>Debris slumps down to reactor pressure vessel bottom (meltdown).</td>
</tr>
<tr>
<td>394</td>
<td>Debris starts to melt through the bottom head.</td>
</tr>
<tr>
<td>426</td>
<td>Vessel bottom head fails. Pressure of the reactor containment vessel rises to 0.34MPa.</td>
</tr>
<tr>
<td>426.04</td>
<td>Debris (initial temperature 1,433°C) reacts with concrete and produces heat.</td>
</tr>
<tr>
<td>513.59</td>
<td>Electric penetration modules in drywell exceed 260°C, and are blown out of the containment. Mass rates are: 4.61 kilogram steam, 0.11 kilogram hydrogen, 1.01 kilogram carbon dioxide (CO2), and 2.35 kilogram CO per second. The leak rate of the containment vessel is 30.4 cubic-meters per second.</td>
</tr>
<tr>
<td>613</td>
<td>Drywell and wetwell pressures are at 0.10MPa, and temperatures are 661°C and 98°C respectively. The leak rate through the containment failed area is 29.6 cubic-meters per second.</td>
</tr>
<tr>
<td>695</td>
<td>Drywell and wetwell temperatures are 623°C and 97°C respectively. The leak rate through the containment failed area is 64.7 cubic-meters per second.</td>
</tr>
<tr>
<td>Around 840</td>
<td>700 centimeter thick concrete (basemat of reactor building) fails.</td>
</tr>
<tr>
<td>1028</td>
<td>Drywell and wetwell temperatures are 614°C and 97°C. The leak rate through the containment failed area is 1.34 cubic-meters per second.</td>
</tr>
</tbody>
</table>
more than 100 kilogram of carbon monoxide (CO)—a sensible amount— every minute. A chemical reaction takes place continuously for few hours before the concrete is penetrated, and results in a massive release of flammable gas after a melt through.

d. More severe nuclear accident

According to the above analysis, it is assumed that the HPCI core cooling would continue for four hours. However, a more severe nuclear accident including LOCA may happen in combination with a SBO. Oak Ridge National Laboratory set the following six scenarios:

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Events in addition to SBO</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>HPCI/RCIC are initiated and are available for 4 hours.</td>
</tr>
<tr>
<td>2</td>
<td>HPCI/RCIC are not initiated. Open SR valves to rapidly depressurize. Steam cool fuels. Manually start RCIC.</td>
</tr>
<tr>
<td>3</td>
<td>Steam cool fuels by operating SR valves. Manually start RCIC. SR valves are stuck open.</td>
</tr>
<tr>
<td>4</td>
<td>HPCI/RCIC are not initiated.</td>
</tr>
<tr>
<td>5</td>
<td>HPCI/RCIC are not initiated. SR valves are stuck open.</td>
</tr>
</tbody>
</table>

The accident at Fukushima Daiichi Nuclear Power Plant Unit 1, which will be discussed in more detail in the following sections, is almost a “Scenario 5” accident due to the loss of IC system at its early stage. If the potential LOCA is taken into consideration, it is almost a “Scenario 6.”

Coincidence of several such events, as expected, may significantly accelerate the core meltdown, reactor pressure vessel damage, and containment vessel damage. (See Table 2.1.1-4) “Wetwell rupture” in the table below is a postulated failure of the pressure suppression chamber from steam blowout or dynamic loads associated with condensation oscillation. MARK I containments vessels were reinforced against LOCA dynamic loads in Japan in the 1980s, but the reinforcement did not cover a severe accident of this scale.

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Reactor, reactor pressure vessel</th>
<th>Containment vessel</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1st uncovery</td>
<td>Re-flooding</td>
</tr>
<tr>
<td>1</td>
<td>302</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>315</td>
<td></td>
</tr>
<tr>
<td>4</td>
<td>11</td>
<td>12</td>
</tr>
<tr>
<td>5</td>
<td>33</td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>17</td>
<td></td>
</tr>
</tbody>
</table>

The accuracy of the above analysis may be questionable, considering that it was performed over 30 years ago. However, there have been no major discrepancies or deviations from this analysis in the many evaluation reports that followed. In fact, NUREG/CR-6042, training material for NRC staffs issued by the Sandia National Laboratories twenty years later, contains no major revisions from the analysis above.

The NUREG/CR-6042 material contains detailed insights including the hypothetical danger posed by the melt and slump of low melting point control rods as the core melt progresses. This would cause red-hot fuel assemblies to remain within the core. In this case, criticality would not happen because of the lack of water as a moderator, but water injection could result in supercriticality (excursion). This hypothesis is based on the assumptions that only the fuel assembly remains unmelted and the shape of fuel rod is maintained even against the thermal shock of water injection. However, the assumption is extremely remote and therefore is considered inconceivable in reality. There is no discussion of the further progression of core damage or the criticality of
the debris accumulated on the bottom of the reactor pressure vessel and the debris in
the pedestal leaked out from the vessel melt through. The possibility of re-criticality in
an evolving nuclear accident is not treated as a realistic concern. After the water level
drops down to the bottom of the reactor pressure vessel, the core support plate starts
to melt. The slump of debris into the water supposedly causes a steam explosion, but
the analysis concludes that it is not a practical concern on the grounds of various
experimental outcomes.

The corium erosion of concrete was measured in a hypothetical experiment using a
heated cylinder iron and a pile of iron nails to imitate corium. Corium melts concrete
and slumps down. It creates gaseous substances such as steam, hydrogen, CO2, and
CO. Iron, if present, acts as a catalyst to create methane. A crust that permeates gases
may exist on the surface of the eroded concrete. Because the crust accumulates gas
underneath, it inhibits any water cooling effects. The gas includes particles generated
from concrete and becomes a media to carry different radioactive materials (radioac-
tive aerosol).

The possibility of a severe meltdown of the core, burning through the basemat and
through the crust and body of the earth in the so-called China Syndrome, has been test-
ed and analyzed. According to a result of an analysis by a German research institute on
a PWR reactor of a typical size, the debris melts the concrete layer to a 19-meter depth
in 1,050 days, but does not erode further as the heat release and heat generation of the
debris offset each other. Thereafter, debris starts to contract. If there is not a 19-meter
layer of concrete, and a stream of underground water is present beneath the penetrated
basemat, the debris stops expanding vertically before hitting the water table and spreads
laterally. On the 230th day, expansion stops and contraction begins.

e. Release of radioactive substances
Radioactive releases take place in multiple stages during a nuclear accident. The fol-
lowing is based on NUREG-1465 (February 1995), which discusses the source term of
an accident at a light water reactor. When fuel cladding is breached, volatile elements
such as noble gas, halogen, and alkali metal are released from the gap space between
the cladding and the fuel pellet during so-called gap release. It is estimated that
approximately 5 percent of the entire internal volume is released. When the tempera-
ture at which fuel cladding damage occurs is considered, cesium, which is a represen-
tative alkali metal, reacts with iodine, which is a representative halogen, and releases
cesium iodine.

As an accident progresses and fuel pellets start to melt, elements used to fill the gap
will be released. Almost 100 percent of noble gas and 20 to 25 percent of alkali metal
and halogen will be released. Tellurium and strontium will be released during this
early in-vessel release.

During the melt-through, ex-vessel release of radioactivity takes place to emit
radioactive aerosol from CCl as previously mentioned. 30 to 35 percent of alkali metal
and halogen are newly released. Plutonium will be released in addition to 25 percent
of the tellurium and 10 percent of the strontium. In parallel with the ex-vessel release,
later in-vessel release takes place from the residue inside the reactor pressure vessel,
but its quantity is insignificant.

Note that these releases assume that the nuclear accident takes place immediately
after the shutdown and does not take into account any human intervention. If there is
intervention, the release may take place in a significantly different behavior. In terms
of controlling radiation exposure, the significance of the elements and isotopes of
noble gas and iodine varies significantly, depending on the time elapsed from the shut-
down until the release. Krypton, a noble gas, should be treated as a significant radioac-
tive release immediately after the shutdown, but can be ignored against xenon after
one day. Isotopes of xenon Xe-133 and Xe-135 would need attention from the first day
for about three days, whereas only Xe-133 needs emphasis after day three. All iodine
isotopes of I-131, I-132, I-133, I-134, and I-135 need attention immediately after the
shutdown, but I-134 will be excluded after about 12 hours, I-135 after day three, I-133
after day 10, and I-132 after day 30. Only I-131 needs attention thereafter.

Radiation is released from the nuclear reactor and the spent fuel pool. Following is
the inventory of the source and total amount of radiation. This is essential informa-
tion for evaluating the maximum potential risk of the nuclear accident. Table 2.1.1-5 shows the data of the Fukushima Daiichi plant as of March 11, 2011, immediately prior to the accident.

The total amount of radioactivity in the reactors of Units 1, 2 and 3 were 2.90x10^20Bq, 5.00x10^20Bq, 5.00x10^20Bq respectively, relatively large compared to Units 5 and 6 which were shutdown for the refueling and inspection. Total radioactivity of Unit 4 spent fuel pool was 2.10x10^19Bq, higher than any other units. A total of 6,375 spent fuel assemblies were stored in the common pool with total radioactivity of at 1.40x10^19Bq, only second to Unit 4 spent fuel pool.

Table 2.1.1-5: Fuel assembly and total radioactivity in reactors and spent fuel pools [5]

<table>
<thead>
<tr>
<th>Unit</th>
<th>Reactor</th>
<th>Spent fuel pool</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td># fuel assembly</td>
<td>Total radiation (Bq)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>400</td>
<td>2.90E+20</td>
</tr>
<tr>
<td>2</td>
<td>548</td>
<td>5.00E+20</td>
</tr>
<tr>
<td>3</td>
<td>548</td>
<td>5.00E+20</td>
</tr>
<tr>
<td>4</td>
<td>*(548)</td>
<td>**(1.7E+19)</td>
</tr>
<tr>
<td>5</td>
<td>548</td>
<td>1.60E+19</td>
</tr>
<tr>
<td>6</td>
<td>764</td>
<td>1.00E+19</td>
</tr>
<tr>
<td>Common pool</td>
<td>—</td>
<td>—</td>
</tr>
</tbody>
</table>

* rated for Unit 4 ** assuming the fuel was inside the reactor

Table 2.1.1-6: Spent fuel pool storage [6]

<table>
<thead>
<tr>
<th>Unit</th>
<th>Spent fuel assembly</th>
<th>New fuel assembly</th>
<th>Total</th>
<th>Storage capacity</th>
<th>Occupancy %</th>
<th>Decay heat (MW)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>2011/3/M</td>
</tr>
<tr>
<td>1</td>
<td>292</td>
<td>100</td>
<td>392</td>
<td>900</td>
<td>43.6</td>
<td>0.18</td>
</tr>
<tr>
<td>2</td>
<td>587</td>
<td>28</td>
<td>615</td>
<td>1,240</td>
<td>49.6</td>
<td>0.62</td>
</tr>
<tr>
<td>3</td>
<td>514</td>
<td>52</td>
<td>566</td>
<td>1,220</td>
<td>46.2</td>
<td>0.54</td>
</tr>
<tr>
<td>4</td>
<td>1,331</td>
<td>204</td>
<td>1,535</td>
<td>1,590</td>
<td>96.5</td>
<td>2.26</td>
</tr>
<tr>
<td>5</td>
<td>946</td>
<td>48</td>
<td>994</td>
<td>1,590</td>
<td>62.5</td>
<td>1.01</td>
</tr>
<tr>
<td>6</td>
<td>876</td>
<td>64</td>
<td>940</td>
<td>1,770</td>
<td>53.1</td>
<td>0.87</td>
</tr>
<tr>
<td>Common pool</td>
<td>—</td>
<td>—</td>
<td>6,375</td>
<td>6,840</td>
<td>93.2</td>
<td>1.13</td>
</tr>
</tbody>
</table>

[5] Compiled by NAIIC based on the TEPCO documents. In the table, for example, 4.5E+19 represents 4.5x10^19.


f. Loss of coolant accident at spent fuel pool
The loss of coolant accident at the spent fuel pool involves different conditions from that of reactors, including: a lower level of nuclear fissile materials as a result of burnup in the reactors, the time elapsed from the burnup in the reactors, decreased decay heat, potential exposure of spent fuels to the air-atmosphere in case of the coolant water loss, lack of other containment function besides the fifth barrier (reactor building), and the larger amount of stored fuel than in a reactor.

The loss of cooling water in the spent fuel pool that stores hot spent fuel assemblies may result in a “zirconium fire” from overheating, depending on the degree and situation of the pool damage. In order to mitigate this, the National Academy of Science proposed in a 2004 report a concept of reconfiguring spent fuel assemblies in a checkboard pattern. Based on this report, NRC mandated in the Security Order (B.5.b) for nuclear power operators to follow this recommendation as one of the efforts under “Phase I.”

In this way, management of the fuel assembly in the spent fuel pool must be given sufficient consideration in the same way as the ones in the reactor. Table 2.1.1-6 shows the spent fuel pool storage at the Fukushima Daiichi plant as of March 11, 2011, immediately prior to the accident.

Unit 4 and the common pool were almost fully loaded; 96.5 percent and 93.2 percent were occupied, respectively. Also, it is notable that they continue to cast high decay heat as of January 1, 2012.
2.1.2 Key damage and the impact of the earthquake and tsunami

The accident is clearly attributable to natural phenomena: the earthquake and resulting tsunami. The weakness of the nuclear power plants, whose owners procrastinated in implementing necessary safety measures, was exposed by the damage and effects of the disaster. This section provides an overview of the damage and effects of the earthquake and tsunami, as well as observes and assesses issues concerning the safety of nuclear power plants, considering the occurrence of earthquakes and tsunamis.

1. Key damage and impacts

The Great East Japan Earthquake of March 11, 2011 at 14:46 damaged power transmission grids from Shin-Fukushima Electrical Substation of TEPCO to the Fukushima Daiichi plant, and cut the electricity supply. The power plant was connected to a backup 66kV nuclear line from Tohoku Electric Power, but this was not available because of the failure of a cable connected to a metal-clad type switchgear (M/C) for Unit 1. As a result, the plant lost all its off-site power.

In addition, the tsunami hit the plant about 50 minutes after the earthquake and inundated a number of emergency diesel generators, cooling seawater pumps, the onsite power distribution system, and the DC power supply system.

Units 1, 2 and 4 lost all their power supply. Units 3 and 5 lost all AC power supplies. The DC power supply in Unit 3 ran out and it lost power completely on March 13 at 2:42.

The earthquake and tsunami damage was not limited to the power supply systems. The tsunami’s massive energy washed vehicles, heavy equipment, heavy oil tanks, dirt, and other debris over the plant and ruined buildings, equipment and facilities. Tsunami waves reached as far as the ultrahigh voltage switchyard of Units 3 and 4 and the underground radiation waste storage facility building (common pool building) of Units 3 and 4. A large amount of seawater flooded key buildings. The tsunami left debris covering the plant site, hindering efforts to deliver equipment and materials. It blew up manholes and gratings and created holes. As a result, along with the roads within the site ruined by the earthquake, accessibility was severely degraded. In the chaos following the destruction, workers were greatly hindered in their response efforts by continued alerts and aftershocks and tsunami. The main control room functions including monitoring and control, lighting of the plant facilities and communications were completely lost. As a result, the operators and staff on-site had to make spontaneous decisions and responses without relying on effective measures and procedure manuals. They were forced to deal with the accident in uncertain situations.

The loss of power made it extremely difficult to cool the reactors in a timely and effective way, because the execution of the series of steps to cool down the reactor leading to a cold shutdown is heavily dependent on power availability. Such steps include injecting high pressure coolant, depressurizing the reactors, injecting low pressure coolant, cooling or pressure drop in the containment vessels, and removing decay heat to the ultimate heat sinks. The difficult access to the plant site mentioned earlier hindered the alternate use of fire trucks to inject water and constantly stalled efforts to use generator trucks to provide a temporary power supply and to configure power lines for venting the containment vessel.

2. Observation and assessment
   a. Redundancy, diversity and independence of power supply system under a natural disaster

[7] Grating is a steel made storm drain cover.
[8] In Chapter 2, the Japanese text uses two different terms for the main control room.
[9] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident.
[10] Redundancy is referred to as a state of having two or more systems or equipment of identical nature intended to function identically. Diversity is referred to as a state of having two or more systems or equipment of different natures intended to function identically. Independence is referred to as a state of ensuring two or more systems or equipment to be unharmed simultaneously from common factors or dependent factors in design basis environmental condition and operating state.
The loss of the power supply has again shown the nuclear power plant's dependence on electrical power to ensure safety, as well as the significant importance of power. At the same time, the accident has reaffirmed that the power supply system is the system that stretched in and out of the plant facilities.

In order to increase the reliability of the power supply system and prevent a loss of power—especially for equipment and facilities indispensable in addressing a severe accident—attention needs to be paid to more than a single failure. Redundancy, diversity and independence need to be designed based on a perspective of assuring the safety of the entire nuclear power generation system against a potential realization of complex threats that may undermine the safety functions of multiple equipment and facilities.

<table>
<thead>
<tr>
<th>Equipment</th>
<th>Unit 1</th>
<th>Status</th>
<th>Unit 2</th>
<th>Status</th>
<th>Unit 3</th>
<th>Status</th>
<th>Unit 4</th>
<th>Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>Emergency diesel generator</td>
<td>D/G 1A</td>
<td>× Submerged</td>
<td>D/G 2A</td>
<td>× Submerged</td>
<td>D/G 3A</td>
<td>× Submerged</td>
<td>D/G 4A</td>
<td>× Submerged</td>
</tr>
<tr>
<td>Emergency M/C</td>
<td>M/C 1C</td>
<td>× Flooded</td>
<td>M/C 2C</td>
<td>× Submerged</td>
<td>M/C 3C</td>
<td>× Submerged</td>
<td>M/C 4C</td>
<td>× Submerged (under maintenance)</td>
</tr>
<tr>
<td></td>
<td>M/C 1D</td>
<td>× Flooded</td>
<td>M/C 2D</td>
<td>× Submerged</td>
<td>M/C 3D</td>
<td>× Submerged</td>
<td>M/C 4D</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>M/C 2E</td>
<td>× Submerged</td>
<td></td>
<td></td>
<td>M/C 4E</td>
<td>× Submerged</td>
</tr>
<tr>
<td>Normal M/C</td>
<td>M/C 1A</td>
<td>× Flooded</td>
<td>M/C 2A</td>
<td>× Submerged</td>
<td>M/C 3A</td>
<td>× Submerged</td>
<td>M/C 4A</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td>M/C 1B</td>
<td>× Flooded</td>
<td>M/C 2B</td>
<td>× Submerged</td>
<td>M/C 3B</td>
<td>× Submerged</td>
<td>M/C 4B</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td>M/C 1S</td>
<td>× Flooded</td>
<td>M/C 2SA</td>
<td>× Submerged</td>
<td>M/C 3SA</td>
<td>× Submerged</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>M/C 2SB</td>
<td>× Submerged</td>
<td>M/C 3SB</td>
<td>× Submerged</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Emergency P/C</td>
<td>P/C 1C</td>
<td>× Submerged</td>
<td>P/C 2C</td>
<td>× Submerged</td>
<td>P/C 3C</td>
<td>× Submerged</td>
<td>P/C 4C</td>
<td>— Under construction</td>
</tr>
<tr>
<td></td>
<td>P/C 1D</td>
<td>× Submerged</td>
<td>P/C 2D</td>
<td>× Submerged</td>
<td>P/C 3D</td>
<td>× Submerged</td>
<td>P/C 4D</td>
<td>× Submerged (under maintenance)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>P/C 2E</td>
<td>× Submerged</td>
<td></td>
<td></td>
<td>P/C 4E</td>
<td>× Submerged</td>
</tr>
<tr>
<td>Normal P/C</td>
<td>P/C 1A</td>
<td>× Flooded</td>
<td>P/C 2A</td>
<td>× Submerged</td>
<td>P/C 3A</td>
<td>× Submerged</td>
<td>P/C 4A</td>
<td>— Under construction</td>
</tr>
<tr>
<td></td>
<td>P/C 2A-1</td>
<td>× Submerged</td>
<td>P/C 3B</td>
<td>× Submerged</td>
<td>P/C 4B</td>
<td>× Submerged (under maintenance)</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>P/C 1B</td>
<td>× Flooded</td>
<td>P/C 2B</td>
<td>× Submerged</td>
<td>P/C 3SA</td>
<td>× Submerged</td>
<td>P/C 4B</td>
<td>× Submerged (under maintenance)</td>
</tr>
<tr>
<td></td>
<td>P/C 1S</td>
<td>× Flooded</td>
<td>P/C 2SB</td>
<td>× Submerged</td>
<td>P/C 3SB</td>
<td>× Submerged</td>
<td></td>
<td></td>
</tr>
<tr>
<td>DC 125V</td>
<td>125V DC BUS-1A</td>
<td>× Submerged</td>
<td>125V DC BUS-1B</td>
<td>× Submerged</td>
<td>125V DC BUS-1A</td>
<td>× Submerged</td>
<td>125V DC BUS-2A</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td>125V DC BUS-2A</td>
<td>× Submerged</td>
<td>125V DC BUS-2A</td>
<td>× Submerged</td>
<td>125V DC BUS-3A</td>
<td>× Submerged</td>
<td>125V DC BUS-4A</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td>125V DC BUS-3A</td>
<td>× Submerged</td>
<td>125V DC BUS-3A</td>
<td>× Submerged</td>
<td>125V DC BUS-4A</td>
<td>× Submerged</td>
<td>125V DC BUS-4B</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>125V DC BUS-3A</td>
<td>× Submerged</td>
<td></td>
<td></td>
<td>125V DC BUS-4B</td>
<td>× Submerged</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>DC 125V 4D/G B Main bus</td>
<td>× Submerged</td>
</tr>
</tbody>
</table>

Table 2.1.2-1: Locations, damages and availability of on-site power supply system

O=Yes
X=No

[11] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
[12] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
[13] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
The electrical power supply system at the Fukushima Daiichi plant at the time of the accident was reviewed from the on-site power system and off-site power system. Their designs were verified as follows:

(i) On-site power system (Table 2.1.2-1, Figure 2.1.2-1 and 2)
   - Multiple equipment and facilities were installed in same place. All normal M/C and emergency M/C, and normal power center (P/C) for Unit 1 were installed on first floor of the turbine building.
   - Some of the equipment and facilities for upstream and downstream of power supply systems were installed in the same or adjacent buildings. All normal M/C and emergency M/C, normal P/C and emergency P/C, and emergency diesel generator for Unit 3 were installed on the basement floors of adjacent turbine building and control building.

For these reasons, the power supply system was vulnerable to external events such as flooding and fire and threats of intentional attacks in addition to external flooding as materialized by the tsunami. A station blackout could have happened even if a specific building was damaged.

(ii) Off-site power system (Figure 2.1.2-3)
   - There were seven grids connected to Units 1 to 6: Okuma line #1, Okuma line #2, Okuma line #3, Okuma line #4, Yonomori line #1, Yonomori line #2, and TEPCO nuclear line.
There were three transmission routes: 1) Okuma lines #1 and #2, 2) Okuma lines #3 and #4 and Yonomori #1 and #2 multi-circuit, and 3) TEPCO nuclear line.

- Okuma lines #1 and #2 and Okuma lines #3 and #4 and Yonomori #1 and #2 multi-circuit were connected to TEPCO Shin-Iwaki Switchyard and Shin-Fukushima Substation, and TEPCO nuclear line was connected to Tomioka Substation of Tohoku Electric Power.
- The nuclear line had been unavailable due to a cable defect when the disaster took place.

The transmission supply system was vulnerable to external events such as typhoons, tornados, heavy snow and threats of intentional attacks, in addition to the
risk of the collapse of transmission towers (for Yonomori lines #5 and 6) as happened in this earthquake. Yet they were configured in ways that all of the Units 1 to 6 would lose off-site power if the transmission function failed at all three transmission routes or if power was lost at the Shin-Fukushima Substation or Shin-Iwaki Switchyard of TEPCO, and either of these combined with the loss of the Tomioka Electrical Substation.

Diversity and independence of the power transmission system in general were not sufficient to withstand natural disasters. To this end, the design of the entire power transmission system needs to be reflected and realigned for diversity and independence.\(^\text{[18]}\)

b. Earthquake resistance at Shin-Fukushima Substation

The facilities of the Shin-Fukushima Substation had become obsolete over the 34 years since the elevation of operation to 500kV. The developed land had been eroded by rainfall because of the geological nature of the site. It was estimated that if an earthquake the size of the design earthquake ground motion\(^\text{[19]}\) occurred at the Fukushima Daiichi plant, the ground motion would be amplified at Shin-Fukushima Substation because it stands on a complicated ground fault (known as the Futaba Fault). It was estimated that the maximum acceleration of free rock surface would reach 1024 Gal.\(^\text{[20]}\) According to a TEPCO document, without upgrading the current earthquake-resisting capacity, it was considered difficult to restore off-site power within seven days if maximum acceleration at the free rock surface was 1024 Gal.\(^\text{[21]}\)

The anti-earthquake enhancement of transmission systems relating to Shin-Fukushima Substation was scheduled to be completed in 2020. In other words, the substation was still vulnerable to an earthquake as of March 11, 2011. In fact, the electric substation equipment, including a breaker, was damaged by the seismic motion, which contributed to the loss of off-site power.

c. Impact of loss of the main control room function, lighting and communications

(i) The main control room function (the main control room could not address emergency)

Because of the loss of the main control room functions, the operators struggled to correctly understand, judge, and act on the state of the reactors and the rapid escalation of the accident. The operators at the main control room could barely take measures to correct the situation and in some cases several hours were wasted.\(^\text{[22]}\) Even worse, a number of tasks—such as monitoring and control, which are normally performed from the main control room—had to be carried out directly where equipment and facilities were located.

The lack of direct information on the state of reactors and on the rapid escalation of the accident greatly hindered and confused off-site stakeholders\(^\text{[23]}\) in obtaining necessary information on which to judge and act, and had critical rippling effects.

(ii) Lighting (inhibited or delayed on-site emergency response)

Due to the loss of the main control room functions, a number of emergency response tasks had to be carried out on the spot. However, some tasks at the plant were carried out in complete darkness because the plant lost its lights. The workers had to

---

\(^\text{[18]}\) Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident


\(^\text{[20]}\) TEPCO documents

\(^\text{[21]}\) TEPCO documents

\(^\text{[22]}\) Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\(^\text{[23]}\) On-site Emergency Response Center, TEPCO headquarters Emergency Response Center, Off-site Center, NISA and other related agencies, local governments and residents.
memorize the design documents before leaving the main control room or the plant
emergency response center to go to the specific location and carry out tasks. Although
they went to the area where work was needed, the disruption of roads, debris and drifts
from the earthquake and the tsunami added to the loss of lighting in preventing work-
ers from carrying out their tasks and causing delays.

Tremors and roaring sound of aftershocks further threatened the workers who were try-
ing to get to the necessary work location as soon as possible in complete darkness.

(iii) Communications (information was not communicated, delayed or mis-
communicated)

The communication systems which connect the plant and the main control room,
as well as those within the control room (such as the paging system\textsuperscript{24}), and other elec-
tric power security communication equipment such as PHS and landline phones and transceivers, were all disabled by the earthquake and tsunami. As a result, the workers
had to relay messages\textsuperscript{25} and to use the malfunctioning fire alarm as a makeshift sig-
nal.\textsuperscript{26} Such means of communications were neither comprehensive nor prompt, and,
in fact, resulted in a deterioration of work efficiency. It is obvious that limited commu-
nications made it difficult to ensure the safety of the operators.

The plant's main control room and Emergency Response Center\textsuperscript{27} were connected
by two hotlines.\textsuperscript{28} Because of the large volume of information that needed to be pro-
cessed, there was a lot of confusion.\textsuperscript{29} Accordingly, necessary information was mis-
communicated multiple times.\textsuperscript{30}

The flow of information was mainly one-way, from the main control room to the
plant Emergency Response Center. As a result, the main control room had no infor-
mation about events outside, including the status of the other reactors and power
plants, and the safety of their families.\textsuperscript{31} Fear, stemming from a lack of information,
caused mental stress among the workers\textsuperscript{32} and made the emergency response even
more difficult.

The main control room functions and communications are the most critical fun-
damental infrastructure for dealing with emergency situations like a severe accident.
Therefore they must be designed with redundancy, diversity and independence and
operated with contingencies in mind, similar to the power supply system.\textsuperscript{33}

\textit{d. Functionality and habitability\textsuperscript{34} of the main control room}

The main control room must have the highest level of functionality and habitability in
its role as the front of the accident response since a limited number of operators need
to stay in the main control room for many hours under mentally and physically harsh
conditions to respond to an accident. In reality, the Fukushima Daiichi plant lost the
main control room functions, the lighting both on-site and off-site and their means of
communication. As a result, they lost multiple methods that could have lead to a safe
shutdown. It is obvious that the functionality of the main control room was insufficient.

In terms of habitability, the main control room failed to provide radiation prote-
cion. Specifically, the air condition and ventilation systems that were in place to pro-

\textsuperscript{24} Paging is a communication system installed throughout the plant facilities used for communication within
the plant site. It allows for a clear broadcasting and two-way communication.

\textsuperscript{25} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{26} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{27} The on-site Emergency Response Center of Fukushima Daiichi was located inside the Seismic Isolation Building.

\textsuperscript{28} TEPCO documents

\textsuperscript{29} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{30} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{31} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{32} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{33} Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\textsuperscript{34} Habitability of the main control room means an environment which allows for operators to carry out monitor-
ing and operation over a certain period of time until the accident converges.
tect the internal environment from radiation by maintaining normal internal pressure failed to function properly due to the loss of power. As a result, radioactive materials entered the main control room as core damage progressed. The operators inside the main control room were put under the added stress of exposure to radiation. The workers were barely able to eat, sleep or use toilets, despite the fact that these activities are indispensable to supporting difficult emergency efforts over many hours.\[33] The habitability of the main control room was poor.

The functionality and habitability of the main control room in its role at the front of the emergency response were insufficient. On top of the assumption of a severe accident such as loss of power supply, the functionality and habitability of the main control room need to be improved.

e. Effectiveness of the efforts to avert accidents by depending heavily on assistance and supplies from outside the plant

A massive volume of diverse equipment and materials were requested by the Fukushima Daiichi plant: fire trucks, generator trucks, hoses and cables, fuel, batteries, pumps, motors, reactor cooling water, radioactive protection gear, consumables and other supplies. The plant was not prepared to immediately secure and implement a number of the necessary equipment and materials at the plant for response to a situation where all the six reactors from Units 1 to 6 were damaged at the same time and where each could have progressed into a reactor accident. NAIIC questions if it was possible to avert an accident by depending heavily on supplies from outside the plant.

Their efforts must have faced numerous difficulties. But, if the plant had been prepared to procure materials from within the plant or from nearby plants in a timely manner, the reactor accident could have been mitigated if not prevented.\[36]

In this accident, the heavy dependence on the off-site supplies was not effective in many ways because of the limited means of communication,\[37\] the high risk of miscommunication,\[38\] access problems due to the disruption of roads and the debris from tsunami, and the high level of radiation in the vicinity of the plant, which inevitably suspended the logistics.\[39\] Some equipment and goods that were received at the plant were useless without other supplies that had not yet been delivered.\[40\]

Due to these limitations, off-site procurement during a reactor accident tends to be difficult. The equipment and materials necessary for the emergency response\[41\] must be kept on-site or near the plant.

2.1.3 Progression of the reactor accidents

The Fukushima Daiichi plant lost its power from the earthquake and tsunami as verified in 2.1.2 above. In addition to the dangerous and severe working environment, the reactor cooling faced difficult conditions. Despite the continued work by the operators to avoid them, the situations of Units 1, 2 and 3 evolved into reactor accidents, and the reactor building of Unit 4 exploded, with its spent fuel pool exposed to the outside environment. Units 5 and 6, however, succeeded in establishing cold shutdown, although they did encounter some risks.

The following is a comprehensive overview of the escalation of the accident at Units 1 to 4. It provides observation and assessment on arguments regarding the progression of the accident by each reactor. Evolutions at Unit 5 are reviewed through the same process.

1. Progression of the accidents at Units 1 to 4

Some of the other investigation reports have explained the progression of the accident

\[35\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
\[36\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
\[37\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
\[38\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
\[39\] TEPCO documents
\[40\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
\[41\] Equipment such as seawater pump and motor, backup pump, hose, generator truck, cable, high-capacity power supply equipment, etc.
and the level of damage at each reactor of the Fukushima Daiichi plant in chronological order. Instead of repeating a similar time-series description, this report focuses on the attempts by the operators of the plant in response to the damage simultaneously inflicted on Units 1 to 4 and summarizes the flow of their response at a high level.

a. March 11, 2011

(i) Unit 2—regarded as the most critical at the beginning

The scram at Unit 1 took place in response to the earthquake. Heat exhaust to the main condenser was blocked due to a closure of the main steam isolation valve but the reactor had IC to regulate pressure. IC itself is the ultimate heat sink that takes in decay heat from the reactor via the main steam line, and transfers heat to water in the shell side of IC through smaller heat-exchanging tubes. The water in the shell side of the IC started to boil and the water level eventually started to decrease. There was a backup water makeup system, but the loss of AC power from the emergency diesel generator and DC from the battery resulted into a loss of IC operation. IC stopped operating when it was isolated. This contributed to a rapid deterioration of the cooling capability of the reactor core.

Despite the significant change in the situation at Unit 1, it was not recognized by the plant operators nor shared among other plant personnel. A higher priority was given to the RCIC of Unit 2 because its operational condition was unknown. Because Unit 2 also lost both AC and DC power systems, there was a suspicion that RCIC, which had been operating as expected, might have stopped. If that was correct, the reactor water level would go down to the top of the active fuel (TAF) by 21:40. Knowing this, TEPCO determined that the condition fell under Article 15 of the Act on Special Measures Concerning Nuclear Emergency Preparedness and reported the situation to the government at 16:36.

Fukushima Prefecture was the first to issue an evacuation order to the residents within a 2 km radius from the plant (instead of the national government), followed by a 3km zone evacuation order issued approximately 30 minutes later at 21:23 by Prime Minister Kan.

(ii) Unit 1—actually in a more critical condition

The plant personnel on the ground were trying to determine the water level in the reactor as well as the status of RCIC as soon as possible. The water level instrumentation was recovered at 21:40, past the estimated time of the TAF exposure. At that time, the reading was 3,400 millimeters above TAF. Feeling relieved, the operator informed the government. Later, the pressure instrumentations for the reactor and the primary containment vessel were also recovered, showing normal values at 23:25.

Compared to Units 1 and 2, which were in complete darkness, Unit 3 was relatively in a better status due to the survival of the DC power supply. Although it lost AC, the measuring instruments still functioned correctly, allowing the operators to monitor the operation of RCIC. HPCI was also available. A small portable power generator was delivered around 22:00, and the lighting of the main control room was restored for Units 3 and 4.

(iii) Rapid escalation of core melt at Unit 1

The level of radiation had increased in the reactor building of Unit 1 by around 21:50, and the building was declared off-limits. While some workers mentioned later that the emergency response team in the Seismic Isolation Building was less concerned about the status of Unit 1 IC, they made their next moves early on: they configured the fire protection system to mate with the core spray system, activated the diesel pump and started stand-by operation at 17:30 so that the reactor pressure vessel water injection could be performed once the reactor pressure decreased to 0.69MPa or lower. Yet, two hours had already passed since the isolation of IC, and it was estimated that the top of the core had already been exposed above the water and the core melt had already started. In this case, the hydrogen build-up must have been taking place from the zirconium-water reaction.

When DC was restored incidentally at 18:18, an operator turned on the switch in the hope of activating the IC system. The operator heard the sound of steam exhaust
from the shell side of the IC but it faded out after a while. This would make sense if the tube side of IC had been filled with noncondensable hydrogen gas instead of condensable steam.

The IC is capable of depressurizing the reactor very rapidly only if it is activated. Yet, the reactor pressure after 20:00 was reported as 6.9MPa without any makeup from the fire pump starting. The water level of the reactor was 200 millimeters above TAF as of 21:19. But by this time, the water level gauge was showing an erroneous reading.

It is estimated that the core had been damaged to a significant extent and gaseous radioactive material had been leaking from the primary reactor containment vessel to the reactor building by the time the reactor building was made off-limits. This is consistent with the study by Oak Ridge National Laboratory. According to its report, CCI had also started and a massive amount of CO in addition to hydrogen gas had been emitted from the concrete floor where corium had been deposited. Presumably for these reasons, the pressure of the containment vessel exceeded the design pressure and reached 0.6MPa[abs].[^42^] Unit 1 was growing increasingly dangerous by the minute.

**b. March 12, 2011**

(i) Reactor pressure vessel of Unit 1 fails

The condition of the Unit 1 reactor further deteriorated. After midnight, the operators started exploring the possibility of venting. The diesel pump had run out of fuel and stopped running before pumping any water. It never restarted and was never used. Instead, fire trucks were used for water injection after depressurizing the reactor. There were originally three fire trucks at the plant, but one was out of order, and another could not be moved from its location at Units 5 and 6 due to the effects of the earthquake. As a result, only one fire truck was available at Unit 1. However, in the end, water was never injected since the IC was isolated. At 02:30 the reactor pressure vessel failed, and the pressures of the reactor and containment vessels equalized at 0.8MPa. Gas constantly blew from the pressurized containment vessel to the reactor building via electrical cable penetrations, reactor top flange, equipment hatch, etc. Radioactive airborne materials, vapor and hydrogen filled the interior of the reactor building. Radioactive airborne material started to leak from the reactor building to the external environment, and the radiation level at the site boundary continued to rise. All workers, including those dealing with the accident outside the building as well as those working in the main control room with only a flashlight, were instructed to wear full-face respirators. It is considered that short half-life radioactive iodine nuclides such as I-132, I-133, I-134, and I-135 were heavily contained in the emission at this time in addition to I-131, which later became a focus of interest.

At 05:14, the evacuation zone was expanded to a 10 km radius. No effective measures had been taken at all regarding the condition of Unit 1 by this time. The containment vessel pressure (which is also the reactor pressure) started to decline, due either to an increased leak from the containment vessel or the decreased amount of gas from CCI inside the containment vessel. At any rate, the lower pressure finally allowed water injection to the reactor pressure vessel. Water was injected from the water tank using a fire extinguishing pump, but the pressure was still too high for the pump to inject water at a sufficient flow rate.

(ii) RCIC—lifeline for Units 2 and 3

It was thought that the RCIC in Unit 2 was functioning, on the grounds that the reactor water level had been stable, but the actual operation status had not been directly confirmed. And there was no assurance as to how long it would continue to work. Since March 11, use of the high pressure coolant had been explored. Generator trucks were brought near Unit 2 and cabling for the standby liquid control system (SLC) started around midnight. Although the use of a control rod drive (CRD) pump was preferred as far as capacity was concerned, it required operating other supporting equipment at the same time. Because the outlook was pessimistic, further efforts to use the CRD pump were discontinued.

[^42^] Absolute pressure and gauge pressure are the two indicators for the effects of pressure on equipment. Absolute pressure [abs] is inclusive of the atmospheric pressure component (approximately 0.1MPa) whereas gauge pressure [g] is the pressure in excess of atmospheric pressure.
RCIC was stopped at 11:36 at Unit 3 where the DC was still functioning. Fortunately, however, about an hour later at 12:35, HPCI was automatically activated, and the reactor water level was recovered.

(iii) Delayed venting of Unit 1 and the frustration of the government

The Minister of METI had ordered TEPCO to vent Unit 1, and Prime Minister Kan visited the Fukushima Daiichi site. The pressure inside the containment vessel peaked and the reactor building was turning into a dangerous environment due to leak of explosive gases such as hydrogen and CO.

Opening the vent valves was a very time-sensitive task. While TEPCO confirmed the evacuation of nearby residents, members of the special taskforce were administered potassium iodine (KI) tablets and gathered on standby in the main control room of Units 1 and 2. Even though they finally received a go sign, the actual task of opening the vents was unexpectedly troublesome. Additional time was needed to procure other equipment, including an air compressor. The vent valves were finally opened at 14:30.

According to the Site Superintendent, Masao Yoshida:

“We were told to hurry up, but as you've heard, we were out of power, and the valves were stuck. The only way was to manually open [the vent valves]. Since the dose was already rising high in Unit 1, it was getting impossibly hard. We examined many options. We tried the bebicon, a tiny mobile air compressor that activates the vent valves, but it didn't work at all. So we asked the contractors to work on the engineering construction side. In short, it required intensive labor work to open the vents …

Everyone at the time said, 'Open the vents.' But even the people on-site were struggling, trying to confirm whether the venting was actually working. We were devoted to getting it done. I know some say that we were disturbed [by the government intervention.] But people on-site had to think through every possible option all the while we were working on it. So, it wasn't so much that we were disrupted as that the progress was very slow. . . . Maybe the progress was not visible, but we were desperately working on it.”

(iv) Explosion at Unit 1 and its impact on Unit 2

Routing cables to the SLC pump at Unit 2 required extremely hard labor. At 15:36, when the cable routing was almost complete, the reactor building of Unit 1 exploded, ruining all the cable routing efforts. Debris from Unit 1 was thrown everywhere, damaging the cables and injuring five workers. The shock of the explosion also knocked off the blow-out panel of the reactor building of Unit 2.

A monitoring post at the site boundary indicated a radiation level exceeding 1 milli-Sievert per hour. At 18:25, Prime Minister Kan announced the expansion of the evacuation zone to a 20 km radius. At 19:04, the fire trucks finally started to fill Unit 1 reactor with seawater.

Meanwhile, as of 17:30, the RCIC at Unit 2 was still operating. However, due to decay heat, the temperature of the suppression pool had been too high to condense high temperature steam. The pressure inside the containment vessel of Unit 2 continued to rise, and with heightened concerns, actions were begun to prepare for potential venting.

At Unit 3, some DC power ran out at 20:27, and the drywell pressure indicator was no longer displayed. The water level indicator disappeared 10 minutes later. Still, the HPCI continued to operate.

c. March 13, 2011

(i) Unit 3 in crisis

The HPCI of Unit 3 stopped at 02:42, and the reactor lost all means of water injection. The reactor pressure surged, and the diesel fire pump could not inject water. The core started to be uncovered at 04:15, which is when a massive amount of hydrogen is believed to have started to develop from the zirconium-water reaction. The operators went into the torus room to perform the venting operation. It was already extremely hot due to massive decay heat from the reactor as a result of RCIC, HPCI and the main steam safety relief valve (SR valve) operations. The reactor pressure exceeded 7.38MPa.

[43] Hearing with Masao Yoshida, Site Superintendent of Fukushima Daiichi
by 05:00, and the water level was 2,000 mm below TAF and still descending. The con-
tainment vessel pressure was rising, reaching 0.46MPa[abs] by 05:15. The water level in
the reactor dropped to the core support plate by 07:35.

The vents of Unit 3 were opened successfully at 08:41, and the containment vessel
pressure started to decline from the peak of 0.637MPa[abs]. The radiation level at the
site boundary was 882 micro-Sieverts per hour. The workers who had been collect-
ning batteries returned and connected them to open the solenoid valves to actuate the
SRVs. The reactor pressure was decreased sufficiently by 09:25, and water injection
began immediately. Before long, TAF was reflooded.

However, the water tank emptied at 12:20. Because the submerging stopped due to a
shortage of water, the reactor water level again dropped below TAF. At 13:00 the top of
fuel was 2,000 mm above water. Seawater was injected later, but the TAF level was not
recovered. The radiation level at the air lock door of the reactor building reached 300
milli-Sieverts per hour, and 12 milli-Sieverts per hour in the main control room.

The RCIC continued to function at Unit 2, but at 11:00 it was estimated to have
reached a very difficult condition. The operators started preparing for depressuriza-
tion and water injection using fire trucks.

d. March 14, 2011
(i) Explosion of Unit 3

Unit 3 boiled dry, and the core became completely uncovered by 04:30. Fire trucks
and SDF water trucks arrived to assist water injection. While they were preparing for
water injection, orange lights began to flash, and the reactor building exploded at
11:01. The explosion blew wreckage and dust hundreds of meters high and the fall-
ing debris ripped a huge hole in the roof of the turbine building. Seven workers were
injured and all work was interrupted. It took more than five hours to restart the sea-
water injection at 16:30.

The explosion of Unit 3 also affected the work in progress at Unit 2. The hoses and
fire trucks for submerging the reactor were damaged, and the workers had to start
from scratch, again, reconfiguring the watering lines to the reactor. The high level
radiation emitted from the debris, however, made this extremely difficult. Site Super-
intendent Yoshida said:

“Units 1 and 3 had the hoses lined up for seawater injection, but we did not have
water. The work was interrupted sometimes, but they were lined up and ready. So we
thought that we should work on Unit 2. But when the work had reached a certain level,
the Unit 3 hydrogen explosion took place. The resulting debris stopped all the sea-
water injection systems of Units 1, 2 and 4. Because the systems were ruined and Unit 3
had exploded, everybody was upset. So we decided to pull everyone back once. We told
them that there would be no more explosions and begged them to get rid of the debris
and work on realigning the hoses outside. There was a construction company working
on this. Its people were divided into teams; one team to clear debris using backhoes,
the second team to check if the fire trucks and fire pumps were still alive [usable]
and to line them up, and the other team to assist these teams. We sent people out,
but the damned debris was just too much... They were subcontract workers, but
they really did a great job in spite of the high radiation dosage from the debris. We
really thought that our engineers should be able to operate backhoes. But it was the
subcontractors instead who carried out the work. They actually finished the difficult
work, even the configuration and lining up of [hoses], much sooner than I expected.
Thanks to them, at last we were almost there. However, since it still had taken so
many hours, the temperature at the suppression chamber beneath the containment
vessel had climbed very high. High temperature steam was released from the reactor
pressure vessel and discharged into the suppression chamber that contained water
already too hot to condense. Usually, the water temperature is around 50°C and
should condense discharged steam quickly. But at that time, the water temperature
was tens of degrees above one hundred.” [44]
(ii) RCIC stopped at Unit 2

At 13:25, the RCIC, which had been helpful in continuously cooling the Unit 2 reactor, came to a halt. While the reactor water level was maintained at 2,400 mm above TAF, it was estimated that the top of the core would start to be uncovered by 16:30. The operators knowingly had to suspend the recovery work due to repeated aftershocks. By the time they resumed the recovery work at 16:00, the water level had declined to only 300 mm above TAF. The core uncovering started without the situation improving.

The core became fully uncovered at 18:22. The SRV was released to lower the reactor pressure, but the pressure inside the containment vessel did not increase as expected. From this, it was thought that there was a leak from the containment vessel to the reactor building. The reactor pressure decreased to 0.63MPa. Because the fire trucks that had been injecting water ran out of gas, the reactor continued to boil dry. Water injection started at around 20:30 but it had to be disrupted several times until around 21:20, because each injection was accompanied by increasing reactor pressure. Water injection was suspended until depressurization allowed water to be injected again. At 21:20, depressurization of the reactor was accelerated with the opening of two SR valves. This, in turn, facilitated injection into the reactor pressure vessel. The water level thus recovered to 1,600 mm below TAF by 22:00, still far less than enough to cover.

e. March 15, 2011

(i) Breach of Unit 2 containment vessel

After RCIC stopped in Unit 2, the reactor continued to boil dry. Drywell pressure increased to 0.75MPa[abs] at 00:02. By 06:00 it had reached 0.73MPa[abs], and the reactor water level 2,800 mm below TAF.

Then the reactor building of Unit 4 exploded. At the same time, a loud noise was heard in Unit 2’s torus room. Immediately after the explosion, the radiation level measured at the gate of the Fukushima Daiichi plant was almost 0.6 mili-Sieverts.

Because the working environment had degraded and there was an increased potential of other hidden dangers, a number of workers were moved to the Fukushima Daini plant. While there was no monitoring of Fukushima Daiichi Unit 2 from 07:20 to 11:25, the pressure of the containment vessel decreased to 0.155MPa[abs]. It is obvious that the decrease of the pressure was not attributable to venting the reactor containment vessel, but indicated a breach of the containment vessel.

---

[45] Air Photo Service Co., Ltd. (Permission acquired)
2. Efforts at Unit 5 to avoid accident

2.1 Relief valves had been disabled

On March 11, the reactor pressure vessel of Unit 5 had been under a pressure leak test for the next startup. For testing purposes, TEPCO had kept the safety valve functions of 3 of 11 SRVs available, while disabling the remaining 8 valves neither as safety valves nor relief valves. In other words, none of the 11 SRVs functioned as relief valves when the disaster broke out. The isolation valves between the nitrogen accumulator and the actuator cylinder were closed for each SRV, and the blow-down valves downstream were kept open. The reactor pressure vessel was filled with water at about 7MPa in pressure and 90°C in temperature.

Unit 5 also suffered from an SBO under these conditions due to the earthquake and tsunami. It was shut down in January 2011 for the scheduled refueling and maintenance outage. It still had massive decay heat as of March 11, and therefore the reactor pressure vessel rapidly accumulated pressure and exceeded 8MPa after 01:00 on March 12. The safety valve function of one of the three SRVs started after 01:40 when the pressure had reached 8.4MPa.

2.2 How to depressurize the reactor

In order to depressurize the hyper-pressured reactor pressure vessel, the operators did the following.

First, they decided to open the air-operated (AO) valve located on top of the reactor head for venting the reactor pressure vessel. The vessel had been filled with water to the top for the pressure testing. If the AO valve was opened, the water should drain to the drain drywell sump and the pressure should decrease. However, instrument air (IA) needed to operate the AO valve was depleted due to the blackout, and the AO valve could not be opened. According to an operator, IA depleted rapidly after the power had gone. Although there was a backup feature through the tie line with the station air (SA) system for on-site maintenance activities, its pressure dropped before long just like the IA system.

After they realized that the valves could not be operated by IA, the operators tried using high pressure nitrogen gas. There was a liquid nitrogen tank just outside of the reactor building, to make nitrogen gas through evaporation. Piping that continues into inside the reactor building merges with IA piping using a three-way directional valve.

[46] Air Photo Service Co., Ltd. (Permission acquired)
[47] a. to d. below are hearings with workers who were on-site at Fukushima Daiichi at the time of the accident.
In normal times, compressed air from IA piping is distributed via this directional valve to each AO valve installed in the plant. The operator at this time deliberately switched the flow direction by turning a wheel using an extension handle (wheel key) to let nitrogen gas flow instead of compressed air to the plant. As a result, the vent valve of the reactor pressure vessel, which was unable to be opened using IA, was opened successfully. After 06:00, the drain sump of the drywell became full and drainage stopped after the reactor pressure was reduced to approximately 2MPa.

To further reduce the pressure, it was decided to open the SRV manually. It required high pressure nitrogen in the actuator cylinder through accumulator to move the piston to open each SRV. The hatch to the containment vessel was kept open, but the reactor building was in complete darkness without lighting. An operator had to climb ladders in a narrow, hot and dangerous drywell to get to where the SRVs were. And because of the continuing threat of aftershocks, safety communication measures such as phone and paging systems were out of service. The operators decided to stop the malfunctioning fire alarm from ringing by unplugging its cable and using the alarm for signals instead. Three intermittent alarms, for example, signaled emergency, and meant the operators had to return. After confirming this and other temporary rules, a team of operators were sent to the SRV site. As described above, the isolation valve located downstream to the nitrogen accumulator to actuate the SRV was still closed at this moment, and the blow-down valve to the vent actuator cylinder was open. The team of operators going into the drywell needed to open the isolation valve and close the blow-down valve. After a while, the team was able to successfully manipulate the valves. Upon receiving that information, another operator in the main control room activated the SRV and successfully depressurized the reactor pressure vessel at around 05:00 March 14. In parallel, the makeup water condensate system (MUWC) pump had been recovered and ‘feed and breed’ became ready.

c. Power supply recovery and cold shutdown (1)
Neither of the two emergency diesel generators for Unit 5 was working. The air-cooled emergency diesel generator (B) of Unit 6, which had not been affected, was connected to different equipment using temporarily installed cables for recovery work. The actual power supply network was irregular, with cross-tie cable routing, and a backward current flow from low voltage switchboard (P/C) to high voltage switchboard (M/C), etc. Temporary cabling was also a labor intensive effort. The operators continued this struggle towards establishing a cold shutdown of Unit 5, just to face another, even greater obstacles.

The effort to reestablish a cold shutdown was twofold. First, a low pressure water injection pump was activated in the stand-by operation mode, and the reactor pressure vessel was sufficiently depressurized using the relief valve function of the SRV. Feed and breed helped to maintain cooling of the core. However, this was simply a transfer of decay heat from the reactor pressure vessel to water in the suppression chamber pool. The pool would eventually reach the boiling point.

A second step, therefore, had to be taken before the pool started to boil. A system had to be restored to transfer the heat of the suppression chamber pool and the decay heat in the reactor pressure vessel to the ultimate heat sink. The residual heat removal system (RHR), which was in place for this purpose, and the residual heat removal seawater system (RHRS) that emitted heat into the sea via the heat exchanger had been disabled. The M/C that supplied power to the RHR pump had been flooded by the tsunami, and the RHRS pump installed in front of the water intake had been destroyed by the tsunami.

In order to ensure feed and breed as the first step, water injection into the reactor pressure vessel had to take place using the MUWC to pump water from the condensate storage tank. To do this, power for the MUWC pump had to be recovered. The emergency diesel power generator (B) of Unit 6 supplied power at 6.9kV, but it was impossible to supply this power directly to the 480V MUWC pump. To solve this, existing cross-ties between M/C (6D) and its P/C (6D), and another between M/C (6C) and its P/C (6C) became ready.

[48] The “feed and breed” technique in this case involves pumping cold water into the reactor’s pressure vessel at approximately the same flow rate as that of steam leaving the reactor through SRV into the pressure suppression chamber so that water inventory in the vessel remains nearly constant.
C (6C) were turned on. P/C (6C) supplied power to the turbine building motor control center (MCC) (6C-1) and MCC (6C-2). Additional cabling was installed from MCC (6C-1) to the MUWC pump. With this cabling, the MUWC pump was activated on March 13 at 20:54. By the time the SRVs were opened and depressurization and makeup of the reactor pressure vessel started, the reactor temperature reached 170°C.

d. Power supply recovery and cold shutdown (2)
Recovery of the system to discharge heat to the ultimate heat sink, which is the second step to cold shutdown, required materials such as temporary pumps, hoses and fittings. They were transported and finally installed on March 18. Radiation contamination from the failed Units 1 to 3 affected the installation work.

A temporary cable was connected from the M/C (6C) to the RHR pump. A temporary pump was installed to replace the RHRS pump knocked out by tsunami. The temporary pump was connected to the temporary cable supplying supplied power from a generator truck. The temporary pump was activated on March 19 at 01:55, and the RHR pump was also activated at 05:00. Unit 5 established a cold shutdown thanks to these pumps on March 20 at 14:30.

3. Analysis and evaluation
a. Unit 1
(i) The appropriateness of decisions and actions regarding the operation of IC

In retrospect, the absolute priority during the station blackout situation was to assess the status of the IC and—if it were off-line—to return it to in-service status. When, by chance, the DC power supply temporarily recovered, the operators noticed that the IC was in off-line status, and tried to return it to in-service. However, this was most likely too late, as certain functions of the important IC system had been irreversibly lost by this time.

Presumably, non-condensable hydrogen gas had accumulated in the IC system's heat-exchanging tubes, hindering the natural circulation within the system even though there was sufficient amount of coolant left in the shell side of the IC cylinder. The hydrogen gas which blocked the circulation flow through the tubes supposedly developed by a chemical reaction of the steam and overheated fuel rods exposed by decreasing cooling water. The cooling water was supposedly lost due to venting of steam via the SRV to mitigate high vessel pressure or possible leakage of coolant from damaged piping.

Once the natural circulation is hindered, it becomes practically impossible to recover the functionality of the IC, due to its design. We believe that it is not so important to discuss the appropriateness of the decisions made and the actions taken by the operators regarding the situation, isolating IC again, because it would not have changed the outcome.

Nonetheless, the operators could not quickly assess the status of the IC system and recover its function immediately after the earthquake and the subsequent loss of DC power supply caused by the tsunami. This suggests that there might have been significant technical weaknesses in decisions made and actions taken, as follows: operators left the main control room to inspect the IC system at 17:19, more than one hour and a half after the IC operation status became uncertain if not lost; the main objective of the inspection was not to confirm the IC; the operators easily gave up inspecting the water level of the IC shell because of a small increase in the contamination level in the reactor building, despite the importance of the inspection; the operators had not thought of a possible situation where non-condensable hydrogen had accumulated in heat-exchange tubes and stopped the natural circulation in the IC system although they were taking action to makeup the cooling water in the IC shell assuming a possible loss of coolant; and the operators did not question the water level reading of TAF+2,000mm as of 21:19.

However, individual operators are not to be blamed for these weaknesses. The underlying problems lie in TEPCCO's organizational issues, such as the lack of nuclear safety preparedness against severe accidents—as exemplified by a lack of planning and implementation of adequate operator training, the lack of experience in activating the IC before during the normal operation or periodical inspection, etc.
See 2.2.4, 2 for a detailed discussion regarding the IC.

(ii) Possibility of avoiding the hydrogen explosion

Subsequent to the loss of the IC function, the situation of Unit 1 started to quickly deteriorate towards the possibility of core melt. The question is whether there were any means to avoid the hydrogen explosion. One method was to inject water by the HPCI automatically into the reactor, which was in a state of high internal pressure, but the HPCI had been disabled by the loss of DC power supply. Of all the damage caused by the tsunami, the loss of the DC power supply in this emergency situation was especially fatal.

Even if the DC power supply had not been lost and the HPCI had operated automatically, the plant would not have reached a stable state. As was later tried at Unit 3, adjusting the flow to maintain the water level inside the reactor would be difficult, and the injection pressure and water flow might have dwindled as the pressure of the reactor decreased. Moreover, a battery-operated DC power supply would have depleted sooner or later, resulting in a loss of control of the HPCI. The HPCI might have reached its limit to delay the development of the reactor.

After the loss of the IC function, it was very likely impossible to stop the hydrogen explosion at Unit 1.

(iii) Effect of and response against the short half-life radioactive elements

The situation at Unit 1 was unique in comparison to Units 2 and 3 in the very short time it had until core damage started. The operators working at the main control room and contract workers who were working in the open air to remove batteries out of vehicles and to install hoses and cables outside the building may have been exposed to short half-life radioactive iodine.

The isotopic effect from radioactive iodine on human health is insignificant after about 12 hours (in the case of I-134), or three days (in the case of I-135) after a plant shutdown. In fact, many TEPCO employees and contract workers were exposed to a severe environment, where radioactive elements had filled the reactor building and the plant site of Fukushima Daiichi, from the night of March 11 until the explosion of Unit 1 at 15:36 on March 12. There are still many unanswered questions, such as whether full-face masks and potassium iodine tablets were distributed and administered properly to every worker, and whether they were effectively used. It is hardly conceivable that adequate instructions were given and followed up, or that the detailed actions taken had been understood, considering the chaotic situation at the site. Furthermore, there probably was no chance at the time to inspect the radiation effects of the site in detail. By the time an investigation was performed and the workers started taking the whole body counter test, I-134 and I-135 had disappeared completely.

The issue of radiation effects outside the plant side needs to be considered as well. According to weather reports, the wind was blowing to the west at 15:00 on March 12, immediately before the explosion at Unit 1. Then the wind direction shifted to the northwest at 16:00, to the north-northwest at 17:00, to the north at 19:00, and to the north-northeast at 20:00. A monitoring post at the Onagawa Nuclear Power Plant, 116 kilometers north-northeast of the Fukushima Daiichi Nuclear Power Plant, exceeded the five micro-Sievert level per hour as of midnight March 13, and reached as high as 21 micro-Sieverts per hour as of 01:50 on the same day. This implies that the radioactive materials were carried from the Fukushima Daiichi plant at an average wind velocity of three to four meters per second. As the accident progressed, the decision to expand the evacuation area from a 10km radius from the plant to a 20km radius was made at 18:25 on March 12.

b. Unit 2

(i) Operation of the RCIC, what if it did not last long

The RCIC at Unit 2 remained in service for about 70 hours. It is presumed that all safety interlock functions for the RCIC were disabled due to the loss of the DC power supply.

There is a protective feature to stop the RCIC turbine automatically when the reactor water reaches the preset level (i.e. L-8 level) after water is injected by the RCIC pump. This automatic feature prevents excessive water injection and protects water from entering the steam pipes that drive the turbine. The feature also protects the
SRVs from being stuck in the open position. But the protective feature did not work. The reactor water level must have reached beyond the designed upper limit, and a large amount of water, together with steam, must have entered the steam pipes of the turbine. However, the RCIC kept working and cooled the reactor. It was also lucky that the SRVs did not stick open.

Eventually, the temperature of the suppression chamber water increased as well as the pressure in the exhaust pipes of the RCIC turbine. The protective feature should have been activated by the excessive pressure in the exhaust pipes to stop the RCIC turbine automatically, but the signal was not sent due to the loss of the DC power supply and the RCIC kept working.

While the loss of the DC power supply added many difficulties, it might also have allowed the operation of the RCIC in Unit 2 to last unexpectedly long.

Yet, the status of the RCIC was not known for sure, and no one knew when its operation might stop. To the same extent, exactly what eventually stopped the RCIC has not been determined. If the RCIC had not continued to work for 70 hours and had stopped much earlier, the development of the nuclear accident at Unit 2 would have overlapped with that of Unit 3, and might have made the accident response far more difficult. Under the same hypothetical conditions, radiation could have been released from Unit 2 much earlier, and the situation regarding radiological contamination could have been completely different.

The loss of the DC power supply and the availability of the RCIC invited discussions in the United States after 3.11. The RCIC is supposed to cease when the battery supplying electricity to the DC runs out. Thus, a “manual operation of the RCIC” was introduced as a part of B.5.b. to ensure continued cooling of the reactor core. However, there were doubts even at NRC about the feasibility of “manual operation.” In fact, the RCIC of Unit 2 kept operating consequentially without being instantly influenced by battery depletion and loss of the DC supply after it was last activated, even without “manual operation.”

(ii) Drop-off of the blow-out panel

Unit 2 never did explode. Considering the estimated reactor core damage and the development of hydrogen, an explosion like the ones at Units 1 and 3 was anticipated, but it did not take place. One hypothesis is that the drop-off of the blow-out panel helped to avoid an explosion.

The most likely cause of the drop-off of the blow-out panel was the shock from the explosion at Unit 1. There is no objection to this because a large area of the northern outer wall of Unit 2 seems to have been marked by debris from the explosion of Unit 1. Under different conditions, the blow-out panel would not have dropped and Unit 2 might have exploded.

A large portion of the radioactivity dispersed from the Fukushima Daiichi plant was found to have come from Unit 2. This fact makes an explosion seemingly unrelated to the amount of radioactivity released to the outside environment.

However, if an explosion had happened at Unit 2, it would have injured more workers in addition to those from Units 1 and 3, spread a large amount of highly contaminated debris, and hindered recovery activities. The situations at Units 1 to 3 would have deteriorated further, complicating the situation and making it too difficult to contain.

(iii) The cause and process of damage to the suppression chamber

(1) Mark I containment vessel—less durable against a design basis accident and a severe accident

Since the late 1970’s, some structural deficiencies, or the insufficient margin in the structural capability, of the Mark I type containment vessel have been brought up as important safety concerns in the United States, and necessary reinforcements were implemented at each plant as one of the backfitting tasks. Plants in Japan were also subjected to reinforcements in the 1980’s. Suppression chambers in this case had structural deficiencies in withstanding uneven and asymmetric impulsive dynamic loads during LOCA. The series of reinforcements implemented included enhancing

[49] Examples are melt-through from the reactor pressure vessel, a spread of radioactive aerosol created from the reaction of reactor debris and concrete, and a major breach to the containment vessel.
pipe penetration points where the strength margin was small and adding parts to mitigate the dynamic loads.

Severe accidents and design basis accidents such as LOCA are defined under different categories of accidents. The capability of containment vessels has been also discussed from the perspective of severe accidents—especially a nuclear accident resulting from SBO. As a matter of fact, an analysis report released by Oak Ridge National Laboratory in 1981 suggests that the suppression chamber could be damaged in a very short time in an extreme case such as an SBO accident where initial cooling by RCIC or HPCI fails.

As stated above, damage to the suppression chamber had been considered as a very realistic scenario under an SBO.

The following sections explain why the Unit 2 suppression chamber was possibly damaged.

(2) Reactor pressure vessel boiled dry, initial cooling became inefficient

The tremendously long-term operation of the RCIC at Unit 2 increased the temperature of the pool water in the suppression chamber. Obviously, the internal pressure of the suppression chamber increased accordingly.

The RCIC finally stopped at 13:25 on March 14 and the water in the reactor pressure vessel lowered to TAF by 16:30. The situation in the reactor started to change drastically thereafter.

The water level decreased further to the bottom of active fuel (BAF), which is 3,700 millimeter below TAF, and by 18:22, the reactor core had become completely exposed. Meanwhile, there was no water injection to the reactor pressure vessel, and the reactor pressure vessel boiled dry. The core started to melt from the center.

Seawater injection finally started at 19:54, an hour and a half after the exposure. Water at the bottom of the reactor pressure vessel seems to have evaporated due to the molten core slumped from the core support plate. Also, some of the high temperature residue that remained on the surface of the core support plate continued to generate radiant heat. Under these presumed reactor conditions, the water injection took place from 20:37 to 21:18.

In the beginning, water coming through the core spray system evaporated at the spargers. The water injection continued, and the spargers cooled down sufficiently. The liquid water was discharged from the spargers, but its flow rate was not high enough to spray water to the center of the reactor. Instead, water dripped from the spargers. When dripping water came in contact with the red-hot residue, it must have evaporated instantly and filled the gaseous phase of the reactor pressure vessel with super-heated steam instead of saturated steam. Steam created by the instant evaporation of the pumped in water raised the pressure and eventually the water injection stopped due to the developed pressure. By this time, the reactor coolant pressure boundary might have been significantly damaged and several leakage areas might have already existed.

Examples of the damage are as follows: The teflon coating applied to the metal O-rings used for penetration points of the in-core monitor housing and the CRD housing on the bottom head of the reactor, as well as on the bottom flange of the CRD housing had deteriorated; water in the suction pipes of the primary loop recirculation system had been pushed out by the pressure of the reactor pressure vessel into the containment vessel through the mechanical sealing of the pump shaft; the bonnet flanges and the gland packing on the valves constituting the pressure boundary had already lost their sufficient sealing functions; even the bolts fastening the head of the reactor pressure vessel might have crept and loosened from the high-temperature, causing the metal O-ring of the vessel head flange to lose its sealing function; debris slumped from a hole of a melting shroud cylinder might have damaged piping surrounding the reactor pressure vessel cylinder.

The soundness of the so-called reactor coolant pressure boundary had probably deteriorated remarkably by this point.

Water injected into the reactor pressure vessel evaporates instantly and immediately stops the pumping operation due to high-pressure steam. The high pressure steam eventually leaks into the containment vessel through the leakage points, lowering
the pressure inside the reactor pressure vessel while the pressure in the containment
vessel increases. Once the reactor pressure vessel is depressurized, water is injected
again. This cycle is repeated until the reactor pressure has been sufficiently cooled and
depressurized. The situation of Unit 2 followed changes similar to those described in
the above scenario over a long time period.

(3) Large scale failure of the suppression chamber

Increased pressure in the containment vessel pushes out atmospheric gas into
the suppression chamber. When water reaches an excessive temperature in the sup-
pression chamber, it no longer condensates vapor, but creates steam bubbles on the
surface, causing intermittent or continuous vibrations in the suppression chamber in
combination with an increase of internal pressure. The situation is as severe as having
both a pressure test and seismic test occur simultaneously. Under such circumstances,
a large scale breach or a burst can take place anywhere and at any time. After repeti-
tious attempts to vent failed at Unit 2, a burst is presumed to have occurred at 6:00 on
March 15.

c. Unit 3

(i) The effects of the long operations of the RCIC and the HPCI

An intense burst of white steam rose from the top of Unit 3 reactor building after
it exploded. In the wake of the explosion, Self-Defence Forces started dumping water
from helicopters.

It is estimated that the inside of Unit 1, which also exploded, was in a similar situation
at the time of its explosion. However, Unit 3 had a large-scale release and lost the means
to contain it because the containment vessel had been exposed to considerably high
 temperatures and pressure for a long time before the reactor was effectively cooled.
The prolonged RCIC and HPCI operations implicitly had these effects.

(ii) The DC power supply survived, but the accident was not avoided

The DC power distribution panel of Unit 3 was not flooded and the SRVs and the
air operated valves of the vent system as well as the HPCI were operable until 02:42 on
March 13. However, in the confusion, the lucky situation was not effectively utilized.

On a later day, a foreign BWR plant operator released a paper assessing operational
responses that could have been taken in order to contain the accident at an early
stage. In essence, the reactor should have been quickly depressurized below the
outlet pressure of the water injection pump, and water should have been instantly
injected to reflood the reactor pressure vessel, instead of focusing on maintaining the
water level in the reactor.

Only Unit 3 had DC power supply, which would have enabled operation of the SRV,
so the suggested procedure would only have worked for this unit. However, allocation
of the disaster relief resources from both inside and outside the plant Unit 3 was not
as highly prioritized as Units 1 and 2, which had lost all electricity sources. All of the
fire trucks for water injection activities were assigned to Unit 1. It was also difficult to
carry out tasks in an efficient time frame due to the complicated requests for disaster
relief and the disruption of the traffic network. The opportunities to contain the acci-
dent were not fully leveraged.

Eventually, the reactor water level started to decrease. The DC power supply was still
available when the HPCI started automatically at 12:35 on March 12, but it ran out at
02:42 on March 13. For all practical purposes, this made the condition of Unit 3 the same
as Units 1 and 2. Forced depressurization and venting of the containment vessel of Unit
3 became difficult. Later, venting was performed successfully, playing an indispensable
role in recovery activities, but it induced an explosion of the Unit 4 reactor building.

[50] Chunkuan Shih, Tsong-Sheng Feng, Kai-Chuen Huang, Chin-Cheh Chang, Jong-Rong Wang., “On RPV
Depressurization Strategy and Alternate Water Systems in SBO of Nuclear Power Plants,” Transactions of the

[51] Hearing with the workers who were on-site at Fukushima Daiichi at the time of the accident
(iii) Orange flame from the explosion and high-dosage debris

A close look of the actual footage of the explosion at the Unit 3 reactor building at 11:01 on March 14 indicates a flash of orange light instantly before the explosion. Then dust burst out of the roof, rising as high as 500m. The explosion blew off some large sections of concrete, and one of them allegedly fell on the roof of the turbine building, making a hole as large as a fire truck. The surface of the hole shows many steel reinforcement bars bend downward.

There were approximately 40 tons of zircaloy in the Unit 3 reactor, of which 25 tons were in the fuel claddings and 15 tons in the channel boxes. If all the zircaloy reacted with water, the amount of hydrogen created would be about 2,000 kilograms or 20,000 standard cubic meters. The calorific value would be about 280GJ, an equivalent of about 58 tons of trinitrotoluene (TNT).

However, not all of the zircaloy in the reactor actually reacts with water. According to the analysis report by Oak Ridge National Laboratory, only about 20 percent of the zircaloy in the reactor is estimated to react. Venting took place several times from March 13 until the time of the explosion, as evidenced by the sudden increase in the radiation readings at the monitoring posts. However, venting let hydrogen flow into the Unit 4 reactor building, causing it to explode.

It is questionable whether the zirconium-water chemical reaction in the Unit 3 reactor could create enough hydrogen to cause this result.

As mentioned earlier in this report, the CCI phenomenon creates an enormous amount of steam, hydrogen, carbon monoxide, and carbon dioxide together with radioactive aerosol. Based on the hypothesis that CCI has occurred, the volume of the explosive gas should be larger than the seemingly small estimate above. Also, the orange flash of light observed immediately before the explosion can be explained as an imperfect combustion of the carbon monoxide contained in the explosive gas.

Aerosol created by CCI contains highly concentrated radioactive material. Based on this, it can be presumed that the debris scattered by the explosion had high radiation readings because of CCI. The explosion of Unit 3 can be explained logically taking into account the contribution of CCI, which possibly means that the melt through of the reactor pressure vessel and erosion of concrete in the pedestal may have progressed on a much larger scale.

d. Unit 4

(i) Why the Unit 4 reactor building exploded

The explosion at the Unit 4 reactor building was caused by the back-flow of hydrogen from Unit 3 through the SGTS to the Unit 4 reactor building, which created an explosive atmosphere inside the reactor building. The explanation for the explosion is that something must have caught fire, causing the hydrogen to explode. [52]

This is only a presumption, however, not a proven fact. Further analysis, discussion and verification of this are desired, as there is a lack of evidence that the back-flow of hydrogen alone could have created an explosive atmosphere in the reactor building of Unit 4.

(ii) Why the pessimistic speculation regarding Unit 4 spent fuel pool came up

There was much speculation about the explosion that caused the major damage at the reactor building of Unit 4, and the white smoke from the spent fuel pool that continued immediately after the explosion. NRC advised US citizens in Japan to evacuate from areas within 50 miles from the plant, and later released a statement confirming that there was an internal document pointing out the possibility that the hazardous zone might have to be expanded to the Tokyo metropolitan area. It was later confirmed that the spent fuel pool was filled with a sufficient amount of water, and as a result, the speculation ceased. The cause of this negative speculation was a lack of confirmed information at the initial time and the following technical reasons:

• There was no water level gauge or surveillance camera in the spent fuel pool to monitor its status.

• The strong earthquake and the explosion were enough to cause concerns about

It was difficult to analyze the information regarding the radiation dose accurately due in part to the explosion at Unit 3.

There was no significant insight and research regarding the phenomena of a zirconium fire. There was no analytical tool to practically assess the situation.

The arrangement of hot spent fuels in a checkerboard pattern had not even been considered in Japan, whereas it has already been put in place in the United States. It was thought that there was a possibility that the hot spent fuel assemblies were concentrated locally in the pool.

The B.5.b measures had not even been discussed in Japan, whereas they had already been introduced in the United States. Consequently, there was no spent fuel pool cooling system using external water sources.

(iii) If there had been no leak from the reactor cavity to the spent fuel pool

The negative speculation about Unit 4's spent fuel pool ended with the confirmation of the fact that there was a sufficient amount of water still left in the spent fuel pool. But this, in turn, raised a new question. The water was there because of the structural character of the spent fuel pool's gate. The water that fully filled the reactor cavity and the water in the dryer separator pit connected to the reactor cavity had flowed into the spent fuel pool when its water level decreased from evaporation. This explanation makes sense logically and matches with the fact that the water levels in the reactor cavity and the dryer separator pit had decreased.

The water levels of the reactor cavity, the dryer separator pit, and the spent fuel pool are maintained at an equal level only during a scheduled refueling outage, which constitutes no more than 10 to 20 percent of the entire operation cycle. Therefore, the influx of water from the other pools in the event of loss of cooling function of the spent fuel pool is a hypothetically aggressive expectation. For a normal scenario in which the spent fuel pool remains uncooled for a long time, water in the spent fuel pool would be the only water that can be assumed. Based on this, it is estimated that the water in the spent fuel pool would evaporate completely sooner or later.

There is another issue. Some previous technical literature indicates the possibility of temperatures reaching a point high enough to cause a zirconium fire under certain conditions. In this situation, the zircaloy fuel cladding is compromised and a large amount of radioactive material in the fuel cladding is released by heat to the outer environment. Outside Japan, analytical codes have been developed to accurately assess whether such an event could pose practical concerns, and have been tested and verified. [53]

e. Unit 5

(i) If Unit 5 were in normal operation

Unit 5 was able to avoid a state of crisis when the effort to incorporate the electric power supply from Unit 6's air-cooled emergency diesel generator (B) succeeded at 08:13 on March 12. It achieved a cold shutdown after intermittent adjustments of the water level by operating the SRV and MUWC.

Was the accident inherently avoidable? Hypothetically, if Unit 5 had been in normal operation, the water level and pressure of the reactor at the time of the accident and the decay heat that started to develop immediately after the accident would have been at similar levels to Units 1, 2 and 3. The conditions at Unit 5, where all AC power and the heat discharge route to the ultimate heat sink were lost and the adjacent reactor was out for a scheduled refueling, were the same as Unit 3, which eventually experienced a reactor accident.

Accordingly, high pressure water injection would be performed to prolong the life of Unit 5. However, the DC supply could have been lost prior to accommodating electricity from the Unit 6, and the containment vessel might not have cooled or depressurized enough to allow low pressure flooding to finish before the high pressure water injection reached its limit. In that case, Unit 5 faced the risk of reactor excursion, following a similar pattern of accident development as the Unit 3.

[53] The underlying concern is about the intentional destructive attacks including those by terrorists, and is not about a loss of cooling system due to a loss of power supply or about a breach of the spent fuel pool by earthquake.
2.1.4 Process of radiation release based on the parameters of the reactor

The accident is a “severe accident” in the sense that the reactor cooling functions were lost and the cores melted. The accident, however, became a catastrophic accident that went beyond the assumptions of a severe accident, because a large amount of the radioactive materials emitted from the molten cores were released to the environment and the core melt downs became uncontrollable due to the loss of the containment vessel and reactor building functions.

The process of releasing the core-derived radioactive materials will be assessed in this section according to the estimated progress of the accident, which is based on the parameters obtained from the accident site at the Fukushima Daiichi Nuclear Power Plant. Because the exact present condition of the plant damaged by the accident is not yet completely understood and an adequate analytical model cannot be created, neither TEPCO nor NISA has recreated large-scale complex simulated analysis of the process of the radioactive material emissions from the Fukushima Daiichi plant. On the other hand, it is possible to evaluate logically, although qualitatively, the emission of the radioactive material by appropriately analyzing and assessing the data retrieved by the reactor operators who were on duty at the time of the accident. In addition, this section highlights some of the key facts in the development of the accident, including some lucky coincidents that prevented further escalation of the accident.

1. Large amount of radioactive materials measured by a radiation monitor

Figure 2.1.4-1 depicts the radiation dosage measured within the boundary of the Fukushima Daiichi plant using a monitoring car. Radiation measured on March 12 was emitted from Unit 1, which had a core meltdown starting late at night on March 11. The radiation measured on March 13 and 14 also included radiation emitted from Unit 3, which had a core meltdown on March 13. After March 15, a higher radiation dosage was measured near the front gate. The increase is considered largely attributable to the radioactive material released from Unit 2.

The radiation dosage readings from March 12 to 14 show that the radiation at MP4 to the northwest of Units 1 through 4 was about 10 times higher than the readings in the southwestern and the southern directions. The difference implies that the radiation dosage depends heavily on wind direction. Also, the high radiation dose measured on March 15 to 16 near the main gate shows several peak dosage times followed by an even higher dosage. The source of this high radiation is inferred as Unit 2, judging from the accident’s progress, stated in 3 of this section.

The main causes for the apparent differences in radioactive material released between Units 1 and 3 and Unit 2 are the wind direction and the activation of the suppression chamber (S/C) vents of the containment vessel. The S/C vents of Units 1 and 3 enabled the containment vessel to be depressurized significantly. On the other hand, as the containment vessel of the Unit 2 had not been vented, the pressure in the D/W continued to stay high at 0.6 to 0.7 MPa, and damaged the containment vessel at an early stage of the accident.

2. Development of the accident at Unit 1 up to the release of the radioactive materials

The symptoms of Unit 1 were simply the loss of all electrical power and the subsequent failure of the cooling system. Presumably, it took the shortest course from the meltdown to the breach of the reactor pressure vessel, and the slump of the molten core to the bottom head of the containment. However, there is little data on the reactor parameters between late night of March 11 and early morning of March 12 amid the confusion.
immediately following the earthquake and tsunami. Therefore, we referenced the station blackout simulation[56] created by the Oak Ridge National Laboratory, which was commissioned by NRC, because of the many similarities. (See Figure 2.1.1-2 to 4)

a. The reactor boiled dry from the functional loss of the reactor cooling system in the absence of all power

(i) Nearly 100 percent of the volatile radioactive materials were released by the core meltdown and molten core

When the core cooling function ceased, the core temperature increased and the reactor water level descended to the top of the active fuel in about two hours and a half. The zirconium-water reaction started to progress rapidly in four hours, and the core started to melt four-and-a-half hours after the loss of all electric power. As the core meltdown progressed, the temperature of the molten core increased to 2,500 degrees Celsius or higher. Almost 100 percent of the volatile radioactive materials, such as noble gas, iodine, cesium, and tellurium evaporated[57] from the molten core and were released into the steam phase within the reactor pressure vessel.

(ii) The reactor pressure vessel breach, and high temperature and high pressure gas leaks from the reactor to the D/W of the containment vessel

The core meltdown later caused damage near the bottom head of the reactor pressure vessel. This incident presumably occurred at 02:45 on March 12, when the pressure readings of the reactor and the D/W were equalized as shown in Figure 2.1.4-2. High temperature and high pressure volatile radioactive materials transferred from the reactor pressure vessel to the D/W of the containment vessel. The transfer was supposed to be represented by a rapid increase in the readings of the aerial radiation monitors D/W (CAMS) and S/C (CAMS) in the containment vessel D/W and the suppression chamber, respectively. But the radiation dose monitoring in the containment vessel of Unit 1 only started from March 14 (see Figure 2.1.4-2), so the outflow of radiation that took place in the early morning of March 12 cannot be directly observed.


(iii) Hydrogen and radiation leak from the containment vessel D/W to the reactor building, and the hydrogen explosion

The high temperatures and high pressure steam released to the containment vessel D/W exceeded the design basis temperature and pressure of the containment vessel, and deteriorated the packing of the flange and the service entrance of the containment vessel. Hence, its airtightness was breached. The hydrogen explosions at Units 1 and 3 are evidence of the massive leakage of hydrogen, radiation, and vapor from the containment vessels to the reactor buildings due to the breach of airtightness. The radiation that leaked to the reactor building was released to the external environment by the hydrogen explosion.

Neither the leak of radiation and hydrogen into the reactor building due to the breach of airtightness of the containment vessel nor the hydrogen explosion had been postulated even under the severe accident case.

(iv) Molten core slumps to the containment vessel floor

The bottom of the reactor pressure vessel of Unit 1 was damaged by 02:45 on March 12. It is estimated that it took about one hour for the majority of the liquid highly dense molten core to fall to the floor of the containment vessel as a result of the widening rupture. According to the estimate, some of the fallen molten core spread horizontally from the opening of the pedestal because of its liquidity, while the majority of it thermally decomposed concrete and moved downward. Yet, the whereabouts and the condition of the majority of molten fuel that has allegedly fallen to the containment vessel floor remain completely unknown to date.

(v) Depressurization of the containment vessel by the containment vessel's S/C vent

The pressure of the containment vessel D/W of Unit 1 had been higher than about 0.7MPa[g] since 01:00 on March 12, far exceeding the design basis, as seen in Figure 2.1.4-2. The containment vessel was in jeopardy of a breach. Despite the very poor working conditions, the workers successfully opened the containment vessel S/C vent at 14:30. The pressure of the D/W decreased significantly, and the breach of the containment vessel was avoided at this early stage of the accident. However, the venting operability had not taken any of the protective measures against high radiation dosage in an accident situation into account. and it ended up taking more than 13 hours to complete the mission. The significant delay in venting became one of the factors that led to the failure in preventing the Unit 1 hydrogen explosion and the release of radiation to the external environment.

b. Cooling corium on the containment vessel floor by spraying water using fire trucks

The pressure in the reactor decreased to about 0.8MPa[g] at 02:45 on March 12, enabling water injection using fire trucks. However, the destruction at the facilities
caused by the earthquake, the tsunami and the consequential confusion hampered the preparation of water injection. Water injection started at 05:46 on March 12, but only a small quantity—about 1 ton per hour—was injected by 7 o’clock. Without the water injection, the corium would have retained its high temperature, melting through the bottom of the containment vessel and directly contacting the underground water.

c. Objectives, background, and significance of installing reactor coolant injection system using fire trucks

Water injection using fire trucks was the only useful method employed to cool the corium, and contributed to reaching the present “seemingly stationary condition.”

By 2002, the Fukushima Daiichi plant had installed a system for injecting water into a reactor using a fire extinguishing spray system with water from filtrate tanks operated by electric pumps and diesel pumps. A water filler to this reactor water injection system using fire trucks was installed in June 2010, about nine months prior to the accident.

The installments were originally intended to enhance the fire-fighting and fire-extinguishing facilities at the Fukushima Daiichi plant,[58] but did not take an accident of this scale into consideration. From this perspective, it may seem like pure luck that this water injection system became a factor. In fact, the system was installed to assure redundancy and diversity, so it cannot be simply attributed to pure luck.

d. An enormous amount of highly contaminated water leaked into the basement of the reactor building

Coolant water injection to cool the corium is continuing at the present time, some 15 months after the accident. The location and condition of the corium and the process of the cooling are still not known, but the water injection is continuing. The coolant water that is flowing out to the reactor building contains radiation of presumably almost all of the cesium and about 5 percent of the strontium contained in the core. Details are in Chapter 4.

3. Development of the accident at Unit 2 until the release of radiation discharge

a. Continuous operation of RCIC and its shutdown

The RCIC of the Unit 2 did not stop and continued to operate even after the loss of the off-site power supply.

As illustrated in Figure 2.1.4-3, the water level in the reactor started to decrease while the pressure started to increase by 10:00 on March 14, more than three hours prior to the shutdown of the RCIC. This phenomenon suggests the possibility of the loss of the RCIC core cooling system function.

Figure 2.1.4-3: Reactor depressurization after the shutdown of the RCIC, the SR vent release, the reactor water level decrease at the Unit 2

---

b. Releasing the SRV and the decreasing reactor water level
While an effort was made in the main control room to release the SRV to rapidly reduce the pressure in the reactor pressure vessel, the batteries were not ready. The SRV was finally released after five or six hours, while the reactor water level continued to drop, because considerable time was needed to prepare enough batteries amidst the chaos from the hydrogen explosion at Unit 3. When the reactor pressure started to decrease from the opening of the SRV, the reactor water level lowered to the middle of the reactor core and reached BAF soon after. The core lost coolant water.

c. Reactor pressure vessel breached, the D/W pressure increased, and radiation flow from the reactor to the D/W and the reactor building
As seen in Figure 2.1.4-4, the pressure in the D/W started to increase after 19:00 on March 14, and the pressure in the D/W and the reactor equaled at around 21:00. This indicates a breach of the reactor pressure vessel. Because only a small amount of time had elapsed since the core started to melt, the breach is estimated to have occurred at piping connected to rather than at the bottom of the reactor pressure vessel. At this time, the radiation dosage reading in the D/W started to increase, indicating the start of radiation outflow from the breached reactor pressure vessel to the D/W. Thereafter, the pressure readings of the reactor and the D/W became identical and increased together. They reached and stayed high at 0.6 to 0.7 MPa[g] for more than seven hours, far exceeding the design basis pressure, which is 0.427 MPa[g]. The radiation within the reactor had transferred into the D/W during this period, represented by a rapid surge of the radiation dosage in the D/W.

Hydrogen, iodine, cesium, tellurium and other radioactive gases leaked from the high pressure and high temperature containment vessel flanges to the reactor building. This phenomenon is identical to Units 1 and 3.

During this period, the pressure of the S/C was following a decreasing trend, unlike that of the D/W, and indicated -0.1 MPa[g] by around 06:00 on March 15. It is thought that this decline is a result of the failure of the pressure gauge, due to the breach of the containment vessel.

d. Sudden decrease of the D/W pressure following the breach of the containment vessel D/W
A significant drop in pressure from 0.65 MPa[g] level in the D/W and the reactor was observed from 07:00 to 11:00 on March 15. The pressure of the D/W dropped to the atmospheric level. This sudden decrease in pressure infers a relatively large rupture somewhere in the D/W. In other words, the airtightness of the containment vessel was breached, and a large, highly contaminated gaseous body was released to the reactor building in a short time. The pressure readings were collected only twice during the four hour period due to the effect of the hydrogen explosion at the Unit 4.
e. Meltdown of the debris and breach of the bottom of the reactor pressure vessel

Immediately after the decrease of the D/W pressure, the pressure readings of the reactor and the D/W spiked synchronously. The pressure in the reactor turned around from a sharp drop to a sudden rise to 0.65 MPa[g], the level before the depressurization, and soon dropped sharply again. The peak pressure of the D/W was about half of the pressure spike of the reactor, but their curves synchronized.

The sudden surge and plunge in reactor pressure readings indicates the development of a very large amount of steam inside the reactor in a short time. This was released to the D/W, and then to the reactor building via a breach of the D/W. It is estimated that a large amount of steam was created by the meltdown of core debris at the bottom of the reactor pressure vessel, which still had coolant water that had contacted with the meltdown debris. As a result, it is thought that a relatively large new rupture was created at the bottom of the reactor pressure vessel.

The large body of steam created in the reactor contained a large amount of volatile radioactive material that had evaporated from the core meltdown. A gaseous body of such was released to the D/W from the reactor. As the pressure inside the reactor decreased towards an atmospheric level, the radioactive gas released decreased, as evidenced by the gradual decrease in the readings of the D/W dosage after reaching the maximum value of 138 Sievert per hour.

Even when the pressure in the reactor and the D/W decreased nearly to the atmospheric level at around 05:00 on March 16, the radiation dose of the D/W decreased only to 100 Sv/h and more than two thirds of the radiation remained in the containment vessel. The decrease of the dosage was mainly due to a decline in Xe-133. Only about 1 to 5 percent of the cesium and iodine were emitted to the atmosphere. Most of them, it is estimated, still remain in the containment vessel.

f. Actual condition of the accident analysis by TEPCO using a code “MAAP”

TEPCO released the most recent results of its analysis of the nuclear reactor accident using the analytical code “MAAP” on March 12, 2012. [59] There is a fundamental problem in the analytical method presented in the release that needs to be pointed out. Figure 3-3 on page 30 of the report shows the results of an analysis of the changes in the containment vessel pressure. Likewise, Figure 3.2.2.2 on pages 13 to 18 of the Appendix of the same report shows a simulation of the reactor pressure vessel pressure. These pressure data correspond to Figure 2.1.4-3, 4. The result of the analysis by TEPCO indicates a moderate decrease in the pressure, but ignores the pulse-like behavior of the pressure readings that have been commonly observed both in the reactor pressure vessel and the containment vessel. Such a pressure curve does not provide the estimated release of a large amount of radiation-contaminated steam to the reactor pressure vessel, the containment vessel, and the reactor building. It does not provide visible reference of the large amount of steam developed in the reactor pressure vessel. A fluid flux is indicated by the pressure changes, which are the differentials of the pressure levels. When the changes are averaged and expressed in moderate curves, the actual drastic changes of the fluid flux are represented in the averaged values. The significance of the projection using a complex and expensive simulation code remains questionable.

g. Reason why Unit 2 did not have a hydrogen explosion and the course of radiation release from the reactor building to the environment

It is estimated that a large amount of radiation and hydrogen transferred into the reactor building was released to the external environment through the large opening of the blowout panel on the fifth floor of Unit 2.

Workers attempted to open the blowout panel at Unit 3 in order to prevent a hydrogen explosion, but failed and were unable to prevent the explosion. On the other hand, there is a photograph from March 12 showing the opened blowout panel of Unit 2. It is assumed that the blast from the hydrogen explosion of the Unit 1 reactor building on
March 12 opened the blowout panel of Unit 2.

It might have been a coincidence that the blowout panel was opened, but it not only helped to prevent a hydrogen explosion, but also resulted in the immediate environmental release of the radiation leaked to the reactor building, and mitigated the amount of radiation to the external environment, according to estimates.

The main objective of a blowout panel is to prevent destruction of the reactor building from a rapid increase in internal pressure in the case of a mass amount of steam flowing out to the reactor building or turbine building from events such as a rupture of the main steam pipe.

**h. The containment vessel vent did not function**

Depressurization of the Unit 2 containment vessel was attempted three times by configuring vent lines, but none was successful. The containment vessel venting did not succeed.

### 4. Development of the accident at Unit 3 up to the radiation release

**a. Activating and stopping the core cooling system subsequent to the SBO at Unit 3**

Subsequent to the SBO at around 15:40 on March 11, the RCIC of Unit 3 was activated at 16:03 using the surviving DC power supply. However, the RCIC was stopped at 11:36 on March 12. Because the reactor water level had lowered, the HPCI was automatically activated at 12:35 on March 12. Subsequent to the activation of the HPCI, the reactor pressure that had been at 7.5MPa[g] was reduced to 4.8MPa[g] at 13:05, 30 minutes after the activation of the HPCI. It continued to decline to 3.5MPa[g] by 14:25, to 0.8MPa[g] by 20:00, and to 0.58MPa[g] by 02:42 on March 13—the time the HPCI stopped (see Figure 2.1.4-5). The reactor pressure went up again to 4.0MPa[g] in the hour after the HPCI stopped, and to 7.38MPa[g] in two hours.

**b. Rapid drop of the reactor pressure from the release of the SRV and rapid increase of the containment vessel pressure**

The reactor pressure rapidly declined from 7.3MPa[g] to 0.46MPa[g] at 08:55 on March 13 upon the opening of the SRV. Simultaneously, the D/W pressure rapidly exceeded the design basis pressure level, due to an inflow of high temperatures and high pressure coolant, and rose as high as 0.537MPa[g], almost equal to the reactor pressure.

**c. Depressurization of the containment vessel by S/C venting and the outbreak of intense steam at the core**

The D/W pressure exceeded the design basis pressure of 0.427MPa[g] and became very high, as illustrated in Figure 2.1.4-6. The containment vessel S/C was then immediately vented, and the D/W pressure rapidly decreased. The vent valve was “open” but was unstable, and closed...
before long. The vent valve was reopened, but it closed again shortly thereafter. This process was repeated five times. Each open/close is marked with a red number in Figure 2.1.4-6.

During this period, rapid changes occurred in the reactor pressure and the D/W pressure. Rapid increases in these values indicate the development of a large amount of steam at the core and, more likely, the core meltdown. Sudden drops in pressure in the reactor and in the D/W during the venting clearly suggest the significant effect of depressurization as a result of venting. The role of the containment vessel S/C venting performed during this period was certainly very significant in lowering the containment vessel pressure. As shown in Figure 2.1.4-6, Unit 3 was repeatedly depressurized, and was never exposed to pressure higher than 0.6 MPa[g] for a long time, unlike Unit 2.

At 11:01 on March 14, immediately after the venting was performed for the fourth time, a hydrogen explosion occurred. This means that a large amount of hydrogen, radiation, and steam had been transferred out to the reactor building, and that it was now being released directly to the external environment.

The exact time when the reactor pressure vessel was breached remains unknown, because there is no data available on when the radiation dose at the D/W started to rise. However, it is estimated to be around the time when the venting of the containment vessel was performed for the third time, based on the fact that the radiation dose of the D/W reached 168 Sievert per hour at 04:00 on March 14.

Figure 2.1.4-6: Sudden drop in pressure and intensive outbreak of steam by containment vessel venting at the Unit 3

2.1.5 Efforts to prevent accidents and the accident risk at other nuclear power plants

The Fukushima Daiichi Nuclear Power Plant was not the only plant stricken by the Great East Japan Earthquake. Other nuclear power plants such as Fukushima Daini, Onagawa, and Tokai Daini, also incurred immense damage. They were significantly affected by the earthquake and tsunami to the extent that they would have attracted more attention from interested parties if the Fukushima Daiichi plant accident had not occurred. It should be emphasized that these nuclear power plants could potentially have suffered nuclear accidents if the damage, the effects and the emergency responses against the earthquake and tsunami took a different turn.

This section provides observation and assessment of the emergency response taken mainly at the Fukushima Daini plant, and includes a summary of the accident risk at the Onagawa and Tokai Daini Nuclear Power Plants.

1. Fukushima Daini Nuclear Power Plant
   a. Main damage and the effects
      (i) Damage induced by the earthquake and the consequences
Out of four off-site lines—namely, the two 500kV Tomioka lines and two 66kV Iwaido lines—three lost their transmission capabilities.[60] The Fukushima Daini Nuclear Power Plant barely managed to maintain the power transition line of 500kV Tomioka line #1, which helped the plant to avoid the total loss of its off-site power supply.

Specifically, the system side of a porcelain insulator disconnect switch of Tomioka line #2 had been damaged. Iwaido line #1 was under an inspection and maintenance program when the disaster broke out. Its power transmission was suspended along with that of Iwaido line #2, which was intact but needed to repair its lightening arrester of the transformer.[61]

(ii) Damage induced by the tsunami and the consequences
Nine out of 12 emergency diesel generators, two out of 36 M/Cs and eight out of 36 P/Cs of the on-site power distribution systems, and seven out of eight RHRS pumps[62] lost capabilities directly and indirectly due to flooding by the tsunami.

The breakdown and unavailability of the equipment and facilities badly affected the process of accident recovery. The summary of efforts to prevent accidents at the Fukushima Daini plant is stated below.

b. Summary of efforts to prevent accidents
Units 1 through 4 of Fukushima Daini Nuclear Power Plant had been operating at the constant rated thermal output when they were all shut down through scram operations at 14:48 on March 11 in response to a signal from the seismic accelerometer that had recorded the “high acceleration of the Great East Japan Earthquake.” Although three off-site power transmission lines lost their transmission capabilities, an off-site power supply was barely secured by maintaining Tomioka line #1. At 15:22, the first wave of tsunami reached the Fukushima Daini plant from the southeast, inflicting damages and affecting the site.

Activities were undertaken at each unit to overcome the crisis, as follows.

First, the water level and pressure of the reactors were controlled by RCIC. Meanwhile, water injection to the reactors using external water sources was prepared as a next step measure. This was done under time constraints because the temperature of the suppression chamber pool would rise due to the success of RCIC, and depressurization of the reactor would become more difficult if it progressed excessively, which would make any subsequent low-pressure water injection even more difficult.

The MUWC pumps were in good condition except for the pumps of the Unit 1 A and C systems, which became unusable as their power sources were submerged. The shift supervisor at each unit made the operation staff start the MUWC pumps according to the emergency operating procedures (EOP), and then depressurize the reactor by opening SRVs, and started to gradually switch reactor water injection of the Units 1 through 3 from the RCIC to the MUWC after 03:00 on March 12. Only Unit 4 did not use the MUWC, instead using a high pressure core spray (HPCS) which later was succeeded to the residual heat removal operation using the RHR system. This was because the HPCS was available and used, for its wide operational range from high pressure to low pressure spraying, to replace the RCIC.

Once the water injection method was switched from the RCIC to the MUWC or the HPCS, the reactor cooling proceeded to the next step. However, it had not yet reached a sustainable situation because the accumulated heat in the reactor and the containment vessel had to be transferred to the sea, which was the ultimate heat sink, using the RHR. Activities to restore the ultimate heat sink at each unit will be stated below.

(i) Unit 1
All three emergency diesel generators were destroyed, and two M/Cs, (C) and HPCS, were lost. However, one switchgear of the M/C (D) was intact. Because undamaged
components belonged to the subsystem-B, it was decided to restore other subsystem-B components—the RHR pump (B), the EECW pump (B), the RHRC pump (D), and the RHRS pump (B)—and to immediately procure replacement motors for these pumps. Unlike the RHRC or the RHRS, the EECW pump had not been made redundant, and the only EECW pump (B) belonging to the subsystem-B was submerged and needed to be replaced. Even more difficult was the fact that the 480V P/C power panel for the pumps was disabled. After all, many of the P/Cs at the Unit 1 were disabled by submersion, so it was decided to obtain the power supply from the P/C of the radwaste building, which was not damaged.

Despite facing many difficulties in mounting the motor, which was procured for the RHRC pump (D) and to install the cables to restore the RHRS pump (B), the pumps were eventually restored on March 13.

The last pump remaining out of action was the EECW pump (B), which was disabled because the motor had been submerged. A generator truck was deployed, a transformer and cable were installed, and the EECW pump (B) was reactivated on March 14.

The RHRC pump (D), RHRS pump (B) and EECW pump (B) were restored in addition to the RHR pump (B) – the only pump that had not been damaged. A complete train was configured using a minimal combination of equipment. At 1:14 on March 14, the reactor cooling was finally shifted from the MUWC to the RHR residual heat removal operation.

(ii) Unit 2

Three emergency diesel generators and all of the three M/Cs remained intact. However, P/C (A) and (B) were flooded and EECW pump (A), RHRS pump (A) and (C).

[63] TEPCO document. Note that the systems of the Units 2 through 4 at Fukushima Daini are almost the same as that of the Unit 1.
which all belong to the subsystem-A, were disabled. Hence, the restoration team chose to restore the subsystem-B, which had suffered relatively little damage, and cables to supply power to the pumps needed to be laid out.

Like Unit 1, a several-hundred-meter-long cable was extended and installed from the radwaste building P/C to the RHRC pump (B) and RHRS pump (B).

It was decided to supply the power to the remaining EECW pump (B) from a cable stretching from a spare outlet of the P/C of the adjacent Unit 3 that was unharmed. The benefits of doing so included a significantly shorter length of cables that needed to be installed and absence of concern about the amount of fuel needed for generating electricity (unlike the case of using a power-generation vehicle).

Restoration work was completed on March 14, and the reactor cooling method was switched from the MUWC to the RHR residual heat removal operation at 07:13 on the same day.

(iii) Unit 3
The P/C on the subsystem-A and the three pumps that use a power supply were flooded and disabled, but three emergency diesel generators and three M/Cs as well as the P/C on the subsystem-B and its load were intact. For this reason, immediate restoration work was not needed, and the RHR system on the subsystem-B could be put in service.

Very early on, at 12:15 on March 12, RHR-based residual heat removal operation was put into operation.

(iv) Unit 4
The damage incurred by the flooding which broke in through the service entrance of the seawater heat exchanger building was significant: both subsystems-A and B lost the P/C, and the five pumps which were the loads of the subsystem A were destroyed. Only the EECW pump (B) and the RHRS pump (D) were not damaged in the subsystem-B. Despite this, replacing the motor of the RHRC pump (B) and securing the electricity for this pump ensured one complete train—the minimum necessary. Based on this, it was decided to use a generator truck for the EECW pump (B), and obtain power supply from the P/C of the Unit 3 for the RHRC pump (B) and the RHRS pump (B).

The restoration work was completed on March 14, and water cooling by the MUWC was switched to the RHR residual heat removal operation at 15:42 on the same day.

As stated above, the shift from the MUWC system and the HPCS to the RHR-residual heat removal operation was completed at all Units from 1 to 4. The cold shutdown was achieved at Unit 1 at 17:00 on March 14, Unit 2 at 18:00 on March 14, Unit 3 at 12:15 on March 12 and Unit 4 at 07:15 on March 15, respectively.

c. Observation and evaluation

(i) Possibility of SBO
In order to obtain power from Tomioka line #1, which was the only line where the startup transformer for all units of Fukushima Daini survived, power needed to be transmitted through a high voltage startup transformer. The high-voltage startup transformer, which was a bottleneck, was damaged by the earthquake, and oil had been leaking from the conservator (expansion tank). [64]

Luckily, the damage was not fatal to the transformer’s functions. But if other important functions had been damaged and the transformer had been interrupted, Units 1 and 2, which had lost all emergency diesel generators, could have possibly ended up in an SBO state. [65]

(ii) What contributed to the avoidance of a reactor accident
The water temperature of the S/C pools at Units 1, 2, and 4 of Fukushima Daini were

[64] Hearing with workers who were on-site at Fukushima Daini at the time of the accident
[65] There is a possibility that a method to supply the power to the Units 1 and 2 from the intact emergency diesel generators at the Unit 3 and 4 through the 66kV startup switchyard.
reported to exceed 100 degrees Celsius at 05:22 on March 12 at Unit 1, 05:32 at Unit 2 and 06:07 at Unit 4. The situation developed very similarly to Fukushima Daiichi Units 2 and 3. This indicates that the Fukushima Daini plant might have been looking a nuclear reactor accident in the face.

The plant avoided a reactor accident, however, thanks to some lucky factors, including the fact that every unit managed to escape SBO, as well as the availability of the MUWC pumps. Those pumps were located on the first basement floor of the turbine buildings of Units 1 and 2, and on the second basement floor of the turbine buildings for Units 3 and 4. In the case of Fukushima Daiichi Units 1 to 4, the rolling shutter doors of the turbine buildings’ truck bays were destroyed by the tsunami, and a large body of water instantly flooded the Fukushima Daiichi site. But this did not happen at the Fukushima Daini turbine buildings. Although there was water leakage at the auxiliary building of the Unit 1 reactor building and at the trench connecting the seawater heat exchanger building and the turbine building of Unit 3, the water passed down to the sump within the building below. Thus the MUWC pump was not flooded. The situations at the Units 2 and 4 were even less critical.

Except for the Unit 1 MUWC pumps (A) and (C), which became unavailable because their power sources submerged, everything remained in sound condition, so the shift supervisor at each unit was able to have their operation staff start the MUWC pumps according to the emergency operating procedure manual (EOP). The reactor was depressurized by opening the SRV, the reactor water injection method was gradually shifted from the RCIC to the MUWC, and the plant successfully overcame the crisis.

(iii) How the accident response would have been in case of SBO

As the feature of MUWC pump is a flow amount of 120 to 160 cubic meters per hour at the discharge head of 85 to 90m, and as the shut-off head was 150 to 200m, it is assumed that the shift of water injection from the RCIC to the MUWC took place gradually after the reactor pressure was lowered to below 2MPa. The specifications are important because if the water injection does not shift from the RCIC until the reactor is considerably depressurized, then the temperature and pressure of the water in the S/C pool would rise further. As a result, the reactor depressurization would reach its limit and make a shift from RCIC to the low pressure water injection difficult. This case may require additional operation to “vent the containment vessel,” which was avoided thanks to the availability of AC power supply to run the MUWC pumps. This point is considered a significant difference compared to Fukushima Daiichi.

If the AC power were lost, it would be difficult to actuate SRVs and air operated valves to vent the containment vessels in case DC power is quickly depleted. It would have made the accident response even more complicated.

(iv) Difficulties in restoring a discharge route to the ultimate heat sink

Because M/Cs and the P/Cs were damaged by the tsunami which struck at 15:22 on March 11, the motors which lost their power source needed to be reconnected immediately and directly to other M/Cs and P/Cs that did not break or generator trucks with the use of routing cables. This required a quick and accurate grasp of the availability of the pumps, M/Cs and P/Cs, and decisions on how to combine them and establish discharge routes from the reactor and the containment vessel to the ultimate heat sink.

On the other hand, it would take too long to grasp the whole picture of all the damage, to decide on the heat discharge route to the ultimate heat sink, and to begin searching for necessary types and quantities of motors and cables and power generat-

### Table 2.1.5-1: Parameters of the Units 1, 2, and 4 at the Fukushima Daini plant

<table>
<thead>
<tr>
<th>Unit</th>
<th>Maximum temperature</th>
<th>Maximum pressure (gauge pressure)</th>
<th>Saturation temperature of the maximum pressure on the left</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>130°C</td>
<td>282kPa</td>
<td>131°C</td>
</tr>
<tr>
<td>2</td>
<td>139°C</td>
<td>279kPa</td>
<td>131°C</td>
</tr>
<tr>
<td>4</td>
<td>137°C</td>
<td>245kPa</td>
<td>127°C</td>
</tr>
</tbody>
</table>

[66] A written reply from TEPCO, and hearing with workers who were on-site at Fukushima Daini at the time of the accident
ing trucks. Therefore, the restoration team in the on-site emergency response center in the seismic isolation building became the window to contact various parties with rough ideas to procure all necessary materials and equipment without a shortage. As a result, an enormous amount of goods and supplies were delivered.

The cables had to be routed over several hundred meters immediately after the tsunami had subsided. The mess of debris on the roads had to be cleared as the cables were routed. When the cable arrived in response to the urgent request of the restoration team, the workers faced the issue of how to handle the heavy triplex conductor cable and the huge mandrel on which the cable was wound. As many people as possible were gathered and the cable was routed in a labor intensive manner. There was no choice other than to route the cable directly on the ground without curing, although it was not a preferred method. More than 200 TEPCO employees and contract workers were deployed to complete this physical task.

In order to restore the motor of the RHRC pump (D), a terminal of the routed cable had to be connected to the terminal box of the motor, but there were only a few experienced technicians to complete such task. An on-site worker later said that he keenly felt at this point there were structural issues with TEPCO, which for decades had depended on contractors to provide the majority of practical tasks. In any case, the task was finally completed and the RHRC pump (D) went into operation on March 13.

Even when the ultimate heat sink was provisionally restored, the workers still did not feel relieved or confident. They were concerned about whether the motors, which were coupled, aligned and connected to the pumps in a rush in a harsh environment, would continue to operate stably. The interlock signals for safety protection that were designed to be automatically initiated by the system operation parameters were out of order, so instead, workers manually monitored changes in the situation. The tsunami brought mud and sand, which became dust and saline when dried, that could have damaged bearings and other moving parts of machineries or delicate electrical components. Accordingly, the workers monitored electrical currents, observed temperature by thermography, analyzed lubricants, and stopped their work temporarily from time to time for inspection. The generator truck quickly consumed fuel, necessitating frequent refueling. There also was fear of more earthquakes, aftershocks and tsunami.

The plant managed to avoid falling into an immediate crisis, but the situation was still precarious and the workers were not fully relieved. The residual heat removal operation, which was finally established, depended on coordination between the RHR, the RHRC, the RHRS, and EECW, and it needed to be made redundant as soon as possible to prepare against future danger. While successfully avoiding the immediate crisis made workers feel confident, there was no assurance that the next event would not be more severe than the previous event. It took one more week of hard work for the on-site workers to feel assured that restoration was achieved.

(v) Lessons learned from the series of accidents

In reviewing the series of accident responses taken at the Fukushima Daini plant, it is obvious that there were a number of factors that could have made things more severe, although some were handled with a flexibility that suited the circumstances.

For an example, the loss of one more P/C could have created an enormous amount of additional work. As such, subtle differences in the natural conditions can have significantly different results, from severe to harmless. In this light, it must be acknowledged that it was partially a matter of good fortune that the situation at the Fukushima Daini plant was not as tragic as that of the Fukushima Daiichi plant.

In order not to leave nuclear reactor accidents to the mercy of the forces of nature, we should learn to incorporate prudent considerations in the design and to be prepared at any time.
2. Onagawa Nuclear Power Plant and Tokai Daini Nuclear Power Plant

a. Risks of accident at Onagawa Nuclear Power Plant

At Onagawa, the 275kV Matsushima Main Line #1, both of the two 275kV Ojika Main Lines, and one 66kV Tsukahama Branch Line were shutdown by a protective circuit breaker triggered by the earthquake. But the off-site power supply from the 275kV Matsushima Main Line #2 was secured,\(^\text{[70]}\) simply because of luck.

The area around the entire plant subsided by 1.0 meter because of the earthquake, and as a result the elevation of the key buildings subsided to 13.8m above O.P.\(^\text{[71]}\) In comparison, the tsunami had a approximate height of O.P. + 13m,\(^\text{[72]}\) leaving a very small margin of only 0.8m between the height of the tsunami and the elevation of the land. The ocean was at low tide at the time of tsunami, so the fact that the tsunami did not reach the key buildings of the plant was purely coincidence.

The attributes of tsunamis are the extremely high uncertainties around their probability of occurrence, size, route, damage, scope of effects and scale when one reaches the area of key buildings.\(^\text{[74]}\) Besides, tsunamis have a cliff-edge effect.\(^\text{[75]}\) What would have happened if the height of the tsunami was higher than the site ground elevation at Onagawa? The seawater pumps of the residual heat removal system and component cooling water system, the on-site power distribution boards such as M/Cs and P/Cs, and other equipment and facilities of Units 2 and 3 which are located along the coastal line and Unit 1 which is located behind Unit 2, would have been damaged or affected considerably. In addition, the deterioration of the surrounding environment due tsunami debris and seawater residue may have hindered the on-site accident responses. These possibilities should be considered as well.

It would have been very difficult to avoid a reactor accident if the situation was different at Onagawa Nuclear Power Plant.

b. Risks of accident at Tokai Daini Nuclear Power Plant

At the Tokai Daini Nuclear Power Plant, all of the off-site power supply was lost from the earthquake. In addition, a seawater pump (2C) located on the north side of the seawater pump area at roughly H.P.\(^\text{[76]}\) + 5.1m above sea level along the coastline, used to cool the emergency diesel generator, was submerged by the tsunami which was as high as H.P. + 5.4m. Due to the loss, the emergency diesel generator (2C) became unusable. On the other hand, the residual heat removal system's seawater pumps (A) and (C), which sat adjacent to the seawater pump (2C) for the emergency diesel generator, remained functional; although they were flooded to the pumps, the electrical parts above were safe. The electrical parts survived the flood with a good amount of luck—not because of well-thought measures prepared in advance.

Other seawater pumps such as (2D) for the emergency diesel generator, the (HPCS), and (B) and (D) for the residual heat removal system were not flooded either, although they were located on the south side of the same seawater pump area where the pumps (2C) for the emergency diesel generator and (A) and (C) for the residual heat removal system were flooded. If the scale, power, frequency and route of the tsunami were different, the seawater pumps (2D) for the emergency diesel generator, (HPCS), and (B) and (D) for the residual heat removal system could have been flooded and disabled along with the seawater pumps (A) and (C) for the residual heat removal system.

This would have resulted in a loss of all emergency diesel generators and the loss of on-site AC power, causing a complete loss of AC power supply (coupled with the loss

\[^{[70]}\] Tohoku-epco documents

\[^{[71]}\] ‘O.P.’ represents datum plane for construction at Onagawa Nuclear Power Plant only in 2.1.5, 2).

\[^{[72]}\] Tohoku-epco documents

\[^{[73]}\] At Ayukawa-hama, Ishinomaki-city, Miyagi-prefecture on May 11, 2011, the level of the ocean's surface of high tide recorded 234 centimeters at 6:14 and that of low tide did 136 centimeters at 13:15.

\[^{[74]}\] Hearing with workers who were on-site at Fukushima Daini at the time of the accident

\[^{[75]}\] The cliff edge effect refers to a significantly irregular behavior of a power plant caused by a rapid shift of a state of a nuclear power plant from one to another as a result of a small deviation of a parameter of a power plant. It is a sudden and significant change of a plant condition in response to a small change in input.

\[^{[76]}\] The ‘H.P.’ is Hitachi Peil. H.P. -0.0m is -0.89 of the Tokyo Peil (T.P.).
### Chapter 2 | page 47

#### The National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission

### Maximum acceleration

<table>
<thead>
<tr>
<th>Earthquake</th>
<th>Unit 1</th>
<th>Unit 2</th>
<th>Unit 3</th>
<th>Unit 4</th>
<th>Unit 5</th>
<th>Unit 6</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design basis earthquake</td>
<td>460</td>
<td>530</td>
<td>507</td>
<td>319</td>
<td>548</td>
<td>444</td>
</tr>
<tr>
<td>Difference</td>
<td>27</td>
<td>Δ 112</td>
<td>Δ 66</td>
<td>126</td>
<td>Δ 96</td>
<td>4</td>
</tr>
</tbody>
</table>

### Tsunami

<table>
<thead>
<tr>
<th>Inundation height (main building area)</th>
<th>Unit 1</th>
<th>Unit 2</th>
<th>Unit 3</th>
<th>Unit 4</th>
<th>Unit 5</th>
<th>Unit 6</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>15.5</td>
<td>14.5</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Difference</td>
<td>Δ 5.5</td>
<td>Δ 1.5</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

### Shutdown

<table>
<thead>
<tr>
<th>Scram</th>
<th>AC pwr source</th>
<th>Transmission, transformation</th>
<th>Emergency diesel generator</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>On-site pwr source</td>
<td>On-site pwr source</td>
<td>On-site pwr source</td>
</tr>
<tr>
<td></td>
<td>DC pwr source</td>
<td>DC pwr source</td>
<td>DC pwr source</td>
</tr>
<tr>
<td></td>
<td>P/C</td>
<td>P/C</td>
<td>P/C</td>
</tr>
<tr>
<td></td>
<td>M/C</td>
<td>M/C</td>
<td>M/C</td>
</tr>
<tr>
<td></td>
<td>AC pwr source</td>
<td>AC pwr source</td>
<td>AC pwr source</td>
</tr>
<tr>
<td></td>
<td>D/C pwr source</td>
<td>D/C pwr source</td>
<td>D/C pwr source</td>
</tr>
<tr>
<td></td>
<td>Containment vessel cooling</td>
<td>Containment vessel cooling</td>
<td>Containment vessel cooling</td>
</tr>
<tr>
<td></td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
</tr>
<tr>
<td></td>
<td>Cooling</td>
<td>Cooling</td>
<td>Cooling</td>
</tr>
<tr>
<td></td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
</tr>
<tr>
<td></td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
</tr>
<tr>
<td></td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
<td>Reactor cooling</td>
</tr>
</tbody>
</table>

### Damage and its effect and success or failure of accident preventive efforts

<table>
<thead>
<tr>
<th>Damage and its effect and success or failure</th>
<th>Fukushima Daiichi nuclear power plant</th>
</tr>
</thead>
<tbody>
<tr>
<td>Containing</td>
<td></td>
</tr>
</tbody>
</table>

### Table 2.15-2: Summary of damages and its effects and accident preventive efforts at each nuclear power plant

<table>
<thead>
<tr>
<th>Containing</th>
<th>Fukushima Daiichi nuclear power plant</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pellets, fuel rod cladding</td>
<td>Unit 1</td>
</tr>
<tr>
<td>Pressure vessel, containment vessel</td>
<td></td>
</tr>
<tr>
<td>Reactor building</td>
<td></td>
</tr>
</tbody>
</table>

### Notes

- **O**: no damage or success
- **Δ**: partial functionality loss or failure
- **X**: complete functionality loss or failure

The numbers indicated as (i/j) on the table mean that i number out of j number kept their functionality. Where i number is not zero and “complete functionality loss or failure” are both indicated, the subject equipment and instruments are disabled due to the functionality loss of other equipment and instruments.

**Table with details for Fukushima Daini, Onagawa and Tokai Daini nuclear power plants on the following pages.**

---

[77] Compiled by NAIIC
## Damage and its effect and success or failure of accident preventive efforts

<table>
<thead>
<tr>
<th>Earthquake</th>
<th>Maximum acceleration</th>
<th>Among all the recording maximum acceleration date on the base mat of the reactor building, the record set with the largest difference from the design basis earthquake is indicated (Units: Gal)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>254</td>
<td>243</td>
</tr>
<tr>
<td>Tsunami</td>
<td>Onahama Peil for Fukushima Daiichi and Daini, Onagawa Peil for Onagawa, and Hitachi Peil for Tokai Daini, are indicated (same as above)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>180</td>
<td>185</td>
</tr>
<tr>
<td>Shutdown</td>
<td>Scram</td>
<td>×</td>
</tr>
<tr>
<td>Power source</td>
<td>AC pwr source</td>
<td>On-site pwr source</td>
</tr>
<tr>
<td></td>
<td>DC pwr source</td>
<td>D/C pwr source</td>
</tr>
<tr>
<td></td>
<td>M/C</td>
<td>Δ (9/11)</td>
</tr>
<tr>
<td></td>
<td>P/C</td>
<td>Δ (9/10)</td>
</tr>
<tr>
<td>Cooling</td>
<td>High-pressure water injection</td>
<td>(RCIC)</td>
</tr>
<tr>
<td></td>
<td>Depressurization</td>
<td>(SR valve)</td>
</tr>
<tr>
<td></td>
<td>Low-pressure water injection</td>
<td>(MUWC)</td>
</tr>
<tr>
<td></td>
<td>Containment vessel cooling or depressurization</td>
<td>(MUWC)</td>
</tr>
<tr>
<td></td>
<td>Removal of residual heat to the ultimate heat sink</td>
<td>(RHR-LPCI)</td>
</tr>
<tr>
<td></td>
<td>Seawater cooling instrument system (CCSW, RHRS, RSW, and so on)</td>
<td>× (0/2)</td>
</tr>
<tr>
<td>Containing</td>
<td>Pellets, fuel rod cladding</td>
<td>×</td>
</tr>
<tr>
<td></td>
<td>Pressure vessel, containment vessel</td>
<td>×</td>
</tr>
<tr>
<td></td>
<td>Reactor building</td>
<td>×</td>
</tr>
</tbody>
</table>

---

**Continued from previous page:**

Table 2.1.5-2: Summary of damages and its effects and accident preventive efforts at each nuclear power plant.
### Damage and its effect and success or failure of accident preventive efforts

<table>
<thead>
<tr>
<th></th>
<th>Onagawa nuclear pwr plant</th>
<th>Tokai Daini nuclear pwr plant</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Unit 1</td>
<td>Unit 2</td>
</tr>
<tr>
<td>Earthquake</td>
<td>387</td>
<td>607</td>
</tr>
<tr>
<td></td>
<td>529</td>
<td>594</td>
</tr>
<tr>
<td>Tsunami</td>
<td>△ 38</td>
<td>△ 13</td>
</tr>
<tr>
<td>Shutdown</td>
<td>13</td>
<td>6.3</td>
</tr>
<tr>
<td>Scram</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Power source</td>
<td></td>
<td></td>
</tr>
<tr>
<td>AC pwr</td>
<td></td>
<td></td>
</tr>
<tr>
<td>DC pwr</td>
<td></td>
<td></td>
</tr>
<tr>
<td>M/C</td>
<td></td>
<td></td>
</tr>
<tr>
<td>P/C</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cooling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reactor cooling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>High-pressure water injection</td>
<td>(RCIC,RD)</td>
<td>–</td>
</tr>
<tr>
<td>Depressurization</td>
<td>(SR valve)</td>
<td>–</td>
</tr>
<tr>
<td>Low-pressure water injection</td>
<td>–</td>
<td>–</td>
</tr>
<tr>
<td>Containment vessel cooling or depressurization</td>
<td>–</td>
<td>–</td>
</tr>
<tr>
<td>Removal of residual heat to the ultimate heat sink</td>
<td>(RHR-SHC)</td>
<td>(RHR-SHC)</td>
</tr>
<tr>
<td>Seawater cooling instrument system (CCSW, RHRS, RSW, and so on)</td>
<td>(4/4)</td>
<td>△ (2/4)</td>
</tr>
<tr>
<td>Containing</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pellets, fuel rod cladding</td>
<td>–</td>
<td>–</td>
</tr>
<tr>
<td>Pressure vessel, containment vessel</td>
<td>–</td>
<td>–</td>
</tr>
<tr>
<td>Reactor building</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Continued from previous page:

Table 2.1.5-2: Summary of damages and its effects and accident preventive efforts at each nuclear power plant.
of the off-site power supply that actually happened). Then, because all the seawater pumping functions for the residual heat removal system would have been lost, the discharge line to the ultimate heat sink would have been lost. Loss of all AC power supply and the ultimate heat sink is a much harsher scenario than what actually happened at the Fukushima Daini plant.

As stated, it was possible that avoiding a reactor accident could have become extremely difficult at the Tokai Daini plant. To the same extent, it is worth noting that flooding in the battery room was actually confirmed as a threat to the DC power supply system.

3. Summary

This section verified the damage wrought by the earthquake and tsunami, how other nuclear plants were affected by it, and the risks of accidents. From this effort, we found the following:

- Other nuclear power plants also suffered from various and diverse damages and effects caused by the earthquake and tsunami. They were not solidly prepared against threats to the safety of nuclear power plants.
- Other nuclear power plants could have also suffered a nuclear reactor accident if the damages and effects of the earthquake and tsunami, and the accident prevention efforts, were even slightly different from what they actually were.

Figure 2.1.5-2 summarizes the damages and effects of the earthquake and tsunami, actual accident prevention efforts performed, and how successful the efforts were at each nuclear power plant.

2.1.6 Discussion

This section discusses the accident by assessing the issues covered in the preceding 2.1.2 to 2.1.5. We also looked at issues from a macroscopic point of view, beyond the scope of the accident itself.

1. Hampering factors to the accident response

   a. Containment vessel venting

   At Units 1 to 3, the pressure inside the containment vessel exceeded the design level because the heat in the reactor and the containment vessel could not be discharged to the ultimate heat sink. The venting operation, therefore, was necessary in order to prevent the vessel's rupture. The emergency operating procedures (EOP) manual that describes the venting procedure was written on the assumption that the control panel in the main control room would be operable for monitoring the status and manipulating the equipment of all the plant systems. It was extremely difficult to vent the containment vessel without the use of the main control room's control panel and any DC power.

   The vent line added for the sake of accident had been installed by sharing a part of the existing facilities, such as the Heating, Ventilating and Air Conditioning (HVAC) system of the reactor building, SGTS, and Air Conditioning (AC) system of the containment vessel. The line had interface points with these systems, and an isolation valve had been installed at each of the nine interfaces. The manual stipulates that the operators must ensure that all these valves are closed before starting the venting process.

   In their response to the accident, the operators were not able to check the status of the valves due to the loss of DC power, so the venting operation was executed without fully ascertaining the status of the valves. One year after the accident, TEPCO has still not been able to verify whether the rupture disk in the vent line was activated (broken out). The inability to ascertain the isolation valves' status and the inability to verify the activation of the rupture disk suggest that the gas emitted from the containment vessel most likely flowed into the reactor building via the systems shared with the vent line and remained there. If the execution of the venting operation induced the explosion that devastatingly destroyed the reactor building, i.e., the final barrier of the “five barriers” to “contain” radioactivity, it can never be said that “the containment vessel venting was successful.”

   According to the plant workers, they did not think of the possibility of gas flowing into
other systems through interfaces on the vent line when they implemented the venting operation. This lack of awareness can be attributed in part to how the reference manual drawings, that were to be referred to when performing venting, had been prepared. On one hand, the set of drawings provided in the main control room did not include a piping and instrumentation diagram dedicated to the vent line as an independent system. The vent line was split into sections and added separately in the piping and instrumentation diagrams of the HVAC, SGTS and AC systems. It was a painstaking task to understand the overall picture by finding all the depictions of the vent line in the entire set of drawings. Though the emergency operating manual contained an (A3-sized) insert diagram which roughly illustrates the full scope of the vent line, it did not include drawings of the other systems that partly share the flow paths with the vent line. The insert alone was not sufficient in describing how the venting operation would affect other systems. It was extremely difficult for the plant workers to see and understand the poorly developed drawing of the vent line, which they had never used before nor were trained to use, especially under the extreme pressures of time, using only torch lights in the darkness.

Considering these conditions, simple criticism of their failure to configure the vent line to perform the vent operation in an competent manner is not appropriate. The root cause here is that the design of the vent line was complex and inefficient. All utilities that have similar situations at their nuclear power plants should immediately take remedial actions.

b. Basic knowledge, operating procedures, provision of equipment and materials, and training necessary for responding to a severe accident

Among all the personnel involved in the accident response, some had a certain level of knowledge on the progression of a reactor without coolants, and a clear understanding of the essence of the measures to be taken in such a case. However, according to one of the workers involved in the response, when they actually confronted the reactor accident, “everybody remained silent, lost in their thoughts on how the workers’ morale in dealing with the accident” would be affected if those lacking such knowledge learned of the possible consequences. [78]

At the time of a reactor accident, which develops very quickly and changes the status significantly, responders are expected not only to be sufficiently familiar with operating procedures but to know the time required for each procedure to be performed. It is also important for the entire operator team in the main control room that is responsible for responding to the accident to share that knowledge. Without a confirmed common awareness of these important factors, mitigation measures that are decided upon and then communicated to the workers, would probably not lead to an optimal response with intent by the workers in charge of performing that particular mission.

What if the plant workers across the board had acquired a high level of background knowledge about severe accidents through mandatory classroom and hands-on training programs? What if they had undertaken exercises in a tense atmosphere based on the obtained expertise, and conducted inspections on necessary equipment and materials? Perhaps they would have been better prepared, experienced fewer missing and lacking elements, and could have implemented the post-accident measures more effectively and efficiently. One example: the fire engines at the site and the self-contained breathing apparatus (SCBA) in the reactor building had been provided there essentially for firefighting, not as necessary equipment and materials for responding to a reactor accident. It took a considerable time to check the operational status of the IC for Unit 1, which was suspected of being offline, and to bring it back online, and to ascertain the condition of the RCIC for Unit 2 after its operational status became inaccessible.

We must give fair consideration to the fact that the challenging situation hampered the accident response to a certain extent. After interviews with the responders and other parties concerned, however, there still remains uncertainty about how clearly a feeling of urgency was communicated before beginning the response actions.

c. Lack of training and exercising on operating procedures for a severe accident

BWR Operator Training Center Corp. designed its training and exercise courses for

[78] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
severe accidents based on the assumption that DC power would remain available and the control panel in the main control room operable. None of the training courses had postulated a severe accident without DC power and readings on the control panel, as in the case of this accident.

BWR's training courses focused too much on desktop training aimed at making trainees "capable of explaining" the content of the "severe accident response." Practical training sessions were not a part of the courses. According to the Center, that is partly because the response to a severe accident involves a broader community than just the plant operators stationed in the main control room. The response consists primarily of coordination and concerted action with organizations and personnel outside the main control room, such as the emergency response unit to be set up at the time of an accident. Therefore, the presence of a simulator and a group of trainees in a simulated control room is insufficient for performing as-intended exercises. The Center also mentioned, as another reason, that no nuclear power operator had demanded practical training on severe accidents.

That being the case, the accident response procedures following a station blackout had never been examined. In the case of the accident at Fukushima, the response depended inevitably on a sequence of trials and errors by the plant workers.

It is unfair to compare the actual decisions and actions of the on-site operators at the time of the accident to the ideal practices identified, based on a post-accident investigation, or to blame them for any inappropriateness. Rather, we should be questioning what the true cause was—what led the operators to confront a severe accident with no prescribed procedures and with no training or exercise.

d. Demarcating normal operation and severe accident response

TEPCO’s work management group, the periodical inspection group and the power generation unit assisted the operators in the main control room in responding to the accident, directly on the site and from the back office. Many of these personnel had experience in plant operation. They were therefore considered able to provide assistance in normal plant operation and in severe accident conditions as taught in typical training sessions.

As verified in 1.3, however, TEPCO had disregarded the possibility of the occurrence of severe accidents in its severe accident management policy. Their classroom and hands-on training programs were lacking both quality and effectiveness. The members of the work management and periodical inspection groups and the power generation unit were not specialized in or qualified and responsible enough to provide technical assistance in a severe accident. They were not competent enough to provide timely and effective technical assistance in keeping with the rapid progression of a severe accident, as in the case in Fukushima.

The utility needs to draw a clear line between normal plant operation and the emergency response to a severe accident. It should set up an organization dedicated to providing technical assistance at the time of a severe accident, and properly manage the organization with constant training and exercising.

e. Other factors

The following factors may also have caused difficulties in responding to the reactor accidents at Units 1 to 3.

(i) Giving up on high-pressure water injection

At the early stages of response to the reactor accident at Unit 2, the responders attempted to restore the CRD and SLC pumps—which had spared the flooding of the P/C and were capable of receiving power—with a view to performing high-pressure injection. Despite the many difficulties anticipated in using these for the operation, they managed to complete the routing of the temporary power cable. It was only a few
minutes later that an explosion occurred at Unit 1, scattering debris that damaged the cable and forced them to give up on the idea of high-pressure injection. Again and again, they faced a situation where the water injection operation had to be suspended because they could not depressurize the reactor. The absence of water injection may have worsened the core damage, enhanced the degradation of the air-tightness of the containment vessel, and eventually increased the amount of radioactive material released into the environment.

(ii) A real-time analysis tool to predict and update progression of a severe accident
Such an analysis would have helped the parties involved in responding to the reactor accidents in sharing information effectively. The lack of such a tool both at the plant and at the TEPCO head office created discrepancies in the awareness of the progression of the accidents between the two sides[83] and in turn adversely affected domestic and international communications.

2. Factors contributing to averting reactor accidents
   a. Contribution of the main seismic isolation building
Units 5 and 6 at the Fukushima Daiichi plant, and the Fukushima Daini, Onagawa and Tokai Daini plants all successfully avoided reactor accidents. Of course, the conditions following the earthquake and tsunami were different at these plants. More specifically, each site faced a different level of damage to the power source and the ultimate heat sink and had a different range and severity of flooding in the premises and the buildings. They were, however, all forced to react under extremely-high levels of tension.

   In particular, the Fukushima Daini plant was hard-pressed. One of the workers involved in the response described the experience as “having no leeway to pay attention to the situation at Fukushima Daiichi plant.”[84] Obviously, what counted at those crucial moments was assessing the status in an appropriate and speedy manner. At the same time, securing necessary equipment, materials and sufficient manpower was equally important in order to take action based on those assessments.

   The presence of the “seismic isolation building,”[85] an emergency management facility, played significant roles at all of these plants when the earthquake struck. In terms of logistics, it allowed the responders to take and complete actions necessary for averting reactor accidents; the building provided sufficient space for the few hundred workers involved in the restoration activities on the site to stay and allowed them to take meals and rest, though minimally, in a relatively favorable setting considering the emergency situation.

   The building demonstrated the expected capacity for resisting earthquakes, as indicated by its name. Nonetheless, based on the hindsight obtained by NAIIC from the visits to the facilities, there are some issues related to independence; i.e., the building was supplied with power from the plant’s emergency power system. Furthermore, some provisions, such as the whole-body counters, radiation analysis room, and refiller for air cylinders for airline masks, were not adequate. There is still room for improvement in radiation shielding and air-tightness at the Fukushima Daiichi plant and flood prevention on the first (ground) floor at the Fukushima Daini plant, as well as in other areas.

   b. Importance of assistance provided by subcontractors
The seismic isolation building served as the frontline base inside the plant, and the employees of the utility and the supervisors as well as workers of subcontractors, who courageously stayed there in the extreme circumstances, were able to work toward a common objective. Subsequently, the resulting mutual trust and sense of solidarity naturally gave rise to a good moral environment.[86]

[83] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
[84] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
[85] The seismic isolation building was put in place based on the lessons learned at TEPCO’s Kashiwazaki-Kariwa Nuclear Power Plant at the time of the Niigataken Chuetsu-oki Earthquake in 2007.
[86] Hearing with workers who were on-site at Fukushima Daini at the time of the accident
TEPCO employees took part in front-end tasks, such as the parallel operation of generator truck and cable installation, which had previously been left entirely to subcontractors. This experience shed light on the fact that these tasks involve a diverse range of special skills, and that TEPCO employees lack practical experience and knowledge in these skills; in other words, it unveiled how dependent TEPCO was on the subcontractors. At the same time, the employees keenly realized the importance of the trust between TEPCO and their subcontractors built in peacetime in case of an emergency.\footnote{87}

Nuclear operators should, first of all, identify the areas in which their plant workers have scarce experience and knowledge. Then, they must work, at normal times, to raise the skill of the workers while promoting the sharing of knowledge and experience and the collaboration with subcontractors.\footnote{86}

Proactively examining emergency response strategies is also indispensable. They can effectively create readiness for a swift response and the procuring of necessary items in case of emergency by listing the kinds of capabilities and where they are.\footnote{88}

c. Spirits of plant workers

It is not hard to imagine that those plant workers with knowledge about reactor accidents had to mentally prepare themselves before stepping into the dark reactor building, where the situation of the reactor was worsening every minute. At Fukushima Daiichi Unit 1, skilled workers voluntarily went into a 300 mSv/h environment.\footnote{89}

Plant operation was the only domain in which the utility bore the sole responsibility and did not rely on subcontractors. In reply to a question asked by NAIIC on the state of mind when they entered dangerous situations, the employees mentioned the professionalism as workers in charge of plant operation and their affection for the land where their families reside.\footnote{90} Those plant workers who were fortunately spared the need to go into such an environment also had similar spirits.\footnote{92} It must be noted that the courage and actions of these high-spirited operators realized the cold shutdowns of the reactors while in grave danger.

In addition, every shift team was called “family,” and the shift members were engaged in plant operation and trained together. Through these daily practices, the members of the teams forged a sense of unity and solidarity. This seemingly played a part in their ability to immediately react to the sudden transition from normal operation to the crisis of a reactor accident and also in their ability to perform tasks to avert accidents.\footnote{93}

d. Concerns over future accident responses

Some are concerned that such success factors might be weakened, rather than strengthened, as a result of this accident. Now that the dangers and fears associated with a reactor accident are publicly known, the same level of response may not be attained should another reactor accident occur.

To further reinforce these workers, parties in the nuclear circle should focus on the importance of “courage endorsed by knowledge” and be committed to support actions taken by each responsible person individually. It is worth noting that some sources are worried that, certainly if individuals are expected to make such a commitment, and even if this topic is just openly discussed, that it may lead to difficulty in securing human resources to continue handling Japan’s nuclear power.

3. Potential factors contributing to difficulties in response to the accident

a. If DC power had been lost before the start of RCIC …
At Unit 2, both AC and DC power sources were lost after about two minutes from completing activating the RCIC. The system went inoperable then, since DC power was required. If AC and DC power had both been lost any earlier, the RCIC could not have been started. The HPCI could not have been activated either, for the same reason, and that would have soon triggered a core damage event.

Unit 3 encountered a station blackout when the RCIC was not in service. Fortunately, however, the RCIC was brought online because the batteries and distribution system for DC power survived. If, like Unit 2, the DC power had been lost, Unit 3 would have seen the sudden development of a core damage event, maybe with a limited time margin.

At the Fukushima Daini Nuclear Power Plant, the operators actuated the RCICs of all the units immediately after the plant was hit by the tsunami. If the damage at each unit had been severer and the AC and DC power supplies had been lost all at the same time, the operators would not have been able to manipulate the RCICs and HPCIs for these units. In turn, the units would have veered toward core damage.

Some may think that the possible early release of an enormous amount of radioactive material as a result of the catastrophic failure of the containment vessel was fortunately averted in the accident. Conversely, if the condition had been worse, even faintly, or the conjuncture had been somehow different, a more severe accident could have developed so quickly that there would have been no time to evacuate the local residents. The same applies to the other reactors that did not experience an accident. We do not know the rational explanation why these possible worse consequences were averted.

b. If the solenoid valves of SRVs had gone down …

The depressurization of the reactor pressure vessel using SRVs took place after the core damage had progressed and the temperature inside the containment vessel had risen quite high. Without the success of the depressurization procedure, it would have been impossible to inject water into the reactor using fire pumps with low discharge pressure.

In reality, however, the level of certainty about the successful depressurization after the in-containment temperature had risen was not very high. The SRVs are operated by sending high-pressure nitrogen gas that has accumulated in the accumulator to the drive cylinder. In other words, the normal behavior of the solenoid valves, which are responsible for this switching operation, was a prerequisite for functioning of the SRV. Meanwhile, switching the solenoid valves requires DC power. At the Fukushima Daiichi plant, many workers were engaged in scavenging batteries from vehicles to make up for the lost DC power. The security of DC power is not the only precondition for the normal operation of the solenoid valves. The solenoid valves are comprised of delicate parts made of non-metal materials. Had these components deteriorated due to the high temperatures, the solenoid valve could have failed.

TEPCO had performed operational checks on solenoid valves for the SRVs at 171 degrees (Celsius) and those for vent valves at 100 degrees (Celsius). It was unknown whether the valves would behave normally under more severe circumstances, as happened in this accident.[94]

According to sources at the Fukushima Daiichi plant,[95] some of the SRVs did not respond to the actuation command. In those cases, the operators tried other SRVs, one
after another, to find ones that worked.

Even in cases where all the solenoid valves for the SRVs are out of order because of the heat, there may be a way to pour water into the reactor pressure vessel later on.\[^{[96]}\] It is not very likely, however, that such an injection method could have been secured in a timely manner. And even if the operators managed to find and put such a method in place, we cannot rule out the question of what other feasible alternatives would have been available for depressurizing the reactor pressure vessel, a process that would be required sooner or later. The fragility of the solenoid valves, therefore, could have made the accident more severe.

c. If the earthquake had occurred at a difficult time…

The earthquake, which caused the accident, occurred at 14:46 on Friday, March 11, during an ordinary work shift on a weekday. The weather was fine. There was no strong wind or rainfall, which would have disturbed outdoor response activities. The climate continued like this for several days to follow. Incidentally, the earthquake hit during low-tide hours and Units 4 to 6 were under scheduled outage.

What if the earthquake had occurred at a different time? What if the weather had turned unfavorable on the day or over the following days while the initial response was under way? What if the tide level had been higher? What if Units 4 to 6 had been in service? There would have been far more hurdles to overcome.

Had any of these been the case, there would have been fewer personnel involved in the response, delaying the restoration work. The working environment would have been more dangerous, increasing the number of workers suffering from injuries or sickness. There would have been delays in preparing fire pumps, laying hoses and scavenging batteries. As a result, the accident could have developed faster, making it more difficult for the responders to grasp the situation in a timely manner and hence worsening the situation. Furthermore, the evacuation of residents would have been affected. Depending on the wind direction and/or rainfall, the level of radioactive contamination in the neighborhood could have been significantly higher.

The facts that it was an ordinary, daytime work shift time and that Units 4 to 6 were shut down for scheduled outage contributed to the sufficiency of the workforce engaged in the response at the time of the accident: 24 operators at the main control room for Units 1 and 2, 29 for Units 3 and 4, and 44 for Units 5 and 6.\[^{[97]}\]

The accident occurred at a right time for the response, in many aspects.\[^{[98]}\] But no one can predict the occurrence of a nuclear disaster. A viable response system needs to be built to ensure it functions under any conditions. The effective operation of such a system is crucial.

d. Is an SBO during a plant outage safe?

When the disaster occurred, the reactor pressure vessel of Unit 5 was undergoing a leak and hydrostatic test. For this, special arrangements had been made: the relief valves of SRVs were inactivated at all eleven valves as a countermeasure against malfunctions and human errors during the testing; and the safety valve functions, as well, were disabled at eight of them. The SBO occurred only six weeks or so after Unit 5 being shut down, under unusual circumstances. This is partly why the pressure in the reactor pressure vessel rose to 8.4 MPa in the ten hours or so after the accident occurrence, and why the temperature kept on increasing thereafter, reaching approximately 170 degrees (Celsius) at five o’clock on March 14.

---

\[^{[96]}\] Such alternatives include high-pressure injection using the CRD and SLC pumps, which was given up at Unit 2.
\[^{[97]}\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
\[^{[98]}\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident
We have the following questions:

- What would have happened if the period of the scheduled outage had been shorter and the decay heat had been greater?
- How would the temperature and pressure have increased, if the safety valve functions of all the SRVs had also been disabled like the relief function?
- What would have been the consequences if it had taken more time to change the test configuration to normal configuration?
- Are safety control measures and procedures for averting accidents established, specifically for responding to an SBO during a plant outage?
- Are plant workers provided with training and exercises for responding to an SBO during a plant outage?

During a plant outage, various safety functions and systems may be stopped or the plant parameter settings may be altered purposely from those for normal operation. Thus, a plant needs a specific response strategy for a cold shutdown, apart from the general response strategies.

The utilities must recognize once again that there are risks unique to an accident during a plant outage. They need to prepare themselves in advance through classroom and hands-on training specifically for a severe accident during an outage.

4. **Applied study**

- **a. If an accident had occurred at a nuclear power plant owned by a utility other than TEPCO**

  We examined the accident from a broader perspective: what if the same accident had occurred at a plant owned by a utility other than TEPCO? Or what if it had happened to a different reactor or containment design? We found that some of the examined cases could have resulted in far more serious consequences.

  NAIIC strongly recommends that this kind of discussion be held, not merely theoretically, but in earnest search of definite solutions. The following looks at four particular cases.

(i) Location of headquarters

Assume the accident in Fukushima occurred at a nuclear power plant owned by a utility other than TEPCO, which is headquartered in Tokyo. In actuality, Hokkaido EPCO, Tohoku EPCO, Chubu EPCO, Kansai EPCO, Chugoku EPCO, Kyushu EPCO and Shikoku EPCO are all based in a major city within their respective service area. This would make it difficult for the related parties in the government and competent authorities and responsible executives of the utility operating the plant where the accident occurred to sit together and discuss how to respond to the accident. In such a situation, the aforementioned real-time analysis tool to forecast possible developments of a severe accident would be more relevant.

(ii) Scale of enterprise

TEPCO is the largest utility in Japan, and of world-class. Not all the other utilities operating power-generation reactors can be classified on a similar scale. The relatively small sizes of Hokuriku EPCO, based in Toyama City, and Japan Atomic Power Co. (JAPCO), for example, are shown in Figure 2.1.6-3 below.\[99\] Even at this scale, the utilities must have satisfied the requirement for the “financial basis for

<table>
<thead>
<tr>
<th>Utility</th>
<th>Number of reactors</th>
<th>Total asset (100 million yen)</th>
<th>Annual turnover (100 million yen)</th>
<th>Number of employees</th>
</tr>
</thead>
<tbody>
<tr>
<td>TEPCO</td>
<td>17</td>
<td>147,904</td>
<td>53,685</td>
<td>32,970</td>
</tr>
<tr>
<td>Hokuriku EPCO</td>
<td>2</td>
<td>13,812</td>
<td>4,942</td>
<td>6,568</td>
</tr>
<tr>
<td>JAPCO</td>
<td>4*</td>
<td>8,165</td>
<td>1,752</td>
<td>2,198</td>
</tr>
</tbody>
</table>

* Including one decommissioned reactor

\[99\] Taken from their financial statements for fiscal year 2010
installing a nuclear reactor,” provided for by Article 24.1.3 of the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (“Nuclear Reactor Regulation Law”).

We learned from the accident in Fukushima that these utilities are utterly lacking the financial base and the human resources necessary for dealing with an accident like this. And the current law on compensation for nuclear damage does not provide for effective means of complementing such insufficiency.

Of a reactor accident occurred at a plant operated by a utility of this enterprise scale, the accident resolution process would entail extreme difficulties. We must say, very realistically, that they may not be able to handle the response work on their own.

(iii) Geographical conditions

The ground zero and the surrounding areas of the Fukushima Daiichi plant were inflicted with extensive damage from the accident. Nevertheless, the impact may have been relatively small, compared with an accident at any of the following nuclear power plants, whose geographic conditions put them in particular situations.

1. Higashidori Plant: 54km south of the plant are the Misawa military bases of the Japanese Self-Defence Forces and the U.S. Air Force. Should the equipment on the bases be contaminated by airborne radioactivity, the defence activities of both Japan and the U.S. would likely be affected.

2. Tokai Daini Plant: 80km south of the plant is the Narita International Airport. If radioactive materials released from the plant reached the airport, the aircraft of Japanese and foreign carriers parked there, the cargoes stored in the warehouses, and the vehicles parked in the parking space would all be exposed to airborne radioactivity. These goods and people could become carriers to spread the contaminants not only throughout Japan but overseas. In some cases, the transportation of all these goods and people would have to be halted.

3. Hamaoka Plant: A 20km evacuation zone would include sections of the Tokaido Shinkansen route and the Tomei Expressway. Should these transportation modes be paralyzed, the impact on the traffic would be enormous. There is no alternative to these networks today.

4. Genkai Plant: Approximately 30km north-northwest lies the island of Iki, where 30 thousand people reside. If evacuation was ordered or recommended, the residents would have no means to flee the island. In particular, if an accident coincided with a typhoon, they would be stranded on the island. The isles of Taka, Ogawa, Kakara and Madara are within a 10km radius area, and Kabe Island is situated even closer, a mere five kilometers away. One of these islands is connected to the Kyushu mainland by a bridge, which could be damaged by an earthquake. It would be difficult to find a way for the islanders to evacuate.

(iv) Reactors and other nuclear facilities with different designs

Units 1 to 3 at the Fukushima Daiichi plant are the BWR/3 and BWR/4 reactor types. They use MARK I containment vessels. On the other hand, as evident in the fact that Units 1 to 4 at the Fukushima Daini plant went through serious crises, BWR/5-type reactors and MARK II containment vessels are vulnerable to disaster situations of the same kind. Even the ABWR or PWR design is not equipped with particular capabilities that could possibly avert a similar reactor accident, should the same events involving an SBO and losses of DC power and the ultimate heat sink happen all at once as they did in Fukushima. This accident could have happened to any reactor design and with any containment vessel type.

Basically, the reactor and containment designs have nothing to do with the likelihood of an SBO, DC power loss and ultimate heat sink loss. It depends on the layout, seismic design and water-resistant performance of each plant as well as continued efforts of the nuclear operator to ensure safety.

What is more, for all the debate about causes and probabilities, a nuclear disaster at a fast breeder or a reprocessing installation would require an entirely different response from light water reactors. Careless water injection might cause a large-scale

[100] Advanced or next-generation reactor designs that incorporate “passive designs”, as the implementation is advocated lately, have better durability. Some of the existing plants have more durable units that are operated in compliance with the “B.5.b guidelines.”
explosion and fires for the former, for example, and uncontrolled criticality for the latter. No analyzing code dedicated to severe accidents at these installations has been developed. The number and level of engineers sufficiently experienced and knowledgeable are unknown as well.

5. Issues regarding multi-unit plants and neighboring nuclear power plants

a. Which is safer, a single-unit or multi-unit plant?

Unit 5 of the Fukushima Daiichi plant was shut down at the time of the disaster. It was undergoing a leak and hydrostatic test for the reactor pressure vessel, in preparation for a next cycle of operation. Fortunately, its DC power batteries were spared damage from the disaster. Without AC power, however, the batteries could not be charged and would be completely discharged before long. Important reactor parameter readings could not be monitored in the main control room. The relief valve function of SRVs would also become ineffectual. These potential issues were solved by cross-tying from the MCC (6C-2) of Unit 6, previously implemented as an accident management measure. This fact suggests a design advantage for a multi-unit plant over single-unit one.

On the other hand, the serious reactor accidents experienced by Units 1 to 4 of Fukushima Daiichi highlight the negatives of a multi-unit plant, as problems interacted with and amplified each other. Units 5 and 6, which spared reactor accidents, however, underline the positives of complementary interaction between the units. In short, multi-unit plants work favorably in terms of accident prevention but seem to work adversely at the post-accident mitigation stage.

This section examines the degree of difficulty in averting core damage accidents related to combinations of the presence or absence of a backup power supply from neighboring units and the operational status of plants. Figure 2.1.6-4 tabulates what happened to Units 1 to 3 and 5 at the Fukushima Daiichi plant and Units 1 to 4 at the Fukushima Daini plant. The cell with “?” indicates the question of what could have happened to Unit 5 of Fukushima Daiichi “if the power from the emergency diesel generator (B) of Unit 6 could not have been transferred.” Put in this perspective, Unit 5 could still have averted an accident, but that was not guaranteed.

b. Plant technical specifications on safety based on interactions between units at a multi-unit plant

Suppose a reactor accident occurs to one of the reactors at a twin-unit plant. What should be done to the other unit? Should the plant operators initiate the procedure for a cold shutdown right away? Should the air-conditioning system of the other unit be kept operational when performing the vent operation at the accident unit? Although the current plant technical specifications require continued operation in this case, operation should probably be halted in order to prevent the unnecessary inflow of radioactive material.

At Unit 5 of the Fukushima Daiichi plant, the workers quickly restored the power supply of the SGTS. Meanwhile, the core damage was progressing at Units 1 to 3, suggesting there was a high likelihood of contamination of the surroundings by radioactive material. In fact, the SGTS was brought back on line as soon as the restoration work was completed. The plant’s technical specifications surely require the SGTS to be operated when the air-conditioning functionality of the reactor building is lost in a hot shutdown. This may be, contrarily, perceived as allowing radioactive contamination to invade the building. More consultation and examination should be conducted with regard to the appropriate judgment under these unusual conditions.
c. Readiness for responding to simultaneous occurrence of multiple events

The accident in Fukushima imposed significant challenges in terms of the redundancy, diversity and independence of the various safety systems at a reactor facility when hit by a large-scale natural disaster. It also pointed at the possibility that an accident could simultaneously have a similar impact on multiple reactors at a single power plant as well as on neighboring nuclear power plants.

In particular, the explosions were largely responsible for complicating the interaction among the multiple units and neighboring plants. The debris strewn by the explosion at Unit 1 damaged the power cable that had been routed for supplying power to the distribution panel of Unit 2. One of the options for the restoration measure was thus ruled out. The explosion at Unit 3 brought the restoration work at Unit 2 back to the beginning. The explosion of the reactor building at Unit 4 is attributed to hydrogen mixed in from Unit 3, implying the possibility of inter-unit impact. Moreover, the incidents at Units 1 to 4 exerted an impact on the adjacent Units 5 and 6 by raising the radiation dose level in the vicinity of the plant. The restoration work at the Fukushima Daini plant, located about 12 km away, was similarly affected.

Additionally, the Fukushima Daiichi plant included three reactor designs: BWR/3, BWR/4 and BWR/5. Each of the six units had its own uniqueness, and this may be another factor that made the accident response more challenging. The Fukushima Daini plant employs only one reactor design, BWR/5, so the workers at the plant, in some cases, were able to successfully apply the response for one unit to the other units based on a prediction that the same events could happen to these units.

All the nuclear power plants in Japan, except Higashidori and Tokai Daini Nuclear Power Plants, are of multi-unit configuration. The nation must give due consideration to potential issues related to the peculiarities of these multi-unit plants. One possible option to mitigate the complexity is to allocate resources, materials and equipment among the constituent units in advance and build a response structure dedicated to each unit. That said, it will not be easy to manage the practical tasks of several response teams from a single emergency response center in the seismic isolation building. It is more realistic to find the best solution through repeated mock exercises.

More detailed examination is needed with respect to what events could cause ripple effects on neighboring units and neighboring plants, as these must be determined based on case-by-case assessments.

d. Safety goals to be applied to multi-unit plants

In Japan, “safety goals” are separately set to individual reactors. This approach may be unreasonable from the standpoint of local residents near a multi-unit nuclear power plant or when multiple plants are sited within the vicinity. Japan has several areas where two nuclear power plants exist within a 20 km radius. The people living in these zones are exposed to higher risks.

To achieve risk equitability from the viewpoint of these residents, a concept of setting more conservative safety goals for nuclear power plants should be examined for locations where reactors are concentrated.

6. Establishing redundancy, diversity and independence against large-scale disasters

The accident impressed on the world the lesson that the redundancy, diversity and independence, which were meant as defensive measures to a single failure, were utterly powerless against large scale natural disasters.

The collapse of a single pylon led to a loss of two off-site power systems. Flooding in a single room caused the failure of two pump systems. The switchgears collectively installed in a single room went down altogether, due to flooding. Consequently the presence or absence of off-site power sources, on-site emergency power supply and DC power had almost no significance.
What design can achieve viable redundancy, diversity and independence against large-scale natural disasters? The utilities must go back to the starting point and find a clear answer to this question.

7. **Appropriate design basis against natural disasters**

   **a. Design basis against earthquakes and tsunami**

   The maximum seismic acceleration observed at the Fukushima Daiichi plant and Onagawa Nuclear Power Plants at the time of the Great East Japan Earthquake exceeded the design seismic acceleration. Including these two cases, there have been at least five such cases on the record in Japan since 2005. This exceedance is anomalously high. Compared with major European countries where the exceedance frequency is set as lower than once in 10 thousand years, the Japanese design basis is extremely optimistic.

   The same applies to the design basis for tsunami. In consideration of waves caused by hurricanes, rather than earthquake-generated tsunami, the U.S. NRC has deterministically set “Probable Maximum Wave Heights” for the East Coast of the Mainland and the coast along the Gulf of Mexico (Regulatory Guide 1.59). The Probable Maximum Wave Height at the mouth of Chesapeake Bay, on the East Coast, is set as 6.8 meters, and that in the estuary area of the Mississippi River, where it enters the Gulf of Mexico, at 10.6m. The Diablo Canyon Nuclear Power Plant, which is sited on the west coast in California, where, like Japan, earthquakes occur frequently, had been given a maximum wave height of 10.7m, based on a conservative deterministic guidance. In 2010, a probabilistic tsunami hazard analysis was conducted with consideration given to submarine landslides. It verified that the estimated frequency of submarine landslides occurring near Diablo Canyon was nearly once in one million years. The safety measures formulated by the nuclear operator in charge of the plant include an additional margin on top of the conservative design basis. The utility provided snorkels with a height of 13.5m to protect the seawater pumps from being submerged, as they serve the ultimate heat sink.

   As opposed to these examples, the design basis setting approach in Japan lacks a conservative approach. And safety control has not been satisfactorily practiced by the nuclear utilities.

   The accident at the Fukushima Daiichi plant showcases a perfect example. The design tsunami height was determined as 6.1 meters in February 2009, based on re-evaluation results. But, in response, TEPCO simply reinforced the sealing of the seawater pump motor. This reinforcement may have been effective if the tsunami is merely a gentle rise of the sea level. In fact, the “height of tsunami,” “height of inundation” and “height of runup” are three different things: Height of tsunami < Height of inundation < Height of runup. It is widely known that the danger of tsunami is not just a matter of water flow but that it carries various suspended solids in the water and smashes them into objects on the way to its destination. TEPCO believed that the reinforcement of the sealing for the motor based on the design tsunami height, and not even the inundation height, was an effective and sufficient countermeasure against the motor being submerged by tsunami. It is clear that TEPCO’s position was far behind today’s global-standard principles of nuclear safety design.

   **b. Design bases for other phenomena and threats**

   Tornados are frequently observed in Japan these days, and damage from the natural phenomenon has been widely covered by the media. In the U.S., some nuclear operators have voluntarily installed “tornado relief vents” to protect the roof of the reactor building from tornados. The NRC stipulates the scale of tornados to be postulated in nuclear designs as to have a frequency of once in 10 million years (Regulatory Guide 1.76). In this scenario, the wind speed is assumed to be 103 meters per second in typical frequent occurrence zones, and a tornado missile—an automobile with a mass of 1,810 kilograms—collides at a velocity of 41m per second.

   On the contrary, there is no such thing as a “tornado relief vent” attached to the reactor buildings of nuclear power plants in Japan. If a tornado passes above the reactor building, its roof will be destroyed due to the large pressure difference. If fragments of the building or any other large flying object falls into the spent fuel pool and damages the structure, the water level will decrease and the stored spent fuels will be
exposed. Eventually, radioactive material may be released. The utilities must assess this risk and implement necessary preventive measures.

At the present time, typhoons are the only natural strong wind phenomenon considered in Japanese nuclear designs. The assumptions are based on existing meteorological records. Although tornados are not necessarily a new phenomenon, they are not factored in the design basis of any of the nuclear power plants in the country.

In this way, the utilities’ safety control should look at a wider spectrum that covers not only tornados, but fire protection design, internal flooding, cyber terrorism, and so forth, in addition to earthquakes and tsunami, in order to enhance the safety of existing plants. To achieve a higher level of safety at existing plants, the utilities should share design principles and good practices among plants and nuclear operators, so as to make constant improvements.

8. Issues identified from the perspective of counterterrorism measures
   a. Counterterrorism measures helpful for severe accident countermeasures

Some believe that the accident in Fukushima unintentionally provided potential terrorists who regard nuclear power plants as ideal targets of attack with vitally effective tactical suggestions. We must be aware of the fact that terrorists have learned that they could gain extremely advantageous negotiation conditions by artificially creating the same level of enormous damage as the impact brought about by the nature at the Fukushima Daiichi Nuclear Power Plant, or by blackmailing, following the creation of a setting similar to the time immediately before these perilous moments. Europe was quick off the mark to discuss countermeasures against possible terrorist acts and executed desktop exercises (EUROSAFE Forum, November 2011).

The U.S. had taken similar action prior to the accident, through in the reverse direction of causation according to a report released by the NRC’s taskforce in July 2011. Because of the measures implemented following the 9.11 incident, according to Clause B.5.b of an NRC order dated February 25, 2002, the U.S. nuclear power plants had already prepared for a possible severe situation with the concurrence of an SBO and DC power loss before the reactor accident occurred in Fukushima.

Terrorism is the third threat to nuclear safety after internal events and external events. The U.S. case above suggests that the fortified defence against the third threat automatically helps in the defence against the first and second threats.

This implication itself is not surprising. But there were a number of important items not embraced by Japanese nuclear power plants, and that obviously casts doubts on Japan’s enthusiasm for promoting nuclear safety. We are not saying with absolute certainty that, if these had been implemented, the accident could have been averted. But perhaps it would have been mitigated, at least.

In addition, the U.S. became aware of the necessity for further improving Clause B.5.b, after analyzing the details of the accident, and has already started taking action. In essence, countermeasures against internal events, against external events, and against terrorist attacks are not completely independent of each other; there are actually strong commonalities among them. Japan needs to practice nuclear safety promotional activities based on this recognition, to prepare itself for responding to possible contingencies in the future.

b. If safeguards against aerial terrorist attacks had been in place . . .

The Ordinance of Establishing Technical Standards for Nuclear Power Generation Equipment, a METI Ordinance under the Electricity Business Act, for which NISA is responsible, specifies in Article 4.3 as follows: Provided that there is a risk of an aircraft crash undermining the safety of a nuclear reactor, safeguards and other appropriate measures must be put in place. The provision appears to presumably correspond to the U.S. Code of Federal Regulations, 10CFR50.150 “Aircraft Impact Assessment.”

The purpose of this U.S. regulation is to assume aircraft crashes by terrorists, and require nuclear operators to implement response measures for aerial attacks in the design of future nuclear power plants. In accordance with this regulatory require-
ment, the ABWR, a candidate design for new nuclear power plant construction in the U.S., was redesigned to include a new water injection system. With this mechanism, while depressurizing the reactor pressure vessel, the reactor feedwater piping can be supplied with water directly from a special fireproof building built sufficiently far away from the reactor building, in case the reactor building is burning and inaccessible. This system uses a high-pressure pump with a capability commensurate with the high-pressure core spray pump. If such a system had been in place at the Fukushima Daiichi plant and survived the quakes and tsunami, the subsequent response would have very likely been improved.

Article 4.3 of the above ministerial ordinance of Japan was drawn up in a totally different fashion. The first half of the provision presents a precondition that “there is a risk of an aircraft crash undermining the safety of a nuclear reactor.” But by saying that such risk is substantially low through use of a probability theory, the precondition was nulled, making the “safeguards and other appropriate measures” in the last half of the provision unnecessary. This probability theory is given in the “Criteria for Probability of Aircraft Crash on Commercial Power Reactor Facilities,” formulated by the Nuclear Reactor Safety Subcommittee under the Nuclear and Industrial Safety Subcommittee of the Advisory Committee for Natural Resources and Energy. The secretariat of the subcommittee was installed in the Agency of Natural Resources and Energy, and the criteria were approved as appropriate by NSC. NISA approved the results of the assessment conducted by the nuclear operators based on this methodology and declared that the utilities had no need for implementing “appropriate measures” (Document dated June 17, 2010).

The decision paper was released after aerial terrorism had become a realistic threat. The concerted action of the above three parties in deriving such a conclusion may not have been unbiased. We believe the “appropriate measures” stated in Article 4.3 of the ordinance require more sincere and proactive discussions while referring to the initiatives taken by the U.S. Incidentally, to earn the trust of the public through a series of assessments and the decision-making process, involvement of a trusted independent organ is indispensable. In this as well, the concerted action of the three parties described above, was not appropriate.

2.2 Analyses and discussions on some issues

The accident is clearly attributable to the natural phenomena of the earthquake and resulting tsunami. Yet a number of important factors relating to how the accident actually evolved remain unknown, mainly because much of the critical equipment and piping that are directly relevant are inside the reactor containment vessel, and beyond the reach of on-site inspection or verification for many years to come. Despite this fact, in its interim investigation report, TEPCO attributed the main cause of the accident to the tsunami; it specified that no major damage from the earthquake to reactor facilities important for safety functions had been recognized—though they did add the conditional phrase “thus far.” The government also came to a similar conclusion in its accident report that was submitted to the International Atomic Energy Agency (IAEA). We conducted our investigations and hearings with great care, conscious of neither jumping to conclusions by intentionally screening out certain possible causal factors nor accepting simplistic measures. NAIIC believes there is a need for the regulators and TEPCO to investigate and verify causes of the accident based on the following facts:

1) A violent tremor struck the plant about 30 seconds after the SCRAM (the emergency shutdown of a nuclear reactor), and lasted for more than 50 seconds. Therefore, the activation of the ‘stop’ function did not necessarily mean that the nuclear reactors were protected from the earthquake motion. It is thought that the earthquake ground motion from the earthquake was strong enough to cause damage to some key safety facilities, because very few of the seismic backchecks against the design basis earthquake ground motions and

---

[107] Namely, Units 3 and 4 at South Texas Project.
[108] This new system is called Auxiliary Feedwater Injection (AFI).
[109] Such as the National Academy of Sciences of the U.S.
The reactor pressure and water level record before the tsunami hit makes it obvious that a massive loss of coolant accident (LOCA) did not occur immediately following the occurrence of the earthquake. However—as has been published by the Japan Nuclear Energy Safety Organization (JNES) in the “Technical Findings” composed by NISA—a small-scale LOCA, from small through-wall crack(s) in the piping and a subsequent leak of coolant, would not noticeably affect the variations in the water level or pressure of a reactor. If this kind of small-scale LOCA were to remain uncontrolled for 10 hours or so, tens of tons of coolant would be lost, leading to core damage or core melt.

3) The government-run investigation committee’s interim report, NISA’s “Technical Findings,” and TEPCO’s interim report all concluded that the loss of emergency AC power—which definitely impacted the progression of the accident—“was caused by flooding from the tsunami.” TEPCO’s report says the first wave of the tsunami reached the site at 15:27 and the second at 15:35. However, these are the times when the wave gauge set 1.5km offshore detected the waves, not the times of when the tsunami waves actually reached the plant. This suggests that at least the loss of emergency AC power supply A at Unit 1 might not have been caused by flooding. This basic question needs to be logically explained before making a final judgment that flooding was the cause of the station blackout.

4) Several TEPCO vendor workers working on the fourth floor of the nuclear reactor building at Unit 1 at the time of the earthquake witnessed a water leak on the same floor immediately after the occurrence of the earthquake. Two large isolation condenser (IC) tanks and their piping are housed on this floor. NAIIC believes that this leak was not due to water sloshing out of the spent fuel pool on the fifth floor. However, since we cannot go inside the facility and perform an on-site inspection, the source of the water leakage remains unconfirmed.

5) The isolation condensers (A and B systems) of Unit 1 were automatically activated at 14:52, but the operators of Unit 1 manually stopped both IC systems only 11 minutes later. TEPCO has consistently maintained that the explanation for the manual suspension was that “it was judged that reactor coolant temperature change rate could not be kept within 55 °C/hour (100 °F/hour), which was the benchmark provided by the operational manual.” The government-run investigation committee’s report, as well as the government’s report to IAEA, states the same explanation. However, according to several control room operators directly involved in the manual suspension of IC who responded to NAIIC’s hearing investigation, they stopped IC to check whether coolant was leaking from IC and other pipes because the reactor pressure was falling rapidly. The operator’s explanations are reasonable and their judgment was appropriate, while TEPCO’s explanation does not make sense.

6) In terms of the safety relief valves (SRVs) of Unit 1, there isn’t any “valve open/close record” to support that the SRVs really functioned properly in every phase of the accident in
which they were supposed to open or close (such records are available for Units 2 and 3). We found that the sound of the Unit 2's SRV moving was frequently heard in both the main control room and Unit 2, but no control room operator in charge of Unit 1 heard the sound of the Unit 1 SRV opening. There is therefore a possibility that the SRV did not work in Unit 1. In this case, a small-scale LOCA caused by the earthquake motion could have taken place in Unit 1.

2.2.1 Seismic ground motion at the Fukushima Daiichi Nuclear Power Plant due to the Great East Japan Earthquake

The maximum acceleration and duration of the seismic ground motion at the Fukushima Daiichi plant exceeded the standards of the earthquake resistant design on the foundation of the plant ground on the side of Units 1 to 4. Units 1 through 3, which were in operation at the time of the earthquake, were automatically scrammed. However, approximately 30 seconds later, strong tremors began shaking the plant hard; it lasted for more than 50 seconds, far longer than its design standards. Thus, although the “shut down” function worked, it is not clear if the “cooling” and “containment” functions were active. NISA presumes that the safety functions were kept unaffected despite the earthquake vibrations, but their argument lacks supporting evidence and is illogical and not convincing. Considering that the seismic backchecks for Design Basis Earthquake Ground Motions and the seismic reinforcement of the reactors were incomplete, it can be concluded that the seismic ground motions could have damaged important equipment and piping systems necessary for safety.

1. Outline of the earthquake

On March 11, at 14:46, a magnitude (M) 9.0 earthquake occurred off the coast of the Pacific side of the Tohoku region of Japan. (Officially named the 2011 Off the Pacific Coast of Tohoku Earthquake by the Japan Meteorological Agency, and hereafter called the “Great East Japan Earthquake” or “the earthquake”). The hypocenter was about 24km deep in the area 130km east southeast of the Oshika Peninsula, Miyagi Prefecture. The fault movement stretched in the northern and southern directions. The earthquake source fault length was about 450km north to south, and the width was roughly 200km east to west. The duration of the fault movement was approximately 180 seconds, during which seismic waves were constantly released.

Strong tremors shook a wide area over a long period. The seismic intensity on the Japanese seismic scale reached a maximum of 7 in Kurihara City in Miyagi Prefecture (equivalent to 11 “very disastrous” in the modified Mercalli scale in the US), and reached 4 or higher in the area from eastern Hokkaido all the way to the Chubu region (equivalent to 6 “strong” to 7 “very strong” in the Mercalli scale).

The intense upheaval of the ocean floor caused the tsunami, bringing particularly high waves to the coasts of Iwate, Miyagi, and Fukushima prefectures. Although the name, “the Great East Japan Earthquake Disaster,” implies damage caused by the earthquake, in fact, it was the tsunami that accounted for the majority of the approximately 20,000 fatalities and missing persons.

The tsunami hit the Fukushima Daiichi plant 40 minutes after the plant was shaken by the earthquake of an intensity of 6+ on the Japanese seismic scale (an intensity of 6.1 was recorded by the seismometer installed in Futabamachi, Shinzan, the nearest monitoring station to the plant). The plant was overcome by a large tsunami wave about 10 minutes later. According to GPS-based surveying and other measurements by TEPCO, the entire premises sank approximately 60 centimeters. We cover the tsunami in detail later on; in this section, we will focus only on the key observations related to the seismic ground

motion. Both the interim reports by TEPCO and the Investigation Committee on the Accident at the Fukushima Nuclear Power Plants of Tokyo Electric Power Company (Government’s Investigation Committee) only briefly mention the seismic movements. It is extremely essential, however, to accurately grasp their predispositions, in order to examine whether the earthquake damaged important equipment and piping systems.

2. Earthquake shaking on the reactor building basements

Table 2.2.1-1: Comparison of the observed maximum accelerations on the reactor building basements due to the Great East Japan Earthquake with the maximum response accelerations to the design basis earthquake ground motion (DBEGM) Ss for Units 1 to 6 of the Fukushima Daiichi Nuclear Power Plant

<table>
<thead>
<tr>
<th>Unit (Observation point)</th>
<th>Observed maximum acceleration</th>
<th>Maximum response acceleration to DBEGMs</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>North-south (NS)</td>
<td>East-west (EW)</td>
</tr>
<tr>
<td>Unit 1 (1-R2)</td>
<td>460</td>
<td>447</td>
</tr>
<tr>
<td>Unit 2 (2-R2)</td>
<td>348</td>
<td>550</td>
</tr>
<tr>
<td>Unit 3 (3-R2)</td>
<td>322</td>
<td>507</td>
</tr>
<tr>
<td>Unit 4 (4-R2)</td>
<td>281</td>
<td>319</td>
</tr>
<tr>
<td>Unit 5 (5-R2)</td>
<td>311</td>
<td>548</td>
</tr>
<tr>
<td>Unit 6 (6-R2)</td>
<td>298</td>
<td>444</td>
</tr>
</tbody>
</table>

Based on the interim report by TEPCO,[111] Table 2.2.1-1 indicates the observed maximum accelerations and maximum response accelerations[112,113] to the design earthquake ground motions Ss,[114] on the reactor building basements[115] of Units 1 through 6. The comparison between the acceleration response spectra of observed motions and the calculated response spectrum to the design basis earthquake ground motion for Units 1 to 3 will be provided separately (see Reference Material [in Japanese] 2.2.1-2. See also Reference Material 2.2.1-1 for the distribution of earthquake observation points at Fukushima Daiichi at the time of the earthquake.).

According to Table 2.2.1-1, the maximum accelerations of the east-west direction of Units 2, 3, and 5 exceeded the maximum response accelerations by 25 percent, 15 percent and 21 percent, respectively. The interim report of TEPCO states, “Although some were over the maximum response accelerations, most were below them.” It also states, “The actual [response spectra] exceeded the response spectra based on the design basis earthquake ground motion Ss in some period bands, but they were generally at the same level” And, it claims, “The earthquake motion was at about the same level as those assumed in the seismic capacity evaluation of the facilities.” In this way, the TEPCO report deems that there was no problem with the earthquake resistant design of the plant. However, from the viewpoint of seismic design, it is unacceptable that the actual accelerations even partly exceed the response accelerations of the design basis earthquake ground motion.

Another big problem is that, due to the malfunctioning of the seismic observation system, the recordings of all 18 components in Table 2.2.1-1 stopped approximately 130-150 seconds after the start of recording[116] (see Figure 2.2.1-1 (d) as an example). TEPCO’s report states that a complete record has been obtained by another nearby

---


[112] The maximum values among the calculated response values against Ss-1 to Ss-3

[113] See 1.1.5


3. Earthquake motion on the site basement

a. The design basis earthquake ground motion

Figure 2.2.1-1 (a) shows the observed wave form in the east-west (EW) direction at “the southern free field borehole seismic array,” in the southern part of the site (the area where Units 1-4 are located) at a depth of O.P. (Onahama Peil) -200m\(^{[118]}\) (see Reference Material [in Japanese] 2.2.1-3) for the vertical section of the observation point. The depth is almost the same as that of the “free surface of the base stratum” (O.P.-196m)\(^{[119]}\) used for setting the design basis earthquake ground motion (DBEGM). It is necessary to conduct “hagitori analysis” (calculating the earthquake ground motion on the hypothetical free surface of the base stratum by “stripping off” [hagitori] the effect of surface layers from observed seismograms) in order to compare the observed wave form with the waveform of the DBEGM.\(^{[120]}\)

Figure 2.2.1-1 (b) shows the EW component of the hagitori wave.\(^{[121]}\) Figure 2.2.1-1 (c) shows the DBEGM Ss-2H (horizontal component of Ss-2)\(^{[122]}\) for comparison. The maximum acceleration of the hagitori wave is 675 Gal, exceeding that of the DBEGM, 600 Gal. When comparing (b) and (c), another important point is that the duration of the considerably strong motion of the hagitori wave is about 120 seconds, or 50 seconds or more for strong motion (over 300 Gal) alone, while the overall duration of the DBEGM Ss-2H is only around 60 seconds, including 20 plus seconds of strong motion.\(^{[122]}\) This must have caused the entire nuclear power plant to go through “cyclic

---


\(^{[119]}\) See 1.1.4

\(^{[120]}\) The free surface of the base stratum is assumed to have neither surface layer nor structures above it. On the contrary, the vibration condition of the observed waves of -200m, which is covered by the surface layer, is different from that of the DBEGM. The “hagitori analysis” uses observed waveforms to estimate the ground motion on the hypothetical ground surface at -196 meters under an assumption where the surface layer does not exist. This analysis has some issues in general, but the issues are not covered in this report. The seismic wave obtained as a result of this analysis is called “hagitori wave (rock outcrop ground motion).”


---

As mentioned in 1.1.3, the DBEGMs include Ss-1H, which has a slightly longer duration maximum acceleration of 450 Gal, but the condition is not so different from Ss-2H (maximum acceleration, 450 Gal), has a shorter duration.
loading," making it vulnerable to fatigue fracture. Further, the earthquake motion had a tendency to magnify the “floor response spectrum,” [124] augmenting the impact on the equipment and piping systems on each floor of the reactor building.

We also developed the same illustration as Figure 2.2.1-1 for the observed waveform at the north free field borehole array, in the northern part (Units 5-6 side) of the site. See Reference Material [in Japanese] 2.2.1-4. Comparisons between the response spectra of the hagitori waves and the DBEGM at the south and north free field borehole arrays are also provided separately in Reference Material [in Japanese] 2.2.1-5. The comparisons reveal that the response spectrum of the EW component hagitori wave slightly exceeds those of the three kinds of DBEGM at the southern point.

At the northern point, on the other hand, the hagitori wave stays under the DBEGM in most of the cases in terms of both the waveform and the response spectrum.

b. Possibility of north-south difference in underground structure and site amplification characteristics at the Fukushima Daiichi Nuclear Power Plant

The fact provided by Reference Material 2.2.1-5 suggests a possibility that—although the southern part of the area in which Units 1-4 are located and the northern area in which Units 5-6 are located, are only 1 to 1.5km apart—there is a difference in the underground structure and the site amplification characteristics. This is also discernible when comparing the earthquake wave form at each of the five depths (between O.P. -300 m and the ground level) between the south and north free field bolehole seismic arrays. Particularly, regarding the EW-component earthquake ground motions at the depths deeper than O.P. -100m, those in the southern area are stronger than those in the northern area. NISA has also acknowledged the difference in earthquake ground motion between the northern and southern areas. [125]

Even on the northern side, the maximum accelerations in the shallower part and on the reactor building basemats are not small. Therefore, there is a possibility that there was a problem with the seismometers installed deep in the northern area, in addition to the complexity of the underground properties. However, considering all the discus-

---

[124] Response spectrum of the oscillation of each floor of the building against earthquake motion
sions above, the seismic records and the results of the earthquake response analyses of Units 5 and 6 must not be directly applied to Units 1-4.

4. Time of the SCRAM and subsequent long, violent earthquake ground motion
As indicated in Figure 2.2.1-1 (d), the exact time of the scram in Unit 1 was estimated to be around 14:47:33 seconds on March 11. This time of the scram is considered to be valid by deliberations in the Subcommittee on Earthquakes, Geology and Ground of the Niigata Prefectural Nuclear Power Plant Technical Commission for Safety Management. The obvious and significant point about the figure (d) is that the violent earthquake shaking struck the nuclear reactor building nearly 30 seconds afterwards. The horizontal axes of (a) (b) (d) in Figure 2.2.1-1 indicate the time elapsed since a certain point, and the time scale and time point have been aligned across the graphs (also for the acceleration scale on the vertical axes). The extremely important point is that, although the seismic recording on the basement of the Unit 1 reactor building stopped at around 140 seconds as mentioned in 2), it is highly possible that after the interruption of recording, there was a very large acceleration, judging from the time-history waveform of the rock outcrop ground motion (hagitori wave).

Although it may be suspected that the maximum acceleration occurred 150 seconds after the recording initiation, NISA has stated that "the seismic observation devices on the basement of the reactor buildings detected and recorded the maximum acceleration even after the recording was interrupted. For the units for which there are only interrupted records, analysis suggests that the maximum acceleration occurred before the interruption." However, according to the hearing, NISA only repeated the explanation by TEPCO and did not confirm for themselves how the seismic observation devices acquired the correct maximum accelerations. Although the earthquake motion became slightly weaker after the scram, violent motions shook the reactor building about 30 seconds later and lasted for over 50 seconds. In other words, it appeared that the earthquake motions became weaker at the time when the recording stopped, but a severe shaking then hit 10 seconds afterwards. A similar situation likely occurred at nearby Units 2-4. Comprehensive research and examination must be conducted to find out exactly what happened during this long, violent earthquake motion. Until then we cannot easily conclude that "the reactor was able to withstand the strong earthquake ground motion because the emergency shutdown worked."

5. Problems with the earthquake response analysis based on the observed records
Based on the results of the earthquake response analyses reported by TEPCO, and the field investigation of Unit 5, NISA has presumed that the equipment and piping important to retaining safety functionality were not damaged at any of the units of the Fukushima Daiichi Nuclear Power Plant. However, NISA's conclusion lacks

[126] See 1.1.2.1
[127] Ishibashi, Katsuhiko. Figure 1 in Gempatsu Shinsei - Keisho no Kiseki (Earthquake-Nuclear Combined Disaster: Warning Tracks) (Nanatsumori Shokan Inc. 2012) [in Japanese]. reading of changes in reactor output data in the transient event records that is publicized by TEPCO.
[128] The time of scram and other issues at Fukushima Daiichi were discussed, and explained by TEPCO in responses to questions by a Subcommittee member, at the Subcommittee's 27th meeting on August 30. According to the meeting minutes and handouts, the scram had presumably started at around 47 minutes and 31 seconds in Unit 1, at around 47 minutes and 32 seconds in Unit 2, and at around 47 minutes and 29 seconds in Unit 3, and had finished within 3.5 to 5 seconds as designed. Niigata Prefecture, "27th Jishin, Chishitsu/Jiban ni kansuru shou iinkai (The 27th meeting regarding earthquake, geology and ground)," [in Japanese]. Accessed May 4, 2012, www.pref.niigata.lg.jp/genshiryoku/27jisingiji.html.
[130] Hearing with NISA official.
The earthquake response analyses were conducted using the observed records for the basemats of the reactor buildings as input values, with respect to the reactor buildings, the turbine buildings, and the seven major facilities\footnote{Isolation condenser system piping of Unit 1, primary loop recirculation system of Unit 1, vent pipe / downcomer / ring header of Unit 1, vent pipe / downcomer / suppression chamber of Unit 2, core spray system piping of Unit 2, and high pressure water injection system piping of Unit 3.} of class S in seismic design which have the ‘shutdown, cool, and contain’ functions and additional six facilities\footnote{See Table 1.1.1-1. Here, the units are compared as of both the reactor installation permit application date and the operation commencement date.} at all units. The results showed that the calculated values fell within the evaluation standards; for this reason the report states that safety was supposedly maintained both during and immediately after the earthquake.

TEPCO used Unit 5 as a representative unit—since it was the only unit unaffected by both the tsunami and the hydrogen explosions, and therefore accessible for an on-site investigation—and conducted earthquake response analysis to the DBEGMs, after screening all the devices and piping of class S in seismic design, other than the seven major facilities. The result of the analysis showed that the calculated values were within the evaluation standards, excluding some piping and piping support. Regarding the piping and piping support where the calculated values exceeded the evaluation criteria, TEPCO and NISA carried out a visual inspection and confirmed that there was no damage; accordingly, they reasoned that the safety functions had been maintained. However, such results and conclusions are very unreliable, as more detailed investigations, such as NDT (Non Destructive Testing) were not conducted (see 1.1.5, 5).

NISA has claimed that the interrupted records can be used as valid data for input of the earthquake ground motion for analysis. However, even if it may be acceptable to use the interrupted record only for Unit 6, of which the earthquake record has been examined, it should not be applied to Units 1-4 in the southern area. And as Unit 1 is five to seven years older than Unit 5,\footnote{In general, the largest earthquake in the prominent seismic activities within a concentrated time and space is called the “main shock.”} it would be utterly illogical to conclude that Unit 1 was not damaged because Unit 5 suffered no damage.

NISA explicitly states that TEPCO’s final reports on the seismic backcheck of the Fukushima Daiichi Nuclear Power Plant had not yet been submitted nor assessed by the government, and as a result, seismic reinforcement work against the DBEGMs had not been conducted. NISA clearly says the evaluation of some piping and piping support (with the use of the response spectrum of the DBEGMs) shows that the calculated values exceeded the evaluation standards. As was stated in 1.1.5, it is most remarkable that the seismic reinforcement work had not been conducted at Unit 1, leaving this unit the least robust against earthquakes. Therefore, the judgments based on the analyses and on-site inspection described above are generally meaningless.

NISA states that no result has been obtained from plant parameter examinations and plant behavior analyses showing damage to the basic safety functions of the facilities of Units 1 to 4. But this is a separate issue from the earthquake ground motion, and will be discussed later in 2.2.2.

### 6. Aftershocks

Innumerable aftershocks followed immediately after the main shock\footnote{In comparison to the direct aftershocks (aftershocks in a narrow sense) which occur along the fault plane of the main shock (plate boundary surface), these earthquakes are brought on inside the Pacific plate east of the Japan Trench, the subducted Pacific plate, and a shallow part of the continental plate.} and still continue to occur even as this report is being written. The aftershocks have been occurring over an area with a length of 500km and width of 200km, mostly corresponding to the source region of the main shock, which stretches from east of Iwate Prefecture to Ibaraki Prefecture. There have been many earthquakes that have occurred in the area surrounding the source region as well.\footnote{The same as the interim reports on the seismic backcheck; reactor pressure vessel, main steam system piping, primary containment vessel, residual heat removal system piping, residual heat removal system pump, reactor core support structure, control rods (evaluated for insert ability).} For a detailed table of the aftershocks, see
According to witnesses at a NAIIC hearing,\[^{[137]}\] the work in the main control room for Units 1 and 2 at the Fukushima Daiichi plant was often disrupted by aftershocks. The aftershocks mentioned may correspond to the ones recorded at 14:51, 14:54, 14:58, 15:05, 15:12, 15:15 (the largest aftershock according to the latest records) and at 15:25, which registered a seismic intensity of 4 on the Japanese scale at Shinzan in Futaba town. It is believed, however, that there was little possibility that these aftershocks damaged the equipment and piping or escalated the damage already incurred, because the maximum acceleration at the site was only 43 Gal or lower (see Reference Material [in Japanese] 2.2.1-6). But the possibility cannot be ruled out that the aftershocks caused or augmented damage to the upper floors, which sway to a larger extent than the lower floors. The aftershocks may have caused objects that had been damaged or become unstable, due to the main shock, tsunami and explosion, to topple or fall.

\[2.2.2\] **Possibility of damage to important devices due to the earthquake motion**

The Fukushima Daiichi nuclear power plant was struck by “prolonged, violent earthquake motions” due to the Great East Japan earthquake. As was discussed in detail in 2.2.1, the intensity of the earthquake was around the same level as the DBEGM Ss in the new guideline, but the duration of the strong tremors was exceptionally long. It is unclear whether the Fukushima Daiichi plant had enough robustness against an earthquake, since seismic backchecks had not been conducted on the plant. In addition, the long-lasting tremors may have increased the number of seismic cyclic loading applied to the important piping, and as a result, so called “metal fatigue fracture” may have appeared in the piping. But no one can enter the containment vessel to investigate what really happened. It may be possible to deduce what could possibly happen and what could not happen through fault tree analysis (FTA). The FTA conducted on Unit 1 by the Japan Nuclear Energy Safety Organization (JNES) has indicated that it cannot be denied theoretically that a small-break loss-of-coolant-accident (SB-LOCA) may have occurred in Unit 1. If a SB-LOCA is left for a long period of time, it may develop into reactor core damage or core meltdown.

\[1.\] **Small-Break Loss-of-Coolant-Accident (SB-LOCA)**

The inability to directly inspect the site of the accident has made it extremely difficult to investigate the physical cause of the accident. Almost everything that is necessary to investigate the cause lies inside the containment vessel, which cannot be directly accessed by investigators. The inside of the containment vessels can be viewed through the use of cameras and small robots that only allow an understanding of the general conditions. Thus, it cannot be determined which pipe, out of the numerous piping which run up and down the containment vessel, was affected by the earthquake motions and caused an SB-LOCA. An SB-LOCA can occur when a pipe is cracked completely through. In order to find the crack, all of the insulation and steel covering must first be removed, followed by a careful inspection of the pipe surface. Such an inspection, however, will not be possible for many years to come.

We collectively refer here to the various types of important piping directly connected to the reactor pressure vessel—including the main steam piping, feed water, recirculation inlet and outlet piping, ECCS piping, and IC piping – as “reactor piping.” If cracked, the coolant (light water) gushes from the piping and a loss-of-coolant-accident (LOCA) will occur. The degree of the LOCA depends on the type of pipe and the level of damage. If a complete break (guillotine rupture) occurs in a pipe with a large diameter, it will be a large-break LOCA (LB-LOCA).\[^{[138]}\] Even if the pipe is large in diameter, when the fracture is a small trough-wall crack, it results in a SB-LOCA. Additionally, when a pipe has a medium sized through-wall crack, the result is a medium-break LOCA (MB-LOCA).

The only thing almost certain is that neither an LB-LOCA nor an MB-LOCA occurred at Units 1 through 3 as a result of the earthquake motions of the Great East Japan

---

\[^{[137]}\] Hearing with workers who were on-site at Fukushima Daiichi at the time of the accident

\[^{[138]}\] LB-LOCA means Large Break LOCA. SB-LOCA means Small Break LOCA. MB-LOCA means Medium Break LOCA.
earthquake. If an LB or MB-LOCA had occurred, the water level and pressure in the reactor would have fallen rapidly in a short period of time, but this was not observed in the time between the earthquake and the time of the total station blackout SBO in the data released by TEPCO. An SB-LOCA, however, can occur without a drastic decrease in water level or pressure in the reactor; therefore no one can decisively conclude, based only on the published plant operation data, that an SB-LOCA never occurred.

2. “Fault Tree Analysis:” effective in accident cause analysis

Fault Tree Analysis (FTA) can be used as one of the ways to examine the possibility of an SB-LOCA immediately after the earthquake even if the containment vessel cannot be accessed to check the reactor piping. This is done by conducting an accident progression analysis for particular piping, postulating that it has small cracks of various sizes and comparing the results with the records of the actual reactor water level and pressure.

Conducting the FTA would be extremely effective in identifying the causal factors of the Fukushima Daiichi nuclear power plant accident, but TEPCO has yet to publicly announce if an FTA is being considered.

On the other hand, JNES did conduct a series of FTA at the request of NISA since last summer. The results of the analyses were discussed at the “Forum for Opinions on Technical Knowledge” established in October 2011 by NISA, and were compiled and publicly released on March 2012. Table 2.2.2-1 shows the FTA data put together by JNES[139] and the main points of the discussion and evaluation of the “rapid pressure drop in the Unit 1 reactor,” which inquires why the pressure fell rapidly from approximately 6.8 MPa to 4.5 MPa in the 11 minutes from the automatic start of the IC in Unit 1, which occurred at 14:52.

Generally in an FTA, the actual events which occurred (in this case, the rapid pressure drop in the reactor) are called “top events,” the potential causal factors are listed up in detail, and potential causes of the top events are analyzed in detail. As the process of analyzing all potential causes would take a large amount of time and work, those which are most likely to be the cause are examined first.

In Table 2.2.2-1, the “top events” are indicated in yellow and the events which have been analyzed are in blue.

3. FTA of Small-Break Loss of Coolant Accident (SB-LOCA) in Unit 1

A number of important results were found in the FTA by NISA and JNES. Figure 2.2.2-1 shows the areas in which the leaks occurred (the areas in which the coolant leaked in its vapor phase and the liquid phase) according to the FTA (see Table 2.2.2-1). There are five places in which the leak could occur: the drain pipe in the IC (D-1), the steam pipe (D-2), the B recirculation pipe on the side connected to the IC drain pipe (B), the A recirculation pipe on the side which is not connected to the IC drain pipe (A), and the main steam pipe (C).[141]

In this FTA leak analysis, for each of the five assumed leak points indicated in Figure 2.2.2-1, two to three different leakage areas (the size of the cracks) are hypothesized, as is indicated in Table 2.2.2-2.

The leak analysis uses the plant dynamic behavior analysis code “RELAP5 MOD3.3.”[142]

[141] See 2.2.4, 2 for a detailed explanation of the isolation condenser (IC).
[142] NES implemented the latest version of RELAP5 MOD3 released by NRC in 2005.
### Table 2.2.2-1: FTA of rapid decrease in reactor pressure in Unit 1 at Fukushima Daiichi Nuclear Plant: FTA of the rapid decrease in nuclear reactor power

<table>
<thead>
<tr>
<th>Top events</th>
<th>Analyzed case</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rapid decrease in nuclear reactor pressure (following the rise in reactor pressure due to the earthquake and closure of MSIV)</td>
<td>No damage</td>
</tr>
<tr>
<td>Malfunctioning of IC capability (excessive cooling)</td>
<td>Intermittent use of IC (automatic/manual)</td>
</tr>
<tr>
<td>Lowering of pressure due to use of other systems</td>
<td>Rapid increase in heat exchange</td>
</tr>
<tr>
<td>IC steam piping leakage</td>
<td>Erroneous opening of safety relief valve (automatic)</td>
</tr>
<tr>
<td>RPV (steam phase)</td>
<td>Opening of safety relief valve (manual)</td>
</tr>
<tr>
<td>Main steam piping leakage (incl. MSIV)</td>
<td></td>
</tr>
<tr>
<td>RPV piping small leakage (small LOCA)</td>
<td></td>
</tr>
<tr>
<td>RPV (liquid phase) small leakage</td>
<td></td>
</tr>
<tr>
<td>IC return line leakage</td>
<td></td>
</tr>
<tr>
<td>Abnormally rapid cooling of primary system</td>
<td></td>
</tr>
<tr>
<td>Rapid increase in flow of water supply (condensation of air bubbles [void])</td>
<td></td>
</tr>
<tr>
<td>Erroneous activation such as ECCS (condensation of steam)</td>
<td></td>
</tr>
<tr>
<td>Erroneous activation of recirculation (condensation of air bubbles [void])</td>
<td></td>
</tr>
</tbody>
</table>

### Evaluation Results

#### Nuclear reactor pressure behavior

It is possible to simulate the rapid decrease in reactor pressure based on knowledge of the characteristics of the devices and the design of the primary system and IC of the nuclear reactor.

<table>
<thead>
<tr>
<th>Improved performance due to the earthquake is unlikely</th>
</tr>
</thead>
</table>

The nuclear reactor pressure is not high enough to meet the minimum pressure necessary to open the safety relief valve. (However, there are no records taken by the transient recording device)

The results of the analysis which simulates the use of the safety relief valve shows that the residual heat of one valve is larger than the capacity of one IC, the change in nuclear reactor pressure is larger than when the IC is in use. Also, the valve must be opened and closed frequently in order to keep the pressure within a certain range. However, there is no data which shows such pressure behavior.

The analysis which simulates a small leakage in the gas phase part (IC steam piping) shows that the amount of heat exchange decreases but the decrease in the reactor pressure due to the residual heat from the steam leakage is larger. The increase in pressure after the closure of the MSIV is gradual.

The effect on the reactor pressure vessel (RPV) pressure is determined by the balance of the steam due to the leakage and the steam due to the decayed heat, and can be considered to be the same as the simulated effect of the steam leakage of the main steam pipe mentioned below.

The analysis which simulates a small leakage in the gas phase part (main steam piping) shows that once the leakage is formed, if the leakage is large, the increase in reactor pressure is gradual at the time of the closure of the MSIV.

The analysis which simulates small leakages of 3 square centimeters, there is not much decrease in reactor pressure in the early stages because enough steam is produced from the decay heat. For this reason, it is necessary for the IC (or SRV) to be in use. Thus, the existence of a small leakage cannot be assumed based on the reactor pressure behavior alone.

A small leakage in the RPV liquid phase part affects the RPV pressure depending on the steam produced due to the leakage amount and the decay heat. It is the same as the small LOCA mentioned above.

A small leakage in the IC return line results in a large decrease in reactor pressure. This is because a large amount of heat is exchanged due to the increase in the IC flow and the heat removal is speeded up, and because the steam from the RPV is released. (Similar to the leakage of the gas phase part).

When the AC power is lost, the water pump stops and there is no water supply. A sudden increase in the water supply is not possible. A sudden decrease in air bubbles (void) due to an increase in water supply is not possible.

There is no injection of water to the reactor from ECCS, etc. Thus, a change in pressure cannot be caused by the injection of water by the ECCS.

The recirculation pump stopped, did not have any power when the AC power was lost, and it did not start again.

As the average power range monitor (APRM) was already at zero (there was only decay heat), there cannot be any further loss of power. (A sudden decrease in steam production and decrease in pressure is not possible).

#### Nuclear reactor water level behavior

It is possible to simulate the nuclear reactor water level behavior based on knowledge of the features of the devices and the design of the primary system and IC of the nuclear reactor.

<table>
<thead>
<tr>
<th>Improved performance of the IC due to the earthquake is unlikely</th>
</tr>
</thead>
</table>

The nuclear reactor pressure is not high enough to meet the minimum pressure necessary to open the safety relief valve. If the safety relief valve was opened, the nuclear reactor water level would change in a serrated form. However, the actual recorded data do not show such records.

The analysis which simulates the use of the safety relief valve shows that the nuclear reactor water level gradually decreases as the safety relief valve is opened. However, the actual recorded data do not show such tendencies.

The analysis which simulates a small leakage in the gas phase part (IC steam piping) shows that the nuclear reactor water level gradually decreases due to the gradual draining of the RPV inventory amount. (Almost the same as the leakage in the main steam piping mentioned below).

The effect on the RPV inventory is determined by the amount of water supply (no supply after the loss of power) and the amount of steam leaked, and can be considered to be the same as the simulated effect of the steam leakage in the main steam piping mentioned below.

The analysis which simulates a small leakage in the gas phase part (main steam piping) shows that the nuclear reactor water level continually decreases once the leakage is formed. On the other hand, the actual recorded data do not show a decrease in the water level until the opening of the SRV.

The analysis which simulates a small leakage of 3 square centimeters shows the nuclear reactor water level gradually decreases. A simulation of the 0.3 square centimeter leakage shows a decrease in the water level. However, the actual recorded data does not show a decrease in the water level before the SRV is in use.

In the case of a small leakage in the RPV liquid phase part, the effect on the RPV coolant inventory is the same as the above. The decrease in the water level is the same as the above as well.

The analysis which simulates a small leakage of 3 square centimeters shows the nuclear reactor water level gradually decreasing. A simulation of the 0.3 square centimeter leakage shows a decrease in the water level. However, the actual recorded data does not show a decrease in the water level before the SRV is in use.

When the AC power is lost, the water pump stops and there is no water supply. It is not possible for an increase in water supply to cause a change in the air bubbles (void) or a sudden change in the RPV inventory.

When the AC power is lost, the water pump stops and there is no water supply. It is not possible for an increase in water supply to cause a change in the air bubbles (void) or a sudden change in the RPV inventory.

There is no injection of water to the reactor from ECCS, etc. (If there was an injection of water, it should be apparent in an increase in the nuclear reactor water level)

There is no injection of water to the reactor from ECCS, etc. (If there was an injection of water, it should be apparent in an increase in the nuclear reactor water level)

The recirculation pump stopped, did not have any power when the AC power was lost, and it did not start again.

The recirculation pump stopped, did not have any power when the AC power was lost, and it did not start again.

As the average power range monitor (APRM) was already at zero (there was only decay heat), there cannot be any further loss of power. (A decrease in the APRM and a change in the amount of reactor air bubbles (void) and subsequent change in the water level) is not possible.

---

**Table continued from previous page**

**Table 2.2.2-1**: FTA of rapid decrease in reactor pressure in Unit 1
4. SB-LOCA with a leakage size smaller than 0.3 square centimeters is not contradictory to the actual change in water level and pressure in the reactor

Figures 2.2.2-2~4 show a comparison of the results of cases D-1, A-3, C-3 in Table 2.2.2-2 and the actual measured values, regarding the reactor pressure and water level. These cases are hypothesized to have a rather large leakage area of 3 square centimeters, and the analysis shows the water level in the reactor falling rapidly. However, as the water level in the analysis differs significantly from the actual measured water level, it can be concluded that the earthquake did not cause such significant damage to the IC piping (the drain piping), recirculation piping (the side not connected to IC), and the main steam piping (although NAIIC’s report does not cover all cases, this applies to other cases as well).

On the other hand, Figures 2.2.2-5 and 2.2.2-6 show the results when the leakage area is set at one-tenth the size, at 0.3 square centimeters. As these figures indicate, when the leakage area is small, there is hardly any difference between the analysis results and the actual monitored values in both the water level and the pressure in the reactor. Even if there was damage to the piping at the time of the earthquake and there was an SB-LOCA with a leakage of less than 0.3 square centimeters, it would be realistically impossible to presume or deny such a leakage based on the monitored changes in pressure and water level in the reactor.
Figure 2.2.2-2: Leakage of 3 square centimeters in IC drain pipe (Case D-1)

Figure 2.2.2-3: Leakage of 3 square centimeters in recirculation pipe (Case A-3)

Figure 2.2.2-4: Leakage of 3 square centimeters in main steam pipe (Case C-3)

On the other hand, the analysis shows that in case A-2, the loss of coolant was 2,000 cc per 1 second, despite the extremely small size (0.3 square centimeters) of the leakage area. At this rate, the loss of coolant would be 7.2 t per hour, and 72 t every ten hours. This significant loss of coolant could result in fuel damage within 10 hours.[143]

5. Ordinary earthquake response analysis cannot be used to analyze accident causes
As is discussed in detail in 1.1.5, the seismic backchecks outlined in the revised Guide (Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities 2006) were not conducted in Units 1 to 6 of the Fukushima Daiichi plant. Table 2.2.1-1 shows that the maximum acceleration level of the basemat of the nuclear reactor building is around the same as the DBEGM Ss.

As is discussed in 2.2.1.5, TEPCO chose Unit 5 (which was not affected by a hydrogen explosion) as the representative unit of the Fukushima Daiichi plant on which to perform its earthquake response analysis. Although the analysis shows that some of the piping and piping supports faced events exceeding the evaluation standards, TEP-
CO claims that the visual inspection found no damage. However, it should be noted, considering that TEPCO used Unit 5 to represent Units 1 to 3, which are five to seven years older, that even if there were no problems with Unit 5, it cannot be presumed that there would be no problems with Units 1 through 3.

Because Units 1 to 3 cannot be accessed, it was extremely inappropriate for TEPCO to use the “ordinary” earthquake response analysis to examine the effect of the earthquake motions on the reactor piping and the piping supports. The reason is simple: when theoretically inferring the soundness of the piping—i.e. when discussing the possibility of the reactor piping breakage due to the earthquake ground motion – the analysis is conducted based on the unconditional assumption that all of the piping support structure has not been affected by the earthquake.

The situation inside the containment vessel is unknown. There is a possibility that something completely unexpected occurs, which is common in the event of an accident. The support structure for the piping may come loose and been damaged by the long, violent earthquake motions, for example. In that case, the seismic load applied to the piping would be different, with the possibility of damage to the piping. It is necessary to examine various possibilities using many different cases. It is also necessary to use various damping coefficients, which have a large influence over the earthquake response analysis, in the sensitivity analysis.

In accident analysis, there is a need to examine various possibilities. The cases TEPCO used in its earthquake response analysis were meant for use in designing the plant and conducting seismic backchecks, not for accident analysis.

Moreover, there have been criticisms pointing out problems with the earthquake response analysis itself. The drastic improvement in computer software does not necessarily guarantee that the response of the actual situation will be correctly predicted.

**6. Possibility of SB-LOCA at Unit 1 cannot be denied**

Based on what NAIC heard from the operation workers, if there was damage to the piping system in the reactor as a result of the earthquake motions on March 11, and a SB-LOCA occurred, it most likely occurred at Unit 1.

According to material NAIC obtained from TEPCO, a control room operator at Unit 1 heard a sound that he described as “unnatural.” At the NAIC hearings, the operator said that he had heard the sound immediately after the scram, but also stated that at that time the IC had automatically started. The scram began at 14:47 and the automatic start of the IC occurred five minutes later at 14:52. So it is likely that the sound was heard just before 15:00. At that time, another worker asked, “What is that rumbling sound?” and the operator replied, “Isn't it the sound of the IC [the sound of the steam coming out of the exhaust piping]?” At the request of the operator, the other worker opened the back-side door to the main control room and confirmed that the sound was from the two exhaust ports of the IC (also called the pig's snout).

It is unlikely, however, that this was the cause of the sound. The IC was stopped manually at 15:03 and had been in operation for only 11 minutes. When the IC was stopped, the temperature of the water in the tank was only 70°C. At this temperature, it is unlikely that vapor or steam would come out of the IC exhaust ports, so the possibility that the sound came from the IC is quite low. Then what caused the sound?

Someone wrote on the white board for Unit 1 in the main control room that “there was a hissing sound from the hallway side,” but it is unclear what the sound was, what time it was heard and who heard it. In the NAIC hearings, more than one worker has stated that they did not know who had written it on the white board.

On the other hand, it is strange that no one in Unit 1 heard the sound of the SRV being operated, which is more likely to have been heard. This will be discussed in more detail in 2.2.4-2.
2.2.3 Relation of tsunami and total station blackout

1. Judgment of past reports
All publicly released investigative reports on this accident regard the loss of the emergency AC power supply, which significantly worsened the accident, as having been caused by the flooding by the tsunami.

The interim report by the Government’s Investigation Committee is a prime example:

On 15:27 and 15:35 of March 11, the tsunami came to Fukushima Daiichi Nuclear Power Plant twice and overran the emergency seawater pumps which were set on a platform 4m high. The tsunami also overran the 10m and 13m platforms, flooding the reactor building, turbine building and many of the facilities. Although Units 1 to 6 were receiving AC power from the emergency diesel generators, due to the tsunami, the seawater pumps for cooling the water-cooled emergency diesel generators and a number of emergency diesel generators were submerged (excluding 2B of Unit 2, 4B of Unit 4, 6B of Unit 6), and almost all of the power panels were damaged by the flood. Because of this, between 15:37 and 15:42, all of the AC power was lost in Units 1 to 6, with the exception of the air-cooled diesel generator in Unit 6 (6B).

Although the specific phrases used in the explanation vary, the technical knowledge compiled by NISA, the report submitted to the IAEA by the Japanese government, and the interim report by TEPCO all present similar conclusions.

2. Fundamental error in tsunami arrival time in past reports and monitored data
All of these past reports took their data from the TEPCO report, which states that the first wave arrived at 15:27 and the second wave arrived at 15:35. However, it must be taken into account that these records were taken by a wave gauge that is located 1.5km off-

---

[147] NAIIC added the time lines of the first and the second waves of the tsunami on the graph in NISA Presentation 2-1-1 for the discussion panel “Jishin, Tsunami ni kansuru Iken Choshukai (Hearing regarding earthquake and tsunami),” October 5, 2011 [in Japanese].
These are the arrival times for the point 1.5km offshore, not the arrival times of the tsunami waves at the Fukushima Daiichi Nuclear Power Plant.\footnote{A written reply from TEPCO, May 15, 2012}

The only data for the monitored tsunami at the Fukushima Daiichi Plant shows that the first wave was only around 4m high and the second wave was much higher. The height of the second wave is unclear, as the wave gauge was limited to monitoring waves at a maximum of around 7.5m.

**3. Conditions necessary for the tsunami to have been the cause of the loss of the AC power supply**

As the reactor buildings in which the emergency diesel generators are stored are 10m high in Units 1 to 4, and 13m high in Units 5 and 6, it would be unlikely for a tsunami significantly lower than 10 meters to flood the reactor buildings. On the other hand, the seawater pumps of the diesel generators are placed on 4-meter-high platforms located near the ocean, and there is a possibility of damage if the water reaches 1.6m above the platform. If the seawater pumps stop, the diesel generators, which are cooled by the seawater, also stop.\footnote{A stop signal of the diesel generator will be sent as the sea water pump keeps its discharge pressure under fixed value or less for 60 seconds (for 10 seconds only for Unit 3), after the pouring water is taken away from the electronic motor of the sea water pump; a written reply from TEPCO, February 27, 2012.} However, the air-cooled diesel generators (system B in Unit 2, Unit 4 and Unit 6) and the water-cooled diesel generators of system A in Unit 1 for which no stop signal mechanism is installed will not stop, even if the seawater pumps are damaged by flooding.\footnote{A written reply from TEPCO}

It is impossible for the tsunami to have been the cause of the loss of the AC power supply for system A of Unit 1, system B of Unit 2 and system B of Unit 4 unless the second wave arrived before the AC power was lost. The tsunami could not have been the cause of the loss of power from the other power sources either, unless the second wave arrived before the power was lost, or the seawater pump was stopped due to flooding damage from the first wave. At this point, there are no reports which have investigated this in detail.

**4. Further examination required of the cause of the loss of the AC power supply**

The series of photographs of the second tsunami wave shows that the second wave came from the eastern direction.\footnote{Part of 44 pictures released by TEPCO on May 19, including 11 pictures taken from the waste central treatment building located on the south side of Unit 4. The significant pictures of them will be published as reference documents.}\footnote{According to NAIIC hearing, someone said that he/she saw his/her PHS reading 15:39 at the parking underneath Shiomizaka on the north side of Unit 1, and that he/she went up Shiomizaka to escape from the second tsunami, which was 10 meters tall.} In the end, however, the wave coming from the southern direction first reached the ocean area near Unit 4. There was an interval of 56 seconds between two photographs, one which shows the wave reaching the ocean area near Unit 4, and the other which shows the wave reaching the tip of the breakwater. It takes 70-80 seconds for a tsunami to move the distance of 800m at a depth of 10 meters between the location of the wave gauge and the tip of the breakwater. Hence, it is likely that the second tsunami that passed the point 1.5km offshore at 15:35 reached the ocean area near Unit 4 at around 15:37. It also took some time for the tsunami to move forwards and submerge the emergency power generation devices on the 10m high platform.

It is also likely that the seawater pumps\footnote{Part of 44 pictures released by TEPCO on May 19, including 11 pictures taken from the waste central treatment building located on the south side of Unit 4.} in Units 1 to 4 were not stopped due to flooding damage from the first tsunami wave. The series of photographs taken at the time of the first tsunami wave\footnote{According to NAIIC hearing, someone said that he/she saw his/her PHS reading 15:39 at the parking underneath Shiomizaka on the north side of Unit 1, and that he/she went up Shiomizaka to escape from the second tsunami, which was 10 meters tall.} shows that the bottom part of the wall of the Unit 4 building on the shore.\footnote{It is general knowledge that it takes about two minutes for a tsunami to travel 1.5km across an ocean with a depth of 10 meters. According to the written reply from TEPCO mentioned in footnote #147, TEPCO’s simulation result estimated that the tsunami took about two and a half minutes to travel 1.5km from the wavemeter installation point to the point the tsunami hit the shore.}
4m-high platform was still visible. According to a crewman of a ship that was in the harbor and workers who sought refuge by moving from the eastern side of Unit 3 towards Unit 1, the wave did not completely pass over the breakwater from the eastern side.

The tsunami was, therefore, not the cause of the loss of the power in system A of Unit 1, which occurred at 15:35 or 15:36\[155\] according to the NAIIC hearings.\[156\] It is also questionable whether the tsunami was the cause of the loss of power in system B of Unit 1 and in system A of Unit 2, which occurred at 15:37, or the cause of the loss of power in both systems A and B of Unit 3, which occurred at 15:38. At this stage, when the emergency power devices have yet to be thoroughly investigated, we cannot rush to the conclusion that the SBO would not have occurred without the tsunami.

2.2.4 Issues which need to be examined

1. Leaked water in the nuclear reactor building of Unit 1

TEPCO's subcontracted workers in the reactor building of Unit 1 have reported in interviews that there was a gush of water on March 11, right after the earthquake occurred at 14:46. They were working near the water gush.

a. “Water rushed in at me, sliding like a ‘Tatami-mat’”

According to the interviews with the subcontracted workers, the water leakage occurred in the area near the southern wall on the fourth floor of the reactor building. On the same floor, there were two large IC tanks and complex IC piping.

When the gush of water occurred, four subcontracted workers were installing the scaffolding necessary for inspections on the switchboard on the same floor. NAIIC interviewed two workers, A and B, who are from different subcontractors, on different days. Although the two workers' accounts differ in some details, they are generally consistent.

According to B, he shouted to the other workers to stay where they were, as the earthquake tremors were getting stronger. After this, water gushed into the area near the southern wall of the reactor building. At this time, B was standing a little away from the wall, with his back to it. On the left, there was a five-meter-square opening in the floor for moving large devices and equipment objects between floors and a jib crane (a small, fixed rotary crane). B states that water rushed at him in the form of a “tatami-mat” from the upper right side. He thought, “If we get wet, everything will be over,” so he shouted, “Run!” to the other workers, and ran between the two IC tanks and down the stairs on the north side with some other workers. He was hurrying, and does not recollect if the water was cold or hot, or if steam came out with the water.

A heard B shout, “Stay!” but A ran between the IC tank and the containment vessel and grasped a handle attached to a nearby pipe to support himself from the tremors. After he heard a voice (in B’s direction) shout, “Run!” he saw water gushing from above at a 45 degree angle, and hastened to escape past the tank and down the stairs on the north side.

b. The cause of the gushing water has yet to be identified

On the fifth floor, the very top of the spent fuel pool was exposed. There is a possibility that the origin of the gushing water was overflow from the spent fuel pool. It is estimated that the pool water was shaken strongly by the earthquake (causing sloshing) and overflowed onto the floor, spilling to the fourth floor\[157\]. It is possible that the

---

[155] TEPCO stated on May 30, 2012 that TEPCO heard from the same witness subsequent to NAIIC hearing, and that the witness reversed the testimony made at NAIIC hearing. This point is further discussed and explained in Reference Material [in Japanese] 2.2.3.

[156] The tsunami can hardly be a cause given that the A system was shut down before the B system at Unit 1 and considering the locations of both system. This point is further discussed and explained in Reference Material [in Japanese] 2.2.3.

[157] TEPCO's Kashiwazaki Kariwa Nuclear Power Plant was damaged in the Niigataken Chuetsu-oki Earthquake in July 2007. The fuel pools overflowed from sloshing at Unit 1 through 7. Especially at Unit 6, the radiation-contaminated pool water eventually flowed to the outside of the plant through the fuel exchanger cable penetration point and drainage equipment in the uncontrolled area. After this event, a one-meter high fence was installed around the spent fuel pools at all units of Kashiwazaki Kariwa, Fukushima Daini, and Fukushima Daini Nuclear Power Plant.
water spilled from the fifth floor to the fourth floor through the opening in the floor, but this contradicts with B’s narrative. He stated that he was standing almost right below the opening and that the water that came gushing through was from his right.

As there are many ventilation openings at the top of the wall of the spent fuel pool, it is possible that the water overflowed into the ventilation openings to the exhaust duct and to the fourth floor.

As stated in 2.2.4 2, the issue of whether the IC piping was damaged by the earthquake movement has been raised numerous times. There is a complex IC piping system on the fourth floor of the nuclear reactor building where the gushing water was witnessed, and part of it extends close to the spot.

Thus NAIIC informed TEPCO that, in spite of the risk of being exposed to a certain level of radiation, NAIIC wanted to conduct an on-site inspection of the fourth floor (TEPCO was not told the purpose of the inspection). Entering the reactor building for inspection is incredibly dangerous, as the interior of the building is pitch dark even in daytime due to the lack of lighting, wreckage from the hydrogen explosion is everywhere, and there are large openings for moving equipment in each floor. TEPCO informed NAIIC that, because accompanying NAIIC members into the building would subject their workers to unnecessary radiation exposure, TEPCO personnel would not enter the building. After much consideration, NAIIC gave up on the idea of investigating the interior of the nuclear reactor building.

So at this point, the only conclusion that NAIIC can come to is that immediately after the earthquake, there was a gush of water near the southern wall of the fourth floor of the nuclear reactor building of Unit 1 that TEPCO and NISA need to thoroughly investigate.

TEPCO must have been aware that there were subcontracted workers working there at the time of the earthquake, and the TEPCO Fukushima Nuclear Accidents Investigation Committee should have immediately interviewed the workers. But workers A and B stated that they had not been interviewed by TEPCO prior to NAIIC’s interview with them about gushing water.

2. Problem with isolation condenser (IC)

Seeing that the pressure in the nuclear reactor was falling rapidly, the control room operators of Unit 1 manually shut down the IC in order to check if there was any piping leakage and to control the falling pressure. The direct reason of the manual shutdown of the IC is not that the “reactor coolant temperature change rate could not be kept within 55°C/hr (100°F/hr)” as TEPCO claims. The manual shutdown was conducted according to the appropriate judgment and cooperation of three operators. On the other hand, as the site cannot be entered and thoroughly investigated at this point, it is impossible to judge whether the earthquake motions could have caused small fractures in the IC piping system, which could have then led to a small loss of coolant accident.

a. Why the control room operators manually shut down the isolation condenser (IC)

(i) Role and operating principles of the IC

Normally, the nuclear power plant generates power by using the nuclear fuel in the nuclear reactor to boil water and create a large amount of steam (at a pressure of approximately 6.8 MPa, temperature 285°C) that is sent through the main steam piping to the turbines and generators to produce electricity. However, at 14:47, the main

[158] After the accident, worker A asked TEPCO several times about the source of the flooding water, and the possibility of radiation exposure, and so forth. But TEPCO ignored worker A for quite some time, finally conducting an inspection for internal radiation exposure at the end of June in response to worker A’s inquiry.

[159] January 18 and February 13 in 2012

[160] At the Fukushima Daiichi Nuclear Power Plant, each operation team for Unit 1 and 2, Unit 3 and 4, and Unit 5 and 6, is an 11-person team consisting of one duty operator director, one duty operator deputy director, two duty operator managers, one duty operator deputy manager, two propulsion machinery operators, and four auxiliary machinery operators. Unless it is necessary to distinguish among them, the operation team members are referred as “operators” in this report.
steam isolation valve (MSIV) suddenly closed, and there was no place for the massive amount of steam inside the reactor pressure vessel to vent, and the pressure began to rise in the vessel. Immediately after the reactor scram, the fission product decay heat was especially high, and the reactor pressure rose rapidly. At 14:52, the IC (see Figure 2.2.4-1) sensed the rise in reactor pressure and automatically turned on.

According to TEPCO, it was the first time that the IC automatically started and was ever used in Unit 1 since it started operation in 1971.

The IC in Unit 1 was a vestige of the early days of boiling water reactors (BWRs) and only Unit 1 in the Fukushima Daiichi plant has an IC.\[161\]

As shown in Figure 2.2.4-1, the IC facility is composed of two trains, A and B. Each train is composed of the condenser tank which contains the cooling water, the steam piping which leads the steam from the top of the reactor pressure vessel to the condenser tank, and the drain piping which leads the water, formed when the steam is cooled and condensed in the condenser tank, to the recirculation piping at the bottom of the reactor pressure vessel, and four motor operated (MO) valves.

Of each of the four valves of trains A and B, valve 3A in the A train and valve 3B in the B train are always closed during operation. In contrast, the other valves (valves 1A, 2A, 4A, 1B, 2B, 4B) are always open. However, valves 3A and 3B are designed to open automatically if, for any reason, the reactor pressure continues to stay at more than 7.13 MPa for over 15 seconds (for example, if the MSIV suddenly closes).

When valves 3A and 3B open, the high temperature and high pressure steam from the reactor pressure vessel goes through the steam piping into condenser tanks A and B, which are installed outside the containment vessel. In the condenser tanks, heat is exchanged between the steam and the cooling water, so that the steam is condensed into water, which has a lower temperature than the original steam. As the volume is greatly reduced when steam is turned into water, the reactor pressure decreases. The water that comes out of the condenser tanks A and B goes through the drain piping into the containment vessels, combining at point P in the diagram, entering the recirculation pump of the loop B recirculation piping (point Q in the diagram), and going back into the nuclear reactor pressure vessel.

The most prominent characteristic of the IC is that the above process is possible through "natural circulation," without the use of a special pump. For natural circula-

---

\[161\] In Japan, it has been also installed at the oldest BWR, namely Japan Atomic Power Company Tsuruga Unit 1 which started commercial operation in 1970.
tion, tank A and B are placed at almost the same level as the very top of the reactor pressure vessel. Also, the water level of the reactor does not change greatly, as the coolant circulates in a closed loop (reactor→steam piping→condenser→drain piping→reactor).

(ii) Issue of the manual shutdown of IC at 15:03

Figure 2.2.4-2 shows the reactor pressure recorded by the pen recorder, starting from slightly before the scram following the earthquake, to the SBO which occurred 50 minutes later. According to the record, the reactor pressure was approximately 6.8 MPa while Unit 1 was still in operation shortly before the earthquake. After the automatic scram occurred due to the earthquake (Point 1 in Figure 2.2.4-2), the air bubbles (void) of the reactor coolant were crushed and the reactor pressure decreased, but as the MSIV closed, the reactor pressure started to increase (Point 2). Then as the reactor pressure reached the benchmark of 7.13 MPa, the automatic IC started at 14:52 [162] (Point 3) and the reactor pressure began to decrease. However, approximately 11 minutes later at 15:03, the falling reactor pressure began to rise again (Point 4). TEPCO has stated that the reason for the rise in pressure is that the workers in the main control room shut down the IC by manually closing the 3A and 3B valves. As previously stated, reactor pressure rapidly increases if the IC is shut down, especially when there is a large amount of steam as a result of decay heat immediately after the scram. TEPCO’s explanation that the rapid rise in reactor pressure occurred because of the manual shutdown of the IC is not problematic. However, the question remains why the IC was shutdown manually 11 minutes after it was automatically started.

In the 11 minutes from 14:52 to 15:03 during which the IC was working, the reactor pressure fell rapidly from approximately 6.8 MPa to 4.5 MPa. It is questionable wheth-

---

[162] The time stated is according to TEPCO, ‘Kakushu Sosa Jisseki Torimatome (Summary of various operation results),’ May 16, 2011 [in Japanese].
er this decrease was normal. Furthermore, the IC and other piping may have been damaged by the lengthy, strong earthquake motion, and the coolant may have leaked from the fractured areas. These are serious questions to consider when examining the development of the accident in Unit 1. Justifiably, the Government’s Investigation Committee discusses this issue at length in its interim report.\[163\]

(iii) The explanation that they observed the operation rule of the reactor coolant temperature change rate of 55°C per-hour reactor is irrational

Regarding the manual shutdown of the IC, TEPCO has claimed in many places—including on its website, in press conferences, and in its reports—that the control room operators manually shut down the IC in order to observe the operation rule that the per-hour reactor coolant temperature change rate must be kept within 55°C.\[164\] TEPCO’s Interim Report,\[165\] released on December 2, stated:

*The operating manual states that the IC should be operated so that the per-hour reactor coolant temperature change rate should not exceed 55°C in order to lessen the effect on the nuclear reactor pressure vessel. After the temperature of the IC fell rapidly, the shutdown was conducted according to the directions in the manual.*\[166\]

The Government’s Investigation Committee accepted TEPCO’s view without question.

According to Table 37-1 in Section 1 of Article 37 of TEPCO’s “Nuclear Reactor Facility Safety Regulations at Fukushima Daiichi Nuclear Power Plant,” the per-hour reactor coolant temperature change rate should not exceed 55°C. At 15:03 on March 11, as the reactor pressure was falling rapidly, the control room operators presumed that if the two trains of IC system were used to cool the reactor, the coolant temperature would fall rapidly, exceeding the coolant temperature change rate provided in the safety regulations, and that it would not be possible to abide by the regulations. Then, they stopped manually both trains (trains A and B), closing the drain pipe isolation valves (MO-3A, 3B) of the trains.\[168\]

The investigations by both TEPCO and the Government’s Investigation Committee in essence explain that the decision to manually shut down the IC was based on the judgment that it would not otherwise be possible to follow TEPCO’s requirement that the per-hour reactor coolant temperature change rate must be kept within 55°C. However, the facts below make clear that this explanation is illogical. TEPCO’s position of clinging to this view is reason enough for doubt, raising suspicions that there may have been problems with the IC or damage to the IC system piping.

The reason that the IC started automatically in the first place is to control the reactor pressure, which rose due to the sudden closing of the MSIV. And it is, of course, TEPCO itself that set the IC autostart conditions. TEPCO should have been aware of how the reactor pressure and coolant temperature would change when both trains A and B automatically started at the same time.

If the IC was manually shut down because the per-hour reactor coolant tempera-

---


\[164\] There are two main reasons to adhere to the maximum coolant temperature change per hour of 55°C. The first is to avoid potential damage to instruments and piping by additional excessive thermal fatigue. The second reason is to prevent brittle fracture on the core in the reactor pressure vessel by a rapid temperature change. The limit of the rate temperature change follows the “100 degrees Fahrenheit” rule created in western countries as a rule of thumb of the operational experience on thermal power plant and chemical plants, and not a rule created especially by a scientific theory. It is simply to operate softly by minimizing the temperature differences between instruments and piping.


\[166\] In reality, the subject procedural manual (i.e. ‘Procedural manual for MSIV shutdown’) does not include a statement about 55°C in a applicable section. Therefore, the statement by TEPCO, “the operation has been conducted according to the procedural manual,” is considered to be close to a false statement.

\[167\] “The Technical Specification for the Nuclear Reactor Facility at Fukushima Daiichi Nuclear Power Station” is established by TEPCO as required by Electric Utility Industry Law. The limit value of 55°C is mentioned in the guideline.

ture change rate could not be kept under 55°C, it is likely to be attributable to a defect of the cooling capacity of the IC, which was too strong, or the damage of the piping of IC system. TEPCO’s explanation that it manually shut down the IC in order to adhere to the 55°C limit is evidently self-contradictory. A more logical reason for the manual shutdown of the IC is required.

Additionally, the reactor coolant temperature change rate of each moment is not shown in the main control room, either in a graph or in words. If the control room operators wanted to know the reactor coolant temperature change rate for a certain period of time, they would have had to calculate it based on the change in the reactor pressure during that time. However, interviews with the operators have made clear that, the workers did not conduct any such calculations after the IC had automatically started.

(iv) Operators conducted the manual shutdown in order to confirm whether there was leakage in the piping

NAIIC conducted numerous interviews with the control room operators who were working at Unit 1. Below is a summary of an account of an operator who was involved with the operation of the IC.

After the strong tremors of the earthquake occurred, the strongest I had ever experienced, the operators in the main control room of Unit 1 lay down on the floor to protect themselves. As the tremors lasted a very long time, the operators looked up at the operation board and pointed at the blinking lamps while still lying flat on the floor. They also confirmed the automatic start of trains A and B of the IC system during this time. Afterwards, as the operators dealt with the operations, I was notified by an operator that the reactor pressure had fallen drastically, from 7 MPa to 4.5 MPa. I stopped the IC in order to gain control of the reactor pressure. After the reactor pressure was under control, as directed in the operating manual regarding the closing of the MSIV, the IC was manually started and stopped so that the reactor pressure was kept to between 6 and 7 MPa. The train A was used while the train B remained stopped. At that point, I was confident that a cold shutdown could be accomplished according to the manual. Yet, although the operators followed the manual when possible, they could not refer to it all the time. The operators had undergone simulation training at the BWR operation training center. However, there had been no simulation training for Unit 1, so they had not received simulation training for the IC. All the control room operators were well aware of the 55°C limit, and since changes in the temperature correspond to changes in the reactor pressure, they tried their best to be sensitive to changes in the temperature. However, the IC was not stopped based on the rate of change in temperature. It was stopped in order to gain control of the reactor pressure.

Below is another important statement, obtained on another day of hearing, from a Unit 1 worker who was directly involved with the manual shutdown of the IC. This is a definitive statement regarding the shutdown of the IC and appears almost without any editing. The words within brackets were added by NAIIC.

Hearing that the IC was functioning [from other workers] I told the supervisor, “Since the reactor pressure has decreased, I want to confirm if there are any leakages. As the pressure is falling rapidly, the pressure vessel will not be kept in proper order. I want to stop it in order to check if there are other leakages. Is it alright to do so?” Since the reactor pressure was falling, the reactor coolant temperature change may also have been in danger, and the reactor pressure may have been falling in places other than the IC. If the IC was stopped and the reactor pressure recovered, it would mean that there were no other leakages. I wanted to check, so I wanted to stop the IC in order to do so. When asked if the IC could be stopped, the supervisor gave me permission, so I said, “Close the IC valve once.”

[169] NAIIC has conducted numerous hearings of operators at the Fukushima Daiichi Nuclear Power Plant from March 6 to April 27, 2012.

[170] TEPCO, “Genshiro Sukuramu Jiko / Genshiro Sukuramu / (B) Shu Joki Benhe no Baai (Scram trouble of nuclear reactor/Scram of nuclear reactor/(B) When the main steam valve is closed),” in Igo-ki Jikojji Unten Sosa Tejunshe Jioso Bessu (Accident Operation Manuals of Unit 1 at Fukushima Daiichi Nuclear Power Station [phenomenon base]), February 5, 2011 [in Japanese]
As written above, the manual shutdown of the IC at 15:03 was based on the appropriate judgment and cooperation of three workers, including the supervisor. The direct reason for the manual shutdown of the IC was not the reactor coolant temperature change, but that was in order to check if there was leakage in the piping, to regain control of the reactor pressure, to go back to following the manual and to eventually achieve a cold shutdown.

The important point for the manual shutdown of the IC was the confirmation of leakage, not the “within 55°C” rule. In explaining the manual shutdown of the IC, TEPCO focused on the rule that the reactor coolant temperature change rate must be within 55°C in order to avoid using the phrase “confirmation of leakage,” which could mean the involvement of problems with piping damage from the earthquake.

b. Was the piping of the IC system damaged by the earthquake?

The Government’s Investigation Committee has reported various results from their investigation regarding the IC over many pages of their interim report dated December 26, 2011. In one of the investigation topics, titled “Possibility of the IC piping rupture of the IC right after the occurrence of the earthquake”, the Government’s Investigation Committee’s conclusion was to completely deny the possibility of damage by the earthquake, based on the following three reasons.

First, the IC piping features a “rupture detection circuit” which automatically shuts off the valves of the IC system as a failsafe function, so that the IC would not have worked after the earthquake if there was a rupture. Second, if there was a rupture, then the reactor pressure and water level would have decreased rapidly. Third, if there was a rupture of the IC piping outside of the reactor containment vessel, then steam containing a large volume of radiation would have leaked through the rupture, resulting in “a situation involving worker fatalities.”

The rupture detection circuit was designed to function when the IC piping had been completely ruptured, and would not be activated by a Small Break LOCA (“SB-LOCA”) of the piping. A rapid decrease in the reactor pressure and water level would be caused only by the Large Break or Medium Break LOCA, and usually not by the SB-LOCA (see 2.2.2 for details). The third reason itself is erroneous. Even if the IC piping ruptured, the amount of radiation emitted in the surrounding area would not be large enough to immediately affect human lives, because there would not be so much radioactive material contained in the coolant all the time. There would be a large amount of radioactive material in extremely limited circumstances, such as a case where the nuclear fuel rods have been significantly damaged by the earthquake and discharged a fission product into the coolant prior to the piping rupture.

TEPCO reported in their interim report of the investigation that “there was no damage to the main body of the IC, a rupture in the piping, a leak from the flanges, or damage to the valves,” as a result of their visual inspection. TEPCO also publicly released several photographs of the visual inspection—seen on attachment 6-8 (3). However, as seen in the photos, most of the piping cannot be visually observed directly as it was covered by insulation and metal cover. A narrow and small crack that might cause a SB-LOCA cannot be easily found by a rough visual inspection. Some IC piping is located inside the containment vessel, but the visual inspection by TEPCO was conducted only on the piping on the exterior of the containment vessel.

In conclusion, even if the earthquake motion never caused a large scale break on the IC piping which would activate the rupture detection device, as any inspection of the containment vessel interior is still not feasible at this stage, there are no grounds for denying the possibility of SB-LOCA as a result of coolant leakage from a small, narrow crack on the IC piping caused by the earthquake.

c. Did the IC system function after the SBO?

(i) Were the isolation valves of the IC closed by the failsafe function?

The investigation by the Government’s Investigation Committee regarding the opera-

tion of the IC was very detailed. Many statements in the report contain valuable information, but the discussion on the failsafe function of the IC system is not acceptable.

The Government's Investigation Committee, along with TEPCO and NISA, concluded that all of the IC valves (i.e. 1A through 4A and 1B through 4B) were closed right after SBO, because the DC-driven piping rupture detection device ceased to function due to a loss of DC power, triggering the remittance of a signal to be safe. NAIIC does not agree in defining the remittance of such a signal to be safe as a "failsafe" feature, or to the view that the feature was actually triggered as designed. The reasons are:

Equipment that is meant to be "failsafe" should not be designed only to trigger the signal of the failsafe feature; the design should consider the entire composition of the equipment so there is a failsafe function throughout. For instance, important points include the consideration of the equipment's power source, and a default feature where the safety function is launched by a passive mechanism in case the control signal is lost. Examples of equipment with a passive mechanism are air-driven valves and electromagnetic valves, which fulfill their failsafe function through the passive momentum of included parts such as springs when the opposite side of the balance is lost, as air pressure for air-driven valves or an electromagnetic field for electromagnetic valves. Specific examples include scram valves and MSIV.

The Government's Investigation Committee's point of view on the meanings of both "fail" and "safe" is subjective. "Fail" would certainly include a loss of DC power, but also should include a situation in which the detective circuit does not function when a rupture in the piping is not recognized by the system. Action to isolate is safe in the case of a piping rupture, but rather unsafe if the IC is isolated when the IC should be in-service. The concept used in designing reactors is not as simple as the feature considered "failsafe" by the interim report of the Government's Investigation Committee. Both erroneous actions and erroneous inactions must be included when considering the design concept. This is achieved by a logic circuit that uses signals from multiple detectors, such as 2-out-of-3 or 2-out-of-4. The reactor protection system (RPS), which starts the scram process, employs this concept.

All eight of the isolation valves of the IC (i.e. 1A through 4A and 1B through 4B) are electric valves, so they are not "failsafe" in the case of a power loss but are in a condition described as "fail as is." The degree of the valve opening in the case of a loss of power is directly dependent on the degree of the valve at the time of the loss—whether it was completely open, completely closed, or somewhere in between. The degree of the valve position in such a case is completely remote from all signals, with or without any intentions from a signal system.

As the final safety analysis report (FSAR) of Oyster Creek Nuclear Power Plant states (see Reference Material [in Japanese] 2.2.4-1), isolating the IC by operating the isolation valve is not always defined to be safe. The report states some circumstances where the isolation of the IC should be bypassed instead. The "failsafe" concept accepted by the Government's Investigation Committee is an arbitrary definition, whereas the actual design was not intended to be truly "failsafe."

If the definition of "failsafe", as assumed by Government's Investigation Committee, is to automatically close all the isolation valves of the IC system in case of a loss of DC power, then AC power needs to be supplied even after the loss of DC power, because both MO-1 (i.e. 1A or 1B) and MO-4 (i.e. 4A or 4B) are AC motor operated valves. This situation is the reverse of the SBO, which assumes a loss of all AC power, but not DC power. The DC power can be supplied from batteries, or by the battery charger driven by the AC power supply, so as long as there is sufficient AC power, DC power will be available. In other words, a loss of DC power implies a prior loss of AC power. This makes the "failsafe" function defined by the Government's Investigation Committee impossible to achieve in principle. The AC power source to drive the AC motor operated valves MO-1 and MO-4 is limited only to the off-site electricity source or the on-site emergency diesel generators. There is no doubt that all off-site electricity power sources were lost right after the earthquake. As to the on-site diesel generators, there were records implying a loss of them at or prior to the arrival of the second wave of the tsunami (see 2.2.3), indicating that the AC power had been lost prior to the DC power, with no evidence indicating the opposite had occurred.

The actual process of the loss of electric power to the main control room of Unit 1 after the tsunami was as follows. It was reported that lighting and illumination of the
gauges, and the control system were gradually lost from 15:37 to 15:50 on March 11. Eventually, lighting on the operation status indicators for the HPCI and the IC were lost. What provided electrical power to the main control room in this situation for a while was not the power supply from the emergency diesel generators, but the 120V vital AC power supply created by the emergency batteries via an inverter. This electric power was not enough to drive the AC electric valves of the IC, namely the MO-1 and MO-4, because they required 480V three phase AC power.

There is no possible scenario proving the Government's Investigation Committee's presumption that “for an unknown reason, the AC power kept working even after the loss of DC power.”

(ii) The truth about what made the IC dysfunctional

The Government's Investigation Committee based their reconstruction of the process of losing the function of the IC on an unnatural and unrealistic scenario, in which DC power was lost prior to the loss of AC power. Their reconstruction of the process was as follows: a signal of damage (i.e. of rupture) on the IC was remitted; the failsafe design wrongfully closed almost completely the AC motor operated isolation valves MO-1A and MO-4A, which are inside the D/W; a loss of AC power followed; and the IC system thereby fell into the dysfunction state. It argues that, thereafter, opening the isolation valves outside of the D/W, namely MO-2A and MO-3A, would not have improved the situation in any way. It goes on to state that a concentrated effort to depressurize the reactor and low pressure water injection by D/D-FP should have been conducted as soon as possible.

We could agree with the conclusion of the Government's Investigation Committee only if we knew that their assumed unnatural scenario actually took place. However, we are very skeptical about a realistic occurrence of such a scenario.

The reason that the IC system (A) did not respond properly to the operator actions subsequent to 18:18 on March 11, was not because MO-1A and MO-4A were disabled at the closed position by the failsafe feature, but because the natural circulation had been stopped by the IC narrow tubes being clogged with non-condensable hydrogen, which was created from the zirconium-water reaction in conjunction with the damaged reactor core at a high temperature without coolant water. Since this reason does not contradict other known facts, we consider this the real cause of the IC becoming dysfunctional.

(iii) Reply from the plant makers to the questions from NAIIC

We stated above that the direct cause for the improper response to the operative actions on MO-1A and MO-4A subsequent to 18:18 on March 11 is that a head drop of the condensed water needed to drive the natural circulation in the IC had been lost because the narrow tubes became clogged with hydrogen created from the zirconium-water reaction in the midst of the progressive core damage.

Such a circumstance can easily come to mind if the principle mechanism of the IC is understood. As a matter of fact, under normal operation, vent lines at the top of the IC steam piping are specifically used to continuously send steam to the downstream of the MSIV in order to prevent non-condensable hydrogen and oxygen gases generated by the radiolysis of the reactor water from clogging the piping.

To clarify this, we asked two domestic BWR plant makers for their opinions regarding this possibility.

The replies from the two plant makers were what we had expected, and were the same in principle: that once the IC piping is clogged, the IC subsequently becomes inoperable.

In response to a question on how to revive the function of the IC under those conditions, one maker simply answered that it would be impossible to revive as such a situation was not considered in the design basis. The other maker responded with a conceptual remodeling method of reviving the IC in such a situation, which implies that there was no way to revive the IC at the time.

We also sent questions in writing to TEPCO on how to revive the IC under those conditions, asking whether TEPCO had considered excreting hydrogen through the vent line to the downstream of the MSIV. TEPCO replied that this methodology would be very dangerous as it might cause a hydrogen explosion.
3. Did the SR valve of Unit 1 go into action?

Though it is difficult to ascertain, according to the plant data released by TEPCO and the results from our hearing surveys with control room operators, the safety relief valves (SRVs) of Unit 1 at the Fukushima Daiichi plant might have never (or almost never) gone into action during the accident progression. If that were true, some reactor system piping (meaning the pipes directly connected to the reactor pressure vessel) might have been broken by the earthquake motion immediately after the occurrence of the earthquake. The subsequent small-break LOCA, combined with the misfortunate coincidence of a station blackout (SBO), may have evolved into the fuel damage and meltdown.

a. Two possible scenarios for the loss of coolant at Unit 1

The cause of the core melt that occurred at Unit 1 at the early stage of the accident was the sudden loss of coolant (lightwater) in the reactor pressure vessel. Basically, there are two possible accident scenarios in which this sudden loss of coolant could have occurred.

Figure 2.2.4-3 schematically depicts the water and steam systems of Unit 1 at 14:47, the moment when the main steam isolation valve (MSIV) was closed immediately after the earthquake had occurred.

![Figure 2.2.4-3: State of Unit 1 at the moment when the main steam isolation valve (MSIV) was closed. The pressure in the reactor starts to rise soon after, due to the decay heat of the fission products. The inverse triangles in black show that the valves are closed; those in white mean that the valves are open.](image)

**Accident Scenario 1** The loss of coolant was considered to have stemmed exclusively from the open-close motions of any of the several SRVs (see Figure 2.2.4-4). This was the scenario TEPCO and NISA essentially believe in. The pressure inside the reactor rose due to the decay heat of the fission products because the MSIV was closed. For about 50 minutes, at the minimum, before the SBO occurred, however, the pressure was kept below 7.13 MPa by the isolation condenser (IC). There is practically no data on the pressure and water level in the reactor for the several hours following the SBO. This makes it difficult to definitively identify what was controlling the pressure during those hours. Most likely, the pressure was automatically controlled, mainly by the SRV, starting from relatively soon after the SBO. This is the actual reason why the sudden loss of coolant occurred in the following sequence.

Initially, the reactor pressure rose to 7.7 MPa, which caused a SRV to automatically

[173] Although a possibility of having the two scenarios taking place at the same time exists, this possibility is not a part of our investigation.
As a result, an enormous amount of steam in the pressure vessel flew into the suppression chamber (S/C) all at once, where it was condensed to water (see Figure 2.2.4-4). The volume significantly condensed then, lowering the reactor pressure. As the reactor depressurized, the SRV was automatically closed. As the SRV was closed, the reactor pressure rebounded to 7.7 MPa due to the decay heat. The SRV then opened, allowing the vast amount of steam inside the pressure vessel to make headway to the S/C . . . and the whole process started again. Every time the SRV was opened, a large amount of coolant was transferred from the pressure vessel to the S/C. Consequently, the water level in the reactor sharply fell, and the fuel ended up getting damaged and melting.

**Accident Scenario 2** Part of the reactor piping was ruptured immediately after the Great East Japan Earthquake (just before or after the MSIV was closed), due to the prolonged intense earthquake motion. The coolant burst through the rupture into the drywell (D/W). It moved through the vent tube, vent header and downcomer and joined the water in the S/C (see Figure 2.2.4-5). The loss of coolant through the rupture resulted in the fuel damage and core melt. In this scenario, the reactor pressure would not go up because of the broken pipe, so it is not very likely that the SRV was automatically actuated. TEPCO and NISA almost completely deny this scenario.

b. No evidence supporting the activation of the Unit 1 SR valve

Units 2 and 3 are equipped with a system that automatically records the opening and closing motions of the SRVs. As a matter of fact, the plant data TEPCO disclosed on May 16, 2011 tell when the SRVs of Units 2 and 3 were opened and closed. There is no such record for Unit 1, however. The system to automatically record SRV motions was not in place at Unit 1. Hence, there is no objective data that proves that the Unit 1 SRV opened and closed repeatedly.

NAIIC detected an unexpected yet extremely important fact during our hearings with the control room operators at the Fukushima Daiichi plant. They mentioned that

---

[174] The SR valve of Unit 1 comprises two functions: a function as a “relief valve” which opens at a pressure of approximately 7.3 MPa and a function as a “spring-loaded safety valve” which opens at a pressure of approximately 7.7 MPa. The relief valve function, however, mechanically requires power supply. Thus, the SR valve is estimated to have functioned only as the spring-loaded safety valve after the SBO.
it was so quiet in the main control room and the reactor building, particularly after the SBO, that they only heard the voices of operators. Those in charge of Unit 2 said that, in the dark silence, they heard “sounds of the SRV.” We quote them:

(i) The SRV of Unit 2 was very frequently in motion and I heard a loud banging noise each time.

(ii) It sounded like an earthquake or rumbling. It was more like a heavy buzzing noise than a banging noise.

(iii) All the sounds heard in the main control room came from Unit 2. No sound from Unit 1.

(iv) The interval between the buzzing sounds was not that short. I heard the next sound after a while.

(v) We took turns going to the Unit 2 site (reactor building). When I went there, I heard that sound a lot more than several times.

At another hearing, we asked the interviewees if the SRVs make a sound when they are in action. An operator for Unit 5 replied, “I heard the sound on the site when I was younger, but have never heard it from the main control room.” Another operator at the interview said, “I manually opened the SRV at Unit 4 a long time ago . . . I think it was . . . for a test. I recall that I experienced a thud kind of pulse in the control room. It was exactly when the steam flew into the suppression chamber. But I probably felt it that strongly because I was paying careful attention in a quiet setting.”

Based on these statements, we could conclude that operating sounds of an SRV can be heard at the site as well as in the main control room if it is quiet. In a group interview we

---

[175] Operators for Unit 1 and Unit 2 were engaged in the operation of the respective units in a spacious control room without partitions.

[176] According to TEPCO’s internal investigation documents obtained by the NAIIC from the utility, an operator for Unit 3 mentioned nearly the same thing. According to this operator, “In the main control room also, the rumbling sound was audible from the beginning.”
conducted later, we asked the four operators for Unit 1 if they had heard some sounds at the time like those made when the Unit 1 SRV was operating. No one heard such sounds.

SRVs for nuclear power plants do not operate very often. It is difficult to reach a definitive conclusion about the operating noise of the Unit 1 SRV, based solely on the limited numbers of interviews we carried out. Yet, the fact that no one heard any noise from the operation of the SRV of Unit 1 carries significant weight.

Repeated actions of the SRV are a fundamental prerequisite for the Accident Scenario 1 that TEPCO and NISA believe. In this respect, NAIIC should evidently report herein the “fact” that there is no single piece of evidence, in the form of data, audible sound or otherwise, that supports an actuation of the SRV of Unit 1.

If the SRV were not in motion, it is more likely that the loss of coolant at Unit 1 resulted from Accident Scenario 2—a rupture of the reactor piping due to the earthquake motion—than Accident Scenario 1.

4. Recriticality issues and hydrogen explosion of Unit 4

We looked into the possibility of new nuclear fission (recriticality) and hydrogen generation in the reactor and spent fuel pool after the accident.

a. Data from other monitoring posts

We checked data from monitoring posts outside the Fukushima Daiichi plant for short life nuclides that can be produced by recriticality.

Around March 15, 2011, the figures for many nuclides, including short life nuclides, increased at the CTBT Monitoring Post in Takasaki, Gunma Prefecture. However, they could have been products of normal plant operation, or nuclides converted from such products.

The Japan Chemical Analysis Center detected a rapid increase of tellurium129, tellurium132, Iodine132, Xenon133, but they could have derived from normal operation.

Although these monitored data do not present obvious sign of recriticality, it is clear that Unit 1 discharged a major amount of radioactive material around March 15th to 16th, and around the 21st. There is a high possibility that the main cause of discharge during the 15th to 17th period was the damage to the Unit 2 suppression chamber (S/C) and the drywell (D/W) as well as the action of venting and the hydrogen explosion at Unit 3. As for the increase from the 21st to 22nd, we suspect the re-melting of Unit 3 debris.

b. Hydrogen explosion of Unit 3 and heat source within the spent fuel pool.

The hydrogen explosion resulted in a plume of white smoke immediately after the explosion, and on the two following days (see Photograph 2.2.4-1).

Observation of the spent fuel pool after the explosion shows the possibility of substantial damage to the fuel. But the dumping of a large amount of water after the explosion might have kept the radioactive material in the pool to be within or around the building, reducing the further spread of radioactive material. It is also possible that rainfall could have caused part of the radioactive material to fall into the ocean.

The decay heat of spent fuel in the pool could maintain the water temperature at around 75°C, calculated without considering the heat release to the heat pool. The calculation would also mean the water level decrease of approximately 0.17m/day was realized by evaporation.
oration, which is consistent with TEPCO’s evaluation of approximately 0.1m/day.

What was the source of the massive amount of heat that caused intermittent water evaporation in the form of white smoke to come out of the pool? The white smoke was generated not only immediately after the hydrogen explosion but on both of the next two days. There was, therefore, the possibility of damaged fuel inside the pool causing temporary massive heat generation.

The layout at the Unit 3 spent fuel pool[183] shows that: 1) fifty-two of the unspent fuel assemblies were arranged together in almost one lump, while the surrounding racks were empty, and 2) nearly half of the spent fuel was arranged together in one lump. Therefore, if the pool was impacted from the hydrogen explosion, it is probable that the used and unspent fuel assemblies were moved closer together and became compressed against one another, creating a condition of criticality inside the pool.

c. Hydrogen explosion of Unit 4

1) Unexplained points.

There was an explosion in the Unit 4 reactor building a little after 06:10 on March 15th. TEPCO presumes that it was caused by hydrogen from Unit 3 entering the fourth floor of Unit 4 through the standby gas treatment system (SGTS) pipes, and that something on the fourth floor must have ignited and triggered the hydrogen explosion.[184] The amount of the hydrogen in question has not been specified. There is no recorded footage even though it was already light enough at the time to record images. There is no objective record of accurate time of the explosion either. The reason for this is unknown.

2) Hydrogen generated by radiolysis of the spent fuel pool water in Unit 4.

At the time of the March 11 accident, Unit 4 was under regular inspection, in the phase of replacing the shroud for the reactor pressure vessel. There was a substantial number of fuel assemblies in the Unit 4 spent fuel pool, continuing to release decay heat.

The amount of hydrogen generated by water radiolysis was insignificant, if the water was around room temperature. But studies by JAEA and the University of Tokyo point out that at higher water temperatures where air bubbles can be observed, the amount of hydrogen gas generated is multiplied by digits.[185] They state that 13.7m³ of hydrogen would be capable

[183] TEPCO documents
of producing detonating gas, considering the volume of Unit 4. This is an amount that could be generated within one day if water boiling temperature in the pool continued, where hydrogen generation per day at that temperature could reach 18.1 m³.

Therefore, the exploded hydrogen could have come from Unit 3 as well as the Unit 4 spent fuel pool, but no quantitative evaluation can be given at this stage.

### 2.2.5 Problems inherent in the Mark I type primary containment vessel

1. Why did the primary containment vessel pressure exceed the design pressure?

Although details of the accident development could have been different at Units 1 to 3, the pressure inside the containment vessels substantially exceeded their designed capacity, up to almost twice the capacity in the case of Unit 1. The fundamental role of primary containment vessels (PCVs) is to contain the radioactive material at the time of accidents such as piping ruptures. When such an accident occurs, the MARK I type PCVs (see Figure 2.2.5-1) restrain pressure by channeling steam through the vents from the drywell (D/W) to the pressure suppression pool, condensing it into water.

The design pressure of the Mark I type PCV is approximately 4 atm (atmospheric pressure; gage) which is the highest transient pressure expected to be caused during the accident of the so-called sudden double-ended guillotine break of the largest diameter pipe in the recirculation outlet loop. However, the design presupposes that ECCS (emergency core cooling system) activates automatically at the moment such a pipe break occurs. It does not assume scenarios where design pressure is exceeded.

It is not really clear what caused the pressure inside the containments to substantially exceed the design pressure. The scenario given by TEPCO and NISA (see Table 2.2.4-4) is this: Following the station blackout (SBO), reactor pressure was mostly controlled by safety relief...
valves, which was accompanied by the coolant moving to suppression chamber (S/C). Since
the water in the S/C was not cooling down due to the SBO, steam coming in from the reactor
was not condensing the way it should, leading to a pressure rise inside the containment ves-
sel. Following the reactor damage and reactor melt, an enormous amount of hydrogen and
other non-condensable gases and steam poured into the S/C, increasing the containment
vessel pressure. Further down the line, the reactor pressure vessel (RPV) was damaged, and
non-condensable gasses like hydrogen and water steam burst directly into the dry well (D/ W),
causing a rapid increase of containment vessel pressure, exceeding the design pressure.

We should also note that the MARK I type PCVs at the Fukushima Daiichi plant is
smaller in volume than the improved version of MARK I, which contributed to the fast
rise in pressure.

On the other hand, as noted earlier in “Scenario 2” in 2.2.4. 3, a, if a small break LOCA
through pipes occurred immediately after the earthquake, water vapor and non-condens-
able gases like hydrogen, would all blow out directly to the dry well through the damaged
openings of the pipes and the reactor pressure vessel, resulting in a rapid increase of con-
tainment pressure.

2. Hydrodynamic loads

During normal operation, the PCV is filled with approximately one atmospheric pres-
sure of nitrogen. In a case of large break LOCA, a massive amount of nitrogen and steam
would rapidly gush out of the drywell (D/W) into the suppression chamber (S/C), creat-
ing a complex dynamic loads, such as pool swell load and condensation oscillation load.
These are called a “hydrodynamic loads.” Strength of the suppression chamber and other
related structures are evaluated based on the technical guideline. [187] Although such
hydrodynamic loads are expected to occur in any type of BWR-containment vessel, it is
presumed that a much more severe one would occur especially in MARK-I as shown in
Figure 2.2.5-2. It is notable that the only types of gaseous matter considered in the guide-
line are steam and nitrogen.

![Figure 2.2.5-2: Nuclear Safety Commission of Japan. “Evaluation guideline for energy load on BWR/MARK I containment pressure suppression system.” November 5, 1987](image)

<table>
<thead>
<tr>
<th>(1) - ① Loss of Coolant Accident</th>
<th>(1) - ③ Vent clear</th>
<th>(1) - ③ Formation of bubbles</th>
<th>(1) - ③ Rise of pool water surface</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1) - ③ Compression of atmospheric section</td>
<td>(1) - ⑥ Breakthrough</td>
<td>(1) - ② Fallback</td>
<td>(1) - ⑨ Chugging induced by steam condensation and oscillation</td>
</tr>
</tbody>
</table>

NOTE: ➡️ indicates the direction of load application.

In the Fukushima accident, however, we presume the reactor pressure vessel was damaged at the end, releasing into the drywell not only steam but very high temperature, non-condensable gases, including hydrogen. It is possible that the high temperature steam and gases added impact load to the drywell, then flowed into the suppression chamber, creating severe hydrodynamic loads under high temperature. If the containments of Unit 1 to 3 were damaged, one possible cause could have been the impact load and hydrodynamic loads under high temperature.

3. Sloshing
Earthquake motion may have created cyclical waves on the water surface of the suppression pool (see Figure 2.2.5-3). This motion of water surface is called sloshing.

When The Niigataken Chuetsu-oki Earthquake in 2007 struck the Kashiwazaki-Kariwa nuclear power plant, the spent fuel storage pools of all the units went through severe swaying, resulting in the massive overflow of coolant from the pools. We presume something similar happened at the Fukushima Daiichi plant—that earthquake motion shifted the water both in the spent fuel storage pools and in the pressure suppression pools. In the Tokachi-oki Earthquake in 2003, a fire started at a petroleum tank located more than 150km away from the epicenter, due to damage caused by sloshing. The characteristic cycle of sloshing in mechanical structures such as pipes is only around 0.1 second. But the sloshing cycle in large sized tanks can reach 5-10 seconds. The sloshing cycle of MARK-I containments is estimated to be around 4-5 seconds. When hit with long-period earthquake motion, the shifting water surface in the suppression chamber could cause the tips of downcomers to be exposed. At that moment, steam enters the gaseous space of the S/C, undermining the designed function of suppression, resulting in over-pressure, even impostulated LOCA. Sloshing in the pressure suppression pool can occur with the MARK-II or ABWR-RCCV types, but the MARK-I has the highest possibility of downcomer exposure. Thorough study is necessary, especially for long-period earthquake motion continuing over a substantial duration.

Figure 2.2.5-3: Sloshing simulation of MARK-I suppression pool water

The Table shows a simulation of pool water sloshing within an S/C under earthquake motion. The areas marked in red are where high level sloshing is occurring. Downcomers are exposed outside the water surface in the blue areas due to big sloshing. At that time, the downcomer emits water steam into the gaseous space of the S/C, resulting in a rise in pressure inside the S/C.

[188] TEPCO documents