The Fukushima Daiichi Accident

Technical Volume 2/5
Safety Assessment

IAEA
International Atomic Energy Agency

THE FUKUSHIMA DAIICHI ACCIDENT

TECHNICAL VOLUME 2
SAFETY ASSESSMENT

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TECHNICAL VOLUME 2

SAFETY ASSESSMENT

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The IAEA thanks the large number of experts who were involved in this report. It is the result of the dedicated efforts of many people. All participants listed at the end of this technical volume made valuable contributions, but a particularly heavy load was borne by the Co-Chairs and coordinators of the working groups. The efforts of many expert reviewers, including members of the International Technical Advisory Group, are also gratefully acknowledged.

THE REPORT ON THE FUKUSHIMA DAIICHI ACCIDENT

At the IAEA General Conference in September 2012, the Director General announced that the IAEA would prepare a report on the Fukushima Daiichi accident. He later stated that this report would be "an authoritative, factual and balanced assessment, addressing the causes and consequences of the accident, as well as lessons learned".[1]

The report is the result of an extensive international collaborative effort involving five working groups with about 180 experts from 42 Member States (with and without nuclear power programmes) and several international bodies. This ensured a broad representation of experience and knowledge. An International Technical Advisory Group provided advice on technical and scientific issues. A Core Group, comprising IAEA senior level management, was established to give direction and to facilitate the coordination and review. Additional internal and external review mechanisms were also instituted. The organizational structure for the preparation of this publication is illustrated in Fig. 1.

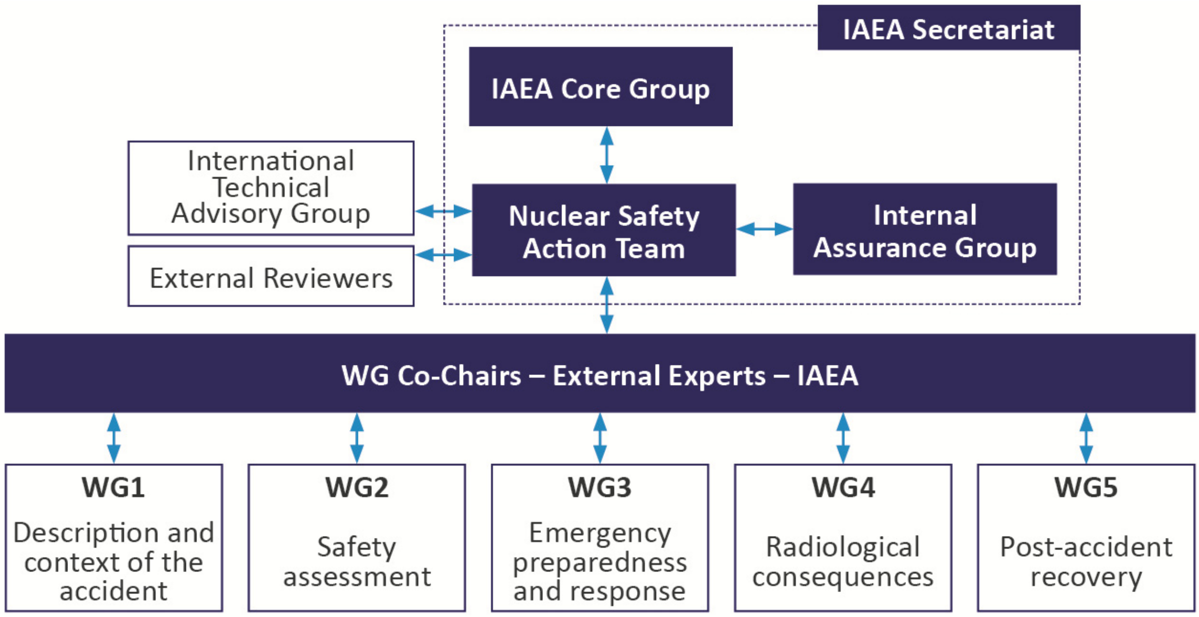

FIG. 1. IAEA organizational structure for the preparation of the report on The Fukushima Daiichi Accident.

The Report by the Director General consists of an Executive Summary and a Summary Report. It draws on five detailed technical volumes prepared by international experts and on the contributions of the many experts and international bodies involved.

The five technical volumes are for a technical audience that includes the relevant authorities in IAEA Member States, international organizations, nuclear regulatory bodies, nuclear power plant operating organizations, designers of nuclear facilities and other experts in matters relating to nuclear power.

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Introductory Statement to Board of Governors (2013), https://www.iaea.org/newscenter/statements/introductory-statement-board-governors-3.

The relationship between the content of the Report by the Director General and the content of the technical volumes is illustrated in Fig. 2.

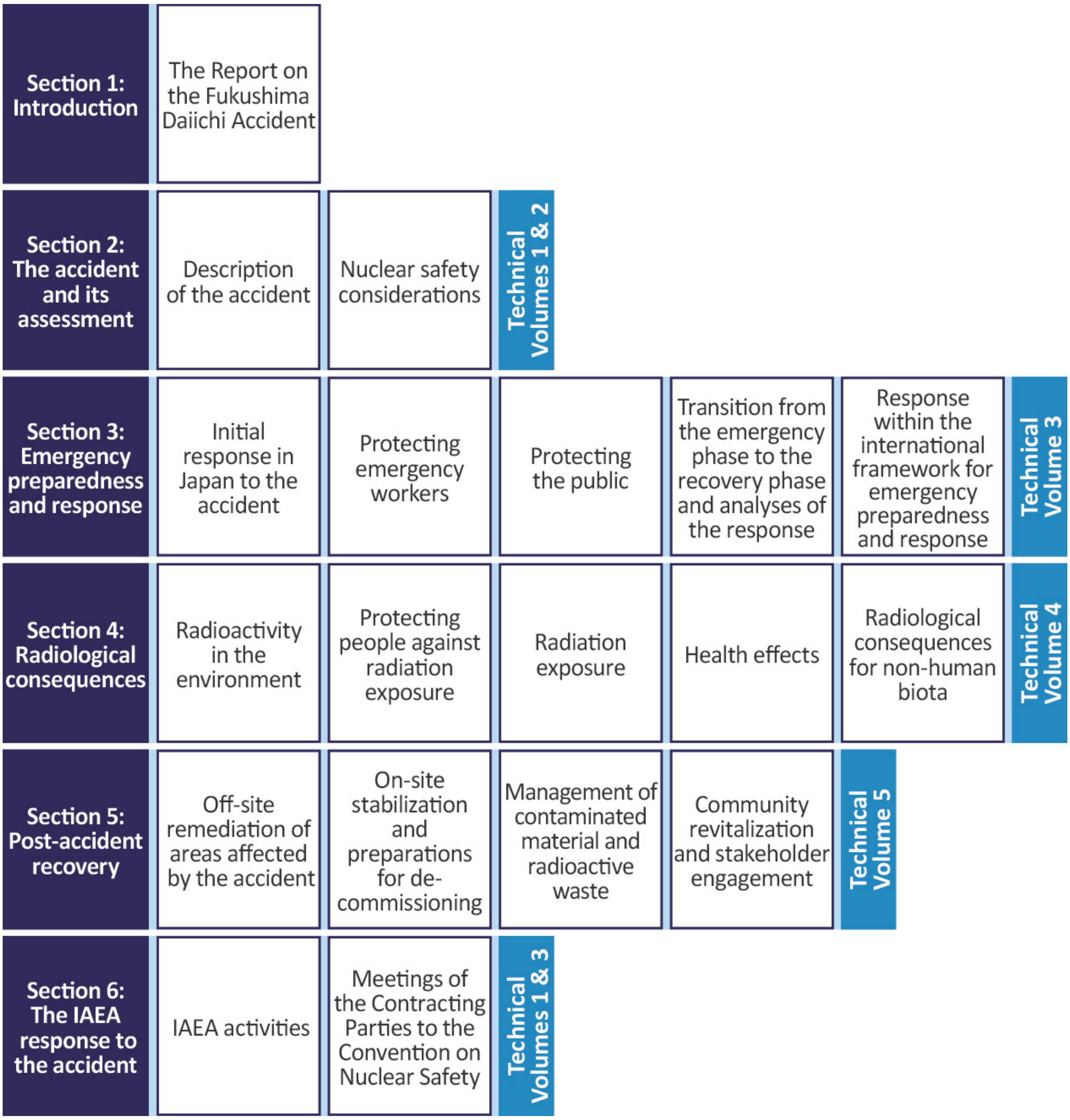

FIG. 2. Structure of the Summary Report and its relationship to the content of the technical volumes.

CONTENTS

SAFETY ASSESSMENT

2. INTRODUCTION

Technical Volume 1 of this report has described what happened during the accident at the Fukushima Daiichi nuclear power plant (NPP). The accident exceeded the design basis of the Fukushima Daiichi units in several respects. It was a severe accident, it affected multiple units and it was an accident which left the operators with little indication of what was happening, rendering them unable to control the situation. This volume describes why the accident occurred the way it did. To do so, multiple assessments have been performed to answer the following questions:

- Why did the site suffer from an extended station blackout?
- Why was site staff unable to cool the reactors and maintain the containment function?

The methodology used in this assessment is based on the IAEA safety standards in force at the time of the accident. The IAEA Safety Standards Series provides a system of Fundamental Safety Principles, Safety Requirements and Safety Guides. As the primary publication in the IAEA Safety Standards Series, the Fundamental Safety Principles No. SF-1 [1] establishes the basic safety objectives and principles of protection and safety for ensuring the protection of the public and the environment, now and in the future, from the harmful effects of ionizing radiation. Safety Requirements publications establish the requirements that must be met to ensure the protection of people and the environment, both now and in the future, in accordance with the objective and principles of the Safety Fundamentals. Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus on the measures recommended. The IAEA Statute makes the safety standards binding on the IAEA in relation to its own activities. They are not by default binding on Member States, but any State entering into an agreement with the IAEA concerning any form of IAEA assistance is required to comply with the requirements of the safety standards that pertain to the activities covered by the agreement. Many States elect to use IAEA safety standards as templates for their own legislation and regulations. Many other States, including Japan, use legislation and regulations adapted to their own situation and traditions.

All nuclear accidents result from a failure to maintain one or more of three fundamental safety functions, as described in IAEA Safety Standards Series No. NS-R-1, Safety of Nuclear Power Plants: Design, [2]:

- Control of reactivity;
- Removal of heat from the core;
- Confinement of radioactive material and control of operational discharges, as well as limitation of accidental releases.

The accident at the Three Mile Island NPP occurred owing to the loss of the second safety function, but releases were minimized because the containment successfully prevented any significant radioactive release to the environment. The Chernobyl accident occurred owing to the loss of the first safety function, and in this case no containment was available, resulting in the core being exposed to the environment and a very large release of radioactive material. The accident at the Fukushima Daiichi NPP occurred owing to the loss of the second and third safety functions as a result of an unanticipated severe external event — an earthquake followed by a resultant tsunami of extreme height. In this volume, the reasons behind the failure to maintain the second and third safety functions at the Fukushima Daiichi reactors are assessed in detail.

The volume begins (Section 2.1) with a review of how the design basis of the site for external events was assessed initially and then reassessed over the life of the NPP. The section also describes the physical changes that were made to the units as a result. The remainder of the volume describes the treatment of beyond design basis events in the safety assessment of the site, the accident management provisions, the effectiveness of regulatory programmes, human and organizational factors and the safety culture, and the role of operating experience. Further background information is contained in three annexes included on the CD-ROM of this Technical Volume which describe analytical investigations of the accident along with information on topics such as system performance, defence in depth and severe accident phenomena.

Section 2.2 provides an assessment of the systems that failed, resulting in a failure to maintain the fundamental safety functions in Units 1–3, which were in operation at the time of the tsunami and in which the reactor pressure vessels (RPV) and containment vessels failed. The section also describes Units 4-6, which were shut down at the time of the tsunami, and the site's central spent fuel storage facility.

Section 2.3 discusses the probabilistic and deterministic safety assessments of beyond design basis accidents (BDBAs) that had been performed for the plant and the insights from these assessments that had led to changes in the plant's design. The section pays particular attention to the assessment of extreme natural hazards, such as the one which led to the total loss of AC power supply on the site. The additional loss of DC power supply in Units 1 and 2 played a key role in the progression of the accident because it impeded the diagnosis of plant conditions and made the operators unaware of the status of safety systems.

Section 2.4 describes the accident management provisions and their implementation. All components of accident management are discussed, both preventive (before core melt) and mitigative (after core melt or severe accident). The section covers hardware provisions, emergency operating procedures, severe accident operating procedures, human resources and organizational arrangements, including training and drills. Interface with the off-site emergency arrangements is also discussed.

Section 2.5 deals with the governmental, legal and regulatory framework for nuclear safety in Japan up to the time of the Fukushima Daiichi accident. It evaluates this framework and its contribution to the accident, and identifies lessons learned.

Section 2.6 analyses the human and organizational aspects of the accident. It examines the main stakeholders of nuclear safety in Japan and shows how their actions were interrelated and interconnected, thereby reinforcing basic assumptions about nuclear safety that prevented them from adequately preparing for such an accident. The section analyses why the accident happened despite advancements in nuclear safety in areas such as solid design, peer reviews, regulatory frameworks, safety assessment methodologies, years of successful operating experience, defence in depth, emergency preparedness, severe accident management guidelines (SAMGs) and a strong international commitment to nuclear safety.

Finally, Section 2.7 addresses the role of operating experience in improving plant design and operation in order to continuously improve nuclear safety and support defence in depth. The section assesses the TEPCO operating experience programme and the extent to which lessons were learned from events both in Japan and internationally, and the design changes made.

2.1. ASSESSMENT OF THE PLANT IN RELATION TO EXTERNAL EVENTS

2.1.1. Site characteristics: Reassessment of the design bases for the Fukushima site and selection of the main plant grade level

2.1.1.1. Reassessment of the design bases of the site

The Fukushima Daiichi NPP site was selected for the location of six reactor units at the beginning of the 1960s. The site permissions were issued for Unit 1 in 1966, Unit 2 in 1968, Unit 3 in 1970, Unit 5 in 1971 and for Units 4 and 6 in 1972. With the exception of Unit 1, which was connected to the grid in November 1970, all of the other units were constructed and put into operation during the 1970s.

The process of site selection and site evaluation followed the practice and regulations at the time. Detailed information on this aspect as well as on the site characteristics is given in Technical Volume 1 in accordance with the information obtained from the Establishment Permit document [3, 4] drawn up for all six units. It can be considered as being equivalent to the section on site characteristics in a safety analysis report (SAR) for NPPs, as outlined, for example, in IAEA Safety Standards Series No. GS-G-4.1, Format and Content of the Safety Analysis Report for Nuclear Power Plants [5]. However, attention should be paid to the age of the data provided in the Establishment Permit, checking, in particular, that they have been updated regularly, as needed (see, for example, Ref. [4]). In this case, the population data originate from the end of the 1960s, the meteorological data are more recent, and the data and studies on seismic matters for some structures, systems and components (SSCs) date from the beginning of the 2000s.

During the operational life of the units, a comprehensive reassessment of the site characteristics and all its aspects was not required by the regulatory authority, regardless of evidence on new hazards or new hazard levels, updated regulatory requirements on periodic safety reviews or the availability of new scientific and technical findings or experiences [6]. However, the Tokyo Electric Power Company (TEPCO) performed the reassessment of specific subjects related to seismic and tsunami hazards, as described later in this volume.

Before March 2011, safety reviews in Japan resulting from a regulatory request were mainly conducted in relation to specific hazards or issues — called ‘backchecking’ in Japanese practice — rather than as a comprehensive reassessment and implementation of physical plant upgrading or ‘backfitting’.[1] Thus, the backchecking of seismic hazards had been requested since 2006, while tsunami hazards were not given the same priority. This was confirmed by the newly established Nuclear Regulation Authority (NRA) during the IAEA Consultancy Meeting on Regulatory Activities and Operating Experience held in Tokyo from 20 to 24 January 2014. The implementation of physical upgrades to enhance safety were carried out only for specific aspects, with a grace period of five years being granted for executing the relevant plant upgrading [7]. This situation changed after March 2011 and the current regulatory system requires the mandatory implementation of safety backfits.

Thus, prior to the Fukushima Daiichi NPP accident, no formal legal or regulatory requirements existed in Japan that required the comprehensive reassessment of the original site related design basis and site characteristics, either periodically or in response to new knowledge that might have been gained. This situation precluded periodic safety reassessment of the full range of external hazards that may affect plant safety under new conditions. On the other hand, since the beginning of the 1990s, periodic safety reviews have been an established practice in several Member States, where comprehensive

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[1] Assessment is a process aimed at providing information that forms the basis of a decision on whether or not something is satisfactory. Reassessment is a new assessment, performed after the original assessment and triggered by any of the reasons described above. The final decision on whether or not something is satisfactory may lead to the launch of a plan for determining effective corrective measures, including implementation of physical plant upgrading.

reassessments of site related aspects have been implemented on a regular basis. Furthermore, relevant guidance has been provided in IAEA safety standards [8], including technical and scientific support services. In this regard, the IAEA site and design Safety Requirements and related Safety Guides have been updated to take account of developments in the methodology, new requirements from regulatory authorities and new data. Such safety review and reassessment programmes must include consideration of the site related aspects of existing operating nuclear installations in a systematic and comprehensive manner. All topics and potential hazards are to be considered in this process, and the subsequent execution of safety upgrades has to enhance significantly the safety of those installations. A comprehensive reassessment of site related aspects would generally include the following aspects:

- Geological and geotechnical hazards (e.g. surface faulting, soil liquefaction, cavities);
- Earthquake hazards;
- Volcanic hazards, which are of particular importance in Japan due to its high volcanic risk;
- Hydrological hazards, such as external flooding hazards, from several potential sources (e.g. tsunamis, storm surges, downbursts with heavy precipitation);
- Extreme and rare meteorological phenomena (e.g. hurricanes, cyclones, typhoons, tornadoes);
- Human induced events of accidental origin (e.g. aircraft crash, explosion pressure waves).

Many of the safety reassessment programmes of existing nuclear installations have had the support and guidance of the IAEA and have resulted in modern and updated safety standards [8]. An important example of such measures to assess and enhance safety is the re-evaluation of site related safety aspects (mainly seismic) that was carried out for most of the water cooled water moderated energy reactor (WWER) type reactors in Eastern European countries mainly during the 1990s.

Before the accident, due to the lack of a regulatory framework for updating and upgrading plant safety in relation to the characteristics of the site, the Fukushima Daiichi NPP’s site related characteristics were not reassessed in a systematic and comprehensive manner. Such a reassessment would have considered all site related aspects and external events (i.e. seismic and geological, meteorological and hydrological, volcanic, and human induced hazards) as well as environmental issues. Regarding the specific regulatory framework for assessing tsunami hazards, at the time of the issuance of the Establishment Permit and during the entire operational period, the regulatory authorities did not issue any requirements or guidance concerning the need to reassess the tsunami hazards and, correspondingly, the plant safety with regard to these extreme natural events. The new guidance developed and released by the Nuclear Safety Commission (NSC) of Japan in 2006 [9] as part of the seismic safety guidelines does not contain any requirements, criteria or methodology that could be used in the reassessment of tsunami hazards in view of new and updated methodologies, data, evidence or concerns, except for a generic statement on the need to take into account concurrent events, in addition to the seismic hazards.

The situation described above led to an underestimation of the tsunami hazards that affected the site and the plant. It also resulted in a lack of appropriate measures to cope with higher than design basis tsunami hazards. It was characterized, in addition, by the absence of systematic reviews, as well as by the lack of interim measures before confirmation of emerging information from trial calculations.

2.1.1.2. Selection of the main grade level of the plant units

The decision on the finished top level of the ground at the location of the main plant buildings — at an elevation defined with respect to a reference level as, for example, the mean sea level — is an important aspect of the site that affects plant safety owing to its significance in relation to flooding hazards. This aspect refers to the adoption of the main plant grade level[2] at the time of the design and

[2] The main plant grade level is the top level of the ground, as finished, after plant construction.

construction of the plant with respect to the natural topographical situation prevailing on land and offshore.

The reference marker for all plant elevations in the region corresponds to the Onahama Port datum line (Onahama Port, or OP)[3], located about 50 km south of the Fukushima Daiichi NPP. TEPCO selected a main plant grade level of OP +10.00 m for the nuclear island and main buildings of Units 1–4 when the 'natural' ground level, i.e. the terrain topography, was approximately +30–35 m, well above such a level. For Units 5 and 6, this main plant grade level was defined as OP +13.00 m. The selection of such main plant grade levels implied the removal of a significant portion of natural soil/rock to reach that grade level.

Some general considerations about the main plant grade level are worth noting. Many factors are usually taken into consideration for a decision on that grade level, in view of the requirement of maintaining, throughout the operational life of the plant, a 'dry site' concept. The dry site concept implies that all items important to safety will be constructed above the level of the design basis flood, taking into account wind wave effects and any accompanying event(s) that may affect the reference level of the water at the time of the design basis flood (such as storm surge, sea level rise, tectonic movement, accumulation of debris, sediment transportation and ice). This can be accomplished by locating the plant at a sufficiently high elevation or, if necessary, by means of construction arrangements that raise the ground level at the site above the estimated maximum flood level. The site boundary should be monitored and maintained to keep such dry conditions during the operational life of the plant. In particular, if any filling is necessary to raise the plant above the level of the flood conditions for the design basis flood, this engineered plant item should be considered as an item important to safety and should therefore be adequately designed and maintained.

The dry site concept is considered a key measure against site flooding hazards that may affect safety. The original plant layout is to be defined on such bases, and it should be reassessed during the operational life of the plant to confirm these conditions. If the reassessment yields negative results, adequate protective measures and mitigation actions should be implemented in a timely manner.

If the above conditions are not met, the site is considered a 'wet site', where the level of the design basis flood is determined to be above the plant main grade level. Consequently, permanent site protective measures must be taken during the construction and operational stages, and, as mentioned above, these engineered plant protective measures are to be considered as items important to safety and should therefore be appropriately designed and maintained. Thus, the factors that are usually taken into account for the decision on the main plant grade level include the following:

- Specific plant design aspects (e.g. the plant and building layout, particularly the layout of the reactor building and reactor room level);
- Maximum and minimum levels of the estimated flooding caused by storm surges, rare meteorological phenomena (e.g. typhoons, hurricanes) or tsunamis (generated by, for example, earthquakes, landslides or volcanoes), i.e. as a result of the evaluation of all hydrological hazards;
- Geotechnical aspects related to foundation soil properties, the competent soil layers for the foundation of the buildings and, accordingly, the type (shallow or deep) of the selected foundation system;
- Hydrogeological aspects related to the presence of the groundwater table (aquifers), with its influence during the construction and operation stages;
- Construction methods and costs for digging, excavation and backfill, as well as considerations regarding the transport and assembly of heavy components;

[3] Onahama Port datum lines are 0.727 m below Tokyo Bay standard mean sea level as indicated in section 2.2.2 of the Establishment Permit [3].

— Operational aspects, e.g. those related to pumping water in and out to the main condensers, with its impact on in-house energy consumption;
— Embedment effects of the main buildings in the site response to seismic loads and the soil–structure interaction effects in the dynamic response of the SSC.

In the civil construction of the plant, TEPCO considered these factors, as indicated in the TEPCO Fukushima Nuclear Accident Analysis Report [10], through interviews with former employees who were involved in this work [10]. Although the topography provided a natural ground level of OP +30–35 m, TEPCO decided to build the plant at OP +10.00 m, i.e. much closer to the sea water level. This decision entailed the removal of soil and rock layers with a thickness of more than 20 m over a large area.

Figure 2.1–1 shows a cross-section of the construction site, including soil characteristics, based on the figure included in the Establishment Permit [3], to illustrate this discussion.

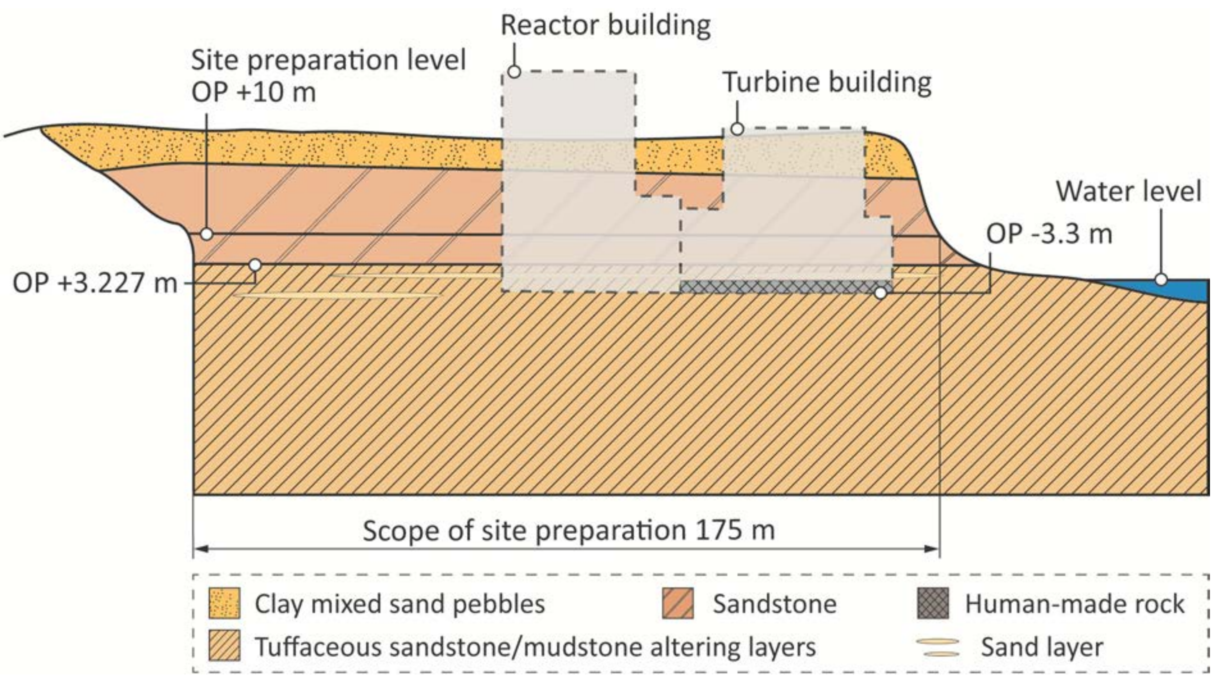


FIG. 2.1–1. Cross-section of the Fukushima Daiichi construction site showing soil characteristics (all levels are above mean sea level) [3].

Figure 2.1–2 shows a typical cross-section of the main buildings of Unit 1, illustrating the location of the grade level at OP +10.00 m and the relationship with the layout of the buildings.

The TEPCO report on the accident [10] indicated that the main plant grade level was defined in accordance with the information available from the historical records on tsunamis in the Fukushima Daiichi region and the topographical and bathymetric conditions of the shoreline in the area. In terms of construction costs, it would have been preferable not to excavate to a low level, though on the other hand, the lower level provided access advantages for water intakes and loading wharfs as well as significantly lower installation (i.e. required pump power) and operating costs for pumping cooling water. The report [10] also listed economic factors. These included the total excavation cost to develop the power station site area, including road access and the work area required, as well as the need to remove the clay and sandstone upper layers to reach the stable strata to obtain firm foundation soil where the major buildings were founded.

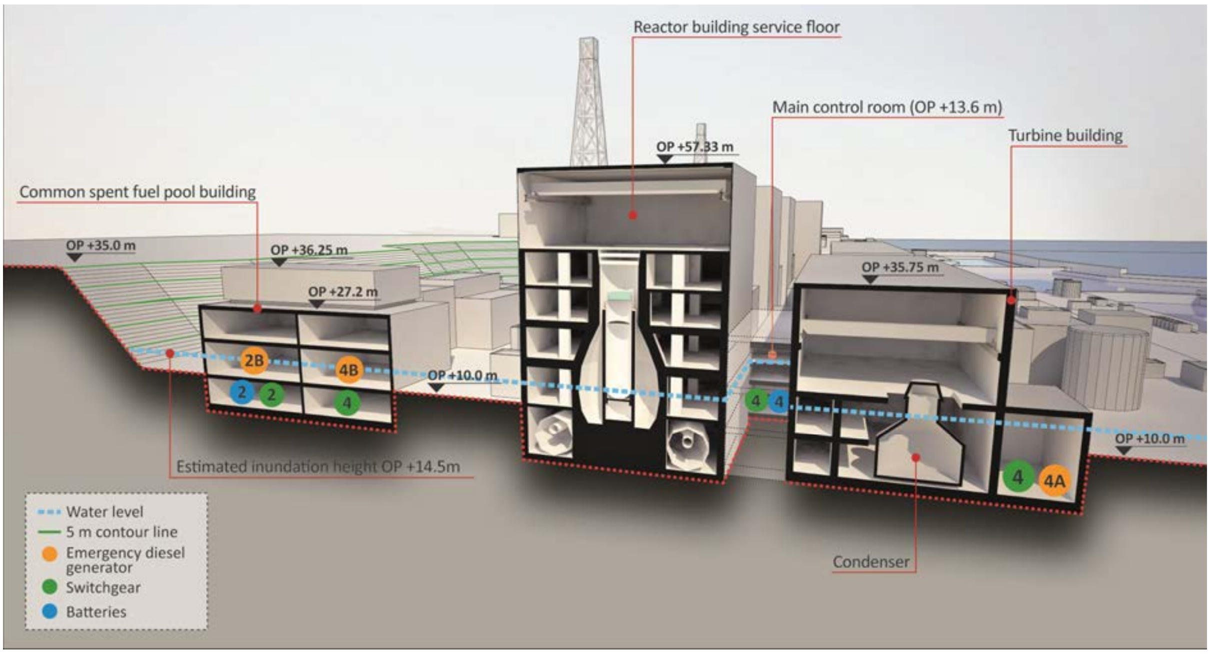


FIG. 2.1–2. Cross-section of the main buildings at Unit 1 of the Fukushima Daiichi NPP [3].

Thus, the main reason for the decision on the main plant grade level would have been the economics of water cooling supply (i.e. installation cost during the construction stage and transport energy cost during the operational life of the installation) based on the assumption that the external flooding levels would not impose a risk according to recent historical records in this area [10].

These plant grade elevations, particularly those at OP +10.00 m and +4.00 m, played a key role in the runup to the flooding from the tsunami in March 2011. Thus, a site that was considered as a dry site in the original design basis became a wet site during its operational period as result of the analyses of the flood levels performed before March 2011. These analyses were under review in March 2011.

2.1.2. International safety standards in relation to earthquake and tsunami hazard assessments and design aspects

2.1.2.1. Earthquakes: Hazards and design considerations

The IAEA has issued a number of publications focusing on earthquake and flood hazards since the 1970s through its Nuclear Safety Standards (NUSS) programme. These publications include Safety Requirements and Safety Guides, which were developed on the basis of consensus among Member States and recognized engineering practices. The safety standards in these areas, of generic and specific nature, have been periodically updated in accordance with scientific and practice developments, and considering also the lessons learned from occurred external events in the world.

IAEA safety standards require that, before the construction of an 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. Adequate design bases are required to be established to provide sufficient safety margins over the life of the NPP. These margins need to be sufficiently large to address the high level of uncertainty associated with the evaluation of external events. Site related hazards 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.

The guidance provided in the IAEA safety standards on seismic hazards (fault displacement and ground motion hazard) has been revised three times.

The first Safety Guide on this topic was published in 1979 as Safety Series No. 50-SG-S1, Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting [11]. At that time, the concept of capable fault was first introduced to differentiate those active faults which may have the potential for relative displacement at or near the ground surface.

In the 1960s and 1970s, it was common international practice to use historical records when applying methods for estimating seismic and concomitant (e.g. tsunami) hazards. This common practice included increasing safety margins by increasing the maximum recorded historical seismic intensity or magnitude in the site region, and assuming that such an event would occur at the closest distance to the site [11]. 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 potential values might not be attained in a relatively short period of observation when, typically, the observation period needs to include pre-historical data in order to provide robust estimates for the hazard assessment. However, the seismic hazard assessment for the design of Units 1 and 2 at the Fukushima Daiichi NPP was conducted mainly on the basis of regional historical seismic data without increasing the safety margins as described above. 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 [3, 4]. At that time, the regulations on deterministic seismic hazard assessment in many countries did not require a given frequency of occurrence for this extreme event. However, later, in the 1990s, studies demonstrated that the mean period of recurrence of those design bases spanned from 1000 to 100 000 years, with the median being 10 000 years [12]. These studies were performed in support of the revision of siting regulations in some countries based on a comparison of the original design basis for earthquakes with the available probabilistic hazard results.

The first version of the Safety Guide on seismic hazard assessment [11] was substantially revised in 1991 and published as Safety Series No. 50-SG-S1, Rev. 1 [13]. At this time the concept of seismic hazard analysis based on a seismotectonic model using a reliable geological, geophysical, geotechnical and seismological data set was established. This meant that the seismic hazard at an NPP site would be controlled not only by seismicity (i.e. the historical record of occurred earthquakes), but also by tectonics, which gives an indication of the long term potential for geological structures to generate earthquakes. This concept has been retained in the subsequent revisions of this Safety Guide, first in 2002 [14] and, finally, in 2010 with the current Specific Safety Guide, IAEA Safety Standards Series No. SSG-9, Seismic Hazards in Site Evaluation for Nuclear Installations [15].

In the Safety Requirements publication IAEA Safety Standards Series No. NS-R-3, Site Evaluation for Nuclear Installations [16], a preference was indicated for probabilistic hazard analyses for external events in order to take into consideration the potential for hazards beyond the design basis and, as a consequence, the need to avoid cliff edge effects affecting safety. Following this requirement, IAEA Safety Standards Series No. SSG-9 [15] provides detailed recommendations for a probabilistic seismic hazard analysis (PSHA).

Although the probabilistic approach is recognized as an efficient tool to evaluate hazards beyond the design basis, there are examples of deterministic approaches for beyond design basis seismic evaluation. The European Utility Requirements (1998) provide a factor of 1.4 beyond design for checking the adequacy of the plant, while the United States Nuclear Regulatory Commission (NRC) requires a plant to demonstrate a high confidence of low probability of failure (HCLPF) of 1.67.

2.1.2.2. Tsunami: Hazards and design considerations

Tsunami waves and associated phenomena may produce severe damage to installations located in coastal areas. With regard to nuclear installations, IAEA safety standards require that the potential for tsunamis that can affect safety and the determination of its characteristics should be assessed, taking into consideration pre-historical and historical data as well as associated hazards, with account taken of any amplification due to the coastal configuration at the site (see paras 3.24–3.28 of Ref. [16]).

If such a potential exists and detailed hazard characterization is done, the facility, installation or plant should be designed to withstand the event according to adequate design bases, including specific performance criteria determined as a result of a tsunami hazard assessment, as indicated in IAEA Safety Standards Series No. NS-R-1, Safety of Nuclear Power Plants: Design [2] (see paras 5.16 and 5.17). These aspects have also been considered in the current revision of NS-R-1, IAEA Safety Standards Series No. SSR-2/1 [17], in Requirement 17.

To comply with these requirements, IAEA Safety Standards Series No. NS-G-3.5, Flood Hazard for Nuclear Power Plants on Coastal and River Sites (2003) [18], and IAEA Safety Standards Series No. NS-G-1.5, External Events Excluding Earthquakes in the Design of Nuclear Power Plants [19], provided detailed recommendations based on recognized practice and consensus among Member States at that time. Characterization of runup, drawdown and associated phenomena (i.e. hydrodynamic forces, debris and sedimentation) was recommended in NS-G-3.5 [18]. This version of the Safety Guide on flood hazards reflected the operating experience from the flooding affecting the Le Blayais NPP site in France in 1998.

As stated in the IAEA safety standards [18], the dry site concept needs to be applied (as defined in the previous section) particularly in relation to flood events. Thus, all items important to safety should be constructed above the level of the design basis flood, with account taken of wind wave effects and any accompanying event(s) that may affect the reference level of the water at the time of the design basis flood (such as storm surges, sea level rise, tectonic movement, accumulation of debris and ice). In many Member States, this concept is preferred to the alternative solution of permanent external protective barriers such as levees, sea walls and bulkheads, which require being classified as safety related SSCs with strict design and construction engineering, periodic inspections, maintenance and monitoring features, among other aspects important to safety. In both cases, redundant and conservative measures should be implemented owing to the intrinsic cliff edge characteristics involved in surmounting the protective barriers in place. These measures include ensuring waterproofing and the suitable design of items necessary to provide the capability to properly perform the fundamental safety functions of shutting down the reactor and maintaining it in safe shutdown condition [18].

A revision of NS-G-3.5 was recently undertaken to consider new data, information and lessons learned, mainly from the December 2004 Indian Ocean tsunami. The result was IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [20], which was published in 2012. This new version maintains the concepts and recommendations from the previous Safety Guides and provides more detailed recommendations related to the protection of NPPs against the hazardous effects of tsunamis. An annex in this safety standard — which is not considered as part of the safety standard — refers to current practices in some Member States, with Japan and the United States of America being the examples.

2.1.3. Japanese regulatory practices in relation to earthquake and tsunami hazards and design aspects

2.1.3.1. Earthquakes

At the time the construction permit for Units 1–6 of the Fukushima Daiichi NPP was issued, between 1966 and 1972, the applicable criteria in Japan for defining the site related design bases including the review guidance were, in general, those established in the:

— Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria, issued by the Japan Atomic Energy Commission (JAEC) in 1964 [21] (revised in 1989).
— Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities, issued by JAEC in 1970. This guidance document was revised in 1977 and 1990 [22] and provides very general requirements on the need for safety functions not to be affected by this type of natural hazard.

Regarding the seismic hazard assessment, a document entitled Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities was issued by the NSC in 2006 [9] [4] which was valid at the time of the Fukushima Daiichi accident.

Active and capable faults — Japan

Until very recently, the Japanese nuclear regulatory guides and Japanese practice did not make a clear distinction between active faults and capable faults as established by the IAEA. In other words, seismic hazard analyses involved only vibratory ground motion and not fault displacement. Starting with the 16 July 2007 Niigata-Chuetsu-Oki earthquake that affected the Kashiwazaki-Kariwa NPP, and following several IAEA missions that recommended that fault displacement hazard issues be specifically addressed, more attention was given to this topic by TEPCO. In fact, TEPCO agreed to conduct a research study, including performing specific and detailed site investigations in relation to the fault displacement issue at the Kashiwazaki-Kariwa site and to initiate probabilistic fault displacement hazard analysis. This was also identified as a task within the framework of the IAEA Extrabudgetary Programme on Seismic Safety. TEPCO presented their results during a seminar on fault displacement hazards in January 2014 at the IAEA, which was organized as part of the annual Donors' Meeting of the Extrabudgetary Programme of the IAEA's International Seismic Safety Centre (ISSC).

The NSC Regulatory Guidelines of 2006[9] increased the definition of an active fault from 50 000 to 120 000–130 000 years. The latter period refers to the late Pleistocene. This change was significant since the faults near the NPPs, which may have been previously considered not active during the original site selection and design process, had to be considered active after 2006 and the seismic hazard values, consequently, were increased. Even though the older faults were considered to be active by the NSC Regulatory Guidelines of 2006 [9], the Japanese regulatory authority at that time, the Nuclear and Industrial Safety Agency (NISA), did not have in place specific requirements or regulations regarding fault displacement issues and only required that the vibratory ground motion be re-evaluated.

The regulatory outlook on the fault displacement issue has changed significantly in Japan since the Fukushima Daiichi accident. Fault displacement hazard has now become a subject that is raised frequently by the current Nuclear Regulation Authority (NRA). However, even though regulatory requirements were not in place, NISA had already asked several utilities to investigate the fault displacement hazard at their NPPs. After the 2007 Niigata Chuetsu-Oki earthquake and the

[4] Hereafter referred to as the NSC Regulatory Guidelines of 2006

identification of the potential fault displacement issue at the Kashiwazaki-Kariwa site by the IAEA, NISA became more proactive in its oversight of the problem. The issue was formally raised by NISA only after the Fukushima Daiichi accident and in the context of the backcheck programme at the Expert Hearing on Earthquakes and Tsunamis [23].

NISA/NRA have asked for reviews of the following NPPs regarding the potential for fault displacement hazard at the site according to information provided by the official NRA web site:

- On 29 August 2012, NISA directed the Kansai Electric Power Company (KEPCO) to develop an additional plan on the fracture zones at the site of the Mihama NPP.
- In November 2012, the NRA conducted investigations and held evaluation meetings relating to the fracture zones at the site of KEPCO's Ohi NPP. Additional investigations are planned, including a feature called the F-6 fracture zone.
- NISA received additional investigation plans from the Japan Atomic Power Company (JAPC) in May and August 2012 regarding the fracture zones at the site of the Tsuruga NPP. The NRA conducted field investigations related to on-site capable faults at Tsuruga in December 2012.
- NISA directed the Hokuriku Electric Power Company to conduct additional investigations of the fracture zone at the site of the Shika NPP on 18 July 2012.
- NISA received a report on the evaluation results of the fracture zones at the site of the Higashidori NPP from Tohoku Electric Power Company in March 2012.

A new regulation prepared by the NRA in July 2013 defines fault activity. The document puts the new limit for fault capability to 400 000 years in cases where the late Pleistocene dating cannot be established due to the absence of these horizons [24]. This may have some major consequences for Japanese NPPs, as follows:

- Faults in the vicinity of the site, previously not identified as active (or 'capable' in IAEA terminology) under old regulations, may now become a seismogenic source which would be able to generate seismic ground motion and/or be a source for fault displacement hazards.
- In the case of seismic ground motion, there will yet be another increase in the vibratory ground motion hazard with respect to the level for which the SSCs were designed or previously re-evaluated.
- In the case of fault displacement hazards, there would now be a serious issue if they were found at or near an NPP site.

The M 6.5 default earthquake that most of the NPPs in Japan were designed against may have concealed some small faults (or folds) in the site vicinity (i.e. about a 5 km radius around the NPP site) as they contributed much less to the ground motion hazard than the M 6.5 default earthquake. Because fault displacement was not explicitly considered a separate hazard, this did not cause an issue before the new NRA regulations. It is likely that small capable faults may exist in the vicinity of some Japanese NPPs. Without resorting to probabilistic fault displacement hazard analysis, it is difficult to resolve issues related to these faults. The NRA will make deterministic decisions regarding these faults because Japanese regulations do not support probabilistic fault displacement hazard analysis.

Vibratory ground motion

The underestimation of the seismic hazards for the Fukushima Daiichi and Daini NPPs is shown in Tables 2.2–1 and 2.2–2, provided by TEPCO. At these sites, the original design basis as well as the values obtained from recently conducted seismic hazard reassessments were exceeded with respect to the maximum acceleration values recorded in March 2011.

The underestimation of the seismic hazard is related to the reliance of past Japanese practice on basing the seismic hazard assessment mainly on observed earthquakes (recent historical data) and not on the tectonic potential of the faults (including the subduction zone), coupled with the lack of conservative assumptions for taking into account the uncertainties that exist in the assessment of pre-historical extreme events. In this regard, consideration of an active fault survey for assessing vibratory ground motion has been strengthened after the issue of the NSC Regulatory Guidelines of 2006.

It is important to point out the difference in the Japanese approach with respect to the international practice. As indicated in the previous section, in the 1960s and 1970s, it was common international practice to use historical records when applying methods for estimating seismic and concomitant (e.g. tsunami) hazards. This approach was basically deterministic. The international practice [11] was to add a safety margin to supplement the lack of information on non-observed extreme events of very low annual frequency of occurrence by increasing the maximum historically recorded seismic intensity or magnitude and by assuming that such an event may occur at the closest distance to the site, as explained in detail in Safety Series No. 50-SG-S1 [11]. This was not done in Japan in order to compensate for the fact that the maximum values might not be attained in a relatively short period of observation. The pre-historical data are to be included in order to consider extreme events and provide robust estimates for the hazard assessment of very low annual probability of occurrence events. In addition to the criterion to use pre-historical and historical data commensurate with the low annual frequency of occurrence of these extreme external events, the internationally recognized practice also recommended the use of global analogues in order to cope with the lack of such pre-historical data. This is another important tool, particularly as an earthquake with M 9.5 (the largest in history) had occurred previously in the same tectonic environment of the Pacific tectonic plate.

When the IAEA Fact Finding Expert Mission of 24 May–2 June 2011 [6] said in its report that the "tsunami hazard was underestimated", it meant that the magnitude associated with the subduction zone was underestimated because of the emphasis on the use of historical data only (see the detailed description in Section 2.1.4.2).

At the beginning of the 1960s, when the first unit of Fukushima Daiichi was designed, there were two major subduction earthquakes (in Chile in 1960 at M 9.5 and in Alaska in 1964 at M 9.2) in the Circum-Pacific Belt, on which Japan is also located. At that time, an integrated approach to large scale seismotectonic modelling was not adopted in Japan. The tectonic potential of faults was not considered because at that time the earthquake occurrence model relied heavily on seismicity data and a belief that a mega earthquake like the M 9.5 earthquake in Chile in 1960 would not occur in this area which proved to be incorrect.

Due to the conservative approach of the Japanese regulatory guides for seismic design of NPPs — e.g. using the static approach with three times the static equivalent seismic loads established for conventional building codes — the site specific seismic hazard may not be the actual design basis. The difference between S1 and S2 earthquake levels[5], as they were defined by the Japanese regulations, is related to the time frame for which the faults that may generate significant earthquakes are considered to be active. Until the publication of the NSC Regulatory Guidelines of 2006 [9] for calculating S1, it was required to postulate that faults that had moved within the past 10 000 years were to be considered as being active. For the calculation of S2 this time frame was 50 000 years. This meant that many more faults had to be considered active in the calculation of S2. This increased the estimated value of the seismic ground motions hazard.

[5] S1 and S2 are the two levels of severity of design basis ground motions that should be taken into account. IAEA Safety Series No. 50-SG-S2 defined the application of these two levels in design as follows: (1) ground motion level 1 (S1), which is the maximum that reasonably can be expected to be experienced at the site area once during the operating life of the nuclear power plant; (2) ground motion level 2 (S2), which is considered to be the maximum earthquake potential at the site area [11, 25].

With the publication of the NSC Regulatory Guidelines in September 2006 [9], the 50 000 year period was increased to 120 000–130 000 years, as was described in the previous section. This meant that every NPP in Japan was required to perform a backcheck in order to understand the impact of the additional faults that needed to be considered to be active for assessing the vibratory ground motion.

Since September 2006, the derivation of the seismic design basis using the seismic hazard approach has stipulated the following approaches:

- Following a methodology similar to the deterministic seismic hazard analysis outlined in IAEA Safety Standards Series No. SSG-9 [15], using the ground motion prediction equations (GMPEs) approach for the identified seismic sources as formulated, for example, by the Japan Electric Association (JEA) Response Spectrum;
- Using the Green's Function approach for numerical simulation of the ground motion generated by the identified seismic sources;
- Considering diffuse seismicity, formulating the ground motion by collecting and analysing relevant strong motion records that cannot be assigned to an identified seismic source with account taken of site characteristics.

During the preparation of IAEA Safety Standards Series No. SSG-9 [15], the following paragraph was added in order to represent the Japanese approach regarding the second item in the list above:

> "5.14. In seismically active regions for which data from ground motion caused by identifiable faults are available in sufficient quantity and detail, simulation of the fault rupture as well as of the wave propagation path is another procedure that should be followed. In cases where nearby faults contribute significantly to the hazard, this procedure may be especially effective. The parameters needed include:
>
> (a) Fault geometry parameters (location, length, width, depth, dip, strike);
> (b) Macroparameters (seismic moment, average dislocation, rupture velocity, average stress drop);
> (c) Microparameters (rise time, dislocation, stress parameters for finite fault elements);
> (d) Crustal structure parameters, such as shear wave velocity, density and damping of wave propagation (i.e. the wave attenuation Q value).
>
> "For complex seismotectonic environments such as plate boundaries, thrust zones and subduction zones, and in particular for offshore areas, the specific seismotectonic setting of the earthquake that affects those seismic source parameters mentioned in (a)–(d) should be considered in the characterization of the ground motion."

Minimum magnitude (M 6.5) event

Historically, postulating an M 6.5 default earthquake at the site may have had favourable and unfavourable consequences. The favourable consequence is that a certain level of robustness is ensured for every NPP in Japan as a minimum. However, this assumption also had an unfavourable consequence because it led plant operators to ignore nearby faults if they produced lower ground motions that were less than the enveloping response spectrum due to the M 6.5 postulated earthquake. While this had no implication in the evaluation of the vibratory ground motion hazard, it led to the potential for the fault displacement hazard being ignored. As expressed in the previous subsection, this topic was recognized as a regulatory issue in Japan after the 2007 Niigata Chuetsu-Oki earthquake, at the Kashiwazaki-Kariwa site. It has been formally recognized as a regulatory issue in Japan since the Fukushima Daiichi NPP accident, and the corresponding reassessments of several Japanese NPPs to verify safety against earthquake related hazards are ongoing.

Ground motion prediction equations for the Niigata-Chuetsu-Oki and Great East Japan earthquakes

For approximately the same epicentral distance and distance from fault rupture (about 200 km), the base mat motions recorded at the two plants, Fukushima Daiichi and Daini (only 10 km apart), are significantly different (see Tables 2.1–1 and 2.1–2). The soil properties are similar (at ~50 m, a layer with shear wave velocity of V_s = 700 m/s). The plant structures are also similar, as is the embedment depth of ~10–12 m for all units.

The two earthquakes (Niigata-Chuetsu-Oki in 2007 and the Great East Japan in 2011) which were recorded in the basement levels of the reactor buildings of the seven units of the Kashiwazaki-Kariwa NPP and all the units of the Fukushima Daiichi and Daini NPPs show that when the location of the earthquake focus with respect to the NPP is of the same order of magnitude as the causative fault dimensions, the values obtained using the conventional GMPE and site response (considering only soil amplification) will not be accurate in terms of predicting ground motion.

The variability between the records obtained at Units 1–4 and Units 5–7 of the Kashiwazaki-Kariwa NPP during the Niigata-Chuetsu-Oki earthquake is significant and cannot be predicted using conventional tools. As shown in Tables 2.1–1 and 2.1–2, the same is true for the records obtained at the Fukushima Daiichi and Daini units during the Great East Japan Earthquake. This may explain the emphasis given to the fault rupture simulation approach by Japanese scientists.

2.1.3.2. Tsunamis and external flooding

At the time the construction permit for the Fukushima Daiichi NPP was issued, from 1966 to 1972 for Units 1–6, the applicable criteria in Japan for defining the site related design bases including the review guidance, were those established in the:

- Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and Application Criteria, issued by the JAEC in 1964, (revised in 1989) [21].
- Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities, issued by JAEC in 1970. This guidance document was revised in 1977 and 1990 and provides very general requirements on the need that safety functions must not be affected by natural hazards [22].

Specifically regarding the tsunami hazard assessment, the 1990 version of the Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities [22] provided very generic requirements and only stated that the effect of the tsunami should be considered in the design, but it did not prescribe an approach or methodologies to be used or performance criteria to be fulfilled in that regard. It also stated that the design tsunami should be determined, for example, by numerical simulation.

Later, in 2002, reflecting the advances and progress in the area of numerical simulations for tsunami hazard evaluation, a standard assessment methodology was developed by the Japan Society of Civil Engineers (JSCE) through its Tsunami Evaluation Subcommittee of the Nuclear Civil Engineering Committee. Thus, the Tsunami Assessment Method for Nuclear Power Plants in Japan [26] was published and its main elements are described in Section 2.1.5.4. It is based on a deterministic approach, and the uncertainties in the tsunamigenic source data are considered by processing a number of studies with a variation of the significantly involved parameters.

However, the most important characteristic of the JSCE methodology, from the point of view of its incidence in the calculations performed before March 2011 by TEPCO and which led to an underestimation of the tsunami wave heights at Fukushima Daiichi, is the fact that the tsunamigenic sources are fixed or deterministically established by the guidelines, and the applicant should apply only those sources as indicated by the procedure. It should also be highlighted that this standard provides a method for calculating the maximum and minimum tsunami heights, but does not include specific guidance on how to deal with the associated effects such as, for example, hydrodynamical loads, sand drift, or missiles from transported debris (see chapter 2, point 2 of Ref. [26]). Moreover, the JSCE methodology indicates that "it is assumed that the effects of the other phenomena are less important than that of the water level". These associated effects are only referred to as future challenges in the last sentence of the final chapter.

The JSCE methodology had been applied by TEPCO and all other utilities in Japan since 2002, as described in detail in Section 2.1.5.4 of this volume. It is also included in an annex of IAEA Safety Standards Series No. SSG-18 [20] as an example of practice of a Member State (Japan). Practice in the USA is also included in that annex.

In relation to regulatory requirements and guidance, the new NSC Regulatory Guidelines of 2006 address tsunami safety in chapter 8 as an accompanying event of an earthquake, stating that "safety functions of Facilities shall not be significantly affected by the tsunami which could be postulated appropriately to attack but very scarcely in the operational period of Facilities". No detailed or specific requirements or guidance is provided on the way to comply with this statement, in particular regarding the meaning of "very scarcely" [9].

Regarding the combination with other flooding hazards, it was indicated that in practice only the high tide is added to the calculated tsunami water levels. In the meeting with NSC authorities at the time of the IAEA Fact Finding Expert Mission, it was clarified that the NSC Regulatory Guidelines of 2006 were not legally binding and they were not regulations, although in practice they were considered as such. It was also expressed that these guidelines would be revised in view of recent experience and lessons learned [6].

No regulatory guidelines existed, also, regarding the consideration of external flooding as a hazard that may result from a combination of meteorological and hydrological phenomena.

2.1.4. Design basis and reassessments of the earthquake hazards and remedial actions taken during the operational life of the Fukushima Daiichi NPP

2.1.4.1. Background information on seismic hazards

IAEA safety standards require that the site of a nuclear installation be adequately investigated with regard to all characteristics that could be significant to safety and possible external natural and human induced hazardous phenomena. IAEA Safety Standards Series No. NS-R-3, Site Evaluation for Nuclear Installations [16] states the key requirements to be complied with, as follows:

> "3.2. Information on prehistorical, historical and instrumentally recorded earthquakes in the region shall be collected and documented.
>
> "3.3. The hazards associated with earthquakes shall be determined by means of seismotectonic evaluation of the region with the use to the greatest possible extent of the information collected.
>
> "3.4. Hazards due to earthquake induced ground motion shall be assessed for the site with account taken of the seismotectonic characteristics of the region and specific site conditions. A thorough uncertainty analysis shall be performed as part of the evaluation of seismic hazards.

“3.5. The potential for surface faulting (i.e. the fault capability) shall be assessed for the site. The methods to be used and the investigations to be made shall be sufficiently detailed that a reasonable decision can be reached using the definition of fault capability given in para. 3.6.”

The IAEA Safety Requirements mentioned above, which updated similar safety requirements established in Safety Series No. 50-C-S (Rev. 1), Code on the Safety of Nuclear Power Plants: Siting [27], published in 1998, are supported by the detailed recommendations provided in the newly revised IAEA Safety Standards Series No. SSG-9 [15], in which methodologies and criteria for assessing seismic hazards and, particularly, seismic ground motions and the potential for fault capability are provided.

As indicated in IAEA Safety Standards Series No. SSG-9 [15], the general approach to seismic hazard evaluation should be directed towards reducing uncertainties at various stages of the evaluation process in order to obtain reliable results driven by data. Experience shows that the most effective way of achieving this is to collect a sufficient amount of reliable and relevant data. There is generally a trade-off between the time and effort necessary to compile a detailed, reliable and relevant database and the degree of uncertainty that the analyst should take into consideration at each step of the process. All pre-instrumental data on historical earthquakes (that is, events for which no instrumental recording was possible), extending as far back in time as possible, should be collected. Palaeoseismic and archaeological information on historical and pre-historical earthquakes should also be taken into account.

Since its first revision in 1991, Safety Series No. 50-SG-S1 (Rev. 1), Earthquakes and Associated Topics in Relation to Nuclear Power Plant Siting [13, 27] recommends the application of a seismotectonic approach which is established on the basis of a database comprising geological, geophysical, geotechnical and seismological information. The integration of the geological data (physical capability of tectonic structures to generate earthquakes) with historical and pre-historical seismicity (empirical data) is the cornerstone of the seismotectonic approach regardless of the method used for the calculation of the seismic hazard specific to the site (deterministic or probabilistic).

After detailed hazard characterization is done, the plant should be designed to withstand the seismic events according to specific design bases determined as a result of this hazard assessment, as indicated in paras 5.16 and 5.17 of IAEA Safety Standards Series No. NS-R-1, Safety of Nuclear Power Plants: Design [2]. Moreover, para. 5.22 of NS-R-1 [2] states that: “The seismic design of the plant shall provide for a sufficient safety margin to protect against seismic events.”

To comply with such design requirements, mainly applicable to the design of new installations, IAEA Safety Standards Series No. NS-G-1.6, Seismic Design and Qualification for Nuclear Power Plants [8], provides detailed recommendations for the design of SSCs according to their safety significance and following recognized international engineering practice and consensus at the time.

Finally, and more applicable to the situation at the Fukushima Daiichi NPP, for existing operating power plants the evaluation of seismic safety should be conducted as required by the following factors, as prescribed by para. 2.10 of IAEA Safety Standards Series No. NS-G-2.13, Seismic Safety Evaluation for Existing Installations [28]:

“(a) Evidence of a seismic hazard at the site that is greater than the design basis earthquake arising from new or additional data (e.g. newly discovered seismogenic structures, newly installed seismological networks or new palaeoseismological evidence), new methods of seismic hazard assessment, and/or the occurrence of actual earthquakes that affect the installation;
(b) Regulatory requirements, such as the requirement for periodic safety reviews, that take into account the ‘state of knowledge’ and the actual condition of the installation;

(c) Inadequate seismic design, generally due to the vintage of the facility;
(d) New technical findings, such as vulnerability of selected structures and/or non-structural elements (e.g. masonry walls), and/or of systems or components (e.g. relays);
(e) New experience from the occurrence of actual earthquakes (e.g. better recorded ground motion data and the observed performance of SSCs);
(f) The need to address the performance of the installation for beyond design basis earthquake ground motions in order to provide confidence that there is no 'cliff edge effect'; that is, to demonstrate that no significant failures would occur in the installation if an earthquake were to occur that was slightly greater than the design basis earthquake...;
(g) A programme of long term operation of which such an evaluation is a part."

2.1.4.2. Design basis in relation to earthquake hazards

The following is based on the Establishment Permit Report of the Fukushima Daiichi NPP [3] and Nuclear Reactor Establishment Change Permit Application (Nuclear Industry Report to the Government No. 5–11) submitted on 13 April 1993 [4]. The site description parts of the original Establishment Permit Report [3] have been included in Technical Volume 1. Those parts that relate directly to the derivation of the seismic design basis are treated in this section.

The contents of the section on earthquakes in Ref. [4] are as follows:

- Seismic motion earthquakes and earthquake damage that have occurred in Fukushima Prefecture and surrounding areas;
- History of earthquake damage in areas near the Fukushima Daiichi site;
- Ground conditions of the Fukushima Daiichi site;
- Seismic past earthquakes;
- Recent seismic activity;
- Active faults;
- Seismic geotectonics;
- Site ground vibration characteristics;
- Standard seismic motion;
- Reference documents.

Earthquakes and earthquake damage that has occurred in Fukushima Prefecture and surrounding areas

As indicated in Refs [3] and [4], records of earthquake damage in Japan date back as far as A.D. 599 and have been used to create chronological tables which provide information on the magnitude, hypocentre and degree of damage from each earthquake. Earthquakes around Fukushima Prefecture can be classified into two groups, those with hypocentres in the ocean off the coast of Iwaki and Sanriku, and those occurring inland around Lake Inawashiro.

According to isolines that indicate the number of years between recurrences in the region around Fukushima Prefecture, earthquakes with intensities greater than severe earthquakes and those greater than disastrous earthquakes occur only once every approximately 150 years and once every approximately 400 years, respectively. Furthermore, a very severe earthquake is not known to have occurred in the vicinity of Fukushima Prefecture, making it one of the least earthquake prone areas in Japan.

According to historical research, other than Aizuwakamatsu, in the vicinity of Lake Inawashiro, no other areas in the vicinity of Fukushima Prefecture have suffered damage in the past, and areas around the site of the Fukushima Daiichi and Daini NPPs have never been damaged by an earthquake.

History of earthquake damage in areas near the Fukushima Daiichi NPP

Of all the earthquakes that have occurred in Fukushima Prefecture and its surrounding areas, those that are thought to have caused some damage near the site of the Fukushima Daiichi NPP and for which records of damage still exist have been extracted as follows:

- Aizu earthquake in 1611 (27 September 1611) M 6.9;
- Sendai earthquake in 1646 (9 June 1646), M 7.6;
- Iwashiro-no-kuni Koori earthquake in 1731 (7 October 1731) M 6.6;
- Earthquake off the coast of Shioyazaki to the southeast in 1938 (23 May 1938) with M 7.5 and intensity V in Onahama, Fukushima, Aizu;
- Earthquake off the coast of Fukushima Prefecture to the east in 1938 (5 November 1938), with M 6.5 and intensity V in Onahama, Fukushima, Aizu.

Reference [3] provides a description of the effects of the above mentioned earthquakes and concludes that the area in the vicinity of the Fukushima Daiichi NPP site is a region of Fukushima Prefecture with little seismic activity.

One important point to note is related to the offshore earthquakes that occurred on the subduction zone to the east, the Sendai earthquake in 1646 to the north-east and the earthquake off the coast of Shioyazaki in 1938 to the south-east of the site have been estimated at M 7.6 and M 7.5, respectively. They are both within a distance of 70 km from the Fukushima Daiichi NPP site.

The basic data that have been used in checking the seismic hazard of the site are the historical seismicity starting with an earthquake from the year 1611. This means that the data are constrained to about 400 years.

Ground conditions at the Fukushima Daiichi NPP site

As mentioned in Section 2.1.1, the construction site of the power plant was prepared to an elevation of approximately OP +10.00 m. The main nuclear island structures, such as the reactor buildings of the six units, were built directly on mudstone bedrock that lies near an elevation of OP −4.00 m. This mudstone bedrock belongs to the Sendai Group from the Pliocene Epoch of the Neogene Period and is approximately 400 m thick.

On the basis of the results of the bearing capacity tests and on the values obtained of the compression (P) and shear (S) soil wave velocities, the ground conditions were deemed to be good.

Original seismic design basis

Applying the above described methodology and data available at the time of the licensing of the Fukushima Daiichi NPP units — based more on historical seismicity than on tectonics — the original design basis values for the seismic hazard, expressed in terms of zero period ground acceleration (ZPGA), are quite modest, with values ranging from 245 Gals (~0.25g) to 294 Gals (~0.29g) for the north–south and east–west horizontal components, respectively, for Units 1–5. For Unit 6, higher values of ~0.50g were evaluated as the original design basis.

In addition, and in compliance with the seismic design criteria in Japanese practice, a static horizontal acceleration of 470 Gals (~0.47g) is used for designing the buildings and structures. ZPGA values, measured in Gals, for each unit are included in Tables 2.1–1 and 2.1–2 [6].

2.1.4.3. Reassessment of the earthquake hazards

Information on the evolution of the assessment of seismic hazards was obtained mainly from the chapter on Seismic Motion of the Nuclear Reactor Establishment Change Permit Application (Nuclear Industry Report to the Government No. 5–11) [3, 4].

New investigations were performed in order to obtain a change permit for the construction of structures as described below. The importance of these investigations is that they are based on much more recent earthquake catalogues as well as information on tectonics, i.e. active faults in the region of investigation. A new set of ground motion parameters were derived for these new structures. However, it is not clear whether or not the new parameters were applied to the existing structures, i.e. whether or not backcheck evaluations were performed.

The following explains the methodology used in the determination of the design basis seismic motion as it should be applied to "common facilities used to assist operation, as well as concrete ducts and internal equipment connected to these facilities" mentioned as part of changes made to the Nuclear Reactor Establishment Change Permit Application (Nuclear Industry Report to the Government No. 5–11) [3, 4]. Specifically, these include the establishment of the spent fuel pool and dry storage facility, the establishment of the spent fuel transport vessel storage area and specialization and extra installation of emergency diesel generators.

The change permit application states that the USAMI Catalog (1979), Utsu Catalog (1982) and Meteorological Agency Earthquake Catalog were the most reliable at that time. A figure provided in the change permit application shows the distribution of recent damaging earthquakes having epicentres within 200 km of the site from all of the damaging earthquakes mentioned in the USAMI Catalog and the Meteorological Agency Earthquake Catalog. With regard to the magnitude and epicentre locations of the earthquakes, the USAMI Catalog was used for earthquakes that occurred prior to 1884, the Utsu Catalog was used for earthquakes occurring between 1885 and 1980, and the Meteorological Agency Earthquake Catalog was used for earthquakes that occurred after 1981. This decision was likely made on the basis of the confidence that was placed in the three catalogues for the respective time periods mentioned above.

While there are some differences between the information available earlier and the new earthquake catalogues, the impact of the new seismicity information on the design does not seem to be very significant. Nevertheless, the following important conclusions were made after the review of the recent earthquakes from the Meteorological Agency Earthquake Catalog:

— According to the hypocentre distribution, earthquakes frequently occur around the site in ocean areas offshore of Miyagi, Fukushima and Ibaraki Prefectures.
— According to the vertical distribution, earthquakes that occur on the Pacific Ocean side are generated in conjunction with subduction of the Pacific plate, and the hypocentres get deeper as they near land.
— Earthquakes that occur in conjunction with the subduction of the Pacific plate as mentioned above occur at a depth of between 60 and 90 km near the site.

Active faults

The latest comprehensive survey on active faults at the time was Active Faults in Japan (New Edition) [29]. The report shows active faults within 100 km of the Fukushima Daiichi NPP site. According to this publication, the main faults having late Quaternary activity in areas around the site are the Futaba fault and the western marginal fault zone of the Fukushima Basin:

(1) Futaba fault:
The length of the Futaba fault (on land) is 18 km, from the vicinity of south-western Kayakura in Soma City to the vicinity of Ogai in Haramachi City. Its activity extends to the late Quaternary.

No connection was considered to have existed between this fault and observed past earthquakes or microearthquakes. As Active Faults in Japan (New Edition) [29] puts the activity level of this fault at Class B, earthquakes occurring on the Futaba fault (M 6.9, $\Delta = 35$ km) are considered to be extreme design basis earthquakes. This means that deterministically, a design basis earthquake of M 6.9 has been considered to have occurred at a distance of 35 km from the Fukushima Daiichi NPP.

(2) Western marginal fault zone of the Fukushima Basin:
According to Active Faults in Japan (New Edition) [29], there are a number of faults, from the vicinity of Shiroishi City in Miyagi Prefecture to the vicinity of Tsuchiyu in Fukushima Prefecture along the western marginal fault zone of the Fukushima Basin. The longest of these extends 15 km, and all are of Class B in terms of activity level. Earthquakes occurring near this fault zone were the Iwashiro earthquake of 1731 (M 6.6) and one in southern Miyazaki Prefecture in 1956 (M 6.0). Neither of these extended along the full length of the fault zone; the impact on the NPP site is described in section 3.2.1 (Past Earthquakes) of Active Faults in Japan (New Edition) [29]. No connection was found between this fault zone and the observed microearthquakes.

Further consideration of this issue was based on the connection between this fault zone and the seismic geological structure described below. With regard to other faults, their size and distance from the site were considered to have a minimal impact at the site.

Seismotectonic structures

Seismically active tectonic structures near the Fukushima Daiichi NPP site can be classified as follows:

— Earthquakes occurring in north-eastern Japan:
  (1) Near the Japan Trench at shallow depth;
  (2) At plate boundaries;
  (3) In the Earth's crust, mainly onshore;
  (4) Within the sinking Pacific Plate.
  The analysis of destructive earthquakes that have occurred in these areas reveals the following:
  - The M 8 class earthquakes occurring near the Japan Trench and further eastward have little impact on the NPP site, given the distance.
  - Earthquakes of around M 7.5 are occurring off the coast of Miyagi, Fukushima and Ibaraki Prefectures, at the plate boundaries.
  - Earthquakes around M 7.0–M 7.5 are thought to be occurring in the Earth's crust west of the Ou Mountains, the Kitakami River Basin and near Nikko.
  - No large earthquakes that would impact the NPP site are occurring in the intra-Pacific Plate slab.

  While M~8 earthquakes occurring near the Japan Trench would not have much of an impact on the site due to their distance, M~9 earthquakes would likely have an impact on both the seismic hazard as well as the tsunami hazard. The underestimation of the tsunami hazard seems to be based on the assumption of M~8 earthquakes instead of using M~9 for the Japan Trench earthquakes.

— Earthquakes that may occur on seismotectonic structures:
  The following earthquakes may occur in seismically active tectonic structures near the Fukushima Daiichi NPP site:
  (1) Earthquakes near the plate boundary: The largest earthquakes occurring in the ocean near the Fukushima Daiichi NPP site were one in Rikuzen in 1646 and an M 7.6 earthquake in Sendai in 1835. The potential magnitude of earthquakes in this area is a maximum of M 7.75. Due to the potential for earthquakes of this maximum magnitude

to occur anywhere near the plate boundary from off the coast of Fukushima Prefecture to the Japan Trench, it is assumed that an M 7.8 earthquake will occur near the plate boundary off the coast of Fukushima Prefecture (Δ = 50 km, focal depth (H) = 40 km). This means that deterministically, a design basis earthquake of M 7.8 has been considered to have occurred at a distance of 50 km (and at a depth of 40 km) from the site.

(2) Crustal earthquakes: The largest earthquake occurring on land near the NPP site was one in Nikko in 1683 with a likely magnitude of between M 7.3 and M 7.5. Due to the connection with active faults in the area, an M 7.5 earthquake is expected to occur in the western marginal fault zone of Fukushima Basin (Δ = 65 km).

Regarding the seismic hazard assessment, there are several aspects of the applied methodology that would be important to understand, as they may have a bearing on the underestimation of the seismic hazards and, consequently, on the eventual underestimation of earthquake concomitant events as the tsunami hazard.

- The estimation of the maximum magnitude was made using a combination of historical earthquake information and the geomorphological fault dimensions. The latter is specifically for 'on-land' events and not for those generated at the Japanese Trench in the Pacific subduction zone.
- The information regarding the on-land faults was taken from official sources, but conservative parameters are assumed for the analysis.
- For the Japan Trench, it is assumed that the magnitude would be about M 8. Furthermore, because of the distance from the site, it was assumed that these events would not impact the site, because closer sources would dominate the seismic hazard.

A point that has not been considered here is the fact that the maximum magnitude associated with the Japan Trench was estimated without much tectonic based justification and was based mostly on observed historical data. An approach similar to the on-land faults (deriving maximum magnitudes in relation to physical fault dimensions) was followed for the maximum magnitude estimation (M 8) of the Japanese Trench, but the number of segments to be mobilized during a single event was underestimated.

2.1.4.4. Actions taken to cope with reassessed earthquake hazards

At the time of the Fukushima Daiichi accident in March 2011, TEPCO was in the process of backchecking or reassessing the seismic safety of the plants in compliance with the requirements from NISA following the issue of the new NSC Regulatory Guidelines of 2006 [9] and as a result of the 2007 Niigata-Chuetsu-Oki earthquake that affected the Kashiwazaki-Kariwa NPP.

Tables 2.1–1 and 2.1–2 [6] present information for all units of the Fukushima Daiichi and Daini NPPs on:

- The original design basis for the seismic horizontal ground motion, including the static horizontal acceleration;
- The revised seismic design bases as part and as a result of the backcheck reassessment process started in 2006;
- The observed maximum acceleration values during the Great East Japan Earthquake. The figures in boxes indicate exceedance with respect to the original design basis values.

As can be seen in Table 2.1–1, for the first two units of the Fukushima Daiichi NPP, the original design basis values are quite modest, with about 0.25g for the ZPGA and with 0.47g for the static horizontal acceleration. The latter is applied for specific design purposes of the building and structures as part of the seismic design process of Japanese NPPs, which is quite conservative, follows three distinct paths and chooses the most conservative result. This point is illustrated in a recent IAEA publication on the results of a benchmark project conducted for the Kashiwazaki-Kariwa NPP (the KARISMA benchmark project) [30].

Another interesting point to be noted is that, for Unit 6, the original design values of the peak ground acceleration are higher than the reassessed values, in line with the application of the new NSC Regulatory Guidelines of 2006 and also higher than the ZPGA values recorded from the March 2011 event.

Regarding Units 1–5, a significant increase can be observed in the reassessed ground motion (obtained after 2006–2007) as compared with the original design ground motions. This may be attributed to two factors: the use of a more seismotectonic based approach to seismic hazard analysis (less dependence on historical seismicity), as described previously for the new seismic guidelines; and a more conservative estimation of the ground motion for a given set of magnitude/distance pairs due to the availability of a much richer ground motion database feeding into more robust ground motion prediction equations (GMPEs).

The reassessed ground motion parameters for the Fukushima Daiichi NPP were exceeded at Units 2, 3 and 5 in the east–west (E–W) direction. It should be noted that similar exceedance is not observed in the north–south (N–S) and vertical (U–D) components, where a comfortable margin still remains between the reassessed design ground motions and the observed accelerations.

Furthermore, the observed accelerations at the Fukushima Daini units (at least those in the two horizontal directions) are significantly less than those for the Fukushima Daiichi units, although the distance from the causative fault may differ only slightly (about 10%). It can be concluded that some directivity effect (E–W component) or local soil response at the Fukushima Daiichi NPP site may be responsible for this difference in the exceedances observed at the two sites.

It is possible to perform a brief comparison of the consequences of the exceedance of the seismic accelerations and the tsunami heights observed in March 2011 with respect to the re-evaluated seismic and tsunami hazard values. The reassessed seismic hazard was exceeded at three units and only in one direction. While the seismic source parameters of the 11 March 2011 earthquake (e.g. the magnitude of the subduction earthquake) may have been underestimated, the consideration in the seismotectonic model of other seismogenic sources (as part of the regional seismotectonic model which may be located onshore) and the use of new GMPEs for calculating the reassessed hazards may have compensated for part of this underestimation in the maximum magnitude of one of the sources (i.e. in the offshore subduction zone).

Regardless of the underestimation of the seismic hazard, TEPCO’s Progress Report No. 2 [31] confirmed that the earthquake did not have a serious impact on plant safety. It did not cause a loss of coolant accident (LOCA) or the loss of emergency diesel generator (EDG) functions in Unit 1, as speculated after the accident and indicated in the Report of the National Diet of Japan [32]. Thus, it has been concluded that pipe breaks causing leakage on a scale that would have affected the development of the accident did not occur. Concerning the loss of EDG functions, it has been shown that this was not caused by the earthquake, since the recorded data clarified that the EDG function loss followed immediately after the loss of function of the seawater pumps, which is considered to have been caused by the tsunami.

Tsunami hazards are caused by tsunami waves generated by fault dislocation at the seabed produced by earthquakes as the root cause. The tsunamitectonic model for calculating the tsunami wave height differs greatly from the seismotectonic model for calculating the seismic hazard expressed in terms of ground accelerations. Only subduction earthquakes which can generate out of plane components of fault displacement are the cause of the generated tsunami water waves. Therefore, the underestimation of the source parameters (e.g. the magnitude) would be more difficult to compensate. Nevertheless, the tsunami hazard as re-evaluated in the trial calculations shows little difference from the one that actually occurred.

TABLE 2.1–1. MAXIMUM ACCELERATION VALUES OBSERVED AT UNITS 1–6 OF THE FUKUSHIMA DAIICHI NPP, AND COMPARISON WITH ORIGINAL DESIGN BASIS VALUES AND RE-EVALUATION VALUES [6]

Fukushima Daiichi NPP unit	Maximum measured acceleration value (Gal)			Maximum response acceleration value (Gal)					Static horizontal acceleration (Gal)
				Revised design basis (2008)			Original design basis (1966)		
	N–S	E–W	U–D	N–S	E–W	U–D	N–S	E–W	
Unit 1	$\boxed{460}$	$\boxed{447}$	258	487	489	412	245		470
Unit 2	$\boxed{348}$	$\boxed{550}$	302	441	438	420	250		
Unit 3	$\boxed{322}$	$\boxed{507}$	231	449	441	429	291	275	
Unit 4	281	$\boxed{319}$	200	447	445	422	291	283	
Unit 5	$\boxed{311}$	$\boxed{548}$	256	452	452	427	294	255	
Unit 6	298	444	244	445	448	415	495	500	

Note: Values in boxes indicate that the maximum recorded value was beyond the original design basis.

TABLE 2.1–2. MAXIMUM ACCELERATION VALUES OBSERVED AT UNITS 1–4 OF THE FUKUSHIMA DAINI NPP, AND COMPARISON WITH THE ORIGINAL DESIGN BASIS VALUES AND RE-EVALUATION VALUES [6]

Fukushima Daini NPP unit	Maximum measured acceleration value (Gal)			Maximum response acceleration value (Gal)					Static horizontal acceleration (Gal)
				Revised design basis (2008)			Original design basis (1966)		
	N–S	E–W	U–D	N–S	E–W	U–D	N–S	E–W	
Unit 1	254	230	305	434	434	512	372	372	470
Unit 2	243	196	232	428	429	504	317	309	
Unit 3	$\boxed{277}$	$\boxed{216}$	208	428	430	504	196	192	
Unit 4	$\boxed{210}$	$\boxed{205}$	288	415	415	504	199	196	

Note: Values in boxes indicate that the maximum recorded value was beyond the original design basis.

A final point to be highlighted is related to the design practice to cope with seismic and tsunami hazards. As mentioned before, the Japanese practice of seismic design incorporates several approaches, adopting the one that is most conservative. This results in a robust design of SSCs in NPPs. In the case of the Fukushima Daiichi accident, this design clearly prevented significant consequences from the exceedance of ground motion parameters, which would have been the result if other design criteria, like the application of more than three times the static acceleration of the conventional building codes, had been applied for the design of the buildings at the Fukushima Daiichi site. This was also observed and documented by the IAEA after the Niigata-Chuetsu-Oki earthquake in 2007 at another TEPCO plant, the Kashiwazaki-Kariwa NPP.

However, the design against tsunami hazards is quite sensitive to cliff edge effects, and this leads to less robust solutions than those that can be implemented for seismic events. A minor exceedance of flood levels may lead to severe consequences by causing internal flooding. This difference was highlighted in the report of the IAEA International Fact Finding Expert Mission of the Fukushima Daiichi NPP [6].

Figure 2.1–3 illustrates the location of the measurement points for the measurements shown in Table 2.1–1 for Fukushima Daiichi NPP units [6].

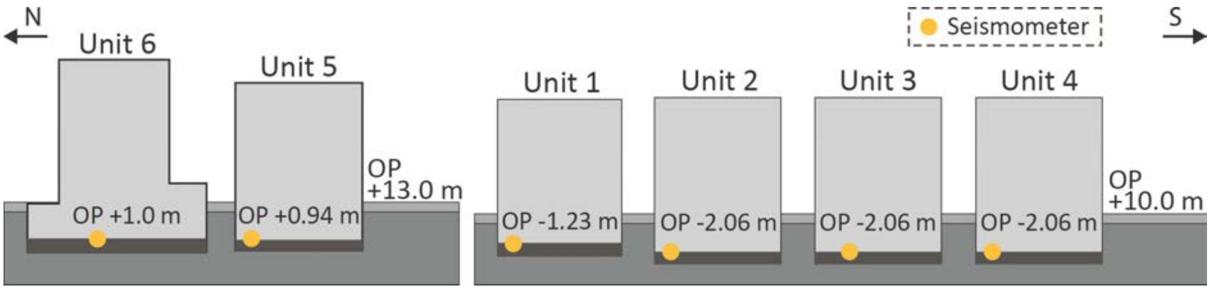


FIG. 2.1–3. Maximum acceleration values observed at Units 1–6 of the Fukushima Daiichi NPP, and comparison with the original design basis values and re-evaluation values [6].

2.1.5. Design basis and reassessments of tsunami hazards and remedial actions taken during the operational life of the Fukushima Daiichi plant

2.1.5.1. Characteristics of tsunami hazards

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. Large meteorites impacting the ocean can also generate a tsunami. All oceanic regions and sea basins of the world, and even fjords and large lakes, can be affected by tsunamis.

Tsunami waves and associated phenomena may produce severe damage to installations located in coastal areas. With regard to nuclear installations, IAEA safety standards require that the characteristics of potential tsunamis that can affect safety be assessed, taking into consideration pre-historical and historical data and other types of associated hazards, with account taken of any amplification due to the coastal configuration at the site, (see paras 3.24–3.28 of Ref. [16]).

Consequently, if the potential of a tsunami exists and detailed hazard characterization is carried out, the facility, installation or plant should be designed to withstand the event according to design bases, including specific performance criteria which are to be determined as a result of a tsunami hazard assessment and the intensity or magnitude of the postulated event.

In addition to the effects produced by a variation in water levels (maximum and minimum), the hazardous effects of tsunami waves include strong currents in harbours and bays, bores in rivers, estuaries and lagoons, and huge hydrodynamic forces. Sedimentation phenomena, including deposition and erosion, may also be generated owing to large forces at the sea floor.

Two important issues should be considered in the process of assessing tsunami hazards when estimating the flooding design basis for a nuclear installation:

- The minimum water level produced by the receding wave of the tsunami plays a key role in safety, since, for a period of time, the cooling water may be disrupted or interrupted.
- Tsunami waves are strongly dependent on the configuration of the coast (bathymetry and topography). Therefore, the hazard assessment (e.g. the runup and water level at the shoreline) not only needs to consider the coastal configuration at the site evaluation stage before the plant has been built, but also — and significantly — it needs to consider the final layout of the installation with all modifications to the ground elevation, slopes, grade levels, etc.

Finally, as shown in Fig. 2.1-4, tsunamis are one of the factors to consider in the assessment of the final flood level (maximum and minimum levels). Thus, the tide height, wave heights produced by other meteorological and hydrological phenomena, tectonic subsidence or uplift have to be considered in combination with the occurrence of the tsunami waves. The final flooding level reaching the site is the combination and result of all those factors at the time of the occurrence of the event.

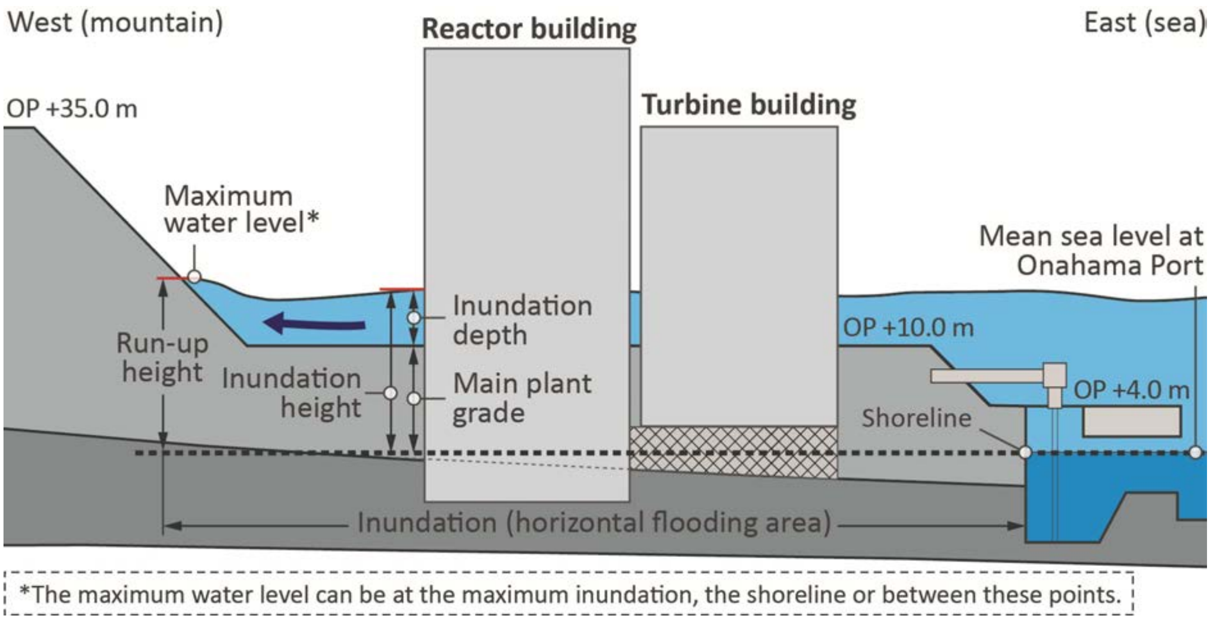


FIG. 2.1–4. Tsunami parameters at the shoreline [20].

2.1.5.2. Design basis in relation to tsunami hazards

As indicated in Section 2.1.2, according to the approach applicable in the 1960s, when the application for the Establishment Permit for the Fukushima Daiichi NPP was submitted, it was common international practice to use historical records for assessing the design basis tsunami height for designing the installation but adding conservative assumptions and using deterministic approaches to count for the potential occurrence of extreme events with very low annual frequency of occurrence. Later, during the 1970s, the methods for assessing the tsunami levels evolved and numerical simulations based on source models of the tectonic mechanisms for characterizing the tsunamigenic sources that generate the sea floor deformations were developed and used. In line with that approach, the design basis tsunami was determined for each site on the basis of the available information on the maximum historical observed tsunamis and the greatest tsunamis induced by submarine active faults (Fig. 2.1–5).

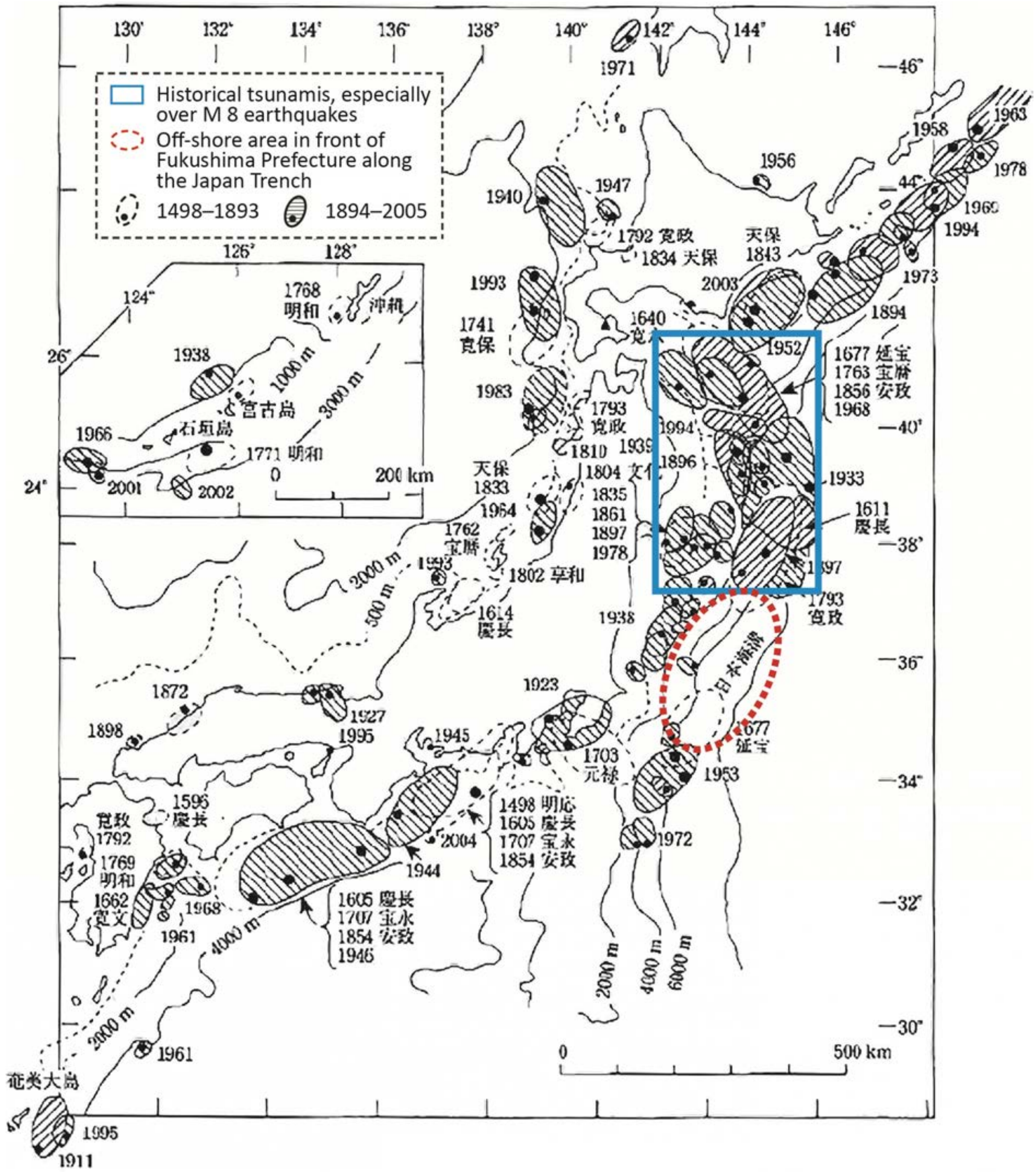


FIG. 2.1–5. Historical earthquakes in the Japan subduction trench [33].

Following these criteria, the tsunami hazard at the Fukushima Daiichi NPP site was initially estimated based on the data and observations from the tsunami generated by the M 9.5 earthquake in Chile in 1960 that reached Japan's east coast. Thus, the design maximum height was defined at OP +3.122 m in accordance with the tide level records observed during that event at Onahama Port, in Fukushima Prefecture, located 50 km south of the site (see also Section 1.2 of Technical Volume 1).

As reported by NISA to the 2011 IAEA Fact Finding Expert Mission [6], this value still represented the licensing design basis for flooding generated by tsunamis at the time of the accident in March 2011. Based on that practice, according to the Japanese reports prior to the accident in

March 2011, the regulatory body had no authority to impose new requirements on the licensee or to change the design bases.

Consequently, the OP +4.00 m of the plant grade level for locating the safety related SSCs at the water intake area corresponding to the location of the seawater cooling pumps would have been considered sufficient by TEPCO to cope with the maximum flood level evaluated as was indicated above. Moreover, the main plant grade levels (i.e. the elevation of the buildings and structures of the nuclear island and balance of plant) that were established at OP +10.00 m for Units 1–4 and at OP +13.00 m for Units 5 and 6 would have been considered enough margin.

Table 2.1–3 summarizes the values of the maximum and minimum tsunami flood levels adopted for the Fukushima Daiichi and Daini NPP sites and for the different units.

TABLE 2.1–3. ORIGINAL DESIGN BASIS: MAXIMUM AND MINIMUM FLOOD LEVELS FOR TSUNAMI HAZARDS AT THE FUKUSHIMA DAIICHI AND DAINI NPP SITES [33]

Site	Unit	Year of Establishment Permit	Rise	Drawdown
Fukushima Daiichi NPP	1	1966	OP +3.122 m	OP −1.918 m
	2	1968		
	3	1970		
	4	1972		
	5	1971		
	6	1972		
Fukushima Daini NPP	1	1974	OP +3.690 m	
	2	1978		
	3	1980	OP +3.705 m	
	4	1980		

OP −1.918 m	Historical low water level
OP +3.122 m	Height of the tide from the Chilean tsunami at Onahama Port on 24 May 1960
OP +3.690 m	= OP +1.490 m + 2.2 m
OP +1.490 m	Mean of high tides at Onahama Port
2.2 m	Height of the tsunami component from the Chilean tsunami at Onahama Port on 24 May 1960
OP +3.705 m	= OP +1.505 m + 2.2 m
OP +1.505 m	Mean of high tides

At this point, it should be noted that the approach applied used only known historical data, leading to the determination of the plant design basis, as shown in Figs 2.1–5 and 2.1–6:

— Figure 2.1–5 shows the location and rupture areas of the earthquakes that generated those tsunamis at the subduction offshore zone (Japan Trench) of the Eurasian and Pacific tectonic plates. The figure shows that no records were available of earthquakes occurring along the Japan Trench in the offshore area in front of Fukushima Prefecture.
— Figure 2.1–6 shows that at the Fukushima Daiichi site the tsunami flood levels were low (of the order of a few metres) compared with the levels in locations to the north of the site — as in Iwate and Miyagi Prefectures — that had recorded maximum levels of around +38 m for tsunamis that occurred in 1611, 1677, 1896, 1933, 1938 and 1960.

The following points should be highlighted in this approach:

- The use of historical records dating from a very recent period of a few hundred years only. No conservative assumptions were made regarding the need to take account of the potential occurrence of extreme events (i.e. a typical period of recurrence of the order of 10 000 years).
- The correspondence between the lack of historical records of tsunami flood levels at the specific location of the Fukushima Daiichi and Daini sites and the lack of data on the occurrence of earthquakes in the offshore area in front of the sites, i.e. a seismic gap for that seismogenic source which coincides with the lack of a high level of tsunami flooding phenomena.

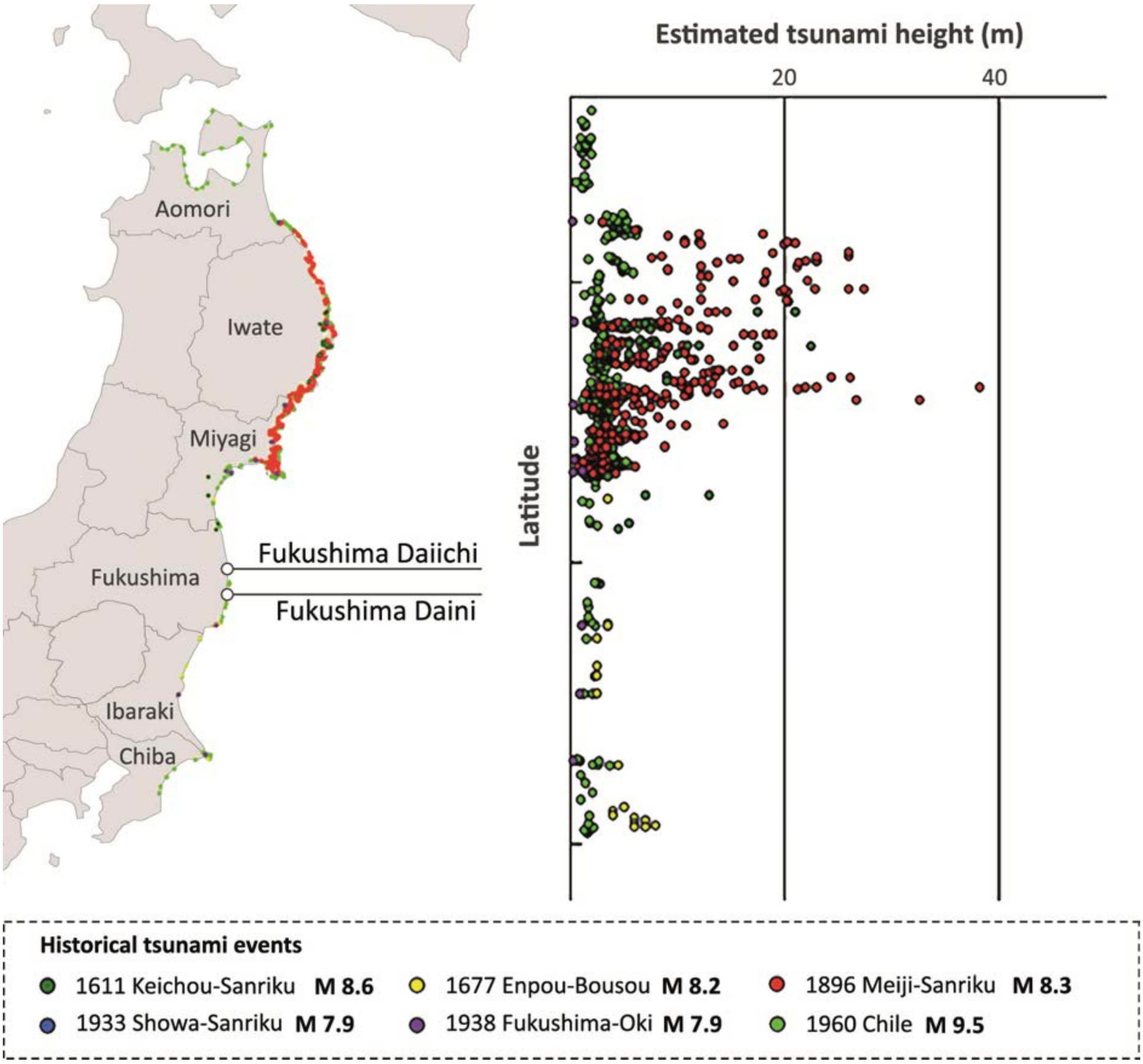
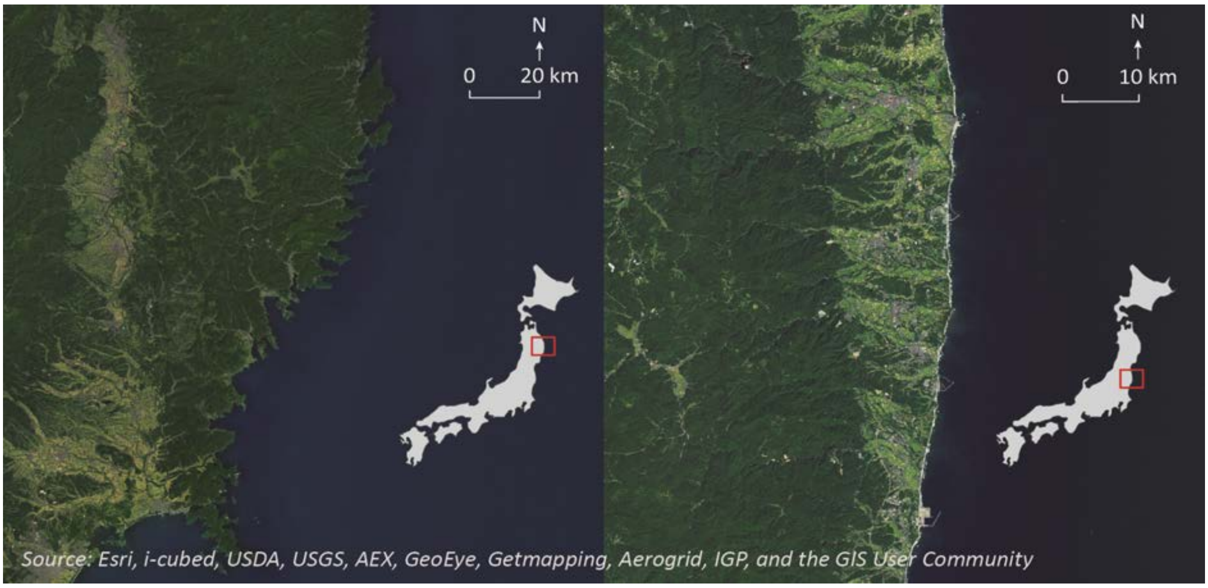


FIG. 2.1–6. Historical tsunami records [33].

The practice in the 1960s, as described above, was not unique to TEPCO. Other utilities in Japan also used the same approach, for example, at the Onagawa NPP operated by the Tohoku Electric Power Company, located about 120 km to the north of the Fukushima Daiichi NPP site in Miyagi Prefecture. In this region, most of the larger earthquakes occurred in historically recorded times in the Japan subduction trench located off the eastern coast, as shown in Fig. 2.1–5, in comparison with the ones recorded at Fukushima Prefecture.

Using the same approach as in the Fukushima Daiichi NPP design basis, the maximum level adopted for the flood design event for Unit 1 of the Onagawa NPP site in 1970 was OP +2.00–3.00 m, as indicated in its Establishment Permit. This level corresponds to values obtained from the literature survey. Later, for Unit 2, in 1987, the highest flood level was determined at OP +9.10 m using numerical simulation techniques. However, the main plant grade level of the Onagawa NPP site is OP +14.80 m. Another reason for the increased level of tsunami runup historically recorded at the north of the Fukushima Daiichi NPP site, in the region of Miyagi and Iwate Prefectures, is the influence of the coastal topography and bathymetry. The Fukushima Daiichi site is located along a rectilinear coastline, while at the Onagawa NPP site, the coastline is a ria (a coastal inlet) with a V shape that would significantly amplify the wave heights of tsunamis. Figures 2.1–7 and 2.1–8, illustrate these issues [33].



(a) (b)

FIG. 2.1–7. (a) Aerial view of Iwate and Miyagi Prefectures; and (b) Fukushima Prefecture.

Finally, there is the issue of floods other than tsunamis. The Establishment Permit [3] document indicates the following in Section 2.2.3:

> “2.2.3 Wave Height
>
> “Large waves occurring near the site are produced by mainly typhoons or low pressure systems and according to observational records after February 1965 the largest wave was produced by Typhoon #28 (1965) and had a significant wave height at a water depth of 10 m of 6.51 m and a maximum wave height of 7.94m.
>
> “Waves usually hit perpendicular to the coast flowing in an easterly direction and the majority of waves fall between the directions of ESE and ENE.”

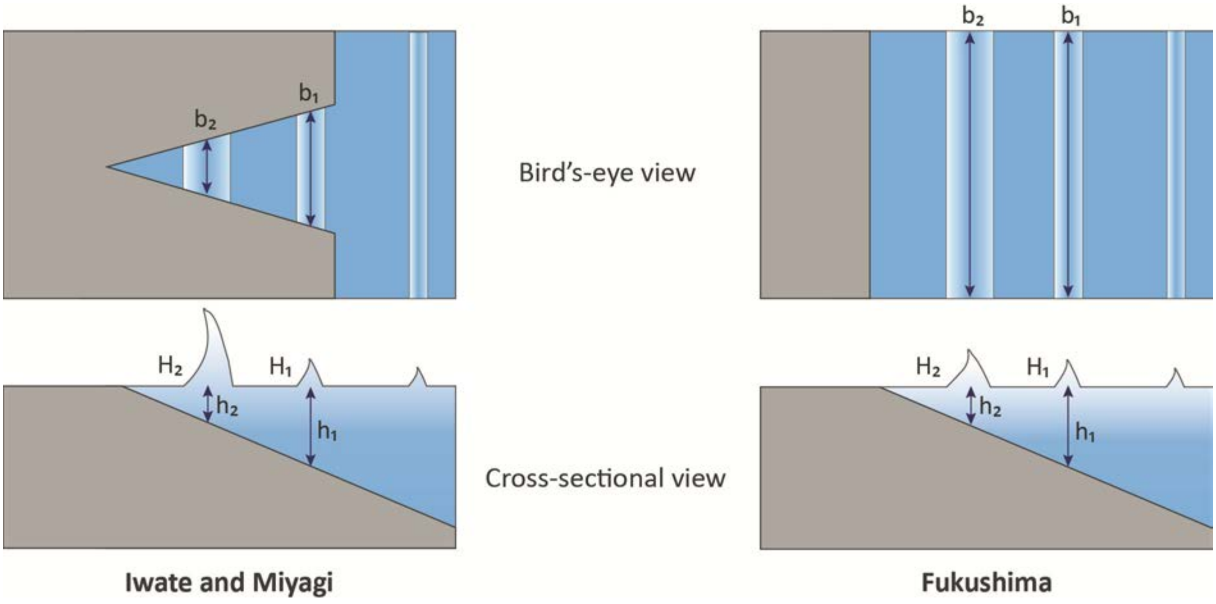


FIG. 2.1–8. Influence of topography and bathymetry in Iwate and Miyagi Prefectures in comparison with Fukushima Prefecture [33].

This would mean that flood levels significantly higher than the estimated value of OP +3.122 m for a tsunami would be considered for other natural phenomena such as the rare meteorological phenomena of powerful typhoons. In this regard, TEPCO has clarified that site protective design countermeasures adopted for coping with the high waves from typhoons include breakwater structures which have the function of damping the waves, and thus the heights of waves in the harbour would stay below the adopted level of OP +4.00 m.

2.1.5.3. Re-evaluation of tsunami hazards

The numerical simulation of earthquake induced tsunamis based on a tectonic mechanism and source modelling was carried out only after the mid-1970s. These simulation techniques involve the characterization of tectonic sources of earthquakes that occur at the bottom of the ocean and which cause the uplift and subsidence of the seabed and, subsequently, lead to the generation of tsunami waves at the sea surface, which propagates in the sea to reach shorelines far away. These relative displacements between both sides of the colliding tectonic plates are the tsunami source. Table 2.1–4 summarizes the evaluations performed by TEPCO after the Establishment Permit, including the remedial measures taken at each time.

As can be seen from the table, from 1966 to 2002, there were no developments and tsunami hazard levels were not reassessed. After 2002, following new guidance from the JSCE in 2002 and 2009 and to strengthen safety according to the disaster prevention plans of the local governments (Ibaraki and Fukushima Prefectures in 2007), the original maximum flood level due to tsunamis was revised upwards from the design basis of OP +3.122 m to the higher level of OP +5.70 m in 2002, and to OP +6.10 m in 2009 using latest bathymetry and tidal data. In addition to those re-evaluations, TEPCO conducted a number of trial calculations using approaches and assumptions different than those proposed by the JSCE. They are described in the following sections.

TABLE 2.1–4 SUMMARY OF TSUNAMI EVALUATIONS PERFORMED BY TEPCO BETWEEN 1966 AND 2009 [33]

Year	Tsunami height	Evaluation method	Countermeasures
1966	OP +3.122 m	Establishment Permit (observed height as result of Chilean tsunami in 1960)	—
2002	OP +5.7 m	Japan Society of Civil Engineers (JSCE) assessment method	To raise elevation of the pumps To make buildings watertight, etc.
2007	OP +4.7 m	Disaster prevention plan by Ibaraki Prefecture	Unnecessary
2007	Approx. OP +5 m	Disaster prevention plan by Fukushima Prefecture	Unnecessary
2009	OP +6.1 m	Latest bathymetric and tidal data on the basis of the JSCE assessment method	To raise pumps' elevation, etc.

2.1.5.4. The JSCE methodology

Using the JSCE methodology, the Fukushima Daini, Tokai and Onagawa NPPs also revised their tsunami flood levels upwards as follows:

- Fukushima Daini NPP: OP +5.20 m (increasing 2.078 m from the original design basis of OP +3.122 m);
- Tokai NPP: OP +4.88 m (with no provisions in this regard in the original design basis);
- Onagawa NPP: OP +13.60 m (increasing 4.5 m from the value of OP +9.10 m estimated in 1987).

Specific aspects of the JSCE methodology included the following:

- It deals only with earthquake generated tsunamis.
- It deals mainly with local or near field tsunamis, i.e. tsunamis generated close to the Japanese shorelines, since the effects of near field tsunamis are greater and more destructive than those of far field tsunamis. It recognizes the occurrence of far field, distant tsunamis (e.g. tsunamis originating in the Alaska/Cascadia subduction zone in North America or in the Nazca plate subduction zone in South America), but it uses them for validating the historical data. The simulation models assume that in those regions of the world the largest historical earthquakes have already been experienced.
- It provides specific guidance only on maximum–minimum wave heights, i.e. on the variation of water levels; it does not provide specific recommendations on how to address the issue of other tsunami related hazards (hydrodynamic forces of the waves, debris impact, sedimentation, etc.) as indicated in chapter 2, point 2 of the JSCE document [26].
- It defines the 'design tsunami' as the one that causes the maximum and minimum water waves at the site among all the various possible scenario tsunamis. The scenario tsunamis are a large number of postulated tsunamis for which numerical simulations are performed, each with different characterizations of the source model. Accordingly, a parametric study is conducted, varying some of the source parameters (e.g. fault position, depth of upper edge, strike direction, dip angle, dip direction), with the uncertainties taken into account. The design tsunami should exceed all recorded and calculated historical tsunamis at the target site. The historical tsunami records are used to validate the numerical simulations.
- It specifies standard tsunamigenic sources, i.e. those common to any target sites. This is one of the most critical aspects of this guidance, since the tectonic sources that can generate tsunamis are modelled with all their characteristics provided by the established guidance, which corresponds to historical tsunamis. The scenario tsunamis corresponding to the different tectonic regions around Japan are defined in Table 2.1–5.

TABLE 2.1–5. TSUNAMIGENIC SOURCES FOR JAPAN AS DEFINED IN TABLE 4-1 OF REF. [26]

Classification	Sea area	Types of earthquakes
Tsunamis due to earthquakes along the plate boundaries	Sea areas related to the subduction of the Pacific plate	Typical interplate earthquakes Tsunami earthquakes (slow earthquakes) Intraplate earthquakes with a reverse fault Intraplate earthquakes with a normal fault
	Sea areas related to the subduction of the Philippine Sea plate	Typical interplate earthquakes
Tsunamis due to earthquakes in the eastern margin of the Sea of Japan	Eastern margin of the Sea of Japan	Shallow inland earthquake
Tsunamis due to earthquakes in the submarine active faults	Entire area around Japan	Shallow inland earthquake

An important consideration discussed explicitly in the JSCE methodology is tectonic plate displacement — either subsidence or uplift — produced by the tsunami generating seismic event and affecting the onshore area. This issue is not indicated in the methodology for determining the tsunami flood level, since the main plant grade level changes owing to the crustal movement, although TEPCO has considered it in the model simulation using an elastic theory.

During the 11 March 2011 event, the onshore ground level at the Fukushima Daiichi site experienced a subsidence of approximately 0.66 m, meaning that the main plant grade level had subsided by such an extent. Therefore, instead of an OP +10.00 m main grade level, the plant was located about 0.66 m below, i.e. approximately OP +9.34 m, when the tsunami reached the coast and flooded the plant site. In any case, the new main plant grade level should count for this phenomenon.

In summary, the tectonic subsidence is usually considered in estimating the tsunami flood level, but the level of water reaching a plant grade level is not the difference between the tsunami estimated level minus the original plant grade level, but minus the 'subsided new' plant grade level.

Figure 2.1–9 and Table 2.1–6 illustrate the geodetic measurements after the earthquake, at locations P1 to P5 and T1 to T3 of the main plant grade level and the subsidence values that the plant has experienced at those points, with an average of 0.662 m.

TABLE 2.1–6. GEODETIC MEASUREMENTS AND SUBSIDENCE AFTER THE 11 MARCH 2011 EARTHQUAKE (AFTER THE TEPCO REPORT TO NISA IN OCTOBER 2011) (ORIGINAL IN JAPANESE) [34][a]

T1	T2	T3	P1	P2	P3	P4	P5
OP +13 m	OP +10 m	OP +4 m	OP +4 m	OP +4 m	OP +4 m	OP +4 m	OP +10.2 m
OP +12.375 m	OP +9.338 m	OP +3.300 m	OP +3.379 m	OP +3.358 m	OP +3.209 m	OP +3.370 m	OP +9.579 m

[a] Top row: original measurements; bottom row: measurements following the earthquake.

FIG. 2.1–9. Geodetic measurements and subsidence after the 11 March 2011 earthquake (after the TEPCO report to NISA in October 2011) (original in Japanese) [4].

The application of the JSCE methodology in 2002 for re-evaluating the design tsunami at the Fukushima Daiichi NPP is illustrated in Fig. 2.1–10, where the eight zones of potential tsunami sources are indicated. It should be highlighted that the offshore trench fault source in front of the Fukushima Daiichi NPP is not included in that re-evaluation.

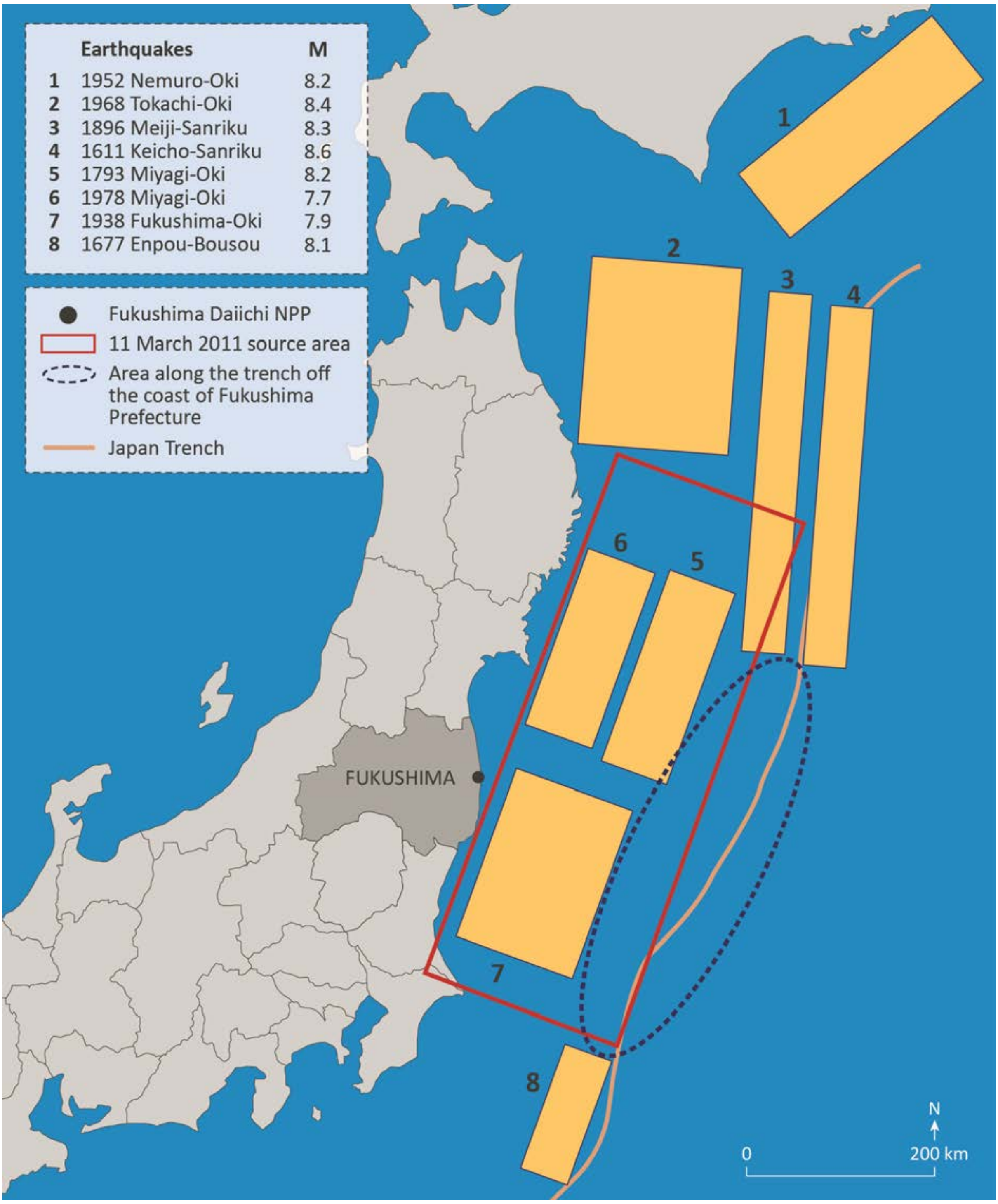


FIG. 2.1–10. Standard source model as defined by JSCE guidance, 2002 [26].

One hundred and forty-five simulations (see Fig. 2.1–11) were performed, with the dominant source being the one identified as No. 7 with M 8, corresponding to the earthquake at Fukushima-Oki (M 7.9) in 1938, which was assumed at M 8 in the calculations.

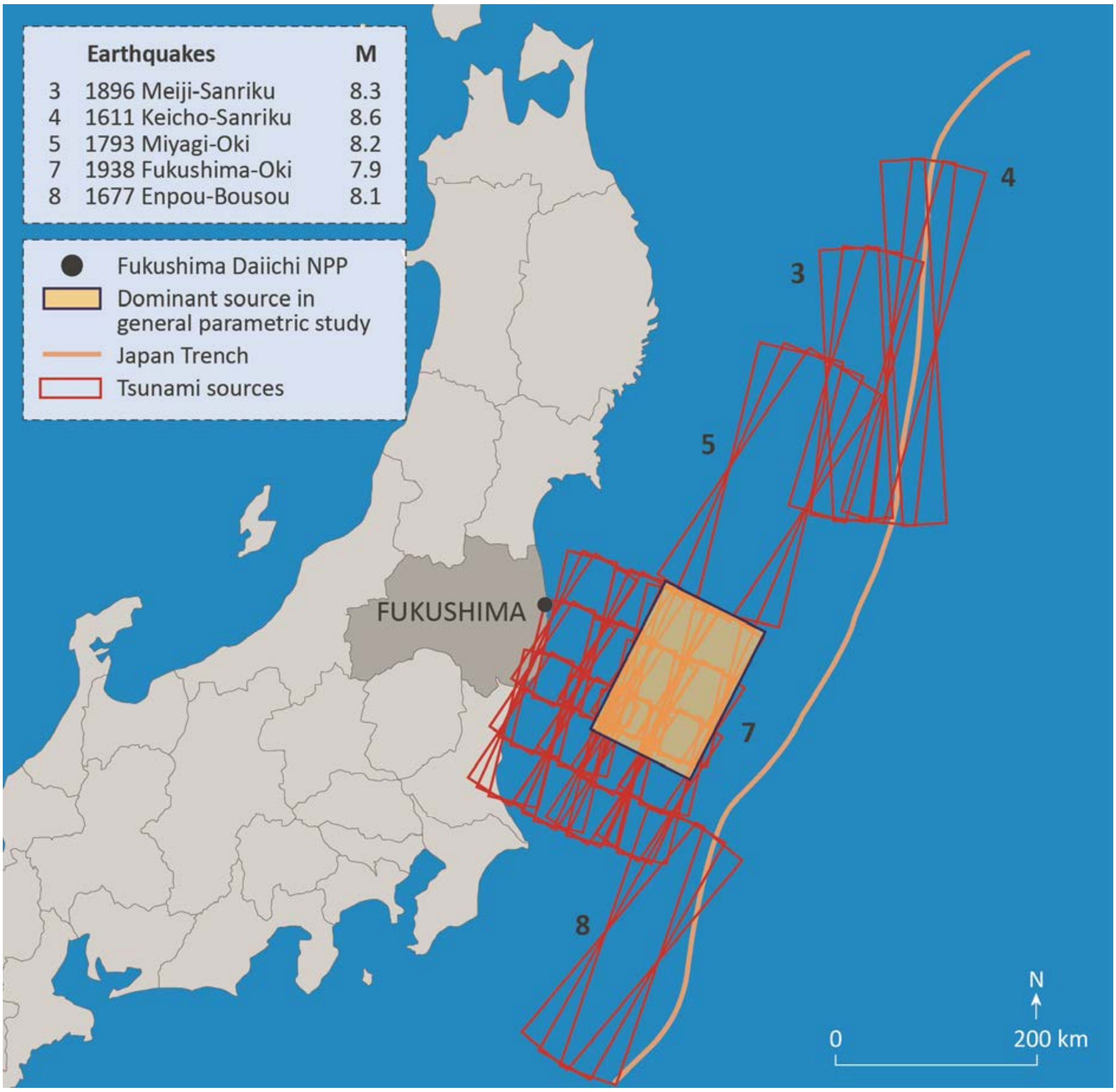


FIG. 2.1–11 Scenario of tsunamis for calculations performed by TEPCO in 2002 [33].

As a result of these calculations by TEPCO, the tsunami hazard water levels (maximum and minimum) were obtained with a maximum high water level of OP +5.7 m, while the minimum water level of OP −3.60 m was calculated on the basis of the 1960 Chile earthquake (M 9.5).

Another important consideration is the runup value, which is the water height, reached at the maximum inundation point, as shown in Fig. 2.1–4. The estimated value corresponds to the tsunami height at the location that can be called the water cooling intake point, which is the point at shoreline at which the water reaches the level of the cooling intake structures. TEPCO indicated [33] that the runup was taken into account and did not significantly increase the calculated value of OP +5.7 m since it did not reach the main grade level of OP +10 m.

In 2007, to confirm the level of nuclear safety in relation to the disaster prevention plans of Ibaraki and Fukushima Prefectures, TEPCO voluntarily performed new evaluations with the source models corresponding to historical tsunamis, which were defined as illustrated in Fig. 2.1–12.

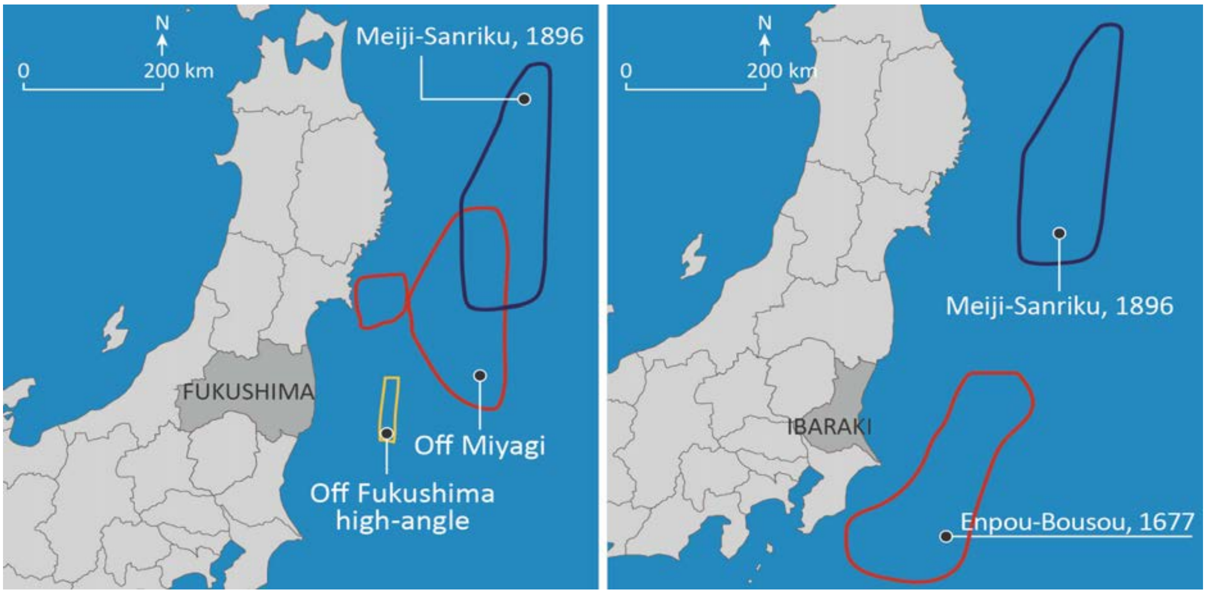


FIG. 2.1–12. Tsunamigenic sources for the evaluations performed in 2007 to ensure safety in relation to the disaster prevention plans of Ibaraki and Fukushima Prefectures [33].

The Japan Atomic Power Company (JAPC), the operator of the Tokai Daini NPP, recalculated the design tsunami heights for its site based on these new wave source models to maintain consistency with the assumptions in the disaster prevention plans. In the case of the Tokai Daini NPP, the results of these evaluations showed maximum water levels of OP +4.70 m and +5.00 m, which were similar to the results obtained in 2002. The design tsunami height had to be revised upwards, and the JAPC decided to construct a 7.0 m high wall at Tokai Daini to protect the emergency seawater pump room [10].

Later, in 2009, with more precise topography (onshore), bathymetry (offshore) and tidal data of the Fukushima Daiichi NPP site, TEPCO again performed this evaluation using the JSCE source models, and obtained the maximum water level of OP +6.10 m. Figures 2.1–13 and 2.1–14 illustrate the results obtained by TEPCO of the tsunami re-evaluations performed between 2002 and 2009 for the Fukushima Daiichi and Daini NPPs.

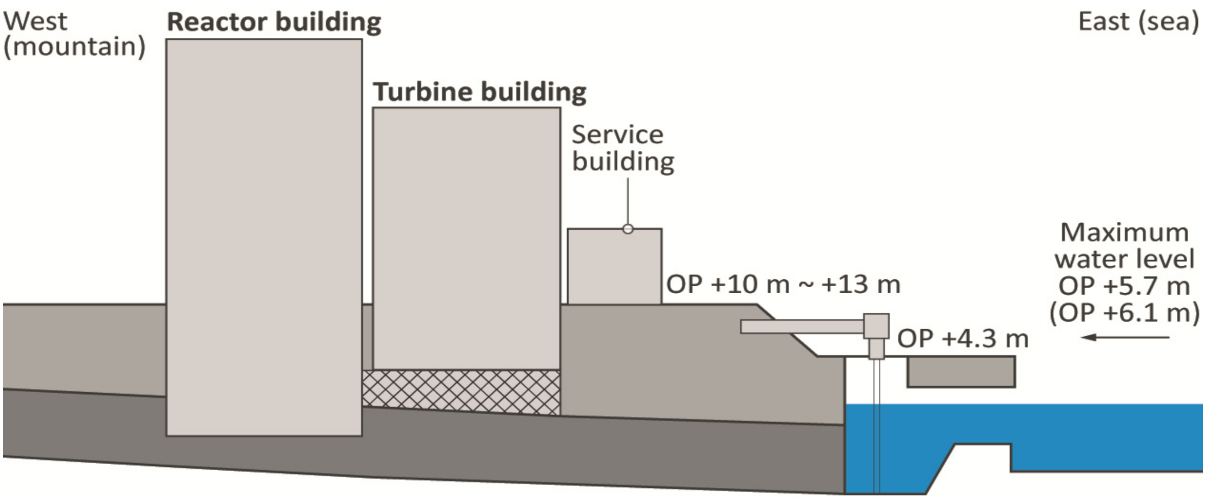


FIG. 2.1–13. Summary of 2002 and 2009 tsunami re-evaluations of the Fukushima Daiichi NPP using the JSCE methodology (topography and tide data were revised in 2009) [33].

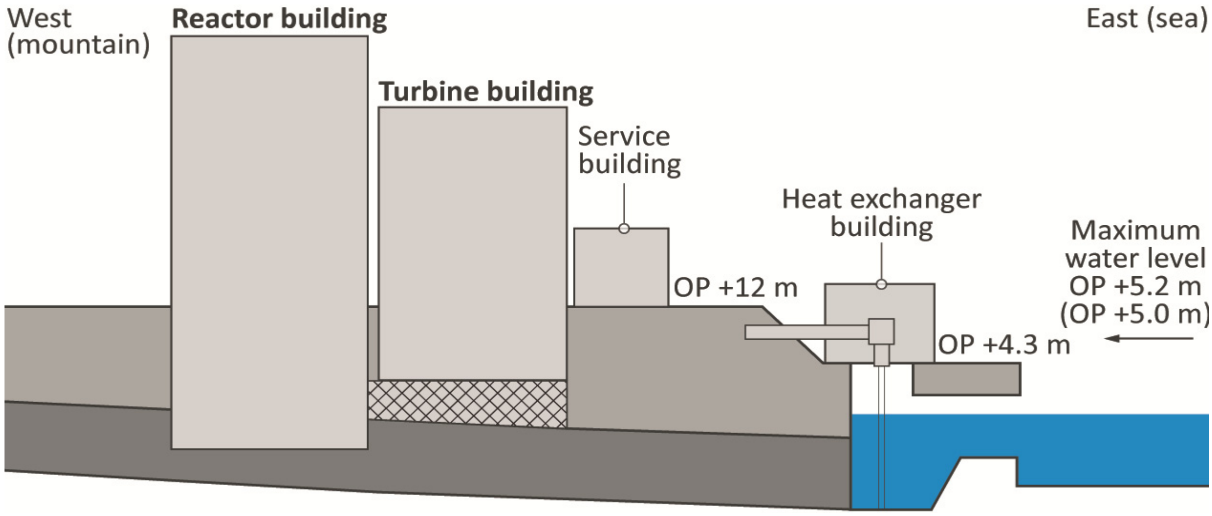


FIG. 2.1–14. Summary of 2002 and 2009 tsunami re-evaluations of the Fukushima Daini NPP using the JSCE methodology (topography and tide data were revised in 2009) [*33*].

In addition to the studies completed by TEPCO between 2002 and 2009 based on the JSCE deterministic methodology and wave source models, as described above, a number of trial calculations were performed using wave source models or methodologies that went beyond the JSCE tsunami assessment method. In the following section, these trial calculations, and the results obtained, are described.

2.1.5.5. First trial analysis using the probabilistic tsunami hazard assessment approach

The first trial analysis was conducted using a probabilistic approach under discussion at the JSCE and identified as a prototype approach. Thus, a probabilistic tsunami hazard assessment was performed by TEPCO to confirm the adaptability and improvement of the JSCE prototype method [33].

This first trial analysis resulted in a mean annual frequency of exceedance of 10^{-5} to 10^{-6} for a tsunami exceeding 10 m at the Fukushima Daiichi NPP site. TEPCO did not interpret this result as the actual frequency of tsunamis that could strike the nuclear power plants at the Fukushima Daiichi NPP site. These results were presented in the International Conference on Nuclear Engineering-14 (ICONE-14) on 17–20 July 2006.

2.1.5.6. Trial analyses using the JSCE methodology with different source models

Two additional trial calculations were performed in 2008–2009 by TEPCO using the JSCE methodology, but with different source models, as described in Fig. 2.1–15:

— Using as a tsunami source the model proposed by the Headquarters for Earthquake Research Promotion (HERP) with an M 8.2 earthquake rupturing an area of 200 km × 50 km located anywhere in the offshore zone, as indicated in the figure on the left in Fig. 2.1–15;
— Using as a tsunami source the Jogan 869 earthquake with M 8.4 and with the assumed location as proposed by Satake et al. [35], as indicated in the graph in the figure in the middle in Fig. 2.1–15.

The comparison of these two models with respect to the rupture and magnitude of the 11 March 2011 earthquake is also illustrated in the figure on the right in Fig. 2.1–15, as well as with respect to the 2002 re-evaluation that strictly applied the JSCE prescribed methodology.

The results obtained from these two trial calculations using different source models are described in the following sections.

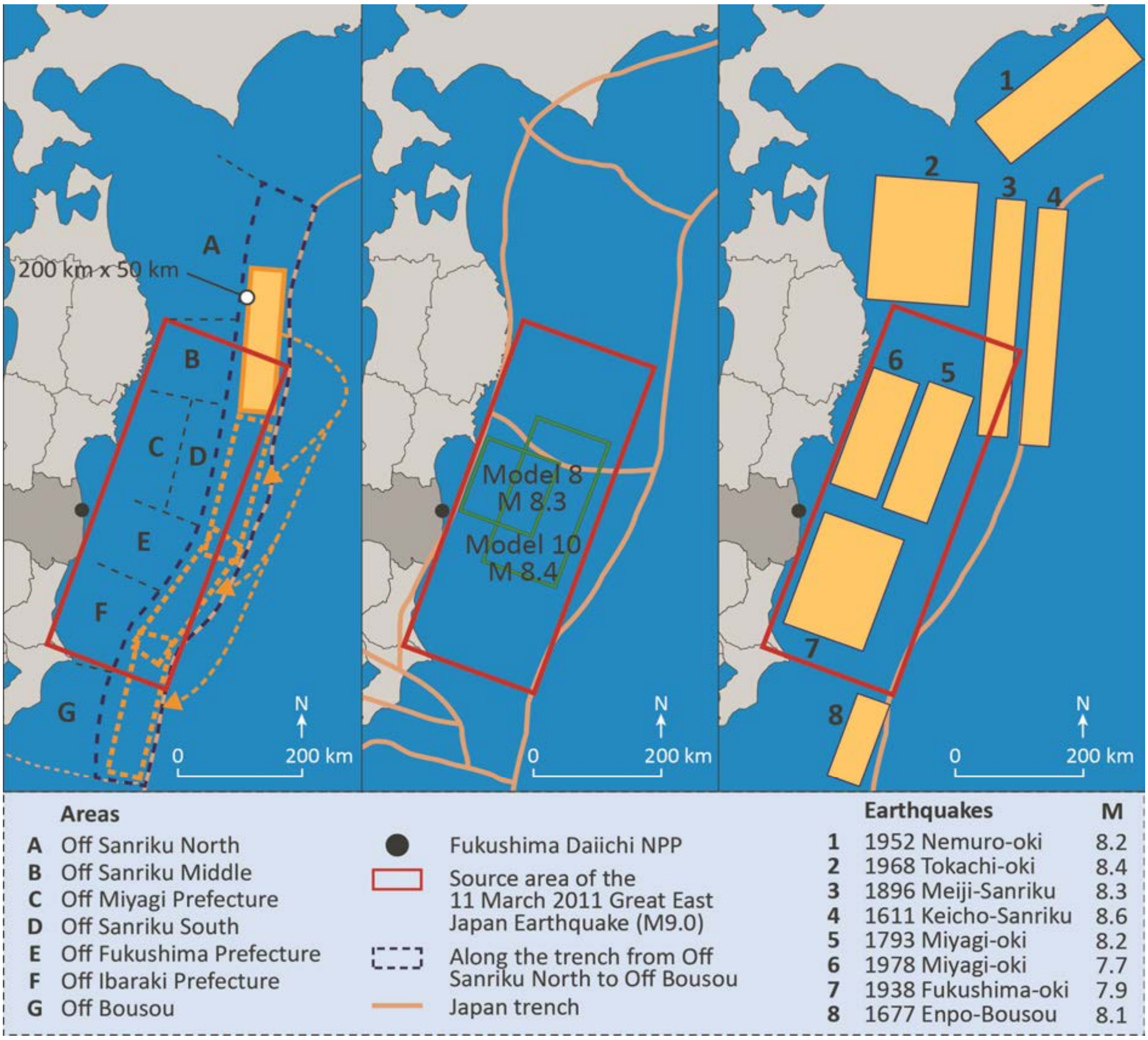


FIG. 2.1–15. Trial calculation performed by TEPCO in 2008. Comparison of source models (see Ref. [36] for the figure on the left, Ref. [35] for the figure in the middle, and Ref. [26] for the figure on the right).

2.1.5.7. Trial analysis based on the JSCE methodology with HERP source models

As stated above, this trial calculation was carried out using the model proposed by the Headquarters for Earthquake Research Promotion (HERP) with an M 8.2 earthquake rupturing an area of 200 km × 50 km, located in the subduction zone. Here, the basic difference with respect to previous re-evaluations by TEPCO applying the JSCE methodology is the assumption that an M 8.2 earthquake may occur in the offshore trench of the Japan subduction fault facing Fukushima Prefecture. This assumption was not considered previously because it was accepted that there was a seismic gap, reflecting that there was no record of a large, M 8 level earthquake off the coast of Fukushima Prefecture.

In 2007, the Central Disaster Management Council (CDMC) of the Cabinet Office discussed the proposal by HERP of the source model and concluded that a large, M 8 level earthquake along the

Japan Trench off the coast of Fukushima Prefecture should not be taken into account because it was unclear whether or not it would be feasible to assume the existence of a wave source in areas where no earthquake had previously occurred. It was decided to use only records available from a given historical time.

Because the HERP proposal did not specify a wave source model, TEPCO conservatively used the wave source model of the M 8.3 Meiji Sanriku earthquake of 1896. This earthquake occurred off the coast of Iwate Prefecture, more than 100 km north of Fukushima, and resulted in a tsunami of 38 m.

The results of this second trial calculation carried out by TEPCO in 2008 for the Fukushima Daiichi and Daini NPPs are shown in Fig. 2.1–16, as presented by TEPCO during the meeting with the IAEA of 10–14 February 2014 [33].

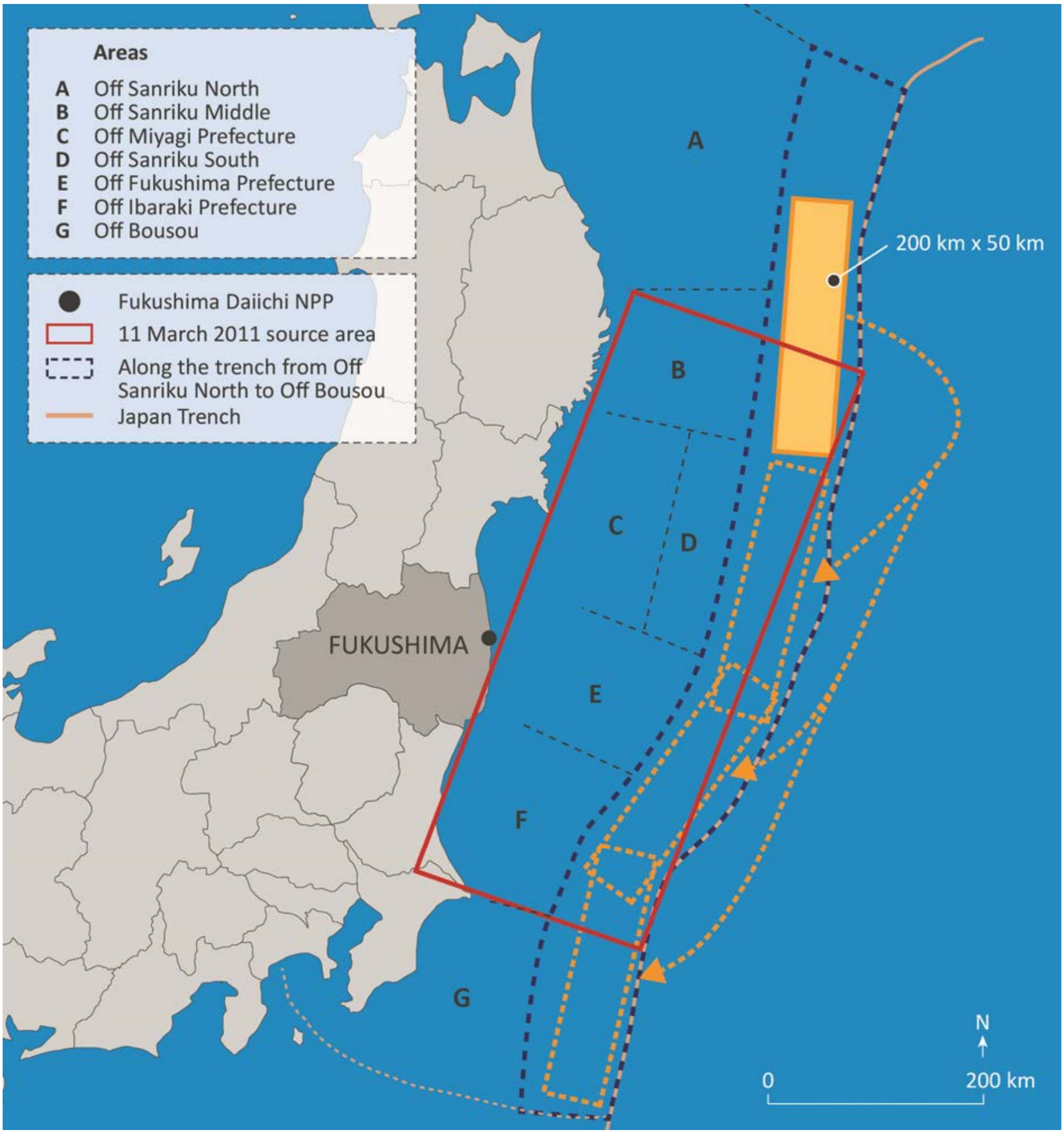


FIG. 2.1–16. Earthquake regions as determined by the Headquarters for Earthquake Research Promotion (HERP) for the Fukushima Daiichi and Daini sites as used by TEPCO in 2008 [36].

TABLE 2.1–7. RESULTS OF THE TRIAL CALCULATIONS FOR THE FUKUSHIMA DAIICHI NPP SITE

	Fukushima Daiichi							
Unit	1	2	3	4	5	6	Northern part (OP +13 m)	Southern part (OP +10 m)
Tsunami height (m)	8.7	9.3	8.4	8.4	10.2	10.2	13.7	15.7

TABLE 2.1–8. RESULTS OF THE TRIAL CALCULATIONS FOR THE FUKUSHIMA DAINI SITE

	Fukushima Daini				
Unit	1	2	3	4	(OP +12 m)
Tsunami height (m)	7.6	7.2	7.8	8.2	15.5 (southern part)

The results of the trial calculation presented in Tables 2.1–7 and 2.1–8 are summarized as follows:

- In front of Units 1–4, at the location of the seawater cooling pumps where a maximum level of OP +5.7 m was calculated in 2002, a maximum tsunami height of OP +9.3 m was estimated.
- In front of Units 5 and 6, the maximum tsunami height was estimated as OP +10.2 m.
- Other locations at the site, at the southern and northern parts, show maximum tsunami heights of OP +13.7 and 15.7 m, significantly higher than the previous re-evaluation values.

2.1.5.8. Trial analysis using Jogan 869 tsunami source models

Regarding the trial calculation with consideration of the Jogan 869 earthquake and tsunami, Fig. 2.1-17 and Table 2.1–9 shows the evolution of the knowledge and hypothesis between 1990 and 2008 about the location of the epicentre in the Japan offshore trench.

TEPCO conducted the trial calculation with magnitude 8.4 and with the assumed location of the tsunami source as proposed by Satake et al. [35]. This model is based on data obtained from soil deposits from the Jogan tsunami through paleo-tsunamigenic investigations. As indicated by TEPCO [33], Satake et al. [35] did not determine the Jogan tsunami source model because of the lack of deposit data of that tsunami and the need to conduct additional tsunami deposit survey investigations in Fukushima and Ibaraki Prefectures.

Five boreholes were dug along the coast in the vicinity of the Fukushima Daiichi and Daini NPPs to investigate the existence of evidence of the Jogan 869 tsunami. Three boreholes, located to the south of the site, yielded no evidence of tsunami deposits, while two boreholes located to the north of the site showed evidence of tsunami deposits at 0.5 m depth in one of them and between 3 m and 4 m depth in the other. TEPCO indicated that the results obtained from the deposit investigations showed some inconsistencies with respect to the trial calculation using the source model proposed by Satake et al. [35]. Therefore, additional investigations were proposed to be carried out.

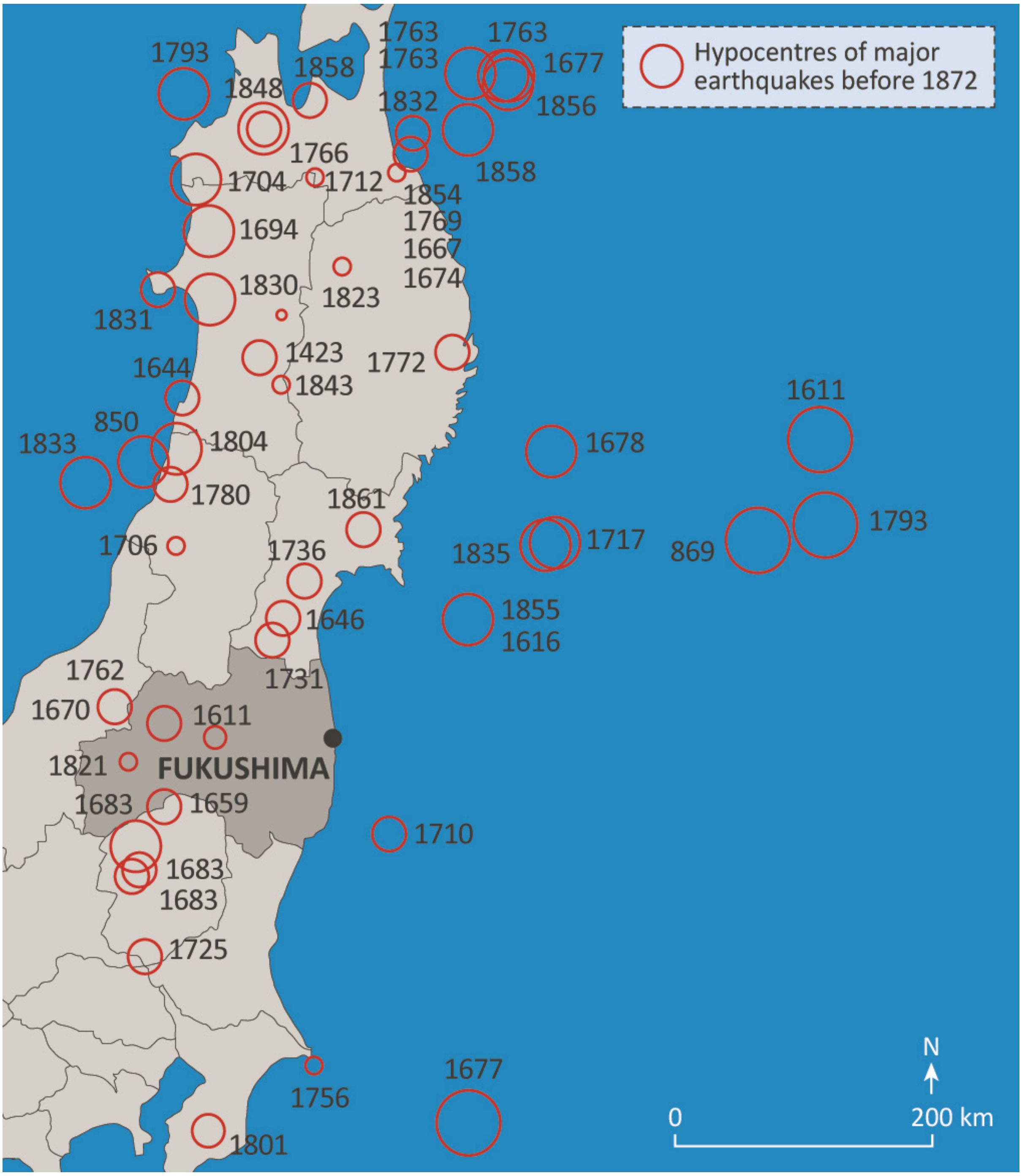


FIG. 2.1–17. Jogan 869 earthquake and tsunami potential locations [33].

TABLE 2.1–9. POTENTIAL LOCATIONS OF THE JOGAN 869 EARTHQUAKE AND TSUNAMI [33]

1990 Abe et al.	Tsunami source was assumed to be located off Sanriku Tsunami height in Sendai plane was smaller than 1611 Keicho Sanriku tsunami
2001 Minoura et al.	Tsunami source was assumed to be located off Miyagi Tsunami height in Fukushima coast was approximately 2–4 m
2006 AIST and Tohoku University	Tsunami deposit survey was carried out in Miyagi Prefecture at first and then in Fukushima Prefecture
2008 Satake et al.	Tsunami source was supposed to be located from off Miyagi Prefecture to off Fukushima Prefecture

Because information about the Jogan tsunami was limited and the source model was uncertain before 2005, the JSCE (2002) [26] did not take the Jogan tsunami into account.

The source model adopted by TEPCO for this calculation assumed an M 8.4 earthquake with a rupture area of 200 km × 100 km for Model 10 and an M 8.3 earthquake with a rupture area of 100 km × 100 km for Model 8 [33]. These were facing Miyagi and Fukushima Prefectures, offshore in the Japan subduction trench.

This trial calculation using the Jogan 869 tsunami resulted in maximum tsunami wave heights of OP +8.7–9.2 m for the six units of Fukushima Daiichi (Table 2.1–10) with no inundation in other northern and southern parts of the site at the grade levels of OP +13.00 m and OP +10.00 m, and tsunami wave heights of OP +7.8–8.00 m for the four units at Fukushima Daini and no inundation for the grade level OP +12.00 m (Table 2.1–11).

TABLE 2.1–10. TRIAL CALCULATION RESULTS FOR THE JOGAN 869 TSUNAMI FOR FUKUSHIMA DAIICHI [33]

	Fukushima Daiichi							
Unit	1	2	3	4	5	6	Northern part (OP +13 m)	Southern part (OP +10 m)
Tsunami height (m)	8.7	8.7	8.7	8.7	9.1	9.2	No inundation	No inundation

TABLE 2.1–11. TRIAL CALCULATION RESULTS FOR THE JOGAN 869 TSUNAMI FOR FUKUSHIMA DAINI [33]

	Fukushima Daini				
Unit	1	2	3	4	(OP +12 m)
Tsunami height (m)	8.0	7.8	7.8	7.9	No inundation

2.1.5.9. Conclusions by TEPCO on the trial calculations

In view of the results obtained from the trial calculations performed, TEPCO management considered that it was necessary to review the appropriateness of the tsunami source models. Thus, electric utilities requested that the JSCE review the suitability of the tsunami sources in 2009. In parallel, in August 2010, TEPCO constituted the Tsunami Measures Working Group as an internal body in the utility to conduct a full scale examination to study measures for reducing the impact of tsunamis.

Each of these trial calculations — performed between 2006 and 2009 — predicted maximum tsunami wave heights considerably greater than either the original design tsunami height of OP +3.122 m or the results of OP +5.7–6.1 m from the re-evaluations performed between 2002 and 2009 using the consensus JSCE methodology. The prediction to be properly considered in this case was for a runup that would reach and flood the main plant grade level at OP +10.00 m and +13.00 m if the HERP data were considered in the source model.

Another issue is the difference between the tsunami heights produced by the Great East Japan Earthquake at the Fukushima Daiichi and Daini NPP sites and for which studies presented by TEPCO show that the rupture that caused the Great East Japan Earthquake was so large that several trains of tsunami waves were generated from different locations along the rupture, as if they had been produced by multiple separate earthquakes. At Fukushima Daiichi, some trains arrived almost in phase, causing them to reinforce each other and produce a much bigger tsunami. The superposition of peaks did not occur at Fukushima Daini, leading to a lower wave height. The possibility of this phenomenon occurring had not been realized before the Great East Japan Earthquake and was not explicitly considered in the JSCE methodology. As shown by the trial analyses, the JSCE methodology was able to provide conservative predictions of tsunami heights if the correct assumptions were made about the source model and the magnitude of the earthquake in the offshore trench of the Japan fault in front of Fukushima Prefecture (the HERP source model).

2.1.5.10. Actions taken to cope with reassessed tsunami hazards

Actions taken by TEPCO

As a result of the reassessment processes of the flood level caused by a tsunami, which were carried out by TEPCO and other utilities in Japan with NPPs operating on the east coast, and which were triggered mainly by: (i) the issue of the 2002 JSCE methodology; (ii) the issue of the NSC Regulatory Guidelines of 2006 and (iii) the Niigata-Chuetsu-Oki earthquake in 2007 affecting the Kashiwazaki-Kariwa NPP, a number of countermeasures, through plant modifications or safety upgrades, were implemented at four NPPs, as summarized in Table 2.1-12.

In the case of the Fukushima Daiichi NPP, considering the fact that safety related items for removing the reactor decay heat (linked to the ultimate heat sink) and for cooling the emergency diesel generators (EDGs) for emergency power supply are located at the plant area at OP +4.00 m, which would be flooded in case of the newly reassessed tsunami flood level of OP +5.70–6.10 m, TEPCO mentioned during the IAEA Fact Finding mission in 2011 [6] that the motors of the safety related pumps (residual heat removal (RHR) system) were accordingly elevated to avoid disruption of function. However, no additional details were provided about the adequacy of these measures to cope with such an event for the protection of all related mechanical, electrical and instrumentation and control (I&C) components of the RHR system. Table 2.1–12 shows that TEPCO raised the pumps. Whether all the pumps or the pump motors were elevated was not clear.

Until 2009, TEPCO carried out actions in response to the newly calculated values for the tsunami flood level which were obtained using only the JSCE methodology. The results of the trial calculations carried out from 2008 using (i) the HERP assumptions of an M 8.3 earthquake in the Japan Trench offshore, facing the Fukushima Daiichi and Daini sites, and (ii) the model proposed by Satake et al. [32] for the Jogan 869 showed significantly higher values of the tsunami wave heights. This implied the need to cope with several more metres of flood, runup and inundation areas, including the main plant grade where the nuclear buildings are located. TEPCO did not implement interim corrective actions to develop protective measures for the plant to cope with such higher levels of flooding resulting from the trial calculations while conducting further examinations of the assessments, based on the following reasons:

- There was no historical record of a large, M 8 level earthquake off the coast of Fukushima.
- The JSCE methodology was the applicable standard developed with the consensus of all participating institutions in Japan, and it was also accepted and used by other utilities.
- Other institutions, not only the JSCE but also the Central Disaster Management Council (CDMC) and the Prefectural Governments in Ibaraki and Fukushima, did not consider the tsunami source located off the coast of Fukushima Daiichi and Daini NPP sites.
- An earthquake with a magnitude higher than 9 was not considered to be a credible event in the Japan Trench by the Japanese scientific community.
- Historical events such as the Jogan 869 earthquake and tsunami, which caused flood levels similar to the ones reached in March 2011, required additional investigation and collection of data for better knowledge of their causes and effects before applying them to assess those hazards for nuclear installation sites.

TABLE 2.1–12. SUMMARY OF THE RESULTS OF TSUNAMI REASSESSMENTS (2002–2009) AND PLANT MODIFICATIONS BY JAPANESE ELECTRICAL UTILITIES [33]

	TEPCO		JAPC	Tohoku Co.
Event	Fukushima Daiichi	Fukushima Daini	Tokai Daini	Onagawa
Ground level of main building	OP +10 or +13 m	OP +12 m	HP* +8.9 m	OP +14.8 m
Establishment Permit	Unit 1 in 1966 OP +3.122 m	Unit 1 in 1972 OP +3.122 m Units 3 and 4 in 1978 OP +705 m	— in 1971	Unit 1 in 1970 OP + 2–3 m (literature survey) Unit 2 in 1987 OP +9.1 m (numerical simulation)
JSCE method in 2002	OP +5.7 m (tsunami off the coast of Fukushima is dominant)	OP +5.2 m	TP* +4.88 m	OP +13.6 m (tsunami off the coast of Sanriku is dominant)
	Countermeasure such as raising the seawater pumps was completed	Countermeasure such as making the buildings watertight was completed	Countermeasure was unnecessary	Countermeasure was unnecessary
Scenario tsunami for disaster prevention was published by Ibaraki Prefectural Government	OP +4.7 m	OP +4.7 m	TP* +5.72 m	Unexplained
	Countermeasure was unnecessary	Countermeasure was unnecessary	Countermeasure such as raising the wall around seawater pumps was completed	
Scenario tsunami for disaster prevention was published by Fukushima Prefectural Government	Approximately OP +5 m	Approximately OP +5 m	Unexplained	Unexplained
	Countermeasure was unnecessary	Countermeasure was unnecessary		
Latest bathymetric and tidal data in 2009	OP +6.1 m	OP +5.0 m	Unexplained	Unexplained
	Countermeasure such as raising the seawater pumps was completed.	Countermeasure was unnecessary		
Tsunami in 2011	OP +13.1 m (tsunami height) OP +15.5 m (inundation height)	OP +9.1 m (tsunami height) OP +14.5 m (inundation height)	TP* +5.4 m	OP +13.8 m

* See water reference level for Fukushima Daini (HP: Hitachi Port; TP: Tokyo Peil).
Note: JAPC — Japan Atomic Power Company, Tohoku Co. — Tohoku Electric Power Company.

Actions taken by the regulatory authority

The reassessments of the tsunami hazards at the Fukushima Daiichi NPP site conducted by TEPCO during the operational period values were not requested by NISA (the regulatory authority at that time) according to the information collected at the time of the IAEA Fact Finding Expert Mission in May 2011 [6]. This was also confirmed by NRA (the new regulatory authority established in 2012) during the meetings held in Tokyo in January 2014 [7].

Regarding the actions taken by NISA as a response to the newly developed guidelines for tsunami assessment issued by the JSCE in 2002, NISA and TEPCO concurred that JSCE guidance was adequate for such a purpose. However, NISA did not issue a specific request to conduct a reassessment based on that methodology [6, 7].

The results of tsunami reassessments performed by TEPCO using the JSCE methodology were not formally submitted to NISA. Consequently, the results were not reviewed, commented, approved or rejected by the regulatory authority, although NISA was aware of their existence [10]. The fact that the tsunami estimate increased by a factor of almost two and the main tsunamigenic source shifted from the distant (Pacific subduction tectonic plate in Chile) to the near (Pacific subduction tectonic plate in the offshore Japan subduction trench Shioyazaki) source did not catch the attention of NISA for a long time (from 2002 until March 2011).

As these reassessments and countermeasures were undertaken by TEPCO voluntarily without any instruction from NISA, they did not lead to changes in the licensing documents, and thus the design bases remained as they were before. Regarding the physical measures taken by TEPCO at the Fukushima Daiichi NPP as a consequence of the reassessed higher tsunami wave heights, described in the previous section, and aimed at enhancing safety measures against tsunami flooding, NISA considered them as sufficient according to the Japanese reports [7].

In general, the actions taken by NISA in relation to reassessment of the external hazards were triggered mainly by the issue of the new NSC Regulatory Guidelines in 2006 [9]. NISA requested all NPP licensees by letter to undergo a backcheck on the basis of the new guidelines. This included the need to re-evaluate the tsunami hazards as an accompanying event, although it was not specifically requested in the letter. By the end of 2010, a few months before the accident, NISA received the reports for Units 3, 4 and 5, but they did not include a tsunami safety evaluation, which was planned to be performed by 2016 [7].

2.1.6. Extreme external events in a multi-unit site and multiple sites in the same region

The complexity of the events of the Great East Japan earthquake and tsunami stems from the fact that the natural external hazards impacted multiple units in the Fukushima Daiichi NPP. In addition, the four other NPPs along the coast were also affected to different degrees by the earthquake and the tsunami. However, all operating reactor units at these plants were safely shut down.

This meant that 14 reactor units at 4 sites were simultaneously exposed to high intensity natural external hazards (main shock and aftershocks, and tsunami warnings) during a long period of time in a geographical region that was simultaneously affected by those natural hazards and whose infrastructure was severely damaged.

Although the lessons learned from the Kashiwazaki-Kariwa experience in 2007 were very useful, particularly for on-site emergency measures at the Fukushima Daiichi NPP, and despite a significant number of lessons being properly implemented before 11 March 2011, there was still major regional disruption, which hampered immediate recovery actions at the plant. The nature of the destruction and the damage that occurred at the regional level at multiple units at multiple sites caused significant delays and disruptions in recovery actions for all sites and units.

Although many Member States have multi-unit sites, and in some cases have NPP units in a number of neighbouring sites that may be exposed to a large natural hazard simultaneously, guidance is lacking regarding how to deal with the safety of multi-reactor unit sites when affected by external hazards that occur in a continuous sequence of events.

The established general design criteria require that SSCs important to safety not be shared among nuclear reactor units unless it can be demonstrated that such sharing will not significantly impair the performance of the assigned safety functions, including, in the event of an accident in one unit, the orderly shutdown and cooldown of the remaining units. It should be questioned how these design criteria have been applied, how safety was assessed and how the successful performance criteria were demonstrated in the case of the numerous existing multiple unit sites when most of them were sharing the external electrical power grids, the switchyards and the ultimate heat sinks, and especially for the occurrence of external hazards as common cause failure events. It is known that most of the safety assessments performed consider shared SSCs to be an extra layer of redundancy because of the assumption that the accident occurs only in one unit while the remaining are kept safe.

The common cause nature of these extreme hazards plays an important role in off-site emergency preparedness and response because they affect the feasibility of implementing local, regional and national emergency plans. The situation described above should be carefully considered at the very beginning of an NPP project, from the stage of selection and evaluation of the site to the design and construction of the installation, to the final stage of its operation when all the response procedures and measures are duly established.

IAEA Safety Standards Series No. NS-R-3 [16] provides the following general requirement in para. 2.29:

> "The external zone for a proposed site shall be established with account taken of the potential for radiological consequences for people and the feasibility of implementing emergency plans, and of any external events or phenomena that may hinder their implementation".

The details of the implementation of the measures to fulfil such a requirement in all stages of the nuclear installation's life cycle is the challenge to be faced and resolved, since one of the lessons of the Fukushima Daiichi NPP accident was the occurrence of this complex scenario of extreme natural events affecting many reactor units in many sites located in the same region. This situation should also be analysed in relation to the need to comply with the concept of defence in depth. The question is how well this concept is being applied in such complex scenarios as those described earlier and in all stages of the nuclear installation life cycle.

2.1.7. Summary

During the pre-accident operational period, the site characteristics of the Fukushima Daiichi NPP were not reassessed in a systematic and comprehensive manner to consider all site related aspects and external events (i.e. seismic and geological hazards, meteorological and hydrological hazards, volcanic hazards and human induced hazards) as well as environmental issues. A regulatory framework for requiring such full reassessment of all site characteristics did not exist. Regarding specific external hazards, only seismic backchecking has been requested in Japan, following the new guidance on seismic safety that was developed and released by the Nuclear Safety Commission in 2006 [9] and as a result of the 2007 Niigata Chuetso-Oki earthquake that affected the Kashiwazaki-Kariwa NPP. But, specifically for tsunami hazards, this new guidance does not contain any concrete requirements, criteria or methodology that could be used for reassessment purposes, and includes only generic statements and no specific request for reassessment was issued.

The lack of a comprehensive regulatory framework for external hazards is one of the reasons that led to an underestimation of the tsunami hazards and to insufficient measures to cope with extreme external events. The Japanese approach was not in line with international and other national safety standards, resulting in significant discrepancies in the level of assessed hazard. No recommendations were expressed at the international level, since no international reviews had been requested. Even though predicting the height of a tsunami is difficult and subject to a variety of scientific and expert opinions, an international review team of independent experts assessing the Fukushima Daiichi NPP's level of protection against flooding would have recommended use of methodologies consistent with international safety standards. This emphasizes the importance of international cooperation on safety.

International independent peer reviews on site characteristics, external events and design are an effective means in assessing and enhancing the safety level of existing facilities.

The definition of the main plant grade level (i.e. at OP +10.00 m) at the time of the Establishment Permit was of great significance. The prevailing reason for the decision on the plant grade was the economics of the water cooling supply (i.e. installation cost during the construction stage and transport energy cost during the operational life of the installation) based on the assumption that the external flooding levels would not impose a risk according to recent historical records in this area. Thus, a site that was considered a dry site in the original design bases later became a wet site, when the resulting maximum flood levels of the reassessment performed showed values higher than the plant grade levels [10].

Regarding the seismic hazard assessment, the original ground motion design bases were evaluated mainly on the basis of historical seismicity data. During the process of obtaining permits for all the units, and particularly after the issuance of new NSC Regulatory Guidelines of 2006 [9], a methodology using also geomorphological fault dimensions was applied. However, this related specifically to on-land events and not to those generated at the Japan Trench, in the Pacific Ocean, which are the ones that generate tsunamis. The information regarding the on-land faults was taken from official sources, but conservative parameters were assumed for the analysis. Regarding the Japan Trench, the associated maximum magnitude was estimated to be about M 8 without much tectonic based justification, largely based on observed historical data and at the locations where they already occurred. Thus, due to the distance from the site, it was assumed that these offshore events would not be relevant for the seismic hazard at the site because closer on-land faults would constitute the main contributors to the seismic hazard. Although an approach similar to the on-land faults (deriving maximum magnitudes in relation to physical fault dimensions) was followed for the maximum magnitude estimation, M 8, of the Japanese Trench, the number of segments to be mobilized during a single event was underestimated.

The assessment of the maximum flooding levels caused by earthquake generated tsunamis, as done for the original design basis at the time of the issuance of the Establishment Permit in the late 1960s as well as in the reassessments carried out during the operational life of the plant, was underestimated in relation to the potential occurrence of extreme flooding events. The tsunami generated by the Great East Japan Earthquake reached values approximately 10 m higher than the originally estimated value (i.e. OP +3.122 m) and 7 m higher than the highest of the accepted re-estimated value (i.e. OP +6.10 m).

The evaluation by TEPCO of the tsunami flood level at the time of the Establishment Permit used the methodology and criteria prevalent in Japan at that time, which were based only on the study and interpretation of historical records of earthquakes and tsunamis. Although those records covered a period of some thousands of years, the distant tsunami that occurred in the Pacific subduction plate of Chile in 1960 was the event used for design purposes, and the OP +3.122 m level reached at Onahama Port was the adopted as the tsunami flood design level. For near sources located in the Japan Trench facing the east coast, a combination of the lack of historical records of tsunami flood levels at the specific location of the Fukushima Daiichi NPP site and the lack of evidence of the occurrence of earthquakes in the offshore area in front of the site were the basis for supporting this decision.

The Japanese approach, at least until 2006, of using mainly historical data of observed events that were available for a very recent period of a few decades or a few hundred years is the main reason for the underestimation of the earthquake magnitudes in assessing the tsunami hazards. The common international practice at the time of the original plant design was to use historical records when applying methods for estimating seismic and concomitant (e.g. tsunami) hazards. To compensate for the lack of pre-historical data commensurate with the low probability required (the usually accepted period of recurrence on the order of 10 000 years), this practice included the following assumptions:

(i) the rule of increasing the maximum historical recorded intensity or magnitude and (ii) to locate the seismic source closest to the site. This internationally recognized conservative and deterministic approach was also reflected in IAEA Safety Series No. 50-SG-S1 [11], in 1979, which was prepared and discussed according to international criteria applied in the 1970s.

In addition to the criterion to use pre-historical and historical data commensurate with the low annual frequency of occurrence of these extreme external events, the internationally recognized practice recommended the use of global analogues in order to cope with the lack of such pre-historical data. This is another important tool, particularly when an earthquake with M 9.5 (the largest in history) had occurred previously in the same tectonic environment of the Pacific tectonic plate. During the same decade as the site characterization of the Fukushima Daiichi NPP, two major earthquakes occurred on the Circum-Pacific Belt (on which the Japan Trench is also located). These were the Chile earthquake of 1960 with M 9.5 and the Alaska earthquake of 1964 with M 9.2.

Considering the above explanation, the maximum seismic magnitude for the Japan Trench could have been postulated to be M 9+ on the grounds of tectonic similarity.

The need to use pre-historical and historical data and the use of global analogues when data are missing in the area under investigation have been included worldwide since the 1970s in requirements, recommendations and practices for dealing with the assessment of extreme natural external events.

In spite of the lack of regulatory requirements for conducting a reassessment of the tsunami hazards, TEPCO had carried out a number of re-evaluations of the tsunami flood level at the Fukushima Daiichi NPP site since 2002, using numerical simulations. In all these reassessments, the tsunami wave heights were revised upwards, with increases from the original OP +3.122 m design flood level to OP +5.70 m in 2002, to OP +6.1 m in 2009 and to approximately OP +15.7 m in the trial calculations conducted in 2009.

Such reassessments were triggered mainly by the issue of the new guidelines for assessing tsunami hazards for NPP sites published by the JSCE in 2002 as a standard practice accepted by all nuclear utilities in Japan. The JSCE methodology used a standard source model for near tsunamis based on historical data in which no tsunamigenic source was assumed to occur offshore facing the Fukushima Daiichi and Daini NPP sites in the Japan Trench. That assumption was key to all evaluations performed using this standard practice.

Other reassessments were called ‘trial calculations’, and they were based on different assumptions, supported by other institutions or experts in Japan. One institution in Japan, HERP, has maintained that an M 8.2 earthquake should be considered elsewhere in the Japan Trench. When this position was applied in the trial calculation performed by TEPCO, the resulting tsunami flood level value was very similar to the flood level which occurred in March 2011, and was much higher than those obtained using the standard practice.

Therefore, if either a conservative approach had been followed in Japan as the one applied at the time of the original design and construction, or if global analogues had been used due to lack of specific pre-historical data, the associated generated tsunami height would have been close to the height calculated in the trial calculations.

In summary, assessments using a conservative approach, based on all relevant, domestically and internationally available data, yielded predictions of tsunamis heights close to the level reached during the March 2011 accident.

The remedial actions adopted by TEPCO to cope with the new situation, derived from higher tsunami water heights resulting from the reassessments performed, were only commensurate with the values obtained by using the JSCE 2002 methodology. The design and implementation of protective measures to cope with higher values than the ones corresponding to the trial calculations, including all associated hazardous phenomena such as hydrodynamic forces and debris impact, remained pending the outcome of the additional studies, investigations and confirmations that were to be carried out.

As a country known for its high level of awareness of natural disasters, Japan has an excellent system of warning, preparedness and response to cope with these external events. However, the magnitude of the 2011 natural disaster was not anticipated in any of the reassessments performed for the NPPs located in the affected region and, consequently, a scenario of extreme natural events affecting the whole region was not included in the design and operation bases of these installations.

Moreover, contingency plans for the failure of multiple units at multiple sites within a regional disaster context were not available. The effective mitigation of common cause mode failures affecting multi-unit plants simultaneously required a large amount of resources in terms of trained experienced people, equipment, supplies and external support. The timely provision of those resources for carrying out the recovery actions was disturbed by the disruption produced at the site and off-site at the regional level due to the earthquake and tsunami.

2.1.8. Observations and lessons

— **The safety of nuclear installations, in general, and the site related aspects, in particular, needs to be reassessed during their operational life in response to new knowledge, new hazards, new regulations and new practices, as part of periodic safety reviews. In this regard, the role of national and/or international independent peer reviews needs to be emphasized as an important tool to assess and enhance safety.**
The requirement for a reassessment process of site related aspects needs to be included in the regulatory framework, and the responsible organization needs to implement the plant safety improvements in a timely manner based on the results of this process. This needs to cover, in a systematic and comprehensive manner, all natural and human induced hazards which may create or exert potential effects on nuclear installation safety, as well as the impact on the environment.
The reassessment process needs to be performed in accordance with periodic safety reviews and international safety standards and recognized engineering practice. In this regard, international peer review missions are key elements for assessing and enhancing safety with another layer of effective actions which may contribute to cope with the lack of timely actions or responses by the responsible organizations and/or the regulatory bodies.

— **National and international standards to cope with external events in siting, site evaluation and design aspects need to be periodically updated and revised in accordance with scientific and technical developments, recognized engineering practices as well as using information from experience of recently occurred extreme natural external events.**
The experience and data obtained during the 11 March 2011 earthquake and tsunami in Japan will be useful in the revision of national regulations in the effort: (i) to bring them in line with modern criteria and methodologies; and (ii) to be able to cope better with the involved uncertainties in the assessment of these extreme natural hazards. Regulatory documents need to ensure that databases take into account pre-historical and historical events commensurate with the low annual frequency of occurrence of the extreme natural phenomena in line with the relevant IAEA safety standards. It has been demonstrated that one reason for the underestimation of the 11 March 2011 tsunami was that only Japanese historical data were taken into account in the evaluations as well as in the use of methodologies applied on the basis of an incorrect consensus approach. Since: (i) the magnitudes of all historical earthquakes were smaller than 9; (ii) the historical earthquake magnitudes and/or intensities were not increased as conservatively done in international deterministic practice; and (iii) none were located in the offshore region facing Fukushima, the earthquake and subsequent tsunami hazards were underestimated. Evaluations using standard practice underestimated the tsunami height that might occur, as happened in March 2011. At the

same time, some experts and institutions using alternative approaches based on the source model proposed by HERP determined tsunami flood levels comparable to the 2011 ones in the Fukushima area. These discrepancies between different expert opinions need to be properly treated, since all of them might contribute to reducing the uncertainties that exist in assessing extreme natural events. Therefore, the use of mainly national historical data is not sufficient to characterize the risk of extreme natural hazards, as highlighted by IAEA safety standards since 2003. The prediction of extreme natural hazards often remains difficult and controversial. Natural hazards assessment, as well as reassessments, should be performed in a conservative way and be updated according to new knowledge, as soon as available.

— **Assumptions of complex scenarios need to be made and adequate conservative estimations need to be applied at the site evaluation, design and different operational stages in relation to the potential occurrence of extreme external events of very low frequency but with high safety consequences. When operating nuclear installations are faced with revised estimates that exceed previous predictions, it is important to take interim corrective actions in a timely manner by the responsible organization and the regulatory authority.**

The consideration of uncertainties involved in the knowledge and determination of the loads on SSCs during the operational life of the installation requires the assumption of complex scenarios in a comprehensive manner from the beginning of the process.

Correspondingly, an appropriate regulatory framework has to be in force and in line with the identified needs to be able to request, control, regulate and provide guidance on the acceptable level of risk and the performance criteria that the installation has to comply with to safely cope with assumptions of extreme external events during the operational life of the plant.

In the case of the Fukushima Daiichi NPP, it has been demonstrated that interim corrective measures were not taken in a timely manner.

— **The assessment of natural hazards needs to be sufficiently conservative. Particularly in relation to the assessment of tsunami hazards, there is a need to use highly conservative assumptions for estimating the tsunami heights (maximum and minimum), runup and other site associated phenomena. They should be based on pre-historical and historical specific data commensurate with the low annual frequency of their occurrence and, if such specific data are not fully available, using appropriate global analogues.**

The consideration of mainly historical data in the establishment of the design basis 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.

Regarding the need to apply a more conservative approach for tsunami hazards than those used for other external natural hazards, the main reasons are as follows:

- Large aleatory and epistemic uncertainties in parameters involved in tsunami hazard calculations, particularly in the characterization of the tsunamigenic sources;
- Significant variations in inundation levels at different parts of the site considering the specific and detailed plant layout and the elevations of different plant sectors;
- Difficulties in incorporating effective tsunami protection measures for operating plants after an increase in tsunami height estimation resulting from periodic reassessments;
- Inability of SSCs at nuclear plants to cope with increased flood heights with respect to the design levels, with possible flood related cliff edge effects seriously affecting the safety of the nuclear installation.

— **Regarding uncertainties in tsunami hazard calculations, special attention needs to be paid to the aleatory and epistemic uncertainties associated with the maximum magnitude earthquake related to tsunamigenic sources such as major subduction zones.**

In general, the assessments of the magnitude of historical tsunamigenic earthquakes contain large uncertainties because they are inferred from damage caused on land, sometimes at a distance of more than 100 km, as well as on tsunamis that also heavily depend on bathymetry and coastal topography. For these reasons, a higher degree of conservatism may be necessary in the estimation of maximum magnitudes for tsunamigenic seismic sources at the time of the original design in order to avoid onerous physical upgrades later on during the design, construction or operational stages, or when such hazards are reassessed.

While the prevailing view among Japanese scientists before the earthquake of 11 March 2011 was that an M 9 earthquake could not be generated by the Japan Trench, as it has been for this Pacific tectonic plate in the past (in Chile and Alaska), it is important that diverse expert opinions from recognized scientific or academic institutions (both nationally and internationally) be considered to account for the epistemic uncertainties for assessing natural hazards.

— **There is a need to use a systemic approach in dealing with the design and layout of SSCs for effective protection against flooding hazards.**

The dry site concept is considered to be a crucial element for coping successfully with flooding hazards, and it has to be formulated from the beginning of the NPP project. It needs to be periodically reassessed and maintained, and, if conditions for a dry site change, adequate protective measures need to be taken in a timely manner.

The selection of the main plant grade level during the first stages of the NPP project is a critical aspect which needs to receive careful consideration due to its importance for the dry site concept. Leaktightness and water resistance also have to be ensured through a comprehensive evaluation of all potential waterways, although this measure can only be used as a redundancy, i.e. in conjunction with a dry site or an effective site protection measure. Thus, the main plant grade level has to be determined with sufficiently large safety margins to avoid flooding hazards due to cliff edge effects.

On the other hand, for those plant design aspects which may be seriously affected by external flooding but for which major uncertainties or insufficient knowledge exist, larger conservatisms have to be applied with respect to other site related aspects and external events for which those issues are better controlled. The same is true for those aspects with more difficulties and complexities in executing effective upgrades, or with higher consequences in case of failure that may affect the defence in depth concept.

— **There is a need to act effectively and promptly in implementing upgrading measures to maintain the defence in depth concept of an installation and to ensure the performance of safety functions when an original dry site becomes a wet site during its operational life as result of a reassessment of the flooding hazards at the site (i.e. having a potential for higher flood levels than the main plant grade level).**

Attention has to be paid to the fact that the upgrading measures to protect an operational installation that is now located on a wet site, and the closing of all possible waterways, may be practically more difficult to implement for an existing facility than for a new site, where such measures would form part of its original design and construction.

In the case of indications of evidence of greater hazards than those originally predicted in the design bases, the responsible organizations have to react effectively and promptly, and ensure safety through the implementation of interim measures while final confirmation of such evidence is obtained.

— **Complex scenarios involving consequential or independent occurrences of multiple external hazards affecting multiple units located on a site and, possibly, multiple NPPs at different sites in the same region need to be considered in accident scenarios and actions to be taken.**

Due to the nature of the Fukushima Daiichi accident, the lessons learned will cover a very wide area, which involves a wide variety of findings. Traditional engineering thinking for verifying the adequacy of a design involves characterizing limit states and comparing the effects of the loads on the installation (the demand) with the strength (the capacity) of the installation.

However, the greatest uncertainties in this process are with the definition of the acting loads, i.e. with the demand imposed on the installation. For this reason, design loads are defined to cover credible, possible and likely situations. In this sense, the design loads need to cover the potential of the occurrence of extreme events in the future. They have to be properly and conservatively estimated by the designer.

The designer, in the final design process, may or may not derive the proper design basis criteria with due account taken of complex scenarios of either extreme or severe natural hazards and with enough conservatism to comply with the defence in depth concept and to ensure adequate safety margins.

The potential for complex scenarios involving multiple external hazards that affect multiple units at the same site and at the regional scale, and possibly multiple sites in the region, needs to be comprehensively considered in the accident scenarios and measures to be taken. If such scenarios cannot be screened out, provisions need to be made for plant layout, site protection measures, design of shared and non-shared SSCs, accident management and off-site emergency preparedness and response in order to protect the plants from natural disasters in an environment where serious disruptions of normal life and infrastructure may occur affecting communications, transportation and utilities (water, electricity, gas, sewage), and logistics, human resources and supplies.

— **Clear procedures establishing measures to be taken before, during and after a tsunami, in particular, and for any external event, in general, adopted for design bases need to be prepared, implemented and exercised during the operation of the nuclear installation.**

For well defined tsunamigenic (fault controlled) sources, a large earthquake will always precede the tsunami and, consequently, if the source is located near the site, the vibratory ground motion will provide a warning. If the source is located at a large distance from the site, warnings from international and national tsunami notification centres are available. For all types of tsunami that may occur at the site, notification from the national and/or international tsunami warning system needs to be transmitted to the control room for immediate operator actions. In addition, a clear procedure needs to be followed by plant management in preparing for a possible tsunami until the warning is lifted. It is also important to coordinate post-earthquake procedures with those of the tsunami response, as an imminent tsunami would likely affect the possible inspections related to post-earthquake actions.

Moreover, as a consequence of a major natural disaster, a severe disruption to the plant may have occurred, and the plant state (with degraded systems and degraded physical conditions of the SSCs) may have lost robustness and may have degraded the defence in depth conditions with respect to the design condition level. The safety profile of the plant needs to be well understood (i.e. the SSCs required for fulfilling the fundamental safety functions) for different plant states (e.g. shutdown), to ensure consistent protection of the plant in case of the occurrence of consequential and/or independent natural events (e.g. aftershocks following the main extreme earthquake or other natural events, such as strong winds) which may be generated during an extended period when plant recovery and upgrading actions are being taken.

2.2. ASSESSMENT OF THE FAILURE TO MAINTAIN FUNDAMENTAL SAFETY FUNCTIONS

2.2.1. Introduction

This section assesses, on a reactor by reactor basis, what led to the failure to maintain the fundamental safety functions in Units 1–3 of the Fukushima Daiichi NPP. Units 4-6 are also assessed, taking into consideration their shut down at the time of the initiating event. The focus of this section is on a systems based review, leaving the assessment of other factors to subsequent sections in Technical Volume 2. This review was conducted using as a basis the IAEA Safety Requirements publication, IAEA Safety Standards Series No. NS-R-1, Safety of Nuclear Power Plants: Design [2][6], which was in effect at the time of the accident. The review was supplemented by reference to relevant Safety Guides as appropriate. Since IAEA safety standards require that a design basis be established, it was instructive to begin this assessment with a review of the design basis of the boiling water reactors (BWRs) in operation at Fukushima Daiichi NPP because this will help to identify why certain equipment was unavailable.

[6] It has, in the meantime, been superseded by IAEA Safety Standards Series No. SSR-2/1, published in 2012 [17].

Much of the discussion presented in this section provides a review and assessment of the conditions experienced at Fukushima Daiichi NPP during the progression of the accident in the units relative to the capabilities of the plants as set out in the design basis. The BWRs were designed in the USA by the General Electric Corporation based on the General Design Criteria in 10 CFR Part 50 [37]. These design criteria were formulated as NPPs were first developed in order to provide structure and organization to the design process and to ensure essential quality processes were included. The primary purpose of the General Design Criteria is to ensure that essential equipment, referred to as safety related SSCs, remain functional to protect people and the environment from the effects of postulated design basis accidents (DBAs). As BWR technology was introduced into Japan in the 1960s, regulations related to design were developed so that the reactors at the Fukushima Daiichi site complied with Japanese and not US requirements.

Severe accidents involving significant fuel damage are not included in the set of DBAs which, by definition, the plant is designed to handle. This statement is true for both Japanese and IAEA requirements in effect at the time of the accident. Severe accidents were at one time considered hypothetical; it was not until probabilistic studies such as the Reactor Safety Study (WASH-1400, 1975) [38] identified them as potential contributors to public risk that severe accidents came under consideration. The accident at the Fukushima Daiichi NPP was a beyond design basis accident (BDBA), known as a station blackout (SBO), which refers to the complete loss of off-site and on-site AC power. Further complicating the situation was the near complete loss of DC power.

The risk from an SBO was not fully appreciated until it was identified by safety research programmes in the 1970s as a significant contributor to the core damage frequency (CDF). An example of this work is NUREG-1150 [39], completed as a draft in 1987, which identified an SBO as often approaching 50% of the CDF for some plants. Following NUREG-1150 [36], regulatory authorities in Member States began to develop requirements to address this newly identified risk. In general, these regulatory actions did not contemplate the complete loss of both AC and DC power because they adopted a ‘coping time’ approach in which NPPs needed to demonstrate that they could cope with the loss of AC power. Typically, these coping times were based on expected station battery lifetimes of between 4 and 8 hours.

To a large extent, the assessments in this section focus on the performance of the relevant SSCs as compared with the requirements in IAEA Safety Standards Series No. NS-R-1 [2] related to the design basis because the requirements in NS-R-1 are prescriptive in nature and an objective assessment can be performed. However, this approach is governed by the range of postulated initiating events (PIEs) included in the design basis and the assumption that AC and DC power and vital auxiliaries such as high pressure air are available. In contrast, the requirements related to beyond design basis events typically use language such as ‘adequate consideration needs to be given’ to design requirements, leaving what constitutes ‘adequate’ to the implementation by individual Member States [8, 28]. Therefore, in order to properly assess the performance of equipment following a BDBA, one needs to consider more than its design, including requirements such as procedures and guidance to recover equipment (referred to as severe accident management guidelines or SAMGs) or the use of alternative means to restore the safety function. This assessment is conducted in Section 2.4.

2.2.2. Electrical power supply systems

2.2.2.1. Off-site power supply

The Fukushima Daiichi off-site power supply system consisted of seven transmission lines in total. Units 1–4 were connected to the four Okuma lines (275 kV) from the Shin-Fukushima substation, and the TEPCO nuclear power line (66 kV) from the Tomioka substation (normally not used). Units 5 and 6 were connected to the two Yonomori lines (66 kV) from the Shin-Fukushima substation for the off-site power supply (these lines could not be connected to the generators of these units) (see Fig. 1.2-17 of Technical Volume 1).

Between the Shin-Fukushima substation and Fukushima Daiichi, pylons were shared by Okuma lines 1 and 2 (27 pylons) and by Okuma lines 3 and 4, together with the two Yonomori lines (22 pylons). Within the site premises, Okuma lines 1 and 2 led to the switchyard of Units 1 and 2, Okuma lines 3 and 4 led to the switchyard of Units 3 and 4, and the two Yonomori lines led to the switchyard of Units 5 and 6 (six additional pylons).

All seven lines were connected via transformers to different 6.9 kV busbars which could be interconnected between Units 1–4 and between Units 5 and 6. The switching facility for Okuma line 3 in the switchyard for Units 3 and 4 was under maintenance when the earthquake occurred. Thus, this line was not available for the off-site power supply.

In terms of seismic standards, equipment for receiving and transforming power from the transmission lines was categorized as Class C equipment designed according to the corresponding general industry guide (Seismic Design Guide for Electrical Facilities at Substations and Other Facilities, JEAG 5003-2010 [40]).

System response during the accident

After the earthquake on 11 March 2011, the off-site power supply to Fukushima Daiichi was completely lost. Okuma line 1 became unavailable due to the opening of a breaker in the switchyard for Units 1 and 2 (presumably triggered by earthquake induced damage to another breaker in the same switchyard). Okuma line 2 became unavailable due to the opening of yet another breaker in the Shin-Fukushima substation (presumably triggered by earthquake induced damage to a related breaker in the switchyard of Units 1 and 2). Okuma line 3 became unavailable due to the opening of a breaker in the Shin-Fukushima substation (presumably triggered by earthquake induced arcs at a pylon of the transmission line; in addition, a ground wire at the Shin-Fukushima substation was disconnected). Okuma line 4 became unavailable due to the opening of another breaker in the Shin-Fukushima substation (presumably triggered by earthquake induced arcs at another pylon of the transmission line). In addition, there was a tilting of the steel structures leading the Okuma lines 3 and 4 into the Shin-Fukushima substation. The Yonomori lines 1 and 2 became unavailable due to the opening of two related breakers in the Shin-Fukushima substation. In addition, an on-site pylon of the Yonomori lines collapsed. The TEPCO nuclear power line was also not available because the power transmission from the Tomioka substation was interrupted due to the occurrence of high voltage discharge in the power lines.

Due to the tsunami, the switchyards for Units 3 and 4 and Units 5 and 6, as well as the six startup transformers for Units 1–6 were exposed to water. In addition, the underground cable of the power line between the metal clad switch gear in the on-site auxiliary substation and the turbine hall of Unit 1 became unavailable due to exposure to water.

Assessment

IAEA Safety Standards Series No. NS-R-1 [2] states that:

> "5.67. In the design of the plant, account shall be taken of power grid–plant interactions, including the independence of and number of power supply lines to the plant, in relation to the necessary reliability of the power supply to plant systems important to safety."

This requirement was formally complied with in the design of the Fukushima Daiichi NPP, which featured redundant off-site power supply lines, but with a low degree of independence.

The off-site power supply to the Fukushima Daiichi NPP was designed against possible failures affecting only a single line by having multiple lines from two different substations (in the case of Units 1–4), leading to three different switchyards as well as options for power transfer between Units 1–4 and between Units 5 and 6. However, the resilience against common cause failures of more than one line was lower due to six lines leading to the same substation, shared pylons for up to four lines, and shared switchyards for pairs of lines. For example, the collapse of a single pylon due to the earthquake on 11 March caused the loss of both Yonomori lines, resulting in the complete loss of off-site power for Units 5 and 6.

The off-site power supply of NPPs can be strengthened not only by sufficient redundancy (e.g. multiple lines), but also by sufficient diversity, for example by being able to connect each unit to different lines which use different pylons or underground cables, leading to different substations of different subgrids. In addition, the overall vulnerability against external events can be reduced, for example by strengthening the seismic design of the transmission lines and the associated switching devices.

A potential loss of off-site power was taken into account in the plant design — as is common practice — by the provision of emergency diesel generators (EDGs) to provide backup power supply, as discussed in Section 2.2.2.2. However, it is desirable to keep the probability for such an event reasonably low.

2.2.2.2. Emergency diesel generators

In total, 13 EDGs were installed at the Fukushima Daiichi NPP. Each unit was equipped with two redundant EDGs; Unit 6 had an extra EDG dedicated to the high pressure core spray (HPCS) system.

All six EDGs of Units 1, 3 and 5 were located in the basement of the respective turbine buildings. This was also the case for EDG 2A of Unit 2 and EDG 4A of Unit 4. The other two EDGs of Units 2 and 4 (EDG 2B and EDG 4B) were placed in the ground floor of a common auxiliary building. In Unit 6, two EDGs (EDG 6A and EDG HPCS) were located in the basement of the reactor building annex and one (EDG 6B) on the ground floor of a separate EDG building.

Most of the EDGs were water cooled, using the sea as ultimate heat sink, whereas three EDGs (EDG 2B, EDG 4B and EDG 6B), installed in the 1990s as part of severe accident management (SAM) modifications, were cooled by air (all being located on the ground floor, rather than the basement of the respective buildings). Air cooling was selected for EDG 2B, 4B and 6B, as the seawater cooling system used for the water cooled EDGs was at its rated capacity. The EDGs were designed to withstand design basis seismic ground motion (Class S equipment) [10].

System response during the accident

When the earthquake occurred, EDG 4A (Unit 4) was not available due to periodic inspection. As designed, all 12 available EDGs started automatically after the complete loss of off-site power following the earthquake and provided AC power to the emergency busbars.

Due to the subsequent tsunami and flooding, the functions of all but one of the EDGs were lost (EDG 6B). The six EDGs of Units 1 to 4, which were located on the basement level of the turbine buildings, were inundated and became inoperable. The two EDGs located at higher elevation in the common auxiliary building were not flooded, but became unavailable for emergency power supply due to the flooding of the associated switchboards in the basement of the building.

Assessment

IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.89. The combined means to provide emergency power (such as by means of water, steam or gas turbine, diesel engines or batteries) shall have a reliability and form that are consistent with all the requirements of the safety systems to be supplied, and shall perform their functions on the assumption of a single failure. It shall be possible to test the functional capability of the emergency power supply.
>
> ....
>
> "2.7. …The safety analysis examines: (1) all planned normal operational modes of the plant; (2) plant performance in anticipated operational occurrences; (3) design basis accidents; and (4) event sequences that may lead to a severe accident. On the basis of this analysis, the robustness of the engineering design in withstanding postulated initiating events and accidents can be established, the effectiveness of the safety systems and safety related items or systems can be demonstrated, and requirements for emergency response can be established."

The earthquake apparently did not affect the functionality of the EDGs. Hence, there is no indication of any weakness in their seismic design. However, as manifested by the failures after the tsunami, the design was not robust against a flooding of the site. Although there was some difference in the locations and cooling mechanisms with the effect that several EDGs were not directly affected by the tsunami, the emergency power supply as a whole was not consistently designed against flooding, as the switchboards for the two air cooled EDGs of Units 2 and 4 (at higher elevation) were located in the basement, where they were subject to flooding damage.

The design of the emergency power supply had taken into account a possible single failure by including two redundant EDGs for each unit. Redundancy criteria were supplemented with some of the elements needed for diversity (e.g. different cooling systems and different equipment elevations). However, this was not done in a systematic way and not supported by a comprehensive common cause failure evaluation, which could have identified the vulnerability in case of a flooding of the site.

As a conclusion of the assessment, it could be deduced that the IAEA safety requirements in effect at the time of the accident were formally complied with in the design of these systems, but the protection against common cause failures was not adequately evaluated. It is observed that it is important to consider, in the plant design, phenomena with a low probability of occurrence but with the potential to produce severe consequences for plant safety, even for situations beyond the design basis.

2.2.2.3. DC power systems

The DC power supply is vital for plant safety, as it is needed for instrumentation and control and provides AC power from inverters to a small number of essential components. Each Fukushima Daiichi unit featured 125 V DC busbars backed by batteries. DC panels and the batteries of Units 1, 2 and 4 were located on the basement floor of the turbine building. For Unit 3, DC panels and batteries were located on the mezzanine basement floor of the turbine building. For Units 5 and 6, the DC panels were located on the mezzanine basement floor of the turbine building and the batteries were located on the basement floor. The batteries were designed to cope for at least 8–10 hours with a total loss of AC power [41].

System response during the accident

No earthquake related effects on the battery backed DC power supply were reported. However, in Units 1, 2 and 4, the DC power installations were flooded by the tsunami, leading to the loss of the battery backed DC power supply. The DC power supplies in Units 3, 5 and 6 were not affected by the tsunami due to their location at higher elevations, i.e. the mezzanine area. In Unit 3, the batteries were not flooded and supplied power until they were completely depleted.

Assessment

With regard to postulated initiating events (PIEs), IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.88 After certain PIEs, various systems and components important to safety will need emergency power. It shall be ensured that the emergency power supply is able to supply the necessary power in any operational state or in a design basis accident, on the assumption of the coincidental loss of off-site power. The need for power will vary with the nature of the PIE, and the nature of the safety duty to be performed will be reflected in the choice of means for each duty; in respect of number, availability, duration, capacity and continuity, for example."

In referring to emergency power systems (EPSs), IAEA Safety Standards Series No. NS-G-1.8 [42] states:

> "2.6. …The EPSs should be designed for high functional reliability and testability and to have the capability to carry out their safety functions. Their design requirements, form and layout should be consistent with all the requirements for the safety systems to be supplied with power".

This Safety Guide also states:

> "4.15. …The DC power system should be divided into redundant divisions .... Each division should consist of at least a battery, a battery charger and a distribution system."

The DC power supply design was robust against single failures, e.g. by having multiple DC busbars per unit. However, the physical layout of the associated switch gear and the batteries made the system vulnerable to common cause failure by flooding. There was neither sufficient diversity in the location of the system's facilities nor adequate protection against flooding (e.g. water-tight compartments for essential components).

For situations in which only DC power backed by batteries is available for a prolonged time, as was the case with Unit 3, high capacity batteries and the capability to replace or recharge them with mobile equipment would be beneficial.

Most of the problems in the event arose from the total unavailability of electrical power, a scenario which had not been anticipated in the plant design. This situation is potentially shared by most of today's operational reactors, as consideration of the total loss of electrical power has not been a design requirement. It is important to implement short term and long term solutions for this challenge in the future at existing NPPs. The design and layout of electrical equipment should be robust enough to avoid the possibility of common cause failures from credible beyond design basis events.

2.2.2.4. On-site electrical power distribution systems

The electrical power distribution system of each unit of the Fukushima Daiichi NPP consisted of a set of buses for different loads and various interconnections. On a generic level, these systems were the same at each unit. During normal power operation, the power required by a unit was provided via one or two house transformers from the generator to two normal buses (6.9 kV) which fed high power loads (e.g. the feedwater pumps), as well as two emergency buses at 6.9 kV which provided power to essential safety systems (e.g. the shutdown cooling pumps). Each of the four 6.9 kV buses were connected to 480 V (460 V at Unit 1) low voltage buses feeding power via separate transformers to smaller pumps and to 125 V DC buses via rectifiers backed by batteries and featuring additional inter-unit connectors [10].

In the absence of electrical power from the generator, that is, when the reactor is shut down, the required power is provided by the external grid. The corresponding transmission lines feed the common buses (6.9 kV) via a startup transformer. These common buses can be connected to other common buses, to normal buses of adjacent units and via transformers to low voltage buses of adjacent units. If off-site power supply was unavailable at the Fukushima Daiichi NPP, the EDGs supplied power to the 6.9 kV emergency buses.

When the earthquake struck the plant, Units 1–3 were in normal power operation, feeding power from the generator to the grid via the Okuma lines 1–3 and to their normal buses via their respective house transformers. The common buses associated with Units 1 and 2 were fed via the respective startup transformers. The circuit breakers and other devices for receiving power from the Okuma line 3 at the switchyard of Units 3 and 4 were not available due to maintenance. Therefore, the common bus of Unit 3 was connected to the common bus of Unit 2 rather than being fed by its startup transformer.

Units 4–6 were shut down for maintenance. They were receiving power to their common buses via their startup transformers from the Okuma line 4 and the Yonomori lines 1 and 2, respectively. The circuit breakers and other devices transmitting power to the Okuma line 4 at the switchyard of Units 3 and 4 were not available due to maintenance. Some buses at Unit 4 were not available due to maintenance of EDG 4A.

System response during the accident

There is no indication that the earthquake affected the function of the electrical power distribution systems in the different units of the Fukushima Daiichi NPP. Later on, after the tsunami, all the metal clad switch gear (M/C) for the 6.9 kV electrical power distribution installed in Units 1–6, except for three M/C in the Unit 6 reactor building, were exposed to water, leading to functional damage. Thus, in Units 1–5, both ordinary and emergency busbars became completely unavailable and could not have supplied power to the necessary equipment, even if off-site power supply or EDGs would have been available. Most of the 480 V power centres were also damaged by water (including all on the lowest basement floors and some power panels on the semi-basement floors), limiting the number of locations where portable high voltage power supplies could be connected during restoration efforts.

Assessment

The electrical power distribution system being the backbone linking sources of power supply to components and controls which are essential for the safe handling of the plant implies a high reliability requirement.

The design of the electrical power distribution system was robust against single failures. However, the physical layout of the associated switch gear made the system susceptible to common cause failure by flooding.

2.2.3. Ultimate heat sink (emergency seawater system)

Sea water was used in the six Fukushima Daiichi NPP units as an ultimate heat sink (UHS) to remove heat as well as to cool equipment in the plant. Emergency seawater system (ESS) pumps had been installed on the ocean side to provide sea water as a UHS. The pumps were located at an elevation lower than the plant grade level. The intake structures were placed inside an artificial seawater pond, partially closed by breakwaters.

System response during the accident

After the earthquake and tsunami, it was found that the facility inspection crane had collapsed (presumably from the earthquake), and the ESS pumps and ancillary equipment were damaged by collision with floating debris from the tsunami. Sea water had also entered into the lubrication oil of the ESS pumps' motor bearing. However, except for the RHS system and Unit 4 seawater system pumps A and C, which had been removed for inspection, the other pumps remained standing in their original locations, even after being flooded by the tsunami. No pumps were washed away, indicating limited mechanical damage to the ESS pumps. The water inside the artificial seawater pond appeared very dirty, with a considerable amount of debris from the backflow of the tsunami waves [10].

Assessment

The UHS water intake structure was designed in the 1960s. In 2006, new seismic safety guidelines were developed and released by the Nuclear Safety Commission [9]. TEPCO evaluated the new design basis earthquake of the Fukushima Daiichi NPP and confirmed that the intake structure could withstand the new design basis earthquake and maintain its function. This new design basis earthquake implied almost the same level of acceleration as the observed earthquake of 11 March 2011. The ground level of the area where the ESS pumps are located is at OP +4 m. Based on the assessment results of tsunami heights, countermeasures were implemented in the first decade of the twenty-first century to maintain functions for tsunami heights of 5.7–6.1 m [33, 43]. However, the tsunami of 11 March 2011 was much higher than this, causing the pump motors to be submerged into the sea water and resulting in the loss of the system's function.

IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.41. Natural phenomena and human induced events shall be taken into account in the design of the systems and in the possible choice of diversity in the ultimate heat sinks and in the storage systems from which fluids for heat transfer are supplied".

The UHS intake structure is part of the first level of defence in depth for normal plant operation and the third level of defence in case of accident conditions. At the Fukushima Daiichi NPP, seawater system pumps were placed at OP +4 m level to ensure sufficient net positive suction head (NPSH). However, the pump motors were placed above ground level. The seawater pump motors were submerged under water when the tsunami wave inundated the plant. The design of the UHS was generally not sufficient to avoid damage by flooding, and hence this requirement of NS-R-1 [2] was not met. However, it should be noted that BWRs with Mark I containments are capable of using the atmosphere for the UHS by using such systems as the hardened containment vent or isolation condensers (see Technical Volume 1). The effectiveness of these systems is assessed in other sections of this technical volume.

The UHS is one of the most critical systems, and its failure may lead to core degradation. Therefore, prevention of loss of the UHS is important for removing the decay heat. The design of UHS structures should be enhanced by considering various means, such as making its doors water tight to avoid flooding. For the UHS, the availability of an additional water source located elsewhere on the site or direct dissipation of heat to the atmosphere (such as a cooling tower) could be considered as a lesson learned.

2.2.4. Main control room and instrumentation and control

The main control rooms (MCRs) were designed for all the plants to be operated safely in all plant states, including accident conditions. In case of a design basis accident, the MCR habitability system was designed to ensure that the MCR would remain habitable by filtering out radioactive iodine through a charcoal filter.

Generally, in the BWR design, emergency procedures are based on a number of key parameters, e.g. neutron flux, reactor pressure vessel (RPV) pressure, containment pressure and temperature. Reactor water level is the key parameter for confirming adequate core cooling. In essence, the level is derived by subtracting the weight of the water of a known reference leg from the weight of the water between two pressure taps at different elevations in the pressure vessel. In case of prolonged loss of both off-site and on-site power, it is expected that boiling will occur in the reference legs of the reactor water level instruments, which may result in false (non-conservative) indications (water level indicating higher than it actually is) [10].

System response during the accident

The tsunami and subsequent flooding wetted or submerged the EDGs, the switch gear and the electrical distribution systems, resulting in a rapid loss of all AC (with the exception of Unit 6) and most of the DC power supplies. Subsequently, the control board instrumentation and control (I&C) room lighting in Unit 1 became unavailable. Units 2 and 4 were affected in a similar way, whereas the emergency lighting and some indications remained functional in Unit 3. The MCR functions, the lighting as well as both on-site and off-site means of communication were lost.

The post-accident reporting of TEPCO mentioned that adverse conditions in the dry well had probably resulted in the false reactor level indications as explained above [10]. Consequently, these conditions, in conjunction with the global loss of indications in the control rooms, strongly affected the decision making process during the accident progression. However, from time to time some indications were restored by the use of power from car batteries.

The capability to monitor radiological conditions, both on-site and off-site, was severely affected due to the unavailability of electronic personnel dosimeters, computer systems for activating and recording dose, contamination monitors and portable survey instruments. Radiation monitors essential for monitoring core, containment and spent fuel pool conditions were also lost when the electrical distribution equipment was flooded. In addition, radiological effluent and environmental and meteorological monitors were also rendered unavailable. This situation severely hampered TEPCO and the off-site agencies.

TEPCO's Safety Data Parameter Display System (SPDS) was designed to function using plant data from the process computer (which displays various data required to monitor the operation of the plant). However, during the course of the accident, the SPDS could not fulfil its role once the process computer became unavailable. The government's Emergency Response Support System (ERSS) that relied on the SPDS data also failed (see Technical Volume 3).

The loss of the MCR habitability system made it difficult for the operators to monitor the plant status and to bring the plant to a stable condition [32].

Assessment

IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.71. A control room shall be provided from which the plant can be safely operated in all its operational states, and from which measures can be taken to maintain the plant in a safe state or to bring it back into such a state after the onset of anticipated operational occurrences, design basis accidents and severe accidents. Appropriate measures shall be taken and adequate information provided to safeguard the occupants of the control room against consequent hazards, such as undue radiation levels resulting from an accident condition or the release of radioactive material or explosive or toxic gases, which could hinder necessary actions by the operator."

Due to the loss of MCR habitability systems and instrumentation, this requirement was not fulfilled, since the MCR of Units 1–4 of the Fukushima Daiichi NPP could not perform their design functions due to the loss of these necessary power supplies. In addition, direct radiation exposure in the MCR, after the beginning of core degradation, was not preventable due to its location. The operators inside the MCR were subjected to the added stress of exposure to radiation. The report of the National Diet of Japan [32] highlighted that the workers were barely able to eat, sleep or use toilets, despite the fact that these activities are indispensable to supporting difficult emergency efforts over many hours.

One of the most significant aspects of the accident at the Fukushima Daiichi NPP was the progressive loss of all I&C, leaving the operators with few, if any, indications to assist them to understand, monitor and control the situation.

The beyond design basis event faced at the Fukushima Daiichi NPP emphasizes the importance of the continued functionality of a pre-defined set of essential instruments to manage an event to prevent the occurrence of a severe accident or to mitigate its consequences. The extent and nature of this minimal set of I&C need to be defined with care, according to the characteristics of each design. It must be protected, hardened and supported in such a way as to be available under all circumstances.

The Fukushima Daiichi accident also highlighted the importance of protecting the location(s) from which the operators will manage the accident (be it the MCR, an alternative control room or remote shutdown panels). The locations require, even in extreme hazard and severe accident conditions, adequate accessibility, functionality and habitability. This requires robustness against severe external hazards as well as radiation protection in case of significant radioactive releases.

2.2.5. Design features of Units 1–3 for maintaining fundamental safety functions

Units 1–3 were in operation at the time of the earthquake and successfully shut down the automatic signals initiated by earthquake monitoring instrumentation. However, decay heat was still being produced and needed to be removed by dedicated cooling systems. In Sections 2.2.6, 2.2.7 and 2.2.8, systems capable of maintaining fundamental safety functions will be discussed and an assessment will be made. Units 4–6 were in cold shutdown at the time of the event and hence will be discussed in separate sections.

2.2.6. Reactivity control

2.2.6.1. Reactor shutdown (trip) systems

The system primarily associated with controlling core reactivity in normal operations is the reactor control system, which acts through the control rod drive (CRD) system. The reactor protection system

(RPS) initiates a reactor scram (trip) signal to the CRD system that makes the reactor subcritical by quickly inserting safety rods.

The reactors also have backup systems to shut down the reactor, including an alternate rod insertion (ARI) system to initiate insertion of the safety rods if the RPS fails and the diverse standby liquid control system (SLCS), which injects borated water into the reactor core in case of failure of the CRD.

System response during the accident

During the earthquake at the Fukushima Daiichi NPP, the operating reactors were all automatically tripped by their RPSs after receiving the earthquake signal. The safety rods shut down the reactor cores, bringing them to the required subcritical condition.

Assessment

IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.13. Means shall be provided to ensure that there is a capability to shut down the reactor in operational states and design basis accidents, and that the shutdown condition can be maintained even for the most reactive core conditions."

The RPS and CRD systems met both the IAEA and Japanese safety requirements.

2.2.7. Maintenance of core cooling

2.2.7.1. Isolation condensers of Unit 1

The isolation condenser (IC) removes the decay heat of the reactor in case the main steam isolating valves are closed. The operating requirement of the IC is high reactor pressure, and if the state of high reactor pressure continues for a certain period of time, the outboard isolation valve on the condensate line (which is the only isolation valve which is normally closed) automatically opens and the system starts to remove heat by natural circulation (see Fig. 1.2-8 of Technical Volume 1). This operational delay is to prevent the IC from operating due to a transient phenomenon of instantaneously high reactor pressure [3].

Unit 1 was equipped with two ICs which could remove decay heat up to 6% of rated power. The IC tank capacity was sufficient for approximately 10 h of operation without make-up water.

The ICs were used in older BWRs to remove decay heat from the reactor core at high reactor coolant pressure when the main condenser was not available. The design of the IC was not part of the emergency core cooling system (ECCS). The IC isolation logic is similar to the logic for other piping systems which are not required for postulated accident mitigation and which penetrate the primary containment and are also connected to the RPV, such as the main steam lines and their associated main steam isolation valves (MSIVs). Because these isolation logic circuits are powered by the DC power systems, an isolation signal is generated upon the loss of DC power to the logic circuits, taking into account that such an event results in the loss of the ability of the logic circuits to monitor for a pipe break or coolant leak.

The two isolation logic circuits, each powered from a different and redundant DC power supply system, are typically arranged so that one logic circuit closes the valves inside the containment and the other closes the valves outside the containment. This overall design arrangement ensures that at least one of the two isolation valves will close upon receipt of an isolation signal in the case of a loss of AC or DC power. Once the isolation signal is received, the associated isolation valve will close, provided the required power is available (AC power for inboard valves and DC power for outboard valves). If motive power to the isolation valve is lost, then the valve will fail in its current position.

System response during the accident

After the earthquake and resultant loss of off-site power, the ICs were used for core cooling to remove the decay heat. Both ICs automatically started at 14:52 on 11 March 2011 on the RPV high pressure signal (7.13 MPa) to cool the reactor and maintain pressure less than the safety relief valve (SRV) lift set points. The operators who were monitoring RPV pressure and the subsequent cooldown rate (a limit of 55°C/h specified in the technical specifications) determined that one train was sufficient to cool the reactor. They closed the control valves (MO-3A and MO-3B) for both trains of the ICs around 15:03, and after that opened/closed only MO-3A to control reactor pressure. It was reported by TEPCO that the operators then manually started and stopped the ICs three times prior to the arrival of the tsunami. When the tsunami hit the site, both control valves were closed and no IC train was in operation. Based on the response of the measured parameters (RPV pressure and IC valve positions) following the earthquake and before the arrival of the tsunami, TEPCO concluded that no damage occurred to the ICs as a result of the earthquake.

During the event at Unit 1, the actual sequence of events concerning the loss of AC and DC power was not fully known, and therefore the final conditions of the isolation valves is also not known.

Assessment

For BDBAs, consideration should be given to the design of systems to remove decay heat which are independent of the normal cooling water systems. Examples of these types of systems are passive cooling systems or mobile equipment. As these systems can eventually be the last option in the case of extreme events, their design must pay special attention to their robustness in case of common cause failure events.

2.2.7.2. Reactor core isolation cooling system of Units 2 and 3

The RCIC system removes the decay heat from the reactor vessel when the main steam lines are isolated or the condensate and feedwater systems are not available. The system includes a steam turbine driven pump, with the capacity to inject at high pressure, whose operation depends only on DC power supply for control. The steam turbine takes its steam supply from the main steam line and the turbine exhaust is piped to the suppression pool to condense the steam. The system takes suction primarily from the condensate storage tank (CST); when this tank is depleted, the suction will be switched automatically (or manually) to the suppression pool.

System response during the accident

After the earthquake, the RCIC system of Unit 2 was manually started after the loss of feedwater as a direct consequence of the failure of external electricity supply. Following the tsunami, it was assumed that the RCIC system in Unit 2 was functioning due to the fact that prior to the tsunami, the reactor water level was stable. The operator manually switched the RCIC water source from the condensate water storage tank to the suppression pool at 04:00 on March 12 after the confirmation of the water level in the tank became low. Efforts were made by the Emergency Response Centre (ERC) to recover the power supply to monitor the water level in the reactor. The water level was confirmed at 21:50 on 11 March 2011 and operation of the RCIC system was confirmed at 02:55 on 12 March 2011. The RCIC continued to function for about three days after the earthquake. What eventually stopped the RCIC system has not been determined [32].

It is worth noting that the Unit 2 RCIC system remained operable far longer (nearly three days) than would have been expected, given the loss of DC power and the resulting uncontrolled operation. It is believed that in this uncontrolled operation mode, the RPV water level rose to the level of the steam lines, causing liquid water carryover into the RCIC system turbine, which in turn led to a reduced water injection rate and an effective self-regulating level control at the elevation of the steam lines. In addition, it is also believed that seawater flooding of the Unit 2 torus room led to significant heat rejection from the suppression chamber (SC), which prevented overheating of the water in the suppression pool and likely played a role in extending the operation time of the RCIC system. These effects need to be clarified in order to develop a better understanding of the operating behaviour of the RCIC system and its ability to prevent onset of core damage.

The RCIC system of Unit 3 was started up manually at 15:05 on 11 March 2011, as feedwater was lost as a direct consequence of the failure of external electricity supply, and maintained the reactor water level. Since the DC power remained functional in Unit 3, operators implemented contingency measures to avoid automatic shutdown of the RCIC system due to the high reactor water level by reducing the flow rate of the system. The system remained operational, allowing the reactor cooling function to be maintained for about 20 h. It was reported that the system unexpectedly shut down, resulting in a decrease of the reactor water level, which in turn led to the initiation of the high pressure coolant injection (HPCI).

Assessment

Based on the above, it is apparent that the RCIC system performed its function as per the design intent and that there was no failure of the system with respect to its design features. Further clarification is needed to understand the mode of operation in Unit 2 and what caused the unexpected shutdown in Unit 3. The system was not designed for prolonged loss of all AC (suppression pool heat removal) and DC (turbine control) power. Further analysis to identify additional ways of maintaining the operation of the RCIC system operation in case of extensive damage to the electrical systems could be carried out to reinforce the defence in depth concept.

2.2.7.3. Safety relief valves

The safety relief valves (SRVs) provide overpressure protection and depressurization for the RPV and associated steam piping. They are located on the main steam lines inside the dry well. The SRVs are spring type valves (with actuators) and function both as safety valves (spring type) and as relief valves (nitrogen type). In the safety mode, the high pressure steam acts on a pilot valve against a pre-set spring pressure. Once its set point is exceeded, it vents the steam which is holding the main valve closed, allowing the high pressure steam to escape through the main valve to the suppression pool. In the relief mode, pressurized nitrogen is applied through nitrogen cylinders, energized by a solenoid valve to move the pilot valve, allowing the main valve to open. The solenoids are energized by switches located in the MCR.

System response during the accident

According to the information released by TEPCO for Unit 1, it is not fully understood yet how the reactor depressurized. Reactor vessel and the containment pressures became equalized approximately 11 hours after the onset of the accident. The possibility of a leakage into the containment from in-core monitor dry tubes and/or SRV gaskets is being investigated. Any of these situations would have been induced by the extreme conditions existing in the reactor during this phase of the severe accident evolution.

For Units 2 and 3, operators encountered difficulties actuating the solenoid valves to operate the relief mode of the SRVs and depressurize the reactor when needed. These problems were caused by a lack of DC power and high pressure air. Indications are that the SRVs in Units 2 and 3 were eventually opened, but the delay in depressurizing the Unit 2 reactor contributed to the amount of core damage because the alternate injection systems are incapable of injecting water into the reactor at high pressure.

Assessment

The reactor coolant system was designed with adequate SRVs to protect the pressure boundary of the reactor coolant system against overpressure to meet the requirement of IAEA Safety Standards Series No. NS-R-1 [2], which states:

> "6.21. The reactor coolant system, its associated auxiliary systems, and the control and protection systems shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded in operational states. Provision shall be made to ensure that the operation of pressure relief devices, even in design basis accidents, will not lead to unacceptable releases of radioactive material from the plant. The reactor coolant pressure boundary shall be equipped with adequate isolation devices to limit any loss of radioactive fluid."

The SRVs were designed to handle DBAs. They performed their functions to control the pressure in the reactor successfully during the initial phase of the accident but lost their relief function when the accident became more severe and DC power and high pressure air were lost. Depressurization of the primary circuit depends on the relief mode of the SRVs and the importance of ensuring the availability of the DC power supply to energize the SRV solenoids and the availability of compressed air supplies to power the SRV pilot valves needs to be emphasized. Permanently installed or portable DC supplies and air supplies should be adequate to support SRV operation over extended times without AC power.

2.2.7.4. Emergency core cooling systems

The ECCS provides core cooling under loss of coolant accident (LOCA) conditions to limit fuel cladding damage. Across the various unit designs at the Fukushima Daiichi NPP, the installed ECCS equipment comprises a combination of two possible high pressure and two low pressure systems, with electrical power supplied by emergency sources and composed of different subsystems. The high pressure systems are the high pressure core spray (HPCS) system or the high pressure coolant injection system (HPCI), which injects water from the condensate storage tank (CST), or the suppression chamber, the core spray (CS) system and the automatic depressurization system (ADS). The low pressure systems are the low pressure core spray (LPCS) system, which injects water into the core upper plenum and the core bypass region, and the residual heat removal (RHR) system working in low pressure coolant injection (LPCI) mode. The LPCI protects against larger LOCAs to maintain the water level in the reactor when the reactor pressure is low enough to allow low pressure injection.

The HPCI or, for the newer BWR units, the HPCS, is the first line of defence in the ECCS. It is designed to prevent fuel cladding damage for small LOCAs of the reactor coolant system. These systems only require DC power to operate (for turbine control). They comprise a turbine driven pump, pipes, valves and ancillary systems. The turbine is driven by high pressure steam extracted from upstream of the isolation valve of a main steam line on the inner side of the dry well. Normally, the CST is used as a water source, but the system was designed so that the suppression pool water can also be used as a backup.

The LPCI system injects large volumes of water for large LOCAs. These are not discussed in detail, since they lost their function after the complete loss of AC power. Operators never tried to initiate them as they were aware of this situation.

The ADS is an I&C subsystem which generates the control signal to open the reactor coolant system SRVs to depressurize the reactor, allowing injection with the low pressure ECCS. In this sense, ADS is functionally redundant to the HPCI (single train) for accident initiators, which do not induce prompt reactor depressurization.

System response during the accident

Unit 1

After the earthquake and prior to the arrival of the tsunami at the site, there were no significant abnormalities found in the alarms and display lights for the ECCS. Reactor water levels were stable and reactor pressure was controlled via the IC. After the tsunami struck, the HPCI system control panel (fed by DC power) went off, and the system could no longer be started. Finally, the ADS was also inoperable as a consequence of the loss of DC power and high pressure air; nevertheless, the reactor eventually depressurized by some yet unconfirmed mechanism.

Unit 2

After the tsunami, the HPCI system control panel (fed by DC power) shut off. The system completely lost its functionality and could not be used as a backup to the RCIC, which had remained in operation and continued so for a long period of time. The loss of DC power equally affected the ADS, which also became inoperable. At some time during the accident, operators decided to depressurize the reactor in order to allow alternative water injection. This action was finally implemented by using portable batteries to open the solenoid valves.

Unit 3

In Unit 3, both the RCIC and HPCI systems remained operable, as the flooding did not affect the DC power supply. The turbine driven RCIC system eventually stopped on 12 March 2011. Shortly afterwards, the HPCI system started automatically on a low reactor water level signal. At this point, the reactor pressure was around 7.5 MPa. The flow rate of the HPCI was controlled by control room operators in such a way as to save battery power and to prevent its automatic shutdown due to high reactor water levels in the same way as had been previously done for the RCIC system. While this mode of HPCI operation avoided starting and stopping the HPCI system and the risk that the system would ultimately fail to restart, it may also have aggravated the attempt to maintain the RPV water level. Due to the depletion of the DC batteries, the water level indication was later lost, but the HPCI remained in operation. Operators monitored the HPCI system operation status via reactor pressure and pump discharge pressure.

The operators worked to switch reactor injection from the HPCI system to diesel driven fire pump (DDFP). The shift crew manually stopped HPCI on 13 March 2011, but their attempts to depressurize the reactor were unsuccessful and alternative water injection via fire trucks was ineffective. Attempts were made to open the SRV, but were not successful. At this stage, the operator tried to restart the RCIC and HPCI systems, but neither could be restored as the DC battery was depleted. The reactor pressure rose again to 4.0 MPa one hour after the HPCI stopped, and it further increased to 7.38 MPa in two hours, later reaching the safety valves' set-point. Based on the TEPCO evaluation, the depressurization of the Unit 3 RPV occurred because the ADS logic was successful [31].

Assessment

The Nuclear Safety Commission (NSC) of Japan issued a Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities (NSCRG: L-DS-I.0) in 1970, which was comprehensively reviewed and revised based on technological improvements and developments of

LWRs [22]. It also incorporated lessons learned from various events that occurred in nuclear power plants, such as the accident at the Three Mile Island NPP, in the USA. The NSC issued the latest revision of this guide on 30 August 1990. The ECCS design requirements given in Guideline 25 are [22]:

> "(1) The emergency core cooling system shall be designed to be capable of preventing serious damage of reactor fuel and of limiting the reaction between fuel cladding metal and water to a sufficiently small amount in case of a postulated loss of reactor coolant resulting from a break in piping, etc.
>
> (2) The emergency core cooling system shall be designed with redundancy or diversity and independence so that the system can fulfill its safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the system.
>
> (3) The emergency core cooling system shall be designed to be capable of being tested and inspected on a periodical basis. The emergency core cooling system shall also be designed to allow testing and inspection of each constituent system independently so that the integrity and redundancy of the emergency core cooling system can be verified."

IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.35. Core cooling shall be provided in the event of a loss of coolant accident so as to minimize fuel damage and limit the escape of fission products from the fuel. The cooling provided shall ensure that:
>
> (1) the limiting parameters for the cladding or fuel integrity (such as temperature) will not exceed the acceptable value for design basis accidents (for applicable reactor designs);
>
> (2) possible chemical reactions are limited to an allowable level;
>
> (3) the alterations in the fuel and internal structural alterations will not significantly reduce the effectiveness of the means of emergency core cooling; and
>
> (4) the cooling of the core will be ensured for a sufficient time."

In addition, NS-R-1 [2] states:

> "6.37. Adequate consideration shall be given to extending the capability to remove heat from the core following a severe accident."

IAEA Safety Standards Series No. NS-G-1.9, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [44], states:

> "4.81. Equipment for emergency core cooling should be adequately protected from the consequences of internal and external hazards such as seismic hazards that have the potential to jeopardize its safety functions."

NS-G-1.9 [44] further emphasizes:

> "3.30. Since redundant or diverse systems are also potentially vulnerable to events (e.g. fires, floods) resulting in common cause failures, appropriate physical barriers or physical separation or a combination of both should be used as far as is practicable."

The design of the ECCS system at the Fukushima Daiichi NPP was in accordance with the requirements of the NSC Regulatory Guidelines of 2006 [9] and those stipulated in IAEA Safety Standards Series No. NS-R-1 [2] because the system was designed for design basis accidents.

In summary, the ECCS was designed as part of defence in depth (DID) level 3 to mitigate the consequences of postulated design basis accidents, i.e. LOCAs. From the above discussion, it is clear that the LPCI systems could not operate in such circumstances, as they depend on AC power to drive valves and pump motors. The HPCI and ADS also were not designed to cope with situations without DC power. Further analysis to identify additional ways of maintaining the HPCI and ADS operation in case of extensive damage to the electrical systems could be carried out in order to reinforce the DID concept.

2.2.7.5. Alternate water injection system

The United States Nuclear Regulatory Commission, in NRC SECY 89-17 [45], January 1989, identified a number of plant modifications that substantially enhance the capability of a plant to both prevent and mitigate the consequences of severe accidents. The recommended improvements include an alternative water supply to the reactor vessel and dry well and updated emergency procedures and training. In light of these recommendations, severe accident countermeasures had been adopted at the Fukushima Daiichi NPP, such as the alternate water injection system provision using the fire protection system. The alternate water can be injected into the reactor core (through the core spray piping) or into the primary containment (through the suppression pool cooling return line) from other water sources (fresh water or sea water).

The alternate water injection system utilizes equipment capable of injecting cooling water into the nuclear reactor when the normal ECCS system fails to function. The purpose of the system is to inject water into the nuclear reactor and primary containment vessel (PCV), and this feature was implemented between 1996 and 2002 to provide water injection in the event of loss of core cooling and/or loss of molten core cooling. Initially, this system was utilizing the stable fire system as a source of water, which was not seismically resistant. This area was further considered and strengthened in 2007, and the system was upgraded by the installation of fixed connection points on the turbine building, which allowed use of the fire trucks as a source of coolant. All of the pumps used for this system are low pressure pumps and, therefore, the reactor and containment need to be depressurized using a combination of the SRVs and the containment venting systems.

System response during the accident

During the Fukushima Daiichi accident, the alternate water injection system injected water into the reactors after the failure of the conventional systems. Nevertheless, due to other circumstances external to the system, it could not be employed in a timely manner. For Unit 1, operators were able to use alternative injection without attempting to open SRVs, indicating that the reactor depressurized on its own via some unknown failure mechanism. For Units 2 and 3, operators were eventually able to open SRVs to depressurize the reactor, but as has been previously mentioned this was too late for alternative injection to prevent core damage. As will be assessed later in this section, the containments for Units 1–3 were also eventually depressurized.

Assessment

This assessment assumes that the system was intended to demonstrate compliance with para. 6.37 of IAEA Safety Standards Series No. NS-R-1 [2], which states: “Adequate consideration shall be given to extending the capability to remove heat from the core following a severe accident.” The system was ready to perform its intended function during the accident at the Fukushima Daiichi NPP and could have played a vital role by injecting coolant into the reactor cores after failure of the capability to inject coolant into the reactors through the usual ECCS.

During the accident, different problems arose in Units 1, 2 and 3, related to the failure or delay of the depressurization of the reactor and/or the PCV and the shortage of readily available equipment. Furthermore, an assessment conducted by TEPCO [31] concluded that, for all units, an unknown amount of water was bypassed to the main condenser or the CST and was not injected into the core to either cool the fuel and fuel debris. Because of these problems the system was not functioning as designed and it is assessed, therefore, that the system did not meet the requirement stipulated in para. 6.37 of IAEA Safety Standards Series No. NS-R-1 [2].

2.2.8. Maintenance of containment integrity

2.2.8.1. Primary containment

The design of the Mark-I type primary containment is a dry well in the shape of a pear shaped bulb, containing the RPV and primary piping, and a wet well, which contains a torus shaped suppression chamber (SC). The SC is partly filled with water, and the large vent pipes that connect the dry well to the SC have their open ends submerged in the water of the suppression pool. Escaping steam from a LOCA in the dry well would be routed to the SC, where the water condenses the steam. The SC also condenses steam discharged from the HPCI/RCIC turbines and from the SRVs. The PCV is inserted with nitrogen to avoid the potential occurrence of hydrogen ignitions during accidents.

The design pressure of the Mark I-type PCV is the highest transient pressure expected to be caused during the worst postulated accident, i.e. a sudden double ended guillotine break of the largest diameter pipe in the recirculation outlet loop (LOCA). However, the design assumes that the ECCS activates automatically at the moment when a pipe break occurs. It did not assume the scenario of a total loss of AC power with extensive loss of DC power, which in the end caused complete failure of the ECCS [3].

Containment heat removal would, under design basis accidents, be provided by the RHR system in its containment cooling mode. The containment cooling mode includes suppression pool cooling and containment spray. As the RHR system is AC powered, it was not available to help with removing the heat from the containment.

System response during the accident

After the core became damaged, large amounts of steam, hydrogen and radioactive material were released inside the primary containment due to opening of SRVs and other potential leakage paths. This release of large quantities of steam pressurized the PCV and increased the temperature as well, since cooling was also lost in the suppression pool. Sometime after core damage, the PCV failed to maintain its integrity, and a large quantity of steam, along with hydrogen and other radioactive effluents, was released into the reactor building. Even in the major consequences of the accident, the primary containment performed its intended functions according to its design characteristics. No hydrogen deflagration was reported inside the PCV; however, its eventual loss of leaktightness was a consequence of continuous operation under environmental conditions far beyond its design limits for an extended period of time.

Assessment

IAEA Safety Standards Series No. NS-R-1 [2] states:

> "6.43. A containment system shall be provided in order to ensure that any release of radioactive materials to the environment in a design basis accident would be below prescribed limits. This system may include, depending on design requirements: leaktight structures; associated systems for the control of pressures and temperatures; and features for the isolation, management and removal of fission products, hydrogen, oxygen and other substances that could be released into the containment atmosphere."

NS-R-1 [2] also emphasizes:

> "6.46. Provision for maintaining the integrity of the containment in the event of a severe accident shall be considered. In particular, the effects of any predicted combustion of flammable gases shall be taken into account."

NS-R-1 [2] further states:

> "6.64. Systems to control fission products, hydrogen, oxygen and other substances that may be released into the reactor containment shall be provided as necessary:
>
> …….
>
> (2) to control the concentration of hydrogen, oxygen and other substances in the containment atmosphere in design basis accidents in order to prevent deflagration or detonation which could jeopardize the integrity of the containment."

Moreover, NS-R-1 [2] requires that:

> "6.65. Systems for cleaning up the containment atmosphere shall have suitable redundancy in components and features to ensure that the safety group can fulfil the necessary safety function, on the assumption of a single failure."

In para. 6.66 of NS-R-1 [2], it is also stated that: "Adequate consideration shall be given to the control of fission products, hydrogen and other substances that may be generated or released in the event of a severe accident."

The containment of the Fukushima Daiichi NPP reactors was designed in accordance with IAEA safety standards. As explained in Section 2.2.8.2, the containment functionality for severe accident conditions depended on the hardened containment venting system, which, due to the accident management strategy in use at the Fukushima Daiichi NPP and the hardware provisions derived from this strategy (i.e. rupture disc set points), did not function as designed.

2.2.8.2. Hardened containment venting system

The United States Nuclear Regulatory Commission, in Generic Letter 89-16 [46], recommended enhancements to the performance of the containment, including the installation of a pressure resistant (hardened) containment venting system. The purpose of the hardened containment vent was to provide a method of relieving containment pressure to maintain it below the primary containment pressure limit and accommodate a steam flow equivalent of 1% decay heat power without adversely impacting the mitigating equipment. The hardened vent was not designed to operate during a severe accident.

The NSC established the Common Issue Committee in July 1987 to investigate approaches to severe accidents, probabilistic safety assessment (PSA) techniques and the containment function during severe accidents. In the 1990s, hardened venting systems were installed in all of the units of the Fukushima Daiichi NPP.

The system had the capacity to vent from either the dry well or the wet well. The operators were capable of controlling the PCV pressure, maintaining it below the threshold that would affect the PCV's integrity and enabling water injection into the RPV with low pressure systems. In the case of a total loss of AC and DC power, the vent system isolation valves could only be actuated manually and locally from inside the reactor building by the use of compressed air.

System response during the accident

In Units 1 and 3, venting from the wet well enabled the PCV to be depressurized significantly. The Diet report [32] concluded that, despite several attempts, the PCV of Unit 2 could not be depressurized. The loss of DC power and high pressure air led to the loss of the remote control capability of the vent valves. In addition, at the time of the decision to open the PCV vent valves, the radiological conditions in the reactor building prevented the implementation of the required manual local actions.

Assessment

Containment venting would have allowed for more effective core cooling via alternative means and the migration of hydrogen from the containment to the reactor building could have been prevented. The combination of core damage, high containment pressure and high containment temperature compromised the containment, allowing hydrogen to escape from the PCV, which led to the explosions in Units 1, 3 and 4. The hydrogen explosion in Unit 4 was caused by the migration of hydrogen from Unit 3 to Unit 4 via an interconnected ventilation system (see Section 2.4).

2.2.8.3. Reactor building

During a design basis accident, the standby gas treatment (SBGT) system — a specially designed safety feature — is expected to maintain the reactor building at subatmospheric pressure in order to collect potential radioactive leakage from the PCV. It is a safety related system which is redundant, with electrical power supplied by EDGs and which is fitted with high efficiency filters in the exhaust line of the stack.

System response during the accident

In the Fukushima Daiichi accident (after the tsunami struck), the SBGT system remained inactive due to the loss of electrical power. This contributed to the consequences of the accident, both with regard to the radiological conditions in the reactor building and the physical process of gravity driven hydrogen migration to the upper parts of the reactor building. After core damage in Units 1–3, steam and hydrogen were released into their respective PCVs and eventually leaked to the reactor buildings. Hydrogen explosions occurred in the reactor buildings of Units 1, 3 and 4.

A factor that influenced the progression of the accident in Unit 2 was the unit's proximity to Unit 1. The impact of the hydrogen explosion in the reactor building of Unit 1 caused a blowout panel on the wall of the reactor building in Unit 2 to open. The main objective of a blowout panel was to prevent damage to the reactor building as a result of the rapid increase in internal pressure. The hydrogen probably leaked from the PCV and did not accumulate in the reactor building, as it escaped directly to the external atmosphere, along with some radioactive products. This was the likely reason why no hydrogen explosion occurred in the reactor building in Unit 2. The explosion in the Unit 4 reactor building occurred due to backflow of hydrogen containing gases from the Unit 3 reactor building through its SBGT system, as explained in Section 2.2.9

Assessment

The reactor buildings of Units 1, 3 and 4 were severely damaged because hydrogen mitigation had not been properly considered. This situation occurred because the leakages induced by the failure of the integrity of the primary containments had not been anticipated and hydrogen mitigation efforts were focused on the PCV with features such as making the atmosphere inert. However, due to the failure of the containment heat removal systems, the long term pressurization and temperature increases in the PCV, the leakage of hydrogen to the reactor building ultimately resulted in multiple explosions, since there were no systems installed to manage hydrogen.

2.2.9. Design features of Unit 4

Unit 4 was a BWR/4 type reactor, similar to those of Units 2 and 3, except for some minor design differences. It was shut down for reload and maintenance at the time of the earthquake and tsunami. All the fuel was in the spent fuel pool (SFP) to facilitate work on the reactor internals. The upper pool cavity gate was installed, isolating the SFP from the reactor well. EDG 4A was out of service for planned maintenance, and EDG 4B was operable and in a standby state. The SFP water temperature was maintained at approximately 27°C.

Response during the accident

After the earthquake and tsunami, the operators in the control room of Units 3 and 4 focused on the stabilization of Unit 3, since it was in operation and had a large amount of decay heat compared with the heat load in the SFP of Unit 4. There was no fuel in the Unit 4 reactor vessel, and hence safety systems were not actuated. The only concern was the cooling of the Unit 4 SFP, which had been lost due to the station blackout condition.

Assessment

The hydrogen explosion in the reactor building of Unit 4 on 15 March 2011 received considerable attention and resources because the explosion was unexpected, as operators did not believe that there was enough decay heat in the SFPs to result in rapid overheating and subsequent high temperature interaction of zirconium and water to produce hydrogen. Given this uncertainty, considerable efforts were undertaken to ensure that the fuel in the SFP was properly cooled because a release from the pool would lead directly to the atmosphere. The subsequent analyses and inspections performed by TEPCO revealed that the water level in the SFP of Unit 4 never dropped below the top of active fuel (TAF), and no significant fuel damage occurred. Nevertheless, according to information released by TEPCO, SFP water did reach saturation conditions [10] and the impact of this uncertainty shows the importance of reliable monitoring systems for transmitting the status of the SFP during accidents.

Regarding the cause of the explosion in Unit 4, recent assessments have concluded that the hydrogen migrated from Unit 3 to Unit 4 via the vent lines of the containments of Units 3 and 4 which were connected to the exhaust vent piping. This piping arrangement allowed gases from the Unit 3 containment to be vented into the Unit 4 reactor building as a result of reverse flow through the Unit 4 SBGT system. One design feature which may have prevented or mitigated the migration of hydrogen is backflow dampers, which were not included in the design of the Unit 4 SBGT system [18]. This indicates that the interfaces between the common exhaust ventilation systems of Units 3 and 4 were not designed to isolate these units appropriately in the event of an accident.

2.2.10. Design features of Units 5 and 6

Unit 6 of the Fukushima Daiichi NPP has a structure consisting of a reactor wing with an annex on the outside. In contrast, Unit 5 has a stand-alone type reactor building, consisting of a reactor wing with no annex. Two of the Unit 6 EDGs (EDG A and EDG HPCS) were installed in the basement of the reactor building annex; a third (EDG B) was located on the ground floor of a separate EDG building. Both EDGs of Unit 5 were installed in the basement of its turbine building. The reason for installing EDGs on the foundation (lowest basement floor) level had been the need to deal with the vibration and seismic safety of large size equipment. At the time of the earthquake, Unit 5 was going through periodic inspection and RPV pressure leakage tests were being conducted with fuel loaded in the reactor. Unit 6 was also undergoing periodic inspection, and the reactor was in cold shutdown condition (i.e. all control rods inserted) with the fuel being loaded.

Response during the accident

Loss of the external power supply was caused by the earthquake, and the subsequent large tsunami knocked out the four water cooled EDGs of Units 5 and 6. While the EDGs were installed in the basement, their air supply louvres were above the inundation level, and no other water ingress pathways, such as ducts or trenches, existed. What caused the EDGs to stop functioning was the loss of cooling using the sea as the ultimate heat sink. The air cooled EDG of Unit 6 and the associated electrical distribution system survived the flooding, and they were used to re-establish cold shutdown in Units 5 and 6. Similarly, the DC power supply in Units 5 and 6 installed in the mezzanine basement floors was not damaged by flooding. The charging of the DC power supply of Unit 5 was restarted by using power from the EDG of Unit 6, using temporary cross-tie electrical connections.

Assessment

Loss of cooling was prevented in Units 5 and 6 by directly connecting a power source of one air cooled EDG to the equipment available for cooling. It was still possible to inject cooling water using the make-up water condensate (MUWC) system, and the cooling of the reactor was maintained with the discharge of steam to the SC by SRVs. However, the RHR and cooling seawater systems, which both discharge heat into the ocean as the UHS, stopped functioning due to the unanticipated size of the tsunami. The seawater pumps and their motors were at a lower level than the reactor building, and they were flooded and damaged. Thus, even if electricity had been available to drive the ECCS, there would have been no way of dissipating the heat. However, temporary submersible pumps were used in Units 5 and 6 to restore the function of each RHR system. The system at Units 5 and 6 was re-established by using substitute RHR pumps. The supply of water to the SFP was also carried out in a similar manner. The mechanical damage to ESS pumps was limited. The seawater pumps to cool EDG A at Unit 6, for example, were restarted on 18 March 2011 without having to be repaired. They then allowed EDG A at Unit 6 to start up on 19 March. Cold shutdown of these two units was declared on 20 March 2011. No other damage to the reactor buildings or SFPs was reported.

The prevention of core damage in Units 5 and 6 was to a large extent the result of the survival of a single EDG. This was due to its different cooling mechanism and its location, but also to the location of the associated switch gear. This example reinforces the assessment that an analysis of flooding hazards could have shown the importance of siting the EDGs in a safer location.

2.2.11. Spent fuel storage facilities at the Fukushima Daiichi NPP

The inventory of spent fuel being stored at the Fukushima Daiichi NPP on the day of the accident is shown in Table 2.2–1.

TABLE 2.2–1. NUMBER OF SPENT FUEL ASSEMBLIES BEING STORED AS OF 11 MARCH 2011 ACCORDING TO THE INDEPENDENT TECHNICAL EVALUATION AND REVIEW, MAY 2011 [47]

Unit	Spent fuel assembly (units)	New fuel	Decay heat (MW)
Unit 1	292	100	0.18
Unit 2	587	28	0.62
Unit 3	514	52	0.54
Unit 4	1331	204	2.26
Unit 5	946	48	1.01
Unit 6	876	64	0.87
Common pool	6375	0	1.13
Cask storage building	408	0	—

Response during the accident

In the course of the accident, the systems of Units 1–5 and the common pool which were designed for cooling the spent fuel inventory and injecting make-up water became unavailable due to the loss of AC power from both the external grid and the EDGs. The UHS was also unavailable due to damage to the intake structure from the tsunami and the loss of power. Moreover, key parameters, such as the water temperature or the water level in the SFP were unknown because of the loss of DC power. However, in the case of Unit 6, one EDG (6B) survived, as explained earlier, but the SFP cooling in Unit 6 was unavailable due to damage to its seawater cooling pump. The cask storage building also experienced SBO, but the dry storage casks were designed to be air cooled through natural convection.

Spent fuel pools

In the first few days following the tsunami, the operators considered that there was sufficient water in the SFPs and that overheating of the fuel was not an immediate issue. This view changed on 15 March, when the Unit 4 reactor building exploded. As previously described, 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 SFP 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 the SFPs.

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 airdrop of water from helicopters. Subsequent analysis and inspections revealed that the water level in the SFPs 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 that the source of the hydrogen was not the fuel in the Unit 4 SFP but the migration of hydrogen from Unit 3 to Unit 4 via a connected PCV venting piping. However, the lack of knowledge about the actual conditions in the SFPs during the accident, due to the loss of instrumentation, led to the effort to add water to the pool. In the days and weeks that followed, the SFPs for Units 1–4 were cooled by a combination of external injection via concrete pumper trucks and, by the end of March 2011, the hook-up of external water sources which were connected to the installed SFP cooling piping. Units 5 and 6 were cooled via the cross-connection of the operational EDG from Unit 6.

Assessment

The result of the assessment of the performance of the SFPs points directly to the lack of knowledge regarding the actual water level and temperature in the pools. While it was necessary to take action to restore lost water inventory and cool the SFPs, these actions were not as high a priority on 15 March as was thought following the hydrogen explosion in Unit 4. The lack of instrumentation to indicate to operators the actual conditions in the SFP led them to prioritize spent fuel pool cooling as a high priority action. It should be pointed out, however, that given the information available at the time of the accident to operators, their concerns were justified.

An inspection of the dry cask storage facility revealed that, while the building was damaged by the tsunami, the dry storage casks did not appear to have been damaged. The casks became wet because of the tsunami, but they were not moved from their storage locations by the force of the waves or debris. No abnormalities have been seen in the dry storage cask area as its cooling was maintained with the circulation of air.

2.2.12. Application of defence in depth

The concept of defence in depth (DID) referenced by the IAEA was first introduced by the International Nuclear Safety Group (INSAG) [48]. It is now widely recognized as a fundamental safety approach for ensuring the safety of nuclear installations. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], states that DID is the primary means of preventing accidents and mitigating their consequences. DID is implemented through the combination of a number of consecutive and independent levels of protection that would have to fail before harmful effects could be caused to people and/or to the environment. If one level of protection were to fail, the subsequent level would be available. When properly implemented, DID ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of a very low probability. The independent effectiveness of the different levels of defence, and thus the resistance to common cause failures, is a necessary element of DID. It ensures that a high level of safety is achieved with sufficient margins to compensate for potential equipment, human or organizational failures.

The strategy for DID is twofold: first, to prevent accidents; and second, if prevention fails, to limit their potential consequences and to prevent any progression to more serious conditions. Accident prevention is the first priority. The rationale for the priority is that provisions to prevent deviations of the plant state from well known operating conditions are generally more effective and more predictable than measures aimed at mitigation of the consequences of such a departure, since the plant's performance generally deteriorates when the status of the plant or a component departs from normal conditions. Consequently, preventing the degradation of plant status and performance will generally provide the most effective protection of the public and the environment, as well as assuring the productive capacity of the plant. Should preventive measures fail, however, mitigatory measures — in particular the use of a well designed confinement function — can provide the necessary additional protection of the public and the environment.

IAEA safety standards [2, 17] reflect today's consensus that for nuclear power plants, DID consists of five levels, which are summarized as follows:

Level 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 (PIE)), to reducing the consequences of a given PIE or to reducing the likely release of the 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.

Level 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 PIEs are likely to occur over the service lifetime of an 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 PIEs.

Level 3. For the third level of defence, it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or PIEs may not be halted 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.

Level 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.

Level 5. The fifth and final level of defence is aimed at mitigation of the radiological consequences of potential releases of radioactive material that may result from accident conditions. This requires the provision of an adequately equipped emergency control (see Technical Volume 3).

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 (DID Level 1). After the earthquake, the power supply was successfully provided from on-site sources, and all safety systems at DID Level 3 continued to function as designed. This indicated that the safety systems and equipment withstood the seismic hazard [10].

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 [2]. Key safety equipment was not protected in leaktight 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 DID 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 DID 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 [49].

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 [10].

The last physical barrier included in DID Level 4 is 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 [10].

The Fukushima Daiichi accident demonstrated that extreme natural hazards have the potential to invalidate or impair multiple levels of defence in depth [50, 51]. A systematic identification and assessment of external hazards and robust protection against these hazards needs therefore to be considered for all levels of DID. 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.

Summary of the Fourth International Conference on Topical Issues in Nuclear Installation Safety

On 21–24 October 2013, the IAEA convened an international conference on Topical Issues in Nuclear Installation Safety, in Vienna [51]. This conference both summarized the latest assessments regarding the application of DID and identified future activities. The Conference President summarized the meeting as follows:

— Although the DID concept remains valid after the Fukushima Daiichi accident, it has to be strengthened and extensively applied in order to meet most recent safety objectives for nuclear plants, such as the ones adopted by the Contracting Parties during the extraordinary meeting of the Convention for Nuclear Safety. DID is not only relevant for the design of new installations, but should also be maintained/improved by periodic safety reviews over the entire life of installations.

— While DID remains an essential tool for safety and should continue to be applied, further development and guidance are required on several subjects such as:
  - Consistent application of design basis definitions at the international level;
  - Postulation of multiple failures in reactor design;
  - Practical elimination of sequences, in relation with the use of deterministic and probabilistic approaches;
  - Assessment of independence and reliability of different levels of DID;
  - Approach to be adopted for very low probability events leading to very large health and societal consequences;
  - Tools to be based on already developed methodologies to ensure that safety provisions are comprehensive enough to ensure DID.

— One important lesson from the Fukushima Daiichi accident is that extreme external hazards can result in common cause failures simultaneously jeopardizing several levels of defence. Such common cause failures can result in complete loss of the instrumentation of the plant, inducing extreme difficulties in the management of a severe accident. Special attention should be given to these risks when implementing DID.

— Hazards, as well as the combination of hazards, to be taken into account in relation to DID need further work and international guidance.

— Effective implementation of DID requires that the most recent knowledge resulting from operational experience feedback as well as research and development are taken into account. Human factors and reliability of instrumentation and control systems are subjects that require further development and guidance.

— Approaches have already been developed and efforts are underway to improve robustness of plants, taking into account the current lessons learned from the Fukushima Daiichi accident. However, such approaches still need to be matured on several topics such as:
  - Criteria to choose between fixed and mobile equipment;
  - Design approach for equipment or ‘hardened safety core’ of equipment ensuring fulfilment of safety functions under extreme conditions.

— As already highlighted by the IAEA, the World Association of Nuclear Operators (WANO) and the OECD Nuclear Energy Agency (OECD/NEA), mitigation levels of DID should be enhanced in operational safety, while prevention should also be maintained. The IAEA’s Operational Safety Review Team (OSART) missions and WANO peer reviews have already been extended to cover some design aspects related to continuous improvement of operating plants.

— Wider use of IAEA review services, especially those related to siting, design and emergency preparedness, should be promoted and established, contributing to the prevention of nuclear accidents and emergency management. Peer pressure should be extended to ensure the implementation of their recommendations. The topics of these peer reviews could be chosen in light of operating experience feedback.

— Realistic drills are essential for the effectiveness of emergency preparedness. They should involve all the key players, at all levels, in decision-making and communication. They should be designed, as much as possible, to train individuals and to prepare organizations to react in a flexible manner to unexpected situations.

— An idea was proposed that the technical concept of DID is necessary but not sufficient to ensure safety. Effective institutional systems need to be set up, applying the same DID concept and principles, involving all stakeholders (operators, regulators, industry, etc.). To address this issue, a peer-review service was suggested to be established jointly by the IAEA and WANO, using the expertise of the OECD/NEA, with initial self-assessment feeding into Nuclear Safety Convention review meetings.

— Following the Fukushima Daiichi accident, WANO adopted a strategic orientation to increase its strength and its focus on nuclear safety. It improved its peer review process and expanded its scope to integrate some design aspects, as well as corporate peer reviews. Overall, the peer pressure was increased in order to enhance commitment to safety of the operators worldwide.

— The safety of nuclear and non-nuclear industries would benefit from closer collaboration, allowing better sharing of experience feedback, as well as education and training methods. Nuclear safety could also benefit from the experience of first response organizations, such as the police or fire brigades.

2.2.13. Summary

The tsunami caused a station wide loss of AC power and, in most units, an immediate loss of DC power. The operators struggled in extremely difficult conditions to re-establish cooling to the cores of Units 1, 2 and 3. In addition, due to the combined impact of the explosion in the reactor building of Unit 4 and the loss of instrumentation, the operators believed that the spent fuel in the storage pool in Unit 4 was damaged. The fundamental safety functions that must be maintained are: (1) reactivity control; (2) core cooling; and (3) containment integrity. Reactivity was successfully controlled. Core cooling was not maintained, primarily because of the failure of the high pressure injection systems and the inability to depressurize the reactors to allow for alternative modes of water injection. The high pressure injection systems failed because the loss of DC power made it difficult to operate steam driven systems. The operators were also unable to deploy alternative water injection systems due to the loss of DC power and the high pressure nitrogen system, which is needed to open the SRVs. Finally, the containment function was lost due to the inability to remove heat from the containment caused by the loss of AC and DC power, which rendered both containment cooling systems and containment ventilation systems inoperable.

The status of the various means of heat removal on Units 1–3 was as follows.

Unit 1 had a system called an isolation condenser (IC), which would have been able to remove decay heat via natural circulation without power for about ten hours if it had been working. The system had started automatically when the off-site power was lost following the initial earthquake. At the time the tsunami struck, the operators had just closed some valves in the system in order not to cool down the reactor too quickly. Without power, it was not possible to reopen them.

In order to use an alternative mode of water injection (from the stationary diesel driven fire pump or mobile pumpers), the operators needed to depressurize both the RPV and the containment. Following successful opening of the containment vent, and after the RPV had depressurized to the same pressure as the containment, the operators were eventually able to use fire trucks to inject water into the reactor vessel to cool the damaged fuel. It is not known exactly how hydrogen escaped from containment, but it led to the explosion on 12 March.

Units 2 and 3 had a different means of removing decay heat, known as the reactor core isolation cooling (RCIC) system, which used a steam turbine driven from the main steam line. The RCIC systems for both Units 2 and 3 were manually started when off-site power was lost. However, DC power was needed to control them, so once the tsunami struck, the systems could only be controlled in Unit 3, whose batteries had survived.

In Unit 2, the RCIC system was believed to have continued to operate without operator control until it failed about three days after the earthquake. The assessment is that in this uncontrolled mode the RPV water level rose to that of the steam lines, causing water carryover into the RCIC turbine, which in turn led to a reduced water injection rate and effective self-regulating level control at the elevation of the steam lines. It is also believed that sea water which had flooded the torus room, cooled the torus, which prevented overheating of the suppression pool and extended the RCIC's operating time. The operators were unable to open the containment vent in Unit 2, resulting in the PCV being maintained at a high pressure for a considerable period of time. On 15 March, the sound of an explosion was heard, followed by a drop in the Unit 2 suppression chamber pressure.

Unit 3 was the most complex accident to analyse because some systems did function and DC power remained available for some time. Because of this, operators had instrumentation and were able to control vital core cooling systems until the station batteries were depleted after about two days. Initially, the RCIC system was functioning and under control, enabling the water level in the reactor to be maintained. After about 20 hours, it failed for unknown reasons, causing the HPCI system to start automatically and inject water into the RPV. Since this is a relatively large capacity system, the operators reduced the flow rate by opening a return line to provide a bypass around the RPV. Even with this flow bypass, the HPCI system caused a dramatic reduction in RPV pressure, which led the operators to shut down the HPCI system after about 14 hours because they were concerned about pump shaft oscillation and failure. Following this, an attempt was made to use the fire trucks but this failed to inject much water into the RPV, leading to the core being uncovered and fuel failure.

On 13 March, the RPV was depressurized and the containment vent was opened which allowed the fire trucks to inject water into the RPV to cool the fuel. However, significant core damage had already occurred due to the prolonged period of high pressure and the lack of an available high pressure injection source. From this time on, the RPV remained depressurized, with water injection from the fire trucks continuing.

At the time of the accident, the fuel from Unit 4 had been removed from the reactor and was in the SFP at the top of the building. An explosion occurred which damaged the masonry structure around the SFP. Because the operators had no information about conditions in the pool, they became concerned that the fuel may be overheating and generating hydrogen itself. This led them to divert resources from re-establishing cooling to the other units. It is now understood that the explosion was a result of leakage of hydrogen from Unit 3 through shared systems.

Since the PCVs of the reactors at the Fukushima Daiichi NPP were not designed to withstand the severe accident conditions, containment venting systems had been installed. At various stages during the accident, the PCVs for Units 1, 2 and 3 either failed for unknown reasons or were opened via the containment hardened vents. Even when the containment hardened vents were successfully opened, the assessment, based on records of containment pressure, found that these PCVs eventually failed as well. The damage to the cores released large amounts of steam, hydrogen and radioactive material inside the PCVs through the SRVs and other leakage paths, which both pressurized and heated the PCVs, causing them to lose their integrity. This led to the release of steam, hydrogen and radioactive material into the reactor building.

Despite these failures, it is important to note that the PCVs performed their primary function. The eventual loss of integrity was a consequence of operation under temperatures and pressures far beyond their design limits, caused by overpressurization and the inability to remove heat for an extended period of time. Even with the PCV failures, leakage of radioactive material from the cores to the environment was likely mitigated by the suppression pools, which would have retained various fission products discharged from the RPVs through the SRVs.

Further work is needed in order to more fully understand additional details of the progression of the accident in the various units, particularly the failures which led to the RPVs of Units 1–3 becoming depressurized and to the PCVs of Units 1–3 being breached.

Regarding the application of DID, the tsunami was so powerful that it defeated Levels 1 to 3 by a common mode failure which meant that operators were forced to rely on provisions designed for Levels 4 and 5. The provisions designed for Level 4, such as containment venting and alternative water injection, proved inadequate which ultimately led to offsite releases.

2.2.14. Observations and lessons

- **Provisions need to be made for ensuring fundamental safety functions in case of loss of DID Level 3, including core cooling, spent fuel cooling and containment integrity.**
  Robust equipment to manage DID Level 4 provides the operators with greater capability to prevent an accident from progressing. For example, operation of the containment vent valves manually from a nearby shielded location would allow for the depressurization of the containments to aid in heat removal. Similarly, opening the SRVs from a nearby shielded location would have depressurized the reactors and allowed water injection to remove heat from the cores.
- **Personnel need to be trained to manage severe plant conditions (Level 4). This training needs to include consideration of the extreme environmental conditions which may prevail during a severe accident.**
  Although the operators had severe accident operating procedures available, the equipment and training on its use proved to be ineffective. Moreover, the possibility of a multi-unit severe accident or of an accident which simultaneously impacted other nuclear stations or damaged the local infrastructure had not been considered.
- **Defence in depth Level 4 provisions need to be independent from those of Level 3. They need to be sufficiently flexible and robust to make options for SAM available to the operators.**
  Although it is not possible to provide mitigating equipment for all possible severe accidents, thought needs to be given to providing the operators with flexible and robust options that would allow them to respond to several plant situations.
- **Interconnections between units need to be designed to prevent an accident from migrating from one unit to another.**
  The interconnection between Units 3 and 4 caused the migration of hydrogen from Unit 3 to Unit 4, causing an explosion in the Unit 4 reactor building. This explosion further complicated the response to the event by causing resources to be diverted to address the situation.
- **Critical instrumentation needs to be designed and maintained so that it continues to function during severe accidents.**
  Equipment important for providing information about the plant status needs to be resistant to severe accident conditions. For example, with instrumentation available in Units 1–3, the operators would understand what was happening in the reactors and would be able to take action to further mitigate the consequences of the event. Similarly, operator awareness of the real condition of the SFP in Unit 4 would have enabled them to know that, in spite of the explosion, there was no immediate threat from the fuel in the SFP in Unit 4.
- **Provision needs to be made for the removal of decay heat by alternative means (such as mobile equipment) should the permanently installed equipment not be operable.**
  The operators were aided in their attempts to restore cooling to the reactors by the connections that had been installed based on the experience of the Niigata-Chuetsu-Oki earthquake that affected the Kashiwazaki-Kariwa NPP in 2007. This allowed them to use diesel driven fire pumps or mobile pumpers to inject water into the reactors to cool the cores. This equipment needs to be reliable and diverse and has to be inspected regularly and tested to ensure its operability under station blackout conditions.

2.3. ASSESSMENT OF THE TREATMENT OF BEYOND DESIGN BASIS EVENTS

2.3.1. Introduction

This section reviews the beyond design basis accident (BDBA) probabilistic and deterministic safety assessments for the Fukushima Daiichi NPP and the insights from these assessments that were applied to the plant design. Of particular note are the measures against the extreme natural hazards that led to the total loss of AC power, referred to as a station blackout (SBO) because of its particular relevance to the accident. An SBO is often considered as a loss of normal and emergency AC power, which does not imply the loss of DC electrical supply or other AC power sources. The loss of DC power at

Fukushima Daiichi Units 1 and 2 played a central role in the development of the accident because it impeded the correct diagnosis of plant conditions, including the status and operation of safety systems.

Certain beyond design basis aspects are considered elsewhere. For instance, Section 2.1 addresses beyond design basis external event (BDBEE) assessments, while Section 2.4 focuses on provisions to mitigate severe accidents, such as the use of fire systems for core cooling injection.

The accident at the Fukushima Daiichi NPP exceeded its design basis in several respects. It was a severe accident, impacting multiple units, and it was an accident during which the operators were left with little indication to aid their understanding of what was happening, rendering them unable to control the situation. The interference between the units (hydrogen explosions and the allocation of limited resources) contributed to the severity of the accident. The level of devastation and isolation of the site, the scarcity of information available to the operators and the multiple unit effects come out as new features to be considered in accident management.

A major characteristic of the Fukushima Daiichi NPP accident was the massive common mode failure induced by the earthquake and subsequent flooding due to the tsunami. This common mode failure reached a scale considerably beyond that usually addressed in the assessment of BDBAs. This common mode failure impaired both the AC and DC electrical systems depriving the operators of almost all means of control over multiple units.

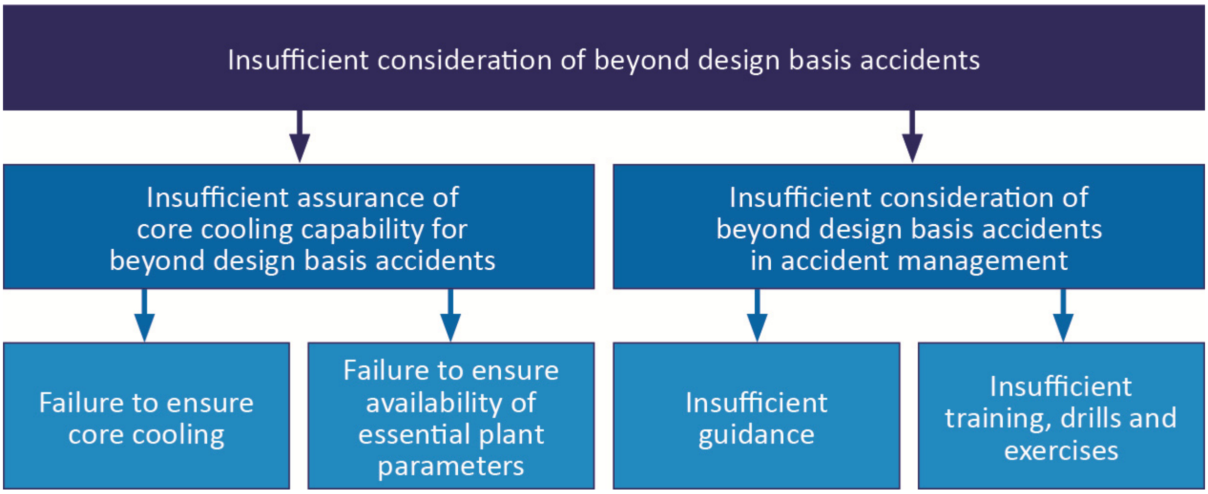


FIG. 2.3–1. Summary of insights from beyond design basis considerations.

Figure 2.3-1 describes some of the specific challenges encountered during the accident which are related to this lack of consideration of beyond design basis conditions. Some of the direct challenges in managing the accident were that components failed in such a way that core cooling was not maintained and the lack of critical safety parameter information did not allow the emergency response centres (ERCs) and main control rooms (MCRs) to have a clear understanding of the status of core cooling, to track the status of safety systems and to facilitate appropriate decision making. These challenges could have been mitigated by considering the ability to remove decay heat during beyond design basis conditions (see para. 6.37 of IAEA Safety Standards Series No. NS-R-1 [2]). Another challenge was that personnel were not prepared to manage the combined loss of power and instrumentation following flooding at the plant. Without proper plant instrumentation, as well as working in a harsh environment with inadequate SAMGs, plant personnel struggled to promptly implement the mitigation actions.

2.3.2. Review of deterministic safety assessment

2.3.2.1. IAEA requirements on deterministic safety assessment

A number of IAEA safety standards recommend that plant specific deterministic BDBA analyses be performed [5, 52] to investigate credible scenarios that lead to improvements in accident management. Thus, it is necessary to determine whether safety functions can be fulfilled in BDBA conditions and to demonstrate the capability of the design to mitigate such accidents.

Neither the Establishment Permit nor the periodic safety reviews (PSRs) for the Fukushima Daiichi units contained deterministic safety analyses of BDBAs, including severe accidents [3], contrary to what would be contained in Refs [5, 52]. One reason for this is that it was not a regulatory requirement, as was noted in the report of the IAEA Integrated Regulatory Review Service (IRRS) mission conducted in June 2007, which concluded that:

> "There are no legal regulations for the consideration of beyond the design basis, as Japanese plants are considered to be adequately safe as ensured by preventive measures" [53].

Some of the reasons why the SBO did not warrant detailed analysis and additional mitigation were an assumed high likelihood of grid recovery within a short space of time and design features to mitigate single unit SBO events such as unit electrical cross-connections [10]. Consequently, a multi-unit extended loss of AC power was not considered [54], though it should be kept in mind that the assessment of multi-unit events was not common practice for nuclear utilities prior to the Fukushima Daiichi accident. In addition, the reactors were equipped with systems able to cool the core for substantial periods of time in the absence of AC power, such as the isolation condenser (IC), reactor core isolation cooling (RCIC) and the high pressure coolant injection (HPCI) systems.

Furthermore, other BDBAs pertinent to the events of March 2011, such as the total loss of the ultimate heat sink, loss of DC power and the loss of SFP cooling, were also not assessed deterministically in the Establishment Permit or in the PSRs, which only include the analyses of design basis accidents.

The EDGs and battery rooms in the basement floor of the turbine building and the emergency electrical switch gear on the first floor of the turbine building were vulnerable to flooding, as was demonstrated by a plant event involving flooding at Unit 1 in 1991 [55]. Sea water leaked from a corroded pipe inside the turbine building, flooding an EDG room through its door and cable penetrations. Given the vulnerability to flooding from internal and external hazards of the EDGs, battery rooms and rooms containing emergency electrical switch gear, an assessment of flooding should have been performed and documented in the plant safety analysis. Such an assessment, required by IAEA safety standards [52], would have highlighted this vulnerability and the need for safety improvements, such as protecting the EDGs, battery rooms and switch gear from flooding.

TEPCO had performed some limited deterministic safety analyses of BDBAs in the early 1990s for those accident sequences that dominated the results of the PSAs. However, given that the PSA only studied single unit internal events, these deterministic safety analyses did not provide enough elements for accident management preparations to cope with the events that occurred in March 2011. Thus, although the plant was located in an area prone to tsunamis possibly larger than that considered in the design basis, these BDBA analyses did not include flooding of the site or the extended loss of AC power. At about the same time, the accident management guidelines were developed to support some of the generic BWR Owners' Group (BWROG) accident management updates. Thus, the severe accident operating procedures at the Fukushima Daiichi NPP contained some of the generic guidance provided by the BWROG. These updates were also inadequate in addressing the multi-unit event of March 2011, since the event substantially exceeded that postulated with particular reference to both the additional resources necessary to cope with such an event and the potential impact of one severe accident on adjacent units [10].

The extent of the deterministic BDBA analyses for the Fukushima Daiichi NPP were inadequate, which accounts partially for the inadequacies in the plant hardware, accident guidance and training to cope with the events of March 2011 [54]. More comprehensive deterministic BDBA analyses enables the identification of the plant vulnerabilities and highlights the need to protect the EDGs, battery rooms and emergency electrical switch gear from flooding. Without current beyond design basis assessments, the extent of the safety margin and the possible consequences of insufficient margin in beyond design basis modifications and BDBA management strategies cannot be determined. These beyond design basis assessments needed to include credible beyond design basis conditions (such as from internal and external flooding hazards which likely lead to the loss of all AC and DC power in the absence of adequate protection).

2.3.2.2. Ensuring the core cooling function under beyond design basis conditions

Paragraph 6.37 of IAEA Safety Standards Series No. NS-R-1 [2] states that: “Adequate consideration shall be given to extending the capability to remove heat from the core following a severe accident.” In addition, para. 4.81 of IAEA Safety Standards Series No. NS-G-1.9 [44] states that: “Equipment for emergency core cooling should be adequately protected from the consequences of internal and external hazards such as seismic hazards that have the potential to jeopardize its safety functions.”

Core cooling at the Fukushima Daiichi NPP was not maintained in Units 1–3, largely because they were not adequately protected from the consequences of external hazards, and insufficient provision was made to ensure core cooling in beyond design basis conditions. Three examples are described below that indicate some of the challenges experienced in maintaining or re-establishing cooling.

The first case involves the valves on the IC in Unit 1. By design, each division of the containment isolation system protective logic closes all valves within that division upon the loss of DC power to the protective circuit because containment isolation is required by the design basis. However, because the operators failed to notice that the system was isolated, it could not then be readily used for core cooling operation without operator intervention. In the March 2011 event, it was preferable to keep the IC operating. Given the loss of system indication (due to the loss of DC power) and the belief by the personnel within the ERC of its continued operation, attempts to recover core cooling using the IC were delayed and core melt resulted.

The second case concerns the safety relief valves (SRVs), which failed in the closed position upon the loss of DC power and high pressure air and could not be readily opened. However, the SRVs failing open would be detrimental to the operation of the RCIC and HPCI, and result in loss of coolant inventory prior to the availability of low pressure injection systems. If depressurization and injection with low pressure systems is a defined accident management strategy, then guidance in accident management procedures needs to be provided on how to achieve this under all plant conditions. This guidance was not readily available, which contributed to the delays in depressurizing the RPV in Units 2 and 3. This in turn contributed to the delays in establishing adequate low pressure injection flow, and exacerbated the consequences.

The third case relates to the containment ventilation system, where the vent isolation valves were closed and did not readily open upon the loss of DC power and high pressure air. The vent valve had to be operated in the field under challenging conditions due to the loss of PCV vent valve actuating power and compressed air, necessitating procedures to be developed and battery power provided to aid the local operation of these valves. The resultant delays in venting the PCV delayed the depressurization of the RPV (because the RPV pressure cannot be lower than the PCV pressure under these conditions), which in turn led to a delay in the injection of coolant into the reactor vessel from low pressure sources. The lack of coolant resulted in uncovering of the core, core damage, high radiation levels, hydrogen production and, ultimately, PCV failure. Thus, because the operators were unable to successfully operate the containment venting system, they were unable to reduce the containment pressure, which inhibited their efforts to cool the core. Accident management measures need to be in place to depressurize the containment earlier (in conjunction with a rupture disc design change) and to allow operators to implement strategies using low pressure water sources to limit or prevent damage to the reactor core.

From the above examples, it can be concluded that greater provision could have been made to ensure core cooling capability in beyond design basis conditions.

2.3.3. Review of the probabilistic safety assessment

2.3.3.1. IAEA requirements on probabilistic safety assessment

This section compares the scope and results of the PSA for the Fukushima Daiichi NPP with the requirements in IAEA standards. For example, para. 4.31 of IAEA Safety Standards Series No. GSR Part 4 [52] states that the "external events that could arise for a facility or activity have to be addressed in the [facility's] safety assessment, and it has to be determined whether an adequate level of protection against their consequences is provided." Paragraph 5.73 of IAEA Safety Standards Series No. NS-R-1 [2] reiterates the expectation that external hazards, in particular those unique to the plant site, are assessed and mitigated. IAEA Safety Standards Series Nos SSG-3 [56] and SSG-4 [57] also highlight the need for the verification of plant safety in relation to potential internal initiating events and internal and external hazards, specifically stating that consideration should be given to cases where one external hazard can induce other hazards (such as seismically induced flooding).

However, it was common practice for Japanese plants to perform PSAs of limited scope only, since existing regulations did not require more detailed assessments to demonstrate plant safety. Internal and external hazards were mostly excluded in the PSA, although NISA's instruction to NPP owners to perform seismic PSAs in 2006 had yet to be completed for the Fukushima Daiichi NPP. Thus, for this NPP, the PSA was limited to the determination of core damage frequency (CDF) and containment failure frequency arising from internal events. A more comprehensive assessment of the core damage risk would have indicated both how vulnerable the plant was to flooding and the high likelihood of flooding. A full scope Level 2 PSA would have indicated a low probability of success for severe accident interventions given the limited training and guidance provided in SAM and thus the plant's vulnerability to common cause effects.

There was an important precursor event involving flooding at the Fukushima Daiichi NPP Unit 1 in 1991 [55], when sea water leaked from a pipe inside the turbine building, flooding an EDG room. Both the EDG room and the seawater pipe were located in the basement of the turbine hall. The corroded pipe leaked water at a rate of 20 m^3/h, which penetrated the room with the reactor's EPS through the door and cable penetrations.

Given such a precursor, the frequency of flooding of the EDGs, battery rooms and both AC and DC electrical switch gear, all located in the lower levels of the turbine hall, would not be lower than 10^{-3} per reactor-year, which, given the challenges faced in mitigating the total loss of both trains of AC and DC electrical power, would likely result in core damage. Thus, an internal flooding PSA would have highlighted this risk, suggesting the need for safety improvements such as protecting the EDGs, battery rooms and switch gear from flooding or moving them to a higher elevation.

The Fukushima Daiichi NPP PSAs did not include tsunamis in their scope, even though they were a significant contribution to the overall risk [32]. Technical evaluations in 2001 of the 869 Jogan tsunami [58] concluded that the recurrence interval for a large scale tsunami was 800–1100 years and that since more than 1100 years had passed since the Jogan tsunami, the possibility of a large tsunami striking the Sendai plain was high. Their numerical findings indicate that a tsunami similar to Jogan would inundate the present coastal plain for about 2.5–3 km inland. TEPCO was aware in 2008 of the importance of evaluating the existence of a tsunami source along the Japan Trench off the coast of Fukushima Prefecture [54]. According to a report by the Institute of Nuclear Power Operations (INPO) [59], TEPCO derived in 2006 the probability of the Fukushima coast experiencing a tsunami greater than 6 m to be less than 1.0×10^{-2} in the next 50 years, and others had drawn a similar conclusion on the relatively high likelihood of a tsunami occurring that would exceed the design basis [60].

Given the location of the battery rooms, switch gear rooms and (some) EDGs in the lower levels of the turbine hall, it was known that flooding could lead to the total loss of all AC and DC power, which would likely result in core damage [10]. Thus, a basic tsunami PSA would have indicated a CDF greatly exceeding that reported for internal events and in excess of the IAEA target for existing NPPs of 1.0×10^{-4} per plant operating year [61] and would be considerably greater than the reported CDF for unit 1 of 3.9×10^{-8} per year. The large uncertainties associated with external hazards do not prevent their risk quantification since reporting uncertainties and performing sensitivity analyses are part of the normal PSA process. TEPCO has used these large uncertainties, combined with the lack of an established formal PSA tsunami methodology, as the reasons for not assessing and mitigating this risk [10].

Comprehensive level 2 PSAs tend to contain human error probabilities higher than those contained in level 1 PSAs because of the reduced guidance, training and knowledge of plant personnel on SAM. In addition, actions within a level 2 PSA are often implemented in a harsh environment, further reducing the probability of success. A comprehensive level 2 PSA for the Fukushima Daiichi NPP would have indicated the challenges experienced in reflooding a molten core using mobile equipment, and the necessity for design changes such as improvements in the ability to depressurize both the RPV and PCV, improved guidance, improved plant indication and improved training.

The limited Fukushima Daiichi NPP level 2 PSAs contained the use of the hardened containment vent system through the application of a fault tree approach to model the equipment failures. The human error probability for manual operation provides a failure probability of 1.9×10^{-3} for Fukushima Daiichi Unit 6. A more thorough assessment that investigated the challenges in SAM would have suggested the improvements that would have been made.

In summary, the Fukushima Daiichi plant design had some weaknesses that would have been discovered by performing a more comprehensive PSA, as recommended by IAEA safety standards [43, 61]. Rectification of these weaknesses may have prevented core damage. Examples include the lack of protection of the EDGs, battery rooms and switch gear from flooding, which would have been found by performing an internal hazards PSA, the lack of protection from tsunamis exceeding the design basis, and the low likelihood of success of severe accident interventions given the limited training, guidance and knowledge of plant personnel in this area. Hence, the calculated frequency of core damage for the Fukushima reactors was underestimated by the Japanese PSA requirements.

The Nuclear Safety Commission (NSC) released a report entitled Accident Management for Severe Accidents at Light Water Power Reactor Installations in May 1992. The amended version of this report, issued in October 1997 [62], established the PSA approach for internal events during plant operations. However, a PSA was not required for internals hazards (such as fire and flood) or external hazards (such as seismic activity and tsunami), and therefore these risks were not assessed. In the report, the NSC recommended that the regulatory body and utilities introduce accident management measures, although in the decision it was stated that:

— The safety of reactor facilities in Japan was sufficiently ensured by current safety regulations and the risk from reactor facilities was considered to be sufficiently low.
— The development of accident management measures was significant in further reducing the risk, which was already low.

This is one of the reasons why the PSAs for the Fukushima Daiichi units prior to the accident in March 2011, which were referenced in the PSRs, were limited to internal events. Internal hazards, external hazards, SFP accidents and multi-unit events were not considered in the PSAs.

After the Niigata-Chuetsu-Oki earthquake damaged the Kashiwazaki-Kariwa NPP in 2007, a PSA standard for seismic events was established and plant specific seismic PSAs were in the process of being finalized by TEPCO. Operating experience from the severe earthquake damage to the Kashiwazaki-Kariwa NPP had resulted in a design change to the Fukushima Daiichi units, with a flange being installed for the connection of mobile equipment. A seismic PSA should have highlighted a potential safety concern, since there was considerable pipework in the turbine building (including pipework carrying sea water through the turbine hall basement) that was not of a high safety grade. Consequently, it would have been assessed as having a significant likelihood of failure under severe seismic loading, which would then have resulted in the flooding of the EDGs and essential switch gear, leading to the failure of AC and DC power.

2.3.3.2. Results of the Fukushima Daiichi NPP PSA

The PSRs issued every ten years for each unit include the PSA results for internal at-power and shutdown events. For Fukushima Daiichi Unit 1, the third PSR was issued in November 2010. It reported a CDF of 3.9×10^{-8} per year and a containment failure frequency (CFF) of 1.3×10^{-8} per year. The CDF for SBO was 2.5×10^{-9} per year (for at-power internal events), which was only 6% of the total risk and was a small fraction of the overall calculated risk of a core damage event. The CFF from SBO was 1.5×10^{-9}, or 12% of the total.

These very low probabilistic results show the limitations of PSA modelling and assumptions applied for internal events. In addition, if internal hazards such as fire and flood and external hazards, such as seismic activity and tsunami, had been included, the risk of core damage from an extended loss of AC power would have been even greater, possibly by several orders of magnitude.

The CDF, CFF and SBO risks for the other units were similarly presented as being very low, as shown in the following:

— Unit 2 (from the second PSR issued in June 2001):
  - CDF for internal at-power events of 9.9×10^{-8} per year (SBO contribution of 44%, with a frequency of 4.3×10^{-8} per year);
  - CFF for internal at-power events of 1.1×10^{-8} per year (SBO contribution of 38%, with a frequency of 4.2×10^{-9} per year).
— Unit 3 (from the second PSR issued in March 2006):
  - CDF for internal at-power events of 1.3×10^{-7} per year (SBO contribution of 16%, with a frequency of 2.1×10^{-8} per year);
  - CFF for internal at-power events of 1.1×10^{-8} per year (SBO contribution of 11%, with a frequency of 1.2×10^{-9} per year);.
— Unit 4 (from the second PSR issued in March 2008):
  - CDF for internal at-power events of 1.6×10^{-7} per year (SBO contribution of 25%, with a frequency of 4.0×10^{-8} per year);
  - CFF for internal at-power events of 3.2×10^{-8} per year (SBO contribution of 20%, with a frequency of 1.6×10^{-8} per year).

Another reason why the SBO risk was reported as being low, and therefore not a major contributor, was the estimated high likelihood of recovering external AC power supplies (estimated at a 95% chance of recovery within the first 30 minutes) and the ability to cross-connect power supplies from an adjacent unit. Consequently, without considering multi-unit accident initiators and the extended loss of off-site power, which could result from external hazards, the SBO risk was very low.

Extreme external events may result in common mode failures, such as an SBO or a loss of ultimate heat sink (LUHS) if adequate provisions are not taken. In particular, flooding can cause massive common mode failures of all components which are not water resistant, whatever their redundancy, their diversity, their classification or their ranking in terms of defence in depth. Thus, with the inclusion of external hazards, an SBO would have been prominent among the overall risks of a severe accident.

The confirmed effectiveness of accident management measures by means of limited PSAs was also a factor contributing to the self-confidence about the high safety level of the plant. This self-confidence was derived from the limited scope of the PSAs, which did not take into account the importance of both internal and external hazards.

Venting of the PCV was considered in both the level 1 and level 2 PSAs for the units at the Fukushima Daiichi NPP. For accident sequences where heat removal from the PCV has failed, there is a possibility that the pressure of the PCV will be in excess of its design value before core damage occurs, in which case there is a procedure for using the hardened venting system to reduce this pressure. This is modelled in the level 1 PSAs. PCV venting in the level 2 PSA is assumed to occur at twice the design pressure at 8.53 kPa (gauge) for Unit 1 and 7.69 kPa (gauge) for Units 2–4. Procedures did not exist to address how to accomplish venting when all power, compressed air, lighting and indications are lost and when the actions need to be performed in a harsh work environment. These performance shaping factors were not adequately considered in deriving such a low human error probability of 1.9×10^{-3} for PCV venting. The use of mobile equipment, such as fire engines for water injection, was not considered within the PSAs.

2.3.3.3. Consideration of cliff edge effects

Paragraph 5.73 in IAEA Safety Standards Series No. NS-R-1 [2] states that: “A probabilistic safety analysis of the plant shall be carried out in order...to provide confidence that small deviations in plant parameters that could give rise to severely abnormal plant behaviour (cliff edge effects) will be prevented.”

A report had been issued by JNES [63] for the development of tsunami PSA methodology, where it was assumed that complete loss of the safety function occurs when the inundation height exceeds its installed level if additional flooding protection was not present. Given the location of the battery rooms, switch gear rooms and EDGs in the lower levels of the turbine hall, it could be foreseen that flooding would lead to total loss of all AC and DC power which would likely result in core damage.

The plant would have been strengthened if critical equipment had been protected from flooding or moved to a higher elevation, or if mobile equipment to mitigate flooding had been stored remotely with the ability to access the reactors. Such measures add additional defence in depth and so would have removed this cliff edge effect. After the accident in March 2011, NISA emphasized that preventing the loss of a safety function due to common cause failure (such as flooding) was of the utmost importance [64] requiring the application of the principles of diversity and independence.

2.3.4. Summary

The accident at the Fukushima Daiichi NPP exceeded the design basis of the units in several respects. It was a severe accident involving multiple units and it was an accident during which the operators had little information on what was happening, rendering them unable to control the situation. Neither the Establishment Permit (under which the NPP was constructed) nor the PSRs for the Fukushima Daiichi units included deterministic safety analyses of BDBAs, which was not in line with IAEA safety standards. Some of the reasons why the particular BDBA of SBO did not warrant detailed analysis and additional mitigation were an assumed high likelihood of grid recovery within a short space of time and the existence of design features to mitigate single unit SBO events, such as unit electrical cross-connections. Consequently, a multi-unit extended loss of AC power was not considered in the assessment. The other BDBAs pertinent to the events of March 2011, such as the total LUHS, loss of DC power and the loss of SFP cooling, were also not assessed deterministically in the Establishment Permit or in the PSRs, which contained only DBAs.

Once TEPCO began preparing probabilistic risk assessments (later PSAs) in the early 1990s, some limited deterministic safety analyses of BDBAs were performed for those accident sequences that dominated the results of the PSA. However, given that the PSAs only covered single unit internal events, these deterministic safety analyses did not help in the accident management of multi-unit events such as the one that occurred. These BDBA analyses did not include flooding of the platform or extended loss of AC power.

The inadequacy of the deterministic BDBA analyses partially accounts for the inadequacies in plant hardware, accident guidance and training to cope with the events of March 2011. A comprehensive deterministic BDBA analyses allows for the identification of plant vulnerabilities and the need to protect the EDGs, battery rooms and emergency electrical switch gear from flooding.

It was common practice for Japanese plants to perform only limited scope PSAs, since existing regulations did not require more detailed assessments to demonstrate plant safety. Internal hazards (such as fire and flood) and external hazards (both natural and human-made) were mostly excluded from the PSAs, although NISA had instructed NPP owners in 2006 to perform seismic PSAs. The PSA for the Fukushima Daiichi NPP had yet to be completed and it was limited to the determination of CDF and CFF arising from internal events. A more comprehensive assessment would have indicated both how vulnerable the plant was to flooding and the high likelihood of flooding. A full scope level 2 PSA would have indicated a low probability of success for severe accident interventions given the limited training and guidance provided in SAM and the plant’s vulnerability to common cause events.

2.3.5. Observations and lessons

— **Deterministic and probabilistic beyond design basis safety analyses need to be comprehensive and take into account both internal and external events, including internal flooding and external hazards such as seismic events and flooding.**
The combination of DSA and PSA needs to be used to assess factors such as cliff edge effects, realistic equipment and personnel performance, and the relative contribution of various accident sequences to the overall plant risk. For the sequences with the highest possibility of contributing to core damage or challenging the containment integrity, the operating organization should evaluate the need for taking actions to mitigate the consequences of these sequences.
— **Extremely low numerical values from PSAs need to be reviewed and confirmed.**
Comparison of the CDF calculated for the Fukushima Daiichi NPPs to the worldwide average for similarly designed BWRs would have indicated that the values calculated by TEPCO were at least two orders of magnitude lower than the other plants. This difference should have been investigated which may have highlighted weaknesses in the procedures and training being used at the Fukushima Daiichi NPP. However, this investigation was not conducted, which highlights the

issue that numerical values obtained from PSAs need to be used with caution when making decisions about the overall safety of the plant.

2.4. ACCIDENT MANAGEMENT PROVISIONS AND THEIR IMPLEMENTATION

2.4.1. Introduction

This section summarizes the status and implications of the accident management provisions available at the Fukushima Daiichi NPP. In accordance with IAEA terminology, both preventive (before core melt) and mitigative (after core melt or severe accident) parts of accident management are covered. All components of accident management (AM) are discussed, including hardware provisions, emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs), human resources and organizational arrangements, including training and drills. The interface with the emergency arrangements is also discussed. Gaps in the provisions are identified based on a comparison of the situation prior to the accident with the relevant IAEA safety standards in effect at that time. The final part of the section is focused on the assessment of actual plant staff actions during the accident, discussing the key factors causing severe consequences of the accident.

2.4.2. Accident management provisions available at the Fukushima Daiichi NPP

2.4.2.1. National regulatory requirements related to accident management

The AM provisions available in Japan before the Fukushima Daiichi accident have been most comprehensively described in Accident Management for Severe Accidents at Light Water Power Reactor Installations, issued by the Nuclear Safety Commission (NSC) on 28 May 1992 [62]. This document describes the role and significance of AM, both preventive and mitigative parts of AM, and the results of the technical evaluation of provisions relevant for AM in BWRs and PWRs in Japan. Practices and measures taken in different countries and associated issues are described. In the conclusions, it was stated, based on a level 1 PSA, that the possibility of a severe accident in Japan was sufficiently low. Nevertheless, it was strongly recommended that AM measures be developed and implemented. Based on the document, there was an NSC decision of 28 May 1992 [62], amended on 20 October 1997, which strongly recommended that the regulatory body and utilities introduce AM measures at NPPs.

The NSC decision stated that:

— The safety of reactor facilities in Japan is sufficiently ensured by current safety regulations and the risk from reactor facilities is considered to be sufficiently low.
— The development of AM measures is significant in further reducing the risk, which is already low.
— Effective AM should be developed by licensees on a voluntary basis, and its proper implementation in the event of an emergency is strongly recommended.
— In order to further improve the safety of reactor facilities, licensees should continue to develop AM using specific proposals made by NSCRG: L-AM-II.01 [62].
— Government agencies should define their roles in the promotion and development of AM, and should continue to work on the details of these roles.
— For reactor facilities in operation or under construction, a report was expected from the government agency to the NSC review on AM implementation policies.
— The authorities concerned and licensees should continue severe accident studies.

Although the document describes the components of AM, it is clear that it is neither a regulation nor a regulatory guide which follows the IAEA safety standards, and that implementation of measures was left to the operators on a voluntary basis. There does not seem to have been an update of the document since 1997, so more stringent recommendations that were included in the relevant IAEA safety standards on design (IAEA Safety Standards Series No. NS-R-1 [2], in 2000) or on SAM (IAEA Safety Standards Series No. NS-G-2.15 [49], in 2009) were not reflected in the NSC guidance.

2.4.2.2. Relevant findings from periodic safety reviews and international missions

The IAEA conducted an Integrated Regulatory Review Service (IRRS) mission from 25 to 30 June 2007 to the Japanese regulatory body, the Nuclear and Industrial Safety Agency (NISA) [53]. Although the mission did not explicitly deal with the safety assessment of individual NPPs, in reviewing the regulatory arrangements it identified some issues relevant for the area of AM. In the mission findings, the continuous efforts by NISA to update the legislative and governmental framework was acknowledged, but the mission also noted that:

> "There are no legal regulations for the consideration of beyond the design basis, as Japanese plants are considered to be adequately safe as ensured by preventive measures. The regulatory body has strongly requested licensees to voluntarily implement severe accident management (SAM) and carry out probabilistic safety assessment (PSA) including preventive and mitigatory measures in line with the guide for SAM review prepared by METI. Accident management measures are taken by licensees on a voluntary basis" [53].

Subsequent to this finding, the mission suggested that NISA should continue to develop a systematic approach to the consideration of BDBAs, and the complementary use of PSA and SAM in the assessment process for risk reduction purposes. In addition, the mission also suggested that the PSR should be more focused in order to provide a comprehensive picture of the plant safety status at certain intervals.

In the mission report it was also stated that all Japanese NPPs older than ten years, including the Fukushima Daiichi NPP, were the subjects of PSRs. The latest PSR for the Fukushima Daiichi NPP was performed in 2010 for Unit 1 and earlier for other units. However, there was no finding in the area of AM, nor were any relevant plant modifications proposed in the reports.

After the IRRS mission, several discussions regarding the need for the development of AM regulations or updating of the guidelines were initiated (including shortly before the accident), but no effective action was taken.

Although the IAEA, through its Operational Safety Review Team (OSART) missions, was invited to some Japanese NPPs, it was not invited to the Fukushima Daiichi NPP, nor was the World Association of Nuclear Operators (WANO) and its review missions.

2.4.2.3. Basic approach and strategies for implementation of accident management programmes

The recommendations in the area of AM made by the NSC (briefly described in Section 2.4.2.1) have been implemented in accordance with the guidelines contained in Refs [65, 66].

In response to the decision by the NSC given in NSCRG: L-AM-II.01 [62], the Ministry of International Trade and Industry (MITI), which was the regulatory body for NPPs at that time, encouraged the utilities in July 1992 to establish AM implementation plans using the insights obtained from PSAs. The utilities submitted their AM implementation plans to MITI in March 1994. MITI reviewed these plans and prepared a report in October 1994 entitled Accident Management for Light Water Nuclear Power Reactors [67]. In the report, MITI recommended that the utilities undertake the AM implementation plans, prepare AM operating procedures and establish the necessary administrative framework by 2000.

The utilities completed implementation of AM measures by February 2002 and reported to NISA, which was the new regulatory body for NPPs. In addition, the utilities submitted evaluations of the effectiveness of AM measures for eight representative BWR and PWR plants to NISA. After reviewing the evaluations, with the assistance of the Japan Nuclear Energy Safety Organization (JNES), NISA confirmed their validity. NISA recognized that it was also important to evaluate the effectiveness of AM measures for NPPs other than the eight representative plants and requested the utilities to perform evaluations of all plants. The utilities performed evaluations of the effectiveness of AM measures for individual plants in two steps: through level 1 and level 2 PSAs. From the results of the first step, summarized in [68], it is seen that:

— For the BWR-3 Mark-1 type, implementation of the AM measures led to a reduction in the core damage frequency (CDF) from 4.3×10^{-7} to 2.8×10^{-7} per year (discounting internal and external hazards which were not assessed).
— For the BWR-4 Mark-1 type, implementation of the AM measures led to a reduction of the CDF from 3.5×10^{-7} to 1.4×10^{-7} per year.
— For the BWR-3 Mark-1 type containment, implementation of AM measures led to a reduction in containment failure frequency (CFF) from 3.7×10^{-7} to 2.1×10^{-8} per year.
— For the BWR-4 Mark-1 type containment, implementation of the AM measures led to a reduction in the CFF from 2.8×10^{-7} to 5.5×10^{-8} per year.

The results were submitted to NISA in March 2004 as PSA Evaluation Report after AM implementation. NISA reviewed these reports with assistance from JNES and published an evaluation report of PSAs for all existing operating plants [69]. Based on comments from NISA, the PSA models were modified and new results were obtained as follows (examples for existing BWR-4):

— Frequency of core damage: 1.6×10^{-7}/reactor-year;
— Frequency of containment failure: 1.2×10^{-8}/reactor-year.

The assumed effectiveness of the AM measures, confirmed by PSA studies, was also a factor contributing to the confidence about the high safety level of the plant, although it was derived from limited scope PSAs which did not take fully into account the importance and significant contribution to risk of both internal and external hazards.

2.4.2.4. Summary of accident management provisions available at BWR plants

The AM provisions implemented at Japanese BWRs are summarized in Tables 2.4–1 and 2.4–2. The utilities selected AM measures focusing on the essential safety functions of NPPs. Specifically, reactor shutdown, coolant injection to the reactor vessel and the containment vessel, heat removal from the containment vessel, and supporting function to the safety systems were chosen as the four essential functions for BWRs, after which relevant AM measures were selected for each safety function.

TABLE 2.4–1. SAM FOR BWRs IN JAPAN — PREVENTIVE AM MEASURES

AM functions	Equipment and systems	Accident sequences	Details
Reactor scram	ARI activation with signals for high pressure of the RCS, or low liquid level. RPT activation with the same signals above.	Transients without scram.	New signals are independent of conventional scram and ECCS signals. These systems have already been implemented to ABWR in the design stage.
Depressurization	The ADS is activated by a signal with low liquid level of the reactor vessel.	Transients with failure to depressurize.	Not applied to BWR-3 and ABWR. BWR-3: isolation condenser was implemented. ABWR: high pressure ECCSs were already reinforced.
Alternative water injection	Use of the make-up line. Water supply from the fire protection system.	Transients with loss of ECCS injection.	—
Alternative heat removal	Containment hardened vent.	Transients with loss of decay heat removal.	—
Supply of AC power	Accommodation of 6.9 kV and 480 V from the adjacent plant.	Loss of all AC power.	This AM measure is applied to the Fukushima Daiichi NPP.

ABWR: advanced boiling water reactor; AC: alternating current; ADS: automatic depressurization system; AM: accident management; ARI: alternate rod insertion; BWR: boiling water reactor; ECCS: emergency core cooling system; RCS: reactor coolant system; RPT: recirculation pump trip.

TABLE 2.4–2. SAM FOR BWRs IN JAPAN — MITIGATIVE AM MEASURES

AM functions	Equipment and systems	Accident sequences
Depressurization (same as prevention)	The ADS is activated by a signal with low liquid level of the reactor vessel.	Transients with failure to depressurize.
Hydrogen control	Inerting of the primary containment by nitrogen.	Any accident including severe accident.
Alternative water injection to the reactor core	Use of the make-up line. Water supply from the fire protection system. External connection allows the use of mobile equipment such as fire trucks.	Transients with loss of ECCS injection.
Alternative water injection to containment	Use of the make-up line. Water supply from the fire protection system. External connection allows the use of mobile equipment such as fire trucks.	Same as above.
Alternative heat removal	Use of the dry well cooler and use of the heat exchanger in the make-up line. Recovery of the RHR system. The containment hardened vent.	Same as above.
Supply AC power	Accommodation of 6.9 kV and 480 V from the adjacent plant. Power supply from the EDGs.	Loss of all AC power.

ADS: automatic depressurization system; AM: accident management; ECCS: emergency core cooling system; EDG: emergency diesel generator; RHR: residual heat removal.

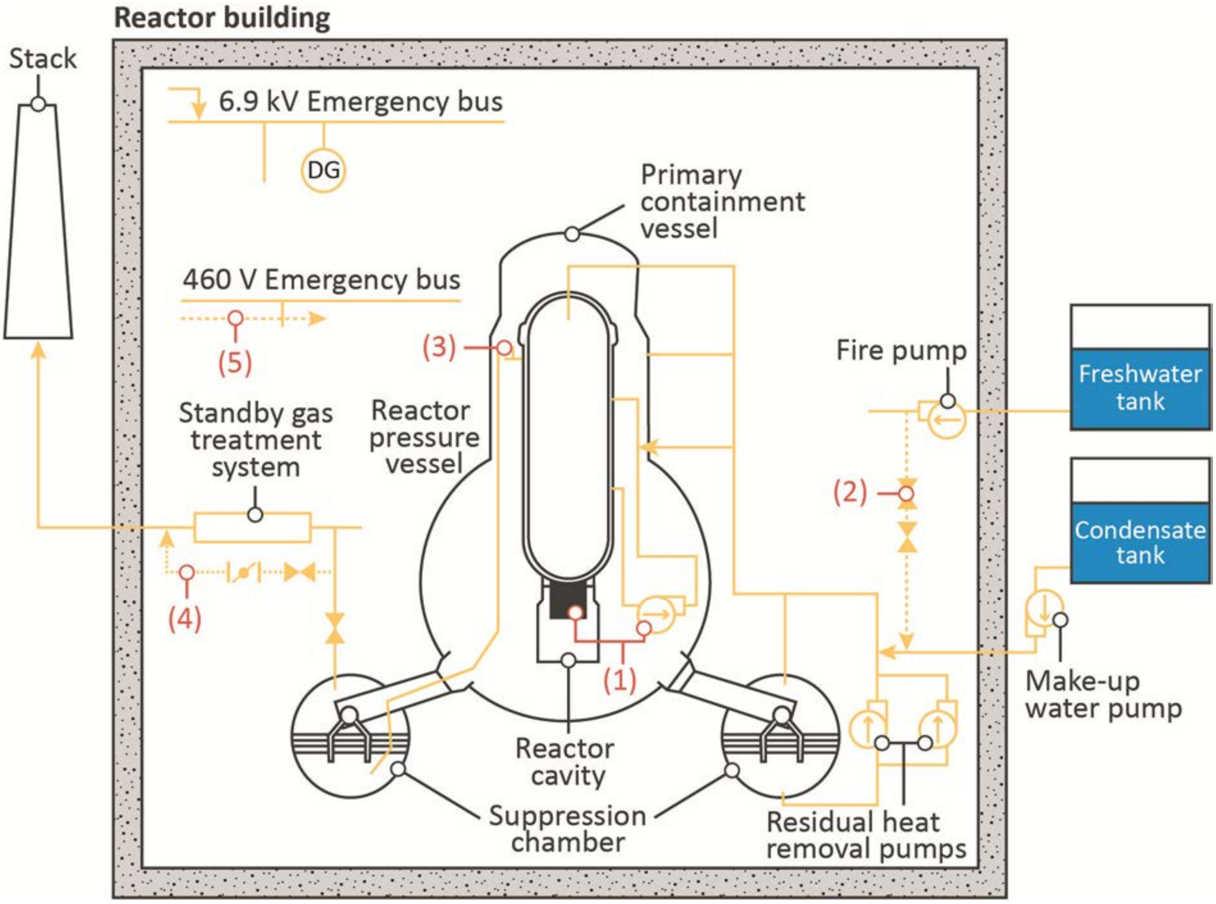


FIG. 2.4–1. Examples of AM actions for BWRs in Japan (DG: diesel generator).

Examples of AM measures are illustrated in Fig. 2.4–1:

(1) Reactivity control (automatic insertion of control rods or automatic stopping of reactor recirculation pumps);
(2) Water injection to the reactor and reactor containment vessel (pumps of condensate make-up water system or fire protection system);
(3) Reactor depressurization system;
(4) Heat removal from the reactor containment vessel (through dry well cooler and reactor coolant cleanup system or hardened containment vent piping);
(5) Means for power supply from adjacent nuclear units.

In addition to the hardware provisions explained above, the organizational provisions that have been established in Japanese NPPs for implementation of AM hardware are summarized as follows [66]:

— Organization: To implement the AM provisions, supporting organizations, in addition to the operators, are also involved focusing on technical evaluation, communication, dose assessment and restoration. Operational actions are carried out mainly at the discretion of a central control room. Furthermore, when an accident has escalated to core damage and requires a decision, the supporting organization provides the technical evaluation to support the decision making of operators. There was a seismically robust technical support centre (TSC) at the plant that was suitable for the supporting team.
— Procedures: Consistent with the plant upgrades, the electric utilities have prepared procedures for operators and for supporting organizations based on the progression of the event. For operators, in addition to emergency operation manuals (for BWR plants) to prevent core damage, emergency

operation manuals for post-core damage were provided to respond to a severe accident. Procedures and manuals, developed in 1987 in the USA by the BWR Owners' Group (BWROG), were used as a basis for development of the plant specific documents. Japanese BWR utilities conducted a joint research programme to develop unique accident management guidelines. These utilities also benchmarked the BWROG procedures in the joint research programme. For supporting organizations, in order to judge the mitigation measures in a comprehensive manner after core damage, the AM guide included procedures and criteria, knowledge bases such as technical data, and projected consequences.
— Education, training and communications system: The electric utilities have programmes in place for periodic education to enhance knowledge on plant behaviour during a severe accident and drills for examining the effectiveness of the organizations for implementing AM. A desktop training method was used for AM training, with the scope and frequency of training established in the plant's technical specifications. The knowledge base consists of information, including types of information and criteria for identifying operating status and making correct decisions during the implementation of the AM measures. Furthermore, systems have been developed to provide training to notify and communicate with the internal and external organizations about the plant conditions and the status of implementation of AM measures, and also to exchange information and to receive guidance and advice.

The course and consequences of the accident demonstrated that the organizational provisions were not adequate.

2.4.2.5. Accident management provisions at the Fukushima Daiichi NPP Units 1–4

From the point of view of AM, the similarities and differences between the individual units at the Fukushima Daiichi NPP can be summarized as follows:

— Unit 1 of Fukushima Daiichi was equipped with the BWR-3 type, while Units 2–4 were equipped with the BWR-4 type; however, this difference was not essential for AM.
— All units had the same Mark-I type containment.
— For a reactor scram, all of the units had the same provisions, consisting of a control rod drive hydraulic control system (CRDHCS) and standby liquid control system (SLCS).
— Automatic depressurization was available in Units 1–4. All units had provision for 6.9 kV and 480 V emergency electric power supply from adjacent units.
— All units had two EDGs.
— For emergency core cooling at high pressure, all units had steam driven high pressure coolant injection (HPCI).
— Unit 1 had two trains with isolation condensers (IC).
— Units 2–4, of more recent design, had, instead of ICs, a reactor core isolation cooling (RCIC) system.
— For emergency core cooling at low pressure, all units had two trains of core spray (CS); in addition to CS, Units 2–4 also had two trains of a low pressure coolant injection (LPCI) system.
— For containment heat removal, Unit 1 had two trains of a shutdown reactor cooling system and two trains of containment cooling systems (CCSs). On the other hand, Units 2–4 had just two trains of a residual heat removal (RHR) system.

2.4.3. Comparison of accident management provisions at Fukushima Daiichi Units 1–4 with IAEA safety standards

This section includes a comparison of AM provisions available at the Fukushima Daiichi NPP prior to the accident with the IAEA safety standards that were in effect at the time of the accident. Although severe accidents were not part of the design basis for existing plants, they were sufficiently well established and described in a number of IAEA safety standards and other guidance published in reaction to the Three Mile Island and Chernobyl accidents, long before the Fukushima Daiichi accident.

The assessment in this section is based on a comparison with IAEA Safety Standards Series No. NS-G-2.15 [49], issued in 2009, while hardware and procedural provisions at Fukushima Daiichi were implemented in 2002, with some updating in 2007. Even though the safety standard was issued following the hardware modifications, such an assessment is still considered as a source of lessons learned.

The most important principles in the IAEA safety standards available prior to the Fukushima Daiichi accident are summarized in the text below, followed by the assessment of the provisions available at the NPP. Information about the equipment, procedures and staffing of the NPP presented in Technical Volume 1, as well as relevant sections of Technical Volume 2, were extensively used for the assessment. Among other sources, Refs [70, 71] provided additional valuable inputs.

2.4.3.1. Availability and results of plant specific analytical studies supporting the development of procedures and guidelines

Recommendations in IAEA safety standards

In accordance with IAEA Safety Standards Series Nos NS-R-1 [2], GS-G-4.1 [5], NS-G-1.10 [72] and NS-G-2.15 [49], the development of the AM programme should be based on insights into possible plant damage states, which can be obtained by best-estimate analysis using up to date, validated computer codes in all phases of the development, from the initial considerations up to the validation of procedures and guidelines. Analyses are needed to investigate the effectiveness of the proposed actions, including assessment of potential negative consequences of the actions. Uncertainties in the determination of timing and severity of phenomena should be taken into account in the use of the results. If generic results are used, they should be adapted to the specific plant. Use of PSA levels 1 and 2 is encouraged as a valuable input for the development of AM guidance. However, in view of the inherent uncertainties involved in determining credible events, the PSA should not be used a priori to exclude accident scenarios from the development of SAM guidance. Identification of potential challenges should be sufficiently comprehensive to provide a basis for guidance for plant personnel even if the scenario constitutes a very unlikely path within the PSA or is not identified in the PSA at all.

Assessment of the plant status using the applicable IAEA safety standards

As described in Section 2.3, the approach to implementation of SAM provisions was based mainly on a generic risk informed approach, where the BWROG analysis was reviewed for the PSA impact. Several results of analysis considered relevant for the development of SAM strategies (symptoms, time windows, prioritization and effectiveness of the strategies, including recovery actions, determination of environmental conditions both for staff as well as equipment) were not included in sufficient detail. Sensitivity studies with varying values of symptoms and time windows important for demonstration of the effectiveness of the strategies were not covered. It is clear that there was no analysis considering the possibility of recovery actions in the worst case scenario. It was assumed that even during the severe accident progression there would be some means available for successful recovery actions, the feasibility of which was not investigated in sufficient detail.

2.4.3.2. Procedures and guidelines for accident management

Recommendations in IAEA safety standards

In accordance with IAEA Safety Standards Series Nos NS-R-1 [2], NS-R-2 [73], NS-G-2.2 [74] and NS-G-2.15 [49], development of an AM programme consisting of EOPs and SAMGs is a requirement to be implemented at any NPP. For the beyond design basis area the emergency operating procedures (EOPs) and SAMGs should be symptom based, with clear interfaces between them. They should cover prioritization of actions, initiations and termination criteria, time windows for actions and their duration, equipment and resources needed for an action, cautions in performing the action and monitoring the plant response. Procedures and guidelines should consider using any plant equipment available and should be consistent with available hardware provisions. Possible positive and negative consequences should be specified in using the SAMGs. The AM guidance should address the full spectrum of credible BDBAs; that is, all accidents considered credible on the basis of possible initiating events, and possible complications during the evolution of the accident that could be caused by additional hardware failures, human errors and/or events from outside including issues which are not considered in the safety analysis.

The procedures and guidelines should be written in a user friendly way so that they can be executed under high stress conditions. Pre-calculated graphs or simple formulas (computational aids) should be developed in order to facilitate decision making. There should be adequate background material available to provide information for understanding the procedures and for training. In the development of EOPs and SAMGs, habitability of the control places should be taken into account.

EOPs and SAMGs should be independently reviewed, and they should be validated in full using appropriate simulation methods (simulator, computer simulation, tabletop exercise). Validation should be done under conditions that realistically simulate conditions present under an emergency.

Assessment of the plant status using applicable IAEA safety standards

Accident management guidance in place at the Fukushima Daiichi NPP units at the time of the accident included abnormal operating procedures (AOPs), EOPs, and severe accident operating procedures used by control room operators. In addition, accident management guidelines (AMGs) were in place for use by the technical support staff in the Emergency Response Organization. Collectively, these documents covered the realm of response from abnormal conditions, DBAs and BDBAs, including severe accidents.

The plant specific EOPs were based on generic emergency procedure guidelines (EPGs) developed by the BWROG in January 1987 [75]. In 1992, the Nuclear Safety Commission (NSC) recommended to Japanese utilities that they implement AM measures to address severe accident conditions. In response, the Japanese utilities performed joint research programmes, which included plant specific evaluations using PSA methodology and the development of AMGs. The completed AM reports were submitted to the Ministry of Trade and Industry (MITI) in March 1996. Based on insights from the PSAs, modifications to the Fukushima Daiichi NPP were made and AM measures, including severe accident operating procedures and AMGs, were implemented between 1997 and 2002. These improvements included the use of alternate injection systems such as the water used for fire fighting and MUWC system, a hardened primary containment vent, alternate rod insertion (ARI) system, and reactor recirculation pump trip (RPT) system. After this effort was completed, a report was submitted to NISA summarizing the actions that were implemented to address severe accident conditions.

During the late 1990s, the US nuclear industry undertook a significant effort to identify vulnerabilities and to develop guidance to address severe accidents which involved the loss of core cooling and subsequent core damage. This industry effort was conducted in accordance with NEI 91-04, Severe

Accident Issue Closure Guidelines [76]. The SAM closure process, as described in NEI 91-04, consisted of the following steps:

- Evaluate industry developed bases and owners group SAMGs, along with the plant individual plant examination (IPE), individual plant examination of external events (IPEEE) and current capabilities, to develop SAMGs for accidents found to be important to each plant;
- Interface SAMGs with the plant's emergency plan;
- Incorporate severe accident material into appropriate training programmes;
- Establish a means to consider and possibly adopt new severe accident information from self-assessments, applicable regulatory communications, PSA studies, etc.

In support of this effort, the BWROG performed a number of improvements to the emergency procedure guidelines (EPGs) and created new SAGs. Revision 2 [77] to the BWROG EPGs/SAGs was implemented in March 2001 and was the most current revision at the time of the Fukushima Daiichi accident.

Some of the key improvements in accident response that were implemented in the BWROG EPG/SAGs included:

- New SAGs which specify optimized strategies for events in which core cooling can no longer be ensured. The SAGs provide specific strategies to deal with phenomena such as core melt progression, recriticality, creep rupture, high pressure melt ejection, Mark I containment liner challenges, hydrogen generation, core–concrete interaction and fission product release.
- Comprehensive anticipated transient without scram (ATWS)/stability modifications, including the injection of soluble boron to mitigate the consequences of large irregular oscillations by limiting their duration.
- Optimized hydrogen control strategies, including strategies to address combustible gas concentrations above the deflagration limits.
- Addition of the IC as a system that may be used to rapidly depressurize the RPV, since the IC rejects heat outside containment with no net loss of inventory and provides a relatively large heat sink.
- Revised guidance on the utilization and termination on dry well and suppression pool sprays, including the authorization to use sprays for fission product scrubbing at low pressures or if the containment has failed. This included the use of alternate systems if normal spray systems are unavailable.
- Restricted the use of external injection and spray sources only under conditions which would not challenge primary containment integrity.
- Authorized defeating interlocks such as HPCI and RCIC high temperature trips and suppression pool and dry well spray interlocks under certain conditions.
- Added new cautions to warn of the effects of high cooling water temperatures upon HPCI and RCIC turbines and to emphasize that reducing primary containment pressure will reduce available net positive suction head (NPSH) for pumps drawing suction from the suppression pool.

Although TEPCO was a member of the BWROG and had acquired the updated BWROG EPG/SAG products, they had not implemented these improvements in AMG at their NPPs. Delays in implementing the improved industry guidance was based on the belief that their existing procedures and guidelines were adequate and the need to prioritize station resources on addressing other issues at the plant.

Although the relevant procedures were available in the plant, their effective use was not feasible. From the description of the provisions available in the plant, it is clear that the AM programme was developed with the assumption that AC power would always be available, either from its own power sources in the unit or that it could be easily re-established by connecting with the neighbouring units.

In addition, it was assumed that DC power would always be available for the proper functioning of the instrumentation and for manipulating valves. These assumptions turned out to be incorrect. For these reasons, the procedures did not cover contingency actions for situations without the availability of instrumentation needed to show key plant parameters. The plant procedures were not established for performing the action, when all safety related electrical distribution systems, and subsequently many of the safety systems, become inoperable due to a single event. In the absence of AC power, with limited possibilities of manual actions and without adequate information about the plant status, it was very difficult to perform any effective actions.

Although mobile sources of coolant (such as fire trucks) were available on the site and provisions for connection of these sources to the plant had been made, the use of these sources failed or was delayed either due to the absence of adequate procedures or other obstacles in the implementation of the procedures. No alternative methods were prepared for the assessment of the plant status, in the case of complete absence or very limited quantity of information obtainable from the instrumentation. Operating procedures also did not adequately cover the situation of prolonged loss of power, e.g. how to vent the containment under conditions associated with limited possibility of monitoring the plant status.

2.4.3.3. Hardware provisions for SAM and their survivability under conditions of extreme external hazards and severe accident conditions (including instrumentation and power supply)

Recommendations in IAEA safety standards

Guidance on hardware provisions is described in IAEA Safety Standards Series Nos NS-G-1.10 [72] and NS-G-2.15 [49]. According to these guides, the plant should be equipped with adequate hardware provisions in order to fulfil the fundamental safety functions as far as is reasonable for BDBAs and severe accidents as well. Modification of the plant should be considered whenever needed for the development of a meaningful AM programme. Hardware implementation or upgrading should focus on preservation of the containment function.

The equipment needed to mitigate a severe accident should be, to the extent possible, independent of the equipment available to fulfil design basis requirements. Special attention should be devoted to the I&C as an essential component for effective AM. The plant parameters needed for AM measures should be identified and be available from the instrumentation. Dedicated instrumentation that is qualified for the expected environmental conditions is the preferred method to obtain the necessary information. The ability to infer important plant parameters from local instrumentation or from unconventional means should also be considered. The need for the development of computational aids to obtain information where parameters are missing or their measurements are unreliable should be identified and aids developed accordingly.

Appropriate measures should be taken to remove the decay heat from the corium debris to an ultimate heat sink. Where it is needed to remove the decay heat by repeated or continuous venting of the containment atmosphere, such venting should take place through a pathway that can provide an appropriate reduction in the fission product releases, e.g. by filtering or scrubbing.

Assessment of plant status against IAEA safety standards

Although some design provisions aimed at strengthening AM capabilities were implemented at the Fukushima Daiichi NPP, there remained several weaknesses that contributing significantly to the transition of the initiating event into a severe accident in parallel at several units. The most important issues were as follows:

— Difficult access to some manually operated valves under degraded environmental conditions;

— Limited means of communication for conditions of prolonged loss of power supply;
— No hydrogen countermeasures in the reactor building, outside the primary containment vessel;
— Insufficient radiation protection equipment available for response to the accident;
— Insufficient supply of fuel and other consumables needed for continuous operation of the fire trucks and of compressed air for the manipulation of valves;
— Unavailability of monitoring equipment independent of DC power (pressure and temperature monitors) as well as equipment allowing local manipulation of valves. Lack of independent battery powered lighting of the MCR and access pathways.

2.4.3.4. Organization and arrangements of the licensee to manage accidents (including training and drills)

Recommendations in IAEA safety standards

IAEA Safety Standards Series Nos NS-R-2 (superseded by SSR-2/2) [73, 78], NS-G-2.8 [79] and NS-G-2.15 [49] require that there should be appropriately staffed and qualified teams responsible for SAMG execution as well as for the technical support centre (TSC). The functions and responsibilities of the staff should be clearly specified, considering that there is a need for specialized experts serving as evaluators, decision makers and implementers of the actions. The transfer of responsibilities between the teams should be clearly defined. Integration of the AM programme within the overall emergency arrangements for the plant should include possible interaction between security measures and the execution of AM actions. Reliable methods of communication, under conditions of prolonged SBO, should be available, in particular between the MCR, TSC and the ERC. Criteria for activation of the TSC should be specified. Decision making authority should be assigned to a high level plant (or site) manager.

In order to comply with these needs, there should be adequate training organized by professionals for all individuals and groups involved in the application of the AM programme, using a combination of classroom training, exercises and drills. Training on severe accident phenomena at the necessary technical level to the appropriate individuals should be provided. The training should cover the use of non-conventional line-ups and equipment, such as mobile equipment. Refresher training should be organized at appropriate intervals, with periodic confirmation of competences of the personnel. Exercises and drills should require the application of a substantial portion of the overall SAMG package and should involve the participation of all individuals and groups, with local, national and, where appropriate, international participation. It should be verified that the emergency response and all related duties can be carried out in a timely manner according to the schedule and under conditions of stress. There should be full scale exercises involving external organizations such as police, fire services, ambulance teams, rescue teams and other emergency services. Training programmes should be reviewed and updated periodically.

Assessment of the plant status against IAEA safety standards

Organizational arrangements and the managerial decisions taken were appropriate for normal operation. However, in spite of large human and equipment resources, summarized in Section 1.2 of Technical Volume 1, the arrangements and decisions were inadequate for this emergency, as explained below:

— Communication between the main control room and the site ERC was insufficient or ineffective.
— The operators did not have adequate access to the external engineering experts who could have provided advice on SAM.
— Insufficient communication between the MCR and the ERC did not provide the latter the opportunity to provide advice on mitigatory actions.

- The transfer of responsibilities between different decision makers was not clear. In particular, the transfer of responsibilities from the MCR staff to other decision makers was delayed, contributing to greater stress for the staff.
- Plans for replacing operating shifts were not adequate for conditions of severely damaged infrastructure and plant surroundings.
- There was no possibility for external qualified engineering support for independent assessments and advice on recovery actions.

Information about the arrangements for staff training is provided in Section 1.2 of Technical Volume 1. Based on the information presented there, it can be concluded that the approaches and means used in training were similar to other NPPs worldwide. Since the occurrence of a severe accident was considered extremely unlikely, staff training did not pay enough attention to severe accidents and was not performed to the level required for an accident situation. More specifically, the following weaknesses in training were identified:

- Infrequent refresher training for dealing with severe accidents (once in every three years) did not provide the basis for sufficient knowledge and depth of understanding.
- Staff was not trained to perform AM actions under conditions of prolonged SBO, with extremely limited information about the plant status and limited possibility to use equipment which was dependent on availability of AC or DC power.
- Operator training for Unit 1 was performed on a Unit 3 reference simulator, which did not include IC systems, and therefore did not correspond to Unit 1; only a limited scope (compact) simulator and plant analyser was available for Unit 1.
- SAM training was not sufficiently detailed regarding assessment of the plant status under conditions of missing or limited information.
- Without information from the TEPCO's Safety Data Parameter Display System (SPDS) for the monitoring of fundamental safety functions during the accident, the personnel were not trained to use contingency means for monitoring the functions.
- Drills did not include consideration of inaccurate or missing information to Off-site Centre (OFC) personnel which would enhance their capability in diagnosing the event.

2.4.3.5. Interface with off-site emergency arrangements

Recommendations in IAEA safety standards

According to IAEA Safety Standards Series No. GS-R-2 [80] and NS-G-2.15 [49], there should be a clear determination of responsibility, coordination and exchange of information between all organizational units involved in emergency actions, including interfaces with off-site emergency arrangements. Off-site and on-site emergency response should be coordinated between all responding organizations. If off-site organizations have responsibilities in AM, this should be described. Roles and responsibilities for the different members of the emergency response organization involved in AM should be clearly defined and coordination among them should be ensured. Where the members of the emergency response organization are not at the same location, a reliable communication network between the different locations should be used. The impact of external events, such as extreme weather conditions, seismic events or events that are disruptive to society should be considered when assigning the decision making authority for SAM at an off-site location. Guidance should be available for measures to be taken if off-site communication fails and only the part of the emergency response organization located at the plant site remains functional. Arrangements for local response should be integrated with the arrangements at the national level concerning functions, responsibilities, authorities, allocation of resources and priorities. This objective requires effective communication between all the organizations involved. It is the responsibility of the regulatory body to ensure that such arrangements are in place.

Assessment of the plant status against IAEA safety standards

As stated by TEPCO [70], the combination of earthquake and tsunami was not considered in disaster management plans and engineering before 2011. Plans for utilizing off-site support, including both national industry as well as international support by operators and vendors, were not adequate for the accident. Coordination between the site ERC and corporate ERC personnel was not effective. Capabilities to predict the evolution of the accident, both for the site staff as well as for corporate ERC staff, were limited. The Off-site Centre, which was essential for coordination of external support at the corporate and governmental levels was not functional. The facility was not designed to be robust enough, so that both normal and backup power was lost, and without filtered ventilation it was not habitable due to high dose rates. There was no contingency plan for relocation of the Off-site ERC to other places under the harsh conditions. Organizational arrangements for sharing responsibility among the site ERC, corporate ERC and the government did not function as planned. There was no access by plant staff to reactor vendors and architect–engineering personnel with specialized knowledge and experience.

2.4.4. Evaluation of accident management actions at the Fukushima Daiichi NPP and identification of key factors leading to severe consequences of the accident

This section evaluates the performance of the AM actions during the course of the accident and identifies the most important factors that increased the severity of the accident. After presenting general issues affecting the performance of the AM actions at all units, more specific issues relevant for each of the units in Sections 2.4.4.1–2.4.4.4. Section 2.4.4, in combination with findings in Section 2.4.2, provides the basis for the formulation of lessons learned in the area of AM presented in Section 2.4.6.

In terms of the general issues affecting the implementation of AM actions at all units, although the staff's response was extraordinary under the harsh conditions, their actions were not optimum. In addition to the issues identified in Section 2.4.2, more specific reasons for the difficulties in preventing and later mitigating the severity of the accident at the Fukushima Daiichi NPP can be summarized as follows:

- The combined impact of the earthquake and tsunami resulted in the complete loss of all AC and DC power sources at Units 1, 2 and 4, and only limited, temporary, DC power at Unit 3, rendering the majority of systems inoperable.
- Core melting occurred in Units 1–3 due to the inability to inject water caused by the SBO and the loss of DC power.
- Primary containment vessel cooling was lost due to the loss of AC and DC power and compressed air.
- The challenges with the depressurization of the reactors contributed to the severity of the accident.
- The lack of immediately available external pumps and coolant supply.
- Depressurization of the reactor without adequate heat removal from the primary containment vessel led to high temperature, leading to the loss of containment integrity.
- Containment venting was delayed until a late stage during the accident, causing the primary containment vessel to lose its integrity. As a result, large quantities of steam, hydrogen and fission products were released into the reactor building, leading to hydrogen explosions.
- The control room became uninhabitable after some time and was accessible only with special protective clothing.
- AM measures relied heavily on the indication of the reactor water level, which due to the loss of DC power and the harsh environmental conditions resulted in false non-conservative indications.
- Lack of instrumentation on plant parameters prevented the staff from having a clear understanding of core cooling or the status of safety systems in order to take appropriate actions.

- Both on-site and off-site radiation monitors for dose rate and contamination were lost due to the loss of AC power.
- In spite of a substantial body of knowledge on severe accidents, misunderstanding of the plant status led to less than optimum decisions on AM actions and their prioritization.
- Lack of effective communication significantly extended the time needed to perform verification and control actions.
- The absence of a proper means of communication and high radiation levels seriously complicated the coordination of containment venting with emergency planning actions.
- Opening of direct release path from the primary containment vessel to the reactor building and to the environment without filters resulted in a substantial release of radioactive material.
- Contaminated debris of the reactor building structure were spread widely due to hydrogen explosions, resulting in large scale contamination of the site. This greatly complicated the recovery actions.
- The relationship of actions contained in the plant procedures and emergency plans was not clear enough (it was not clear if venting could start before the end of evacuation was confirmed), possibly contributing to a delay in containment venting.
- The established strategy delaying containment venting was not optimum since venting took place under conditions of a severely damaged core (high source term, large amount of hydrogen) — both factors significantly increasing the release.
- Insufficient communication between the MCR and the ERC did not allow for the opportunity to provide advice on mitigatory actions.
- There was no ready availability of external qualified engineering support.
- There was a need to perform AM actions on several units in parallel with inadequate human and equipment resources.
- Difficulties in replacing operating shifts led to the necessity for many workers to stay very long on-site under difficult living conditions which led to fatigue and possibly contributed to the difficulty in making complex decisions.
- The effectiveness of external support from the local government was significantly reduced due to the recovery actions that were required as a result of large scale infrastructure damage.

2.4.4.1. Unit 1: Specific issues

Key factors in the progression of the accident at Unit 1 included the following:

- After the earthquake, but prior to the tsunami, the operators, according to procedures, were cooling the reactor by cycling the operation of the IC in order to control the reactor pressure. During this procedure, they switched off the IC because they were exceeding the operational technical specification (OTS) cooling limit of 55°C/h just before the impact of the tsunami. The plant supervisor and operator did not notice that the IC was isolated, which led to a delay in the initiation of recovery actions, so that subsequent core damage and reactor vessel failure occurred much earlier than at Units 2 and 3 The fact that the operator did not have full knowledge of the status of the IC and the valves contributed to the unavailability of this system.
- The unclear and inappropriate instructions for operating the containment vent system, and a lack of appropriate training for the staff where containment exceeded its design pressure, were allowed for a relatively long time. Furthermore, the delay in lowering the containment pressure hindered the efforts of the field operators to implement an alternative type of water injection into the reactor vessel.
- The inappropriate design of the containment venting system, with shutoff valves not designed to be operated under severe conditions, loss of DC power and compressed air and, later on, the harsh environmental conditions in the reactor building.

2.4.4.2. Unit 2: Specific issues

Key factors influencing the progression of the accident at the unit included the following:

- The RCIC was started manually when the feedwater was lost due to the lack off-site AC power following the earthquake. After depletion of the water storage tank, the RCIC suction was switched over to the suppression chamber. There were difficulties in confirming the operational status of the RCIC due to the loss of instrumentation power. Nevertheless, the RCIC maintained reactor inventory and cooling for almost three days.
- The RCIC was not designed to operate during the prolonged loss of AC power (without cooling of the suppression chamber) and without DC power (with loss of turbine control). As a result, the RCIC was lost on 14 March, but a clear reason for this was not determined.
- The HPCI could not be used when the RCIC was lost because of the lack of DC power.
- Effective venting of the containment vessel was not implemented before RCIC failure, preventing heat removal from the primary containment vessel.
- The hydrogen explosion in Unit 1 caused a blow-out panel on the wall of the reactor building of Unit 2 to open; the hydrogen released from the primary containment vessel did not accumulate in the reactor building, but was released directly into the atmosphere.

2.4.4.3. Unit 3: Specific issues

Key factors in the progression of the accident in the unit included the following:

- Emergency lighting and some indications remained functional after the tsunami due to the availability of DC power, attributable to the higher elevation of the batteries and busbars compared with Units 1 and 2 (the battery discharge time being about 8–10 hours).
- The RCIC system was manually started at 15:05 on 11 March. More optimum operation of RCIC by the operators was possible due to the availability of DC power avoiding automatic reactor high level RCIC shutdown (bypassing part of the injected water). The RCIC system operated for more than 20 hours until its unexpected shutdown, with attempts to restart it being unsuccessful.
- Following the failure of the RCIC system, the HPCI system started at 12:35 on 12 March due to the low RPV water level. The operators ran the system continuously to conserve DC power until 02:42 on 13 March, when the HPCI system was manually shut off.
- The longer availability of DC power was achieved by switching off unnecessary loads and controlling the water injection rate manually to prevent the unnecessary start and stop of the HPCI/RCIC systems.

2.4.4.4. Unit 4: Specific issues

Key factors for damage of the unit included the following:

- Since there was a serious concern about the status of the SFP, a great deal of effort was devoted to delivering coolant to the pool by various means (e.g. helicopters and fire trucks).
- Late containment venting under conditions of severe core damage in Unit 3 led to the accumulation of leaked hydrogen in the reactor building of Unit 4 through the shared reactor building ventilation system. This was caused by the inadequate design of the venting system common to Units 3 and 4, which led to the accumulation and eventual explosion of hydrogen in the Unit 4 reactor building.

2.4.5. Summary

Although severe accidents were not part of the design basis for the units, requirements and guidance for severe accidents were included in the IAEA safety standards well before the accident at the

Fukushima Daiichi NPP. Japanese practice at the time of the accident was not fully in line with these requirements and guidance. TEPCO had developed emergency and accident management procedures, which had been strongly recommended by the Nuclear Safety Commission, but which were not a requirement. The IAEA's International Regulatory Review Service (IRRS) mission in 2007 had suggested that the regulatory body continue to develop a systematic approach to BDBAs, and the complementary use of PSA and SAM in the assessment process. However, this was not done.

The AM programme and procedures assumed that AC power would always be available at a unit, either from its own power source or through a connection with the neighbouring units. It was also assumed that DC power would always be available for the proper functioning of the instrumentation and for manipulating valves. These assumptions turned out to be incorrect. The procedures also did not consider the possibility that a severe accident could impact several units simultaneously or that the off-site infrastructure could be seriously disrupted, making it more difficult to receive support in responding to an event or accident.

The combined impact of the earthquake and tsunami resulted in complete loss of all AC and DC power at Units 1, 2 and 4, with only limited DC power being temporarily available in Unit 3. This caused the cores of Units 1, 2 and 3 to melt because of the inability of the operators to inject water to cool them. Primary containment vessel cooling was also lost due to the loss of AC power, which, combined with the delay in venting the containments until a late stage of the accident, resulted in their failure. As a result, large quantities of steam, hydrogen and fission products were released into the reactor buildings, leading to three hydrogen explosions.

Accident management actions relied strongly on the indication of the reactor water level, which due to the loss of DC power and harsh environmental conditions, resulted in no (or false non-conservative) indications. The lack of instrumentation meant that staff did not have a clear understanding of plant parameters on core cooling or the status of safety systems in order to take appropriate actions.

Beyond design basis accidents were not addressed sufficiently, either by deterministic or probabilistic methods. Design basis accident analyses cannot be the sole basis for efficient development of defence in depth, since these analyses do not deal with BDBAs. The Fukushima Daiichi units were reasonably well equipped with systems to cool the core in the absence of AC power. However, the complexity of system performance under beyond design basis conditions together with inadequate training meant that the operators were not able to operate the systems, contributing to the severity of the accident.

The lack of effective communications significantly extended the time needed to perform verification and control actions. Lack of the proper means of communication and high radiation levels seriously complicated coordination of containment venting with emergency planning actions. Problems with communication between the MCR and the ERC did not allow the centre the opportunity to provide advice on mitigatory actions.

Operating experience from the 2007 earthquake that affected the Kashiwazaki-Kariwa NPP had resulted in the fitting of the means for the injection of cooling water via external portable pumps into all reactors. This was essential to the restoration of cooling water injection into the reactors at Units 1-3. However, training in the use of this equipment and other installed equipment under extreme environmental conditions was insufficient to allow for its timely use, given the complete loss of power. The weaknesses in the training and the available strategies for severe accident response are in part related to the fact that although TEPCO had access to the latest guidance from the BWROG, it did not implement this guidance at its reactors. As a result, operators and emergency response personnel were not able to address the degraded plant conditions that occurred following the earthquake and tsunami because the necessary accident mitigation strategies were not in place.

The need to perform AM actions on several units in parallel with an inadequate number of people and equipment caused great problems for the operators. Difficulties in replacing operating shifts led to many workers staying on-site for a very long time under difficult living conditions, which led to fatigue and possibly contributed to the difficulty in making complex decisions. There was no engineering support readily available from off-site sources. External support from the local government was significantly reduced due to the large scale infrastructure damage.

2.4.6. Observations and lessons

- **Accident management provisions need to be clear, comprehensive and well designed.**
  Accident management strategies need to be based on a plant specific analysis performed by using a combination of deterministic and probabilistic approaches. Accident management guidance procedures need to consider events taking place in several units simultaneously and in SFPs. These provisions need to take into account disrupted regional infrastructure, including serious deficiencies in communication, transportation and utilities.
- **Regulatory bodies need to ensure that adequate AM provisions are in place, taking into account severely damaged infrastructures and long duration accidents.**
  The Nuclear Safety Commission (NSC) recommended in 1992 that the regulatory body and the utilities introduce AM measures, although this had not been a requirement. In response, TEPCO had developed emergency and AM procedures. An IAEA IRRS mission to Japan in 2007 noted that AM measures were “taken by licensees on a voluntary basis” [51]. Therefore, it was suggested that the regulatory body develop a systematic approach for BDBAs, and the complementary use of PSAs, and SAM in the assessment process was recommended. However, no effective action was taken.
- **Training and exercises need to be based on realistic severe accident conditions.**
  Special attention in personnel training needs to be devoted to performing actions under conditions of prolonged SBO with limited information about the plant status or the unavailability of 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. Training, exercises and drills need to involve not only on-site AM personnel or personnel executing on-site emergency plans, but also all off-site responders at corporate, local, regional and national levels.
- **Provisions for the proper management of hydrogen need to be considered.**
  The hydrogen explosions at the Fukushima Daiichi NPP significantly affected the ability of operators to respond to the accident. Hydrogen monitoring and removal equipment should be installed in the plant to prevent hydrogen deflagration or explosions.

2.5. ASSESSMENT OF THE EFFECTIVENESS OF REGULATORY PROGRAMMES

2.5.1. Introduction

This section describes and evaluates the governmental, legal and regulatory framework for nuclear safety in Japan up to the time of the accident at the Fukushima Daiichi NPP. It focuses on the framework’s contribution to the accident, and identifies lessons learned.

2.5.2. Governmental, legal and regulatory framework for nuclear safety in Japan

The evolution of the governmental and regulatory organizations prior to the accident at the Fukushima Daiichi NPP has been divided into three periods according to H. Shiroyama [81]: (1) from 1957 to 1978; (2) from 1978 to 1999; and (3) from 1999 to 2011. What distinguishes each of these periods from each other is that significant changes to the Japanese regulatory structure were enacted in response to an event at a nuclear installation in Japan. The historical development of the regulatory framework in Japan is described in Annex I of this volume.

The criticality accident that occurred on 30 September 1999 at the JCO fuel fabrication facility at Tokaimura initiated the third period in the evolution of the Japanese governmental and regulatory structure. The accident was, at that time, Japan's worst nuclear accident, and was classified at level 4 on the International Nuclear and Radiological Event Scale (INES) [82]. Two workers died [83] as a result of exposure to radiation and hundreds of other workers and members of the public received radiation doses. Although the direct causes of the accident were linked to unsafe acts and faulty procedures on the part of the operator, contributing factors included the failure of the former Science and Technology Agency (STA) to adequately assess the hazards of the facility and detect non-compliances with the conditions of its licence [84, 85].

The NSC set up an investigative committee on the JCO's Tokaimura criticality accident which recommended actions to strengthen the regulatory framework. Several changes were made to laws and organizational arrangements following these recommendations [84-86]:

- Strengthening the Nuclear Safety Commission: To strengthen the NSC, it was transferred from the Nuclear Safety Bureau of the Science and Technology Agency (STA) to the Prime Minister's Office on 1 April 2000. The NSC staff was increased from 20 to 92 members, and its involvement in regulatory oversight was subsequently institutionalized, providing a double-check system in the regulatory activities.
- Amendment of the Reactor Regulation Act: The Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (Reactor Regulation Act) was amended in 2000 to strengthen the nuclear safety requirements for the management, operation and inspection of nuclear processing plants and nuclear energy facilities [87].
- Enactment of the Act on Special Measures Concerning Nuclear Emergency: The Act on Special Measures Concerning Nuclear Emergency Preparedness (Nuclear Emergency Act) was adopted in December 1999 [88]. This legislation complemented the provisions for response to natural disasters established in the Disaster Countermeasures Basic Act [89]. The Nuclear Emergency Act set out roles and responsibilities of the national government, local governments, and licence holders in case of emergencies at nuclear facilities.
- Amendments of Ordinances for the Compensation Law and the Law on the Indemnity Agreement for Compensation of Nuclear Damage: The implementing ordinances for the Compensation Law and the Law on the Indemnity Agreement for Compensation of Nuclear Damage were amended by Cabinet Order in 1999 to establish the amounts for which nuclear operators are liable, and to include nuclear damage resulting from transport, storage or disposal incidental to the storage of nuclear spent fuel within their scope.
- Revision of the Emergency Preparedness Guidelines: The guidelines on technical and special matters of nuclear emergency measures were revised by the NSC in May 2000 [90].
- Reorganization of government bodies with responsibilities for nuclear safety: A reorganization of the Japanese central government took effect on 6 January 2001. The reorganization had been planned for several years, according to the Basic Act on Central Government Reform (No. 103 of 12 June 1998) and other laws related to administrative reform. It was aimed broadly at reducing costs and improving the efficiency of government [91]. However, a number of specific measures were implemented with the intention of strengthening the governmental organizations responsible for nuclear safety.

The laws that were in place following the changes during this third period are shown in Fig. 2.5-1.

The former MITI was reformed as the Ministry of Economy, Trade and Industry (METI) with responsibilities for ensuring a stable and efficient energy supply, including the use of nuclear energy. METI was also put in charge of nuclear safety regulation and the licensing of nuclear installations.

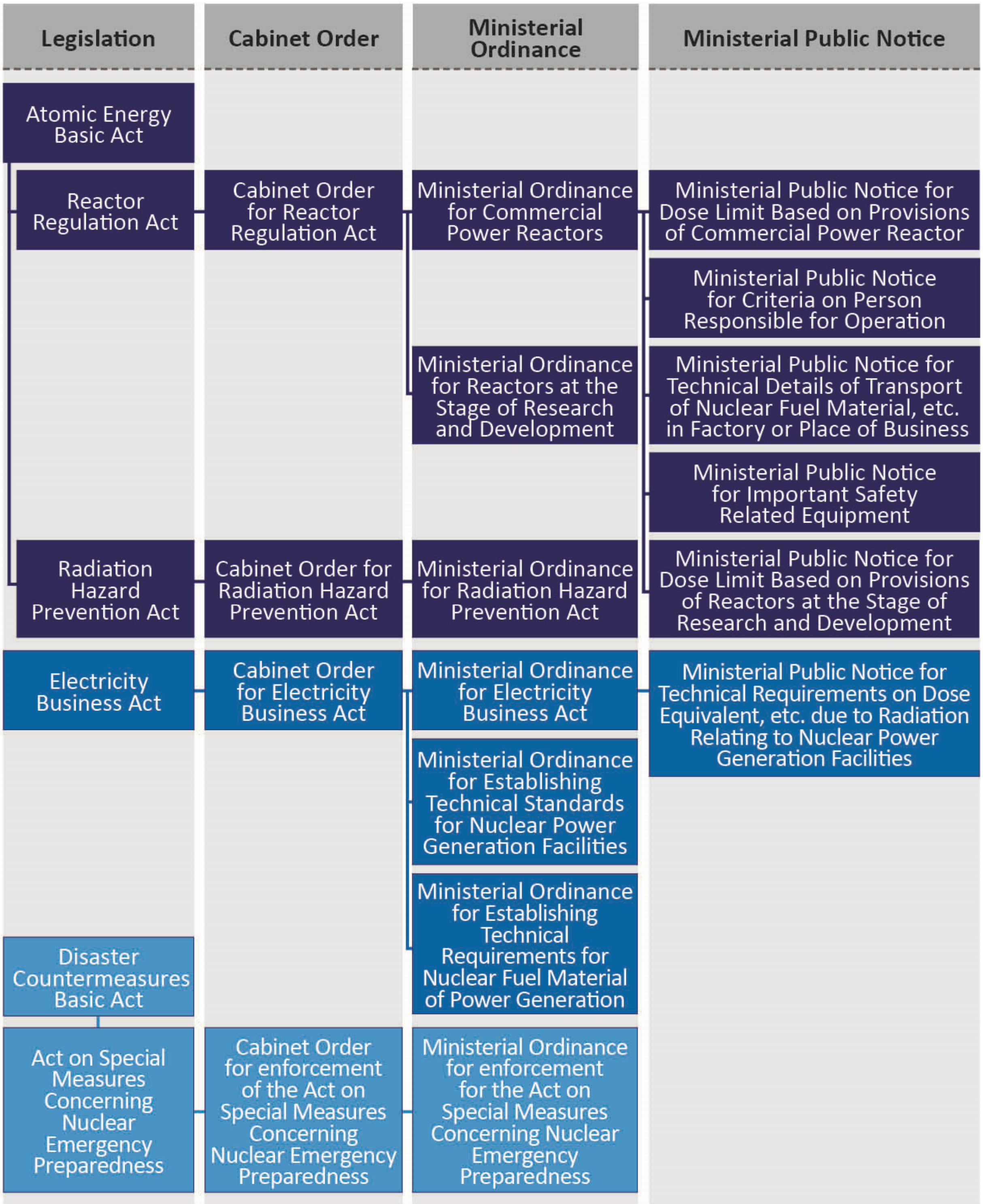


FIG. 2.5–1. Japanese nuclear safety legislation as of March 2011 [92].

Within METI, specialized structures were set up to carry out different tasks. The Agency for Natural Resources and Energy (ANRE) within METI had responsibility for planning and overseeing the national energy supply. ANRE's Department of Electricity and Gas Industry managed nuclear energy policy and radioactive waste management.

The Nuclear and Industrial Safety Agency (NISA) was set up within ANRE to regulate the safety of nuclear and other energy sources as well as industrial safety. NISA's main functions comprised the following, as set out by the Law for Establishment of The Ministry of Economy, Trade and Industry (Law No. 99 of July 16, 1999) [53, 93]:

- Regulating nuclear power refining, fabrication, storage, reprocessing and waste disposal businesses and nuclear power generation installations and matters relating to ensuring the safety of these businesses and installations;
- The safety of nuclear power relating to its utilization as an energy source;
- Control of explosives, safety of high pressure gas, mine safety, and other safety matters under its jurisdiction (hereinafter referred to as industrial safety);
- International cooperation pertaining to affairs under its jurisdiction.

The new Ministry of Education, Culture, Sports, Science and Technology (MEXT) was created through a merger of the STA and the Ministry of Education. The former responsibilities of the STA for nuclear safety regulation were reallocated between METI and MEXT. Accordingly, the regulation of nuclear fuel cycle facilities, including uranium refining and fuel fabrication, spent fuel storage, reprocessing, nuclear waste management business, and related transport of nuclear material, were transferred from STA to NISA. About 90 employees were also transferred from STA to NISA, approximately 35% of NISA's initial complement of 260 people [94].

MEXT retained responsibility for the science and technology aspects of nuclear energy, including: policy and the development of nuclear technologies; safety regulations governing research reactors, protection against radiation hazards; and the use and transport of nuclear and radioactive material; and safeguards. MEXT also supervised the National Institute of Radiological Sciences (NIRS), the Japan Atomic Energy Research Institute and the Japan Nuclear Cycle Development Institute. The latter two organizations merged in 2005 to become the Japan Atomic Energy Agency (JAEA) [94, 95].

The Nuclear Safety Commission (NSC) and the Japan Atomic Energy Commission (JAEC) were relocated from the Nuclear Safety Bureau of the STA to the Cabinet Office to continue their respective roles as senior advisory bodies on nuclear safety and nuclear policy. The NSC was empowered by law to require reports from NISA and performed double-check reviews of NISA's work.

The Ministry of Foreign Affairs (MOFA) was responsible for the international aspects of nuclear energy utilization, including the implementation of the related international treaties and conventions.

Later, in 2003, the Japan Nuclear Energy Safety Organization (JNES) was established in order to enhance regulatory safety activities together with NISA. The law establishing JNES was enacted by an extraordinary session of the Japanese Diet held in autumn 2002, following a crisis in public confidence precipitated by the discovery of TEPCO's falsification of safety records at its nuclear facilities [54]. JNES's main functions, as set out in its establishment law, consisted of conducting inspections at nuclear facilities, review of licensees' periodic inspections, safety analysis and evaluation, nuclear emergency preparedness support, research and testing for code/standard development, and collection, analysis and distribution of safety information. In practical terms, JNES was formed by merging parts of the former Nuclear Power Engineering Corporation (NUPEC), the Japan Power Engineering and Inspection Corp (JAPEIC), and the Nuclear Safety Technology Centre (NUSTEC) [93, 96].

The Japanese legal framework comprised numerous laws which had been revised over the years in response to accidents and incidents and which, as a whole, clouded the authorities and responsibilities for nuclear safety. The National Accident Independent Investigation Commission of the Japanese Diet has commented critically on this matter, describing the legal framework as a 'patchwork' and stating [32]:

> "The laws and regulations governing Japan's nuclear power industry at the time of the accident were outdated relative to those of other countries and, in some cases, obsolete."

The independence of the regulatory body from entities with interests that could unduly influence its safety related decision making is another important principle. The Government of Japan has claimed that NISA and ANRE were effectively separate agencies [93]. The IAEA's International Regulatory Review Service (IRRS) mission to Japan observed that:

> "NISA is effectively independent from ANRE, in correspondence with the requirements of GS-R-1. This situation could be reflected in the legislation more clearly in future" [53].

In retrospect, it is evident that NISA lacked the requisite de jure independence from entities responsible for the promotion of nuclear energy. NISA as a department within METI was subordinate to the Minister. NISA's lack of clarity on its legal authority was illustrated by the repeated cases of data falsification by NPP operators. The Minister was also responsible for national policy on the utilization of nuclear energy, yet at the same time was also responsible for enforcement action against his licensees in the light of, in the NSC's words, their "repeated malicious conduct" [97] in relation to safety.

The complex arrangement of organizations at different levels within the Government of Japan appears to have hampered the effective regulation of safety. Many government ministries, agencies and quasi-governmental organizations played influential roles with regard to the utilization and regulation of nuclear energy in Japan. The NSC was an important body with both advisory and supervisory roles in the system of 'subsequent regulation review'. In the opinion of the IRRS mission, the role of NISA as the regulatory body in relation to NSC was in need of clarification [53]. Inspection activities, an important part of the functions and responsibilities of the regulatory body, were assigned by law partly to JNES. Consequently, NISA had to manage the interface with JNES with regard to its inspections.

2.5.3. Organization and staffing of the regulatory body

This section describes the nuclear regulatory body staffing and organization, management systems, and regulatory functions in place prior to the Fukushima Daiichi accident.

NISA was organized with 11 divisions reporting to the Director General at the Tokyo headquarters. Nuclear safety inspectors and senior specialists for nuclear emergency preparedness were stationed at each nuclear facility NISA's budget was allocated within the government which had the final decision on matters such as staff numbers and the number of divisions. The government was seeking generally to reduce staff numbers in administrative posts. Although NISA had been relatively successful in defending its budget, its allocation of Yen 32.6 billion (US $330 million) in fiscal year 2010 was 12% less than the previous year [53].

NISA was subject to a rule for government personnel whereby staff members were expected to rotate jobs every two to three years, which hampered NISA's ability to develop and retain specialist expertise. NISA tried to retain expertise in two ways. First, experts were rotated within NISA, thus retaining their knowledge. Second, policy makers were expected to stay at least three years and were thereafter rotated to other departments of METI [93].

JNES was established as a technical support organization to NISA in 2003, funded mainly by NISA, 60% of NISA's fiscal year 2010 budget being allocated to JNES, which had approximately 400 employees as of 1 April 2010. JNES had the status of an 'incorporated administrative agency'. As such, JNES was free of some of the administrative rules that applied to central government departments. Consequently, JNES had more flexibility on matters of staffing and budget which better enabled it to retain a cadre of professional experts [93].

The NSC had become considerably stronger since its foundation 30 years earlier. It was supported by its own secretariat, composed of a Secretary-General and six divisions with a total of about

100 members of staff. Several advisory councils and committees on safety, emergency preparedness and other regulatory matters also supported it. In order to increase transparency, the NSC had also taken steps to open the deliberations of its committees to the public [93, 96].

The budget pressures and administrative rules for job rotation of civil servants to which NISA was subject were important factors that limited the technical and regulatory competence of its staff.

2.5.4. Regulatory management systems and safety culture

The IRRS mission to Japan commended NISA for its recognition of the importance of adopting a comprehensive quality management system and for having set out its management policy and developed a quality management manual. However, although some instructions and guides were available at the time of the mission, particularly for periodic and operational safety inspection, they were not available for all key regulatory process tasks and activities; nor did they appear to be linked to an overall process map for NISA as a whole and nuclear safety regulation in particular. The IRRS found that further development was needed to fully establish the management system within NISA [53]. While the Japanese national report to the 2011 review meeting on the Convention on Nuclear Safety (CNS) referred to NISA's management system, there was no further information on its implementation as of that date [92, 96].

NISA embraced a code of conduct based on the following four values:

— Strong sense of mission;
— Scientific and rational judgments;
— Neutrality and justice;
— Transparency in operations.

NISA described its decision making process and public communication as being based on scientific and rational judgment, while less account was taken of issues related to management of safety and human performance [53].

Although NISA had recognized the importance of developing a management system to direct its activities, the management system had not been effectively implemented by the time of the accident. The internal culture of the regulatory body valued hard science over softer subjects.

2.5.5. Authorization of facilities and activities by the regulatory body

The Fukushima Daiichi NPP is one of Japan's oldest nuclear sites, dating from the early days of the nuclear programme in Japan. The licence for establishment of the Fukushima Daiichi NPP was issued on 1 December 1966 by the Prime Minister, after consideration of TEPCO's application by the Nuclear Reactor Safety Review Committee of the JAEC.

The JAEC Safety Review Committee had established five expert groups which conducted the safety review of the proposed facility at the Fukushima Daiichi NPP:

— Reactor Group responsible for review on physics, thermal and mechanical, fuel and material;
— Environment Group responsible for review on meteorological and radiological effects;
— Seismic Group responsible for review on seismic designs;
— Plant and Power Generation Group responsible for evaluation on plant performance;
— Safety Review Group responsible for evaluation on accidents and disasters.

The review was carried out by referring to various guides prepared by the JAEC Committee on Nuclear Reactor Safety Standards as well as information from other countries such as the draft US

General Design Criteria. After the reactor Establishment Permit was granted in 1966 for the first unit, the Ministry of International Trade and Industry (MITI) reviewed and approved the Construction Plan in September 1967, conducted the pre-service inspection and tests before fuel loading and commissioning, according to the Electricity Business Act. The construction of later units at the Fukushima Daiichi NPP was the subject of amendments to the original Establishment Permit.

In the authorization process prior to the formation of the NRA, the planning and design stage was completed by an application for an establishment licence under the Nuclear Regulation Law. The design review was conducted following a double-check approach: a primary review performed by NISA and a secondary review by NSC and JAEC. Since 2001, both commissions provided opinions to METI, which the Minister had to take into account before the establishment licence could be granted.

Having received an establishment licence, the operator had to apply for the construction plan under Article 47 of the Electricity Business Act. This approval was granted in several steps, based on compliance of the detailed design with technical standards. Further approvals were needed for the pre-service inspection, the safety management review for welding, the fuel design and the fuel inspection. Since 2003, JNES had performed assessments and pre-service inspections of equipment in support of NISA. After completion of construction work, the operator applied for approval of the operational safety programme. Approvals were signed by the Minister of METI. There was no requirement for submittal of a Final Safety Analysis Report (FSAR). The same authorization procedure was followed for major modifications that had a safety significant impact on the basic design.

The initial authorization for the Fukushima Daiichi NPP was granted in 1966, in the early years of nuclear power development in Japan. The construction of additional units at the site was treated as an amendment to the Establishment Permit. The operator was not required to submit or to maintain up to date an FSAR or an equivalent document describing the design and operating arrangements for the facility. Not updating the documentation which describes the revised safety case is not in line with IAEA safety standards.

2.5.6. Severe accident assumptions and countermeasures

The NSC published a regulatory guide on Accident Management for Severe Accidents at Light Water Power Reactor Installations in May 1992 [62]. Shortly afterwards, in July 1992, MITI published a document titled Roadmap of Accident Management [98] and requested nuclear operators to take action in response. However, the AM measures were limited in scope. Moreover, the measures were regarded as voluntary initiatives by nuclear operators and not as regulatory requirements [99]. TEPCO prepared AM measures which included primary containment vessel venting systems and EDGs over the period from 1994 to 2002. The authorities in Japan believed that the safety of reactor facilities was sufficiently ensured by existing safety regulations. The NSC decision appended to its guide states [62]:

> "The safety of reactor facilities in Japan is sufficiently ensured by current safety regulations... The possibility of severe accidents is sufficiently low due to these measures, to the extent that such accidents could not occur from an engineering viewpoint, and thus the risk from reactor facilities is considered to be sufficiently low.... The Commission believes that effective accident management should be developed by licensees on a voluntary basis..."

The NSC based its assessments on level 1 PSAs, which showed the estimated frequency of reactor core damage to be comparable to reactors in other countries [62]. The apparent acceptability of the CDF was used to justify the limited measures taken by licensees to mitigate the consequences of a severe accident. However, the PSAs considered only internal events within the NPPs, such as equipment failures and human errors. External hazards, such as earthquakes or tsunami, and internal hazards were not included in the scope of the PSAs (see Section 2.3). Since the PSAs did not consider these initiating events, the estimates of reactor CDF were optimistic. Nor did the PSAs evaluate the potential for large releases of radioactivity outside the reactor containment.

The 2007 IAEA IRRS mission noted the lack of regulatory requirements for BDBAs and made a suggestion that NISA continue to develop a systematic approach to consideration of such events and the complementary use of PSA and SAM [53]. Neither the IRRS mission suggestions, nor the actions in such as the USA and Europe to enhance accident measures in the light of external events and terrorist incidents, appear to have spurred further efforts in this area.

2.5.7. Authorization of long term operation: Requirements for periodic safety review and backfit

The Establishment Permits for Japanese NPPs were issued for an unlimited period. The Ministerial Ordinance under the Nuclear Regulation Law required an ageing management technical review for plants that reached 30 years of operating life, and ongoing Ageing Management Technical Assessments (AMTA) at ten year intervals thereafter. The AMTA covered the functional soundness of selected components and the status of checkout/inspection, functional surveillance test, monitoring and repair/replacement of the portions evaluated. The licensee was required to develop a ten year Long-term Maintenance Programme (LTMP) based on the evaluation results. NISA and JNES checked the operator's implementation status of the LTMP during their Periodic Inspection and the Safety Management Review. The IRRS peer review mission concluded that: "Ageing phenomena in general are carefully studied in Japan" [53].

TEPCO submitted the results of the AMTA for the 40th year of operation and the LTMP for Unit 1 of the Fukushima Daiichi NPP to NISA on 25 March 2010. NISA reviewed the report with JNES and held committee meetings to incorporate the experts' opinions in the review. NISA approved the report and operation of the unit for a further ten years on 7 February 2011, just one month before the accident.

Licensees in Japan also conducted PSRs every ten years after the commencement of operation. The PSRs started as a voluntary activity in 1992 and were made a legal requirement in 2003 through an amendment to the Ministerial Ordinance on Rules for the Installation, Operation of Commercial Power Reactors under the 'Nuclear Regulation Act' [99]. All NPP units more than ten years of age were reported to have conducted at least one PSR at the time of the IRRS mission in 2007.

The licensees were not, however, required to submit the results of their PSRs to NISA. Before the 2003 amendment, PSR reports were provided to MITI or to METI/NISA who reviewed them and reported to the NSC. After the 2003 amendment, PSRs — although now mandatory — were regarded as self-assessments and the licensees were not required to submit them to NISA. The regulatory body confirmed that each licensee had adequately conducted PSRs during its inspections.

The PSRs that were conducted in Japan did not entail re-examination of the adequacy of the design of the NPP or the reassessment of external hazards in the light of new standards, data and methods. The Japanese national report to the 5th Review Meeting of the CNS in 2010 states that:

> "The Periodic Safety Review in Japan covers the operator safety activities at nuclear installations in operation, but it does not cover the design review. The nuclear installations have been designed with sufficient safety margin, than that they just met the necessary design criteria so that back-fitting has not been institutionalized" [92].

The IRRS mission to Japan made the suggestion that:

> "The PSR should be made a more focused and periodic effort to give a comprehensive picture of the plant safety status at certain intervals. All its conclusions should be reported to NISA in one summary report" [53].

There was no other established process in Japan for deciding on backfits to a plant (where backfit means the application of current requirements to an existing facility). NISA did not have a legal basis to order backfits on items covered by the technical standards referenced in the plant establishment permit. NISA's policy was to request the licensees to conduct so-called backchecks when technical standards were revised or new knowledge was gained.

The original Establishment Permit for the Fukushima Daiichi NPP was granted on the basis of a maximum tsunami height of 3.1 m [99]. TEPCO performed a re-evaluation of tsunami hazards in 2002 on a voluntary basis following which the design tsunami height was increased without changing the licensing documents. When the Seismic Design Review Guide, first issued by the NSC in 1978, was fully revised in 2006, NISA requested that the licensees conduct a backcheck to re-evaluate seismic and tsunami hazards. The re-evaluation was done on a voluntary basis and not in the framework of a formal PSR. The licensees' voluntary activities were delayed. TEPCO had not finished its re-evaluation by the date of the accident at Fukushima.

The approach to PSR in Japan was not fully in line with guidance from the IAEA [100]. IAEA Safety Standards Series No. NS-G-2.10, Periodic Safety Review of Nuclear Power Plants [100], recommends a comprehensive review of the design and operation of an existing NPP, including re-evaluation of external hazards, with the objective of determining its adequacy against the current licensing basis and national and international standards, requirements and practices. NS-G-2.10 [96] also recommends that the results of the PSR be submitted to the regulatory body for review.

Had a more rigorous approach to PSR been implemented in Japan in accordance with NS-G-2.10 [100], it would have provided 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. As it was, the lack of a requirement to submit results of the PSR described in the Japanese national reports to the CNS may have contributed to the failure of the licensee and the regulator to rigorously consider new information on vulnerabilities related to earthquakes and tsunami.

The new Japanese NRA has begun consideration of a PSR process aligned with IAEA safety standards. It was enforced in December 2013 as a system which requires NPP licensees to evaluate the safety of reactors comprehensively and submit the results. In order to improve safety at NPP installations, licensees are to submit documents which show the following to the NRA in principle within six months, after the end of periodic inspection:

- The extent of conformity with regulatory requirements;
- Measures for improving safety on a voluntary basis;
- Assessing and reviewing safety margins for improvement and a PSA;
- Comprehensive re-evaluation based on the above results and action plans for improving safety.

2.5.8. Inspections of facilities and activities

The Nuclear Regulation Law was amended after the JCO's Tokaimura criticality accident to correct the regulatory shortcomings the accident revealed and to institute regular inspections by the authorities of the management and operation of nuclear facilities. The IRRS mission in 2007 found that NISA was actively working on tying together the lessons learned from previous incidents and to develop an inspection system that focused not only on hardware, but also on licensees' operational management [53].

Despite NISA's efforts toward improvement, the regulatory inspection programme was rigidly structured by law, including the type and frequency of inspections. The presence of NISA inspectors was required before the operator could complete certain tests and surveillances, and there were only specific periods in which NISA was permitted access for inspection. This approach limited the powers of NISA to perform inspections as needed and made it difficult for NISA to change the inspection type or frequency. The restrictions also affected the operator's conduct of tests and surveillance based on the availability of NISA inspectors [53].

The repeated discoveries of licensees' falsification of safety records in 2002 and 2007 suggest shortcomings in the effectiveness of regulatory inspection and enforcement programmes. The initial discovery was a result of allegations raised by employees. In a sense, therefore, this discovery represented a success for the allegation system that had been instituted after the JCO's Tokaimura criticality accident. However, in spite of follow-up inspections by NISA, regulatory orders to licensees, and various other measures to strengthen self-checking and regulatory oversight, the true extent of records falsification did not come out until several years later. The Fukushima Daiichi NPP was implicated in several cases: one such case involved the past falsification of containment leak rate tests by employees at Unit 3, for which METI imposed a one year licence suspension as punishment [93].

Enhancements to the inspection system were instituted by a revised Ministerial Order issued in January 2009. Greater flexibility was introduced, considering the characteristics of individual nuclear installations, and enhancements were made to inspections during operation and plant shutdown in response to lessons learned from the data falsification events in 2006 [96].

By 2011, the Japanese national report to the Convention on Nuclear Safety stated [92] that operator's safety management activities were governed by the operational safety inspections that NISA had approved. NISA conducted quarterly safety preservation inspections to check the operator's compliance with the PSRs. Periodic inspections were also conducted by NISA and JNES at intervals not exceeding 13 months, focused on the operator's maintenance of the SSCs of the NPPs. The periodic inspections focused on the safety significant SSCs, for example, belonging to the reactor shutdown system, the reactor coolant pressure boundary, the RHR system and the containment system. These regulatory procedures were in addition to the licensees' own walkdown and maintenance management of nuclear installations, periodic safety assessments, and ageing technical evaluations [96]. Figure 2.5-2 illustrates the regulatory inspection process for the operating stage of NPPs at the time of the accident at Fukushima Daiichi NPP.

The regulatory inspection programme at the time of the accident at the Fukushima Daiichi NPP was rigidly structured by law, including the type and frequency of inspections and the rights of access by inspectors. This approach limited the effectiveness of regulatory inspection and enforcement in verifying the safety of licensees' activities and compliance with requirements.

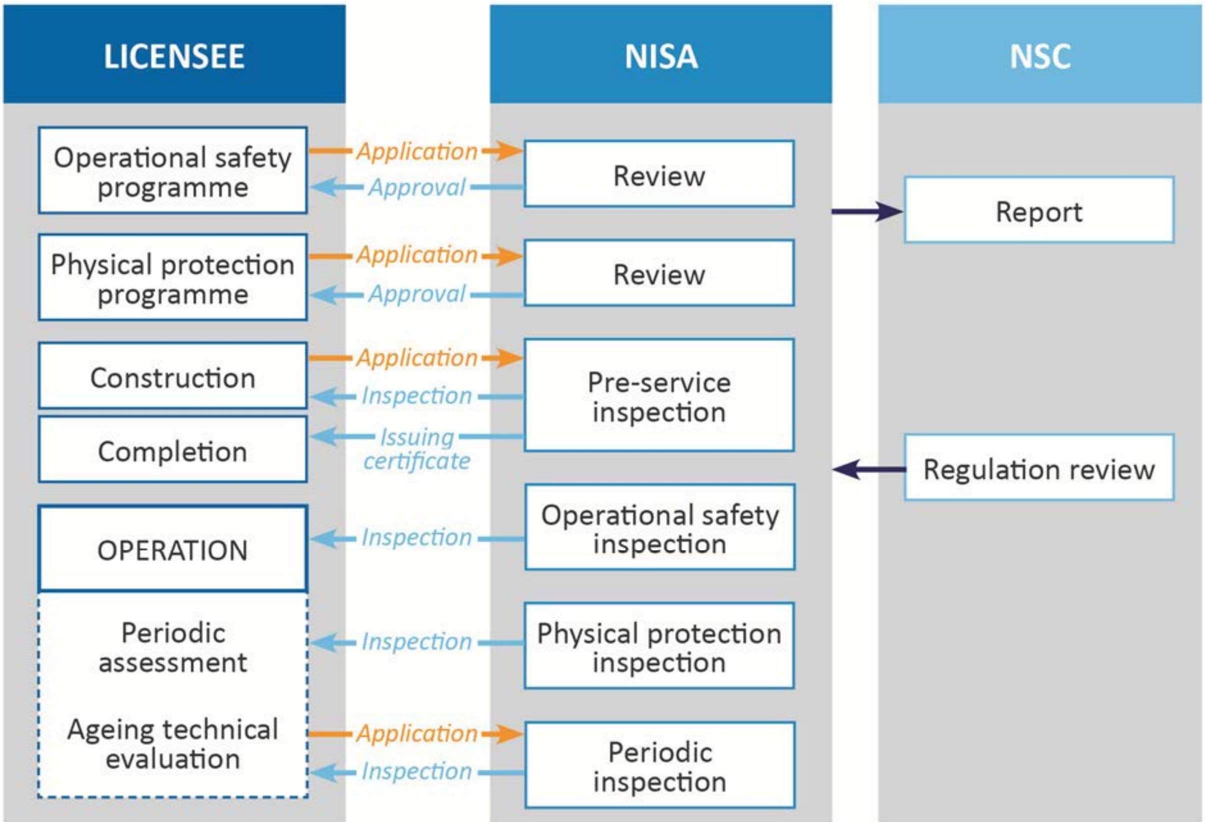


FIG. 2.5–2. The regulatory inspection process at the time of the accident for the operating stage of NPPs [92].

2.5.9. Regulatory requirements and guidance

As noted in previous subsections, the national laws and ministerial ordinances that formed the top two levels of the legal hierarchy were primarily administrative in nature. NISA was not authorized by its statutes to issue regulations binding on applicants and licensees. Two more levels in the hierarchy of documents provided technical criteria for safety.

The NSC had prepared a number of regulatory guides over many years. The NSC guides for power reactors were in five divisions, namely: siting, design, safety evaluation, radiation exposure and accident management. The NSC Regulatory Guides were used by the NSC and NISA to conduct safety reviews. The NSC guides were not requirements, nor were they legally binding, although the IAEA fact finding mission was informed that the NSC guides were considered in practice as requirements [6]. The IRRS mission had discussed these arrangements and had recommended that NISA take the major responsibility in the development and endorsement of regulations and guides for safety [53].

The IRRS mission also learned that NISA drafted technical standards in the form of ministerial orders. Draft standards were discussed with subcommittees and working groups set up under the Nuclear and Industrial Safety Subcommittee and were also subject to external consultation with the public and electric utility stakeholders and to ministerial approval prior to publication.

The fourth level of the hierarchy comprised private consensus standards published by academic and professional societies, such as the Japan Society of Mechanical Engineers, the Atomic Energy Society of Japan, the Electric Association, the Thermal and Nuclear Power Engineering Society in Japan, which NISA endorsed. Figure 2.5-3 outlines the system of technical standards for nuclear power plants in Japan prior to the accident.

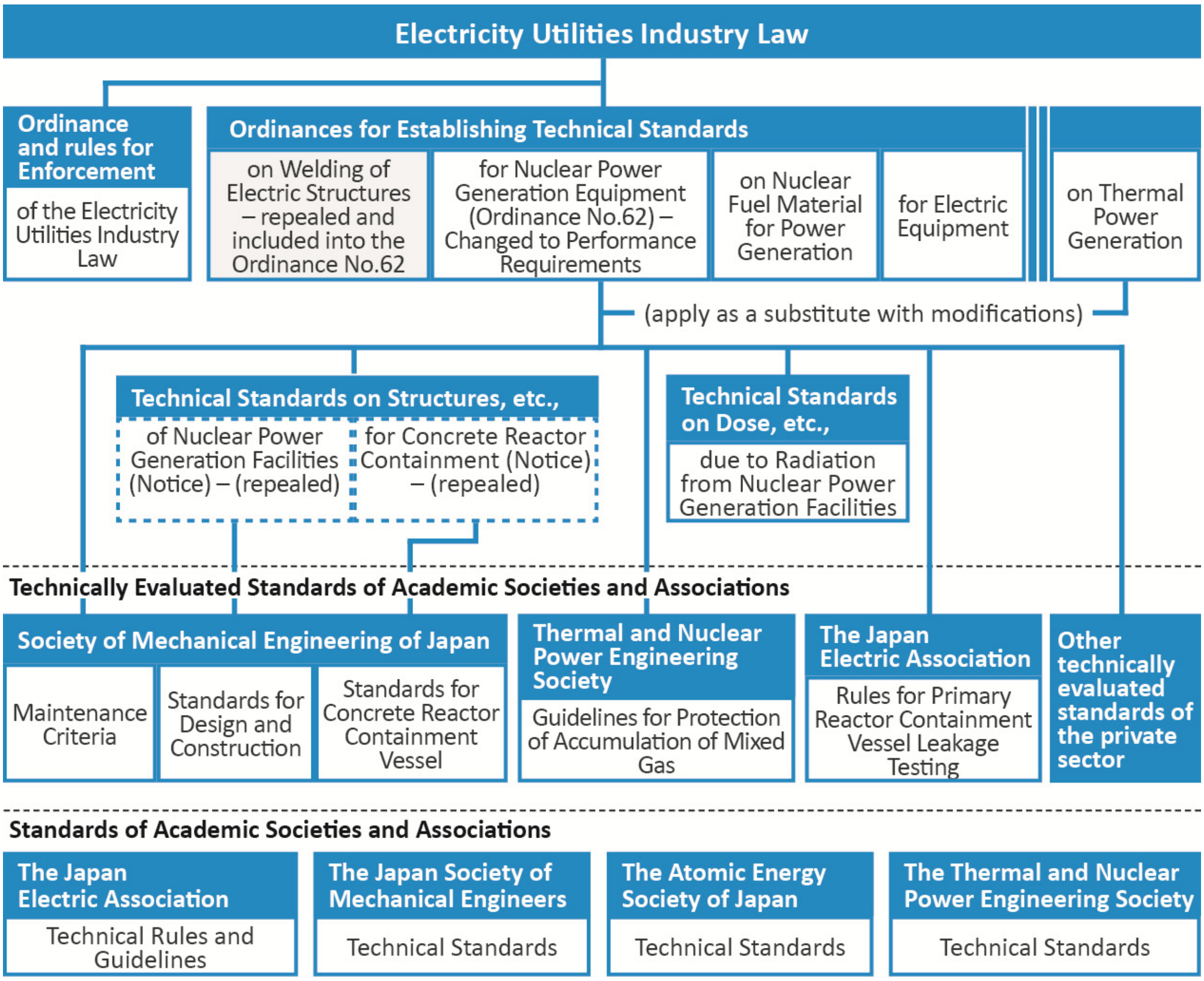


FIG. 2.5–3. System of technical standards for NPPs in Japan.

The Japanese regulatory framework did not include regulations specifying the requirements and criteria for safety upon which regulatory judgements, decisions and actions were based. A series of guides were issued by the NSC which were regarded as requirements in practice. The NSC guides were supplemented by consensus standards published by professional and academic societies.

With regard to tsunami safety, the 2006 revision of the NSC Seismic Safety Guidelines did not contain concrete criteria that could be used in re-evaluation of tsunami hazards. Therefore, an effective regulatory framework was not available to provide for the tsunami safety of the NPPs through their life [99].

2.5.10. International cooperation

Japan has participated in many multilateral and bilateral cooperation activities over the years. Regarding the CNS, Japan participated in the discussion for its creation from the early phase and became a Contracting Party in May 1995. Since then, Japan has participated in all the Review Meetings. In the third Review Meeting, Commissioner Soda of the NSC served as the President. Japan became a Contracting Party to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management in August 2003. Japan was also a party to the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency and the Convention on Early Notification of a Nuclear Accident since 1987.

Japan has also actively participated in other activities of the IAEA. For example, it participated in the International Nuclear Safety Group (INSAG) for many years and also the Commission on Safety Standards (CSS) and its four committees, the Nuclear Safety Standards Committee, Radiation Safety Standards Committee, Waste Safety Standards Committee and the Transport Safety Standards.

In 1997, the IAEA launched an extrabudgetary programme aimed at safety enhancement in South-east Asia, Pacific and Far East countries which evolved into the Asian Nuclear Safety Network (ANSN) in 2002. Japan has played a major role in providing funds and human resources to support the activity. After the Niigata-Chuetsu-Oki earthquake that struck the Kashiwazaki-Kariwa NPP in 2007, the IAEA established the International Seismic Safety Centre (ISSC) in 2008 to which Japan has contributed from the beginning.

With regard to the safety services supplied by the IAEA, Takahama Units 3 and 4 of the Kansai Electric Power, Company (KEPCO) invited an OSART mission in 1988; Hamaoka Units 3 and 4 of the Chubu Electric Power Company in 1995; Kashiwazaki-Kariwa NPP of TEPCO in 2004; and Mihama Unit 3 of KEPCO in 2009. The Nuclear and Industrial Safety Agency (NISA) in METI received the IRRS mission in 2007.

Japan is also a member of the OECD Nuclear Energy Agency (OECD/NEA). It has participated in the activities of the Committee on Nuclear Regulatory Activities (CNRA), Committee on Safety of Nuclear Installations (CSNI), Committee on Radiation Protection and Public Health (CRPPH) and Radioactive Waste Management Committee and on related Working Groups and Task Groups. For example, Japanese delegates have participated for many years in the operating experience feedback activities in the Working Group on Operating Experience (WGOE) in CNRA. Also, many research engineers and experts from the Japan Atomic Energy Agency (JAEA), Japan Nuclear Energy Safety Organization (JNES) and others have joined the safety research activities under CSNI including the Joint Research Projects. JAEA has hosted some of these projects.

The International Nuclear Regulators Association (INRA) has held meetings every year since 1997 where the top regulators from Canada, France, Germany, Japan, the Republic of Korea, Spain, Sweden, the United Kingdom and the United States of America have discussed regulatory issues. From Japan, both the NSC and NISA have attended these meetings.

Japan has had active bilateral cooperation activities for many years. For example, information exchange on nuclear regulation has continued regularly with China, France, Germany, the Republic of Korea, Sweden and the United Kingdom. The JAEA and JNES have also conducted information exchange on safety research with the USA, European countries and others.

2.5.11. Current status of regulatory requirements in Japan

On 19 September 2012, the NRA was established to replace the structure that existed prior to the accident. Following its creation, the NRA carried out a complete review of safety guidelines and regulatory requirements with the aim of formulating a set of new regulations to protect people and the environment. On 8 July 2013, the new regulatory requirements for commercial power reactors entered into force. Based on the concept of defence in depth, importance was placed on the third and fourth layers of defence and the prevention of simultaneous loss of all safety functions due to common causes. In this regard, the previous assumption on the impact of earthquakes, tsunamis and other external events such as volcanic eruptions, tornadoes and forest fires were re-evaluated, and it was decided to enhance countermeasures for nuclear safety against these external events. Furthermore, it is required to take countermeasures against internal fires and internal flooding, and to enhance the reliability of on-site and off-site power sources to deal with the possibility of an SBO.

In addition to the enhancement of countermeasures established at design basis, countermeasures for severe accident response against core damage, containment vessel damage and diffusion of radioactive material, enhanced measures for water injection into the SFP, countermeasures against malicious airplane crashes and an installation of emergency response building are also required.

The items described below are the highlights of new regulatory requirements which have been established by NRA after the Fukushima Daiichi NPP accident:

(1) Reinforced requirements for seismic/tsunami resistance.
   (a) Measures against tsunamis have been enhanced significantly. The standards define a ‘design basis tsunami’ as one which exceeds the largest ever recorded and also require protective measures, such as sea walls to prevent site inundation, and watertight doors to prevent the flooding of buildings.
   (b) Standards to deal with displacement and ground deformation have been clarified. The standards require the construction of S-class (the highest seismic safety classification) buildings and structures on the surface of the ground surface without outcrop of active fault preventing a risk of fault displacement damaging the buildings and equipment therein.
(2) Reinforced or newly introduced requirements for design basis.
   (a) Design basis and protective measures against volcanic eruptions, tornadoes and forest fires have been enhanced significantly for preventing simultaneous loss of all safety functions due to a common cause. The measures against power failure which may trigger simultaneous loss of all safety functions due to common causes other than natural phenomena have been significantly strengthened.
   (b) Measures have been strengthened for fire protection and against internal flooding as events other than natural phenomena which trigger simultaneous loss of all safety functions due to common causes.
(3) Newly introduced requirements for measures against severe accidents.
   (a) Measures to prevent core damage. Based on this requirement, in case of power failure, for example, SRVs will be opened by using mobile power sources to reduce the pressure inside the RPV until water can be injected using a mobile water injection system or other devices. After reducing the pressure inside the RPV, water will be injected into the RPV using a mobile water injection system.
   (b) Measures to prevent containment vessel failure. A filtered venting system should be installed to reduce the pressure and temperature inside the containment vessel and to reduce the escape of radioactive material through the exhaust. In additions, systems (mobile pumps, etc.) are needed to inject water into the lower part of the containment vessel to cool the core to prevent containment vessel failure due to a molten core.
   (c) Measures to suppress radioactive material dispersion. In particular, outdoor water spraying equipment to douse the reactor building and prevent radioactive material from contaminating the atmosphere.

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.

The NRA has adopted a PSR process aligned with IAEA safety standards that was enforced in December 2013. The system requires nuclear power reactor licensees to comprehensively evaluate the safety of reactors and to submit the results to the NRA within six months of the end of periodic inspection, showing the following:

— Compliance with regulatory requirements;
— Measures for improving safety on a voluntary basis;
— Assessing and reviewing safety margins for improvement and a probabilistic risk assessment;

— Comprehensive re-evaluation based on the above results and action plans for improving safety.

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.

2.5.12. Summary

The initial authorization for the Fukushima Daiichi NPP was granted in 1966, construction of the additional units being treated as an amendment to the Establishment Permit. Japanese regulations in effect at the time of the accident did not require the operator to submit or to maintain an up to date FSAR or an equivalent document describing the design and operating arrangements for the facility. Not updating the documentation which describes the revised safety case is not fully in line with IAEA safety standards.

The Japanese regulatory framework has evolved in three distinct phases since the authorization of the first units at Fukushima Daiichi NPP in the 1960s. Based on the conclusion of the 2007 IRRS mission to Japan and the peer reviews conducted under the CNS, the Japanese regulatory system that was in place prior to the accident had the organizations and systems necessary to meet IAEA safety standards. However, there were weaknesses in the implementation of these systems and the performance of the organizations such as unnecessary constraints on inspections, limited enforcement authority, inability to compel operators to make safety upgrades to their facilities, and difficulty in staffing and training at the regulatory authority.

The Japanese regulatory framework as it existed at the time of the accident had been created in 2001 following the JCO's Tokaimura criticality accident in 1999. It was then revised following discovery of the falsification of records by operators in 2002 and 2006. The organizations involved were [93]:

— The Nuclear and Industrial Safety Agency (NISA), which was part of the Ministry of Economy, Trade and Industry (METI). NISA was responsible for regulating the safety of nuclear energy as well as industrial safety and had about 260 staff.
— METI was also responsible for ensuring a stable and efficient energy supply, including the use of nuclear energy.
— The Ministry of Education, Culture, Sports, Science and Technology (MEXT) was responsible for the nuclear energy policy, the development of nuclear technologies, protection against radiation hazards, the use and transport of nuclear and radioactive material and safeguards. MEXT also supervised the Japan Atomic Energy Research Institute and the Japan Nuclear Cycle Development Institute, which merged in 2005 to become the Japan Atomic Energy Agency.
— The Nuclear Safety Commission, which reported to the Cabinet Office and provided a double-check on NISA's regulatory activities.
— The Ministry of Foreign Affairs (MOFA), which was responsible for international aspects of nuclear energy, including the implementation of related international treaties and conventions.
— The Japan Nuclear Energy Safety Organization (JNES), which was established in 2003 as a technical support organization to NISA. Its functions included conducting inspections at nuclear facilities, reviewing licensees' periodic inspections, safety analysis and evaluation, nuclear emergency preparedness support, and research and testing for code/standard development.

The independence of the regulatory body from entities with interests that could unduly influence its safety related decision making is another important safety principle. NISA as a department within METI was subordinate to the Minister and hence lacked the requisite de jure independence from entities responsible for the promotion of nuclear energy. NISA suffered from the expectation that its staff, like all civil servants, rotate jobs every two to three years and had struggled to maintain its budget within the government. NISA's regulatory inspection programme was rigidly structured by law, including the type and frequency of inspections. The presence of NISA inspectors was required before the operator could complete certain tests and surveillances, and there were only specific periods in which NISA was permitted access for inspection. This approach limited the ability of NISA to perform inspections as needed and made it difficult for NISA to change the inspection type or frequency. Following the discovery of the falsification of data records by operators, enhancements to the inspection system were instituted in January 2009 introducing greater flexibility for regulatory inspections [93].

2.5.13. Observations and lessons

— **Where several bodies have responsibilities for safety, the government needs to effectively coordinate their regulatory functions to avoid omissions or duplications that may jeopardize safety.**
  The Japanese nuclear legal framework comprised numerous laws and arrangements at different governmental levels that had been revised over the years in response to accidents and incidents. Many government ministries, agencies and quasi-governmental organizations played influential roles with regard to the utilization and regulation of nuclear energy in Japan. These complex arrangements appear to have hampered the implementation of effective actions and regulatory activities for enhancing safety. The NSC was an important body with both advisory and supervisory roles in the system of subsequent regulation review. In the opinion of the IAEA's IRRS mission, the role of NISA as the regulatory body in relation to NSC was in need of clarification.
— **The regulator should require that the operator of a facility update its safety demonstration on an ongoing basis to reflect changes in the status of the facility.**
  Although improvements in safety rely primarily on the actions of operators, regulatory oversight will be a driving force. In particular, regulatory bodies should promote continuous safety improvement processes, fostering an environment that encourages licensees to invest in improvements beyond national requirements. The regulatory body needs to review and approve the safety demonstration. In addition, any proposed modification of a facility that might significantly affect safety needs to be subject to review and assessment by the regulatory body.
— **Regulatory independence, competence, strong legislative authority and adequate resources, including qualified personnel, are essential in order to perform the required regulatory functions.**
  NISA lacked the requisite de jure independence from the entities responsible for the promotion of nuclear energy. In addition, the budget pressures and administrative rules for job rotation of civil servants to which NISA was subject limited the technical and regulatory competence of its staff. The regulatory body needs to make independent regulatory judgements and decisions, free from any undue influences that might compromise safety, such as pressures associated with changing political circumstances or economic conditions, or pressures from government departments or from other organizations.
— **The regulatory body needs to review and inspect the safety of a facility throughout its lifetime.**
  The goal is to verify: the extent to which the facility conforms to safety standards and practices; the extent to which the licensing basis remains valid; the adequacy of the arrangements that are in place to maintain safety until the next PSR or the end of plant lifetime; and the safety improvements to be implemented to resolve the safety issues that have been identified. These reviews and inspections require unimpeded access to documentation and the plant.

2.6. HUMAN AND ORGANIZATIONAL FACTORS

2.6.1. Introduction

This section analyses the question of why the Fukushima Daiichi nuclear accident happened despite the nuclear community's progress in nuclear safety, brought about by internationally agreed upon safety standards, comprehensive review services, well developed regulatory frameworks, establishment of effective safety assessment methodologies, rigorous approaches to risk management, solid design, years of successful operating experience, defence in depth, emergency preparedness, severe accident management guidelines (SAMGs) and a strong international commitment to enhance nuclear safety.

The causes of the accident go beyond the above mentioned factors and are related to the ability of organizations to detect latent flaws in the sociotechnical system, i.e. the interaction between the human, technical and organizational factors [1]. They reach the deepest level of safety culture — the basic assumptions about understanding and perceptions of reality related to nuclear safety [101]. This level of safety culture is the basis from which people operate — and hence the basis from which decisions and actions are taken. This section examines the basic assumptions on nuclear safety by the main stakeholders involved in the accident at the Fukushima Daiichi NPP. The systemic analysis will show how the actions of stakeholders were interrelated and interconnected, and thereby reinforced basic assumptions about nuclear safety that prevented them from adequately preparing for and preventing the accident on 11 March 2011. It should also be pointed out that the safety culture within the Japanese stakeholders' organizations was not sufficiently evaluated before the accident to identify proactively the areas for improvement [32, 54, 102]. This is not unique for Japan, as effective, continuous safety culture improvements constitute challenges for all organizations globally. There is no perfect safety culture condition; it is an ongoing process to assess and advance the understanding of the organizational basic assumptions which drives nuclear safety [1, 103-106].

The Fukushima Daiichi accident was characterized by a high degree of complexity. Not only did this complexity constrain those who were directly involved in the accident response, but it also continues to affect the understanding of the accident. The following biases (see Annex II of this volume for a more detailed account of these biases) influencing the understanding of, and lessons learned from, complex accidents are discussed in this section:

- The hindsight bias explains the pitfalls of understanding an event retrospectively. Processes are often judged by their outcome, and the knowledge of the outcome thus deeply influences the understanding, potentially leading to an event being seen as "more predictable after it becomes known than it was before it became known" [107, 108].
- Oversimplification: Despite the efforts made to analyse the Fukushima Daiichi accident from many different perspectives, what happened is described in a chronological manner, unit by unit, topic by topic. This natural constraint in writing accident reports can create an oversimplification of the full picture making it hard to acknowledge the full complexity the actors had to deal with.
- Distancing through differencing is a common human mechanism preventing understanding and learning. This happens through people's excessive focus on the "differences, real and imagined, between the place, people, organization and circumstances where an incident happens and their own context", leading to a failure to see lessons applicable for their own operation [109].

The lessons to be learned from a human and organizational perspective are applicable to all States and all organizations worldwide, in order for them to proactively detect the flaws in the sociotechnical system to become resilient before and during the emergence of something unexpected.

2.6.2. Methodology

The human and organizational analysis was conducted in accordance with social and behavioural science procedures, which comprise four equally important elements:

- Recognized methodology;
- Unbiased data;
- Scientifically recognized theory;
- Knowledgeable experts.

The methodology used for this human and organizational assessment is based upon the IAEA safety culture assessment methodology. This has been successfully applied in evaluating safety culture at licensees' sites, corporate centres and their interactions with regulatory bodies. For the purpose of this Technical Volume, the IAEA safety culture assessment methodology was extended and applied to a larger sociotechnical system encompassing not only safety culture but also the whole range of interactions between human, organizational and technical factors as well as interorganizational interactions [1] as outlined in more detail in Annex II of this volume. Consideration was also given to best practices in other industries and state of the art sociotechnical approaches applied to evaluate complex events. In order to capture the relationships and synergies among the human and organizational factors contributing to the Fukushima Daiichi accident, this systemic approach was used for the analysis of the accident and for the development of this section. A systemic approach allows for the consideration of the complex relationships of a sociotechnical system with a broader network of other sociotechnical systems, and how those affect and are affected by each other. This includes "factors that lie far away in time and space from the moment things went wrong" [110].

The methodological approach to conduct a systemic analysis captured factual data from a number of sources, including primary and secondary sources, expert meetings and information provided by Japanese experts, and information provided by other experts from the working groups. The systemic and systematic methodology encompassed a number of steps. As the human and organizational factors permeated the available information sources, the factual related information was first extracted, examined and cross-checked by the experts. The information was subsequently categorized using a classification system, and entered into a sortable database. This resulted in the accumulation of approximately 4900 facts classified into 26 categories, 96 attributes, 3 periods of time (before, during and after the accident of 11 March 2011) and various interrelated stakeholders. The facts were identified and categorized as part of an iterative process involving the examination, validation and discussion within the team of experts on human and organizational factors. The assessment and lessons learned presented in this section are all based on the facts in the database.

Facts from selected categories of the database were further examined for themes[7] through a mapping process to identify relationships, concepts and trends. The final analytical step consisted of determining the important relationships, concepts and trends to fit together into an overarching theme. This process was accompanied by focused discursive analysis that capitalized on the diversity of scientific perspectives, knowledge and experience of the experts.

The facilitated discussions, structured readings, database construction and thematic analysis were all essential parts of this systemic analysis and enhanced the understanding of the sociotechnological features that led to an understanding of the evolution of certain aspects of the accident.

There are inherent biases in understanding a complex accident such as the one at the Fukushima Daiichi NPP, which may constrain the opportunity to learn. Some of these biases were introduced above and are further explained in Annex II of this volume, namely: hindsight bias, oversimplification and distancing through differencing.

The human and organizational assessment of the Fukushima Daiichi accident has taken these biases into account during the analysis of the data. The identified generic systemic safety factors contributing to the accident should be considered for all nuclear stakeholders worldwide, not only for the Japanese stakeholders. The human and organizational learning opportunities are valid for each and every organization in enhancing nuclear safety to become more resilient to unexpected situations.

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[7] Themes are patterns identified through analysing cultural expressions, such as physical artefacts, verbal expressions, espoused values and norms.

It is critical to keep in mind that the concepts and definitions used in this section are not unique to Japan, but, rather, they are applicable to all societies.

2.6.3. Basic assumptions of the main stakeholders regarding nuclear safety and their impact on the conditions for the Fukushima Daiichi accident

2.6.3.1. The basic assumption related to the accident

To understand the Fukushima Daiichi NPP accident, it is necessary to explain the underlying mechanism of human and organizational behaviour in the context of safety culture, i.e. basic assumptions about nuclear safety. Knowledge of safety culture concepts is recognized as a way to enable the understanding of why people in organizations act the way they do in relation to safety. In both theory and practice, there are different levels on which safety culture operates. The first level is the visible, covering, for example, behaviour and physical objects; the second level is more indirect, covering the shared values, attitudes; and, finally, the deepest level covers the basic assumptions about nuclear safety [111]. In this analysis of the Fukushima Daiichi NPP accident, the focus is on the basic assumption about nuclear safety which preconditioned the actions and decisions prior to the accident. These assumptions were shared by the main stakeholders' organizations.

Assumptions are socially constructed when people interact with each other. People develop a common understanding of the things around them and come to share a common assumption of what those things mean [111-113]. When exposed to information from the world around them, people perceive and interpret the information, build expectations of the future based upon it and then act on the basis of these processes. The basic assumption resulting from this process is maintained by people and reinforced by human cognitive processes to uphold a shared reality and to reduce ambiguity about uncertainties. This cognitive process also helps justify the rationale behind decisions already taken and helps create stability. This is a well known phenomenon that is generally a healthy and necessary cognitive mechanism to cope with complexities and uncertainties. Yet, unfortunate effects of this phenomenon have been observed in many accidents in safety critical industries.[8]

Consciously and/or unconsciously, humans tend to select information which does not jeopardize their basic assumption in order to avoid inconvenient information. As will be shown throughout this section, this happened in many instances with regard to the Fukushima Daiichi accident. For example, NISA did not agree to proposals to change the emergency preparedness for a major accident in part because the agency had previously informed citizens that a major accident would not happen [102]. Over time, such collectively developed and basic assumptions and understanding of how the world functions become the socially accepted reality in which people act and interact and which reproduces itself in the way in which new information is integrated from the outside.

2.6.3.2. Why the Fukushima Daiichi accident was unexpected

One feature of improving safety culture is to examine the scope and boundaries of the basic assumption about nuclear safety, i.e. determining what are seen as plausible risks which can jeopardize nuclear safety. Two sets of knowledge are located within the boundaries of the basic assumption. Basically, it is possible to identify issues which can be regarded as known, i.e. so-called known knowns, and issues that humans are aware that they do not know about, i.e. so-called known unknowns [115]. An example of known knowns is the fact that earthquakes and tsunamis occur. An example of known unknowns is when an earthquake will occur or which magnitude it will have — people are aware of not knowing these parameters.

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[8] This has been shown, for instance, in aviation research; see Ref. [114].

Despite knowing about the possibility that it might happen, humans are surprised when a known unknown occurs. This kind of surprise is defined as a situational surprise [116, 117] and can be regarded as a surprise within the boundaries of one’s basic assumption. Its consequences are, in principle, preventable by utilizing available knowledge and information to anticipate and prepare for handling the known unknown (e.g. by building the infrastructure able to withstand earthquakes up to a certain envisaged magnitude). Probabilistic and deterministic safety analyses thus capture potential risks posed by known unknowns within their domains.

However, as the analysis below shows, the Fukushima Daiichi NPP accident was not part of either the known knowns or the known unknowns for the stakeholders involved, as several facts demonstrate that they were not able to imagine that such a complex event could occur. Importantly, when looking at each of the elements that were part of the accident in a separate manner, one may conclude that several of the precursors of the accident were known knowns and known unknowns. For example, as explained previously, calculations were made prior to the accident, showing a relatively high likelihood of a beyond design basis tsunami height at the Fukushima Daiichi NPP, making the tsunami a known unknown for some stakeholders.

Nevertheless, when investigating the accident from a human and organizational perspective, evidence showed that the stakeholders involved lacked an understanding of the systemic implication of the nuclear risk. Simply put, the stakeholders were not able to understand and anticipate the combined effect of the multitude of events taking place almost simultaneously.

Table 2.6-1 shows the different elements of knowledge that belonged to the known knowns and known unknowns of the relevant stakeholders prior to the accident, e.g. the elements of knowledge situated within the boundaries of the basic assumption held by these stakeholders. It further illustrates that, by combining these different elements, one can describe how the stakeholders involved experienced the situation triggered on 11 March 2011. As they had not imagined that the different elements in the table could combine in the way they did when the tsunami hit the Fukushima Daiichi NPP, they were effectively facing a surprise outside the boundaries of their basic assumption. It should be underlined that this table serves as an illustration encompassing only a few of the entirety of events that the people involved in accident response were faced with.

TABLE 2.6–1. EXAMPLES OF KNOWN KNOWNS, KNOWN UNKNOWNS AND UNKNOWN UNKNOWNS RELATED TO THE FUKUSHIMA DAIICHI NPP ACCIDENT

Topic	Known knowns	Known unknowns	Unknown unknowns
	Within the boundaries of the basic assumption of main stakeholders		Outside the boundaries of the basic assumption of main stakeholders
Seismic event	Design basis of the Fukushima Daiichi NPP and prediction of its design basis seismic performance.	Response of the Fukushima Daiichi NPP to a seismic event exceeding its design margin.	—
Tsunami	Tsunamis are co-related events to seismic events.	The prediction of tsunami heights.	—
Loss of off-site power (LOOP)	The Fukushima Daiichi NPP was designed for a loss of off-site power, with the expectation that the grid would be restored quickly.	The extent of grid damage and infrastructure disruption for a very large earthquake and tsunami.	—
SBO	Should SBO occur, the DC power (batteries) would last for about 4-6 hours. If AC was not restored in that time, core damage was to be expected.	—	—

TABLE 2.6–1. EXAMPLES OF KNOWN KNOWNS, KNOWN UNKNOWNS AND UNKNOWN UNKNOWNS RELATED TO THE FUKUSHIMA DAIICHI NPP ACCIDENT (cont.)

Topic	Known knowns	Known unknowns	Unknown unknowns
	Within the boundaries of the basic assumption of main stakeholders		Outside the boundaries of the basic assumption of main stakeholders
Batteries	Designed to provide power for about 4–6 hours.	How the batteries could be managed in an event.	—
EDGs	Expected to function on LOOP to provide AC power. Endurance based on diesel stock replenishment capability	Diesels can fail to start and duration of service may be unpredictable.	—
Switch gear	Interconnections allow cross feeding of power from one unit to its neighbour.	Availability of equipment and/or staff to effect interconnection in a severe accident.	—
Procedure for notification of a nuclear emergency	After NPP Site Superintendent is notified or discovers the specific event defined in Article 10 (1) of the Nuclear Emergency Preparedness Act, he/she shall, within a targeted time frame of 15 minutes, simultaneously notify all relevant entities by fax.	Site Superintendent's perception of severity of event, communication infrastructure available.	—
Staffing on-site	Minimum number of staff available on-site at the beginning of the accident at a given moment.	Capability to relieve staff if severe condition persists over prolonged period in case of damage to outside infrastructure.	—
Capability of staff for accident response	Formal competences of staff to respond to an anticipated type of accident (training, experience).	Psychological and physical condition and ability of staff to respond to an event under severe conditions in a given moment	—
Sum	—	—	**combination of all elements above = Unknown unknown**

When events are experienced as totally unexpected, they constitute so-called unknown unknowns, as people are not aware of the fact that they do not know about them — they happen outside the boundaries of their basic assumption. At the moment they occur, they are so unexpected and astonishing that one can neither understand nor explain them through the normal application of the knowledge and information available before their occurrence. Unknown unknowns profoundly surprise humans and suddenly reveal the incompatibility between one's self-perception of what is possible and what actually happens [118]. As such surprises lie outside the basic assumption of reality, even the availability of advance information is hardly useful in preventing people from experiencing such surprises because of the human tendency to interpret information in a way that suits one's basic beliefs [116].[9]

[9] In studying the accidents at the Three Mile Island and Chernobyl NPPs, researchers such as Reason [119] and Woods [120] differentiate between 'situational' and 'fundamental' surprises and utilize the term 'fundamental surprise' to describe a surprise that "reveals a profound discrepancy between one's perception of the world and the reality" [119], i.e. a surprise occurring outside the boundaries of one's basic assumptions. This term was originally coined by Lanir [116].

As shown in Table 2.6-1, such unknown unknowns can occur through the complex outcome created by the combination of several known knowns and known unknowns. As mentioned above, although advanced knowledge about the different elements present in the Fukushima Daiichi accident was available, much of this knowledge was not effectively shared among the stakeholders but only known by individual groups of people prior to the accident.[10] Both this lack of sharing knowledge and the general lack of realizing that the combined knowledge elements could reveal the possibility of an accident can be attributed at least in part to the basic assumption about nuclear safety held by the main stakeholders. As shown in Fig. 2.6-1, this basic assumption led the stakeholders to believe an accident of this kind would not happen and resulted in their conscious and unconscious rejection of the importance of certain knowledge elements and undermining the importance of sharing and combining them.

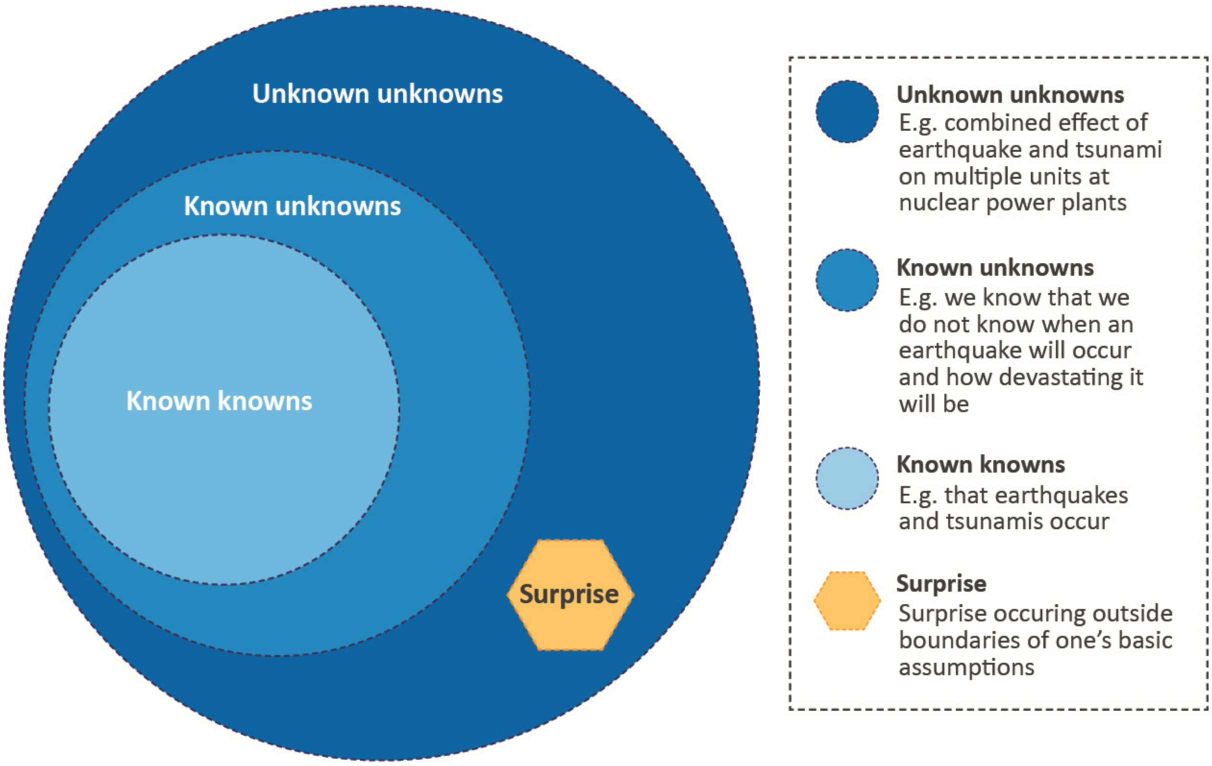

FIG. 2.6–1. The impact of unknown unknowns.

From a human and organizational perspective, it is concluded that the Fukushima Daiichi NPP accident was a surprise outside the boundaries of the basic assumption for the key stakeholders and that this prevented these stakeholders from effectively anticipating, preventing and mitigating the effects of the accident on 11 March 2011.

2.6.3.3. Nuclear safety in Japan

Once Japan decided to embark upon a nuclear power programme, many organizations began building the necessary nuclear infrastructure. TEPCO was one of the first companies to be licensed to construct and operate an NPP in Japan. Unit 1 of the Fukushima Daiichi NPP was the first TEPCO plant to operate. This reactor was a US design and was turned over to TEPCO after construction [54]. After the

[10] It should be noted again that, in hindsight, it is much easier to see the trajectory of an accident than it is to recognize the complexity and the systemic implications before and while the accident occurs [107, 108].

startup of Unit 1, there was a significant expansion in the development and use of Japanese nuclear power technology. Among other factors, the oil crisis of the 1970s was an impetus for this rapid development, as Japan wished to further reduce its dependence on energy imports [121]. By 11 March 2011, nuclear power accounted for nearly one third of the country's electrical energy production. The aim was to maintain or increase this capacity to 30–40% by 2030 [122].

In order for the decision to be made to use nuclear power for electricity production both in Japan and across the world, the risks of nuclear power needed to be understood and controlled. Advanced technological solutions were developed to provide robust safety margins. Over time, it became evident that not only the technical design, but several factors such as the associated infrastructure (procedures, processes, systems, etc.) and human and organizational factors also needed to be considered as part of operational safety. Multiple methods were developed and applied to calculate and assess the risks associated with nuclear power, such as PSAs and deterministic safety assessments. In addition, legislative frameworks, regulatory infrastructures, oversights and assessments (e.g. independent national regulators and IAEA peer reviews), and management systems were gradually put in place.

In addition to meeting the technological challenge of making nuclear power plants as safe as reasonably possible, it was essential to make sure that the public would consider nuclear energy production as safe, in order to gain support for its implementation and use.

The assumptions surrounding nuclear matters and nuclear safety were developed within the different main nuclear industry stakeholders over time. The prevailing assumption was that postulated risks could effectively be eliminated or mitigated through a combination of a robust technical design, the associated infrastructure (procedures, processes, systems, safety standards, etc.) and the capability of humans and organizations of operating and controlling the plant. This basic assumption that the robustness of the technical design and existing measures would maintain and protect the safety of nuclear plants against postulated risks was developed, maintained and mutually reinforced among the main stakeholders[11] in the operation of the Fukushima Daiichi NPP, i.e. the licensee, the regulatory body and the public. As a result of the prevalent technical assumption that the design and existing measures would be sufficient to maintain safety, non-technical factors such as the associated infrastructure and cultural, human and organizational factors were not adequately assessed and strengthened — particularly regarding the treatment of low probability, external events, accident management procedure improvements, and general actions to prepare for a complex accident.

2.6.3.4. Stakeholder: The regulatory body

This analysis of the Fukushima Daiichi NPP accident suggests that relevant regulatory bodies acted on a basic assumption considering the robustness of the technical design and existing measures to protect the nuclear plants against postulated risks as sufficient to ensure nuclear safety. In some instances, NISA did not agree to improve some safety standards proposed by other bodies. For example, when the Working Group for Reviewing the Regulatory Guide for Emergency Preparedness for Nuclear Facilities of the NSC proposed introducing the concept of the precautionary action zone (PAZ), NISA pointed out that:

> "in Japan it was extremely unlikely that a serious accident leading to a release of a large amount of radioactive materials would occur; even if such an accident occurred, it was unlikely to continue for a long period of time, and thus, there was no need to immediately evacuate residents within a 5-km radius of a nuclear power station [102]."

Rationalizations such as these served to protect the collective trust in safety [102]. Prioritization of tasks which were compatible with the basic assumption of nuclear safety and procrastination in

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[11] Other stakeholders were also involved but to a lesser degree.

decisions and activities that would have led to a reconsideration of the accepted approaches and beliefs [102] were among other collective and individual mechanisms that led to a continuation of the existing situation.

The role of the regulatory bodies

IAEA Safety Standards Series No. GSR Part 1 [123] states in Requirement 17 that regulators shall hold a strong and independent position and be able to exert the necessary authority over the industry they oversee. NISA did not hold such a position, was not given sufficient authority and was thus not able to challenge its own basic assumptions or those of others regarding nuclear safety [32]. Regulatory bodies have both a formal role and authority, as provided in laws and regulations [99] and official mission statements (de jure authority), and a more informal role that it is implemented and perceived in day to day situations (de facto authority). In order to be effective in fulfilling its intended mission of ensuring protection of the people and the environment, a regulatory body not only needs to be provided with the necessary formal power and legal framework, but it also needs to be accepted by the operator. NISA was apparently unable to practice either de jure or de facto authority at the time of the accident. The affiliation of NISA with METI, which was responsible for the promotion of nuclear energy in Japan [32], as well as NISA's dependence on other governmental bodies for personnel and budget management issues [32, 53], appears to have hampered its formal independence. Moreover, the distinction between NISA's role and that of the NSC was not clear, as identified by the IAEA Integrated Regulatory Review Service (IRRS) team during its mission to Japan in 2007 [53] (see also Section 2.5.3. for a description of NISA's and NSC's formal roles as provided in the Japanese legal framework). NISA was often perceived as not having or exerting a strong influence on the nuclear industry, including on TEPCO [32]. It appears that NISA was also not fully respected by TEPCO as a strong regulatory body [32], despite some instances in which NISA showed strength in enforcing its decisions, for instance after it became known that TEPCO had covered up problems in the late 1990s and early 2000s [54].

NISA's lack of authority seems most likely to have been at least partially linked to a series of constraints. NISA was inhibited in the fulfilment of its oversight role by the lack of an appropriate regulatory framework [6] and by explicit legal constraints. For instance, the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities [9] was revised in 2006 to also address tsunamis as an event accompanying earthquakes [99], but there was no legal framework to apply this guideline retroactively to NPPs licensed before the guideline was in place [32].

NISA also faced restrictions of its oversight activities due to explicit legal constraints. 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 [53]. In addition, there were also organizational constraints. NISA and the NSC seemed to suffer from a lack of resources [102]. Moreover, NISA's ability to properly adhere to its duties was also hampered by a lack of competence, making it dependent on the competence and expertise of the operators in carrying out its tasks [32]. This lack of competence is considered to be in part linked to the government's staffing policy [32]. NISA's employees comprised two categories, the policy makers and the experts [53]. The policy makers were recruited from universities to METI and rotated within different departments of METI, including NISA. They often had little specific knowledge relevant to the work at NISA and they stayed within NISA for only a few years. The experts were recruited from the industry, with specific knowledge and expertise in the nuclear field. This category of staff rotated within NISA, according to the government rule which required a job rotation every two to three years [53]. However, mastering and understanding of the technology and the tasks associated with a nuclear regulatory body can take a number of years; as a result, it would be very challenging for rotating staff (whether as policy makers or experts) to become fully effective in their nuclear regulatory role over such a short duration.

The relationship of the regulatory body with the licensee

The human and organizational factor analysis suggests a lack of effective independence of NISA and the NSC from the industry they regulated. An example of this is the active influence of the operators, through the Federation of Electric Power Companies (FEPC) of Japan [32], and the active involvement of the licensees in writing reports or guides or their participation in regulatory forums and committees [32, 102]. For instance, the NSC invited licensees to be heavily involved in the process of revising the Safety Design Guide with regard to SBO. The resulting report noted that an SBO lasting for many hours may cause damage to the reactor core, but emphasized the high reliability of external and emergency power sources in Japan; it thus concluded that nuclear power plants had sufficient defences against an SBO. As a result, the Safety Design Guide remained unchanged up to the time of the Fukushima Daiichi accident [32].

Besides the issue of effective independence, the nature of the day to day relationship between a regulatory body and the organizations it oversees and the regulatory approach taken by the regulatory body need to be considered. The interaction between a regulatory body and an operator is a two way relationship, in which both organizations exert explicit or implicit influence on each other. The basic assumptions about nuclear safety are critical — they will direct the behaviour of the actors but will also be mutually shaped through this interaction process. The way the regulatory body fulfils its oversight and regulatory activities influences the licensee's behaviour and its attitude to safety, and vice versa. For example, the way in which the licensee follows requirements will influence the expectations and priorities of the regulatory body in the future.

The analysis has shown that the regulatory body in Japan worked in a non-strategic or reactive manner in its regulatory and oversight work. For instance, since its establishment in 2001, NISA focused on short term activities reacting to various incidents in Japanese nuclear installations [102] and dealing with the various falsifications of safety records that were revealed from the late 1990s onwards [54]. It was therefore not able to address more fundamental and long term issues, such as the consideration and implementation of international fundamental safety principles and developments [102]. NISA's oversight and regulatory activities were often based on compartmentalized thinking, i.e. it did not sufficiently address issues in a broad, systemic manner, considering all aspects relevant to safety [53, 102]. Particular emphasis was placed on technical issues, compared with operational aspects and human and organizational factors [53], although efforts were made by NISA to improve in this respect [53]. In some instances, the regulatory body demonstrated limited risk awareness (e.g. concerning the relevance of tsunamis for the safety of Japanese nuclear installations [99]), acceptance of insufficient assumptions (e.g. about the possibility and probability of a severe accident [32]), a limited sensitivity for safety relevant information and a lack of urgency to act upon such information [99], thereby precluding the swift implementation of appropriate safety measures.

NISA demonstrated a lack of enforcement and failure to take actions in its oversight activities [99], or exhibited a tendency to procrastinate [102]. The relationship between the regulatory body and TEPCO was described as "cozy" in the report by the National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission (Diet Report) [32]. It was a typical pattern for NISA to refrain from formulating clear requirements towards TEPCO in a written form [32] and enforcing their fulfilment [32, 102]. Many issues were recommended to the operators to be implemented on a voluntary basis [53, 99]. The lack of competence at NISA did not contribute to a constructive dialogue on safety issues and regulations with TEPCO [54].

In addition, the regulatory bodies were less disposed to learning from international experience [32], showing a clear tendency for isolation [102], frequently arguing that lessons learned and approaches from other countries were not applicable to Japan [102]. In conclusion, the basic assumption that the regulatory bodies were acting upon dissuaded them from taking firm actions to enhance safety.

The relationship of the regulatory body with the public

There was a frequent concern at NISA that stricter regulations or new safety requirements would be interpreted by the public as an indication that the nuclear installations were not sufficiently safe [102], and that this would cause confusion and anxiety among people [99]. This perception led NISA to develop a lack of transparency in its communication with the public [32]. Although in hindsight there is evidence of failures on the side of the regulators and for flawed regulatory practices, the overall picture conveyed may include some hindsight bias. It should be noted that the regulatory bodies in Japan, just as regulatory bodies all over the world, were indeed engaged in continuous efforts to improve their regulatory practices and frameworks, as was also shown by the IAEA IRRS mission in 2007 [92]. In particular, they include lessons from some events in the beginning of the 2000s or the JCO's Tokaimura criticality accident. Therefore, although the picture might seem clear in retrospect, it might have not been evident to the regulators at the time.

If the regulator requires safety related improvements to an operating plant, it may indirectly trigger the assumption among the public that the plant, without the improvement, would be unsafe. On the other hand, if the regulator does not request improvements, it may indirectly perpetuate the notion that the safety of the plant is adequate and that risks are well managed. The regulator is particularly challenged by having to simultaneously carry out its role and mandate to protect the people and environment, while, at the same time, being conscious of the implications of its decisions, including the public perception of safety.

2.6.3.5. Stakeholder: The public

As the Japanese Government decided to establish its nuclear power programme, there were multiple reasons why, to some extent, the term 'nuclear' elicited negative perceptions among the public. Faced with this situation, the importance of assuring the public that nuclear technology could be trusted, and that an accident was most unlikely to occur, was therefore present from the start of the Japanese nuclear power programme. As described above, the basic assumption about nuclear safety was built upon the technical robustness of nuclear engineering capabilities. Beginning with the first stages of searching for sites for nuclear installations, the national government provided extensive support to the electrical power utilities, in order to assure the local governments and communities of potential sites of the reliability and safety of nuclear power plants. Although governmental support of an important national industry is common, it has been pointed out that the relationship resulted in lost opportunities to increase safety that might have come about through interaction with an engaged and informed civil society [121, 124].

The basic assumption concerning nuclear safety was also affected by the way in which radiation and radiation effects were treated. Several Japanese accident investigation reports pointed out the need to improve the Japanese educational system, to provide a sufficient understanding of radiation and its effects [32, 99, 125]. Investigating "the status of how the topic of radiation had been taken up in the school curricula in the compulsory school education history in Japan", the Investigation Committee on the Accident at the Fukushima Nuclear Power Stations of Tokyo Electric Power Company (ICANPS) concluded that it was:

> "hard to accept that the general public had sufficient opportunities to learn about scientific characteristics of radiation or its effect [on the] human body at schools or in a community" [99].

The Diet report concluded that the Japanese educational programme lacked the inclusion of "social and actual issues" dealing with nuclear matters in order to be able to communicate risk to the public [32]. Furthermore, the Rebuild Japan Initiative Foundation pointed out that the Ministry of Education, Culture, Sports, Science and Technology (MEXT) worked with the objective of emphasizing the positive sides of nuclear power in the educational programmes under its supervision [125].

It was stated, as part of the Diet report that, even in locations near the plant, “only 10–15 percent of residents reported receiving evacuation training, and less than 10 percent were told of the possibility of a nuclear accident” [32]. When the local population in Fukushima was asked about the first thing that came to mind after the earthquake, concerns regarding the nearby nuclear power plant were not a common response [32]. It is important to provide a balanced picture that includes a discussion of the risks posed by the technology in an open and transparent manner. By ensuring such a balanced understanding, the social preparedness to handle a nuclear emergency is enhanced, and the public is able to judge radiation risks on the basis of accurate information, avoiding both unnecessary fear and underestimation of radiation risks.

The relationship of the public with the regulatory body

The reluctance of NISA to update regulations was based, at least in part, on the fear that the public might get the impression that NPPs were not as safe as promoted, and they would consider NISA responsible [91]. While other countries with comparable nuclear programmes implemented significant updates to regulations [32], NISA contributed to maintaining the basic assumptions concerning nuclear safety vis-à-vis the public.

The concern about worrying the public about nuclear risks also characterized the decision making on preparedness for possible nuclear emergencies. After the 2007 event at the Kashiwazaki-Kariwa NPP, resulting from the Niigata-Chuetsu-Oki earthquake, the Niigata Prefectural Government considered conducting evacuation drills assuming the simultaneous occurrence of an earthquake and a nuclear emergency. The initial feedback from NISA was that the likelihood of such a scenario was extremely low, based both on the seismic structural design of nuclear reactors and the fact that the event at the Kashiwazaki-Kariwa NPP did not develop into a nuclear emergency. In addition, the relevant government agencies and local government organizations provided comments to the effect that the implementation of such a scenario would not only require major modifications to the existing emergency framework, but could also result in the public forming “the mistaken view that major natural disasters are likely to initiate a nuclear emergency” [99]. It was finally decided to abandon the idea, although, in the same year, the Niigata Prefectural Government organized a drill combining a nuclear emergency with the occurrence of heavy snowfall [99].

The efforts of the regulators to uphold the safe image of nuclear matters also resulted in a lack of transparency. For example, NISA’s efforts to protect itself from potential lawsuits was shown by its failure to disclose information regarding nuclear safety, such as proceedings from evaluations or results of assessments [32].

The relationship of the public with the licensee

Possible reactions from the public also strongly influenced the way risk management was carried out by licensees. For example, when TEPCO considered risks related to natural disasters, they were considered primarily in terms of jeopardizing public trust and increasing legislation [32]. After the accident, TEPCO admitted that there “was apprehension that, if the possibility was acknowledged that a situation exceeding design standards might occur, it might lead to the cancellation of the establishment permit or prolonged shutdown” [54]. With regards to the treatment of tsunamis, this same logic was applied:

> “We were concerned that, if the risk of a tsunami strike exceeding assumptions had been announced, then there would have been demands for absolutely safe countermeasures to

perfectly protect against any impact resulting from a tsunami. Operations would have been unavoidably shut down until such measures had been completed" [54].[12]

In general, the communication with the public, for example during hearings, on these issues, did not address the challenges of the safety of nuclear plants [32].

2.6.3.6. Stakeholder: The licensee

Many aspects of the role of the licensee have been explained above in the context of the roles of other stakeholders — the regulatory body and the public. For example, the issue of independence, the way the regulatory bodies and licensees interacted with each other on routine and non-routine activities, as well as the approach used by the regulatory body, all impacted the basic assumption held about nuclear safety. Furthermore, the way the licensee acted towards the public with regard to sharing information on the risks posed by nuclear power production is also likely to have impacted the licensee's own perception of these risks. Other aspects specifically related to the licensee are described below.

Confidence

The confidence in the technical features of nuclear safety contributed to TEPCO not implementing certain safety measures in the event of a nuclear accident. While the international community began to advance in the area of SAM, TEPCO did not adopt some of these developments. At the time of the accident, TEPCO's AM measures addressed internal cause events only [99]. TEPCO also did not examine and prepare accident management measures for a tsunami, believing that they could "respond to natural disasters including tsunami within design" [99]. In addition, the measures assumed normal operation of any adjoining reactor units and did not consider an event that could simultaneously affect multiple reactor units [99].[13] As a result, TEPCO had not planned how to address a situation where a unit lost all power and was not able to receive power from an adjacent unit.

Furthermore, after new regulations regarding SBO and the associated countermeasures were established in other countries [32][14], TEPCO and the Kansai Electric Power Company stated that "[r]eflecting SBO in the design guide is going too far". "If the intent is to make SBO a design basis accident, the concept of the design guideline would be fundamentally changed" [32]. It was subsequently agreed that the reliability of DC power sources, external power sources and diesel generators was high and that there were no cases of DC power source failures [32]. At the time of the event, there were "no legal regulations for the consideration of beyond the design basis, as Japanese plants [were] considered to be adequately safe as ensured by preventive measures" [53]. In addition, TEPCO executives and others did not assume that a natural disaster beyond design basis assumptions could occur and therefore did not establish measures for such an event [99].

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[12]TEPCO elaborated on its position as follows:

> "Looking back on the results of this accident, we believe that the risk of a tsunami strike exceeding assumptions should have been announced, and we should have communicated that the possibility of a severe accident occurring even with various safety measures in place was not zero. Also, we should have explained the necessity of measures for mitigating the effects of an accident. We believe that there were problems underlying our not having been able to provide explanations of these risks" [54].

[13] It should be mentioned that prior to the Fukushima Daiichi accident, considerations of multi-unit events had not been made by most countries operating NPPs:

> "Historically, nuclear regulators have not required nuclear operators to include multi-unit accidents at a single site or multi-site accidents in their design basis." [126] .

[14] Although other countries had implemented SBO countermeasures, an SBO of the length that occurred at Fukushima Daiichi had not been considered.

Risk management

TEPCO's decisions related to risk management — and more specifically related to low probability, high consequence events — were not sufficiently evaluated and acted upon. In several instances, the risks identified were not considered to be based on validated facts or identified by sufficiently qualified experts. Developing accurate assumptions for low probability events can be particularly challenging, as fewer examples and much less verified information exist to support them. In the case of the Fukushima Daiichi NPP accident, the height of the tsunami waves which struck the Fukushima Daiichi and Daini NPPs on 11 March 2011 was generally assumed not to be plausible. This assumption was based on the selective collection of scientific evidence and historical records and on the results of different simulations that applied methodologies and models. In 2008, when re-evaluating the tsunami risks at the Fukushima Daiichi NPP, TEPCO requested estimates provided by the Headquarters for Earthquake Research Promotion (HERP), which indicated that the wave height could exceed 15 m. During the same year, TEPCO engineers calculated another estimate of a wave height at the Fukushima Daiichi NPP of approximately 9 m, based on the wave source model presented in a study of the Jogan 869 tsunami [106]. TEPCO did not take concrete measures to protect the Fukushima Daiichi NPP from tsunamis in response to these estimates for two main reasons. First, TEPCO asked the Japan Society of Civil Engineers (JSCE) to determine the validity of the 15 m estimate, which was based on a wave source model and a postulated model with characteristics similar to an earthquake that took place in Iwate Prefecture (approximately 200 km north of the Fukushima Daiichi site). At the time of the accident, this virtually derived value remained under review. Second, the 9 m value was also seen as not sufficiently reliable as "the source model had not yet been finalized" [99, 106].

Taking into account the hindsight bias, it should be noted that TEPCO did take several actions related to tsunami protection. The design basis postulations for tsunami height were re-evaluated at least five times throughout the lifetime of the Fukushima Daiichi NPP. Tsunami height assumptions were increased from sea level +3.1 m to sea level +5.7 m in 2002, and increased again to sea level +6.1 m in 2009 to "address uncertainties in the calculated values based on improved assessment methods developed by the seismic and tsunami experts associated with the Japan Society of Civil Engineers (JSCE)" [106].[15] Seawater pumps were accordingly raised in 2002 and 2009. In order to validate the Jogan tsunami assumptions, TEPCO further arranged studies in 2009 and 2010, performing "core borings at five locations near the Daiichi and Daini sites" [106].[16] These facts illustrate that some actions were being taken. There were other instances where TEPCO did not consider safety measures based on the low probability of the initiating event [32] or lack of sufficient scientific knowledge. For example, following an FEPC discussion in 2010, a decision was made to only take internal events into consideration in safety assessments, as the assessments of "external events were considered significantly more uncertain than assessments of internal events" [32]. At the time of the accident, the Fukushima Daiichi NPP was undergoing seismic and tsunami safety assessments on all units, and associated construction work was in various stages across the units (planning, scheduled and under way). Some countermeasures, such as the water tightening of seawater pumps at Units 5 and 6, had been completed [32].

When sufficient knowledge and experience were present, such as in the case of the earthquake at the Kashiwazaki-Kariwa NPP in July 2007, TEPCO acted proactively to apply countermeasures for a similar type of event. The IAEA sent a fact finding mission to the plant in August 2007; in early 2008, the IAEA follow-up mission reported [127]:

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[15] The JSCE was recognized by all Japanese nuclear organizations as the authority specifying seismic and tsunami design criteria [106].

[16] The maximum tsunami height shown by the deposits was 4 m.

> "Since August 2007 there has been a very significant amount of high quality work performed in all areas that were considered during the follow-up mission including the establishment of required regulations and the participation of recognized institutions in Japan in the area of earthquake engineering and nuclear safety. NISA, JNES, TEPCO and a large number of specialized institutions and universities as well as experts have performed activities relating to the evaluation, regulation and the review aspects of the situation of the plant after the earthquake."

Less than four years after the Kashiwazaki-Kariwa event, seismically isolated buildings were constructed at the TECPO nuclear sites to serve as the ERC in the event of a nuclear accident. These buildings also included communications facilities and meeting rooms [99]. The seismically isolated building at the Fukushima Daiichi site was critical for supporting the event response on and after 11 March 2011. Based on the experience with the Kashiwazaki-Kariwa NPP, TEPCO also installed fire engines that were used as pumps for coolant injection during the accident [10]. These measures demonstrate the importance of the learning attitude.

Human and organizational factors

In the years prior to the Fukushima Daiichi accident, TEPCO completed several actions to improve its management and organizational processes. These were largely triggered by TEPCO's cover-up scandal of 2002 [54].

The large scale scandal was exposed in 2002, when NISA announced the results of an investigation of 29 cases of falsifications of inspection records produced for voluntary inspections at the Fukushima Daiichi NPP, Fukushima Daini NPP and Kashiwazaki-Kariwa NPP between the second half of the 1980s and 2001 [99]. This investigation led to NISA shutting down all of TEPCO's plants for inspection and revising its regulations on voluntary inspections [54].

By the end of the 1990s, TEPCO had already attempted to deal with the issue of falsification, following the discovery of the falsification of some NPP pipe welding data and spent fuel transport container data. In 1998, a Cultural Reform Review Committee was established to investigate the corporate culture and context in order to identify problem areas and review countermeasures and background analyses. This led to several initiatives aiming to reform the corporate culture, including a cultural change action plan with various initiatives to increase cross-hierarchical dialogue and "instill morality and manners of each and every employee" [54]. In spite of these efforts, the 2002 scandal was revealed — and, referring to the reforms prior to 2002, TEPCO admitted in hindsight that "these efforts at corporate cultural reform merely imposed idealism and did not lead to actual reform" [54].

After 2002, TEPCO stated that the company strived to increase transparency, to overcome its closed nature, and to develop an open corporate culture by proactively inviting third party perspectives [54]. The third party perspectives included facilitation of discussions on nuclear safety and quality assurance by external experts, reviews by domestic and foreign specialized organizations and cross-divisional personnel exchanges [10]. In addition, a series of nuclear renaissance activities were launched to make "recurrence prevention measures more widespread and [to become] an outstanding world-class nuclear operator" [54]. These included a leadership development exchange, which was successful according to its students, but seemed to have failed in producing a significant organizational impact owing to a lack of management support in utilizing the developed competences [54]. The latter was also the case with a peer activities programme introduced to improve work processes by learning from international peers. It was ineffective due to a lack of visible sponsorship or commitment and ineffective implementation processes. A final example includes TEPCO's construction of a management framework to monitor progress and achieve overall optimization through performance reviews. In hindsight, TEPCO reported that the ambitious programmes did yield some results, but that, in addition to the problems cited above, a recurring issue was the lack of management support and a lack of accountability for the implementation of the programmes [54].

In November 2009, TEPCO developed 7 Principles of Safety Culture after receiving comments on its safety culture from an external review [10]:

> "Principle 1: All personnel shall be aware of their involvements in nuclear safety
> "Principle 2: Leaders shall autonomously set examples of safety culture principles
> "Principle 3: Promote mutual trust among all concerned parties within or outside TEPCO
> "Principle 4: Make decisions by placing the first priority on nuclear safety
> "Principle 5: Be strongly aware of the inherent risks of nuclear power generation
> "Principle 6: Always maintain a questioning attitude
> "Principle 7: Learn systematically on a daily basis".

In addition to the principles, management communicated the approach of safety first and activities were conducted through safety caravans and case studies to help employees understand safety culture. Action plans, which included actions related to the results of recent safety culture assessments, were also developed and implemented to further assist in cultivating the desired safety culture. In addition, safety activities and events at the nuclear sites were evaluated against the seven principles [54]. Despite these activities, TEPCO concluded the following after the Fukushima Daiichi Accident:

> "[T]he deterioration of a safety culture throughout the entire organization, which underlay the occurrence of events which posed an issue, was limited to the assessment that 'there was no inclination toward a decline in the safety culture.' It did not result in an effort to deduce issues to be addressed based on how the whole operation proceeded. Based on this situation, past efforts to create a culture of safety were confined to so-called campaign-style content, and it is believed that measures did not go deep enough to the heart of measures which should have been addressed and evaluated" [54].

Other accident reports have reached similar conclusions to those of TEPCO, challenging the safety culture within TEPCO and also within the regulatory authority [32, 99]. A full, detailed assessment of TEPCO's safety culture is outside the scope of this t volume. Nevertheless, IAEA safety standards state that effective safety culture improvement activities need to address the deepest level of the culture, i.e. basic assumptions. TEPCO safety culture initiatives were not effective at challenging basic assumptions.

2.6.4. The impact of basic assumptions on the accident response

Those who directly responded in the early stages of the accident did so under extreme circumstances. The anxiety and stress associated with their actions was further exacerbated by the fact that the personnel often did not know the situation of their families nor the condition of their homes. They faced an extremely adverse work environment from both physical and psychological perspectives. Plant personnel worked often in darkness, without any indications of the status of the plant. They had to cope with working among debris and damaged structures and, as time progressed, their environment further degraded owing to the risks posed by hydrogen explosions and radiation exposure. Working under such conditions must have been fraught with fear, uncertainty, frustration and anxiety. However, personnel displayed a strong dedication to their tasks and worked for days without an uninterrupted break for sleep, remaining at the site and performing critical tasks that could affect their survival.

2.6.4.1. Preparedness decisions

As explained above, there was an underlying basic assumption among the nuclear industry stakeholders that the robustness of the technical design would maintain and protect the safety of plants against postulated events and that extreme external events were of low probability. This basic assumption led to an inability to anticipate or even imagine that an extreme external event, such as the earthquake and tsunami of 11 March 2011 causing a multi-unit accident, could take place. From a human and organizational perspective, this surprise outside the boundaries of the stakeholders' basic assumptions contributed to a lack of preparedness that ultimately resulted in personnel being exposed to extremely challenging working conditions in trying to respond to the events at the Fukushima Daiichi NPP. This seriously affected their ability to be effective in the mitigation of the accident.

At the time of the accident, as described in Section 2.6.7:

> "There [were] no legal regulations for the consideration of beyond the design basis, as Japanese plants [were] considered to be adequately safe as ensured by preventive measures" [53].

In addition, a report entitled, Station Blackout at Nuclear Power Plants [128], released in June 1993 by a Working Group on Station Blackout set up under the NSC's Committee on Analysis and Evaluation of Nuclear Accidents and Failures,

> "underscored the high reliability of Japan's external and emergency power sources, concluding that the probability of an SBO occurring was low and that nuclear power plants had sufficient defenses against SBO" [32].

There was a view within NISA that:

> "[W]ith regard to nuclear emergency preparedness, it was not necessary to anticipate an accident that would release enough radioactive material as to actually require protective actions, since rigorous nuclear safety regulations, including safety inspections and operation management, were in place in Japan" [32].

Even in light of the lessons learned from the 2007 Niigata-Chuetsu-Oki earthquake,

> "no integrated efforts had been made by the central government and municipal governments to establish disaster preparedness against complex disasters prior to the accident at the Fukushima Daiichi plant" [32].

The decision to not drastically change the existing emergency preparedness structure is reflected in the following quote by NISA:

> "It is reasonable for us to implement effective and efficient measures against complex disasters in line with the current nuclear emergency preparedness structure, since complex disasters are highly unlikely to occur" [32].

Finally, there was also a concern that developing countermeasures against a nuclear accident coupled with a complex disaster could lead people to think that a major natural disaster could initiate a nuclear emergency and that major modifications to regional disaster prevention plans would be required [99].

The scale of the earthquake and tsunami experienced on 11 March 2011 was not anticipated and the complexity of subsequent events was not considered. As a consequence of this basic assumption, insufficient infrastructure, protocols, procedures, equipment and training were in place to respond to such a situation.

> "The expansion of damage caused by this accident is attributed to the insufficient preparedness on the part of the central government and municipal governments in facing a complex disaster involving earthquakes and tsunamis occurring simultaneously with a nuclear disaster" [32].

The situation faced by operators and other key stakeholders on 11 March 2011 did not fit with what had been previously included in their basic assumptions about nuclear safety. As a result, personnel were forced to improvise and adapt to the evolving situation under severe time pressure, with recurrent earthquake aftershocks, the risks and fear of additional tsunami waves, sometimes in almost complete darkness, in areas scattered with debris and objects, (potentially) contaminated water, high radiation dose rates and a range of other physical safety hazards. The workers' ability to work effectively was also affected by the need to wear cumbersome full body protective clothing and full face masks and, in some instances, self-contained breathing apparatuses.

2.6.4.2. Infrastructure: Buildings, communications and human resources

The infrastructure in place to address emergencies was not adequate to respond to the event at the Fukushima Daiichi NPP. Buildings essential for the implementation of accident response actions were rendered inoperable as a consequence of the accident. For instance, the Local Nuclear Emergency Response Headquarters (NERHQ) at the Off-site Centre (OFC) was located and established without sufficient consideration of access road damage or communication system breakdown that could occur as a result of an earthquake. Consequently, the centre immediately lost its functionality from the beginning of the complex accident [102]. Some personnel were unable to reach the centre because of infrastructure damage. Those who did reach the centre despite the infrastructure damage experienced a loss of normal and backup power and could not remain because the centre had not been built with adequate protection against high radiation or a contaminated environment [106]. No plans were in place for dealing with the situation in the event that such centres and headquarters needed to be relocated [106].

As assessed in Section 2.4.2, the unavailability of proper means of communication and the huge volume of communications also challenged the event response. A situation that involved the loss of all power sources for a long period of time had not been anticipated [99] and, as a result, procedures to restore electronic telecommunication systems did not exist [99]. Such a loss of power rendered normal communications systems as well as some backup systems unavailable. Communications at the site were generally limited to two hotlines that could be used to communicate between the control rooms and the on-site ERC. These limitations not only hindered internal communications, but they also had an impact on external communications making it difficult to understand and mitigate the accident conditions which were changing continuously. For example, the loss of power led to plant data and information being unavailable. This contributed to misunderstandings regarding core cooling and challenged overall decision making. In addition, without the necessary information, government personnel found it hard to provide advice [99]. In some instances, direct calls to the site were made by senior government officials to obtain information or make recommendations.

Staffing issues as assessed in Section 2.4.2 also impacted those responding to the accident at the Fukushima Daiichi NPP. A staffing plan/strategy on how to respond to a multi-unit, high stress, long duration event did not exist [106], and the "corporate and station structure and staffing were not designed to support the number of units that may be affected by a common-cause event" [106]. As a result, operators worked for several days with minimal rest, and personnel in the on-site ERC carried out their functions continuously for a number of weeks. In addition, some contractors who worked on the site during normal conditions left it when radiological conditions worsened. In response, TEPCO personnel remaining at the site were trained, if possible, to perform the critical functions previously carried out by contractors [106].

The radiological conditions present during the event also posed a staffing challenge. As conditions exceeding normal radiologically controlled area (RCA) conditions expanded to include outdoor areas, there were not enough radiation control workers [10]. Insufficient staffing resources also affected TEPCO's communication capability relating to press conferences.

2.6.4.3. Protocols: Roles, responsibilities and decision making

As assessed in Section 2.4.3, clear roles and responsibilities for addressing simultaneous accidents on multiple units had not been established, or roles adopted during the accident differed from those previously assigned. The normal staffing at the site included a site superintendent, two unit superintendents, shift supervisors and assistant shift supervisors for Units 1–4 and Units 5 and 6, in addition to the associated control room crew for each two unit control room [106]. During the accident, the site superintendent, with support from the shift supervisors and general managers of the operations department, was responsible for managing the response to the events at all units. Decisions related to core cooling and venting containment were the responsibility of the site superintendent. However, in some instances, instructions were provided directly by the NSC or members of the Prime Minister's Office [32].

The roles and responsibilities of the headquarters of the Prime Minister, national and local governments, and nuclear plant operators were not clearly defined in the Nuclear Emergency Preparedness Act [88], nor were the defined roles and responsibilities always followed [32]. Several instances serve to illustrate this situation. For instance, even though its official role was to support the station, the ERC at TEPCO's headquarters was forced to exchange information with and answer to inquiries from both the central government and external parties [10]. This ad hoc structure is reported to have caused TEPCO to "lose the sense of responsibility and autonomy that is so essential for an operator in their on-site emergency response" [32].

The conflicting instructions issued by various decision makers also led to confusion. This was the case with the contradictory instructions received by the Site Superintendent regarding the startup of seawater injection into Unit 1 on 12 March 2011 [32]. In this case, METI Minister first ordered TEPCO to inject sea water, and operators at the Fukushima Daiichi site started the appropriate preparations. However, about one hour later, the Prime Minister refused to approve the seawater injection due to concerns over the issue of recriticality. Therefore, the Site Superintendent was asked to suspend the seawater injection while awaiting the Prime Minister's decision. However, based on his understanding of the situation, the Site Superintendent decided on his own to disregard this new instruction, and continued with the seawater injection. As the Prime Minister decided shortly afterwards to approve the injection, this confusion had no real consequences for the Unit 1 seawater injection. Nevertheless, given the time pressure and stress that the stakeholders were under, such misunderstandings and conflicting instructions did not help in ameliorating the situation [10, 32].[17]

A similar situation involving the Prime Minister's Office and the Site Superintendent developed over the decision to vent the primary containment vessel (PCV) of the Unit 1 reactor. The chairman of the NSC notified the Site Superintendent that depressurization and injection should be prioritized over venting; however, at that time, field work was being performed to prepare for venting [10]. After receiving this notification, the operators at the site and officials at TEPCO's ERC in Tokyo deliberated over the request and reconfirmed the policy to prepare for venting before depressurization [10].

Such situations were cited as leading to confusion with respect to the decision making authority and the review and the decision itself, especially since it appeared to some workers that "external opinions were given priority over judgment of the director of the ERC at the power station (Site

[17] While Site Superintendent Yoshida at the Fukushima Daiichi NPP did not follow the instruction to stop the seawater injection, TEPCO Headquarters meant to do so and reported that the company had instructed Mr. Yoshida to stop the injection after conferring with the Prime Minister's Office. The TEPCO Fukushima Nuclear Accident Analysis Report points to several factors as to why TEPCO Headquarters made this decision, highlighting the company's concern that

> "[F]uture coordination with necessary government organizations would be impeded even further if field work proceeded without the approval of the Prime Minister, as the PM is the chief of the Nuclear Disaster Response Headquarters" [10].

Superintendent)” [10]. In general, the lack of clarity about roles and responsibilities increased stress and confusion for all parties, worsening the situation from a human and organizational point of view.

2.6.4.4. Procedures and guidance

Procedures and guidance were also not adequate to address the complex accident conditions experienced at the Fukushima Daiichi NPP. In the area of emergency preparedness and response, “little thought had been given in the Emergency Preparedness Guide to a situation in which commerce and logistics came to a standstill” [32]. The NSC’s Nuclear Emergency Preparedness Guide had been established in 1980 and was subsequently revised; a larger revision to comply with international standards was initiated in 2006 (see Technical Volume 3) [32].All of these reviews did not lead to the guide being fully in line with IAEA safety standards [80]. The underlying reason for this has been the basic assumption that “a Chernobyl type nuclear accident could not occur in Japan” [32]. In addition, the Prefectural Regional Disaster Prevention Plan did not anticipate a nuclear accident of such a scale. Consequently, a 20 km radius evacuation zone was not designated prior to the accident, nor were measures specified to deal with complex accident situations associated with a natural hazard [32]. Other regional disaster plans were developed for an accident similar to those previously experienced, leaving hospitals to develop their own evacuation plans and to evacuate patients on their own [32]. The resulting situation led to challenges in evacuating people from their homes and patients from hospitals.

With respect to operating the plant, the AM measures used at the time of the accident “assumed normal operation of [the] adjoining reactor unit and did not consider the possibility of multiple reactor units simultaneously being affected by external events such as natural hazards. Therefore, TEPCO did not plan any action that can be taken when a reactor unit has lost all power and is not able to receive power from the adjoining reactor unit” [99]. In addition, procedures to direct how to “manually perform critical actions in the field concurrent with a loss of all AC and DC power (including loss of compressed air)” were not in place [106]. Furthermore, procedures to direct how to restore measuring instrument functionality and power, and how to vent and depressurize the reactor under such conditions did not exist [32].

2.6.4.5. Equipment

Following the events at the Fukushima Daiichi NPP, requests were made by TEPCO for a variety of equipment and material, including “fire trucks, generator trucks, hoses and cables, fuel, batteries, pumps, motors, reactor cooling water, radioactive protection gear, consumables and other supplies” [32]. While some equipment was available at the Fukushima Daiichi NPP, in many cases the quantity or capacity was not enough to address the situation at hand. For example, the station had the minimum number of charcoal masks required, but “a sufficient number was not available for all plant workers” [32]. Not only did the station fail to have the equipment readily available, but TEPCO was also “not prepared to immediately secure and implement a number of the necessary equipment and materials” [32].

2.6.4.6. Training and drills

While comprehensive nuclear emergency preparedness drills were conducted annually by order of the national government, they did not simulate the event or conditions experienced on 11 March 2011, nor did they emulate severe accidents or complex disasters. A staff member of the Fukushima Prefecture Hospital Association was quoted as saying that:

> “...neither the earthquake evacuation drills nor the nuclear accident drills were implemented based upon a prior anticipation of having to evacuate all hospitalized patients. Furthermore, they were based on the assumption that the lifeline would be functioning” [32].

In addition, when responding to the proposal to conduct a drill with a scenario that included "the simultaneous occurrence of a nuclear disaster and an earthquake, with no direct cause-and-effect relationship between them," NISA commented that:

> "[T]he national government could not support the drills, since the scenario suggested that even limited damage to evacuation routes and facilities by an earthquake could result in problems at a nuclear reactor, and drills conducted based on this ambiguous scenario could worry local residents unnecessarily" [32].

Emergency drills conducted at the Fukushima Daiichi NPP also did not simulate an event of the complexity and severity of that experienced on 11 March 2011. For example, an emergency drill conducted in February 2011 involved a scenario in which external power was lost to a unit following an earthquake in conjunction with the failure of an EDG to start, resulting in a loss of AC power. However, the drill assumed that the EDG would be recovered after a period of time [102]. In addition, no drills were conducted that included events or abnormalities at multiple units simultaneously, nor were drills that simulated a loss and recovery of information sources or communication challenges [118].

Training to prepare personnel to respond in the event of a nuclear accident was limited. Computer based AM training was conducted, but it lacked the detail necessary to develop the skills and knowledge needed to respond to an accident [32]. Additional training following the computer based simulation was provided at three year intervals. In addition, the simulator used for operator training did not include the isolation condenser, nor did personnel have experience operating the system. Training on the system was based primarily on classroom and on the job-site training [118].

2.6.5. Human and organizational assessment

To acknowledge that what happened on 11 March 2011 was a surprise outside the boundaries of the basic assumption of stakeholders in Japan as well as the international nuclear community, has important implications for what can be learned to enhance the safety of NPPs throughout the world. The difference between surprises within or outside the boundaries of the basic assumption has previously been explained. Although the stakeholders involved were aware of different safety issues related to the accident, it is a complex task to put all these issues together and to realize that such a complex accident may indeed occur. The full complexity of the accident needs to be considered comprehensively in order for the nuclear safety community to learn crucial lessons from the event. The accident would only reinforce the familiar knowledge about nuclear safety, i.e. the known knowns and known unknowns. It would open up the possibility of being again unprepared and exposed to unexpected situations outside the boundaries of the basic assumption. Lessons learned from surprises inside the boundaries of the basic assumption (known knowns and known unknowns) are often limited in scope and involve local, often technology based, fixes (e.g. increased redundancy of components or design changes) [118]. On the other hand, a surprise which disrupts the basic assumptions offers an opportunity for fundamental learning through changing the basic assumptions which are driving behaviours. The very human tendency of avoiding the psychological discomfort emerging from questioning one's basic assumption and from acknowledging the existence of unknown unknowns means that the opportunity to learn from surprises outside the boundaries of the basic assumption could easily be missed. This would often result in failing to see the systemic explanations of why something happened.

To effectively learn from a surprise outside the boundaries of the basic assumptions, it is necessary to overcome three main obstacles which hinder a comprehensive and thorough learning process.

The first challenge is to manage to treat the event as a surprise outside the boundaries of the basic assumptions. This is often avoided by treating it as a situational surprise or rejecting (consciously or unconsciously) any learning opportunity through the mechanism of distancing through differencing. It is certainly necessary to identify attitudes and practices that are unique to the main stakeholders in Japan, but one also needs to attend to general processes, system relations, etc. that may occur at other places.

The second challenge is to synthetize the new lessons and understanding with the pre-existing basic assumption. Being a surprise, outside the boundaries of the basic assumption, incorporating the lessons may require a considerable revision of existing assumptions. In a complex system, there may be processes that appear to fix standard problems, but often these same processes hide more fundamental weaknesses. Integrating lessons requires a willingness to examine systemic relations that may risk disrupting standard practices.

The third and most difficult challenge to be overcome is at the institutional level. Organizational systems become more permanent as they are supported by policies, actions and attitudes of organizational personnel and so forth. A surprise outside the boundaries of the basic assumption may require changes involving much that is deeply entrenched inside the organization. It is thus necessary for the sociotechnical system as a whole to identify and implement the new learning, new practices, a new language and a new way of seeing and interpreting things for it to be institutionalized and brought into the basic assumption within the organizations' safety culture [129].

2.6.6. The implications for the nuclear community

Having expanded on the human and organizational aspects of the accident in the first part of this section, the implications for the nuclear community and society all over the world and the key messages are described below.

2.6.6.1. Treating the accident as a surprise outside the boundaries of the basic assumption

The analysis has shown that the main stakeholders involved in the Fukushima Daiichi accident assumed that the robustness of a nuclear plant's technical design would maintain and protect the safety of plants. The belief in the robustness of the nuclear installations appeared to be so strong that the stakeholders did not recognize the existence of the unknown. Thus, the occurrence of a very high tsunami having a deleterious impact on the NPP constituted a surprise outside the boundaries of the basic assumption that dramatically disrupted and changed the existing reality. For the same reason these stakeholders were not sufficiently prepared to technically and organizationally deal with a long duration SBO or with a multi-unit, long duration accident. This may also be true for other countries and warrants continuing analysis to identify the boundaries of the basic assumption.

2.6.6.2. Difficulty in acknowledging complexity

One reason why the unknown unknowns can be fundamental in nature and their sudden discovery can have such a disruptive impact on one's basic assumption is the fact that it is very difficult to recognize and understand the full complexity of a (sociotechnical) system such as a nuclear power programme and its assessment and regulation. This encompasses many different stakeholders with different roles, tasks and interests interacting with each other regarding the operation and safety of a complex technology. As the analysis shows, not only TEPCO but also the other stakeholders failed to comprehend the full extent the complexity. This led to reduced risk awareness and to a simplified consideration of the sources of risks, including, for instance, external events.

Comprehending and accounting for the complexity was made even more difficult for TEPCO and other stakeholders by the fact that, even though in hindsight it might be perceived as such by some, the degree of safety of an NPP is not simply quantifiable or always easily recognizable. As has been shown, TEPCO as well as the regulatory bodies had made substantial efforts to improve the safety of the nuclear installations over the years, e.g. after events such as the falsification scandal in 2002 or the Niigata-Chuetsu-Oki earthquake in 2007 that affected the Kashiwazaki-Kariwa NPP. These improvements might have contributed to upholding and even reinforcing the belief in the NPPs being sufficiently safe. This suggests that it is necessary to allow for the fact that vulnerabilities can develop in the system over years and remain concealed by the organizations' (long) experience of success as well as by their belief that they have risks under control. "This…is how a successful system produces failure as a normal, systematic by-product of its creation of success" [110].

2.6.6.3. Acknowledging the possibility of the unexpected

This reasoning leads to the necessary acknowledgement of the fact that it will never be possible to consider, anticipate and prepare for all possible contingencies. The possibility of unexpected events — that is, of events no one has thought of before, as they lie outside the boundaries of one's basic assumption and which, as a consequence, an organization did not specifically prepare for — is always present. "Things that never happened before, happen all the time" [130]. It therefore appears necessary to question whether the current practice of the nuclear industry and the engineering culture, which aims to anticipate all possible scenarios and to develop in advance targeted (technical and administrative) solutions, is enough to account for the ever present threat, however improbable it might be, of something unexpected happening. Nuclear safety would be further improved by acknowledging the possibility of the unexpected.

The nuclear industry and its regulators therefore need to become more resilient, i.e. better able to adapt the industry's functioning under varying conditions, both expected and unexpected [131] and to mobilize additional resources when it approaches its margin of manoeuvre [132]. In other words, the nuclear organizations and the industry need to be prepared to be unprepared [114]. Such preparation should explicitly take into account the unexpected and therefore also needs to focus on the ability of the organization and individuals to adapt and respond to unexpected and unknown events, including events implying extreme physical and psychological conditions.

2.6.6.4. Preparation for the unexpected

Organizations need to be prepared for the unexpected, i.e. to be "prepared to be unprepared". This includes providing appropriate training to all individuals on how to respond to unexpected events and to provide adequate equipment and procedures. However, this preparation and training does not only mean to prepare technically and organizationally for all scenarios one can think of. It also means to develop generic competencies and resources within the organization that help the personnel to quickly and flexibly adapt to new situations, to improvise and develop new solutions for unknown problems; in other words: to be resilient in unexpected situations. This kind of human and organizational resilience capabilities needs to developed under normal operation and well in advance.

2.6.6.5. Taking a systemic approach

The analysis of the accident highlights the need for a systemic approach to be taken both in the analysis itself and in learning how to treat nuclear safety in the future. To understand the accident from a systemic perspective, one cannot look at each element of the nuclear system singularly. This accident clearly cannot be attributed to one licensee, one management group, one NPP or one regulator alone, but must rather be considered holistically by looking at and understanding the systemic interactions between all the stakeholders involved prior and during the accident. As explained above, the stakeholders (for instance, the licensees, the regulators, the government and the public) constructed and maintained together and over many decades a perception of risks and nuclear safety that turned out not to have been adequate.

A systemic approach to safety needs to be implemented by all participants and in all types of activities within the nuclear power programme and throughout the entire life cycle of nuclear installations, including review services offered by international organizations. As was shown by the analysis, in Japan, nuclear installations, TEPCO and NISA primarily focused on the technical aspects of nuclear safety. A systemic approach to safety implies that all stakeholders, besides the technical factors, take comprehensively into account the human and organizational factors, including safety culture, to build resilient capabilities.

2.6.6.6. Stakeholders' roles and responsibilities

In line with the systemic approach, each stakeholder plays an important role in achieving the level of safety necessary in nuclear power production. As IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], states: "The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation". Principle 1 emphasizes: "The prime responsibility for safety must rest with the person or organization responsible for facilities and activities that give rise to radiation risks" [1]. Principle 2 states that: "An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained" [1]. This establishes prerequisites necessary for the whole system to reach the safety objectives.

Thus, the licensee always bears the responsibility for the safety of its nuclear installations, as stated by SF-1 [1]. It needs to take ownership of safety and cannot delegate its responsibility to other stakeholders such as the regulatory body.

The regulatory body needs to take ownership of the protection of people and the environment through recognizing its role and its impact on the safety and safety culture of the licensees and of their willingness and ability to take ownership of safety. This entails not only ensuring technical compliance with regulations, but also comprehensively addressing human, organizational and cultural factors [123]. The task of the regulatory body is highly demanding and entails the duty of continuously challenging and questioning the basic assumption held by the industry it regulates and by itself. This implies high demands on the regulator's self-reflecting capability to put its own role and its impact on nuclear safety and on the collective perception of nuclear safety under constant self-scrutiny. As the analysis has shown, the Japanese regulatory bodies, NISA and NSC, seemed to have insufficient capabilities to apply this questioning role owing to a lack of authority, resources and de jure independence.

As for the public, it is important to maintain a transparent dialogue between the public and the other stakeholders that supports this questioning and challenging attitude on an ongoing basis. A constructive dialogue between the public and the nuclear industry and regulators needs to be built on mutual trust and respect for the different roles the stakeholders play in the national nuclear infrastructure. As the analysis has demonstrated, the Japanese licensees and regulatory bodies feared the public reaction to proposed improvements in the regulatory framework, the nuclear installations or the preparation for severe accidents. This led to a lack of transparency toward the public, which in turn made a constructive dialogue with an informed public impossible. In order to allow an open and constructive dialogue among the public and other stakeholders, the public needs to be openly and competently informed not only about the advantages of using nuclear power, but also about the risks that this technology entails as well as how safety is maintained. The regulatory bodies have a particularly important role to play in informing the public in a competent, understandable and up to date way as a reliable and trustworthy body.

2.6.7. Summary

The accident at the Fukushima Daiichi NPP came as a surprise outside the boundaries of the basic assumption of the operators, the utility, the regulator, the government and the people of Japan. It had been accepted that the nuclear power plants in Japan were safe and able to withstand any external events.

When nuclear power began to be developed in Japan, an approach to nuclear safety was developed within the main nuclear industry stakeholders. Over time, a basic assumption emerged that the robustness of the nuclear plant's technical design and existing measures would maintain and protect the safety of plants against postulated risks. This basic assumption was developed, maintained and mutually reinforced over time among the main stakeholders, namely the government departments, regulatory body, the licensee and the public. As a result, the risks faced were not sufficiently assessed, nor the defences against them strengthened — particularly regarding the treatment of low probability external events, accident management procedure improvements, and preparations for a complex accident that could affect all the reactor units simultaneously. The basic assumption prevented the stakeholders from effectively preparing for and mitigating the consequences of the tsunami caused by the Great East Japan earthquake of 2011.

This was manifested by the decisions and actions of all main stakeholders, i.e. the governmental ministries, the regulatory body, the licensee and the public. The regulatory body at the time of the accident was not able to exert the necessary authority over the industry it was meant to oversee. It worked generally in a reactive manner, focusing on short term activities, and did not address more fundamental and long term issues, such as the consideration and implementation of the IAEA safety principles and developments. In its relation to the public, the regulator was, for example, reluctant to update regulations or to allow complex emergency drills out of concern that the public might get the impression that nuclear power plants were not safe.

The operators were so confident that the superior technical features of their plants would make nuclear accidents highly unlikely that they did not prepare sufficiently to mitigate the results of such accidents. TEPCO's AM measures addressed internal events only; they did not prepare for a tsunami, as it was thought that any NPP could respond to natural disasters including tsunami within design. In addition, the management measures assumed that, in the event of an accident at one reactor, adjoining reactor units would continue to operate normally, and they did not take into account an event that could simultaneously affect all the reactor units. When the international community began to advance in the area of severe accident management (SAM), TEPCO did not adopt some of these developments. As a result, TEPCO had not planned how to address a situation in which several units lost all power and were unable to receive power from an adjacent unit.

One part of a healthy safety culture of organizations is the capability to challenge or re-examine the basic assumption for safety. As all stakeholders involved in the nuclear industry will develop their own basic assumptions with regard to nuclear safety, it is of universal importance to be mindful of such assumptions and work proactively to understand their impact on nuclear safety.

The human and organizational factor analysis also concludes that it is important to take a systemic approach to nuclear safety by embracing the complexity of the full range of interactions between human, technical and organizational factors. One of the underlying causes of the Fukushima Daiichi accident was the inability of the stakeholders to comprehend and work with this complexity in order to proactively prevent the accident. Other parts of this volume show how the stakeholders involved in the accident had knowledge of several of the flaws in the system prior to the accident. The human and organizational factor analysis explains how these stakeholders, despite their knowledge, were unable to proactively understand and imagine how these flaws could come together to produce an accident in the way they did following the earthquake on 11 March 2011. Through the analysis, it has been shown how the stakeholders involved were mutually influencing each other. Thus, the scope of the systemic approach needs to encompass the interactions between all stakeholders involved in nuclear safety. In the case of the accident at the Fukushima Daiichi NPP, the unchallenged reinforcing interactions among the stakeholders strengthened the basic assumption that the robustness of the nuclear plant's technical design and existing measures would maintain and protect the safety of plants against postulated risks and prevented the necessary countermeasures from being taken in a timely manner.

Two main observations have been made in this section:

First, the analysis has shown that, over time, the stakeholders of the Japanese nuclear industry developed a shared assumption that the robustness of the nuclear plants' technical design and existing measures would maintain and protect the safety of plants against postulated risks. This shared assumption led the stakeholders to believe that a nuclear accident would not happen and thus thwarted their ability to anticipate, prevent and mitigate the consequences of the earthquake triggering the Fukushima Daiichi Accident.

Second, while the stakeholders involved in the accident at the Fukushima Daiichi NPP were aware of the possibility of the single safety issues related to the accident in advance (e.g. SBO, extreme tsunami heights and multi-unit accidents), they were not able to anticipate, prevent or successfully mitigate the outcome of the complex and dynamic combination of these issues within the sociotechnical system.

2.6.8. Observations and lessons

- **The accident at the Fukushima Daiichi NPP was a surprise outside the boundaries of the basic assumption of the key stakeholders, meaning the stakeholders had not been able to imagine that such an accident could occur. From this, the lesson learned for the international nuclear community is that the possibility of the unexpected needs to be integrated into the existing worldwide approach to nuclear safety.**
  When unexpected situations occur outside the boundaries of the basic assumption of nuclear safety, people and organizations need to be prepared to be unprepared. Resilience competencies and resources have to be developed well in advance within organizations to help personnel to quickly and flexibly adapt to new situations, to develop new solutions for blind spots — in other words: to be resilient in unexpected situations.
- **Individuals and organizations need to consciously and continuously question their own basic assumption and their implications on actions that impact nuclear safety.**
  This is part of sustainable safety culture improvement; the basic assumption about safety is recognized as fundamentally directing safety culture. To enable this, individuals and organizations need to systematically question the nature, boundaries and potential challenges of one's own assumptions of nuclear safety, particularly beyond technical safety matters. Reflection and dialogue are needed within an organization in order to become aware of possible blind spots in basic assumptions. This can be achieved through periodic safety culture assessments (both independent and self-assessment reviews) based on the IAEA's approach to the assessment of safety culture and other IAEA peer reviews, such as the IAEA's IRRS and OSART missions. In addition, the utilization of a diversity of expertise and experience is important in order to avoid undue simplifications in interpretations and to better recognize the full picture.
- **Nuclear organizations need to critically review their approaches to emergency drills and exercises to ensure that they take due account of harsh complex conditions and unexpected situations.**
  Harsh complex conditions include both physical and psychological aspects. Unexpected situations that may be considered for implementation in the framework of emergency exercises include a shaking ground (earthquake), large scale fires and flooding, a dark work environment, unfamiliar sounds and the loss of vital information. In addition, resilience competencies need to be developed

well in advance to prepare humans and organizations for quick and flexible adaptation to unexpected situations (‘unknown unknowns’).

— **The results of research on complex sociotechnical systems for safety need to be taken into account.**
In order to proactively deal with the complexity of nuclear operations, licensees, regulators, designers, peer reviewers and other relevant stakeholders need to consider the research on sociotechnical systems when designing nuclear technology, operating and overseeing nuclear installations and developing industry standards. The development of human and organizational resilience capabilities should also be based upon state of the art research on complex sociotechnical systems.

— **A systemic approach to safety needs to be taken in event and accident analysis, considering all stakeholders and their interactions over time.**
Stakeholders include, among others, licensees, regulators, political leaders and the public. The systemic approach includes human, technological and organizational considerations and is necessary to understand how the components of the overall (sociotechnical) system functioned, interacted and succeeded in everyday situations and over the decades before the accident, as well as during accident response. To accomplish this, a diversity of expertise is needed to cover the human, technical and organizational factors.

— **The regulatory body needs to acknowledge its role within the national nuclear system and the potential for its impact on the nuclear industry’s safety culture.**
The regulatory body has the challenging role of questioning the nuclear industry’s approach to safety. Therefore, the regulatory body needs a critical, profound self-reflecting and questioning ability. This may include institutionalizing an ongoing dialogue within the organization and with other stakeholders on the regulatory body’s safety culture and its impact on nuclear safety.

— **Licensees, regulators and governments need to conduct a transparent and informed dialogue with the public on an ongoing basis.**
This may include explanation of the risks that the use of nuclear technology for energy production entails.

2.7. APPLICATION OF OPERATING EXPERIENCE TO IMPROVE PLANT DESIGN AND OPERATION

Applying operating experience is a key contributor to maintaining continuous improvement in nuclear safety; it supports the defence in-depth philosophy. One of the key lessons learned from the severe accident at the Three Mile Island NPP in 1979 is that the accident could have been prevented through the effective application of operating experience. Furthermore, as noted by the International Nuclear Safety Group in its publication Improving the International System for Operating Experience Feedback [133]:

> “It is widely observed in all fields of human activity that serious accidents are nearly always preceded by less serious precursor events. If lessons can be learned from the precursors and these lessons put into practice, the probability of a serious accident occurring can be significantly reduced.”

The global nuclear industry has recognized the vital importance of operating experience in improving operational performance and preventing nuclear accidents, and has made significant strides in improving its operating experience programmes. The Convention on Nuclear Safety, which entered into force in October 1996, reinforces the international commitment to the application of operating experience to improve nuclear safety. Article 19 of the convention [134], concerning operation, states:

“Each Contracting Party shall take the appropriate steps to ensure that:

....

(vi) incidents significant to safety are reported in a timely manner by the holder of the relevant licence to the regulatory body;

(vii) programmes to collect and analyse operating experience are established, the results obtained and the conclusions drawn are acted upon and that existing mechanisms are used to share important experience with international bodies and with other operating organizations and regulatory bodies”.

As part of their obligations under the convention, contracting parties are required to report on their compliance with the articles of the convention approximately every three years.

The term ‘operating experience’ as used in this section includes:

- Industry events;
- Research findings;
- New analytical methods;
- Learning acquired through periodic safety reviews.

2.7.1. Significant operating experience events relevant to the accident at the Fukushima Daiichi NPP

There have been three well documented, beyond design basis natural events (i.e. combinations of earthquakes, tsunamis, high winds and floods) at NPPs that provide important operating experience that could have impacted the consequences of the accident at the Fukushima Daiichi NPP:

- In December 1999, a storm surge caused flooding at two reactors at Le Blayais NPP, in France.
- The Indian Ocean tsunami of 26 December 2004 flooded seawater pumps at the Madras Atomic Power Station, in India.
- On 16 July 2007, an earthquake exceeded the design basis of TEPCO’s Kashiwazaki-Kariwa NPP, in Niigata Prefecture.

The IAEA and the OECD/NEA analyse events reported through the Incident Reporting System (IRS) and jointly issue a report every three years highlighting significant trends. The three events listed above, as well as others, were included in reports produced between 1999 and 2008. These triennial reports represent a synthesized summary of the most important learning points from operating experience and are openly available to all members of the public and provide focus areas for utility self-assessment as well as regulatory review. They also provide an opportunity for other external reviews, such as the IAEA’s OSART missions, whose scope includes a review of operating experience. The relevant sections from these respective reports are reproduced below:

“Experience with system losses due to severe weather conditions

....

“Severe weather conditions resulted in the formation of waves moving up the river along which the plant is located, causing a partial flooding of two reactor units. External power supplies were lost momentarily due to the loss of the grid. Mechanical and electrical penetrations were damaged. Rooms housing safety related components were flooded causing their unavailability.

This included portions or all of the Essential Service Water System, the Safety Injection System, and the Containment Spray System.

“Moreover, due to the consequences of the storm, the site was momentarily isolated, impairing emergency help from outside. The plants were placed in a safe shutdown state.

“Safety significance

“Bringing and maintaining the reactor in a safe shutdown state to cope with the consequences of the flood requires the availability of the corresponding equipment. Some of the equipment was made unavailable by flooding. A failure of a penetration protecting one train of the Essential Service Water System, similar to the one having affected the redundant train, would have led to a loss of the ultimate heat sink. This scenario is addressed by the safety related studies, but the flooding conditions of the rooms would have complicated the management of the situation” [135].

“Experience with external hazards

“In the period 2005–2008 three events caused by earthquakes were reported. In the most severe event, a strong earthquake occurred in the vicinity of a multi-unit plant site. The actual damages from the earthquake did not affect the safe shutdown and cooling of the reactor cores. But some minor failures led to an on-site fire and the release of a small amount of radioactivity. No protective measures for the public had to be taken. The technical analysis of the earthquake impact revealed that the design acceleration values had been exceeded. The event resulted in a complete recalculation of the acceleration values to be expected due to earthquakes.

“Another event occurred during the disastrous tsunami in the Indian Ocean. Thousands of kilometres from the site of the tsunami’s origin, a nuclear power plant site was flooded by the unexpected high waves.

“One nuclear power plant reported an all time high seawater level. The level just missed the declaration setpoint for a site emergency. Due the conservative design of the plant, no actual flooding occurred.

“A plant at a riverside that is known for water level fluctuations and the forming of sand banks experienced a massive silting of the raw water intake channel. The plant decided to remove the sand from the sand trap installed in the intake water tunnel. Because there was a risk that the sand heap might become unstable during the removal operation and thus could block the entire intake, the plant operating organization introduced a specific operating instruction to cope with the situation in case of sand heap collapse. Contributing to this event was the fact that the sand had not been industrially extracted from the river for some years and the plant had not re-evaluated the potential consequences.

“Safety significance

“External hazards may lead to multiple system failures in a plant, either by flooding or mechanical loads, e.g. due to earthquakes. Because these hazards cannot be prevented, the design must take into account the challenges they pose. The vital functions of the plant — criticality control, removal of the residual heat and enclosure of the radioactivity — have to be ensured through all hazards. Special attention must be paid to environmental changes due to global warming, as well as to industrial activities in the vicinity of a plant. Tide heights, flooding by the sea as well as rivers, and draughts have to be taken into account to a greater

> extent than 30 years ago. Environmental changes may require backfitting measures at the plants" [136].

A more detailed description of these important events is provided in Annex III of this Technical Volume.

The review of the National Reports to the Convention on Nuclear Safety noted similarities in the actions taken in the responses by France to the Le Blayais event; by Japan to Kashiwazaki-Kariwa and by India to the Madras event [92, 137, 138]. These three events demonstrate that operators and regulators appear to put more emphasis on domestic operating experience than on operating experience from other countries [138]. The review determined that the application of domestic operating experience led to safety improvements, whereas the application of international operating experience was not as comprehensive.

Paragraph 1.2 of IAEA Safety Standard Series No. SSG-25, Periodic Safety Review for Nuclear Power Plants [139], specifies that:

> "Routine reviews of nuclear power plant operation (including reviews of modifications to hardware and procedures, significant events, operating experience, plant management and personnel competence) and special reviews following major events of safety significance are the primary means of ensuring safety."

In addition, para. 62 of INSAG-23, Improving the International System for Operating Experience Feedback [133], states:

> "Reporting on events and other safety relevant observations needs to be coupled with programmes that transform the lessons learned into risk reducing measures. The recipients of such reports, both operators and regulators, should close the feedback loop by providing information on their actions in response to a report. This information from recipients should be collected, summarized and made available to all potential users."

In 2011, the IAEA updated the IRS process to allow operators and regulators to close the feedback loop, and encouraged Member States to provide information on their actions in response to a report. In 2013, the IAEA received a total of only 15 items of feedback from only six Member States in response to approximately 80 operating experience reports. In the absence of more comprehensive reporting on the actions taken in response to operating experience, it is difficult to draw conclusions on the appropriateness of these actions.

2.7.2. Effectiveness of application of TEPCO's operating experience programme

2.7.2.1. Application of learning from industry events

The three major events noted in Section 2.7.1 were screened by TEPCO through its operating experience programme for applicability to its plants. In the aftermath of the earthquake at the Kashiwazaki-Kariwa NPP, TEPCO took extensive action at all of its plants to address lessons learned, demonstrating that its process for screening and acting on operating experience can be effective if appropriately applied. At each of its stations, it constructed seismically isolated emergency buildings, improved the firefighting systems, and installed modifications to enable the use of portable pumpers as an alternate source of water injection into the reactor. It also enhanced its site evacuation plans. All these improvements proved to be of significant benefit in the response efforts following the tsunami at the Fukushima Daiichi NPP.

Following the river flooding and high wind events at the Le Blayais NPP, the operator, Electricité de France (EDF), took extensive action to improve defences at all its plants [140], such as:

- Identifying all phenomena which can result in a flood at any of its 19 sites, including those located near an ocean or an estuary, and those located near rivers;
- Completing a reassessment of flood hazards and their associated impacts at each site;
- Identifying the equipment to be protected;
- Reviewing the existing protective measures, including structures, devices, procedures and organizational factors to identify gaps;
- Completing modifications and improvements to resolve gaps.

Actions taken by EDF included raising, extending, or reinforcing dykes and sea walls and improving resistance to water ingress by installing watertight doors, and sealing openings and penetrations. EDF also identified off-site factors that could impact the site response, including loss of off-site power, site inaccessibility and communications breakdowns, and developed mitigation and coping strategies [140].

Had TEPCO taken similar measures to those implemented by EDF after the Le Blayais flood, this may have impacted the consequences of the earthquake and tsunami at the Fukushima Daiichi NPP.

During the Indian Ocean Tsunami of December 2004, seawater pumps were flooded at the Madras Atomic Power Station, resulting in a potentially significant challenge to fuel cooling that was successfully mitigated through operator action. There was no plant damage other than to seawater pumps at lower elevations.

This event was classified as Level 0, the lowest level, on the International Nuclear and Radiological Event Scale (INES). However, the potential for more serious consequences was recognized by both the operator and the regulator, which resulted in the implementation of a number of improvements, including:

- Installation of an additional diesel generator at 2 m above grade level;
- Relocation of the uninterrupted power supply system to a higher elevation;
- Installation of a diesel driven air compressor at a higher level;
- Installation of a dedicated pump for transfer of de-aerator water to steam generators (emergency boiler feed pumps);
- Installation of two diesel driven fire pumps located 2 m above grade level;
- Construction of a tsunami protection wall;
- Installation of a tsunami warning system [138].

The case of the tsunami and flood at the Madras Atomic Power Station was deemed by TEPCO not to require further evaluation on the basis that it did not result in significant consequences (i.e. it was classified as a Level 0 on INES. The potential for more serious consequences was not recognized by TEPCO during the screening process.[18]

Following the accident at the Kashiwazaki-Kariwa NPP, TEPCO took extensive action in response to the operating experience, demonstrating that its process for screening and acting on operating experience could be effective if appropriately applied. However, in the other two cases, a narrow

[18] IAEA Safety Standards Series No. NS-G-2.11, A System for the Feedback of Experience from Events in Nuclear Installations [141] specifies in the footnote to para. 2.7 that "…the distinction should be maintained between a low level event (with no consequences) as contrasted with a reportable condition, which may have a high significance for risk even though it has no immediate consequences."

perspective led to missed opportunities to learn from valuable operating experience. As discussed in TEPCO's root cause report [54], the following aspects contributed to this narrow perspective:

— Results of analysis of internal events using PSA methods contributed to overconfidence in the safety of the NPPs. Had the PSA analysis been extended to include common cause and/or external events, it would have exposed a higher probability of severe core damage and yielded insight into the potential vulnerability to a loss of all sources of power.
— There was apprehension that publicly acknowledging a potential vulnerability to an event that exceeded the design basis for the plant might lead to the inference that the plant was unsafe to operate and result in the prolonged shutdown of the plant and/or costly backfits.
— There existed an inherent organizational bias to avoid actions requiring a significant expenditure of resources, particularly if such actions could not be directly linked to improved production.
— The screening of operating experience, together with the associated evaluations, was performed by departments external, and not accountable, to those directly responsible for the operation of the plant.

TEPCO's senior management was not closely involved in the initial screening process for operating experience and did not have the opportunity to provide a broader perspective.

These organizational factors, taken collectively, suggest that, within TEPCO's management system, nuclear safety was not given priority over all other demands, which resulted in weaknesses in the effective implementation of the operating experience programme.[19]

2.7.2.2. Application of learning from new analytical methods

The following provides a brief overview of TEPCO's assessment of the tsunami hazard at the Fukushima Daiichi NPP [10]. A more detailed discussion is provided in Section 2.1.

The initial design tsunami height for the Fukushima Daiichi NPP was selected as 3.1 m, based on historical information. Specifically, the highest recorded tidal level at Onahama Port from the Chile earthquake and tsunami of 1960 was defined as the design condition. This approach of using historical information rather than numerical simulation was consistent with the accepted methodology at the time.

Between the time of its initial design and the accident of March 2011, as new knowledge and analytical methods became available, TEPCO re-evaluated and revised upwards its estimate of the design tsunami height several times. For example, the design tsunami height for the Fukushima Daiichi NPP was revised from 3.1 m in 1966, to 3.5 m in 1994, then to 5.7 m in 2002, and finally to 6.1 m in 2009.

This practice was not unique to TEPCO; other NPPs owned and operated by other companies in Japan also revised their design tsunami heights as new knowledge and analytical methods became available. For example, in the case of the Onagawa NPP, operated by Tohoku Electric Power Company and located in Miyagi Prefecture, the design tsunami height was revised upwards from an initial value of ~3 m in 1970 to 9.1 m in 1987 and then to 13.6 m in 2002.

[19] Paragraph 2.2 of IAEA Safety Standards Series No. GS-R-3, Management System for Facilities and Activities [104], states that: "Safety shall be paramount within the management system, overriding all other demands". Annexes I and II of the IAEA Safety Culture Assessment Review Team Guidelines (IAEA Service Series No. 16) [142] provide a set of characteristics, attributes and guiding questions that are helpful in assessing whether an organization is effectively upholding nuclear safety as taking priority over all other demands.

The stark difference in design tsunami heights between the Fukushima Daiichi and Onagawa NPPs, which are located 120 km apart along the east cost of Japan's main island of Honshu, is due largely to two main factors.

First, there are significant differences in the shape of the coastlines at the two sites. The location of the Fukushima Daiichi NPP, along a rectilinear coastline, made it inherently less vulnerable to large tsunamis as compared with the Onagawa NPP, whose V shaped coastline was expected to amplify tsunami height for a given wave source. Figures 2.1–7 and 2.1–8 provide an illustration of this effect.

Second, most large historical earthquakes in the Japan Trench off the eastern coast of Honshu were concentrated closer to Iwate and Miyagi Prefectures as compared with Fukushima Prefecture. As noted in TEPCO's Fukushima Nuclear Accident Analysis Report, there were no records of large earthquakes in the region along the Japan Trench offshore from Fukushima Prefecture. Figure 2.1–5 provides an overview of the regions that had the largest historical earthquakes and tsunamis (see also Section 2.1).

In 2007, Fukushima Prefecture and the neighbouring Ibaraki Prefecture updated their tsunami wave source models for their respective Disaster Prevention Plans. TEPCO and the Japan Atomic Power Company (JAPC), the operator of the Tokai Daini NPP, calculated their tsunami heights based on these new wave source models, to maintain alignment with the assumptions in the Disaster Prevention Plans. In the case of the Fukushima Daiichi NPP, no action was required as the resultant tsunami height was less than 5.7 m; in the case of the Tokai Daini NPP, the design tsunami height had to be revised upwards, and the JAPC decided to construct a 7.0 m high wall at the Tokai Daini NPP to protect the emergency seawater pump room [10]. This wall was effective in protecting the plant from the 5.4 m tsunami wave that struck the plant in March 2011.

In 2009, TEPCO used updated topography and bathymetry data to revise its design tsunami height to 6.1 m, and accordingly made further modifications to the plant.

Table 2.7–1, reproduced from the TEPCO Accident Analysis Report of June 2012 provides a timeline of the revisions to the design tsunami height, and the actions taken in response to these revisions, for the Fukushima Daiichi, Fukushima Daini, Tokai Daini and Onagawa NPPs. It is evident from this information that TEPCO recognized, analysed and took corrective action based on improved analytical methods concerning new tsunami knowledge.

In addition to the studies completed by TEPCO between 2002 and 2009 based on the JSCE methodology and its standard wave source models, TEPCO completed two trial calculations using wave source models or methodologies that went beyond the JSCE approach. Each of these trial calculations predicted a tsunami wave height considerably greater than the design tsunami height of 5.7–6.1 m, calculated by the JSCE methodology [33]. It is important to note that the JSCE method was the accepted consensus standard at the time of March 2011.

In one trial calculation, TEPCO applied a wave source model based on the Jogan 869 tsunami proposed by Satake et al. [143] in 2008. The trial calculation yielded a tsunami wave height of approximately 9 m. Since this model was based solely on tsunami deposit survey results, TEPCO decided to perform a tsunami deposit survey to confirm the validity of the 9 m prediction. Approximately 50 bore holes were dug at five locations along the coast in Fukushima Prefecture. Of these, the three southern locations yielded no evidence of tsunami deposit, one of the northern locations yielded evidence of a tsunami deposit of 0.5 m thickness, and the other northern location yielded evidence of a tsunami deposit of 3–4 m thickness [33].

TABLE 2.7–1. EVALUATION OF TSUNAMI EVENTS [10]

Time of tsunami evaluation	Fukushima Daiichi	Fukushima Daini	Tokai No. 2	Onagawa
At the time of approval for establishment	1966 OP +3.122 m (1960 Chilean earthquake and tsunami)	1972 Unit 1 OP +3.122 m 1978 Units 3 and 4 OP +3.705 m (1960 Chilean earthquake and tsunami)	— (Highest high water level 27 September 1956 Kanogawa Typhoon TP +3.24 m)	1970 OP +2~3 m 1987 OP +9.1 m (1611 Keicho Sanriku tsunami)
1994	OP +3.5 m Measures unnecessary (determined based on the Chilean earthquake and tsunami. Calculations were also made with the Keicho Sanriku tsunami, but the values were below that of the Chilean earthquake and tsunami)	OP +3.6 m Measures unnecessary (Same as left)	—	—
2002	JSCE issues Tsunami Assessment Method			
	OP +5.7 m (Determined based on the Shioyazaki-oki earthquake. Calculations were also made with the Keicho Sanriku tsunami, but the values were below that of the Shioyazaki-oki earthquake) Measures implemented (Pumps made 200 mm higher, etc.)	OP +5.2 m (Same as left) Measures implemented (Watertight heat exchanger buildings, etc.)	TP +4.88 m Measures unnecessary	OP +13.6 m (Determined based on offshore Sanriku earthquakes) Measures unnecessary
2007	Estimation of tsunami height by utility company using the wave source model set by Fukushima Prefecture			
	Around OP +5 m Measures unnecessary	Around OP +5 m Measures unnecessary	—	—
	Estimation of tsunami height by utility company using the wave source model set by Ibaraki Prefecture			
	OP +4.7 m Measures unnecessary	OP +4.7 m Measures unnecessary	OP +5.72 m Measures implemented (Higher walls of the pump room)	—
2009*	OP +6.1 m Measures implemented (pumps made higher, etc.) (Determined based on the Shioyazaki-oki earthquake)	OP +5.0 m Measures implemented (Determined based on the Shioyazaki-oki earthquake)	—	—
2011	Great East Japan earthquake			
	Tsunami height OP +13.1 m	Tsunami height OP +9.1 m	TP +5.4 m	OP +13.8 m

* Evaluated with the same method as that of 2002 using bathymetric data updated to the newest occurrences.

In the second trial calculation, TEPCO applied the JSCE method, but used an approach proposed in 2002 by HERP, which was to assume that an M 8.2 earthquake could occur anywhere along the Japan Trench, even in regions where there was no record of a major historical earthquake. HERP did not specify a wave source model, and TEPCO conservatively used the wave source model of the Meiji

Sanriku earthquake of 1896 (M 8.3). This earthquake occurred off the coast of Iwate Prefecture, more than 100 km north of Fukushima Prefecture, and resulted in a tsunami of 38 m. The result of the trial calculation showed a maximum tsunami height of 8.4 m to 10.2 m at the front of the intake at the Fukushima Daiichi NPP and a flood height of 15.7 m, taking runup into account [33].

As a result of these trial calculations, TEPCO departments responsible for civil engineering and tsunami evaluation had information indicating that an earthquake in the Japan Trench larger than historically observed in the vicinity of Fukushima Prefecture, and comparable to the largest earthquakes observed elsewhere along the Japan Trench, had the potential to result in a tsunami whose height significantly exceeded the site elevation. Notwithstanding, they had doubts regarding the appropriateness of the trial calculations and remained confident that the safety margin was adequate. This confidence was based largely on the long held basic assumption that the site was not prone to large historical tsunamis, as described in the preceding section, even though some experts, including those at HERP, did not share this confidence in the safety margin (see Section 2.6).

The JSCE was asked to review the appropriateness of the wave source models and determine whether a revision to the corresponding standard was warranted. In the meantime, TEPCO did not initiate action to make changes to station design or operation pending confirmation by the JSCE, and communicated the results of its findings and its planned actions to the Japanese regulatory agencies [32]. The review by the JSCE was in progress at the time of March 2011.

The nuclear safety implications of the tsunami evaluations were not recognized. The engineers and managers involved in the tsunami evaluations had not received training on nuclear systems and did not understand that a flooding event had the potential to lead to severe damage of the reactor core due to a total loss of all sources of power. Furthermore, they did not communicate to the departments responsible for the safety and design of the facility that the tsunami evaluation results had the potential to vary greatly depending upon the assumptions used in the analysis, and had the potential to significantly exceed the site elevation [54]. When confronted with new information that is based on new research findings or new analytical methods, it is reasonable to seek expert input and to perform additional confirmatory analysis to determine the appropriate long term corrective action; such re-analysis may take months, perhaps years, particularly for situations where there is a paucity of information. However, when confronted with new information, it is also necessary to assess, without delay, the credibility of the new information and the potential safety consequences; if the new information is judged to be credible and to have a potential safety consequence, it is necessary to establish interim compensatory measures pending confirmation of the new findings and pending the implementation of the long term corrective actions. The urgency of the response to the new findings needs to be commensurate with their potential safety significance.

It is vital in any nuclear organization for senior decision makers and managers to have sufficient knowledge and experience to recognize and understand the nuclear safety significance of new information and to apply a strong questioning attitude to challenge the organization to consider all potential nuclear safety consequences and potential mitigating actions. A good question to ask whenever confronted with new information is: “What is the worst that can happen?”

Paragraph 4.4 of IAEA Safety Standards Series No. GS-R-3 [104] specifies:

> “Senior management shall ensure that individuals are competent to perform their assigned work and that they understand the consequences for safety of their activities. Individuals shall have received appropriate education and training, and shall have acquired suitable skills, knowledge and experience to ensure their competence.”

Moreover, paragraph 6.16 of GS-R-3 stipulates:

> "Potential non-conformances that could detract from the organization's performance shall be identified. This shall be done: by using feedback from other organizations, both internal and external; through the use of technical advances and research; through the sharing of knowledge and experience; and through the use of techniques that identify best practices."

2.7.2.3. Application of learning from periodic safety reviews

TEPCO has been actively engaged in developing improved BWR designs based on its own operating experience, as well as international advances in BWR technologies. Improvements have been reflected primarily in new construction, with less emphasis placed on backfitting safety design improvements to existing reactors [54]. For example, the EPS for Unit 6 at the Fukushima Daiichi NPP featured more separation in the design and layout of its electrical distribution system that made it more resilient to common cause events. However, the earlier units at the Fukushima Daiichi NPP were not similarly retrofitted leaving them more vulnerable to common cause events.

Part of the rationale for not backfitting design improvements to the older units at the Fukushima Daiichi NPP was that the PSAs for these units indicated that they posed an acceptably low risk. However, the scope of the PSA was limited to internal events and did not take account of all hazards, such as internal hazards posed by internal flooding and external hazards posed by tsunamis, and all potential sources of radioactive releases, such as the SFPs and multiple units on a single site. As a result, the PSA did not provide a full characterization of the risk. This led to a widely held belief that the nuclear safety risk posed by the plant was acceptable, and the cost of retrofits was not commensurate with the safety benefit (see Section 2.6).

Although TEPCO implemented a PSR process starting in 1994, its scope was focused narrowly on ageing management and did not include a detailed review of the PSA.

Paragraph 5.61 of IAEA Safety Standards Series No. SSG-25 [139] states:

> "A review of the probabilistic safety assessment (PSA) should be conducted to identify weaknesses in the design and operation of the plant and, as part of the global assessment, to evaluate and compare proposed safety improvements."

In addition, para. 5.62 of SSG-25 specifies that the scope of the PSA should include all operational states and identified internal and external hazards.

2.7.3. Regulatory oversight of TEPCO's operating experience programme

Regulators worldwide have access to international operating experience events reported through the IAEA–OECD/NEA Incident Reporting System, discussed earlier, including access to the triennial reports that provide a synthesized summary of the most important learning points from operating experience.

However, as discussed in Section 2.5, the regulatory framework within Japan was such that NISA had no legal mechanism to enforce new requirements upon the licensees in response to operating experience, and did not have a backfit rule. This limitation extended to PSRs, which were focused more on ageing management [7].

There is one example where NISA requested licensees to take action in response to international operating experience relating to the potential loss of post-LOCA recirculation capability due to insulation debris. Aside from this, NISA was not able to identify other examples where it had requested licensees to take action in response to operating experience [7].

Paragraph 3.3 of IAEA Safety Standards Series No. NS-G-2.11 [141] specifies that:

> "…Regulatory bodies should review the screening of events to gain insights that can be used to inform their inspection programmes, licensing activities, and the elaboration of regulations and requirements for safety backfits. Regulators should screen national reports for their international use."

2.7.4. Summary

Over the past 15 years, there have been several international precursor events where natural hazards exceeded the design basis for the affected NPPs. In general, less emphasis is given to the actions taken by countries in response to events that occurred elsewhere than the actions taken in the country where the accident happened. For an operating experience programme to be effective, it must function within a management system where nuclear safety is paramount and overrides all other demands. It is important to periodically check if the organization is effectively upholding nuclear safety as the overriding priority.

A pipe break in the turbine hall in 1991, which allowed sea water to enter into an EPS room, should have revealed the susceptibility of the EDGs to flood, but no action was taken by TEPCO. The operator should also have recognized from its operational experience programme events, where two other NPPs were exposed to beyond design basis natural hazards. These were a storm surge in December 1999, which caused flooding at two reactors at the Le Blayais NPP in France, and the Indian Ocean Tsunami of 26 December 2004, which flooded seawater pumps at the Madras Atomic Power Station in India. TEPCO had made no design or operational changes as a result of these events.

By comparison, TEPCO had made many changes following the earthquake of 16 July 2007, which exceeded the design basis of the Kashiwazaki-Kariwa NPP in Niigata Prefecture. At each of its stations, it constructed seismically isolated emergency buildings, improved firefighting systems and implemented modifications to enable the use of portable pumpers as an alternate source of water injection into the reactor. All these improvements proved to be of significant benefit for the response to the accident. This demonstrates that TEPCO's process for screening and acting on operational experience could be effective if appropriately applied.

Trial calculations using new analytical methods for assessing tsunami hazards in Japan indicated that the tsunami hazard had the potential to be worse than assumed in the design of the Fukushima Daiichi NPP. When confronted with new information and experiences based on new research findings or new analytical methods, it is important to seek expert inputs and, if necessary, to perform additional confirmatory analysis to determine the corrective actions needed. However, it is also necessary to assess, without delay, the credibility of the new information and the potential safety consequences. The nuclear safety implications of the new tsunami evaluations were not recognized, because the engineers and managers involved in these evaluations had not received training on nuclear systems and did not consider that a flooding event had the potential to lead to severe damage of the reactor core. It is vital in any nuclear organization that senior decision makers and managers have sufficient knowledge and experience to recognize and understand the nuclear safety significance of new information, and to apply a strong questioning attitude to challenge the organization to consider all potential nuclear safety consequences and potential mitigating actions.

The regulatory framework within Japan was such that there was no legal mechanism to enforce new requirements upon the licensees in response to operating experience. While operators have the primary responsibility for implementing operating experience, it is important for regulators to have authority to require operators to take action when they deem that operation of the NPP poses an unreasonable risk to the public.

2.7.5. Observations and lessons

- **The effectiveness of operating experience programmes needs to be confirmed periodically and independently through a detailed review of the specific actions taken in response to international operating experience.**
  This review can be made a standard part of existing oversight processes considering relevant sources such as the Incident Reporting System (IRS) by the IAEA and the OECD/NEA.
- **When assessing the applicability of significant operating experience with limited consequences, it needs to be considered whether the consequences could have been much worse had there been a small difference in the initiating event, or in the progression of the event.**
  When assessing the applicability of significant operating experience, the potential for similar consequences from different initiators needs to be considered.
- **The operating experience programme needs to function within a management system where nuclear safety is paramount and overrides all other demands.**
  The management system needs to include objective risk informed decision making criteria to support decisions for retrofitting safety design improvements. It is important to periodically check if the organization is effectively upholding nuclear safety as the overriding priority.
- **The potential implications of new safety issues need to be assessed without delay, and interim compensatory actions need to be taken to maintain the safety margin pending final confirmation of the problem.**
  New safety issues could arise from research findings, new analytical methods or learning from experience. The consideration and effective implementation of interim actions need to be part of the operating experience programme. It is vital in any nuclear organization that senior decision makers and managers have sufficient knowledge and experience to recognize and understand the nuclear safety significance of new information, and to apply a strong questioning attitude to challenge the organization to consider all potential nuclear safety consequences and potential mitigating actions. While operators have the primary responsibility for implementing operating experience, it is important for regulators to have authority to require operators to take action when they deem that operation of the NPP poses an unreasonable risk to the public.
- **Objective risk informed decision making criteria need to be used to support decisions for retrofitting safety design improvements as part of a periodic safety review process.**
  Risk assessment tools (e.g. PSAs) used to support risk informed decision making need to take account of all hazards, all operation modes and all potential sources of radioactive releases.
- **Regulatory bodies need to perform independent reviews of national and international operating experience to confirm that licensees are taking appropriate action in response to operating experience.**
  This does not override the primary responsibility of the operators for implementing operating experience.

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CONTENTS OF CD-ROM

The following annexes to Technical Volume 2 are included on the attached CD-ROM:

Annex I: Historical development of the governmental, legal and regulatory framework for nuclear safety in Japan.

Annex II: Aspects related to the human and organizational factors of the accident.

Annex III: Detailed description of relevant operating experience.

ABBREVIATIONS

ABWR	advanced boiling water reactor
ADS	automatic depressurization system
AM	accident management
AMTA	Ageing Management Technical Assessment
ANRE	Agency for Natural Resources and Energy
ANSN	Asian Nuclear Safety Network
AOPs	abnormal operating procedures
ARI	alternate rod insertion
ATWS	anticipated transient without scram
BDBA	beyond design basis accident
BDBEE	beyond design basis external event
BWR	boiling water reactor
BWROG	BWR Owners Group
CCS	containment cooling system
CDF	core damage frequency
CDMC	Central Disaster Management Council
CFF	containment failure frequency
CNRA	Committee on Nuclear Regulatory Activities
CNS	Convention on Nuclear Safety
CRD	control rod drive
CRDHCS	control rod drive hydraulic control system
CRPPH	Committee on Radiation Protection and Public Health
CS	core spray
CSNI	Committee on Safety of Nuclear Installations
CSS	Commission on Safety Standards
CST	condensate storage tank
DBAs	design basis accidents
DC	direct current
DDFP	diesel driven fire pump
DID	defence in depth
EBP	extrabudgetary programme
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	emergency operating procedure
EP	Establishment Permit
EPG	emergency procedure guideline
EPS	emergency power system
ERC	Emergency Response Centre
ERSS	Emergency Response Support System
ESS	emergency seawater system
FEPC	Federation of Electric Power Companies
FSAR	Final Safety Analysis Report
GMPE	ground motion prediction equation
HCLPF	High Confidence of Low Probability of Failure
HERP	Headquarters for Earthquake Research Promotion
HP	Hitachi Port
HPCI	high pressure coolant injection
HPCS	high pressure core spray
I&C	instrumentation and control
IC	isolation condenser
ICANPS	Investigation Committee on the Accident at the Fukushima Nuclear Power Stations of Tokyo Electric Power Company

IPE	individual plant examination
IPEEE	individual plant examination of for external events
INES	International Nuclear and Radiological Event Scale
INPO	Institute of Nuclear Power Operations
INRA	International Nuclear Regulators Association
INSAG	International Nuclear Safety Group
IRRS	Integrated Regulatory Review Service
IRS	Incident Reporting System
ISSC	International Seismic Safety Centre
JAEA	Japan Atomic Energy Agency
JAEC	Japan Atomic Energy Commission
JAPC	Japan Atomic Power Company
JAPEIC	Japan Power Engineering and Inspection Corp
JEA	Japan Electric Association
JCO	Japan Nuclear Fuels Conversion Company
JNES	Japan Nuclear Energy Safety Organization
JSCE	Japan Society of Civil Engineers
KEPCO	Kansai Electric Power Company
LOCA	loss of coolant accident
LOOP	loss of off-site power
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LTMP	Long-term Maintenance Programme
LUHS	loss of ultimate heat sink
M/C	metal clad switch gear
MCR	main control room
METI	Ministry of Economy, Trade and Industry
MEXT	Ministry of Education, Culture, Sports, Science and Technology
MITI	Ministry of International Trade and Industry
MOFA	Ministry of Foreign Affairs
MSIV	main steam isolation valve
MUWC	make-up water condensate
NERHQ	Nuclear Emergency Response Headquarters
NIRS	National Institute of Radiological Sciences
NISA	Nuclear and Industrial Safety Agency
NPP	nuclear power plant
NPSH	net positive suction head
NRA	Nuclear Regulation Authority
NRC	United States Nuclear Regulatory Commission
NSC	Nuclear Safety Commission
NUPEC	Nuclear Power Engineering Corporation
NUSS	Nuclear Safety Standards
NUSSC	Nuclear Safety Standards Committee
NUSTEC	Nuclear Safety Technology Centre
OECD/NEA	OECD Nuclear Energy Agency
OFC	Off-site Centre
OP	Onahama Port
OSART	Operational Safety Review Team
OTS	operational technical specification
PAZ	precautionary action zone
PCV	primary containment vessel
PIE	postulated initiating event
PSA	probabilistic safety assessment
PSHA	probabilistic seismic hazard analysis
PSR	periodic safety review

RCA	radiologically controlled area
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RPT	recirculation pump trip
RPV	reactor pressure vessel
PWR	pressurized water reactor
SAM	severe accident management
SAMG	severe accident management guideline
SAR	safety analysis report
SBGT	standby gas treatment
SBO	station blackout
SC	suppression chamber
SFP	spent fuel pool
SLCS	standby liquid control system
SPDS	Safety Data Parameter Display System
SRV	safety relief valve
SSC	structures, systems and components
STA	Science and Technology Agency
TAF	top of active fuel
TEPCO	Tokyo Electric Power Company
TP	Tokyo Peil
TRANSSC	Transport Safety Standards Committee
TSC	technical support centre
U-D	up-down (vertical)
UHS	ultimate heat sink
WANO	World Association of Nuclear Operators
WASSC	Waste Safety Standards Committee
WGOE	Working Group on Operating Experience
ZPGA	zero period ground acceleration

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MEETINGS

Working Group (WG) meetings

18 March 2013
Initial meeting of the WG Co-Chairs, Vienna

21–22 March 2013
1st meeting of all WGs, Vienna

12–14 June 2013
2nd meeting of all WGs, Vienna

12–13 September 2013
3rd meeting of WGs 1 and 2, Vienna

9–13 December 2013
4th meeting of all WGs, Vienna

10–14 February 2014
5th meeting of all WGs, Vienna

14–17 April 2014
6th meeting of WGs 1, 2 and 3, Vienna

International Technical Advisory Group (ITAG) meetings

21–22 March 2013
1st ITAG meeting, Vienna

10 June 2013
1st Joint ITAG/Co-Chairs meeting, Vienna

11 June 2013
2nd ITAG meeting, Vienna

6 December 2013
2nd Joint ITAG/Co-Chairs meeting, Vienna

7 May 2014
3rd Joint ITAG/Co-Chairs meeting, Vienna

23–24 October 2014
4th Joint ITAG/Co-Chairs meeting, Vienna

23–24 February 2015
5th Joint ITAG/Co-Chairs meeting, Vienna

Consultants services (CS) meetings

6–7 August 2013
CS on Source Term, Vienna

29–31 October 2013
CS on Human and Organizational Factors and Safety Culture, Vienna

17–21 November 2013
CS on Human and Organizational Factors and Safety Culture, Atlanta

13–17 January 2014
CS on Human and Organizational Factors and Safety Culture, Vienna

17–21 March 2014
CS on Human and Organizational Factors and Safety Culture, Ottawa

Bilateral meetings in Japan

20–24 January 2014
CS to Discuss Issues Related to Regulatory Activities, Operating Experience and Waste Management Topics in Connection with the Preparation of the IAEA Report

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