Report of JSME Research Committee on the Great East Japan Earthquake Disaster

Chapter 8

Nuclear Codes and Standards Issues and Future Perspectives

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Abstract

Working Group (WG6), titled "Nuclear Codes and Standards Issues and Future Perspectives", focused its investigation on the impacts of the 2011 Great East Japan Earthquake, which was followed by a devastating tsunami, on the nuclear power plants (NPPs) located along the northeastern Pacific coast of Japan. The objectives of WG6 were to survey the damages of the NPP facilities that were caused by the earthquake and tsunami and the subsequent consequences (accidents), to examine and analyze the causes of these damages and accidents within the ability of the JSME as a professional society, to conduct a root cause analysis of the damages and accidents and to give considerations on the relationship between the codes and standards issues, and to analyze cases where countermeasures against the earthquake and tsunami worked well and to obtain insights from the analysis. In order to determine the root causes of the Fukushima Daiichi NPP accident and to determine the roles that the codes and standards have in developing safety enhancement measures to prevent future accidents, WG6 conducted studies on the following three viewpoints

Viewpoint 1: Reasons why the accident at the Fukushima Daiichi NPP was unpreventable

Viewpoint 2: The roles of the codes and standards responding to Severe Accidents (SAs)

Viewpoint 3: The adequacy and issues of the current seismic design technology

Based on these studies, WG6 created a set of tangible recommendations on what the future nuclear codes and standards should be to enhance the safety of NPP facilities.

Keywords: Fukushima Daiichi accident, Nuclear safety regulation, Design basis, External hazards, Codes and Standards, Risk management, Seismic design

1. Introduction

Working Group (WG6), titled "Nuclear Codes and Standards Issues and Future Perspectives", focused its investigation on the impacts of the 2011 Great East Japan Earthquake, which was followed by a devastating tsunami, on the nuclear power plants (NPPs) located along the northeastern Pacific coast of Japan. The objectives of WG6 were to survey the damages of the NPP facilities that were caused by the earthquake and tsunami and the subsequent consequences (accidents), to examine and analyze the causes of these damages and accidents within the ability of the JSME as a professional society, to conduct a root cause analysis of the damages and accidents and to give considerations on the relationship between the codes and standards issues, and to analyze cases where countermeasures against the earthquake and tsunami worked well and to obtain insights from the analysis. Based on the studies that were conducted, WG6 created a set of tangible recommendations on what the future nuclear codes and standards should be to enhance the safety of NPP facilities.

As a first step, WG6 conducted a fact finding survey on the impacts of the earthquake and tsunami on the typical nuclear power stations located along the northeastern Pacific coast of Japan, which included the Fukushima Daiichi NPP (Tokyo Electric Power Co.), Onagawa NPP (Tohoku Electric Power Co.), and the Tokai Daini NPP (Japan Atomic Power Co.). This survey was primarily constructed based on the information available in the public domains and utility

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reports.

Next, in order to determine the root causes of the Fukushima Daiichi NPP accident and to determine the roles that the codes and standards have in developing safety enhancement measures to prevent future accidents, WG6 formulated the following three viewpoints:

Viewpoint 1: Reasons why the accident at the Fukushima Daiichi NPP was unpreventable

Viewpoint 2: The roles of the codes and standards responding to Severe Accidents (SAs)

Viewpoint 3: The adequacy and issues of the current seismic design technology

Then, WG6 conducted an extensive survey study and investigation based on these three questions. Through the investigation and discussion among the members, several issues, perceptions, and concerns were raised relating to these viewpoints, which were finally put together as a set of recommendations and proposals by WG6.

2. Impacts of the earthquake and tsunami

Table 1 summarizes the outline of the impacts of the 2011 Great East Japan Earthquake that was followed by a devastating tsunami that affected the NPPs located along the northeastern Pacific coast of Japan.

Table 1 Summary of the Impacts of the Earthquake and Tsunami on the NPPs

Plant Site	PGA of EQ gal*1	Tsunami Height m	Plant Protective Response		
			Reactor Shutdown*3	Core Cooling *4	Containment
Higashidori	17 / 450 *2	> 4	Success	Success	Success
Onagawa	573 / 512	13	Success	Success	Success
Fukushima Daiichi	550 / 438	14–15	Success	Failure*5	Failure*5
Fukushima Daini	277 / 428	7	Success	Success	Success
Tokai Daini	225 / 400	5.4	Success	Success	Success

*1 Representative value of the peak ground acceleration (PGA) of the earthquake at each plant site

*2 Design value / Observed value

*3 Automatic reactor shutdown (scram)

*4 Structural integrity of the reactor cooling system, secured emergency power supply, and ultimate heat sink

*5 Units 1–3. Unit 4 experienced an outage and units 5 and 6 reached the cool shutdown state safely

The following can be noted after reviewing Table 1:

- The PGA exceeded the design value at the Onagawa and Fukushima Daiichi NPPs, while the PGAs at the other sites were lower than their design values.
- The Fukushima Daiichi NPP was struck by a series of huge tsunami waves that had heights as high as 14–16 m, which significantly exceeded the design tsunami wave height of 5.4–5.7 m. The other sites were also struck by tsunami waves, but their heights were not larger than their design wave heights.

At the Fukushima Daiichi NPP station, although all the offsite power was lost, three operating units (units 1–3) were successfully shutdown by the automatic scram system that detected the earthquake. However, the overwhelming tsunami inundated deep into the plant facility, disabling the emergency diesel power supply and all AC power, and these units fell into the state of a long-term station black out. As a result, the reactor core cooling capability was lost, the reactor cores were severely damaged and melted, and hydrogen explosions occurred. Furthermore, an SA occurred, which caused a large amount of radioactive material to be released into the environment.

On the other hand, at the NPP stations other than that of the Fukushima Daiichi NPP, all of the operating units were automatically shut down after the earthquake was detected and the functions of the emergency power supply systems were maintained in spite of partial loss of the offsite power at some stations. The reactor cooling capability of these units was maintained and all units reached a safe cool shutdown state, and the safety of the reactors was secured. In addition, no significant damage due to the earthquake load has been reported on the structures, systems, and components important for safety.

As for the Fukushima Daiichi NPP accident, several organizations conducted thorough investigations and have

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issued detailed reports; these organizations include the Government of Japan (2011, 2012), the National Diet of Japan (2012), and IAEA (2011).

3. Insights regarding the three viewpoints

3.1 Viewpoint 1: Reasons why the accident at the Fukushima Daiichi NPP was unpreventable

To simplify, the direct cause of the Fukushima Daiichi NPP accident may be attributed to the fact that the facility was struck by a devastatingly huge tsunami that had a height that significantly exceeded the design basis. Naturally, there is a question on the adequacy of the design basis of the Fukushima Daiichi NPP regarding tsunami heights. As another important cause of the accident, there was no supposition for the possibility of a tsunami exceeding the design basis. Therefore, no countermeasures were taken against a tsunami event that exceeded the design basis. Adequate risk management was also implemented. It was unclear to as why the risk management was neglected. Some important issues that are immanent in the fundamental management systems, the preparedness of the owners to secure safety, and the roles of the nuclear safety regulation system are as follows:

- there was a lack of attitude to promptly and flexibly apply new knowledge to facility and equipment improvement and regulatory requirements;
- the arguments on nuclear safety often tended to lapse into the dualism of “safe or dangerous”, which acted as an obstacle to the incentive for improvement;
- there was a lack of objectives that could have been constantly pursued to enhance safety, and although the regulatory requirements and the codes and standards were the “minimum” requirements that needed to be observed, there was a tendency to recognize that only satisfying the requirements was sufficient;
- risk information was utilized not to “determine vulnerabilities and to improve”, which is its primary purpose, but instead for “demonstrating safety”; and
- there was a lack of will to focus on the international trends in safety regulations, such as the activities conducted by IAEA and US NRC, and to achieve international harmonization in terms of the regulatory system.

Reflecting on the lessons learned from the Fukushima Daiichi NPP accident, fundamental changes were made to the government’s nuclear safety regulation, which include the following:

- inclusion of SAs into the scope of the nuclear safety regulation (prior to the Fukushima accident, the SA management was implemented by owners’ voluntary efforts);
- introduction of the “back fit” regulation where any amended requirements based on new knowledge are applied to the existing fleets of the NPP; and the
- introduction of a limitation of 40 years for the lifetime of a NPP as a measure for aging plants.

While questions may arise on the adequacy of scientific and engineering basis of the 40-year lifetime limitation, it is expected that an effective and efficient nuclear safety regulation system can be implemented and operated based on scientific rationalism. Then, in addition to appropriately recognizing and moving forward on the issues mentioned above, efforts that are also important on the systematic documentation of the regulatory requirements to secure regulatory transparency and the training and education for the professionals of the nuclear safety regulation to gain high level of expertise are needed.

3.2 Viewpoint 2: The roles of the codes and standards responding to the severe accidents

The Main Committee on Power Generation Facility Codes of JSME, established in 1998, develops and publishes various codes and standards mainly related to the structural integrity of mechanical components in NPP facilities. The typical codes and standards include the material code, design and construction code, the welding code, and the fitness for service code. These JSME codes are counterparts, in a sense, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Codes, Section II (material), Section III (construction), Section V (welding), and Section XI (ISI). The nuclear codes and standards of JSME have been applied to the nuclear safety regulations of the existing fleets of the domestic and newly built NPPs after technical assessments had been conducted and endorsed by the regulatory authority.

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The important points for exploring the role of the codes and standards reflecting the Fukushima Daiichi NPP accident and the reorganization of the nuclear regulatory system may include the following:

1) Development of codes and standards for the SA conditions; and
2) Continued and enhanced utilization of the codes and standards of private sectors via regulatory authority.

3.2.1 Codes and standards for severe accidents

The scope of the current codes and standards include design basis accidents (e.g., the Loss of Coolant Accident (LOCA)). SAs such as core meltdowns have been defined as beyond design basis events (BDBEs), and have been out of the scope of the current codes and standards. This corresponds to the fact that in Japanese nuclear regulations, the SA issues, such as those associated with the development of an accident management program, were not included in the regulatory scope, but rather left for the owner's voluntary safety effort or activities. However, after the Fukushima Daiichi NPP accident, the Japanese nuclear regulations have been completely restructured, including their organizational aspect, making the SA measure the most important regulatory issue.

On the other hand, the Fukushima Daiichi NPP accident revealed the notion that nuclear power plants should be designed and equipped with the necessary equipment so that the safety of the plant is secured by ensuring that the safety functions of shutting down the plant, cooling the core, and containing any radioactive material are operating properly, even when the plant is exposed to a severe threat of external events (earthquakes, tsunamis, hurricanes, etc.) exceeding the design basis. The codes and standards for embodying such plant designs and equipment installations are needed, as schematically illustrated in Fig. 1.

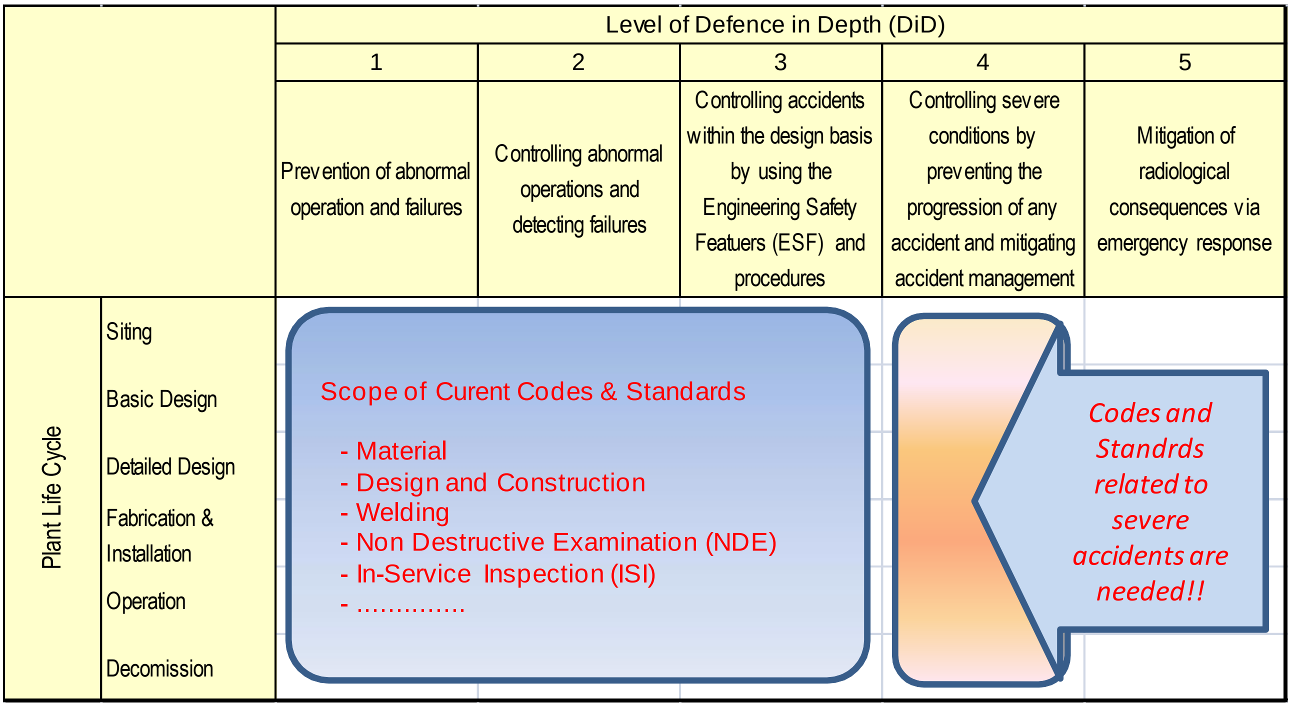

Fig. 1 Scope of the current and necessary codes and standards with respect to the level of the accident

Codes and standards are need for the SAs that are outside the scope of JSME, and thus the three major Standards Development Organizations (SDOs) of Japan (JSME, the Atomic Energy Society of Japan (AESJ), and the Japan Electric Association (JEA)) are working together under the framework of the "Nuclear Codes and Standards Consortium" to draw an overall picture of the necessary codes and standards for the SA measures. It is noted that in this effort of systematic development, an overall view of the entire picture is critical, as is schematically shown in Fig. 2.

In the Main Committee on Power Generation Facility Codes of JSME, as part of the effort for developing the needed codes and standards for this new area of SA measures, work is underway to develop guidelines that are

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considered to be of the highest priority when restarting the existing plants in Japan.

One guideline is the “Severe Accident Management Design Guideline for External Events”, of which the purpose and scope are as follows:

- to provide guidelines or a common basis as references for the utility operators when they develop or refine their accident management programs;
- the guideline provides SA preventing measures by implementing alternative equipment and/or original equipment;
- the guideline also provides SA mitigating measures and strengthening measures for SA management programs; and
- the guideline currently applies to Boiling Water Reactors (BWRs)

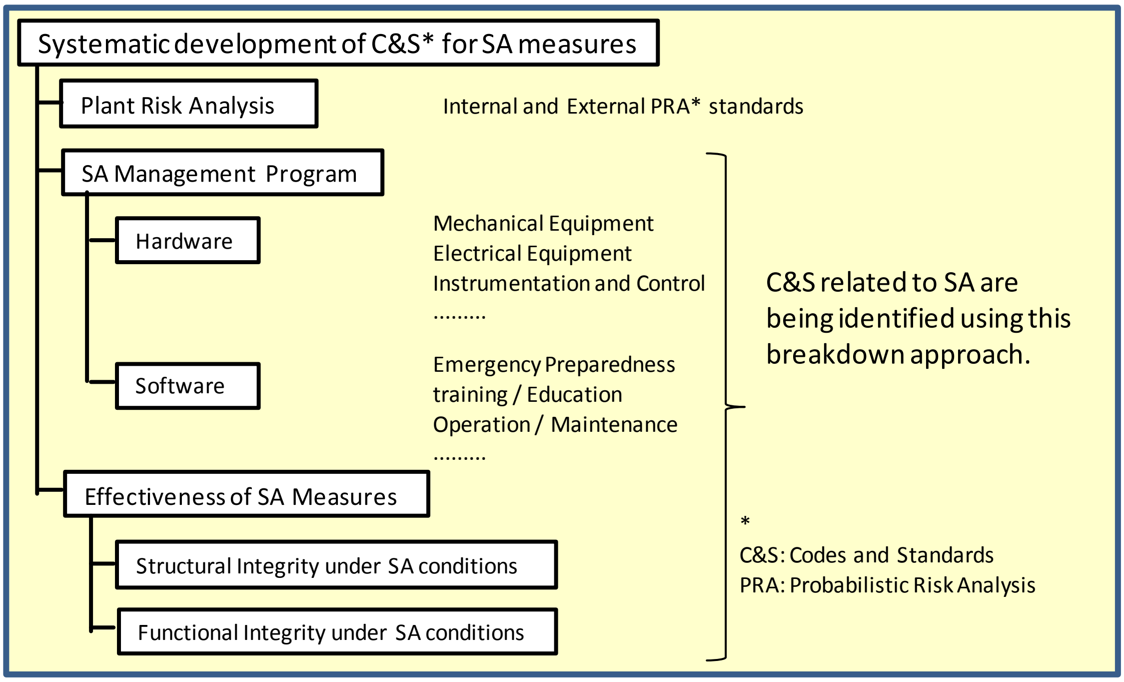

Fig. 2 Systematic development of codes and standards for SA measures

The external hazards that are considered include earthquakes, coastal flooding due to tsunamis and high tides, river flooding due to the collapse of a dam or river bank, windstorms such as typhoons, hurricanes, and tornados, heavy snowfall and very low temperatures, landslides and avalanches, fine particles such as volcanic ash, offsite fires, and the combination of earthquakes/coastal flooding, earthquakes/river flooding, and earthquakes/offsite fires.

The guideline requires the following functions to be secured for the SA prevention measures:

- Reactor cooling capability via a high-pressure injection system
- Reactor cooling capability via a low-pressure injection system
- PCV venting
- RPV depressurization capability
- Long-term reactor cooling capability
- Spent fuel cooling capability

The guideline also requires the following functions to be secured for SA mitigation:

- Capability to cool debris and the containment vessel (CV)
- Capability to prevent CV failure due to over heating/pressure

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- Capability to prevent CV failure due to damage caused by debris
- Capability to prevent hydrogen explosions
- Decrease in radiation exposure and control of the release of fission products

Another effort is also underway to develop the “Guideline for a Structural Integrity Evaluation of Containment Vessels under Severe Accident Conditions”. Here, the basic idea is that keeping the boundary (barrier) function of the containment vessel intact under SA loads is of significant importance to suppress the uncontrolled release of radioactive materials into the environment. Furthermore, its fundamental purpose is to provide guidelines for structural analysis methodologies and failure criteria for structural integrity evaluations under SA loading and SA environments. In this guideline, the loads that are taken into consideration include over pressuring and overheating due to core meltdown in a SA. The failure modes include the following:

- ductile failure or damage to the containment shell
- severe wall cracking at geometrical discontinuities (e.g., reinforced openings)
- buckling of a torispherical, ellipsoidal head by external pressure
- buckling of penetration bellows by internal pressure
- loss of leak tightness at bolted flanges

A detailed three-dimensional elastic-plastic finite element analysis will be applied and a strain-based failure criterion for ductile failures will be provided.

The Main Committee on Power Generation Facilities Codes has a close cooperative relationship with the ASME Boiler and Pressure Vessel Code Committees. In fact, a number of JSME code engineers have participated in several of these ASME committees and their sub-tier groups as members or officers. It should be noted that a cooperation framework between JSME and ASME has also been established for the effort of developing necessary codes and standards for SA measures. The guidelines depicted above were presented to ASME at their draft state and ASME provided a number of useful comments to JSME.

3.2.2 To continue and enhance the utilization of the codes and standards of the SDOs

The nuclear codes and standards of JSME were applied to the nuclear safety regulations of the existing fleets of the domestic and newly built NPPs after a technical assessment was conducted and endorsed by the regulatory authority. This framework, which is called the “performance-based regulatory rules and application of the codes and standards developed by SDOs”, has been in effect since 2006 in Japan. In regard to this framework, there were concerns that the provision of detailed technical rules by the government regulator had sometimes caused a delay in updating the rules with the latest technical knowledge, and as a result, it was difficult to promptly and flexibly apply the advanced technologies (the detailed technical rules were also within the regulatory rules prior to 2006).

Academic societies, such as JSME, provide an arena where the most excellent scientific knowledge and technologies are gathered, and where utilizing this excellent expertise to enhance nuclear safety is one of the most important roles. By developing, updating, and maintaining the codes and standards for the detailed technical rules by the academic societies as SDOs, the regulatory body is able to focus its regulatory resources on truly important safety regulatory issues. Considering these aspects, it is recognized that the utilization of the codes and standards of the SDOs and third party certifications, such as a design conformity assessment and weld inspection, should be further promoted, which will lead to the optimal allocation of limited regulatory resources to important safety issues.

3.3 Viewpoint 3: The adequacy and issues of the current seismic design technology

As was mentioned earlier, although earthquake ground motions that exceeded the design basis were observed at some sites (e.g., Onagawa), failure, damage or loss of the functionality of the structures, systems and components important to safety have not been reported, except for the Fukushima Daiichi NPP units, where inspections are not able to be conducted due to the accident. In this context, it can be considered that the current seismic design technology for mechanical components and the related seismic design rules adequately functioned.

Furthermore, there were cases in the past in which NPPs were hit by earthquakes whose peak ground motion exceeded the design basis. Typical examples include the Onagawa station that was hit by the Miyagiken-oki earthquake in August 2005, and the Kasiwazaki-Kariwa station that was hit by the Niigataken Chuetus-oki earthquake in July 2007. In these cases, the plants were safely shutdown soon after the earthquakes were detected. Detailed inspections

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were conducted in each plant unit after each earthquake and it was determined that there were no significant damages and that the structures, systems, and components important to safety did not malfunction. The current seismic design technology seems to have also functioned. The seismic analysis methods were sufficiently conservative (e.g., elastic analysis with a low damping factor) and the seismic design criteria, such as the allowable stress limits, were also sufficiently conservative.

Onagawa station was also hit by the 2011 Great East Japan Earthquake, and to visually investigate the impact of the earthquake and tsunami on the facility, an IAEA team of international experts conducted a survey to contribute their results to an IAEA database, which is being compiled by the International Seismic Safety Centre (ISSC), to provide knowledge about the impact of the external hazards on nuclear power plants to the member states (Fig. 3). According to the team's initial report (IAEA, 2012),

> *"Onagawa, facing the Pacific Ocean on Japan's north-east coast, was the nuclear power plant closest to the epicenter of the 11 March 2011 magnitude 9.0 earthquake that struck Japan and resulted in a devastating tsunami. The plant experienced very high levels of ground shaking—among the strongest of any plant affected by the earthquake—and some flooding from the tsunami that followed, but was able to shut down safely. In its draft report the team said that 'the structural elements of the NPS were remarkably undamaged given the magnitude of ground motion experienced and the duration and size of this great earthquake'."*

This may also imply that the current seismic design technology is adequate.

Fig. 3 Field investigation by IAEA (upper left and right: survey of the flooded areas; lower left: survey at the front of flood barrier (tsunami survey))

However, these plants were not allowed to restart until after some time. A major reason for this delay is attributed to the fact that the methods and procedures for the structural integrity evaluation of the structures, systems, and components affected by the seismic loads exceeding the design basis were not well prepared. There were also no established restarting criteria.

Among the important issues that need to be investigated in the area of seismic design, efforts need to be made towards the understanding of the ultimate structural integrity limits of components and piping and the ultimate functional limits of active components against the seismic load. The Fukushima Daiichi NPP accident showed that, inherently, there is a very large uncertainty in the severity level of natural hazards and the importance of risk management against natural hazards whose severity levels exceed the design basis. In this context, designing the structure, systems, and components against the seismic load of the design basis may not be sufficient. It is important to conduct an evaluation regarding to what extent the structural and functional integrity of the structures, systems, and components can be maintained against a seismic load exceeding the design basis (real design margin). For this purpose,

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standard methodologies for an ultimate structural integrity evaluation in the exceeding range of the design basis and corresponding criteria are expected to be developed. Inelastic (elastic-plastic) seismic response analysis methodologies and failure criteria based on strain are needed.

Rules and criteria should also be developed and provided to judge when a NPP that has experienced an earthquake exceeding its design basis can be restarted. In order to judge whether the earthquake that hit a plant exceeded the design basis or not, further studies on the governing earthquake ground motion parameters (indices) that have good correlation with the structural damages and failures are needed. The current seismic design methodology is based on the design against seismic inertia force, and therefore the maximum acceleration is used as the primary indicator for the level of an earthquake. However, the maximum acceleration may not always be the contributing measure for the physical value that relates to the structural failure. Conversely, there are cases in which the relative displacement induced by an earthquake or the maximum velocity that is related to the energy of an earthquake is the governing parameter for the structural failure. In fact, in the gas industry, the spectral intensity (SI), which is the integral of the response velocity spectrum of an earthquake within a certain period range (Fig. 4a), is used as the major index for the damageability of earthquakes. It is shown that there is a good correlation between the damage and failure of buried gas pipelines and the SI value. The US NRC requires that, according to Regulatory Guide 1.166, evaluations should be made by using both the response spectra and the cumulative absolute velocity (CAV) to judge whether an earthquake that hit a NPP exceeds the design basis or not (US NRC, 1997). Here, the CAV is the integral of the absolute velocity time history of an earthquake ground motion (Fig. 4b). Studies are needed to determine the appropriate parameter that can indicate whether an earthquake can affect NPP facilities from the viewpoint of structural and functional integrity, and while including the applicability of the SI and CAV.

$$SI = \frac{1}{2.4}\int_{0.1}^{2.5} S_V dT \ \ (h = 0.2)$$

Sv : Velocity response spectrum
T : Natural period (s)
h : Damping factor

$$CAV = CAV_i + \int_{i}^{i+1} |a(t)| dt$$

Summation of the time integral of the absolute acceleration in the range where the maximum value of acceleration exceeds the threshold.

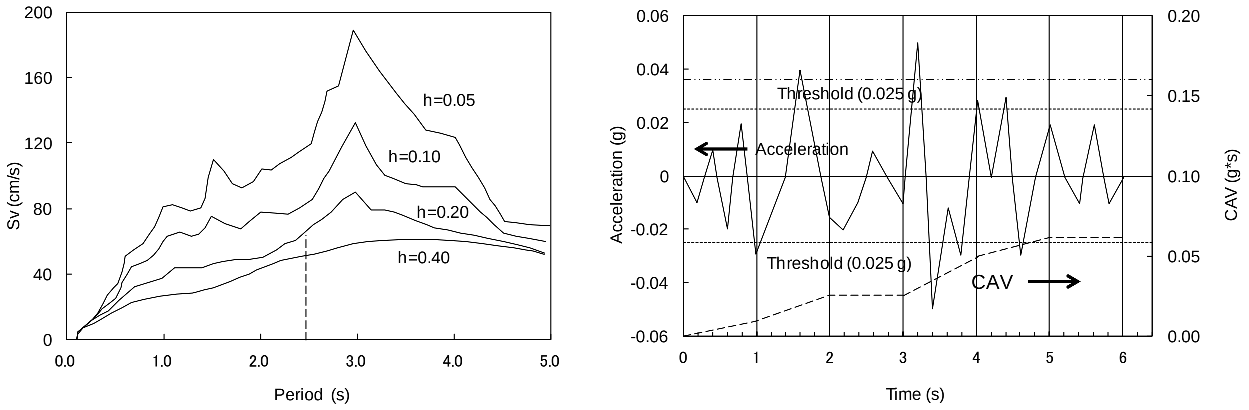

Fig. 4 Definitions of the (a) SI value and (b) CAV

4. Recommendations

WG6 conducted an extensive survey and investigation based three differing viewpoints, as outlined above. Through the investigation and discussion among the members, several issues, perceptions, and concerns were raised relating to these viewpoints. Based on these insights, WG6 developed five recommendations as given below. These recommendations are towards the regulators, the owners and the industry, general public, and codes and standards development bodies.

Recommendation 1: Improving nuclear safety based on the concept of risk

Improving the safety of NPPs should be pursued constantly and continuously. To make these activities effective,

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introducing and applying the concept of risk is essential for safety evaluations or as countermeasures to improve safety.

For the issues regarding nuclear safety, the attempts to distinguish between safe or dangerous should stop, and calm and philosophical discussions should be made based on scientific knowledge and facts. JSME is expected to deliver a proper message to society and the public with such a perspective.

Recommendation 2: Expectations for nuclear safety regulations

As for nuclear safety regulations, a systematically constructed set of safety requirements should be implemented with scientific rationality based on the state-of-the-art technology and high level perspectives. For this purpose, utilization of the codes and standards of the SDO and third party certifications, such as design conformity assessments and weld inspections, should be further promoted, which will lead to the optimal allocation of limited regulatory resources to important safety issues. Transparency, objectivity, and fairness are required for safety regulations and thus the systematic documentation of regulatory requirements open to the public is recommended. It is also important to create an atmosphere where open and flat discussions can be made about the safety regulations among the stakeholders, regulators, owners, industry, SDOs, and the academia. Awareness on the international trend of nuclear safety regulations is also essential to update domestic regulations.

Recommendation 3: Owners with a sense of safety

The owners of NPPs should have a basic attitude with a sense of safety and should continuously revisit and update their design basis and safety evaluation of their plants, including external hazards, with the latest scientific and engineering knowledge. In addition, after carefully considering the large uncertainty of the severity of external events, the owners should be well prepared for situations that exceed the design basis. The owners should be aware that the regulatory requirements and the requirement by the codes and standards are the minimum safety requirements, and should maintain a basic attitude to continuously improve the safety of their plants, since they are directly responsible for the safety of their facilities.

Recommendation 4: Issues with codes and standards

The SDOs should refer to the lessons learned from the Fukushima Daiichi NPP accident and the subsequent researches and investigations, and after collaborating with each other, they should clarify the new role and perspective of the codes and standards to prevent and mitigate SAs caused by severe external events. The codes and standards that are required for this need to be developed in a timely manner and with high priority.

When developing the codes and standards of the SDOs, it is important that the SDOs cooperate and collaborate amongst each other and hold dialogues with a regulator, while recognizing the complementary relationship between the performance-based regulation and technical rules of the codes and standards.

Recommendation 5: Seismic design

The ultimate failure limit of structures and components against seismic loads need to be understood, and the safety margin that is currently accepted in the seismic design methodology and criteria should be quantified. Based on this class of knowledge, structural evaluation methodologies and corresponding failure criteria against severe seismic loads exceeding the design basis should be developed by applying state-of-the-art knowledge and technologies. Systematic and consolidated studies with such a perspective are expected to be conducted.

Rules and criteria should also be developed and provided to judge whether a NPP that has experienced an earthquake exceeding its design basis should be restarted. In this regard, studies are needed to determine the appropriate parameter that can indicate whether an earthquake can affect NPP facilities from the viewpoint of structural and functional integrity.

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This chapter of the report is dedicated to the WG members for their information survey, in-depth discussions, and insights. The author of this chapter expresses his sincere appreciation to the following devoted WG members: Mr. Koji Dozaki, JAPC; Dr. Hitoshi Kaguchi, MHI; Mr. Shoji Kotake, JAPC; Mr. Koji Koyama, MHI; Prof. Shinsuke Sakai,

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