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SEISMIC DESIGN STANDARDS AND CALCULATIONAL METHODS IN THE UNITED STATES AND JAPAN: A SUMMARY

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ABSTRACT

Over the years, a number of nuclear power plants (NPPs) have experienced earthquake shaking and some have experienced shaking in multiple earthquakes. These NPPs have principally been in Japan. After the Niigataken Chūetsu-Oki (NCO) earthquake impacted the Kashiwazaki-Kariwa Nuclear Power Plant (KKNPP) located in the Niigata prefecture of Japan, the United States (U.S.) Nuclear Regulatory Commission (NRC) identified a need to better understand the seismic performance of Japanese NPPs with respect to the codes, standards, and guidance documents under which the NPP designs were made. The NCO earthquake ground motions at the KKNPP site significantly exceeded the DBE ground motions, but KKNPP survived with no significant damage or malfunction to safety-related structures, systems, and components (SSCs). Specifically, NRC wanted to understand similarities and differences in the seismic design processes of Japan and those of the U.S. The subsequent effort sponsored by the NRC resulted in a recently published NUREG/CR report that summarizes information on current and past U.S. and Japanese seismic design standards, calculational methods, and load combinations used for the design of new and currently operating NPPs.

This paper summarizes the subjects covered in the NRC study, including, design basis earthquake (DBE) ground motion definitions; soil-structure interaction analysis; structure analysis and design; equipment, components, and distribution systems design and qualification; and selected operational procedures, such as automatic scram systems. Both design basis earthquake (DBE) and beyond design basis earthquake (BDBE) considerations are presented, discussed, and compared. BDBE issues include common cause nature of seismic excitations, potential cliff edge effects, and the effect of BDBE on multiple units at the site.

Multiple time frames are discussed in this paper. For Japan, the time frames are separated into pre-2006 and post-2006, reflecting changes in requirements and practice due to the lessons learned from the observations of the performance of NPPs in Japan subjected to earlier BDBE ground motions. For the U.S., the time frames considered were pre-2007 and post-2007 reflecting significant changes in U.S. NRC Regulatory Guides and the Standard Review Plan (NUREG-0800).

The new NUREG/CR report only discusses codes, standards, and guidance in place before the 2011 Fukushima accident (i.e., those used for the design of operating NPPs). Therefore, this paper provides brief descriptions of updates to the requirements and procedures for seismic analysis in the U.S., taking into account Fukushima Dai-ichi lessons learned.

INTRODUCTION

Worldwide, the United States (U.S.) and Japan are two of the countries with the most developed seismic design standards and calculational methods for nuclear power plants (NPPs). Thousands of person-years of effort over five decades have been devoted to developing these standards and methods for nuclear facilities, which have evolved over time. As a result, the codes, standards and guidance available for seismic design and risk assessment is considered relatively mature.

In addition, Japan has experienced several earthquakes that have directly affected NPPs with ground motions exceeding the design-basis earthquake (DBE) ground motions (denoted as S_1 or S_2), and in some cases significantly exceeding the S_2. In these cases, minimal or no damage from strong shaking of safety-related structures, systems, and components (SSCs) was observed. Even for the 2011 Tōhoku[1] earthquake, evidence suggests that damage of safety-related SSCs caused by strong shaking was not significant at any of the affected NPPs[2].

Similarly, in the U.S., a few events have occurred affecting NPPs with the most notable being the 2011 Mineral, Virginia, earthquake that caused ground motions at the North Anna Power Station (NAPS) that exceeded the plant's Safe Shutdown Earthquake (SSE) ground motion over a large frequency range. As with the Japanese NPPs, minimal or no damage to safety-related SSCs was observed at NAPS. Although the ground motion experienced at these U.S [Bhargava et al. 2015]. sites exceeded the site-specific Operating Basis Earthquake (OBE) ground motion or SSE ground motion, the overall levels of shaking have been much lower than those experienced by NPPs in Japan. The implication from the aggregate of these experiences is that seismic design of NPPs in the U.S. and Japan has been demonstrated to be effective when tested by actual earthquake shaking.

Field experiments and laboratory testing in the U.S., Japan, and Taiwan over this same period have also illuminated aspects of the standards and some of the conservatism contained therein. The advances in guidance in computational methods, experience in earthquakes, and knowledge from field and laboratory testing have all led to the evolution of seismic design standards and calculational methods over the last five decades.

In this context, a review of the codes, standards and guidance used for design in the U.S. and Japan over time is appropriate and meaningful for understanding the performance and seismic capacity of operating NPPs. The U.S., Japan, and other countries can learn from these experiences and introduce appropriate changes to their current seismic design standards and calculational methods. It is important to note, however, that seismic hazard assessment and seismic design must always be considered in the context of the seismo-tectonic environment in which the country exists. Nearly the entire country of Japan is situated in an area of high seismicity and both subduction and active crustal mechanisms are at work. The U.S. by contrast, is highly varied with seismicity rates that range from very high to very low across its territory and with nearly every seismo-tectonic environment found within its borders. It should be expected, therefore, that differences in assessment, design, and regulatory approaches exist between the two countries.

MOTIVATION FOR THE EVALUATION PROJECT

On July 16, 2007, an earthquake occurred near the world's largest nuclear plant, the Kashiwazaki-Kariwa Nuclear Power Plant (KKNPP) located in the Niigata prefecture of Japan. The Niigataken Chūetsu-Oki (NCO) earthquake ground motion at the site exceeded the plant's DBE ground motions by a significant amount[3] and caused an extended period of shutdown of all seven reactors at the plant. The KKNPP units generally performed well given the exceedance of the DBE ground motion that they experienced. All the reactors at the KKNPP had restarted prior to the 2011 Tōhoku earthquake after

[1] The Tōhoku earthquake is formally called the Tohoku-chiho Taiheiyo-oki earthquake by the U.S. Geological Survey.

[2] Investigation and evaluation of the NPPs impacted by the Tōhoku event indicates that the subsequent tsunami was responsible for damage to safety-related SSCs.

[3] In Japan, the seismic design basis of safety-related SSCs is the envelope of the dynamic responses caused by the DBE ground motions and applied equivalent static loading conditions. The NCO earthquake ground motion exceeded the DBE ground motions by a significant amount, but the induced loading environment of the NCO earthquake on SSCs may not have exceeded the equivalent static loading conditions to the same degree.

following a regulatory process to re-evaluate the seismic safety of the plants. The process is described in [Johnson et al. 2017].[4]

As a result of the experiences at the KKNPP in 2007, the U.S. Nuclear Regulatory Commission (NRC) identified a need to understand and document the lessons learned from this earthquake. NRC staff has been working for a decade since the NCO earthquake to collect lessons learned, to better understand differences between U.S. and Japanese design approaches, and to obtain quantitative data on plant performance during the event. The complete details are contained in NUREG/CR-7230 [Johnson et al. 2017], which was recently published by NRC.

This paper summarizes the key findings of the comparison of U.S. and Japan seismic design standards and calculational methods as described in [Johnson et al. 2017]. The evaluation and documentation effort was part of a multi-phase project with the objective to develop, analyze, and document the impact and lessons learned of the July 16, 2007 earthquake that affected the KKNPP in the Niigata prefecture of Japan. This paper summarizes the similarities and differences in U.S. and Japanese seismic design standards and calculational methods, as well as load combinations. More detailed recommendations are contained in [Johnson et al. 2017]. This paper also discusses U.S. re-evaluation activities undertaken in response to the 2011 events in Japan.

TIME FRAMES

Because both U.S. and Japanese standards have changed over time, specific time frames of interest are consistently used throughout [Johnson et al. 2017] and in this paper. The time frames of applicability of the seismic design standards and methodologies are described below. The project described in [Johnson et al. 2017] covered the time period up to approximately 2011.

For the U.S., there are three time frames of interest when considering the seismic design-basis ground motion: 1973 to 1996, before the NRC issued Regulatory Guide (RG) 1.165, "Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion," [NRC 1997], and enacted Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix S ("Earthquake Engineering Criteria for Nuclear Power Plants") in 1997; 1997 to 2007; and Post-2007, when NRC RG 1.208, [NRC 2007] was issued and numerous changes to other RGs and the Standard Review Plan (NUREG-0800) took effect. "Pre-1997" refers to the seismic analysis and design procedures and requirements implemented for operating NPPs designed and constructed during this (pre-1997) period with emphasis on standards in place after about 1980. "Post-2007" refers to those standards currently in place and being applied to the next generation of NPPs in the U.S. The Post-2007 standards are characterized by the use of Certified Designs. The period of 1997 to 2007 is only important with respect to the methods for developing the seismic design-basis earthquake ground motions. For purposes of this review, the periods of Pre-2007 and Post-2007 are generally meant to represent operating NPP designs (Pre-2007) and new NPP designs (Post-2007). A fourth time frame is "Post-Fukushima - 2011", which is discussed later in this paper but is not covered in detail in [Johnson et al. 2017] as a result of the timing of completion of the project.

For Japan, the time frames of interest are denoted "Pre-2006" and "Post-2006." "Pre-2006" refers to the seismic analysis and design procedures and requirements implemented for NPPs designed prior to 2006. In general, codes and standards of Japan are only available in the Japanese language. However, NRC funded the translation of the Japan Electric Association (JEA) *Technical Guidelines for Aseismic Design of Nuclear Power Plants, Translation of JEAG 4601-1987* [NRC 1994], which was essential for the effort of [Johnson et al. 2017] and summarized herein. "Post-2006" refers to the changes in criteria as stated by Japan's Nuclear Safety Commission (NSC) (on September 19, 2006) [NSC 2006]. The new

4 As of June 2017, five nuclear power units in Japan are in operation (Sendai NPP Units 1 and 2 operated by the Kyūshū Electric Power Company; Takahama NPP Units 3 and 4 operated by Kansai Electric Power Company; Ikata NPP Unit 3 operated by Shikoku Electric Power Company). The remaining units in Japan are in suspension (some listed as undergoing periodic inspection) or in various stages of decommissioning.

criteria are to be considered in the design of new facilities and the reevaluation of existing facilities. Prior to the occurrence of the Tōhoku earthquake and tsunami and the Fukushima Dai-ichi NPP accident, JEA developed a revised set of codes [JEA 2008a] and guidance [JEA 2008b] to address the revisions to the seismic design criteria of NSC. These codes and guidance are in the Japanese language, although, translation into English is slowly proceeding. A new nuclear regulation authority (NRA) was established in Japan in September 2012 and the re-evaluation of existing NPPs and the design of new NPPs are under the jurisdiction of the NRA.

HISTORICAL PERSPECTIVE OF SEISMIC DESIGN

Historically, both the U.S. and the Japanese practices have used deterministic approaches in all aspects of the seismic analysis, design, and regulation. However, over the years, and particularly in connection with the new reactors, the U.S. practices have moved significantly toward a more performance-based, risk-informed regulatory framework. The Japanese practice has recently begun to look at very limited aspects of risk-informed considerations. However, Japan's practice is still basically deterministic. The following describes how the risk-informed aspects are currently being used and provides a brief comparison of the two practices.

Japan introduced the "residual risk" concept in 2006 [NSC 2006]; however, the approach taken in seismic hazard assessment and seismic design is still inherently deterministic in nature. As in most deterministic practices, the focus in Japan is on assuring that a high level of conservatism exists at every step in the design process, such that Japanese NPPs have significant margin above the DBE ground motion used. There is an assumption that the DBE ground motion used is sufficiently rare for the site of interest.

By contrast, the U.S. uses a mixed approach. For existing operating NPPs, meeting the NRC's seismic-safety regulations still means meeting a complex set of deterministic regulations that are demonstrated by deterministic evaluations. This includes how the DBE (or SSE) ground motion that is still in use was selected, although a probabilistic reevaluation of that SSE is now under way for all existing plants. For new designs, the same set of deterministic regulations, demonstrated by deterministic analyses, is still in place, except that the selection of the SSE for a new plant must follow a probabilistic seismic-hazard approach tied to a specific annual frequency of exceedance. What is new is that the regulatory evaluation of the design, which uses deterministic criteria similar to those used for the existing operating plants, is supplemented by a risk-informed and performance-based evaluation of the seismic adequacy of the plant-as-a-whole. This evaluation requires that seismic probabilistic risk assessment (SPRA) be completed prior to loading of fuel[6]. This evaluation provides a clearer way to understand conservatisms inherent in the design and provides an opportunity to risk-inform the entire design practice.

These two philosophies are so different that the relative conservatism of the outcomes of the two approaches cannot be known *a priori*. The conservatism of any regulatory framework for an NPP can only be assessed through a comparison of the true response of the NPP against the true hazard at its site. A SPRA provides a means to evaluate the conservatisms.

Although for new plants the U.S. relies in part on a performance-based, risk-informed framework, the process of seismic analysis, seismic design, and seismic qualification of SSCs is deterministic by choice and the practicality of design. Deterministic procedures (methods and parameter values) are developed and evaluated to assure that the implementation of seismic analysis, seismic design, and seismic qualification for SSCs leads to SSC seismic performance that meets the risk guidelines.

A comparison of the results of the deterministic seismic analysis, design, and qualification process step-by-step is less satisfying than a comparison of SPRA results; however, it is still a valuable exercise. The end result is a comparison of the design loading conditions for SSCs, including loads, in-structure response spectra for qualification of equipment, components, and distribution systems, and other design quantities. This comparison could be conditional on the DBE or include the effects of the DBE. The end

[6] The Standard ASME/ANS RA-Sb-2012 [ASME/ANS 2012] is used as the standard-of-practice for SPRA studies in the U.S.

result quantifies the degree of relative conservatism introduced in various steps of the seismic analysis chain in U.S. procedures compared to the procedures of Japan. The end result could also be interpreted in the risk framework as a surrogate for core damage frequency or large early release frequency, such as onset of inelastic deformation. This is a very valuable and practical assessment process recognizing the multi-disciplinary nature of the process.

For the above reasons, this paper summarizes the results reported in detail in [Johnson et al. 2017]. The comparisons are framed to provide clarity and insights into the similarities and differences of the two regulatory approaches and frameworks.

COMPARISON OF STANDARDS AND COMPUTATIONAL METHODS THROUGH 2011

Based on the assessments performed and reported in [Johnson et al. 2017], the likely relative conservatisms for the operating reactors are summarized as follows:

Elements of the seismic analysis, design, and qualification processes for which Japan is more conservative than the U.S. (Japan Pre-2006 compared to U.S. Pre-2007)

- Structure damping values used in linear analysis are lower in Japan than in the U.S.
- Damping values for some of equipment, components, and piping are specified to be lower in Japan than in the U.S.
- Implementation, testing, and maintenance of modern seismic instrumentation systems are required in Japan and not in the U.S.
- Proof testing for seismic performance is conducted in Japan and the US. However, seismic fragility testing has been performed in Japan for fuel assemblies, CRD assemblies and selected large mechanical equipment and electrical equipment that demonstrates seismic margin beyond the design basis [Kennedy et al., 2011].

Elements of the seismic analysis, design, and qualification processes for which the U.S. is more conservative than Japan (Japan Pre-2006 compared to U.S. Pre-2007)

- All safety-related SSCs (Seismic Category I) are designed to SSE ground motion (by comparison, under the Japan criteria safety-related equipment is designed to S_1 and only a subset is assessed for functionality under the S_2 ground motion).
- Soil-structure-interaction (SSI) analyses to determine the structure response are performed for soil and soft rock sites. The structure response used for design, including as input to subsystems, is defined as the envelope of the responses for three soil profile cases in the U.S.; only a best estimate soil profile is considered in Japan.
- In-structure response spectra are developed with peaks broadened ±15 percent in the U.S. as compared to ±10 percent in Japan.
- Three components of earthquake ground motion are considered simultaneously in the SSI analyses.
- All combinations of loss-of-coolant accident loadings are combined with the SSE ground motions.
- Equipment qualification testing is required.
- Beyond design-basis ground motion evaluations are required for new plants; acceptance criteria for new plants, plant-level High Confidence of Low Probability of Failure (HCLPF) values must be greater than 1.67 times the design-basis ground motion. A SPRA is required prior to loading of fuel in new reactors.

- For existing plants, similar (though early vintage) beyond design-basis assessment procedures were previously implemented and assessments performed during the Individual Plant Examination of External Events (IPEEE) program in the latter part of the 1990s. These assessment approaches subsequently matured and expanded into a number of tools that now exist for a variety of design, assessment, and operational uses. The current risk-informed performance-based operational and regulatory framework is a direct result of that early work and the lessons learned. The NRC is now in the process of re-assessing the seismic hazard for all U.S. operating reactors and will use the beyond design-basis tools for those NPPs whose new estimated ground motion exceeds the original design.

Elements of the seismic analysis, design, and qualification processes for which the relative conservatism is currently unknown (Japan Pre-2006 compared to U.S. Pre-2007)

- Probability of occurrence of peak values of design ground motion (peak ground acceleration (PGA), peak ground velocity, and peak ground displacement) for the operating reactors is currently unknown, although analyses to determine this information are underway.
- In Japan, the maximum of static and dynamic loads are used for design (e.g., a static loading of 0.6g for structures and 0.72g for equipment, piping, etc.).
- In the U.S., the minimum design ground motion at foundation level in the free-field has a minimum PGA of 0.1g anchoring a spectral shape appropriate for foundation level (outcrop or in-column motion). Most commonly a RG 1.60 [NRC 2014][7] spectrum is used.
- Automatic Seismic Trip Systems are required for all NPP units in Japan.

Elements of the seismic analysis, design, and qualification processes that are favorable for Japan in reducing uncertainty in dynamic behavior of SSCs and verifying seismic capacity

- Extensive testing program to verify behavior of soil-structure systems (SSI phenomena and methods of analysis).
- Extensive testing program to define the stiffness, nonlinear behavior, and capacity of structure elements, e.g., shear walls.
- Extensive testing program to define the behavior of equipment, piping, and other components.

ACTIVITIES IN THE U.S. POST FUKUSHIMA – 2011

Activities in the U.S. in response to the Fukushima accident have focused on short- to medium-term reevaluation of NPPs and long-term research, with SPRA and risk considerations playing an increasingly important role. Shortly after the Fukushima accident, the NRC established a task force of senior NRC staff experts to review the events and their causes (both the immediate causes and the underlying causes), and to make recommendations to the NRC. The so-called "NRC Near Term Task Force (NTTF)" gathered information in part by meeting with a large number of nuclear industry experts both in the U.S. and internationally; with NRC staff experts; with U.S. Government officials in many different agencies and the U.S. Congress; and with a large number of private citizens, public interest groups, and state and local agencies. The NTTF published their report [NRC 2011a] on 12 July 2011, only 4 months after the accident. Arguably, the report's most impactful content is a set of recommendations for NRC actions in various technical and regulatory areas.

[7] The NRC recently published Revision 2 of this RG. However, Revision 1, which was published in 1973, was the version that was widely used for operating reactors in the U.S.

Within a very short time, the NRC Commissioners reviewed the NTTF report and issued their own report containing the agency's action plan [NRC 2011b, NRC 2011c]. The NTTF recommendations were prioritized by the NRC Commissioners into Tiers 1, 2, and 3, which were based on both the potential impact of the recommended activities and the realistic timeline associated with implementation of the recommendation. Tier 1 recommendations were to be addressed as soon as feasible. Tier 2 recommendations were to be addressed on a somewhat more extended schedule, usually because certain technical work was necessary before a specific action plan could be developed to implement the recommendation, but also in some cases because the implementation depended on the outcome of a Tier 1 activity. Tier 3 recommendations were to be addressed on a still more extended schedule, usually because considerable research work or policy development was required before a decision could be made on what to do.

For external hazards, the NTTF recommendations are numbered:

- 2.1- Re-evaluate seismic and flooding hazards at the site (Tier 1)
- 2.2- Review seismic and flooding hazards on a 10-year cycle (Tier 2)
- 2.3 Perform seismic and flood protection walkdowns and assess compliance with the design bases (Tier 1)

The approaches to addressing recommendations 2.1 and 2.3 for seismic hazards were developed through a combined effort of the NRC and the U.S. nuclear industry and resulted in agreed upon guidelines described in a report entitled, "Seismic Evaluation Guidance: Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima NTTF Recommendation 2.1 Seismic." [EPRI 2012]. The report, which was published by the Electric Power Research Institute, was endorsed by NRC in 2013 with additional guidance on four issues: (1) the use of the IPEEE submittals for screening purposes; (2) development of foundation input response spectra (FIRS) consistent with the site response used in the development of the site-specific ground motion response spectrum (GMRS); (3) updating the seismic source models; and (4) development of the site response).

At the time of the 2011 earthquake, the NRC has an ongoing project, known as Generic Issue (GI-199), that was focused on re-evaluation of the seismic hazard and seismic risk of NPPs in the central and eastern U.S. [NRC 2010a, 2010b]. The work on Recommendation 2.1 for seismic hazard subsumed and built upon the work done by the GI-199 project and expanded it to all operating U.S. NPPs. The first step in the 2.1 process for seismic (re-evaluation of seismic hazards) was completed on March 15, 2014 as Licensees submitted revised GMRS for comparison with the DBE ground motions for each of the NPP sites. Of the 63 U.S. NPP sites, 32 are required to perform additional risk analysis and other activities. Additional "risk analysis" means SPRA or, for some specific situations in which the exceedance of the GMRS above the DBE are small, seismic margins assessment. Nearly all NPPs required to do further risk analysis committed to perform SPRAs, regardless of the exceedance levels.

During the NRC-industry interactions leading up to the publication of the SPID, and then afterwards when detailed schedules were being discussed, it became clear to both the NRC and the industry that the original completion schedules for the seismic risk analyses would be very difficult (if not impossible) to meet. The resolution to the dilemma was to establish a new program, called the "Expedited Seismic Evaluation Process" or ESEP [EPRI 2013]. The ESEP concept and process was developed by industry and endorsed by the NRC. The ESEP requires additional evaluations (and, if necessary, plant modifications) for each operating NPP for which a seismic risk analysis was ultimately required. The goal of the ESEP is to provide the desired *early* high assurance that the NPP can cope with BDBE ground motions that might produce an extended loss of offsite power or an extended loss of ultimate heat sink. The word "early" in the previous sentence became a requirement to perform this focused assessment before 31 December 2014. The trade-off is that the more in-depth seismic risk analyses were to be completed on a more extended schedule than originally proposed. However, realistically, the extended schedule is more consistent with the overall ability of the U.S. nuclear-power industry and its supporting contractors to perform the large number of plant-specific seismic risk analyses required.

The ESEP assessment focuses on a seismic capacity evaluation of the onsite equipment that the NPP is relying on as part of its new "FLEX" capability. FLEX is a new industry program, endorsed by the NRC, to provide each NPP with augmented capabilities for coping with certain beyond-design-basis external event disruptions of offsite power or of the ultimate heat sink. FLEX is described in [NEI 2012]. The ESEP scope covers a limited number of components and systems that, if they have sufficiently strong seismic capacities, will assure that the plant can withstand safely the challenges at issue: challenges comprising an extended loss of power or an extended loss of ultimate heat sink from a BDBE. The figure-of-merit is that each item under review must demonstrate a so-called HCLPF capacity that is at or above the design-basis earthquake level.

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The NRC staff, in turn, is currently reviewing the industry ESEP reports, which were all submitted by the deadline of 31 December 2014. The revised schedule for licensees' submittals of SPRA results includes several phases, expected to last through 31 December 2019.

CONCLUSION

Seismic analysis, seismic design, and seismic qualification activities are multidisciplinary in nature and many elements comprise the analysis, design, qualification, and construction processes. Evaluating the relative conservatism in the individual elements is valuable but not necessarily indicative of the overall conservatism in one process compared to the other. One way to quantify differences in implementation of the seismic analysis, design, and qualification processes for the U.S. and Japan is to perform a pilot study, whereby a given design is compared step-by-step to quantify more or less conservatism in each step and in the final design. An abbreviated effort of this type was performed for the International Atomic Energy Agency Extra-Budgetary Program Kashiwazaki-Kariwa Research Initiative for Seismic Margin Assessment (KARISMA) project, [IAEA 2013]. Participation in the KARISMA project was an additional phase of this NRC research project. Information gained in the KARISMA project has been incorporated into this paper, as appropriate.

These studies reported herein and in [Johnson et al. 2017] do not, and cannot, provide a strict "apples to apples" comparison of each step in the process.

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- Mr. Noriaki Tomura, Committee Member (Japan Atomic Power Company)
- Mr. Rokuro Endo, Committee Member (JANSI)

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[8] JNES has been incorporated into the new Nuclear Regulatory Authority of Japan.
[9] JANSI was formerly known as Japan Nuclear Technology Institute (also known as JANTI).