However, all attempts to open the pressure relief valves failed, and reactor pressure quickly increased above the level at which the diesel driven fire pump could inject, stopping the cooling of the Unit 3 core about 35 hours after the station blackout. Faced with this setback, the operators tried for nearly 45 minutes to return to injection via the emergency high pressure coolant injection system, but were unsuccessful. Without any capability to cool the reactor, an emergency report for "loss of reactor cooling function" as defined in the regulations associated with the Nuclear Emergency Act [20] was issued for Unit 3, at 05:10 on 13 March. The Unit 3 core remained without cooling for the hours that followed, and Unit 3 became the next unit to lose core cooling.

After the loss of cooling, an alternative water injection method to cool the Unit 3 core utilizing the fire engines was ordered by the site superintendent at 05:15. In view of the deteriorating conditions, he also ordered the Unit 3 containment venting path to be lined up.

Unit 3 alternative core cooling and containment venting

The fire engines from Units 5–6 were dispatched to Unit 3 and work began at 05:21 on 13 March to establish a line for injecting seawater into the Unit 3 core from the Unit 3 backwash valve pit through the fire protection lines. An additional fire engine arrived from the Kashiwazaki-Kariwa NPP at 06:30. The seawater injection line was completed within an hour. However, its use was postponed by the site superintendent as a result of a communication from TEPCO headquarters[50]. As a result, the injection line was changed back to the borated freshwater source through the fire protection line using fire engines.

The efforts to reduce the reactor pressure below the fire engine pump pressure in order to maintain water injection required the activation of the pressure relief valves. This was achieved by use of DC batteries from cars, which were collected in the common main control room of Units 3 and 4.

Meanwhile, the Unit 3 venting line arrangement was also completed in a little more than three hours, at 08:41 on 13 March, but the containment pressure was still below the containment design pressure, not enough to cause the bursting of the rupture disc as designed. As efforts continued to reduce the reactor pressure by opening the safety relief valve, the operators in the control room observed a drop in the reactor pressure in the Unit 3 reactor at 09:08, although the valve status indicators did not conclusively show whether or not the valves were in the open position. Along with this depressurization of the reactor vessel, there was a pressure surge in the primary containment, indicating a discharge from the reactor vessel to the containment vessel. Eventually, at 09:20 on 13 March, the containment pressure exceeded the maximum design pressure of the containment and, subsequently, the containment pressure dropped rapidly, indicating that the venting of the Unit 3 containment had occurred as a result of the bursting of the rupture disc.

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[50] A division director in the emergency off-site centre at TEPCO headquarters who attended a meeting in the Prime Minister's Office earlier asked the site superintendent on the telephone whether there was any fresh water available. He informed the site superintendent of the views of meeting participants, who were inclined to continue freshwater injection as long as possible. The site superintendent interpreted this communication as a directive not to inject seawater as long as fresh water was available.

Following reactor depressurization, achieved by opening additional safety relief valves, the reactor pressure fell below the fire engine pump pressure and injection of borated fresh water into the Unit 3 reactor started at 09:25, after more than four hours without cooling.

The venting of the Unit 3 containment was short lived when a valve on the vent line closed[51] because of the lack of sufficient air supply to keep it open. After 6.5 hours of effort, the valve was reopened by using a mobile compressor.

Unit 2 precautionary measures for fundamental safety functions

At around 10:15 on 13 March, as the conditions for maintaining the relevant fundamental safety functions in Units 1 and 3 became more difficult, the site superintendent ordered the containment vent path for Unit 2 to be pre-emptively established. This was intended to take advantage of still favourable radiological conditions compared to the other units and site wide trend[52] to conduct work in the Unit 2 reactor building where the valves were to be manipulated. The work was accomplished in 45 minutes, but venting did not occur because the pressure inside the Unit 2 containment was not high enough to burst the rupture disc.

At around 12:05, the site superintendent also ordered precautionary preparations for seawater injection into Unit 2 in case the unit's operating cooling system failed. For this purpose, fire engines were to be connected to the fire protection lines of Unit 2 to inject water from the backwash valve pit of Unit 3, if needed.

Injection of seawater into Unit 3 and increase of radiation levels

As the fresh water from the fire protection tanks was depleted at 12:20 on 13 March, the site superintendent decided to inject seawater into the Unit 3 reactor. The fire engines were repositioned, and seawater injection from the backwash valve pit of Unit 3 started nearly one hour later, at 13:12.

At 14:15 on 13 March, a high radiation dose rate (nearly 1 mSv/h) was measured near the site boundary and the relevant government agencies were notified at 14:23 of an "abnormal site boundary radiation level increase" as defined in the regulations associated with the Nuclear Emergency Act [20]. Fifteen minutes later, the radiation dose rate exceeded 100–300 mSv/h at the entry doors of the Unit 3 reactor building. As the dose rates measured on the Unit 3 side of the common main control room of Units 3 and 4 exceeded 12 mSv/h, the shift team moved to the Unit 4 side.

The on-site emergency response centre inferred from these dose levels that radioactive gases had escaped from the Unit 3 reactor, which in turn meant that hydrogen had also escaped. Mindful of the possibility of a hydrogen explosion similar to the one at Unit 1, the site superintendent decided, at 14:45, to temporarily evacuate the workers from the common main control room of Units 3 and 4 and from the areas in the vicinity of Unit 3.

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[51] As discovered two hours later.

[52] Between 05:30 and 10:50 on 13 March, neutrons were detected about 1 km away from the reactor buildings of Units 1–4 near the main gate, indicating a possible breach of the containment vessel, although the source of the neutrons was unknown.

The evacuated areas also included the Unit 3 backwash valve pit area, halting the activities for water injection. The evacuation order was lifted at 17:00, and workers returned to the Unit 3 backwash valve pit area to continue the activities for water injection and venting.

Establishment of core cooling in Unit 5

Meanwhile, the power from the Unit 6 emergency diesel generator was connected at 20:48 on 13 March to the pump of Unit 5's normal, low pressure, heat removal system, and was activated at 20:54. A reactor water injection line to the reactor of Unit 5 through one of the two residual heat removal systems was lined up and interconnecting pipe valves to the makeup water condensate system were opened, 53 hours after the station blackout. However, water injection did not occur, since the reactor pressure had gradually risen and exceeded the injection pressure. In response, a safety relief valve was opened, making use of available DC power and nitrogen supplies. This was successful in reducing the pressure in the reactor pressure vessel and allowed water injection into the Unit 5 reactor at 05:30 on 14 March, which continued afterwards[53].

Loss of seawater cooling in Units 1 and 3

As the injection of sea water into Units 1 and 3 from the Unit 3 backwash valve pit continued into Monday 14 March, the water in the pit fell to such a low level that injection had to be halted at 01:10. After the intake hose was lowered deeper into the pit, the remaining pit water was reserved for injection into Unit 3, which resumed two hours later. Unit 1 core cooling was postponed until the pit could be refilled.

In the following hours, the Unit 3 containment pressure was found to be increasing and the reactor water level indication continued to decrease. The reactor water level in Unit 3 went off-scale at 06:20 on 14 March, indicating to the operators that the core was uncovered. The site superintendent ordered an evacuation of all workers because of concern about a possible hydrogen explosion in Unit 3, halting the pit refilling activities.

The containment pressure in Unit 3 reached a maximum at 07:00, but was found to be slightly lower at 07:20. It subsequently remained stable below maximum design pressure. The site superintendent then decided to resume the work to establish a line to refill the backwash valve pit from the ocean. In the next two to four hours, the seawater injection lines for all units were re-established, and refilling of the pit commenced, utilizing two additional fire engines to pump water from the ocean and tanker trucks from the Japan Self Defense Force, which arrived at the site at 10:26, to carry water to the pit.

Seawater injection into Unit 1 was ready to be resumed when all activities, including the ongoing seawater injection into the Unit 3 reactor, had to stop because of the explosion in Unit 3. This damaged the hoses and fire engines around the Unit 3 backwash valve pit and necessitated a temporary evacuation of workers from the outside areas.

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[53] In addition, the AC power to operate the system for controlling reactor building pressure was supplied from Unit 6's emergency diesel generator. A little over two days after the blackout, pressure in the reactor building was below atmospheric pressure, ensuring secondary confinement.

Explosion in the Unit 3 reactor building

At 11:01 on 14 March, an explosion occurred in the upper part of the Unit 3 reactor building, destroying the structure above the service floor and injuring workers. In addition to the destruction of the alternative water injection arrangement, the capability to vent the containment in Unit 2 was also lost as a result of the explosion, which affected the previously set up Unit 2 containment venting path. After the explosion, the isolation valve on the Unit 2 vent line was discovered to be closed and could not be reopened.

Restart of seawater cooling in Units 1 and 3

The work to re-establish the seawater injection line, this time directly from the ocean, was started again after a break of two hours. After restoration of injection lines, seawater injection was recommenced first for Unit 3 in the afternoon of 14 March and later in the evening for Unit 1. The cores had been without cooling water injection, for 9 hours in Unit 3 and for 19 hours in Unit 1.

Loss of cooling and seawater injection in Unit 2

At around 13:00 on 14 March, Unit 2 became the next unit to experience loss of cooling, with measurements showing that the reactor water level had decreased and that the reactor pressure had increased. This pointed to the possible failure of the Unit 2 reactor core isolation cooling system, as inferred by the unit operators and the on-site emergency response centre. As a result, a "loss of reactor cooling functions" report as defined in the Nuclear Emergency Act [20] was issued for Unit 2.

After the failure of the reactor core isolation cooling system, seawater injection through the fire protection system was attempted at 13:05, but the reactor pressure was too high for the fire engine pumps. It seemed likely that, without water injection, the core would be uncovered very soon. It was therefore decided to use the relief valves to depressurize the reactor in order to enable water injection at low pressure, whilst recognising the potential adverse impact on the confinement as a result of the release of steam from the reactor into the containment[54].

After reactor vessel depressurization and refuelling of the fire engines, seawater injection into Unit 2 through the fire protection system began shortly before 20:00 on 14 March, first with one fire engine and, soon afterwards, with two.

Degradation of the Unit 2 confinement

At around 21:55 on 14 March, recently restored radiation monitoring equipment inside the containment indicated that the radiation levels in the Unit 2 containment had increased substantially since the previous measurements taken eight hours earlier[55]. Also, both the

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[54] The suppression chamber section of the primary containment vessel was already nearly saturated.

[55] A 5000-fold increase in radiation levels in the containment atmosphere (from 1.08 mSv/h to 5360 mSv/h) and a 40-fold increase in radiation levels in the suppression pool section of the containment (from 10.3 mSv/h to 383 mSv/h). Additionally, neutrons had been detected between 21:00 on 14 March and 01:40 on 15 March, again near the main gate. It

reactor and containment pressures showed an increasing trend after 22:30 on 14 March. The containment pressure exceeded the design pressure at 22:50, prompting an emergency declaration of an "unusual rise of pressure in containment vessel" in accordance with the Nuclear Emergency Act [20] for Unit 2. This condition was reported to the relevant government authorities at 23:39. Over the next three to four hours, more relief valves were opened to decrease reactor pressure in order to allow water injection into the Unit 2 reactor. As a consequence, the containment pressure increased further while the unit operations team tasked with establishing the venting line to relieve containment pressure was unable to open the vent valves. In order to protect the confinement function and permit venting as soon as possible, TEPCO staff at the emergency on-site and off-site centres agreed to vent directly from the drywell, recognising that this would increase radioactive releases to the environment. However, it was not possible to open the valves on the drywell vent either, and consequently Unit 2 venting could not be accomplished.

At 04:17 on Tuesday 15 March, the relevant government agencies were notified that depressurization of the Unit 2 containment and of the reactor had not been effective and that containment pressure continued to increase.

Events at Units 2 and 4 followed by site evacuation

At 06:14 on 15 March, an explosion was heard on the site and tremors were felt in the common main control room of Units 1 and 2. This was followed by a drop in the pressure reading of the Unit 2 containment (suppression chamber). The staff in the control room initially reported to the on-site emergency response centre that the Unit 2 suppression chamber pressure had dropped to atmospheric pressure[56], indicating potential loss of the confinement function.

This information indicated a possible containment vessel failure and the possibility of uncontrolled releases from Unit 2. On this basis, the on-site emergency response centre ordered all personnel in all the units to temporarily evacuate to the seismically isolated building where the on-site emergency response centre was located. At about the same time as the event associated with the Unit 2 suppression chamber, an explosion in the upper part of the Unit 4 reactor building was reported by the evacuating personnel.

Following the events in Units 2 and 4, all personnel except those required for monitoring and emergency response were instructed by the site superintendent to go to a radiologically safe location. Approximately 650 people understood this order as a site evacuation and evacuated to the Fukushima Daini NPP site. An estimated 50–70 staff[57], including the site superintendent, remained at the Fukushima Daiichi site. The relevant government agencies were informed by the on-site emergency response centre of the evacuation at 07:00 on 15 March.

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was thought by TEPCO that the neutrons came from the spontaneous fission of actinides that were released following core damage in one of the three reactors.

[56] After rechecking the readings, it was confirmed that the suppression chamber pressure was off the scale, but the drywell section pressure had not decreased significantly in Unit 2.

[57] As noted by different investigation reports, the exact number of staff is not certain [6, 8]. It is also noted, that the staff who evacuated to Fukushima Daini started to return to site on the same day.

About two hours later, white smoke (or steam) was observed being released from the Unit 2 reactor building near the fifth floor. A radiation dose rate measurement of nearly 12 mSv/h was recorded at the main gate at 09:00 on 15 March, the highest measurement since the beginning of the accident. Because of the high radiation levels, an order was issued by government authorities, two hours later at 11:00, requiring all residents within a 20–30 km radius of the Fukushima Daiichi NPP to take shelter indoors.

During this sequence of events, fundamental safety functions of Units 1–3 were lost or severely degraded (Fig. 2.5), and efforts focused on damage assessment and restoration and stabilization of those functions.

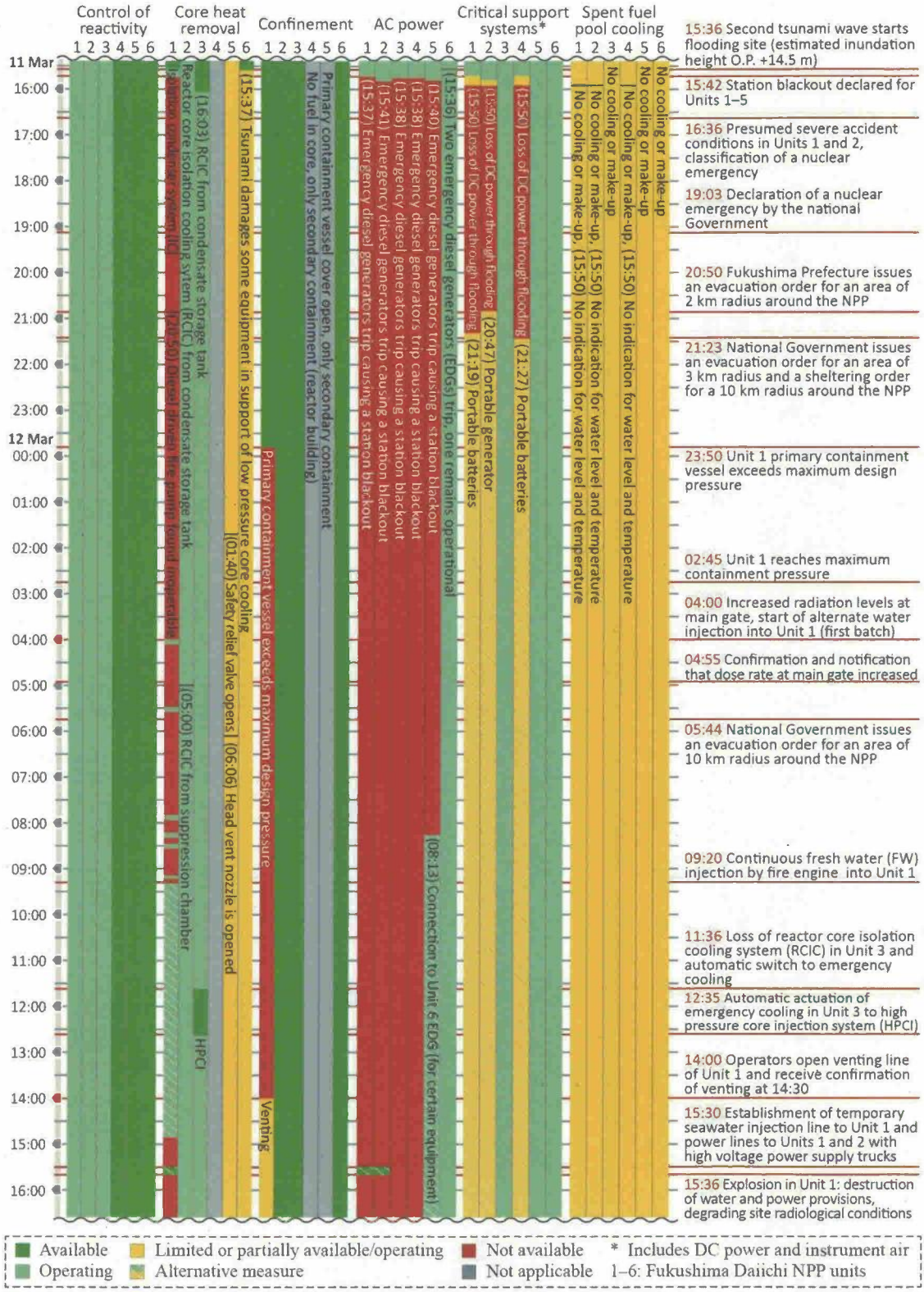

FIG 2.5. Fundamental and supporting safety functions in the accident response at the Fukushima Daiichi NPP (11–15 March 2011).

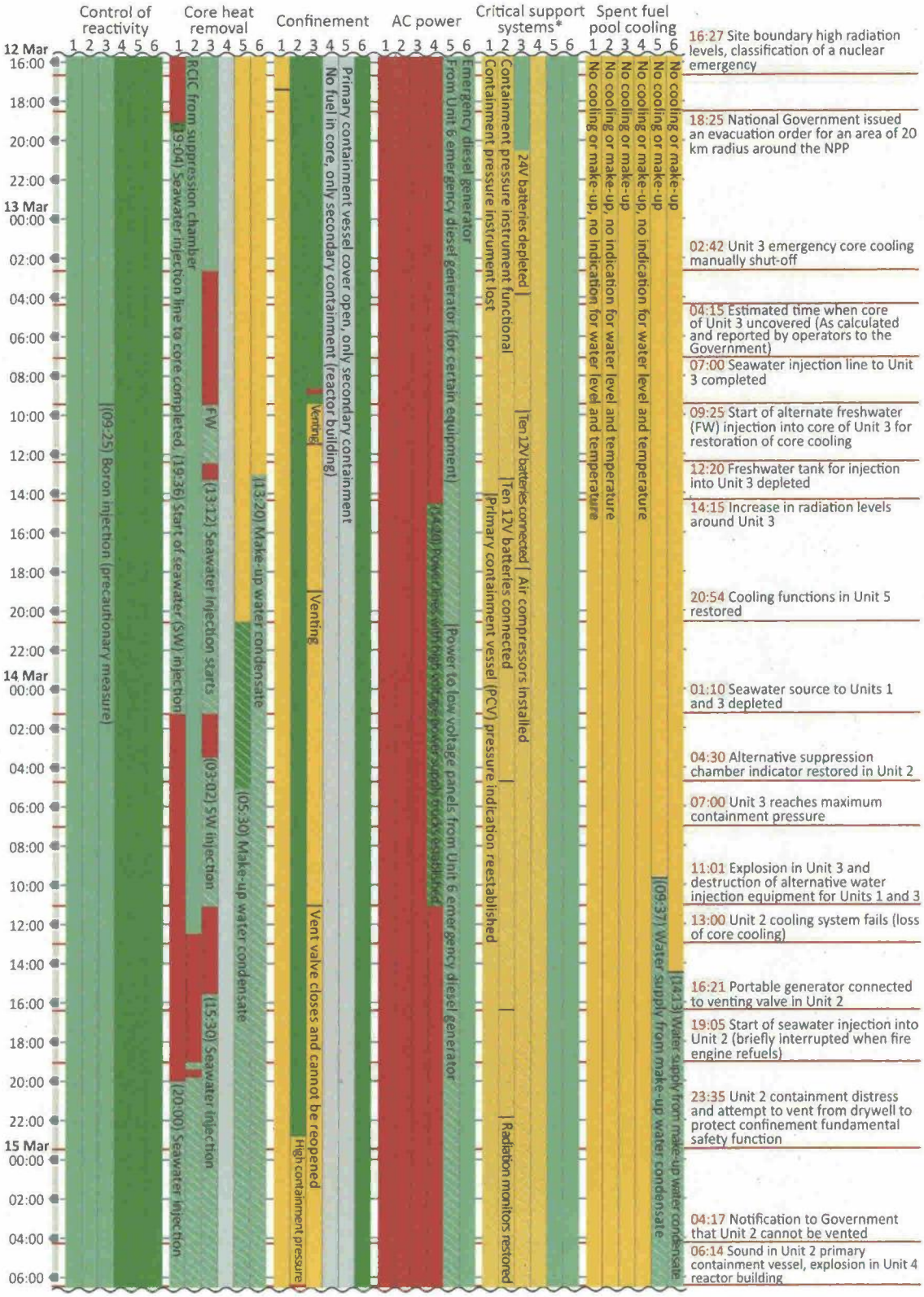

FIG 2.5(cont.). Fundamental and supporting safety functions in the accident response at the Fukushima Daiichi NPP (11–15 March 2011).

2.1.3. Stabilization efforts

Replenishment of the spent fuel pool at Units 3 and 4

In order to perform a damage assessment of the Unit 4 explosion, and in particular to confirm the integrity of the spent fuel pool, a team attempted to enter the Unit 4 building at 10:30 on 15 March, just over four hours after the explosion. The team encountered high radiation levels when they opened the door of the reactor building and could not enter the building.

A remote visual inspection from a helicopter to address concerns about the status of the Unit 3 and 4 spent fuel pools was conducted on the afternoon of Wednesday, 16 March. The inspection confirmed that there was sufficient water in the Unit 4 spent fuel pool to cover the fuel assemblies; however, observations were not conclusive for Unit 3's spent fuel pool, making its replenishment a high priority.

The first supply of water to the Unit 3 spent fuel pool was made between 09:30 and 10:00 on 17 March, when helicopters dropped seawater. Fresh water was sprayed by water cannon trucks later on the same day between 19:05 and 20:07. Sea water or fresh water was sprayed into the Unit 4 spent fuel pool starting on 20 March[58].

Spraying into the pools using water cannon and fire engine trucks or concrete pump vehicles continued intermittently in March, to ensure that the spent fuel was not exposed. The fuel pool cooling and cleanup system was also utilized in April and well into May 2011.

Restoration of power supplies and end of the station blackout

Between 17 and 20 March, work was carried out to lay temporary power cables to Units 1 and 2. At 15:46 on Sunday 20 March, almost exactly nine days after the station blackout, off-site power was restored to Units 1 and 2 through this temporary AC power system, ending the blackout in Units 1 and 2.

In Unit 6, power was restored to the cooling system of the second water cooled emergency diesel generator through a power line connected to the operating air cooled generator. The water cooled emergency diesel generator began operation again at 04:22 on 19 March, supplying AC power to Units 5 and 6.

The blackout at Units 3 and 4 ended, after more than 14 days, when temporary off-site power to these two units was restored on 26 March.

Achieving stable conditions

Unit 5 was the first unit to reach cold shutdown mode when its normal residual heat removal system was put into service, at 12:25 on 20 March. The reactor temperature decreased below 100°C in approximately two hours, placing Unit 5 in the cold shutdown mode at 14:30 on 20 March 2011, nearly nine days after the beginning of the accident.

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[58] The same approach was followed for adding water to the spent fuel pool of Unit 1. Since the Unit 2 reactor building was still covering the Unit 2 spent fuel pool, the spray method could not be used for Unit 2.

The normal residual heat removal system of Unit 6 was put back in service, in a manner similar to that at Unit 5, at 18:48 on the same day. The reactor temperature decreased below 100°C in less than one hour, placing Unit 6 in the cold shutdown mode at 19:27 on 20 March (Fig. 2.6).

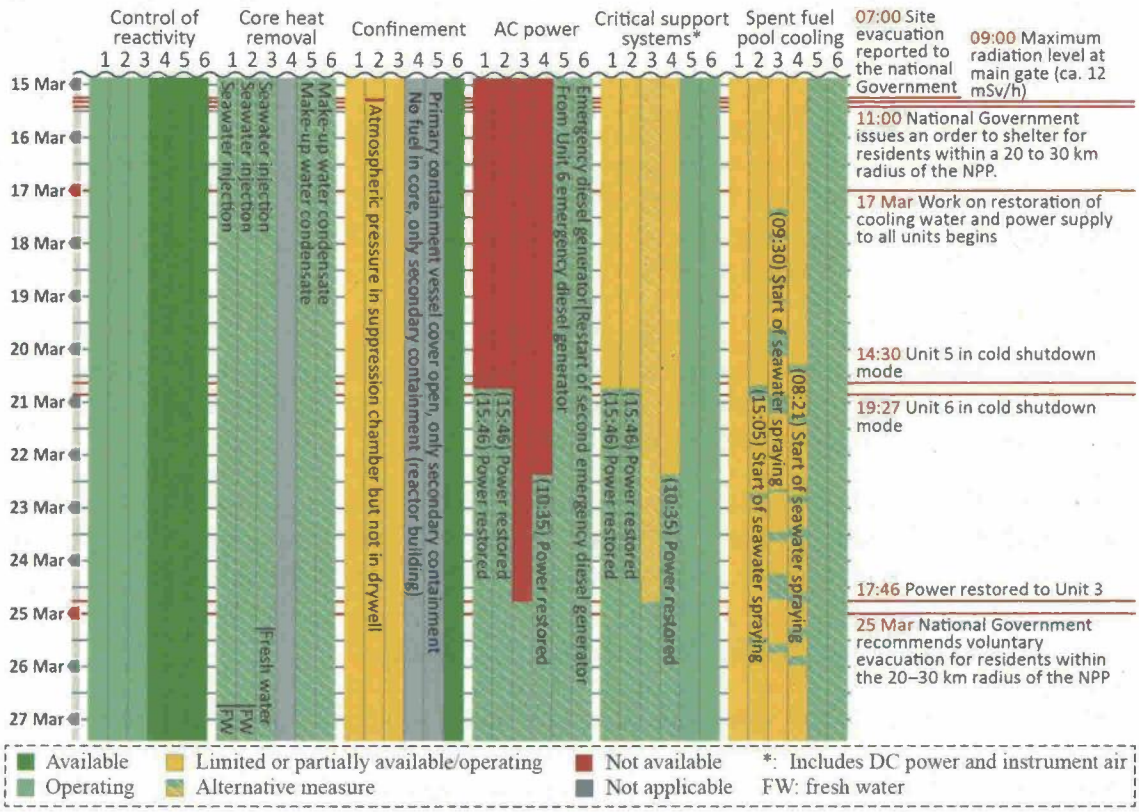

FIG 2.6. Temporary restoration of fundamental safety functions at the Fukushima Daiichi NPP.

For Units 1–3, TEPCO issued an action plan on 17 April 2011, the Roadmap towards Restoration from the Accident at the TEPCO Fukushima Daiichi Nuclear Power Station [24]. The roadmap included measures to be taken for: the establishment of stable cooling of the reactors and spent fuel; reduction and monitoring of radioactive releases; control of hydrogen accumulation; and prevention of return to criticality. These actions were implemented in the nine months following the accident.

The roadmap established two conditions that would define the end of the accident state, or 'cold shutdown state'[59]: achievement of significant suppression of radiological releases and steady decline of radiation dose rates; and achievement of target values for certain plant parameters as prescribed in the roadmap. The government and TEPCO announced on 19 July that the first condition had been achieved in Units1–3 and on 16 December 2011 that the

[59] The term 'cold shutdown state' was defined by the Government of Japan at the time specifically for the Fukushima Daiichi reactors. Its definition differs from the terminology used by the IAEA and others.

second condition had been achieved for these units. This announcement officially[60] brought the 'accident' phase of events at the Fukushima Daiichi NPP to a close.

However, some unstable plant conditions continued, such as fluctuations in temperatures, which had been explained as being caused by instrumentation failures, or fluctuations in the measurement of fission products. More stable plant parameters were achieved between March and April 2012, while post-accident management efforts continued. Additionally, challenges in the management of waste, such as difficulties in dealing with the accumulation of radioactive water due to groundwater ingress to the buildings and occasional failures of equipment, continued. At the time of writing, the Government of Japan considered the Fukushima Daiichi NPP a 'specified facility as an accident site'[61].

2.2. NUCLEAR SAFETY CONSIDERATIONS

2.2.1. Vulnerability of the plant to external events

The earthquake on 11 March 2011 caused vibratory ground motions that shook the plant structures, systems and components. It was followed by a series of tsunami waves, one of which inundated the site. Both the recorded ground motions and the heights of the tsunami waves significantly exceeded the assumptions of hazards that had been made when the plant was originally designed. The earthquake and the associated tsunami impacted on multiple units at the Fukushima Daiichi NPP.

The seismic hazard and tsunami waves considered in the original design were evaluated mainly on the basis of historical seismic records and evidence of recent tsunamis in Japan. This original evaluation did not sufficiently consider tectonic-geological criteria and no re-evaluation using such criteria was conducted.

Prior to the earthquake, the Japan Trench was categorized as a subduction zone with a frequent occurrence of magnitude 8 class earthquakes; an earthquake of magnitude 9.0 off the coast of Fukushima Prefecture was not considered to be credible by Japanese scientists. However, similar or higher magnitudes had been registered in different areas in similar tectonic environments in the past few decades.

There were no indications that the main safety features of the plant were affected by the vibratory ground motions generated by the earthquake on 11 March 2011. This was due to the conservative approach to earthquake design and construction of NPPs in Japan, resulting in a plant that was provided with sufficient safety margins. However, the original design considerations did not provide comparable safety margins for extreme external flooding events, such as tsunamis.

The vulnerability of the Fukushima Daiichi NPP to external hazards had not been reassessed in a systematic and comprehensive manner during its lifetime. At the time of the accident, there were no regulatory requirements in Japan for such reassessments and relevant domestic and international operating experience was not adequately considered in

[60] According to the criteria set by the Government of Japan at the time.

[61] In accordance with the definition of 'specified nuclear facility', i.e. a facility that would require special measures for safety or physical protection of specified nuclear material, established by the current regulatory body, the Nuclear Regulation Authority (NRA), on 7 November 2012.

the existing regulations and guidelines. The regulatory guidelines in Japan on methods for dealing with the effects of events associated with earthquakes, such as tsunamis, were generic and brief and did not provide specific criteria or detailed guidance.

Before the accident, the operator had conducted some reassessments of extreme tsunami flood levels, using a consensus based methodology developed in Japan in 2002, which had resulted in values higher than the original design basis estimates. Based on the results, some compensatory measures were taken, but they proved to be insufficient at the time of the accident.

In addition, a number of trial calculations were performed by the operator before the accident, using wave source models or methodologies that went beyond the consensus based methodology. Thus, a trial calculation using the source model proposed by the Japanese Headquarters for Earthquake Research Promotion in 2002, which used the latest information and took a different approach in its scenarios, envisaged a substantially larger tsunami than that provided for in the original design and in estimates made in previous reassessments. At the time of the accident, further evaluations were being conducted, but in the meantime, no additional compensatory measures were implemented. The estimated values were similar to the flood levels recorded in March 2011.

Worldwide operating experience has shown instances where natural hazards have exceeded the design basis for a NPP. In particular, the experience from some of these events demonstrated the vulnerability of safety systems to flooding.

Box 2.4. Tsunamis [25]

A tsunami — in Japanese meaning a wave ('nami') in a harbour ('tsu') — is a series of travelling waves of long wave length (e.g. from kilometres to hundreds of kilometres) and period (e.g. several minutes to tens of minutes, and, exceptionally, hours), generated by deformation or disturbances of the sea floor (or, in generic terms, underwater floor). Earthquakes, volcanic phenomena, underwater and coastal landslides, rock falls or cliff failures can generate a tsunami. Tsunamis can occur in all oceanic regions and sea basins of the world and even in fjords and large lakes.

Tsunami waves propagate outward from the generating area in all directions, with the main direction of energy propagation determined by the dimensions and orientation of the generating source. During propagation of the tsunami in deep water, they proceed as ordinary gravity waves, with a speed depending on the depth of water. For example, in deep ocean, speeds could exceed 800 km/h, with a wave height generally less than a few tens of centimetres and, in the case of an earthquake source, with wave lengths often exceeding 100 km. During the propagation, submarine topography affects the speed and height of the tsunami wave. Refraction, reflection from a sea mount or its chain (archipelago) and diffraction are important factors affecting the propagation of tsunami waves in deep water.

Tsunami waves become steeper and increase in height on approaching shallow water because wave speed is reduced and wave length is shortened when the depth decreases. In a coastal zone, local topography and bathymetry, such as a peninsula or submarine canyon, may cause an additional increase in wave heights which can also be amplified by the presence of a bay, an estuary, a harbour or lagoon funnels as a tsunami moves inland. Several large waves may occur and the first may not be the largest. A recession of the sea may also occur before the first wave and between each consecutive flooding. A tsunami may cause inland inundation because its wave length is so long that a huge mass of water follows behind the wave front. This can have a destructive impact.

The IAEA safety standards in force at the time of the accident required that before the construction of a NPP, site specific external hazards, such as earthquakes and tsunamis, need to be identified, and the impacts of these hazards on the NPP need to be evaluated as part of a comprehensive and full characterization of the site [26]. Adequate design bases are required to be established to provide sufficient safety margins over the lifetime of the NPP [27]. These margins need to be sufficiently large to address the high level of uncertainty associated with the evaluation of external events. Site related hazards also need to be periodically reassessed in order to identify any need for change as a result of new information and knowledge during the life of the plant [26].

In the 1960s and 1970s, it was common practice to use historical records when applying methods for estimating seismic hazards. This approach was supplemented by using safety margins to increase the maximum historically recorded seismic intensity or magnitude in the site region, and by assuming that such an event might occur at the closest distance to the site [28]. The seismic hazard assessment for the design of Unit 1 and Unit 2 at the Fukushima Daiichi NPP was conducted mainly on the basis of regional historical seismic data. During the process of obtaining construction permits for the later units, a new methodology was applied using a combination of historical earthquake information and the geomorphological fault dimensions [16, 29].

The information regarding the 'inland' faults was taken from official sources as well as from specific surveys conducted by the operator, and conservative parameters were assumed in the analysis to predict the magnitude of a possible earthquake. For the Japan Trench, an earthquake of magnitude 8 class was originally estimated as the maximum event without sufficient tectonic based justification and based largely on observed historical data.

Large magnitude earthquakes (M9) had occurred elsewhere in the Pacific 'ring of fire', for example in Chile in 1960 and in Alaska in 1964, shortly before Fukushima Daiichi Unit 1 was given the construction license. These earthquakes did not lead to a consensus among Japanese seismologists that such an event would be possible close to the shores of Japan in a tectonic environment similar to those that generated earthquakes in other areas of the Pacific tectonic plate.

In the original assessment of external flooding hazards used in the 'Establishment Permit' for the plant, the plant designers applied the methodology and criteria prevalent in Japan at the time, which were based on the study and interpretation of historical records of earthquakes and tsunamis. The distant tsunami that occurred following one of the world's largest known earthquakes in Chile in 1960 was the event used for design purposes against external flooding. This event resulted in a tsunami height observed at Onahama Port in Fukushima Prefecture of 3.1 m above sea level.

With regard to the tsunami sources located at the Japan Trench off the eastern coast, there was a lack of historical records of tsunami flood levels at the location of the Fukushima Daiichi site, as well as a lack of evidence of the occurrence of earthquakes in the area offshore of the site. The absence of data on nearby tsunami sources supported the adopted maximum flood level of 3.1 m for design purposes. TEPCO did not consider large magnitude earthquakes that had occurred elsewhere and did not postulate them as local tsunami source at the Japan Trench.

Despite the lack of regulatory requirements in Japan for conducting a reassessment of seismic hazards and tsunami hazards, TEPCO had conducted several reassessments over the lifetime of the Fukushima Daiichi NPP [30]. TEPCO and other operating organizations in Japan had reassessed tsunami flood levels using a methodology developed by the Japan Society of Civil Engineers and published in 2002 [31]. This methodology used a standard source model for near or local tsunamis, based on historical data, in which no earthquake generating a tsunami is assumed to occur along the Japan Trench offshore of the Fukushima Daiichi site. The assumption of the standard source model, as described above, was applied in all evaluations that were performed using this methodology.

The 2006 guidelines of the Nuclear Safety Commission of Japan (NSC) [32] required consideration of inter-plate earthquakes as well as inland crustal earthquakes. These guidelines on seismic safety and associated events were used for evaluating seismic hazards, but the guidelines for tsunami hazards included only generic and brief statements and did not provide specific requirements, criteria or methodology. These earthquakes were considered by TEPCO as 8 class during its 'back-checking' of seismic safety as requested by the Nuclear and Industrial Safety Agency (NISA). However, owing to the distance of the site from those inter-plate earthquakes, the approach provided smaller hazard values associated with this tectonic structure when compared with the inland seismogenic sources. Consequently, their impact on the ground motion hazard was not considered. At the time of the accident, TEPCO had not completed the re-assessment of the vulnerability of the plant to earthquakes and tsunamis.

In 2009, TEPCO estimated a value of 6.1 m for the maximum tsunami height, using the latest bathymetric and tidal data. As a result of this new estimate, some design changes were made at the Fukushima Daiichi NPP, notably by raising the motors of the pumps used for the removal of residual heat. During the accident, this measure alone proved to be insufficient.

No other safety measures had been implemented to enhance flood protection, such as measures to avoid the flooding of emergency diesel generators.

In addition to the reassessments employing the methodology of the Japan Society of Civil Engineers, trial calculations of the tsunami water flood levels were performed by TEPCO before the accident. One of these trial calculations [30] applied the source model proposed by the Headquarters for Earthquake Research Promotion, which used the latest information and considered different scenarios [30, 33]. This approach examined the potential of the Japan Trench off the coast of Fukushima Prefecture to cause tsunamis. It did not rely only on historical tsunami records for this part of the tectonic subduction zone.

The new approach, applied between 2007 and 2009, postulated an earthquake of magnitude 8.3 off the Fukushima coast. Such an earthquake could lead to a tsunami run-up of around 15 m at the Fukushima Daiichi NPP site (similar to the actual tsunami height on 11 March 2011), which would inundate the main buildings. On the basis of this new analysis, TEPCO, NISA and other organizations in Japan considered that further studies and investigations were needed. TEPCO and other electrical utilities requested the Japan Society of Civil Engineers to review the appropriateness of the tsunami source models; these efforts were in progress in March 2011.

TEPCO did not take interim compensatory measures in response to these increased estimates of tsunami height, nor did NISA require TEPCO to act promptly on these results [30].

With regard to seismic and tsunami hazard assessment, the approach used was not fully in line with international practice at the time of the accident. It was typical international practice to make conservative assumptions, by using a deterministic approach, increasing the maximum historical intensity or magnitude ever recorded in the site region and assuming that the event would occur at the closest distance from the site [34, 35]. This was done to account for the uncertainties in the observations of intensities or magnitudes, as well as to compensate for the fact that the maximum historical values might not be attained in a relatively short period of observation. Typically, the observation period needs to include pre-historical data in order to provide robust estimates for the hazard assessment.

Notwithstanding the difficulties and uncertainties in the seismic hazard assessment, the events at the Fukushima Daiichi NPP demonstrated the robustness of Japanese NPPs in relation to earthquake vibratory ground motions. On 11 March 2011, the maximum accelerations recorded at the base mat of the reactor buildings of Fukushima Daiichi Units 1-5 were significantly larger than had been estimated when the plant was designed. However, there were no indications that the ground motion caused notable damage to safety related structures, systems or components [36]. However, the defences against tsunami induced flooding proved inadequate against tsunami wave heights that were much larger than those used in the design of the Fukushima Daiichi NPP. A scenario involving extreme natural hazards simultaneously and affecting multiple units was not considered in the design of the Fukushima Daiichi NPP. The timely provision of resources for the implementation of severe accident management actions at the Fukushima Daiichi NPP was compromised by the disruption off-site at the regional level due to the extensive damage caused to the infrastructure by the earthquake and the tsunami.

NPP operating experience in Japan and elsewhere in the 12 years prior to the accident showed the potential for severe consequences from flooding. The relevant operating

experience included: a storm surge causing flooding at two reactors at the Le Blayais NPP in France in 1999; the Indian Ocean tsunami in 2004, which flooded seawater pumps at the Madras Atomic Power Station in India; and the Niigata-Chuetsu-Oki earthquake in Japan in 2007. The latter affected TEPCO's Kashiwazaki-Kariwa NPP, causing flooding of the reactor building of Unit 1 owing to the failure of the underground external fire extinguishing piping [37, 38, 39, 40].

2.2.2. Application of the defence in depth concept

Defence in depth is a concept that has been applied to ensure the safety of nuclear installations since the start of nuclear power development. Its objective is to compensate for potential human and equipment failures by means of several levels of protection. Defence is provided by multiple and independent means at each level of protection.

The design of the Fukushima Daiichi NPP provided equipment and systems for the first three levels of defence in depth: (1) equipment intended to provide reliable normal operation; (2) equipment intended to return the plant to a safe state after an abnormal event; and (3) safety systems intended to manage accident conditions. The design bases were derived using a range of postulated hazards; however, external hazards such as tsunamis were not fully addressed. Consequently, the flooding resulting from the tsunami simultaneously challenged the first three protective levels of defence in depth, resulting in common cause failures of equipment and systems at each of the three levels.

The common cause failures of multiple safety systems resulted in plant conditions that were not envisaged in the design. Consequently, the means of protection intended to provide the fourth level of defence in depth, that is, prevention of the progression of severe accidents and mitigation of their consequences, were not available to restore the reactor cooling and to maintain the integrity of the containment. The complete loss of power, the lack of information on relevant safety parameters due to the unavailability of the necessary instruments, the loss of control devices, and the insufficiency of operating procedures made it impossible to arrest the progression of the accident and to limit its consequences.

The failure to provide sufficient means of protection at each of level of defence in depth levels resulted in severe reactor damage in Units 1, 2 and 3 and in significant radioactive releases from these units.

Box 2.5. The concept of defence in depth applicable at the time of the accident [27]

The concept of defence in depth, as applied to all safety activities, whether organizational, behavioural or design related, ensures that they are subject to overlapping provisions, so that if a failure were to occur, it would be detected and compensated for or corrected by appropriate measures. The concept has been further elaborated since 1988 [41, 42]. Application of the concept of defence in depth throughout design and operation provides a graded protection against a wide variety of transients, anticipated operational occurrences and accidents, including those resulting from equipment failure or human action within the plant, and events that originate outside the plant.

Application of the concept of defence in depth in the design of a plant provides a series of levels of defence (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails.

1) The aim of the first level of defence is to prevent deviations from normal operation, and to prevent system failures. This leads to the requirement that the plant be soundly and conservatively designed, constructed, maintained and operated in accordance with appropriate quality levels and engineering practices, such as the application of redundancy, independence and diversity. To meet this objective, careful attention is paid to the selection of appropriate design codes and materials, and to the control of fabrication of components and of plant construction. Design options that can contribute to reducing the potential for internal hazards (e.g. controlling the response to a postulated initiating event), to reducing the consequences of a given postulated initiating event, or to reducing the likely release source term following an accident sequence contribute at this level of defence. Attention is also paid to the procedures involved in the design, fabrication, construction and in-service plant inspection, maintenance and testing, to the ease of access for these activities, to the way the plant is operated and to how operational experience is utilized. This whole process is supported by a detailed analysis which determines the operational and maintenance requirements for the plant.

2) The aim of the second level of defence is to detect and intercept deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions. This is in recognition of the fact that some postulated initiating events are likely to occur over the service lifetime of a NPP, despite the care taken to prevent them. This level necessitates the provision of specific systems as determined in the safety analysis and the definition of operating procedures to prevent or minimize damage from such postulated initiating events.

3) For the third level of defence, it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or postulated initiating events may not be arrested by a preceding level and a more serious event may develop. These unlikely events are anticipated in the design basis for the plant, and inherent safety features, fail-safe design, additional equipment and procedures are provided to control their consequences and to achieve stable and acceptable plant states following such events. This leads to the requirement that engineered safety features be provided that are capable of leading the plant first to a controlled state, and subsequently to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material.

4) The aim of the fourth level of defence is to address severe accidents in which the design basis may be exceeded and to ensure that radioactive releases are kept as low as practicable. The most important objective of this level is the protection of the confinement function. This may be achieved by complementary measures and procedures to prevent accident progression, and by mitigation of the consequences of selected severe accidents, in addition to accident management procedures. The protection provided by the confinement may be demonstrated using best estimate methods.

5) The fifth and final level of defence is aimed at mitigation of the radiological consequences of potential releases of radioactive materials that may result from accident conditions. This requires the provision of an adequately equipped emergency control (see Section 3 on Emergency Preparedness and Response).

A relevant aspect of the implementation of defence in depth is the provision in the design of a series of physical barriers to confine the radioactive material at specified locations. The number of physical barriers that will be necessary will depend on the potential internal and external hazards, and the potential consequences of failures. The barriers may, typically for water cooled reactors, be in the form of the fuel matrix, the fuel cladding, the reactor coolant system pressure boundary and the containment.

The earthquake on 11 March 2011 caused major damage to the infrastructure in the region, including the loss of connections from the off-site power grid to the Fukushima Daiichi NPP. This resulted in a deviation from the normal operation of the plant (defence in depth Level 1). After the earthquake, power supply was successfully provided from on-site sources, and all safety systems at defence in depth Level 3 continued to function as designed. This indicated that the safety systems and equipment withstood the seismic hazard [8].

The plant was built close to sea level and protection against flooding hazards was not sufficient, as the risk of flooding was not appropriately estimated [27]. Key safety equipment was not protected in leak tight compartments or by locating it at higher elevations to provide protection from flooding. This led to the loss of provisions for residual heat removal and containment cooling at defence in depth Levels 1, 2 and 3.

The flooding was the common cause of failure of the emergency power supply system, the near complete loss of systems providing DC power to measuring and control devices, and the destruction of the structures and components providing seawater cooling for the plant.

The objectives of defence in depth Level 4 are the prevention of accident progression and the mitigation of the consequences of a severe accident. For actions at Level 4, the operators needed to use all available means to supply water to the reactor in order to ensure the adequate removal of residual heat. This required the availability of instruments which provide reliable information on the key safety parameters and simple, reliable means for pressure relief in the reactor. In addition, the operators needed clear guidance and be trained to be able to initiate accident management measures [43].

As the accident progressed, operators lost the ability to reliably measure important safety parameters from the control room. This information was needed to assess the reactor status and to take well informed decisions on unusual actions and methods to cool the reactors. Nevertheless, the operators gave high priority to reactor cooling and managed to quickly prepare water supply lines with the intention of injecting coolant into the reactors using available low pressure pumps. However, the attempts to relieve reactor pressure failed because no provisions had been made to carry out this function after a complete loss of power. The required control power could not be restored in time to prevent core damage [8].

The last physical barrier included in defence in depth Level 4 is the reactor containment. Its purpose is to mitigate the consequences of accidents by preventing large radioactive releases to the environment after reactor damage. Depending on the containment type, various systems or kinds of equipment are needed to protect the containment against physical phenomena associated with core damage accidents that could challenge the containment integrity. The units at the Fukushima Daiichi NPP included means for the controlled venting of the containment to relieve the overpressure that might be caused by a steam leakage from the reactor cooling circuit. In addition, the atmosphere inside the containment was filled with inert nitrogen in order to eliminate hydrogen burn and prevent possible explosions.

Measurements taken during the accident indicate that the containment pressures of Units 1, 2, and 3 at certain times increased to levels that were near, or higher than, those for which the respective containments had been designed. This increase in pressure was due to the loss of containment cooling systems and the generation of steam by the overheated reactor cores. Although some containment venting systems were successfully opened, indications are that the containments of Units 1, 2 and 3 failed, leading to the release of radioactive material and

hydrogen. The nitrogen atmosphere inside the containments had effectively prevented hydrogen burn and explosions from occurring in that confined space. However, as hydrogen leaked from the containments to the reactor buildings, hydrogen explosions occurred at Units 1, 3 and 4 [8].

The Fukushima Daiichi accident demonstrated that extreme natural hazards have the potential to invalidate or impair multiple levels of defence in depth [44, 45]. A systematic identification and assessment of external hazards and robust protection against these hazards needs therefore to be considered for all levels of defence in depth. Furthermore, the accident showed that alternative design provisions and accident management capabilities could still ensure the supply of cooling water to the reactor even if all prime safety systems designed to protect the reactor against accidents were lost. However, the timely use of such provisions requires instruments that can provide reliable information on the key safety parameters and simple, reliable means to relieve the pressure in the reactor, so that any means can be used to supply cooling water to the reactor.

2.2.3. Assessment of the failure to fulfil fundamental safety functions

The three fundamental safety functions important for ensuring safety are: the control of reactivity in the nuclear fuel; the removal of heat from the reactor core and spent fuel pool; and the confinement of radioactive material. Following the earthquake, the first fundamental safety function, control of reactivity, was fulfilled in all six units at the Fukushima Daiichi NPP.

The second fundamental safety function — removing heat from the reactor core and the spent fuel pool — could not be maintained because the operators were deprived of almost all means of control over the reactors of Units 1, 2 and 3, and the spent fuel pools as a result of the loss of most of the AC and DC electrical systems. The loss of the second fundamental safety function was, in part, due to the failure to implement alternative water injection because of delays in depressurizing the reactor pressure vessels. Loss of cooling led to overheating and melting of the fuel in the reactors.

The confinement function was lost as a result of the loss of AC and DC power, which rendered the cooling systems unavailable and made it difficult for the operators to use the containment venting system. Venting of the containment was necessary to relieve pressure and prevent its failure. The operators were able to vent Units 1 and 3 to reduce the pressure in the primary containment vessels, however, this resulted in radioactive releases to the environment. Even though the containment vents for Units 1 and 3 were opened, the primary containment vessels for Units 1 and 3 eventually failed. Containment venting for Unit 2, was not successful, and the containment failed, resulting in radioactive releases.

Box 2.6. Fundamental safety functions

The three fundamental safety functions are:

1) control of the reactivity;

2) heat removal;

3) confinement of radioactive materials.

Prior to the accident at the Fukushima Daiichi NPP, there were two accidents involving the failure to maintain one or more of the fundamental safety functions. The accident in 1979 at the Three Mile Island NPP in the USA occurred as a result of the loss of the second of these safety functions, but radioactive releases to the environment were successfully prevented by the third function, confinement of radioactive material by the containment vessel. The accident in 1986 at the Chernobyl NPP in the former Soviet Union occurred as a result of the loss of the first of these safety functions. This plant did not have a containment vessel. Consequently, the Chernobyl accident resulted in a very large radioactive release to the environment. The Fukushima Daiichi accident occurred because of the loss of the second and third of these safety functions following a combination of extreme external events.

Control of the reactivity of nuclear fuel in the reactor core

The safety systems for controlling the reactivity of nuclear fuel in the reactor core are the reactor protection and the control rod drive systems. Prior to the earthquake, Fukushima Daiichi Units 1–3 were operating; Units 4–6 were shut down for maintenance. The reactors of Units 1–3 were automatically shut down by their reactor protection systems, which were activated by the seismic event monitoring equipment. The insertion of reactor control rods by the control rod drive systems terminated the nuclear chain reaction in the nuclear fuel and brought the reactors to a shutdown condition.

Removal of heat from the nuclear fuel

Following the shutdown of Units 1–3, residual heat, produced by the ongoing decay of radioactive substances in the fuel, was removed by the reactor cooling systems. This maintained the second fundamental safety function. These cooling systems comprised both closed circulation loops to transfer heat to the sea water and various means to inject water at high and low pressure into the reactor cores to remove this residual heat (see Section 2.1).

Many of these systems needed AC power to operate, and all of them needed DC power to control their operation. Most sources of power were lost during the course of the accident; this part of the report focuses on the impact of this loss of power.

Unit 1

The isolation condenser (see Box 2.2) for cooling Unit 1 started automatically as the result of a high reactor vessel pressure signal. It opened the isolation valves in the condensate return lines (other isolation valves in the lines were open during normal operation) when the reactor shut down following the earthquake. As required by the operating procedures, the isolation condenser system was stopped and restarted several times by the operators to prevent the reactor from cooling down too quickly and causing thermal stress exceeding the reactor pressure vessel design values. This was done by opening and closing the isolation valves in the condensate return lines [8].

At the time the tsunami inundated the site and electrical power was lost, the operators had just stopped the isolation condenser system by closing a valve on the return line outside the primary containment vessel. Operators had no information available on the isolation condenser valve positions, and it was not until approximately three hours later that they first attempted to manually restart the isolation condenser. The operators were not fully trained to understand how the valves worked under these conditions. They ultimately made two unsuccessful attempts from the control room to restart the isolation condenser by opening the outer isolation valves. The operators had no procedures to manually operate the isolation condenser. At the time of writing, the exact location of all of the valves in the isolation condenser system is unknown, but indications are that the isolation condenser did not function following the tsunami [8].

The steam turbine operated high pressure coolant injection system was not available because its DC power was lost.

Following the loss of the isolation condenser and the high pressure coolant injection system, an alternative means of injecting water into the reactor pressure vessel that relied on low pressure equipment was needed, such as that provided by the pumps for fire-fighting or fire trucks. The operators prepared the injection pathways in due time, but to be able to inject water at low pressure, it was also necessary to reduce the pressure inside the reactor pressure vessel using the safety relief valves. These valves could not be opened because of the loss of control power and high pressure air. The pressure in both the reactor vessel and in the containment was too high to allow the injection of sufficient water to cool the fuel without venting the containment and depressurizing the reactor pressure vessel. The alternative low pressure water injection systems were thus incapable of injecting water into the reactor pressure vessel.

The pressure in the reactor vessel remained high until the core was severely damaged. The most likely cause of pressure relief was a breach of the reactor pressure vessel due to melting [46]. The assumption that pressure relief was caused by a breach, is supported by the pressure increase in the containment a few hours after indications had been received of severe core damage. The consequent reduction in pressure provided the conditions for the first injection of water to the reactor pressure vessel about 12 hours after the tsunami. However, by this time, significant fuel damage had already occurred [8].

It is estimated that the damage to the reactor core occurred approximately 4–5 hours after the tsunami, and that the molten core breached the reactor vessel bottom about 6–8 hours after the tsunami. The first signs of radioactive release to the environment were observed about 12 hours after the tsunami, and a large release took place when the Unit 1 containment was vented to prevent its breach, because of the high pressure, about 23 hours after the tsunami. Chemical reactions between fuel cladding and water had generated large quantities of hydrogen which passed from the reactor pressure vessel into the primary containment vessel and escaped further to the reactor building [8].

Unit 2

Unit 2 had a different design for removing residual heat from the reactor core. The reactor core isolation cooling system (see Box 2.2) used steam from the reactor pressure vessel to drive a turbine which pumped water into the reactor vessel. The Unit 2 reactor core isolation cooling system was manually started after off-site power was lost. To remotely operate this system, DC power was needed, and it was designed to operate for at least four hours. However, the system continued to operate in harsh conditions for about 68 hours without DC power and without operator interventions [8]. This system successfully maintained the water level in the reactor pressure vessel above the fuel and ensured the cooling function.

There are indications that, after about 68 hours, the reactor core isolation cooling system failed. It was therefore no longer possible to inject water into the reactor pressure vessel because it was at high pressure. The water level in the reactor pressure vessel was estimated to drop to the reactor core top in a few hours after the reactor core isolation cooling system stopped functioning. The operators relied on alternative equipment for the low pressure injection of water, similar to that available for Unit 1. After some initial difficulties, they succeeded in reducing the pressure in the reactor pressure vessel by using safety relief valves, although injection was too late to prevent the rapid heating of the fuel and damage to the reactor core.

The containment venting system failed to relieve the pressure at Unit 2. It is assumed that this failure occurred because the rupture disc did not break. It is estimated that the Unit 2 reactor core began to melt about 76 hours after the tsunami. Radioactive releases started about 89 hours after the tsunami, following the breach of the containment boundary as indicated by the rapid drop in containment pressure [47].

Unit 3

In Unit 3, DC power was available for about two days, in contrast to the situation in Units 1 and 2. This meant that the reactor core isolation cooling and the high pressure coolant injection systems using steam driven pumps were available. Initially, the operators were able to maintain water levels in the reactor core by injecting water with the reactor core isolation cooling system. They followed procedures that enabled them to maximize the battery life available for the reactor core isolation cooling system [8].

In addition, steam relief from the reactor pressure vessel to the suppression chamber was available, and the suppression chamber pressure could be controlled using spray water provided by fire pumps. This situation lasted for 20 hours, until the reactor core isolation cooling system stopped and could not be restarted. The high pressure coolant injection system started automatically injecting water into the reactor pressure vessel to maintain the water level.

The high pressure coolant injection system is intended to rapidly refill the reactor pressure vessel following a leak in the reactor coolant system. This system was very effective in reducing pressure in the reactor pressure vessel. However, this led to a situation where inlet steam pressure to the pump turbine dropped below the pump's specifications, and the efficiency of the pump decreased significantly. The operators decided to shut down the system after about 14 hours because of concerns that the system could fail and begin to leak radioactive material outside the containment.

Following the shutdown of the high pressure coolant injection system, operators prepared an injection line to the reactor vessel and were prepared to inject seawater into the reactor vessel. However, due to the high reactor pressure, it was not possible to inject sea water until the reactor was depressurized. Therefore, because of the delay in injecting sea water into the reactor vessel, the water level continued to drop close to the top of the fuel. It has been assessed that an automatic signal, suspected to be false, triggered the automatic fast depressurization by safety relief valves before the operators could open the safety valves in a more controlled manner [48]. The depressurization, together with the low water level in the pressure vessel, is estimated to have caused the remaining water in the reactor core to flash to steam, resulting in the loss of adequate core cooling. The subsequent series of events, leading to the loss of reactor core cooling, were similar to those at Unit 2.

The core started to overheat and the major steam discharge from the reactor pressure vessel to the containment suppression chamber increased the pressure to a level that caused the rupture disc in the vent line to breach, opening a release path to the environment [8]. Melting of the Unit 3 reactor core is estimated to have started about 43 hours after the tsunami. Large radioactive releases started about 47 hours after the tsunami [8].

Unit 4

Unit 4 was undergoing a scheduled inspection and was shut down before the accident. All fuel of Unit 4 was in the spent fuel pool at the time of the accident. Therefore, cooling of the Unit 4 reactor core was not necessary. Cooling of the spent fuel pool was not possible owing to the loss of electrical power, and as a result the pool temperature started to increase.

Spent fuel pools

In the first few days following the tsunami, the operators considered that there was sufficient water in the spent fuel pools, and overheating of the fuel was not an immediate issue. This view changed on 15 March, when the Unit 4 reactor building exploded. At the time, it was thought that the cause of the explosion was hydrogen, and the only possible source of hydrogen in Unit 4 was thought to be from overheated fuel in the spent fuel pool due to the loss of water cover. This immediately raised concerns about how much water remained in that pool, and efforts were made to determine the water level in spent fuel pools.

On 16 March, visual inspections indicated that there was still water in the pool at Unit 4. However, concerns were raised about the status of Unit 3, which led to various mitigation efforts, including the air dropping of water from helicopters. Subsequent analysis and inspections revealed that the water level in the spent fuel pools of both Units 3 and 4 had not dropped to the level of the spent fuel. These inspections confirmed that the explosion in Unit 4 was caused by hydrogen, and the source of the hydrogen was not the fuel in the Unit 4 spent fuel pool, but the migration of hydrogen from Unit 3 to Unit 4 via a common ventilation system. However, the lack of knowledge about the actual conditions in the spent fuel pools during the accident, due to the loss of instrumentation, led to the effort to add water to the pool. A detailed description of the spent fuel pool events is given in Section 2.1.

Units 5 and 6

Units 5 and 6 were also affected by the tsunami, but their reactors were generating less residual heat, because they had been shut down for a considerable period prior to the

accident. In addition, one of the emergency diesel generators at Unit 6 had survived the flooding and was operable. The operators therefore had more time to respond, and the cooling systems for both units were powered by the one remaining emergency diesel generator. This power supply maintained the cooling of the reactor cores and was eventually used to provide cooling to the spent fuel pools in Units 5 and 6, both of which were successfully cooled down to a safe condition [8].

Confinement of radioactive material and control of radioactive releases

As a result of the damage to the Units 1-3 reactor cores, large amounts of steam and hydrogen escaped the reactor pressure vessels. This, in turn, pressurized and heated the primary containment vessels. These vessels were breached and steam, hydrogen and other gases, together with radioactive material, were released into the reactor buildings and eventually to the environment.

The primary containment vessels of the reactors had not been designed to withstand the pressure that could be generated in a severe accident; because of this, venting systems had been installed in the 1990s [22, 23]to limit the pressure in the containment vessels in the event of an accident. There are indications that the primary containment vessels for Units 1–3 failed at various stages in the progression of the accident. This was the result of the pressure and temperature in the primary containment vessel rising to levels that were far in excess of their designed capability before venting could be implemented (see Section 2.1). The leakage of radioactive material from the reactor cores was partially mitigated by the suppression pools, which retained some of the radionuclides released from the reactor pressure vessels.

2.2.4. Assessment of beyond design basis accidents and accident management

Safety analyses conducted during the licensing process of the Fukushima Daiichi NPP, and during its operation, did not fully address the possibility of a complex sequence of events that could lead to severe reactor core damage. In particular, the safety analyses failed to identify the vulnerability of the plant to flooding and weaknesses in operating procedures and accident management guidelines. The probabilistic safety assessments did not address the possibility of internal flooding, and the assumptions regarding human performance for accident management were optimistic. Furthermore, the regulatory body had imposed only limited requirements for operators to consider the possibility of severe accidents.

The operators were not fully prepared for the multi-unit loss of power and the loss of cooling caused by the tsunami. Although TEPCO had developed severe accident management guidelines, they did not cover this unlikely combination of events. Operators had therefore not received appropriate training, had not taken part in relevant severe accident exercises, and the equipment available to them was not adequate in the degraded plant conditions.

The IAEA safety standards in force at the time of the accident required an assessment to be undertaken to determine whether safety functions could be fulfilled for all normal operational modes, accident conditions and beyond design basis accidents, including severe accidents. Important event sequences that could lead to a severe accident need to be identified using a combination of probabilistic methods, deterministic methods and sound engineering

judgement [27]. In addition, specific deterministic beyond design basis accident analyses need to be performed to investigate credible accident scenarios that can be used to introduce improvements in accident management measures [43]. It is therefore necessary to determine whether safety functions can be fulfilled in beyond design basis accident conditions.

Box 2.7. Deterministic and probabilistic safety assessments [49, 50]

Safety analyses are analytical evaluations of physical phenomena occurring at NPPs. Deterministic safety analyses for a NPP predict the response to postulated initiating events. A specific set of rules and acceptance criteria is applied. Typically, these should focus on neutronic, thermohydraulic, radiological, thermomechanical and structural aspects, which are often analysed with different computational tools.

Best estimate deterministic safety analyses should be performed to confirm the strategies that have been developed to restore normal operational conditions at the plant following transients due to anticipated operational occurrences and design basis accidents. These strategies are reflected in the emergency operating procedures that define the actions that should be taken during such events. Deterministic safety analyses are required to provide the input that is necessary to specify the operator actions to be taken in response to some accidents, and the analyses should be an important element of the review of accident management strategies. In the development of the recovery strategies, to establish the available time period for the operator to take effective action, sensitivity calculations should be carried out on the timing of the necessary operator actions, and these calculations may be used to optimize the procedures.

Deterministic safety analyses should also be performed to assist the development of the strategy that an operator should follow if the emergency operating procedures fail to prevent a severe accident from occurring. The analyses should be used to identify what challenges can be expected during the progression of accidents and which phenomena will occur. They should be used to provide the basis for developing a set of guidelines for managing accidents and mitigating their consequences.

While deterministic analyses may be used to verify that acceptance criteria are met, probabilistic safety assessment (PSA) may be used to determine the probability of damage for each barrier. PSA may thus be a suitable tool for evaluation of the risk that arises from low frequency sequences that lead to barrier damage, whereas a deterministic analysis is adequate for events of higher frequency.

Deterministic safety analyses have an important part to play in the performance of a PSA because they provide information on whether the accident scenario will result in the failure of a fission product barrier. A PSA fault tree is a powerful tool that can be used to confirm assumptions that are commonly made in the deterministic calculation about the availability of systems.

The objectives of a PSA are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined. In the area of reactor safety, PSA uses a comprehensive, structured approach to identify failure scenarios. It constitutes a conceptual and mathematical tool for deriving numerical estimates of risk. The probabilistic approach uses realistic assumptions whenever possible and provides a framework for addressing many of the uncertainties explicitly. Probabilistic approaches may provide insights into system performance reliability, interactions and weaknesses in the design, the application of defence in depth, and risks, that it may not be possible to derive from a deterministic analysis.

Improvements in the overall approach to safety analysis have permitted a better integration of deterministic and probabilistic approaches. With increasing quality of models and data, it is possible to develop more realistic deterministic analysis and to make use of probabilistic information in selecting accident scenarios. Increasing emphasis is being placed on specifying probabilistically how compliance with the deterministic safety criteria is to be demonstrated, for example, by specifying confidence intervals and how safety margins are specified.

Several techniques can be used in performing a PSA. The usual approach is to use a combination of event trees and fault trees. The relative size (complexity) of the event trees and fault trees is largely a matter of preference of the analysis and also depends on the features of the software used.

The event trees outline the broad characteristics of the accident sequences that start from the initiating event and, depending on the success or failure of the mitigating safety and safety related systems, lead to a successful outcome or to damage to the core, or to one of the plant damage states (required for the Level 2 PSA). The fault trees are used to model the failure of the safety systems and the support systems to carry out their safety functions.

The fault trees should be developed to provide a logical failure model for the safety system failure states identified by the event tree analysis. The failure criterion that provides the top event of the fault tree for each safety system function should be the logical inverse of the accident sequence success criterion. The basic events modelled in the fault trees should be consistent with the available data on component failures. The fault tree models should be developed to the level of significant failure modes of individual components (pumps, valves, diesel generators, etc.) and individual human errors and should include all the basic events that could lead, either directly or in combination with other basic events, to the top event of the fault tree.

TEPCO started to perform probabilistic safety assessments, as well as some deterministic safety analyses of the more significant accident sequences, in the early 1990s. In line with the practice in IAEA Member States at that time, these probabilistic safety assessments were limited to events at single unit NPPs. Although the Fukushima Daiichi NPP was located in an area where tsunamis were possible, these analyses did not include common cause failures caused by flooding or extended loss of electrical power [8]. The probabilistic safety assessment studies for the Fukushima Daiichi NPP also did not consider internal flooding or fires, and the assumptions related to operator's actions were optimistic.

A comprehensive probabilistic safety assessment, including internal flooding sequences, is needed to highlight the vulnerability to flooding of vital plant systems, such as emergency diesel generators and electrical switchgear. In 1991, a corroded pipe leaked water at a rate of 20 cubic metres per hour, which penetrated the room with the reactor's emergency power system through the door and cable penetrations at Unit 1 at the Fukushima Daiichi NPP. This event demonstrated the vulnerability to flooding with respect to the location of the emergency diesel generators and the electrical switchgear in the basement.

Accident management guidance had also been evaluated at the Fukushima Daiichi NPP through limited scope probabilistic safety assessment studies. For example, these assessments contained the use of the containment vent system through the application of a fault tree approach to simulate the equipment failures with a human error probability for manual operation. But there was no thorough assessment of the challenges in severe accident management, including the limited training and guidance provided to plant personnel. It was not recognized that assumptions about failure probability were overly optimistic and the studies did not lead to improvements in procedures and guidance [49] (see Box 2.8 for accident management).

The Fukushima Daiichi NPP had some weaknesses which were not fully evaluated by a probabilistic safety assessment, as recommended by the IAEA safety standards [49, 51]. Examples include the lack of protection for the emergency diesel generators, battery rooms and switchgear against flooding and the low likelihood of success of severe accident interventions, given the limited guidance, training and knowledge of plant personnel for severe accident management. Beyond design basis accidents were not sufficiently considered, which affected the capability to maintain cooling of the reactor core, the operators' ability to monitor important safety parameters and the management of the severe accident conditions (See Fig. 2.7).

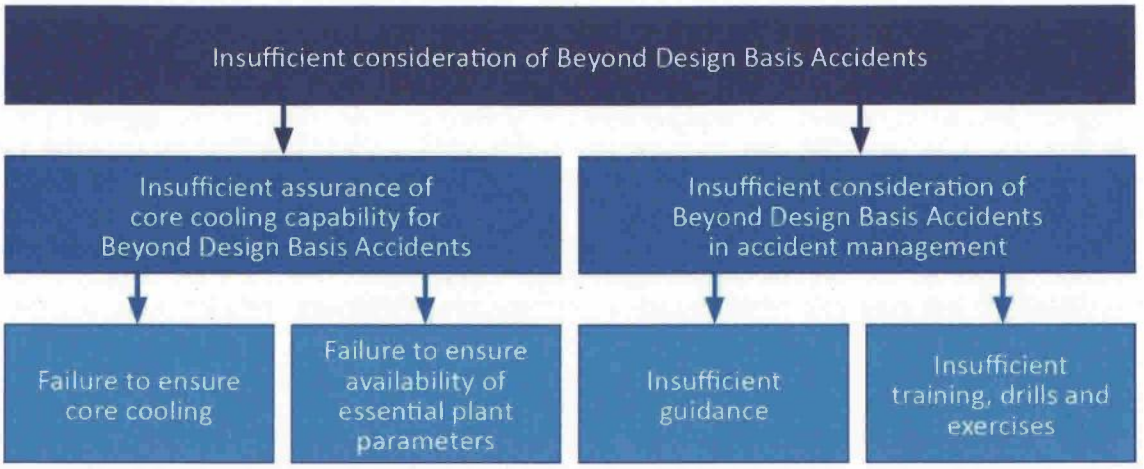

FIG 2.7. Impact of the insufficient considerations of beyond design basis accidents affected the capability to maintain cooling of the reactor core, the operators' ability to monitor important safety parameters and the management of the severe accident conditions [27, 52].

The limited scope of the regulatory body's requirements for beyond design basis accidents contributed to the lack of proper consideration of relevant risks by plant operators. This was highlighted during an IAEA Integrated Regulatory Review Service (IRRS) mission in June 2007, which concluded that "there are no legal regulations for the consideration of beyond the design basis [accidents], as Japanese plants are considered to be adequately safe as ensured by preventive measures". For example, the periodic safety review process in Japan did not require operating organizations to update their analyses to utilize the latest techniques [53].

Box 2.8. Accident management [43]

An accident management programme should be developed for all plants, irrespective of the total core damage frequency and fission product release frequency calculated for the plant. A structured top-down approach should be used to develop the accident management guidance. This approach should begin with the objectives and strategies, and result in procedures and guidelines, and should cover both the preventive and the mitigatory domains.

At the top level, the objectives of accident management are defined as follows: preventing significant core damage; terminating the progress of core damage once it has started; maintaining the integrity of the containment as long as possible; minimizing releases of radioactive material; achieving a long term stable state. To achieve these objectives, a number of strategies should be developed.

From the strategies, suitable and effective measures for accident management should be derived. Such measures include plant modifications, where these are deemed important for managing beyond design basis accidents and severe accidents, and personnel actions. These measures include repair of failed equipment. Appropriate guidance, in the form of procedures and guidelines, should be developed for the personnel responsible for executing the measures for accident management.

When developing guidance on accident management, consideration should be given to the full design capabilities of the plant, using both safety and non-safety systems, and including the possible use of some systems beyond their originally intended function and anticipated operating conditions, and possibly outside their design basis. The point at which the transition of responsibility and authority is to be made from the preventive to the mitigatory domain should be specified and should be based on properly defined and documented criteria.

For any change in the plant configuration, or if new results from research on physical phenomena become available, the implications for accident management guidance should be checked and, if necessary, a revision of the accident management guidance should be made.

The TEPCO accident management programme assumed that AC power would be promptly recovered at the Fukushima Daiichi NPP TEPCO also assumed that other essential utilities, such as DC power and high pressure air, would be available at all times to provide power for instrumentation and to provide for the operation of valves. The programme and guidelines did not cover the possibility that a severe accident could affect several reactor units simultaneously or difficulties in receiving support from outside the site because of serious disruption to the off-site infrastructure. This approach was in line with typical international practice at the time. The accident demonstrated that the operation of certain systems under beyond design basis conditions required exceptionally high skills on the part of the operators in order to maintain fundamental safety functions.

The accident management guidance in place at the Fukushima Daiichi NPP comprised a suite of documentation for use by the operators, including emergency operating procedures and severe accident management guidelines. Accident management guidelines were available for use by TEPCO technical support staff in the emergency response organization. Collectively, these documents covered the range of responses to design basis accidents and beyond design basis accidents, including severe accidents. The absence of electrical power and the lack of adequate information about the plant status made it difficult for the operators to effectively respond to the unfolding events. The accident management guidelines did not cover contingencies for the loss of instrumentation necessary to display the key parameters which allow operators to determine the status of the NPP. In addition, the guidelines did not provide recommendations for managing accidents when all safety related electrical distribution systems, and subsequently many of the dependent safety systems, were inoperable.

Personnel were not trained to perform accident management actions under conditions of prolonged station blackout or where information was limited or missing. Despite this, the operating staff performed their activities properly under the harsh conditions created by the accident. However, the inability to obtain fundamental information on the status of the plant and the need to improvise mitigation actions hampered the response. The absence of severe accident management requirements in the regulatory framework also contributed to the lack of preparation by TEPCO. The NSC published a guide on accident management in 1992 [23], and in the same year, the Ministry for International Trade and Industry (MITI) published a Roadmap of Accident Management. The Ministry also requested nuclear operating organizations to take actions to manage more severe accidents than those considered in the original design. However, this was not a mandatory requirement, and it resulted in limited voluntary actions by nuclear operating organizations. The IAEA IRRS mission to Japan in 2007 suggested the need for regulatory requirements for beyond design basis accidents, and suggested that NISA continue to develop a systematic approach to the consideration of such events, and also to the complementary use of probabilistic safety assessment and severe accident management [54]. The review mission's suggestions did not stimulate further efforts in this area.

2.2.5. Assessment of regulatory effectiveness

The regulation of nuclear safety in Japan at the time of the accident was performed by a number of organizations with different roles and responsibilities and complex inter-relationships. It was not fully clear which organizations had the responsibility and authority to issue binding instructions on how to respond without delay to safety issues.

The regulatory inspection programme was rigidly structured, which reduced the regulatory body's ability to verify safety at the proper times and to identify potential new safety issues.

The regulations, guidelines and procedures in place at the time of the accident were not fully in line with international practice in some key areas, most notably in relation to periodic safety reviews, re-evaluation of hazards, severe accident management and safety culture.

In the legal framework for nuclear safety applicable in Japan at the time of the accident, the Government had established main laws which were supplemented by subordinate laws and ministerial ordinances and statutes. The general structure of the legislative and regulatory framework at the time of the accident is illustrated in Figs 2.8 and 2.9. The regulatory structure in Japan at the time of the accident consisted of a number of government departments and other organizations with responsibilities for nuclear safety. The structure had been revised twice, following the radiation incident aboard the nuclear-powered ship Mutsu in 1974 and the accident at the nuclear fuel processing facility in Tokaimura in 1999, but some fundamental issues related to the clarity of roles and responsibilities had not been addressed [55, 56]. The IAEA IRRS mission in 2007 suggested the need to improve, refine and clarify a number of regulatory aspects [54], such as regulations regarding the treatment of beyond design basis accidents and clarification of the roles and responsibilities of NISA and the NSC within the Japanese regulatory system.

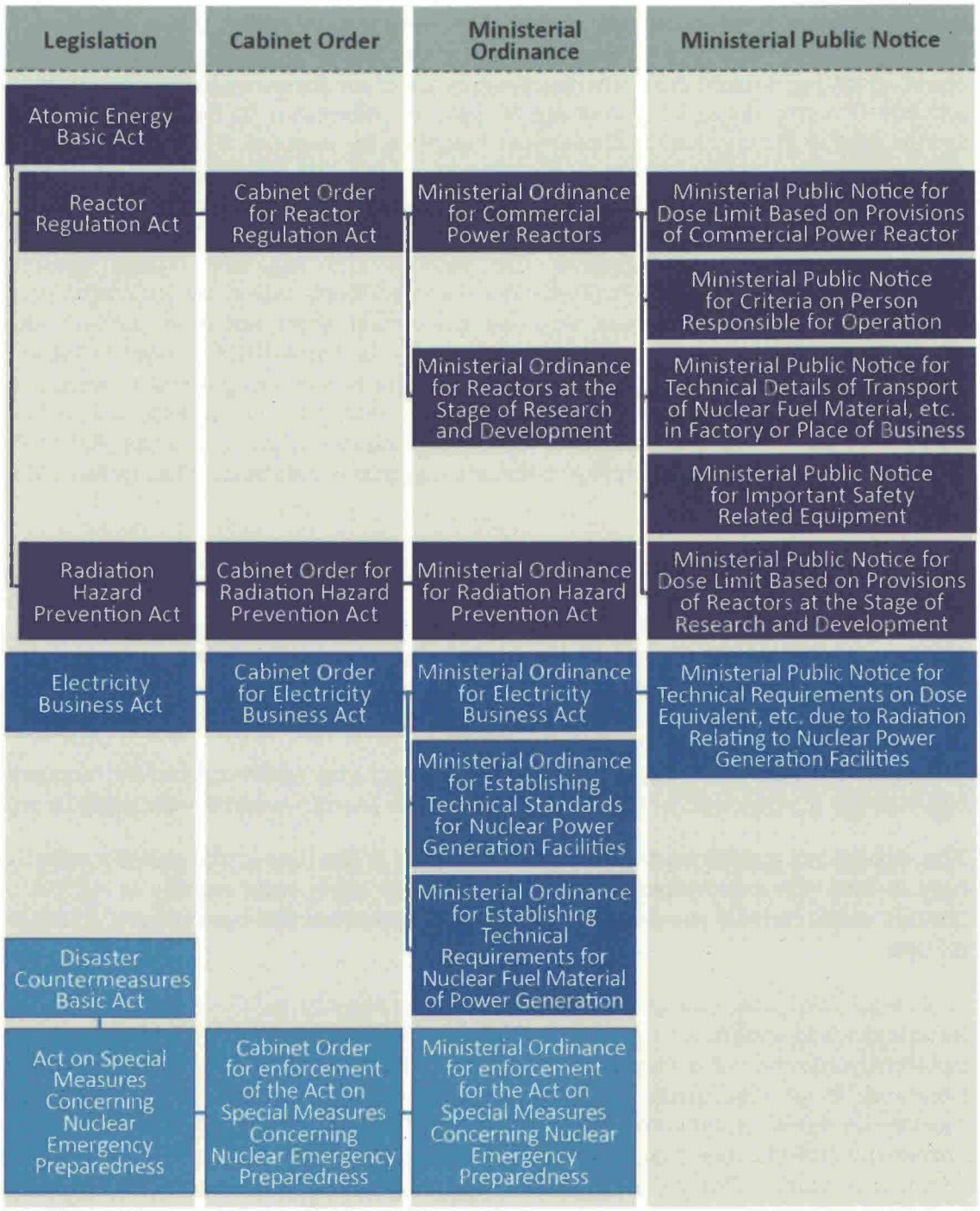

FIG 2.8. The legislative and regulatory framework of nuclear installations safety in Japan at the time of the accident [57].

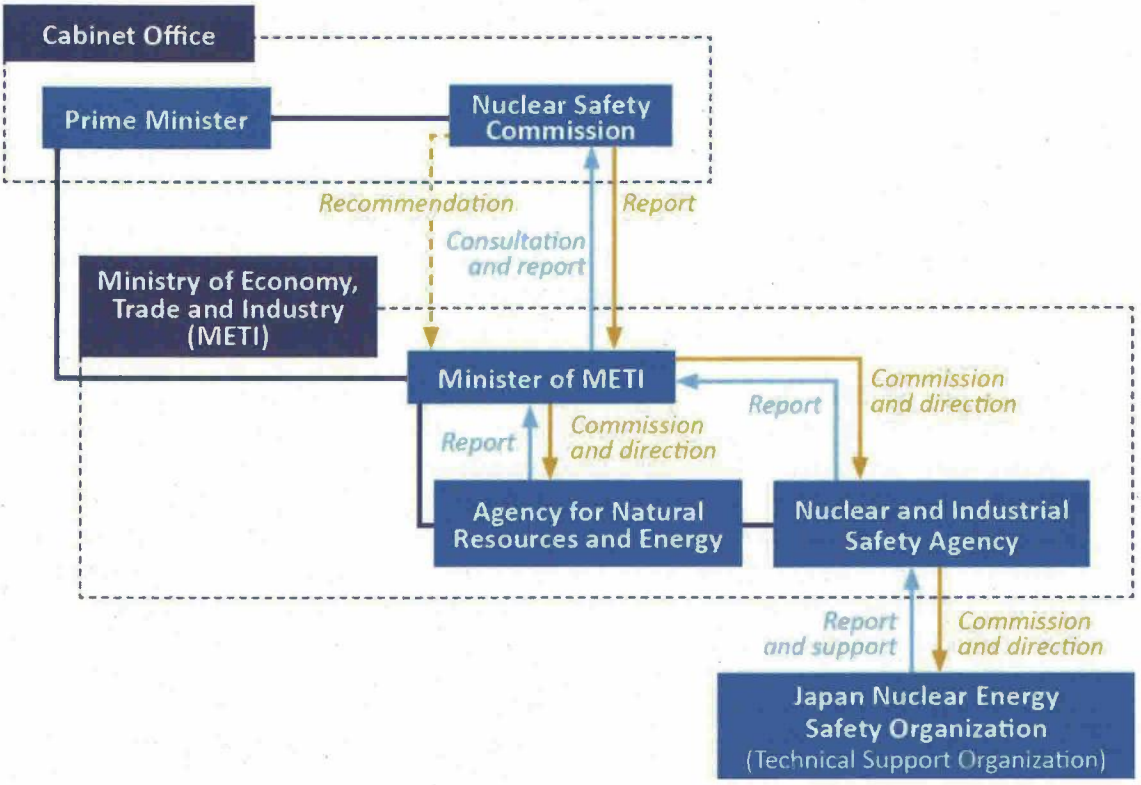

FIG 2.9. Position of NISA in the Japanese Government.

The Ministry of Economy, Trade and Industry (METI) was responsible for policy on the development and utilization of nuclear energy, as well as for the regulation of commercial nuclear installations. Within METI, the Agency for Natural Resources and Energy (ANRE) was responsible for overseeing the national energy supply, including promotion of nuclear energy. NISA was established in 2001 as a special agency attached to ANRE and given the responsibility as the nuclear safety regulatory body. The law required that, in case of conflict between safety and promotion, the Minister of METI should give priority to safety. METI established its National Strategic Plan based on this prioritization, and the IAEA IRRS mission in 2007 concluded that NISA was effectively independent of ANRE in its regulatory decision making. However, the mission also suggested that NISA's independence from METI be reflected more clearly in legislation.

The Ministry of Education, Culture, Sports, Science and Technology (MEXT) also had regulatory responsibilities, including oversight of radiation protection and nuclear material safeguards at NPPs, research reactors and certain research and development facilities for nuclear power. The Ministry also supervised the National Institute of Radiological Sciences (NIRS) and the Japan Atomic Energy Agency (JAEA).

The NSC, located in the Cabinet Office and reporting to the Prime Minister, was an independent body with both advisory and supervisory roles in the framework of nuclear regulation. It developed and issued the nuclear safety related policy documents and regulatory guides that were used by NISA in its regulatory work. The NSC was empowered by law to require reports from NISA and supervised NISA's work. It also had its own staff to

carry out independent reviews and assessments of NPP licence applications and to re-confirm the conclusions made by NISA. The IAEA IRRS mission in 2007 suggested that the role of NISA as the regulatory body in relation to the NSC was in need of clarification.

NISA was supported by the Japan Nuclear Energy Safety Organization (JNES) which was established in 2003 under a law passed in 2002 [54]. JNES's main functions consisted of conducting inspections at nuclear facilities, reviewing the periodic inspections of licensees, providing nuclear emergency preparedness support and coordinating safety related research projects. A comprehensive inspection programme is needed for a regulatory body to be able to independently identify plant safety issues. In Japan at the time of the accident, despite NISA's efforts [58], inspections were rigidly structured, with the type and frequency specified by law. In 2011, Japan's report to the Convention on Nuclear Safety stated that the operators' safety management activities were governed by the operational safety inspections that NISA had approved. NISA conducted quarterly inspections to check operators' compliance with the periodic safety reviews. Periodic inspections were also conducted by NISA and JNES at intervals not exceeding 13 months, focused on the operators' maintenance of structures, systems and components of the NPPs. They concentrated on safety significant structures, systems and components, for example, belonging to the reactor shutdown system, the reactor coolant pressure boundary, the residual heat removal system and the containment system. These regulatory procedures were in addition to the operators' own walkdown and maintenance management of nuclear installations, periodic safety assessments, and technical evaluations related to NPP ageing. NISA had limited ability to expand inspections beyond the legally defined scope, which restricted its ability to identify deficiencies and deviations and ensure that lessons were learned [54]. This approach limited the effectiveness of regulatory inspection to identify safety issues and to verify the safety of licensees' activities and their compliance with requirements.

A series of guidelines were issued by the NSC which were regarded, in practice, as requirements [36]. These guidelines were supplemented by consensus standards published by professional and academic societies. However, the regulations and guidelines in some key areas were not fully in line with international practice at the time of the accident. The most notable differences related to periodic safety reviews, re-evaluation of hazards, severe accident management and safety culture [53, 59, 60].

A periodic safety review provides a formal mechanism for re-examination by the licensees and the regulatory body of the design and external hazards in the light of new information and current standards and technology [53]. In Japan, periodic safety reviews at ten year intervals were required by regulations issued in 2003 [54], but these were of limited scope and not fully in line with international practice because they did not require a re-examination of external hazards [53, 54, 60].

The IAEA IRRS mission in 2007 suggested that NISA should be in a position to make a major contribution to the development of safety regulations. The mission also suggested that NISA needed the ability to modify its inspection programme in a flexible manner to optimize its effectiveness and focus and to be able to conduct safety inspections in locations and at times at its own discretion [54]. The IAEA IRRS mission also suggested that NISA foster frank and open relations with the nuclear industry and operating organizations in order to communicate regulatory concerns directly to the management level.

Establishment of the new regulatory authority

In September 2012, the Nuclear Regulation Authority (NRA) was established [61]. The NRA carried out a review of safety guidelines and regulatory requirements with the aim of formulating new regulations to protect people and the environment. In 2013, the new regulatory requirements for NPPs came into force. Based on the concept of defence in depth, importance was placed on the third and fourth levels and the prevention of simultaneous loss of all safety functions due to common causes. Previous assumptions on the impact of earthquakes, tsunamis and other external events such as volcanic eruptions, tornadoes, and forest fires were re-evaluated, and countermeasures for nuclear safety against these external events were considered. Countermeasures against internal fires and internal flooding and enhancements of the reliability of on-site and off-site power to deal with the possibility of station blackout were also considered.

In addition, countermeasures for severe accident response against core damage, containment vessel damage and a diffusion of radioactive materials, enhanced measures for water injection into spent fuel pools, countermeasures for airplane crashes, and the installation of an emergency response building were also required.

Examples of new regulatory requirements in light of the accident at the Fukushima Daiichi NPP include: (1) reinforced requirements for seismic/tsunami resistance; (2) reinforced or newly introduced requirements for design basis; and (3) newly introduced requirements for measures against severe accidents [62].

The roles and responsibilities that had previously been assigned to different governmental organizations were integrated into the NRA. The NRA holds jurisdiction over some of the activities of the NIRS and the JAEA. The main nuclear safety technical support organization JNES was merged with NRA on 1 March 2014.

Japan requested the IAEA to carry out an IRRS mission towards the end of 2015, aiming at strengthening nuclear safety and enhancing NRA's competence as an independent nuclear regulator through a continuous, transparent and open learning process.

2.2.6. Assessment of human and organizational factors

Before the accident, there was a basic assumption in Japan that the design of NPPs, and the safety measures that had been put in place, were sufficiently robust to withstand external events of low probability and high consequences.

Because of the basic assumption that NPPs in Japan were safe, there was a tendency for organizations and their staff not to challenge the level of safety. The reinforced basic assumption amongst the stakeholders about the robustness of the technical design of NPPs resulted in a situation where safety improvements were not introduced promptly.

The accident at the Fukushima Daiichi NPP showed that, in order to better identify plant vulnerabilities, it is necessary to take an integrated approach that takes account of the complex interactions between people, organizations and technology.

Box 2.9. Safety culture

In INSAG-4, a publication of the International Nuclear Safety Group (INSAG), safety culture was defined as: "that assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance" [63].

IAEA Safety Standard Series No.GS-G-3.5 [64] and Safety Reports Series No. 11 [65] make it clear that safety culture is itself a subset of the culture of the whole organization, with the latter comprising the mix of shared values, attitudes and patterns of behaviour that give the organization its particular character.

Organizations typically go through a number of phases in developing and strengthening safety culture. IAEA Safety Reports Series No. 11 identifies three stages:

1) Safety is compliance driven and is based mainly on rules and regulations. At this stage, safety is seen as a technical issue, with compliance with externally imposed rules and regulations considered adequate for safety.

2) Good safety performance becomes an organizational goal and is dealt with primarily in terms of safety targets or goals.

3) Safety is seen as a continuing process of improvement to which everyone can contribute.

In reality, the three phases are not distinct and any organization may have some parts that are ahead of others in the process of strengthening safety culture.

Prior to the accident, there was not sufficient consideration of low probability, high consequence external events which remained undetected. This was in part because of the basic assumption in Japan, reinforced over many decades, that the robustness of the technical design of the nuclear plants would provide sufficient protection against postulated risks. Consequently, the events that led to the accident at the Fukushima Daiichi NPP were outside the boundaries of the basic assumptions of the operating organizations, the regulatory body and the Government. These basic assumptions influenced the decisions and actions of a wide range of stakeholders, not limited to those directly involved in the operation and regulation of NPPs.

Box 2.10. Basic assumptions [66]

To understand safety culture in its entirety, the artefacts, espoused values and basic assumptions that form the three levels of the concept of culture as it applies to safety must be identified. The application of the Three level model to a specific organization would reflect the uniqueness of that organization, and allow logical links to be made between the artefacts, espoused values and basic assumptions. Logical links will not be apparent in the illustrative examples shown below, as they are not derived from any particular organization.

Artefacts are the easiest to observe, but their meaning are the most difficult to interpret. Knowledge of espoused values will help with the meaning, but it is only when the basic assumptions are understood that the meaning of the components at the artefact level will become apparent.

Artefacts and behaviour: Architecture, greeting rituals, dress, forms of address — visible;

Espoused values: Strategies, goals, philosophies — can be elicited;

Basic assumptions: Human nature, basis on which people are respected — unconsciously held and usually tacit.

Basic assumptions lie at the deepest level of culture. They are fundamental beliefs that are so taken for granted that most people in a cultural group subscribe to them, but not in a conscious way. To understand any culture it is necessary to unearth these basic assumptions that are operating. In the case of an organization, they will also reflect its history, and the values, beliefs and assumptions of the founders and key leaders who have made it successful. Basic assumptions are rarely discussed and confronted and extremely difficult to change.

The basic assumptions influenced NISA to not exercise sufficient authority and, thus, NISA was not able to challenge other assumptions regarding nuclear safety. It was inhibited in the fulfilment of its oversight role by the lack of an appropriate regulatory framework and also by explicit legal constraints [6, 54]. For instance, the IAEA IRRS mission in 2007 found that NISA inspectors did not have unfettered access to licensees' utilities to perform inspections and were only allowed to conduct inspections during certain times. Due to the basic assumption that the robustness of the technical design would provide sufficient protection against postulated risks, NISA worked generally in a less integrative and more reactive manner, in some cases being focused on short term activities, and did not address more fundamental and long term issues such as the consideration and application of IAEA safety standards. In some cases, regulations were not updated or complex emergency drills were not carried out because of a concern that the public might get an impression that NPPs were not safe in contrast to the basic assumption [5].

The same basic assumption that NPPs were safe also influenced TEPCO's actions, giving it confidence in the ability of the technical features of its plants to avoid severe nuclear accidents. This meant that TEPCO was not sufficiently prepared to mitigate the accident of March 2011 [6, 7, 67]. The risk of flooding triggering a nuclear accident was outside the basic assumption, so the latest international guidance on severe accident management was not always followed [68]. The basic assumption also excluded the possibility of a common cause failure which could lead to station blackout for multiple units.

Box 2.11. Systemic approach to safety [69]

The systemic approach to safety addresses the whole system by considering the dynamic interactions within and among all relevant factors of the system — individual factors (e.g. knowledge, thoughts, decisions, actions), technical factors (e.g. technology, tools, equipment) and organizational factors (e.g. management system, organizational structure, governance, resources).

The systemic approach to safety works by addressing this complex system of interactions as a whole. For example, among the important factors to consider in these interactions at a NPP are those related to individuals, such as knowledge, decisions, thoughts, emotions and actions. The technical factors include the physical aspects of the NPP and the range of technical tools and equipment used for operation. The organizational factors include the management system, organizational structure, governance of the plant and human and financial resources Taking into account the interaction between all the individual, technical and organizational factors reveals the complexity and non-linearity of the operations at a NPP. It is necessary to better examine the ways in which the weaknesses and strengths of all these factors influence one another in order proactively to reduce or eliminate risks.

The fact that relevant organizations and their staff did not challenge or re-examine the basic assumptions about nuclear safety illustrates a deficiency in safety culture. As identified in Box 2.9, the third stage of a safety culture programme identifies the need to have a continuing process of improvement, which should include periodic reconsideration of the adequacy of nuclear safety. One means of challenging basic assumptions is to take a systemic approach to nuclear safety and to understand the complexity of the full range of interactions between human, organizational and technical factors. Not sufficiently addressing these interactions was one of the contributory factors to the accident, because the basic assumption remained undetected.

The tendency of the majority of stakeholders not to challenge the adequacy of the existing safety features of the plant strengthened the assumption that the robustness of the plant's technical design and the existing safety measures would be sufficient to protect the plant. This led to necessary safety improvements not being made proactively and promptly [5, 6, 7].

The operators who directly responded in the early stages of the accident did so under extreme circumstances. The anxiety and stress associated with their actions were further exacerbated by the fact that they often did not have information about the safety of their families or the condition of their homes. Individuals at the site did not know how the accident would progress, which created significant uncertainty; despite this, they did all they could to protect people and the environment. The situation faced by the operator was unprecedented — managing a multiple unit accident during a national crisis with a heavily damaged infrastructure. This created an extremely adverse working environment from the physical and psychological perspectives.

The interaction of human, organizational and technical factors across all stakeholder organizations and between different levels inside each organization occur within the broader scope of the safety culture of the organization, and in this way reflects the organization's safety culture. With a systemic approach to safety that analyses the human, organizational and technical factors, an organization can be better prepared for an unexpected event. Nuclear safety will also depend on people's attitudes and behaviour [69]. Human and organizational factors that do not challenge or examine basic assumptions on safety may lead organizations and individuals to take decisions and perform actions that may inadvertently compromise

nuclear safety. It is important to be mindful of such basic assumptions and work to understand their impact on nuclear safety.

2.3. OBSERVATIONS AND LESSONS

A number of observations and lessons have been compiled as a result of the assessment of the nuclear safety considerations of the accident.

— **The assessment of natural hazards needs to be sufficiently conservative. The consideration of mainly historical data in the establishment of the design basis of NPPs is not sufficient to characterize the risks of extreme natural hazards. Even when comprehensive data are available, due to the relatively short observation periods, large uncertainties remain in the prediction of natural hazards.**

Extreme natural events that have a very low probability of occurrence can result in significant consequences, and the prediction of extreme natural hazards remains difficult and controversial due to the existence of uncertainties. Additionally, such predictions may change during the life of an NPP as more information becomes available and methods of analysis improve. It is therefore necessary to use all relevant available data, both domestic and international, to ensure a reliable prediction of hazards; define a reliable and realistic design basis against natural extreme events, and design NPPs with sufficient safety margins.

— **The safety of NPPs needs to be re-evaluated on a periodic basis to consider advances in knowledge, and necessary corrective actions or compensatory measures need to be implemented promptly.**

The periodic safety review programme at the Fukushima Daiichi NPP did not lead to safety upgrades based on regulatory requirements. TEPCO performed the re-evaluation on a voluntary basis considering advances in knowledge, including new information and data. When faced with a revised estimate of a hazard that exceeds previous predictions, it is important to ensure the safety of the installation by implementing interim corrective actions against the new hazard estimate while the accuracy of the revised value is being evaluated. If the accuracy of a new hazard estimate is confirmed, the operating organization and regulatory authority need to agree on a schedule and comprehensive action plan to promptly address the method of coping with such higher hazards to ensure plant safety.

— **The assessment of natural hazards needs to consider the potential for their occurrence in combination, either simultaneously or sequentially, and their combined effects on an NPP. The assessment of natural hazards also needs to consider their effects on multiple units on an NPP site.**

The Fukushima Daiichi accident demonstrated the need to fully investigate the potential for a combination of natural hazards affecting multiple units on an NPP site. The complex scenarios resulting from the occurrence of a combination of natural hazards need to be taken into account when considering accident mitigation measures and recovery actions.

— **Operating experience programmes need to include experience from both national and international sources. Safety improvements identified through operating experience programmes need to be implemented promptly. The use of operating experience needs to be evaluated periodically and independently.**

The operating experience evaluation programme at the Fukushima Daiichi NPP did not lead to design changes that took account of domestic or international experience involving flooding. The review of operating experience needs to be a standard part of plant oversight processes, with account taken of relevant sources such as the Incident Reporting System of the IAEA and the OECD Nuclear Energy Agency. Regulatory bodies need to perform independent reviews of national and international operating experience to confirm that operating organizations are taking concrete actions to improve safety.

— **The defence in depth concept remains valid, but implementation of the concept needs to be strengthened at all levels by adequate independence, redundancy, diversity and protection against internal and external hazards. There is a need to focus not only on accident prevention, but also on improving mitigation measures.**

The flooding resulting from the tsunami simultaneously challenged the first three protective levels of defence in depth, resulting in common cause failures of equipment and systems, Even when faced with this situation, operators were able to apply effective, albeit delayed, mitigation strategies. All layers of defence in depth associated with both prevention and mitigation of accidents should be strengthened by adequate independence, redundancy, diversity and protection so that they are not simultaneously challenged by an external or internal hazard and are not prone to common cause failure. The application of the defence in depth concept needs to be periodically re-examined over the lifetime of an NPP to ensure that any change in vulnerability to external events is understood and that appropriate changes to the design are made and implemented. There is a need for extreme external hazards to be addressed in periodic safety reviews, because such hazards can result in common cause failures that may simultaneously jeopardize several levels of defence in depth.

— **Instrumentation and control systems that are necessary during beyond design basis accidents need to remain operable in order to monitor essential plant safety parameters and to facilitate plant operations.**

The loss of instrumentation and control during the accident at the Fukushima Daiichi NPP left operators with little indication of actual plant conditions. The loss of instrumentation and control systems had a serious impact on efforts to prevent a severe accident or to mitigate its consequences. The extent and nature of the necessary instrumentation and control systems need to be defined with care, according to the characteristics of the design of the plant, including spent fuel pools. Systems need to be protected to ensure they are available when needed. This also demonstrated the need to improve strategies to allow for manual control of vital equipment.

— **Robust and reliable cooling systems that can function for both design basis and beyond design basis conditions need to be provided for the removal of residual heat.**

At the Fukushima Daiichi NPP, the operators were eventually, after some delay, able to deploy portable equipment to inject water into the reactors. Cooling systems based either on installed or portable equipment need to be qualified and tested to ensure that they function and can be deployed by operators when needed.

— **There is a need to ensure a reliable confinement function for beyond design basis accidents to prevent significant release of radioactive material to the environment.**

At the Fukushima Daiichi NPP, the failure of venting the containment, and the subsequent failure of the reactor building due to the hydrogen explosion, led to a significant release of radioactive material to the environment. The confinement function needs to be assessed to ensure that all possible hazards are considered in the design of equipment intended to maintain the integrity of the confinement system.

— **Comprehensive probabilistic and deterministic safety analyses need to be performed to confirm the capability of a plant to withstand applicable beyond design basis accidents and to provide a high degree of confidence in the robustness of the plant design.**

Safety analyses can be used both to evaluate, and to develop response strategies for, beyond design basis accidents and may include the use of both deterministic and probabilistic methods. The probabilistic safety assessment studies conducted for the Fukushima Daiichi NPP were of limited scope and did not consider the possibility of flooding from internal or external sources. The limitations in these studies contributed to the limited scope of accident management procedures available to the operators.

— **Accident management provisions need to be comprehensive, well designed and up to date. They need to be derived on the basis of a comprehensive set of initiating events and plant conditions and also need to provide for accidents that affect several units at a multi-unit plant.**

The accident management procedures available to the operators at the Fukushima Daiichi NPP did not consider the possibility of a multi-unit accident, nor did they provide guidance for the complete loss of electrical power. Accident management provisions need to be based on a plant specific analysis performed by using a combination of deterministic and probabilistic methods. Accident management guidance and procedures need to consider the possibility of events taking place in several units simultaneously and in spent fuel pools. They also need to take into account the possibility of disrupted regional infrastructure, including serious deficiencies in communication, transport and utilities. Accident management provisions should also take into consideration the best available guidance from the international community and be periodically updated to account for new information.

— **Training, exercises and drills need to include postulated severe accident conditions to ensure that operators are as well prepared as possible. They need to include the simulated use of actual equipment that would be deployed in the management of a severe accident.**

Operators at the Fukushima Daiichi NPP had not been specifically trained on how to manually operate systems such as the Unit 1 isolation condenser and fire trucks as an alternative source for low pressure water injection. Special attention is needed in personnel training to perform actions under conditions of prolonged loss of all power, with limited information about the plant status and no information on important safety parameters. Staff training, exercises and drills need to realistically simulate the progression of severe accidents, including the simultaneous occurrence of accidents in several units at the same site. Training, exercises and drills need to involve not only on-site accident management personnel but all off-site responders at the operating organization, local, regional and national levels.

— **In order to ensure effective regulatory oversight of the safety of nuclear installations, it is essential that the regulatory body is independent and possesses legal authority, technical competence and a strong safety culture.**

NISA did not have sufficient authority to take necessary actions, including inspections at regulated facilities. It is essential that the regulatory body is able to make independent decisions on safety over the lifetime of installations. To ensure such independent decision making, the regulatory body must be competent and must possess sufficient human resources, adequate legal authority — including the right to suspend operation and/or to impose improvements in safety on operating organizations — and adequate financial resources. The regulatory body needs the authority to adapt its inspection programme in the light of new safety information. It must also be able to ensure that national regulatory requirements and corresponding guidelines for assessing the safety of nuclear installations are revised periodically in accordance with scientific and technical developments, operational experience and international standards and practices.

— **In order to promote and strengthen safety culture, individuals and organizations need to continuously challenge or re-examine the prevailing assumptions about nuclear safety and the implications of decisions and actions that could affect nuclear safety.**

This can be achieved by individuals and organizations embracing a questioning attitude to identify the nature, boundaries and potential threats of their shared assumptions about nuclear safety. The institutionalization of a continuous dialogue within organizations, and among different organizations, on issues related to nuclear safety, and their significance and impact on decisions and actions, is essential. Periodic assessments of safety culture can help to foster reflection and dialogue on basic assumptions.

— **A systemic approach to safety needs to consider the interactions between human, organizational and technical factors. This approach needs to be taken through the entire life cycle of nuclear installations.**

The accident at the Fukushima Daiichi NPP showed that it is difficult to identify vulnerabilities in systems that involve complex interactions between people, organizations and technology because basic assumption regarding nuclear safety can remain undetected. A systemic approach that includes human, technological and

organizational considerations is necessary to understand how the components of the overall system function and interact in both normal operation and accident conditions.

3. EMERGENCY PREPAREDNESS AND RESPONSE

This section describes the key events and response actions from the onset of the accident on 11 March 2011. It also considers the national emergency preparedness and response system in place in Japan prior to the accident and the international framework for emergency preparedness and response.

Key international requirements for preparedness to respond to a nuclear emergency which existed prior to the accident are summarized in Box 3.1.

Box 3.1. Key requirements for preparedness to respond to a nuclear emergency in the IAEA safety standards prior to the accident

The IAEA safety standards [70, 71] in force prior to the accident required the following:

(1) utilizing an all-hazards approach in developing preparedness and response arrangements[62]; (2) developing an emergency classification system on the basis of observable conditions and measurable criteria (emergency action levels) and initiation of predetermined urgent protective actions for the public (in the predefined zones) promptly following the classification of the emergency by the operator; (3) establishing emergency zones for the full range of possible emergencies, including those of low probability; (4) establishing arrangements for the implementation of protective actions within the emergency zones and beyond, as required; (5) setting national criteria for decisions on public protective actions (evacuation, sheltering, iodine thyroid blocking, relocation, restriction of food and drinking water consumption and distribution, public monitoring and decontamination) in terms of doses and measurable quantities (operational intervention levels), taking account of a range of factors (such as financial and social aspects); (6) making arrangements for carrying out radiation monitoring and environmental sampling and assessment in order to identify new hazards promptly and to refine the strategy for response; (7) identifying, at the preparedness stage, special population groups within emergency zones (e.g. disabled persons, hospital patients) for whom specific arrangements need to be made; (8) establishing arrangements for emergency workers, including setting the dose criteria for different types of tasks, designating emergency workers and ensuring their protection, establishing guidance for managing, controlling and recording their doses, and providing specialized protective equipment, procedures and training; (9) planning for the transition from the emergency phase to long term recovery operations and resumption of normal social and economic activities, including clear allocation of responsibilities, sharing and transferring information, assessing consequences, establishing formal processes to decide on withdrawal of restrictions and other arrangements imposed during the emergency, setting relevant principles and criteria and consulting the public; (10) clearly assigning roles, responsibilities and authorities for emergency preparedness and response at all levels as part of emergency plans; (11) establishing organizational relationships and interfaces among operating and response organizations and preparing operational protocols to coordinate the emergency response at all levels; (12) developing and coordinating emergency plans and procedures at all levels on the basis of assessed hazards; (13) preparing for logistical support through provision of tools, instruments, supplies, equipment, communication systems, specific functional facilities and documentation, including planning for operability and usability of these items and facilities under postulated radiological, working and environmental conditions in the emergency response; (14) planning for and conducting of training, drills and exercises; and (15) establishing a quality assurance programme to ensure that all the supplies, equipment, communication systems, facilities and documentation, etc., are kept continuously up to date, available and functional for use in an emergency.

The types of protective actions in a nuclear emergency are summarized in Box 3.2.

[62] 'Arrangements' — the integrated set of infrastructural elements necessary to provide the capability for performing a specified function or task required in response to a nuclear or radiological emergency. These elements may include: authorities and responsibilities, organization, coordination, personnel, plans, procedures, facilities, equipment and training.

Box 3.2. Types of protective actions in a nuclear emergency [50, 71]

'Mitigatory actions' are immediate actions to reduce the potential for conditions to develop that would result in exposure or a release of radioactive material requiring emergency actions on or off the site, or to mitigate plant conditions that could result in exposure or a release of radioactive material requiring emergency actions on or off the site.

'Urgent protective actions' are actions that must be taken promptly (normally within hours) in order to be effective. The most common urgent protective actions in a nuclear emergency are evacuation, sheltering, iodine thyroid blocking, restriction of the consumption of potentially contaminated food and decontamination of individuals.

'Early protective actions' are those actions that must be taken within days or weeks to be effective. They can be long lasting, even after the termination of the emergency. Unlike urgent protective actions, it is generally possible to base these actions on the results of monitoring that takes account of the specific nature of the release of radioactive material and its dispersion in the environment. Examples of early protective actions include relocation, restrictions on food and drinking water and controls on agriculture.

3.1.INITIAL RESPONSE IN JAPAN TO THE ACCIDENT

At the time of the accident, separate arrangements were in place to respond to nuclear emergencies and natural disasters at the national and local levels. There were no coordinated arrangements for responding to a nuclear emergency and a natural disaster occurring simultaneously.

The arrangements to respond to nuclear emergencies envisaged that, following the detection of relevant adverse conditions at an NPP (e.g. loss of all AC power supplies for more than five minutes or loss of all capabilities to cool the reactor), a notification would be sent from the plant to local and national governments. The national government would then assess and determine whether the situation was to be categorized as a 'nuclear emergency'[63]. If the situation was categorized as a nuclear emergency, a declaration to that effect would be issued at the national level, and decisions about necessary protective actions would be taken on the basis of dose projections.

Based on a report from the Fukushima Daiichi NPP, the national government declared a nuclear emergency on the evening of 11 March and issued orders for protective actions for the public. The response at the national level was led by the Prime Minister and senior officials at the Prime Minister's Office in Tokyo.

The consequences of the earthquake and tsunami, and increased radiation levels, made the on-site response extremely difficult. The loss of AC and DC electrical power, the presence of a huge amount of rubble that hindered on-site response measures, aftershocks, alerts for further tsunamis and increased radiation levels meant that many mitigatory actions could not be carried out in a timely manner. The national government was involved in decisions concerning mitigatory action on the site.

[63] The Act on Special Measures Concerning Nuclear Emergency Preparedness, hereafter referred to as the Nuclear Emergency Act.

The activation of the emergency Off-site Centre, located 5 km from the Fukushima Daiichi NPP was difficult because of extensive infrastructure damage caused by the earthquake and tsunami. Within a few days, it became necessary to evacuate the Off-site Centre due to adverse radiological conditions.

The primary legal basis for the national emergency preparedness and response system in Japan was set out in the Disaster Countermeasures Basic Act [72] and the Act on Special Measures Concerning Nuclear Emergency Preparedness [20] (Box 3.3).

Box 3.3. Key documents defining the national emergency preparedness and response system for a nuclear emergency in Japan at the time of the accident

National legal basis	
Disaster Countermeasures Basic Act*	Act on Special Measures Concerning Nuclear Emergency Preparedness

National planning basis			
Basic Disaster Management Plan*	Order for Enforcement of the Act on Special Measures Concerning Nuclear Emergency Preparedness	Ordinance for Enforcement of the Act on Special Measures Concerning Nuclear Emergency Preparedness	Regulatory Guide on Emergency Preparedness for Nuclear Facilities

Operational plans and manuals		
National	Disaster Management Operation Plan*	Nuclear Emergency Response Manual
Prefectural/City/ Town/Village	Prefectural/City/Town/Village Disaster Management Plans*	Prefectural/City/Town/ Village Nuclear Manuals
Operators	Nuclear Operator Emergency Action Plans	Nuclear Operator Emergency Response Manuals

** These documents address various types of disasters, including nuclear emergencies*

3.1.1. Notification

Notification from the NPP to local and national governments was required under Article 10 of the Nuclear Emergency Act [20] when certain predefined 'specific events' occurred, such as failure of all AC power supplies for more than five minutes [57]. Under Article 15 of the Nuclear Emergency Act, a report of a 'nuclear emergency' would be sent when certain predefined criteria were met or exceeded, such as the loss of all capabilities to cool the reactor [19, 73].

It was assumed that a report of an event under Article 15 would follow a notification of an event under Article 10 [74]. Notification would trigger an assessment and judgement by the national government as to whether the event was a 'nuclear emergency'. If this was judged to be the case, the Prime Minister would be briefed and presented with a draft declaration of a 'nuclear emergency'. The Prime Minister would be responsible for deciding to declare a

'nuclear emergency' and issuing orders[64] and/or recommendations for protective actions to the public [75].

Key actions to be taken if an event fell under Article 10 and/or Article 15 of the Nuclear Emergency Act are summarized in Fig. 3.1 [20, 72, 75, 76, 77].

[64] The Nuclear Emergency Act [20] and the Disaster Countermeasures Basic Act [72] use the terms 'instructions' and 'recommendations' for issuing protective actions. An 'instruction' is mandatory and the public is therefore required to adhere to it. A 'recommendation' is only a suggestion and therefore not mandatory. However, for the purposes of clarity, the term 'orders' is used in this report as an equivalent of 'instructions'.

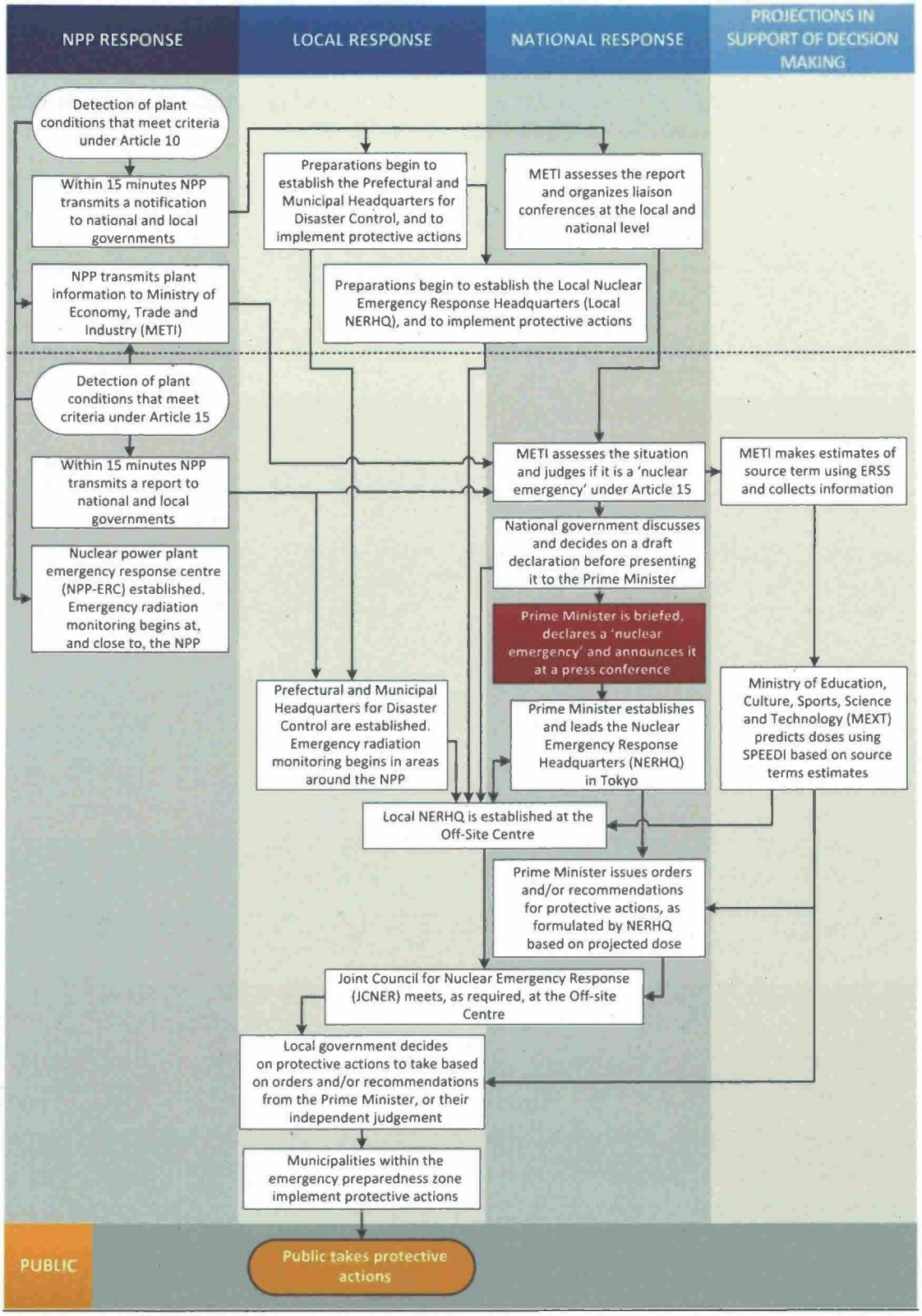

FIG. 3.1. Key actions if an event falls under Article 10 and/or Article 15 of the Nuclear Emergency Act, as planned prior to the accident (based on Refs [20, 72, 75, 76, 77]).

The tsunami wave that flooded the Fukushima Daiichi NPP arrived at 15:36 on 11 March 2011 [10]. Notification by the plant of a 'specific event' for Units 1 to 5 under Article 10 of the Nuclear Emergency Act [20] was sent to national and local governments at 15:42 on 11 March, followed at 16:45 by a report of an event at Units 1 and 2 classified as a 'nuclear emergency' under Article 15 of the Act [3, 8, 78, 79].

The type of the 'specific event' reported under Article 10 was a 'station blackout' for Units 1–5 [78]. The type of the event reported as a 'nuclear emergency' under Article 15 was initially the "inability of water injection of the emergency core cooling system" for Units 1 and 2 [79]. After receiving notification, the national government assessed the situation and made a judgement that the situation was a 'nuclear emergency' [6].

The Prime Minister issued a declaration of a nuclear emergency at 19:03. This was more than two hours after having been notified by the Fukushima Daiichi NPP of an event at Units 1 and 2 classified as a 'nuclear emergency' under Article 15 of the Act, following lengthy discussions among off-site officials [3].

3.1.2. Mitigatory actions

An emergency response centre, headed by the site superintendent, was established at the Fukushima Daiichi NPP, in accordance with TEPCO's Disaster Response Manual, around 15 minutes after the earthquake [6, 8]. It was located in the 'seismically isolated building', which was fitted with special features including an autonomous electrical power supply and ventilation systems with filtration devices. This building had been constructed[65] as a result of lessons learned from the experience of the Kashiwazaki-Kariwa NPP following the Niigata-Chuetsu-Oki earthquake in 2007, and its use enabled mitigatory actions to continue at the site during the response to the accident [8].

The arrangements that existed prior to the accident envisaged that, in case of need, the on-site emergency response centre would send a request for support to TEPCO headquarters, using TEPCO's capabilities or resources gathered from other nuclear operating organizations, through the Agreement on Cooperation between Japanese Nuclear Operators [8, 77].

Following a request from the Fukushima Daiichi NPP, additional staff and equipment from other Japanese NPPs (not operated by TEPCO) were mobilized to support the on-site emergency response. However, extensive damage caused to the transport infrastructure by the earthquake and tsunami, in addition to insufficient pre-planning, impaired the effectiveness of this support. For example, in cases when the request for equipment did not contain an adequate specification of what was required, it led to the procurement of equipment that was incompatible with existing plant equipment (due to mismatched fittings, connectors etc.) [8].

In response to the emergency, personnel from TEPCO, from contractors and from other Japanese NPPs (not operated by TEPCO) were dispatched to the site to assist with various tasks, including restoring power and monitoring instruments, injecting cooling water into reactors, removing rubble and monitoring radiation levels [8]. Personnel from national government agencies and organizations — such as the Japan Self-Defense Force, police and firefighters — were also dispatched to the site. They helped with activities including

[65] The construction started in March 2009 and the building was put into operation in July 2010.

operating the large equipment needed to pour or spray water onto the spent fuel pools in Units 1, 3 and 4 and providing helicopter surveillance of the spent fuel pools [3, 6, 8].

The earthquake and tsunami resulted in the loss of AC and DC electrical power and a huge amount of rubble. There were also aftershocks and alerts for possible further tsunamis. As a result of these factors, as well as increased radiation levels and the hydrogen explosions, and also due to the lack of detailed arrangements, the response was extremely difficult and many mitigatory actions could not be carried out in a timely manner [8]. Workers at the site carried out mitigatory actions under very difficult conditions; they worked longer hours and under far more tiring circumstances than would normally be expected [8].

The national government was involved in decisions concerning mitigatory actions, such as the injection of seawater for fuel cooling [6, 7]. Roles, responsibilities and authorities in this regard had not been clearly assigned at the preparedness stage.

3.1.3. Management of the emergency

The national emergency preparedness and response system in place at the time of the accident envisaged that the core entities in managing the nuclear emergency would be the Nuclear Emergency Response Headquarters (NERHQ)[66] and its Secretariat[67], as well as the Local Nuclear Emergency Response Headquarters (Local NERHQ)[68]. The NERHQ would direct and coordinate the national response, which was to include preparing and issuing orders and/or recommendations on evacuation to the local government [20].

For a national response at the local level, the overall management of the response to a nuclear emergency was to be coordinated, as soon as possible, by the Local NERHQ at the Off-site Centre, located 5 km from the Fukushima Daiichi NPP. The Local Prefectural Nuclear Emergency Response Headquarters and the Joint Council for Nuclear Emergency Response (JCNER) were also planned to be located in the Off-site Centre [75, 76, 80].

For the prefectural response to a nuclear emergency, it was planned that the Local Prefectural Nuclear Emergency Response Headquarters and the Fukushima Prefecture Headquarters for Disaster Control would coordinate activities at the prefectural level. The JCNER would coordinate between the national response at the local level and the prefectural response [20, 75, 76].

Separate arrangements were in place to respond to nuclear emergencies and natural disasters at the national and local levels. These arrangements did not envisage the need to respond to a nuclear emergency and a natural disaster occurring simultaneously [76, 80].

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[66] The NERHQ was to be composed of those appointed by the Prime Minister from among the officials of the Cabinet Secretariat and designated administrative organs [20]. The Prime Minister was to serve as the Director-General of the NERHQ, which was planned to be located within the Cabinet Office in the Prime Minister's Office (see Fig. 3.2).

[67] The Secretariat was to be staffed by representatives of key organizations and headed by the Director General of the Nuclear and Industrial Safety Agency (NISA), which was part of the Ministry of Economy, Trade and Industry (METI). It was planned to be located in the METI/NISA emergency response centre in the METI building (see Fig. 3.2).

[68] The Local NERHQ was to be staffed with individuals from all relevant organizations, with the METI Senior Vice Minister as Director General. It was planned to be located at the Off-site Centre (see Fig. 3.2).

The locations of the core entities in the management of the response to a nuclear emergency, as planned prior to the accident are shown in Fig. 3.2.

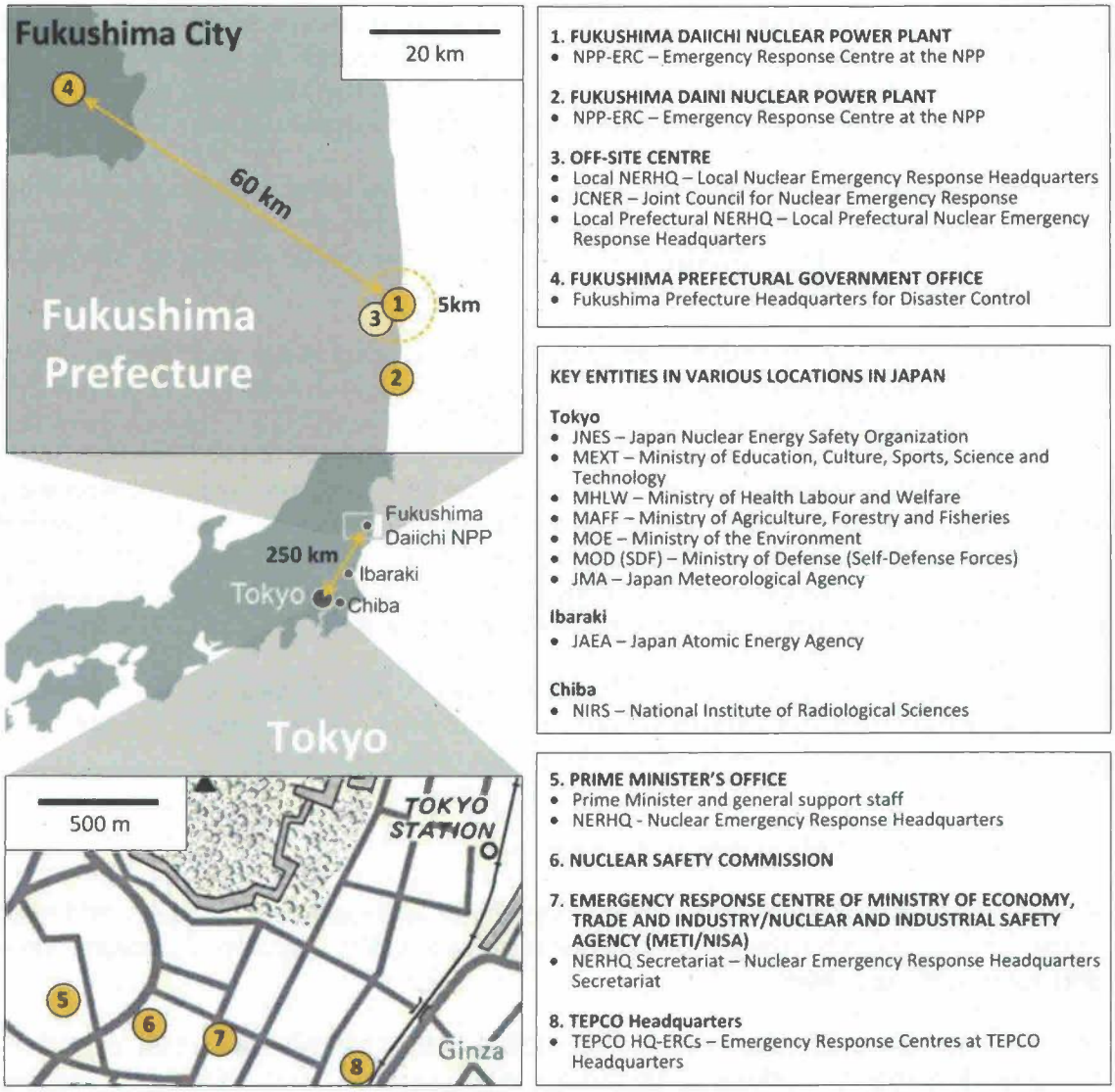

FIG. 3.2. Locations of the core entities involved in management of the response to a nuclear emergency[69].

At 14:50 on 11 March 2011, an Emergency Response Office for dealing with the earthquake was established in the Prime Minister's Office by the Deputy Chief Cabinet Secretary for Crisis Management. At 15:14, the national government established the Emergency Disaster Response Headquarters in the Prime Minister's Office in Tokyo, with the Prime Minister as the Headquarters Director General. At 16:36, the Deputy Chief Cabinet Secretary for Crisis Management established an Emergency Response Office for the nuclear accident at the Prime Minister's Office [6].

[69] Based on Refs [7, 8, 75, 76, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93].

At 19:03 on 11 March 2011, the national government established the Nuclear Emergency Response Headquarters, at the same time as the declaration of a nuclear emergency was issued [3].

As the accident evolved so fast, there was no time for detailed discussions during meetings of the NERHQ. The core group for the emergency response became the Prime Minister and senior officials, located at the Prime Minister's Office. The Prime Minister issued evacuation orders to local governments without the involvement of the Secretariat of the NERHQ [7].

The government — TEPCO Integrated Response Office — an integrated headquarters of the operating organization and the government response organization — was established on 15 March 2011 at TEPCO Headquarters in Tokyo [6] to ensure the timely sharing of information at the national level.

At the local level, the extensive damage caused by the earthquake and tsunami led to difficulties in initiating operations in the Off-site Centre [94]. As a result, the Local NERHQ and other entities, which were supposed to operate from the Off-site Centre (JCNER and Local Prefectural Nuclear Emergency Response Headquarters), could not fulfil their roles. On 15 March 2011, it became necessary to evacuate the Off-site Centre, due to the worsening radiological conditions[70], and to relocate to the Fukushima Prefectural Public Hall, located approximately 60 km from the Fukushima Daiichi NPP [6, 94]. This facility did not have capabilities equivalent to those of the Off-site Centre, which led to difficulties, for example, in sharing information in real-time among the relevant authorities.

For the prefectural response, a new 'nuclear squad'[71] was formed in the Fukushima Prefecture Headquarters for Disaster Control, as part of the structure set up to respond to the earthquake and tsunami, to coordinate activities at the prefectural level.[7].

3.2. PROTECTING EMERGENCY WORKERS

At the time of the accident, the national legislation and guidance in Japan addressed measures to be taken for the protection of emergency workers[72], but only in general terms and not in sufficient detail.

Many emergency workers from different professions were needed to support the emergency response. Emergency workers came from various organizations and public services. However, there were no arrangements in place to integrate those emergency workers who had not been designated prior to the accident into the response.

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[70] The Off-site Centre had not been designed to withstand the increasing radiation levels.

[71] A new nuclear squad was formed because the existing nine functional squads, as specified in the Fukushima Prefecture Disaster Management Plan [76], were engaged in response to the earthquake and tsunami [7].

[72] The IAEA uses the term 'emergency workers' to cover those with specified duties as a worker (any person who works, whether full time, part time or temporarily, for an employer and who has recognized rights and duties in relation to occupational radiation protection) in response to an emergency, including workers employed, both directly and indirectly, by registrants and licensees as well as personnel of responding organizations, such as police officers, firefighters, medical personnel, and drivers and crews of evacuation vehicles. In Japan the term 'emergency preparedness personnel' is used to cover all those who perform emergency response activities in a nuclear emergency, such as "... communication of public information and instructions to residents in the vicinity, guidance of residents in the vicinity for evacuation, traffic control, radiation monitoring, medical treatment provision, and actions to prevent a situation from developing into a disaster in a nuclear facility, and those who perform disaster recovery activities such as removal of radioactive contaminants" [95].