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Seismic Instrumentation System and Its Use in Post-earthquake Decision Making at Nuclear Power Plants

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SEISMIC INSTRUMENTATION SYSTEM AND ITS USE IN POST-EARTHQUAKE DECISION MAKING AT NUCLEAR POWER PLANTS

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VIENNA, 2021

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Printed by the IAEA in Austria
April 2021

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Seismic instrumentation system and its use in post-earthquake decision making at nuclear power plants / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2021. | Series: IAEA TECDOC series, ISSN 1011–4289 ; no. 1956 | Includes bibliographical references.
Identifiers: IAEAL 21-01408 | ISBN 978-92-0-107721-9 (paperback : alk. paper) |
ISBN 978-92-0-107621-2 (pdf)
Subjects: LCSH: Nuclear power plants — Earthquake effects. | Nuclear power plants — Safety measures. | Seismology — Instruments. | Earthquake hazard analysis.

FOREWORD

Over the past decade, many nuclear power plants have experienced relatively strong ground motions caused by earthquakes. Several of these plants have even been subjected to motions that exceeded their design or evaluation bases.

In most cases, no significant damage directly related to the earthquake motion itself was identified in these nuclear units. In a limited number of cases, upgrades were implemented after the earthquake, corresponding either to a redefinition of the design basis earthquake or to new requirements to sustain a beyond design basis earthquake.

In all cases, during the post-earthquake activities, the assessment of potentially hidden damage proved to be particularly challenging. Extensive and time-consuming inspections were carried out to check the plant's integrity and its ability to resume safe operation. For this purpose, an important input is the characterization of the seismic motion actually experienced at the nuclear plant. Such a characterization enables assessors to know whether the seismic motion experienced at the nuclear power plant exceeded the levels considered in the design and, where those levels were exceeded, to determine the damage potential of the motion.

In this sense, seismic instrumentation systems are important elements for plant safety. They provide crucial information for assessing whether a plant can be safely restarted after a shutdown caused by an earthquake. Consequently, in recent years, there has been renewed interest in seismic instrumentation for nuclear power plants and in the development of damage-indicating parameters to evaluate the motions recorded by them.

The purpose of this publication is to provide information on recent experience in the implementation of seismic instrumentation systems and on the utilization of earthquake records to assess the damage potential of the motion. The publication is also expected to be an important reference in developing more reliable and effective post-earthquake actions and procedures.

This publication complements the IAEA safety standards as a technical supporting publication relative to seismic safety of new and existing nuclear installations.

The contributions of all individuals involved in the drafting and review of this publication are greatly appreciated. The IAEA wishes to thank J.D. Stevenson (United States of America) and F. Beltran (Spain) for their contribution to the drafting and review of this publication. The IAEA officer responsible for this publication was N. Stoeva of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Recent developments in seismic instrumentation (seismometers) and the progress in recorded data processing capabilities have resulted in a higher utilization of seismic instrumentation systems for earthquake preparedness in nuclear installations.

Seismic motion waves propagate through a variety of rock and soil layers from the hypocentre of the earthquake to the point of observation, with a high dependence on geological features. The seismic waveforms that reach a particular site are extremely complex and their impact on structures, systems and components is not uniform. Actions taken at an affected nuclear power plant during and after an earthquake depend on the recorded seismic motion and the immediate processing of the records at the plant.

Seismic instrumentation systems play an important role not only in post-earthquake integrity assessment using the recorded seismic motions, but also for during-earthquake actions required from operators and triggered by alarms activated by such systems. In some Member States, the seismic instrumentation system may start a mechanism that automatically shuts down the reactor, depending on the severity of the observed earthquake, without operator intervention. Even where seismic automatic reactor shutdown is not performed, it is often required to carry out real time analysis of recorded earthquake motions and use the results as a basis for operators to proceed to manual reactor shutdown.

Related with the topic of the present publication, IAEA Safety Report Series No. 66 [1] gives detailed guidance on post-earthquake actions, which is consistent with the general guidance provided by Section 7 of IAEA Safety Standard NS-G-1.6 [2] about seismic instrumentation and actions to be taken after an earthquake.

It is characterized therein that the integrity of a nuclear power plant after an earthquake is to be assessed based on the observation of seismic damage, that is, the results of equipment inspection, and on the severity of the seismic motions recorded with the seismometers installed in the nuclear power plant. The details of specific actions to be taken are decided by the matrix-like combination of both inputs.

This publication complements the IAEA Safety Standard and the IAEA Safety Report mentioned above. It describes the typologies of seismic instrumentation systems and their use thereof for defining the ‘earthquake level’ and the damage indicating parameters (DIPs) that will be used to predict the extent of seismic damage based on the observed earthquake motions.

1.2. OBJECTIVE

The primary goals of this publication are to take account of the development of DIPs based on recently acquired seismic experience data in real earthquakes and, additionally, to gather the experience of Member States in the utilization of seismic instrumentation systems and data processing capabilities in the post-earthquake decision making process. The specific objectives are as follows:

1. To support practical implementation and active use of IAEA guidance on seismic instrumentation in nuclear power plants.

2. To provide detailed technical information with practical examples to assist Member States in implementing the recommendations of Safety Guide NS-G-1.6 [2]

3. To provide guidance on the use of seismic instrumentation systems to perform the pre and post-earthquake actions described in IAEA Safety Report Series No. 66 [1].

1.3. SCOPE

The present publication is intended for nuclear power plants and covers both the description of seismic instrumentation systems and the potential use of the seismic instrumentation records for during- and post-earthquake decisions and actions.

This publication is intended for use by regulatory bodies responsible for establishing regulatory requirements related to seismic safety of nuclear power plants. It may also be used as a tool by organizations directly responsible for the design, operation or maintenance of seismic instrumentation systems.

The following topics are presented:

– Manual versus automatic reactor shutdown method (comparison of the advantages and disadvantages of these two methods);

– State-of-practice regarding seismic instrumentation (equipment information and operational status);

– State-of-practice on DIPs (DIPs currently used); and

Data on damage caused by earthquake motions at nuclear power plants and industrial facilities.

This publication addresses the increasing expectations by Member States for guidance on seismic instrumentation systems and on appropriate DIPs to avoid unnecessary outages after an earthquake.

1.4. STRUCTURE

This publication is organized into three additional main sections and seven annexes.

Section 2 provides the overall picture of a seismic instrumentation system. In IAEA seismic design standards and others, seismic instrumentation systems that need to be installed in nuclear power plants are assigned three major objectives: (1) analyse the characteristics of seismic motions; (2) provide operators with alarms, and (3) automatically shutting down the reactor, if applicable. In Section 2, the requirements and recommendations for a seismic instrumentation system are described and examined for each of these three objectives, based on the past seismic experience and on the information collected from Member States.

Section 3 provides an overview of the seismic intensity scales that are used around the world to characterize earthquake motion severity. ‘Seismic intensity’ is qualitative measure of the severity of an earthquake which is defined by the damage caused by the earthquake at a particular location. In general, seismic intensity scales are based on how the motion affects people, everyday objects, structures and other man-made constructions. Section 3 also describes the issues associated with the application of seismic intensity scales to electrical and mechanical equipment of nuclear power plants and elaborates on the current status of the instrumental seismic intensity used by some Member States, which connects a (qualitative) intensity scale with the records of the observed seismic motions.

In Section 4, four types of DIPs are used for trial calculations with the earthquake motions observed in different Member States. The characteristics of those DIPs and the adequacy of their use are also described in this section.

In addition, this publication contains seven annexes, which give additional information on seismic instrumentation systems and post-earthquake decision making. Annex I presents the questionnaires and data sheets, which were provided to the Member States for collecting information; Annex II and III show the status of seismic data acquisition systems and automatic seismic trip systems in nuclear power plants in some Member States; Annex IV provides the definition of the most common seismic intensity scales; and Annexes V to VII provide examples of DIPs computed for recent earthquakes.

2. SEISMIC INSTRUMENTATION FOR DURING- AND AFTER-EARTHQUAKE ACTIONS

2.1. GENERAL CONSIDERATIONS ON SEISMIC INSTRUMENTATION FOR DECISION MAKING

2.1.1. Seismic instrumentation systems at nuclear power plants

As mentioned in Section 1.1, seismic instrumentation at nuclear power plants is progressively increasing its importance within the assessment of safety of a nuclear power plant struck by an earthquake and in the decisions about the actions to be taken during and after the earthquake. In the United States of America, for example, the Code of Federal Regulations, 10 CFR 100 Appendix A and 10 CFR 50 Appendix S, require the installation of seismic instrumentation systems in connection with the seismic safety of nuclear power plants to determine whether reactor operation can be continued by comparing, shortly after an earthquake, the observed seismic motions with the design basis earthquake, as described below:

> "Required seismic instrumentation: Suitable instrumentation shall be provided so that the seismic response of nuclear power plant features important to safety can be determined promptly to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely and to permit such timely action as may be appropriate." (10 CFR 100 Appendix A, VI (3)).

However, seismic instrumentation systems installed in nuclear power plant sites are more than just a tool to compare the observed seismic motion with the design basis, so as to determine whether reactor operation can be continued. Their objectives and required functions are diversifying. These objectives are specified and classified into the following three categories in IAEA Safety Standard NS-G-1.6 [2]:

(a) "For structural monitoring: to collect data on the dynamic behaviour of structural response of structures, systems and components (SSCs) of the nuclear power plant and to assess the degree of validity of the analytical methods used in the seismic design and qualification of the buildings and equipment.

(b) For seismic monitoring: to provide alarms for alerting operators of the potential need for a plant shutdown depending on post-earthquake inspections.

(c) For automatic scram systems: to provide triggering mechanisms for the automatic shutdown of the plant."

What is important to be noticed here is that it is not required for all nuclear power plants to install seismic instrumentation systems to achieve the three objectives mentioned above. In some Member States objective (c) is not required:

The objective of seismic instrumentation at a nuclear power plant may vary depending on the degree of seismic activity at the site location, on each country's seismic damage experience and on seismic design standards and methods. For example, in a low seismicity country such as France, assessment of seismic hazard is not the purpose of the seismic instrumentation systems. However, in a few sites, owners have dedicated local systems, with a lower trigger setup, for the purpose of research on seismic hazard.

In addition, the history of what is generally referred to as a seismometer goes back a long way in time and various types and styles have been proposed. For example, seismic motion detecting sensors can be divided into electrical and mechanical ones. Measured data is recorded and provided for subsequent analysis or communicated in real time as control signals to be further processed as analogue or digital signals. Thus, the seismic motion detection sensors mechanisms and system configurations are diverse.

Particularly, there is a wide variety of seismic instrumentation systems that have actually been installed at nuclear power plants. This publication classifies the seismic instrumentation systems into two categories based on the IAEA's classification of possible objectives and the function/purpose of the seismic instrumentation system. This publication further provides basic advice about their installation and basic specifications, based on actual experienced in Member States.

The three general seismic instrumentation system classes are as follows:

1. Seismic data acquisition systems (SDAS) Section 2.2

2. Seismic alarm/annunciation systems (SAS) Section 2.3

3. Automatic seismic trip systems (ASTS) Section 2.4

The first seismic instrumentation system class, the SDAS, is installed in accordance with the IAEA's installation objectives. In general, it is used to record acceleration time-histories of earthquake motions and to verify the dynamic properties of structures following an earthquake. It is also used to calculate various DIPs, discussed in the following sections to evaluate structural integrity and to decide on post-earthquake actions based on them.

The second class, the SAS, and the third class, the ASTS, correspond as well to the IAEA's installation objectives and have the function of a so-called 'seismic switch'. They identify the earthquake severity during an earthquake, communicate the information to operators and provide SDAS with start-up (trigger) signals and other signals, as appropriate (e.g. for automatic reactor shutdown). Depending on the system configuration, digital SDAS can also function as SAS and ASTS by adding relay contact functions to it.

These three classes, SDAS, SAS and ASTS, are discussed in sections 2.2, 2.3 and 2.4, respectively. Common topics are presented in the rest of this Section 2.1.

2.1.2. Damage level, earthquake level and post-earthquake actions in IAEA Safety Report Series No. 66

2.1.2.1. Earthquake motion severity and post-earthquake actions

As mentioned earlier, IAEA Safety Report Series No. 66 defines post-earthquake actions (i.e. the 'action level'), taking into consideration the severity of the observed earthquake motions (i.e. the 'earthquake level'), and the extent of equipment damage identified during post-earthquake inspections (i.e. the 'damage level').

The ‘damage level’ has four categories, as listed below, depending on the extent of damage identified mainly by visual inspection during operator walkdowns performed immediately after the earthquake (or during the initial focused inspection in the event of a plant shutdown), and the expanded inspection performed as the need arises [1].

> “Damage Level 1: No significant damage or malfunction to SSCs important to safety and those not important to safety.
>
> Damage Level 2: No significant damage or malfunction to SSCs important to safety. Significant damage or malfunction to SSCs not important to safety (NRPG: not required for power generation).
>
> Damage Level 3: No significant damage or malfunction to SSCs important to safety. Significant damage to or malfunction of SSCs not important to safety (RPG: required for power generation).
>
> Damage Level 4: Significant damage to or malfunction of SSCs important to safety (it is highly likely that SSCs not important to safety will experience significant damage at this damage level).”

In addition, the earthquake motions that have been observed by the seismometers installed in the nuclear power plant are to be compared with the earthquake motions envisaged in seismic design. The IAEA’s seismic design standards specify two levels of design motion: SL-1 and SL-2. These two design levels are used to set the experienced ‘earthquake level’ from among the three levels listed below [1]:

> “Earthquake Level 1: Instrumental records indicate that the earthquake motion is less than or equal to the SL-1 earthquake level.
>
> Earthquake Level 2: Instrumental records indicate that the earthquake motion is greater than the SL-1 earthquake level and less than or equal to the SL-2 earthquake level.
>
> Earthquake Level 3: Instrumental records indicate that the earthquake motion is greater than the SL-2 earthquake level.”

In IAEA Safety Reports Series No. 66 post-earthquake actions are determined by the combination of ‘damage level’ and ‘earthquake level’.

It should be noted that comparison of observed earthquake motions with the design earthquake motions in terms of their impact on equipment may not be straightforward. At first sight, the simplest way would be to compare the recorded peak of acceleration with the maximum acceleration of the earthquake motion considered at the design stage. However, it is well known from past experience that the peak ground (or floor) acceleration of an earthquake motion is a parameter that does not correlate well with seismic damage. If it were possible to predict equipment damage using indicating parameters calculated from motion records, then plant locations to inspect and the inspection methods would be easier to determine.

Consequently, it is important to determine what kind of equipment DIPs to use in order to set the ‘earthquake level’ based on the observed earthquake motions and to determine as well what kind of seismic instrumentation system to install for that purpose, thereby improving the reliability of post-earthquake actions.

2.1.2.2. Damage indicating parameters

At the seismic design stage, the response spectrum of absolute acceleration is often used to calculate the seismic loads acting on equipment due to design earthquake motions. Thus, it is a general practice to calculate the response spectrum using the acceleration records of the observed earthquake motion (in many cases, a damping ratio of 0.05 is used) and compare it with the design response spectrum to determine the potential impact. However, the response spectrum only shows the maximum response acceleration value as a function of natural frequency and it does not include any information about cumulative fatigue failure, for example. As mentioned earlier, the peak value of the observed acceleration waveform is not necessarily related to equipment damage. Moreover, the acceleration response spectrum is influenced by the peak acceleration of the acceleration record. This may lead to an erroneous judgment, especially in the high frequency region in which such influence tends to be significant.

In one Member State, where no automatic reactor shutdown is performed based on earthquake motion observations, it is required for operators to perform manual reactor shutdown when the observed earthquake motions exceed the SL-1 level (sometimes designated as the operating basis earthquake). A possible set of exceedance criteria are presented below for the purposes of illustration [3, 4]. Both criteria need to be met in order to consider that the operating basis earthquake was exceeded.

1. Response spectrum check: The 5% damped ground response spectrum for the earthquake motion at the site at frequencies between 2 and 10 Hz exceeds the corresponding operating basis earthquake design response spectrum or 0.2 g, whichever is greater; and

2. Standardized cumulative absolute velocity (CAV) check: The computed standardized CAV value from the earthquake record is greater than 0.16 g-sec.

Here, CAV stands for cumulative absolute velocity, which is one of DIPs to be discussed later. Note that the response spectrum check criterion does not consider frequency contents higher than 10 Hz, by limiting the frequency range of response spectrum comparison to 2 to 10 Hz. The intent is that earthquake motions that do not affect equipment are eliminated by using both the response spectrum and CAV criteria.

In IAEA Safety Report Series No. 66 [1], Section 3.3.1, the response spectrum check is described in paragraph *(a)* (details of which are omitted in this publication) and the DIP check is described in paragraph *(b)*, which is quoted below.

> "Exceedance of a damage indicating parameter. An additional check needs to be part of the SL-1 or SL-2 exceedance criteria, utilizing a parameter that suitably describes damage from earthquake motions (damage indicating parameter). One such parameter is the CAV, which has been correlated with observed damage to ductile components that have experienced earthquake motions. Another potential damage indicating parameter is the JMA [Japan Meteorological Agency] intensity."

IAEA Safety Report Series No. 66 recommends performing a check combining *(a)* and *(b)*, that is, not using only one of the conditions.

As it can be seen in the quoted material above, IAEA Safety Report Series No. 66 points out CAV and the JMA intensity as candidate DIPs, but specific studies are yet to be completed. In this publication, several DIPs are discussed as parameters that can suitably describe the damage caused by earthquake motions (Section 4).

2.1.3. Reliability requirements

Seismic instrumentation systems installed at nuclear power plants have an important role in terms of seismic safety, to continue or to shut down reactor operation, either manually or automatically, by detecting earthquake levels and comparing them to design bases as described in the sections concerning DIPs. In this regard, the regulations or the technical specifications, depending on the reactor type, require different reliability levels for each of the installation patterns A, B and C defined as follows:

A. Requiring that operators' prompt decision on manual reactor shutdown can be made within the prescribed time.

B. Focusing on sending alarms to operators by SAS and requiring operators' prompt decision on manual reactor shutdown.

C. Focusing on safe and appropriate automatic reactor shutdown by ASTS.

These patterns are compared in Table 1.

TABLE 1. COMPARISON OF INSTALLATION PATTERNS AND RELIABILITY REQUIREMENTS

Pattern		**A**	**B**	**C**
With or without ASTS		Without ASTS	Without ASTS	With ASTS
Typical Member States		United States of America	Hungary / China	Japan
ASTS Automatic Seismic Trip System	Safety Requirement: to automatically shut down reactor	-----	-----	Reliability is required including redundancy under which ASTS is defined as a reactor protection system
SAS Seismic Alarm System	Safety Requirement: to manually shutdown reactor promptly after an earthquake	Reliability and seismic quality are required	Reliability and seismic quality are required, and redundancy is also required, focusing on the reliability of alarming SAS in main control room, in particular	Reliability and seismic quality are required, depending on utilities
SDAS Seismic Data Acquisition System			Reliability and seismic quality are required	

Both Pattern A and Pattern B do not install an ASTS, which is a key difference with Pattern C. In Pattern A, the function of SAS is shared by SDAS, which means the signal for SAS usually comes from the SDAS. On the other hand, typically in low seismicity countries, SAS is independent from SDAS and the installation of SDAS is not necessarily required. This type of seismic instrumentation system is categorized as Pattern B in this publication.

2.1.4. Ground motion observation points and definition of design basis ground motion

Safety-related SSCs installed at nuclear power plants are generally designed based on dynamic analysis, taking into account the seismic response for the design ground motions. The seismic response associated with SSCs is developed based on a design basis earthquake, which is defined at a specified control point. This control point is typically established at the ground surface or at some elevation below the surface associated with a geological feature (e.g. rock layer) [5]. In the event of a significant earthquake, a comparison between the design basis of the nuclear plant and the recorded earthquake needs to be conducted. This comparison between the design basis and the actual earthquake needs to be conducted at the same control point. Thus, the seismic instrumentation system needs to be located consistently with the design basis control point. For most nuclear plants, the position at which the design basis ground motion is defined fall into one of the following cases (Fig. 1):

– Case A: Ground surface in the free field.

– Case B: Virtual outcrop of the base stratum (bedrock).

The definition of control point is a critical issue in any comparison of the actual earthquake and the design for the plant. As for Case A, the ground surface at a position that is not affected by surrounding structures, that means, well away from structures, is considered as the measuring point. Case B defines the motion as on the free surface of bedrock that has more than a certain level of stiffness. Generally, there are surface layers and structures on the bedrock

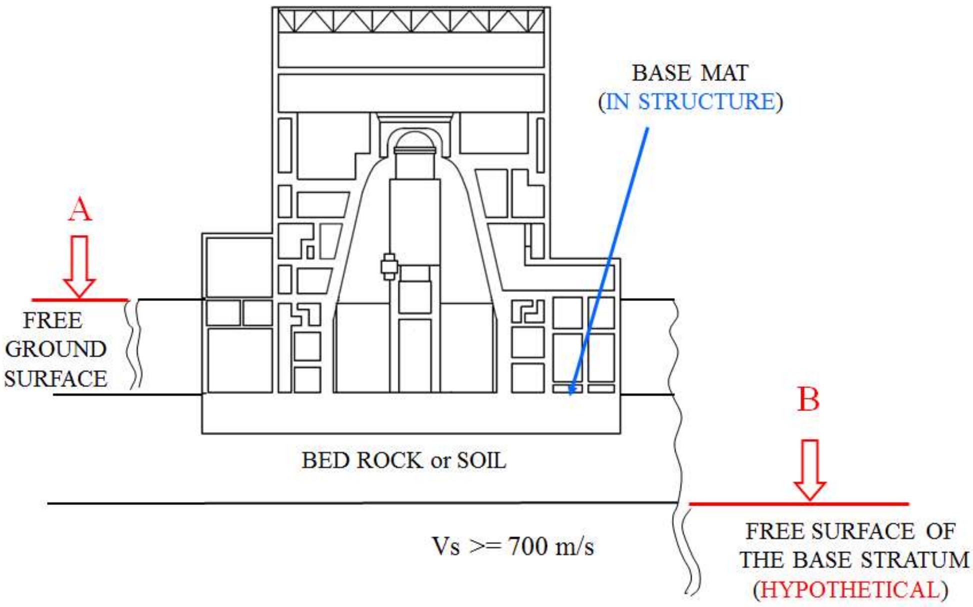


FIG. 1. Positions to define design basis ground motion and measuring points.

layer, making direct measuring impossible (it needs to be noticed that even if a seismometer is installed on the bedrock boring a hole in the ground, the observed earthquake motion cannot be compared with the design basis ground motion due to the effects of reflection of propagating waves).

The characteristics of ground motions are strongly affected by the geological conditions and the geography. The seismic response of the buildings in which safety-related components are installed is further affected by the supporting ground conditions. Thus, it is considered that seismic sensors located within safety-related facilities (e.g. buildings) would provide the most accurate evaluation of seismic input to equipment and of DIPs.

Based on the above, at a nuclear power plant where the design basis ground motion is defined at the ground surface in the free field (Case A), evaluation of effects is performed by combining records from measuring points on the ground surface, well away from structures, and those inside the structures. At a nuclear power plant, where the design basis ground motion is defined at a virtual outcrop (Case B), measuring points are located on the building basemat and at the locations (floors) where safety-related SSCs are installed.

For illustrative purposes, in Japan, rock with an approximate shear wave velocity of 700 m/s or higher, which has not been subjected to significant weathering, is considered as a base stratum (bedrock). The 'free surface of the base stratum' is defined as a free surface, which is hypothetically set, assuming that there are no surface layers and structures on the ground surface, to define a design basis ground motion (virtual outcrop, Fig. 1). Therefore, in Japan, the control point is categorized as Case B. In this regard, because the direct measurement of the earthquake motion at such a hypothetical position (virtual outcrop) is impossible, it is usual to compare the observed earthquake motions at the building basemat, in structures or in boreholes, if any, with the motions computed with the design basis ground motion, at those observation positions.

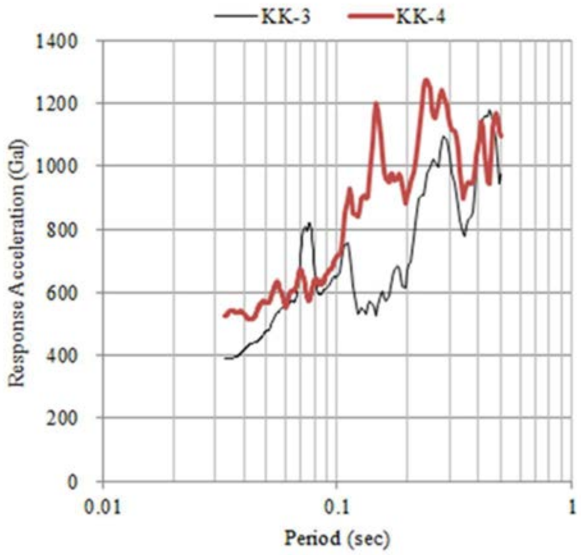
2.1.5. Sharing seismic instrumentation systems among several units (multi-unit sites)

The requirements about sharing of seismic instrumentation systems in a multi-unit site are different, depending on the seismic design philosophy of each Member States, as shown in Table 2.

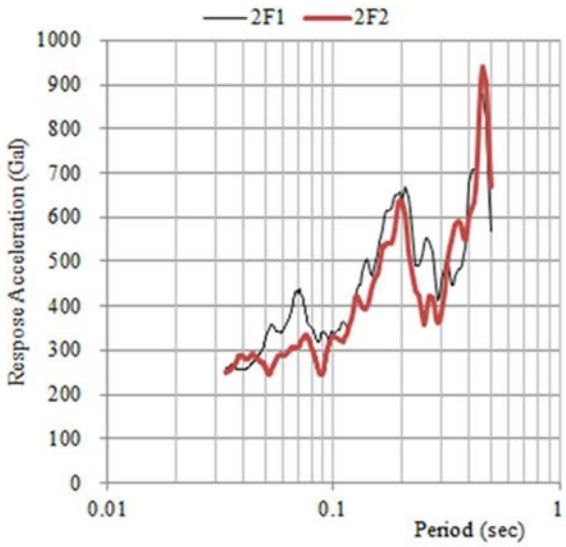
Both the US-NRC Regulatory Guide 1.12 [6] and the Kerntechnischer Ausschuss (Germany) (KTA) 2201.5 [7] have the prerequisite for non-installation of seismic instrumentation systems at all units that seismic responses are expected to be identical among the units. However, even though responses may be analytically identical among the units, different responses have been experienced in real earthquakes as shown below.

Fig. 2(a) shows the acceleration response spectra (damping ratio: 5%) obtained from the observed acceleration time-history in the east-west direction on the foundation mats of reactor buildings at Units 3 and 4 of Kashiwazaki-Kariwa nuclear power plant during the 2007 Niigata-ken Chuetsu-Oki earthquake. Both Units 3 and 4 are boiling water reactors with Mark II advanced-type containment, with electric output of 1100 MWe and located adjacent to each other at a distance of approximately 160 m. However, geological data are slightly different from one location to the other. They can be considered to belong to a 'heterogeneous site' category in the Règle Fondamentale de Sureté (France) I.3.b [8], as shown in Table 2. For this earthquake, a great difference was foumd in the shapes of the spectra due to the differences in the physical properties of the supporting rock layers and to the strong directivity of earthquake wave propagation caused by the short hypocentral distance from the site.

Fig. 2(b) shows the acceleration response spectra, which were observed in the north-south direction on the reactor building foundation mats at a twin plant with almost the same ground conditions, that is, Unit 1 (boiling water reactor with a 1100 MWe Mark II containment vessel) and Unit 2 (boiling water reactor with a 1100 MWe Mark II advanced-type containment vessel) of Fukushima Dai-ni nuclear power plant. The response spectra correspond to the 2011 Tohoku earthquake, which is considered to have less directivity effects because of a large hypocentral distance. Note that spectra in both foundation mats are similar. Around the natural period of 0.07 seconds (14 Hz), however, a difference exceeding 10% can be recognized.



(a) Units 3 and 4 of Kashiwazaki-Kariwa nuclear station (east-west direction on the reactor building basemats in 2007 Niigata-ken Chuetsu-Oki earthquake, h=0.05)



(b) Units 1 and 2 of Fukushima Daini nuclear station (north-south direction on the reactor building basemats in 2011 off-the-Pacific-coast-of-Tohoku earthquake, h=0.05)

FIG. 2. Comparison of the response spectra observed at adjacent units in multi-unit sites.

TABLE 2. EXAMPLE OF REQUIREMENTS ON SEISMIC INSTRUMENTATION SYSTEM AT MULTI-UNIT SITES

Member States (Guideline)	United States [6] (NRC Regulatory Guide 1.12)	Germany [7] (KTA 2201.5)	France [8] (RFS I.3.b)	Japan (none)
Sharing Requirements	Instrumentation in addition to that installed for a single unit will not be required if essentially the same seismic response is expected at the other units based on the seismic analysis used in the seismic design of the plant. However, if there are separate control rooms, annunciation need to be provided to each control room as specified in Regulatory Position C.7.	In the case of multi-unit plants, each reactor building will be equipped with seismic instrumentation. In well-justified cases, it is, however, sufficient to equip only one reactor building with seismic instrumentation (e.g. similar building structure and similar subsoil conditions).	For heterogeneous site[1], in addition to instrumentation installed in an homogeneous site, the following devices will be installed: -An additional triaxial accelerometer in free-field placed in an area of geological and mechanical characteristics or topography different from that which is already instrumented; -A triaxial accelerometer at the base mat of the reactor building of each unit. In case of a significant earthquake, an alarm is delivered in every unit control room.	Although no specific rules have been established on seismic instrumentation system for multi-unit sites, ASTS is installed at every unit as reactor protection system.

Note: 1: Site with heterogeneous geological and mechanical properties of its soil or irregular topography

The boiling water reactor buildings with Mark II type containment usually have a basemat width of around 80 m. In this regard, earthquake wave incoherency effects cannot be ignored in consideration of the aforementioned difference, depending on the site conditions. As shown in these instances, the responses of buildings are affected by not only the physical properties of the ground but also by the directivity of wave propagation. Therefore, it can be said that a careful investigation is needed to determine whether the actual seismic responses can be deemed identical, even at the units that are located adjacent to each other.

2.2. SEISMIC DATA ACQUISITION SYSTEMS

2.2.1. Introduction

A SDAS is a complete seismic monitoring system consisting of sensor(s) and data acquisition units, including communication hardware and software, that acquire, store and transmit digital data recorded during the seismic motion. A SADS plays a key role in collecting site specific seismic instrumental data during the life cycle of the nuclear installation from site selection, to site characterization, to the operational stage until decommissioning.

Site specific seismic instrumental data serve for various purposes during the lifetime of the installation. For example, providing information for the assessment of the seismic hazard at the site; recording the actual seismic response of SSCs, in the event of an earthquake occurrence that is felt at the site; and providing necessary information for post-earthquake actions leading to installation shutdown, restart, and other considerations.

In addition to on-site instrumentation, IAEA Safety Standard SSG-9 [9] recommends that a network of weak-motion sensitive seismographs be installed and operated in the near region of the site (i.e. not less than 25 km in radius around the site). Its purpose is to acquire detailed information on potential seismogenic sources for seismotectonic interpretation. This local network is usually connected to the regional and national seismological networks. This local network will not be discussed in this publication.

2.2.2. Types and characteristics of seismic data acquisition systems

For nuclear power plant applications, the current practice is for sensors to be accelerometers.

In general, the following considerations are important in a SDAS:

– Robustness: Equipment needs to operate reliably over long periods of time – at least ten years in the environment of the nuclear power plant (site and in-structure). This environment could include ranges of temperature, high humidity, dust and/or other conditions. This may lead to requirements for protection against these environmental factors, such as thermal insulation, cases or covers, etc. Instrument output needs to be unaffected by reasonable changes in magnetic fields and atmospheric pressure and reasonable levels of radio frequency interference.

– Measurement type: Acceleration, displacement, deformation, strain and DIPs need to be considered as physical quantities to be recorded. Time varying quantities need to be recorded as time-histories. Peak values of time varying quantities may also be recorded for specific applications such as automatic or manual shutdown. Derived quantities may be useful in determining the expected level of damage in the nuclear installation and may define whether an operating plant may continue operating or needs to be shutdown. For purposes of this section, the discussion focuses on acceleration time-histories.

– Directions of recorded motions: In general, for nuclear installations, three directions of motion (two horizontal and the vertical) need to be recorded. These triaxial sensors need to be aligned along the principal directions of the installation, for ease of use in subsequent evaluations of SSCs. It is most convenient if these directions coincide with the principal directions of analytical models of the SSCs. Locations of instruments are discussed later in Section 2.2.3.1.

– Dynamic range: The dynamic range of the system is the range of amplitudes that can be accurately measured, bounded below by system and site noise or digital resolution and bounded above by the sensor characteristics. The dynamic range is typically defined as the signal to noise ratio. Dynamic range is measured in dB and equivalent bits.

– Frequency range or bandwidth: The frequency range is the range of frequencies that can be accurately reproduced by the recorded data. The overall bandwidth is a function of the system (i.e., sensors, cabling and digitizer bandwidth). Minimum frequency range is 0.02- 50 Hz. Typically, the low frequency limit is at 0.01 Hz. The minimum sampling rate needs to be 200 samples per second.

- Cross-axis sensitivity: The cross-axis sensitivity is the sensitivity of the measurements in one direction to motions in the other two directions. Cross-axis sensitivity needs to be as low as possible. It usually is measured as a ratio of amplitude of motion to that of the main direction of interest.

- Absolute timing accuracy: The recorded motion from multiple instruments needs to be based on a common time scale. These records are appropriately correlated in time for further data assessments. For example, in the free field, the assessment of ground motion incoherency could be made based on the recorded data from an array of instruments. On the foundation, rotations of the foundation (rocking and torsion) can be derived from multiple instrument recordings to permit post-earthquake dynamic analyses of structures subjected to appropriately correlated base translations and rotations. In-structure instruments recording motions correlated in time with free-field and base mat motions can be analysed to determine structure dynamic characteristics from transfer functions derived from the Fourier transforms of the recorded motions.

- Pre-event memory: Pre-event memory times need to be sufficient to capture the P wave motions, when the sensor is triggered to save data by the S wave motions. A minimum of 30 seconds is recommended in general. For a low seismicity site, earthquakes that can produce significant acceleration are generally near field, so the time difference between P and S waves is low. For example, at least 15 seconds is recommended as a pre-event duration in France.

- Recording capacity: Recording capacity needs to be adequate to capture the entire free-field record and the free vibration response of the structure after the strong shaking has reached a minimum level. In France, Règle Fondamentale de Sureté I.3.b [8] requires that the recording continues 30 seconds after the last acceleration higher than 0.01g.

- Multiple event recordings: Adequate provisions are needed to permit recording of multiple events that may occur within a short time interval such as a few hours.

Table 3 summarizes the performance characteristics of seismic instrumentation systems from the 1970s to the current practice.

- Early vintage systems were analogue with very limited capability to meet the current objectives as itemized above. Early vintage systems may have included response spectrum recorders – basically scratch plate devices to directly record response spectral ordinates for comparison with design basis earthquake parameters.

- The evolution from the early vintage systems to today is shown in Table 3. In the United States of America, guidance for seismic instrumentation of nuclear power plants was first published 1971. The first version of US NRC Regulatory Guide 1.12, on Nuclear Power Plant Instrumentation for Earthquakes was published at that time. Revisions of this Regulatory Guide were published in 1974, 1997 and 2017. In all cases, revisions were made to accommodate changes in the state of knowledge of earthquakes and their characteristics, and changes in the area of instrumentation systems.

- Recent practice for SDASs in the United States of America is defined in Table 3, Column 5 (2009 Commercial nuclear power plant Systems, Kinemetrics or Syscom).

Table 3, Columns 6 and 7 identify minimum requirements for two classes of seismic instrumentation – class A and B. Class A instruments are sensors and data acquisition units that were considered at or near the state-of-the-art in 2010 [10]. Class B is one step down from Class A [11]. Both class A and B closely match the specifications in Column 5, with some exceptions. The system requirements of Column 8 indicate the current state-of-practice [6].

2.2.3. Important considerations for seismic data acquisition systems

2.2.3.1. Locations of sensors

Instrument locations need to be selected by the operator of the nuclear installation to obtain adequate information to meet the overall regulatory and mission critical requirements. Instrument locations need to be selected taking into account the following considerations:

(a) Will the recorded data be used to calculate motions at other locations in the structure? If so, are translational inputs at a single location adequate or does a small array of instruments need to be placed to permit definition of rotations? Are multiple triaxial instruments needed on the base mat to define rotations (torsion and rocking)? This issue may be important for new nuclear power plant designs that implement very large basemats.

(b) Will the recorded data be used to compare with in-structure design basis data? This will guide the locations of the instruments within the structures of interest.

(c) Are the instruments located such that equipment-structure interaction effects are minimal? Generally, structural response is desired without the complication of local effects like flexible slabs, equipment-structure interaction, etc.

(d) Are results from a site-specific probabilistic risk assessment used in the decision-making process? The location of instruments needs to be based on the site-specific probabilistic risk assessment insights (which buildings, which elevations within the building, etc.). An alternative is to use results from other probabilistic risk assessments on similar designs coupled with engineering judgment to account for differences in design basis.

Instrument locations need to be selected with appropriate consideration given to items (a) through (d) above, and to obtain data that can be directly related to 'input' and 'output' vibratory motion used in the seismic design. Locations of sensors to capture free-field ground motion, basemat motion and in-structure motion are further discussed below:

– Free-field ground motion: Especially for Case A in Section 2.1.4, the free-field is defined as those locations on the ground surface or in the site soil column that are sufficiently distant from the site structures to be essentially unaffected by the vibration of the site structures. To the extent practical, the location of the instruments needs to be on a soil profile similar to the soil profile used in the seismic design process. Practical definitions of 'sufficiently distant' to satisfy the requirements vary from Member State to Member State. In France, the Règle Fondamentale de Sureté I.3.b [8] considers as free field any point with a minimum distance of 100 meters from any heavy building (buildings of the nuclear island, turbine hall, cooling towers). A sensor located in the free-field records essentially the free-field ground motion. This free-field motion is used to determine whether the design bases earthquake ground motion (SL-1 or SL-2) have been exceeded. Instruments placed in the free field may directly determine a DIP; alternatively, the DIP may be calculated from the recorded acceleration time-histories. The free-field motion may be the starting point for confirmatory seismic analyses to be performed of the nuclear power plant structures, if deemed necessary.

- Foundation or basemat motion: One or more sensors located on the foundation or basemat may serve multiple purposes, for instance, comparison of the seismic design motion with that of the recorded data in the form of response spectra comparisons, input to the calculation of the detailed response of the structure (if deemed necessary to do so) etc.

- In structure motion: Time-history accelerometers located at key locations in safety-related structures provide data for direct comparison with the seismic design parameters for SSCs, benchmarking of the dynamic models, input to supported subsystems at these locations, etc. The locations selected for these instruments need to adhere to principles such as locations important to overall soil-structure response, locations where secondary effects are minimal (flexible sub-structures like floors or slabs, equipment-structure interaction, etc.), locations where systems and components important to safety are located, as determined from risk importance concepts such as those contained in US NRC Regulatory Guide 1.201 on Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance, or from a seismic probabilistic safety assessment or from a seismic margin assessment. Examples inside the containment are reactor equipment supports, reactor piping supports, and other safety-related equipment and piping supports.

Four examples of guidelines on the number and placement of seismic instruments are presented below:

IAEA safety standard on Seismic Design and Qualification for Nuclear Power Plants [2]

It recommends that a minimum amount of seismic instrumentation is to be installed at any nuclear power plant site as follows:

> "One triaxial strong motion recorder installed to register the free field motion;
>
> One triaxial strong motion recorder installed to register the motion of the basemat of the reactor building;
>
> One triaxial strong motion recorder installed on the most representative floor of the reactor building.
>
> The installation of additional seismic instrumentation needs to be considered for sites having an SL-2 free acceleration equal to or greater than 0.25g."

TABLE 3. COMPARISON OF NUCLEAR POWER PLANT SEISMIC INSTRUMENTATION PERFORMANCE IN THE UNITED STATES OF AMERICA (COURTESY OF PROF. NIGBOR)

Performance Parameter	70s-Vintage, Kinemetrics SMA-3	RG 1.12 Rev. 2 1997	ANS 2.2 – 2002	2009 Commercial NPP System, Kinemetrics or Syscom	Advanced National Seismic System Class B	Advanced National Seismic System Class A	RG 1.12 Rev. 3 2017
Dynamic range, dB	40	60	60	86-108	>86	>110	>=110
Dynamic range, equiv. bits	7	10	10	16-18	>= 16	>= 20	>=18
Full-scale, g	1	1	1	3-4	3.5	3.5	4.0
Frequency range, Hz	0.2-30	0.2-50	0.02-50	0-50	0.1-35	0.02-50	0-100
Sample rate, samples/second	Analog	200	200	200 (min)	200 (min)	200 (min)	250 (min)
Cross-axis sensitivity, g/g	0.03	Not specified	0.03	0.01	0.01	0.01	< 0.01
Absolute timing accuracy, msec	No timing	Not specified	Adequate to differentiate main	<1 with GPS	<1	<1	5
Pre-event memory, seconds	0	3	3	99 or greater	>=60	>=60	>=60
Recording capacity, minutes	10	25	25	>100	>=60	>=60	>=120

US NRC Regulatory Guide 1.12 on Nuclear Power Plant Instrumentation for Earthquakes [6]

It states that triaxial acceleration sensors need to be provided at the following locations:

1. “Free-field.

2. Containment foundation.

3. Two higher elevations (excluding the foundation) on a structure inside the containment.

4. A Seismic Category I structure foundation included in a certified standard design or facility where the expected response is different from that of the containment structure.

5. An elevation (excluding the foundation) on a Seismic Category I structure selected in 4 above.

6. An elevation in a non-containment Seismic Category I structure supported by the containment foundation.

7. A Seismic Category I structure foundation not included in a certified standard design or facility.

8. A higher elevation in the instrumented Seismic Category I structure selected in (7).

9. Alternatives to sensor locations (2) through (8) may be necessary in order to support other instrumentation criteria in this regulatory guide.”

Other Considerations in US NRC Regulatory Guide 1.12 are as follows:

- Instruments operational in all modes of plant operation including periods of plant shutdown (Regulatory Guide 1.12, C 3).

- Instruments at multi-unit sites (Regulatory Guide 1.12, C 2).

- Control Room notification (Regulatory Guide 1.12, C 7).

Further to selecting sensor locations, a design review of the location, installation, and maintenance of proposed instrumentation, including considerations for keeping exposures as-low-as-reasonably-achievable needs to be performed.

Kerntechnischer Ausschuss (Germany): KTA 2201.5 on Design of Nuclear Power Plants against Seismic Events, Part 5: Seismic Instrumentation [7]

It states the following requirements regarding number and location of seismic instruments. They apply to plant sites for which the maximum acceleration of the design basis earthquake does not exceed 0.25 g.

For single unit plants:

> "3.2 (1) Accelerographs shall be provided for in the free-field and inside the reactor building."
>
> "3.2 (3) The placement location of the accelerograph in the free-field shall be chosen such that any influence of buildings on the data to be measured can be ruled out. The accelerograph shall be placed at a distance away from the reactor building that it is equal to at least twice the largest length of the reactor building foundation and, also, away from other buildings by at least the largest ground-plan dimension of the respective building"
>
> "3.2 (5) At least three accelerographs shall be installed in the reactor building. Two of these shall be installed in the lowest building level and one in an upper level of the reactor building (e.g., pool floor level); the horizontal distance between the lower acceleration sensors should be as large as possible. The placement locations of the accelerographs should be chosen such that a direct comparison of the measured data with the corresponding design quantities is possible. At their placement location, there should be only negligible operation-related effects on the measurements."
>
> "3.2 (6) The acceleration sensors of the accelerographs should normally be oriented such that their axes are parallel to the axes of the coordinate system used for the seismic analysis of the reactor building."
>
> "3.2 (7) The accelerographs shall be accessible for the necessary operating and maintenance procedures. The accelerographs shall be designed and installed such that an evaluation of the recorded data is not adversely affected, e.g., by damages to components or civil structures that fail during an earthquake nor by a superposition of the measured seismic signal with seismically induced oscillations of neighbouring components."
>
> "3.2 (8) The acceleration sensors of the accelerographs shall be mounted such that no movements relative to the mounting support can occur."

Règle Fondamentale de Sureté (France) on location of instrumentation devices (Ref. [8], I.3.b 2.2.3)

The implementation of seismic sensors depends on the type of site:

Homogeneous sites

For a homogeneous multi-unit site, only one unit may be equipped with a seismic instrumentation. Triaxial accelerometers need to be located:

- At the basemat of the reactor building;
- At one or more floors of the reactor building, with sufficient elevation to have significant amplifications in acceleration and also to be able to estimate with better accuracy the effects on some important safety-related components located above the base mat. These accelerometers are located approximately in the same vertical;

– At the basemat of another building containing systems important to safety and whose foundations are different from those of the reactor building;

– In the free field.

These triaxial accelerometers are arranged so that their respective three orthogonal directions coincide with each other along the principal axes of the buildings of the instrumented unit.

Heterogeneous sites

For a heterogeneous site, in addition to instrumentation installed on a uniform site, the following devices need to be installed:

– An additional triaxial accelerometer in the free field placed in an area of geological and mechanical characteristics or topography different from that which is already instrumented;

– A triaxial accelerometer at the basemat of the reactor building of each unit.

2.2.3.2. Other considerations

– Seismic qualification: The SDAS needs to be seismically qualified to perform its functions during and after the earthquake ground motion. The minimum level of seismic qualification needs to be to the SL-2 level. It is preferable for the seismic qualification to be at a level greater than the SL-2 to assure operability if an earthquake produces ground motion greater than the SL-2 at the site. Seismic qualification needs to include any support systems necessary for system operation (e.g. emergency power, battery backup).

– Maintenance and testing: To reasonably assure operability of the SDAS, appropriate periodic maintenance needs to be performed. In many instances, maintenance may be sub-contracted to a third party (often the system supplier) for an extended period of time, for instance, 10 years. The operating organization needs to verify that maintenance is performed even after such maintenance agreements have expired. In addition, testing of the operability of the system needs to be performed on a regular schedule, for example, quarterly. Schedule maintenance to keep maximum number of instruments in service.

– Operability of SDAS in all operational modes of the nuclear installation: Operational modes of the installation include low power states and plant shutdown (scheduled and unscheduled outages).

– Multi-unit sites: Provisions need to be made to coordinate the SDAS for multi-unit sites. In some cases, one set of sensors may serve multiple purposes on-site (e.g. free field sensors). For sites, where there are common systems for more than one unit, these common systems may be monitored by, and the recorded data may be announced in multiple control rooms. Immediate post-earthquake actions for multi-unit sites need to be closely coordinated between units.

– Control room notification: Appropriate notification of recorded earthquake motions needs to be announced in the control room. These data include necessary information for operator actions, such as manual shutdown of the installation. In addition, notification to the control room needs to be made if earthquake records data storage is approaching the storage maximum capacity and additional storage devices need to be added. The operational data of the SDAS, such as outage due to loss of the electric source or malfunction are also recommended to be notified.

– Installation and configuration control: Installation of all elements of the SDAS needs to be such that all seismic systems interaction concerns are resolved. Seismic systems interaction concerns are the consequence of failure of not-important to safety, SSCs causing failure or malfunction to important to SSCs. The latter could include the SDAS. These concerns are sometimes referred to as 'II/I interaction'. The phenomena associated with seismic systems interaction are: *falling* of items impacting the important to safety item, *proximity* meaning impact of adjacent not important to safety components causing malfunction or damage to important to safety components, and *spray/flooding*. Installation and on-going configuration control procedures need to assure that potential II/I issues do not prevent the SDAS from performing its functions. In some cases, enclosures may be used to prevent inadvertent damage to portions of the system (or accidental impact by plant personnel), for instance, for sensors that initiate the ASTS.

– Anchorage of sensors: It is advisable that when an acceleration sensor is installed on structures, it is installed in a safe location, taking account protection from accidental impact. When installed on ground surface, lightness and rigidness of its foundation structure (concrete pad) need to be enhanced to the extent possible to avoid interactions with the supporting ground. Fig. 3 shows the configuration recommended in Ref. [12]. When the sensor not installed on rock, a pile foundation needs to be used to guarantee a minimum bearing capacity. Exposure to water needs to be prevented as well (e.g. by a rainproof shed).

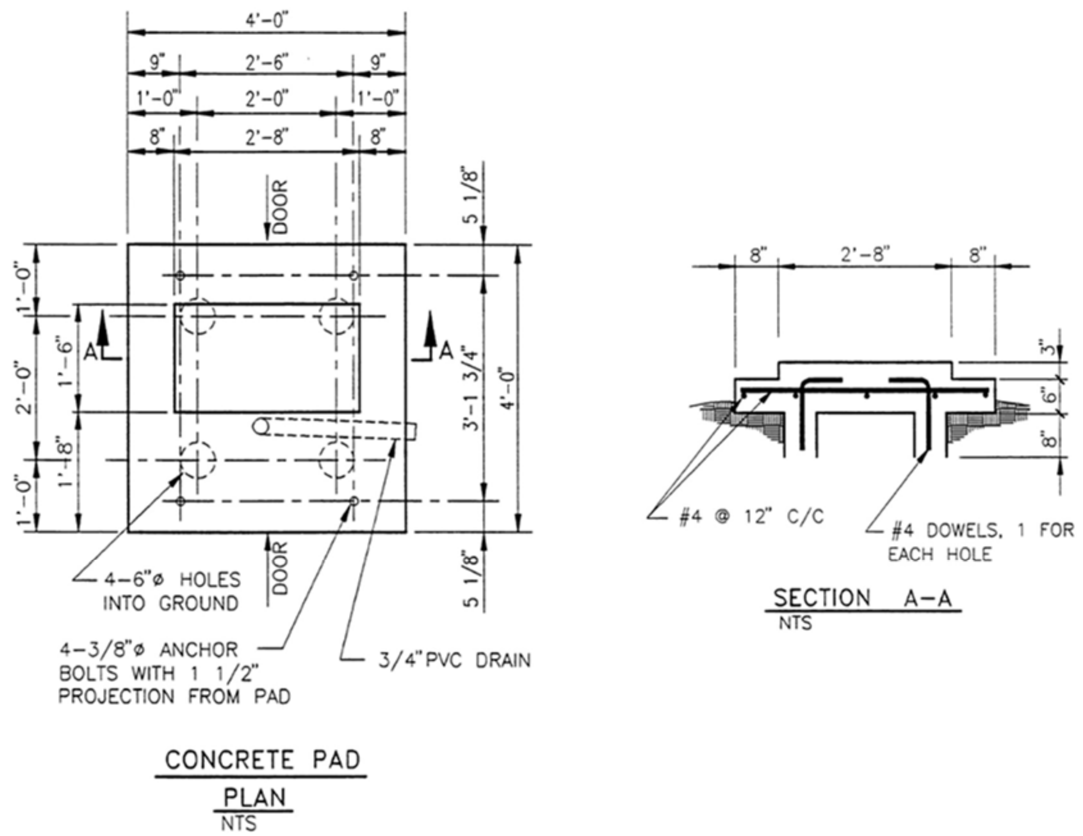


FIG. 3. Example of foundation when sensors are installed on the ground surface [12]. (Courtesy of EPRI)

- Electric power supply: A SDAS needs to have a power supply system anticipating a loss of off-site power in the event of an earthquake. It should be noted that a backup power supply from an emergency diesel generator has a window time until the diesel generator starts up upon a loss of off-site power. In this context, SDAS needs to be also energized by an uninterruptible power supply. In the US Regulatory Guide 1.12, for example, it is required to install a battery that allows for more than 120 minutes of recording, and is rechargeable with an uninterruptible power supply or another power supply. Even in the event of a loss of off-site power, it is recommended that 24-hour recording, and 10-minute continuous recording are possible and verifiable.

- Data processing time: At a plant without ASTS, the time available to analyse SDAS data for the decision on the necessity of a reactor manual shutdown is limited (within four hours after an earthquake in the United States regulation, for example). It is necessary to install a solid-state digital SDAS capable of processing data in a short time or to develop a system that will promptly export the recorded data to a computer (see the lessons learned in Section 2.2.4).

- Maintenance and management: Similarly, a plant without an ASTS needs to enhance the reliability of SDAS. It is required that online maintenance and repairs are performed when the maximum number of SDAS instruments are available; system channels need to be checked every other week during the first three months from the start of plant operation, followed by monthly checks (including batteries). Thereafter, channel functions need to be checked every six months and channel calibration needs to be checked during plant refuelling outage [12].

- Logging of start-up signals: Many transients in plant systems may occur in the event of an earthquake, even at a plant without ASTS. To analyse these transients, their timing in relation with seismic motions becomes very important. It is recommended that the initiation of the SDAS (start-up trigger signal) be recorded on the general plant transient recorder, so as to make it possible to relate in time the earthquake motion signals with the transients recorded in plant systems.

2.2.4. Lessons learned and other observations

A number of lessons learned from actual earthquake experience and from reviews of seismic instrumentation aspects for operating nuclear power plants need to be taken into account for future installations of new and replacement SDAS.

- Data lost due to overwriting main shock data with aftershock data: SDAS needs to have the provision to store tens of individual recorded earthquake scenarios without accidentally overwriting the data from any individual event with that of another. One example of such a provision was described at the Diablo Canyon Power Plant, San Luis Obispo, California, United States of America. The installed system at Diablo Canyon power plant has the ability to store up to 60 events without switching out the storage device for the system. When the storage device approaches its maximum capacity, the control room is signalled that a new storage device is required. Redundant storage needs to be in place, that is, data may be transmitted to an off-site location, but a redundant set of data needs always to reside at the plant or another independent location.

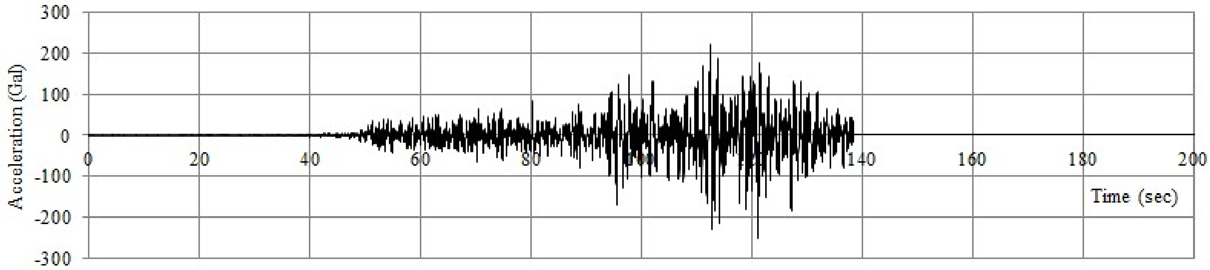


FIG. 4. Experienced acceleration time-history data (data after approximately 138 seconds is lost).

- Software verification for digital instrumentation systems: Fig. 4 shows an instance in which the last part of an acceleration time-history was lost because the time at which an SDAS ended to record and storing digital data was inappropriate. In this instance, there was an error in the recorder software. Recording would end after the elapse of a 'certain time' upon recording accelerations below the level at which recording is programmed to end. Even though accelerations equal or greater than that level were observed during that certain time, the device failed to extend the recording time. Moreover, when acceleration in excess of the recording level was detected after the end of recording, the system's recording check was not yet completed, resulting in activation of an exclusive control that prevented further recording operations. In the event of a usual earthquake, with a short duration of the strong motion, seismic motions will attenuate within a 'certain time' upon detection of acceleration below the level at which recording is programmed to finish. In this instance, however, the strong motion lasted a very long time and no data after 138 seconds was recorded as shown in Fig. 4. In this instance, the maximum acceleration is captured in the 138-second long recording, which allows for peak parameter evaluation, but the missing part reduces the accuracy in evaluation of integral DIPs. Generally, a 'certain time' is set to several tens of seconds, which makes it difficult to find programming errors by verifying the program using usual acceleration time-history data over the normal duration of seismic motions. This has raised an issue about how SDAS software verification needs to be performed.

- Digital and analogue data processing: Seismic instrumentation systems have developed over time and data storage and processing is shifting from analogue to digital systems. Although nuclear power plants where seismic instrumentation systems were installed in past times still have analogue systems, the advice is to use solid-state digital systems [6]. Even if digital systems are employed, however, the entire seismic instrumentation systems do not have to be part of a network. Importing data from data storage into a personal computer to compute DIPs, for example, is acceptable as long as the necessary information can be produced within the required time. Ref. [13] has proposed graded system configurations in relation to installation costs.

- As aforementioned, the time limit by which DIP information is required is set to within a few hours. For instance, four hours are requested by Ref [6], at a nuclear power plant without an ASTS. When analogue data is recorded and DIP computation is performed using digital data, the digitization of analogue data needs to be performed first. In the past, an instance was reported in which operators' decision was delayed because it took three days to analyse the seismic instrumentation data, as analogue data was digitized by the instrument manufacturer. Having at least a minimal digital system is recommended.

– Data transmission capability: If the expectations or requirements are to transmit the recorded data from the recording device to another location on-site or off-site for review, processing, decision making, etc., a seismically qualified transmission mechanism needs to be available. This could be ‘hard’ technology, such as cables, or ‘soft’ technology, such as wireless transmission. In either case, the transmission mechanism needs to be operable if an earthquake occurs.

2.2.5. Status of existing nuclear power plant seismic instrumentation

The IAEA issued a questionnaire soliciting responses from eighteen Member States. Responses were received from twelve of them. The completeness and level of detail of the responses varied considerably. Generally, for Member States with multiple nuclear power plant sites, some information was missing for specific plants or units at a specific site. Given these facts, the responses constitute a valid sample of the nuclear power plant population with respect to existing seismic instrumentation systems.

Tables II-1 through II-5 in Annex II summarize key data of interest for this publication.

Table II-1 A summarizes general information about nuclear power plants in the Member States, including plant name, sites, number of units, and descriptive notes if available. Table II-1 B and C itemize the same information for Japan and India respectively.

Table II-2 A contains the plant seismic design ground motion definitions (peak ground acceleration, PGA, and response spectra/time-histories) for the SL-1 (operating basis earthquake, design basis earthquake ground motion S1) and the SL-2 (safe shutdown earthquake, design basis earthquake ground motion S2, design basis earthquake). Where final information was provided by the respondents as to control point location, it was noted in the table. Table II-2 B and C itemize the same information for Japan and India, respectively.

Table II-3 contains a tabulation of the free-field instruments plant-by-plant and unit-by-unit, to the extent possible, based on the information provided by the respondents. In the free field, four types of instruments are identified as follows:

1. Accelerometers measure time-histories of acceleration. The types listed were analogue or digital. Newer installations are of the digital type. Generally, they are triaxial and measure three directions of motion. They may be placed on a soil or rock surface or downhole in the free field.

2. Peak acceleration recorders measure peak acceleration. They may be triaxial or measure peak acceleration in one direction only. The measured peak accelerations may be announced in the control room and may be used to trigger the scram of the reactor.

3. Response spectrum recorders measure response spectra directly. A few plants have in place response spectrum recorders. Generally, these are of the scratch plate type where a limited number of frequency-tuned arms etch the maximum response on a plate, which can then be read manually or transmitted in some form to the control room.

4. Cumulative absolute velocity meter (CAV-meter) generates a CAV value directly. The parameter CAV is a measure of the damage potential of the ground shaking. It is discussed in detail in Section 3 and Section 4 below.

In general, plants in high seismicity areas have many more instruments recording earthquake motions in the free field and within structures than plants located in low to moderate seismicity areas.

Japan is a high seismicity area. In Japan, site free-field motions are, typically, measured on the free surface of the soil or rock and at several locations downhole (see Table II-3 B). Outside Japan, many Member States follow the general guidance of US NRC Regulatory Guide 1.12, with a revision depending on the vintage of the plant (See Section 2.2.3.1).

Table II-4 contains a tabulation of the in-structure instruments plant-by-plant and unit-by-unit, to the extent possible, based on the information provided by the respondents. The same types of instruments as described above (except for the CAV-meter) are installed on basemats and in structures at nuclear power plant sites. As above, plants in high seismicity areas install many instruments in structures and on components to record the response or the input motions to SSCs. In low to moderate seismicity areas, guidelines such as those of US NRC Regulatory Guide 1.12 are followed. In Japan, it is common to have tens of accelerometers installed in structures at a plant site.

Table II-5 presents the seismic instrumentation system configuration and major instrumentation specifications at nuclear power plants in China. Tianwan nuclear power plant has pressurized water reactors imported from the Russian Federation (VVER reactors), which are equipped with an ASTS system. Table II-5 shows the concept of an entire seismic instrumentation system combining an SDAS and a SAS (seismic switches), and their redundancy concept.

2.2.6. Future requirements and recommendations

Each of the items from Section 2.2.3 are addressed in this section, with advice for specifications as follows:

- Specify characteristics or performance criteria of SDAS as a function of the specified purposes of the system. Considerations include number and locations of instruments, as-low-as-reasonably-achievable criteria, and the following items:
    - Common time scale and trigger for all instruments.
    - Data storage space sufficient to record tens of individual events. Redundant storage required.
    - Provisions for permanent storage and retrieval of recorded data.
    - Data transmission capability. Transmit data to decision-makers and other professionals needing to process and evaluate data.
- Seismic qualification. The SDAS needs to be seismically qualified to perform its functions during and after the earthquake ground motion to at least the level of the SL-2 design earthquake.
- Maintenance and testing. Regular maintenance and testing need to be specified for the SDAS and its supporting systems, to assure operability when an earthquake occurs.
- Installation and configuration control. Provide assurance that no seismic systems interaction concerns will prevent the SDAS from performing its functions.

– Operability of SDAS in all operational modes of the nuclear installation.

– Multi-unit sites. Provisions need to be made to coordinate the SDASs for multi-unit sites. This consideration will also affect the number of sensors and their locations.

– Control room notification. Appropriate notification of recorded earthquake motions needs to be announced in the control room in a timely manner.

2.3. SEISMIC ALARM/ANNUNCIATION SYSTEMS

2.3.1. Introduction

Immediately after an earthquake, operators in the control room are required to take emergency actions. Operators will be heavily burdened as they will also need to deal with plant transients that may occur indirectly due to the earthquake. For example, in IAEA Safety Report Series No. 66, Section 4.2.3, on immediate operator actions, the following necessary actions are listed:

(a) "Confirm the felt earthquake or other requirements stipulated by the regulatory body;

(b) Determine if the felt earthquake is significant;

(c) Stabilize the plant by normal and/or emergency operating procedures;

(d) Activate the on-site response plan, including walkdown inspections;

(e) If reactor shutdown has not occurred (e.g. no seismic scram and no reactor shutdown due to other plant trip sources), determine whether the plant should be shut down (coordinate with designated seismic engineers after reviewing the earthquake ground motion records and derived quantities);

(f) Perform pre-shutdown inspections."

To take these emergency actions immediately after an earthquake, it is essential that accurate information about the severity of the earthquake that struck the nuclear power plant is promptly communicated to operators. The information needed by operators is discussed later in Section 4.6 in relation with DIPs. The necessity of this information is described below.

2.3.1.1. Judgment on 'felt earthquake' and 'significant earthquake'

A 'felt earthquake' is an earthquake that is felt at a nuclear power plant. It is also defined as a tremor that is recognized as an earthquake by multiple operators in the main control room (consensus) and a quake that will start-up the SDAS [14]. As seen from the above, a 'felt earthquake' is recognized by the seismic instrumentation system and operators are firstly required to determine whether the 'felt earthquake' is a 'significant earthquake' that necessitates some response actions. In such determination the seismic instrumentation system is also deeply involved.

In IAEA Safety Report Series No. 66, a significant earthquake is defined as shown below, when in Case A of Section 2.1.3 (when the design seismic motion is defined on a free ground surface).

“(b) Definition of felt earthquake and significant earthquake:

> (ii) *A significant earthquake* is a felt earthquake having a free-field surface peak ground acceleration at the threshold of damage or malfunction of non-seismically designed power plant (either nuclear or conventional) SSCs. Some typical definitions of a significant earthquake are earthquakes with: a free-field surface peak ground motion of greater than 0.05 g or a standardized CAV greater than 0.16 g·s or an earthquake with spectral accelerations in the 2–10 Hz range greater than 0.2 g (5% damping). The designation of a significant earthquake needs to be a function of the site-specific characteristics and the seismic design basis of the nuclear power plant, since it may determine actions to be taken by the licensee and the regulatory body. The definition of the significant earthquake is the responsibility of the licensee and may require agreement or approval by the regulatory body.”

When an earthquake is determined to be a ‘significant earthquake’, operators will declare an emergency and the plant’s response organization will be established.

2.3.1.2. Emergency response and exceedance of the design basis earthquake SL-1

At a plant without an ASTS, it is important for operators to examine whether safety-related facilities have been affected by an earthquake through walkdown inspections, as part of emergency actions taken immediately after the earthquake, so as to decide the necessity of emergency manual reactor shutdown. The challenge here is that, usually, many of the active safety-related systems that need to be inspected for operability were not operating when the earthquake occurred. Therefore, seismic consequences may not be adequately assessed by plant data examination in the main control room and/or by visual inspections during the walkdown. As for hidden damage to active components, rotating equipment, for example, potential onset of plastic deformation in moving parts can be determined based on the earthquake severity, in addition to actually operating those components to see if there are anomalies. That is, when two design basis earthquake levels are defined for dynamic seismic design, such as SL-1 and SL-2 in the IAEA seismic standards, and the allowable limit for the seismic load by SL-1 is considered almost as the elastic limit, it is very important to determine whether the observed seismic motion exceeded the level of SL-1. Since SDAS data is used to make this exceedance determination, high reliability is required of a SAS that sends out a trigger signal to start-up the SDAS.

At a plant with a digital SDAS, SL-1 (or operating basis earthquake) exceedance determination will be made by analysing the response spectrum described in Section 2.1.2.2 and the CAV, for example, within the prescribed time (within four hours after an earthquake, for instance). It is also advisable that an alarm be activated with independence of those analyses, to promptly notify operators of the potential occurrence of a ‘significant earthquake’. The alarm can be substituted by a DIP computation in a short time, depending on the seismic instrumentation system installed.

Instances in the United States of America

In the United States of America, it is recommended that a plant equipped with a digitalized SDAS calculates the CAV and response spectra, as mentioned earlier, to determine operating basis earthquake exceedance. Flexible exceedance determination methods are also proposed depending on the configuration of the seismic instrumentation system installed in the plant.

That is, operating basis earthquake exceedance determination methods are proposed by classifying plants into the four types listed below, depending on the configuration of the seismic instrumentation system [12].

- Plants with free-field online digital instrument, 200 sample per second, sufficient pre-event memory, bandwidth 0.2-50Hz, capable of calculating CAV and response spectra.
- Plants with instrumentally determined response spectra available.
- Plants with only instrumentally determined peak acceleration available.
- Plants with no instrumental data available.

For example, at a plant which is classified within the third plant category mentioned above (i.e. its seismic instrumentation system is capable of measuring the maximum acceleration only), the observed earthquake motion is determined to exceed operating basis earthquake when:

(a) The maximum acceleration exceeds the operating basis earthquake acceleration value, and

(b) When any of the following is applicable:

  - Modified Mercalli intensity scale within 5 km is VII or higher; or
  - An earthquake of magnitude 5.00 or higher within 100 km; or
  - Magnitude 6.00 or higher.

This method is characterized in that the required decision time for a plant at which seismic intensity grade is added to the exceedance criteria is set to within 24 hours. This is thought to be because it takes time to define the seismic intensity grade when it is based on the observation of seismic damage, that is, when no instrumental seismic intensity in real time is used. Due to this, the decision to manually shut down the reactor may be delayed, compared with a plant equipped with a digital SDAS.

Instances in Germany

The operating basis earthquake exceedance and post-earthquake actions are defined in KTA 2201.6 [15]. It is assumed that continued operation of the plant is acceptable as long as the stresses caused by the earthquake in question remain below the elastic limits and plastic deformations remain restricted to the vicinity of geometrical discontinuities. The latter is the case, provided that the stresses of Level C are not exceeded. In the case of a design against the design basis earthquake (DBE) for Level D, the Level C stresses are attained at α × DBE where:

$$\alpha = \left(\frac{\sigma^{C}_{allowed}}{\sigma^{A}_{as\ per\ design}} - 1\right)\left(\frac{\sigma^{D}_{as\ per\ design}}{\sigma^{A}_{as\ per\ design}} - 1\right) \tag{1}$$

Which results in α equal to 0.6 to 0.7.

Therefore it is assumed that the earthquake with $PGA_{inspection} = 0.4 \times f \times PGA_{DesignBase}$ could not cause damages in the structures designed in accordance with KTA 2201.6, KTA

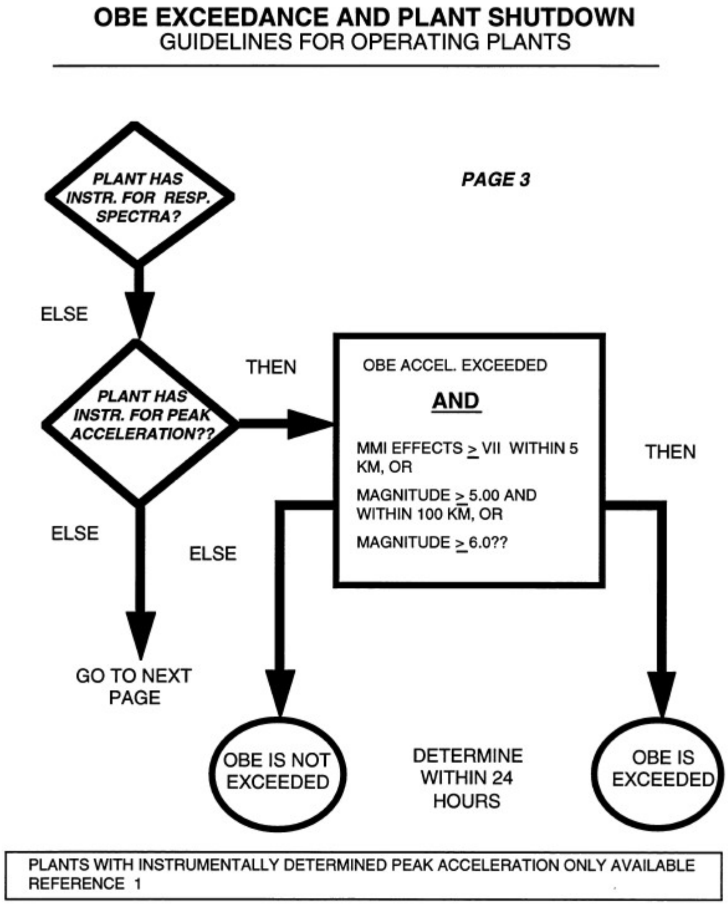


FIG. 5. Example of operating basis earthquake (*OBE) exceedance determination [12] (Plant at which only the maximum acceleration value is measured). (Courtesy of EPRI)*

2201.4 [16], and KTA 3201.2 [17]. Here, f is a conservative factor equal to 1.5, which may be increased to 1.75.

The operating basis earthquake exceedance level or inspection level is $PGA_{inspection} = 0.4 \times PGA_{DesignBase}$. The inspection level is assumed to be essentially exceeded if $PGA = \alpha \times f \times PGA_{DesignBase}$, where f=1.5.

The post-earthquake actions are graded in accordance to whether the inspection level is exceeded, or the inspection level is 'essentially' exceeded. In first case a plant walkdown and a check of the normal plant conditions has to be done. In the second case, when the inspection level is essentially exceeded, extensive assessment of the plant conditions, including control calculations and analyses, are required.

Instances in France

In case of exceedance of the level of earthquake causing an alarm to be generated by the triggers in the control room of the various units on the site (0.01 g), the reading of the data from the monitoring device and a first analysis of time-histories recorded will determine whether the peak acceleration corresponding to half of the amplitude of the design spectrum adapted to the site (SL-1 = 1/2 site specific SL-2) is exceeded at any of the measuring points.

In case of exceeding this seismic level in any of the records, the operator needs to immediately reach the shutdown state considered as the safest for each unit.

The restart of operation can be performed only after justification to the French regulator of the innocuousness of the earthquake for the future behaviour of the plant.

Currently, in the framework of new periodic safety reviews of Électricité de France nuclear power plants since 2012, the SL-1 is re-defined as an ‘inspection earthquake’ corresponding to a fraction of SL-2 set to 0.05 g PGA in the horizontal direction, which is equal or lower than the previous SL-1 depending on the site.

2.3.1.3. Walkdown inspection level and earthquake intensity

At a plant without an ASTS, the role of operators is very important during walkdown inspections, in particular, for the decision on the necessity to shut down the reactor manually in a short prescribed time (e.g. within eight hours after an earthquake [14]). Even at a plant equipped with an ASTS, plant inspection immediately after a significant earthquake is an important task of operators and the results thereof will normally be communicated to regulatory authorities and the public.

For a plant without an ASTS, where the results of walkdown inspections affect safety decisions, the recommended details of walkdown inspections are described in IAEA Safety Report Series No. 66, Section 4.2.3.3, on walkdown inspections by the operator, for example. On the other hand, plants with ASTS are often located in areas of high seismicity. The ASTS can reduce operators’ burdens to specify walkdown details based on the observed ‘earthquake intensity’.

Instances in Japan

In Japan, electric utilities have developed post-earthquake inspection procedures as part of their safety preservation rule, in which the scope of inspection is specified according to the earthquake severity. Table 4 provides an example. This scope of inspection is determined in consistence with the JMA seismic intensity or observed peak acceleration value, making it possible to explain plant conditions corresponding to the seismic conditions of neighbouring municipalities. Basically, the idea is to enhance walkdown inspections at a JMA seismic intensity grade of about 4, and shift to inspection combining functional inspections and others from intensity grade 5 onwards [18].

2.3.1.4. Obligation to report post-earthquake plant conditions

Post-earthquake actions require administrative and public acceptance, for which it is vital to promptly report plant condition. The results of walkdown inspections mentioned in the previous section need to be reported.

Instances in Japan

In Japan, when an event occurs that may have a social impact, the licensee will voluntarily report plant condition. With regard to earthquakes, when a strong earthquake is observed around the power plant, plant condition will be reported upon JMA seismic intensity 5 or higher within a radius of 100 km, or JMA intensity 4 or higher within a radius of 50 km. Here, JMA seismic intensity scale is used as a guide because it is considered difficult for the public to interpret the earthquake motion severity from its maximum acceleration, and the maximum acceleration is not always related to seismic damage. JMA seismic intensity scale is very popular in Japan as a generally used DIP.

In addition, local municipalities require prompt reporting from the plants with which nuclear safety agreements have been signed and entered into force. The earthquake motion level at which reporting is compulsory is different from one municipality to the other. For example, in Ibaraki-prefecture, that has nuclear-related facilities within the prefectural region, it is required of the licensee to immediately perform plant inspections and promptly report the results thereof (by telephone or facsimile) when an earthquake of JMA seismic intensity grade 4 or higher occurs at the prefectural capital, Mito-city, or in the municipality in which the plant is located.

This JMA seismic intensity 4 is a threshold value and a very conservative value, at which some kind of damage may occur to conventional industrial facilities no-seismically designed, as discussed later in Section 3.2.3.

2.3.2. Types and characteristics of seismic alarm/annunciation systems

Here, the functions of the seismometers that are used for operators to take emergency actions and their background are discussed.

As a function of seismometers used for emergency actions, a seismic switch is installed to trigger a signal indicating that the pre-set value has been exceeded. This seismic switch does not have to record the acceleration time-history. A seismic switch generally consists of a triaxial accelerometer, equipped with sensors to close relay contacts, power cables and relays inside the seismic instrumentation panel. It has the function of instantaneously indicating at a remote location that a preset acceleration value has been exceeded. Depending on the seismic instrumentation system configuration, the seismic switch may constitute a SDAS, rather than being an independent device. Here, the intent is to clarify the function of a SAS by discussing its functional requirements considering the seismic switch as independent. However, the ideas may also be applicable when an SDAS functions as a SAS.

The functions of the seismic switch can be summarized as follows:

- Generation and transmission of trigger signals. The SDAS will be started up when the preset seismic acceleration value is exceeded. In general, the trigger signal is indicated in the main control room, and the time of occurrence is recorded for analysing its relations with plant transients resulting from the earthquake event. The data recorded by the SDAS, started up by the trigger signal, is to be used for automatic or manual DIP calculations to decide post-earthquake actions.

- Main control room alarm and indication of seismic parameters. The peak acceleration value or seismic parameters will be indicated as a control room alarm and printer output to initiate the emergency response procedures according to the earthquake levels thereof.

As seen from the above, the function of a seismic switch is to trigger the SDAS and provide indications and alarms in the main control room. Although, its basic performance is similar to that of SDAS sensors, it is strongly required from a SAS to prevent malfunctions. In IAEA Safety Guide NS-G-1.6, for example, spurious signal prevention is emphasized as described below [2]:

TABLE 4. SCOPE OF POST-EARTHQUAKE INSPECTIONS AND EARTHQUAKE LEVELS (EXAMPLE) [18]

Utility A		Utility B	
8 Gal or higher ~ below 25 Gal	Check main control room alarms, instruments, and others to see whether plant anomalies have occurred	JMA seismic intensity degree 3 or lower	Check main control room alarms, instruments, and others to see whether plant anomalies have occurred and take emergency steps as necessary
25 Gal or higher ~ below 50 Gal	Check main control room alarms, instruments, and others to see whether plant anomalies have occurred Perform round inspections in the scope of normal patrol	JMA intensity degree 4 or higher	Check main control room alarms, instruments, and others to see whether plant anomalies have occurred and take emergency steps as necessary Perform round inspections on plant equipment (excluding special access-controlled areas
50 Gal or higher ~ below 125 Gal or intensity degree 5-lower or higher	Perform round inspections in areas under special control with locks and the like in addition to class II inspections Personnel responsible for equipment will perform round inspections on plant equipment (including areas under special control with locks and the like) Perform radiation control assessment Check to see whether chemicals, instruments, tools, and others have fallen or scattered in permanent materials storage areas and job sites under control	JMA intensity degree 5-lower or higher (without reactor scram)	Check main control room alarms, instruments, and others to see whether plant anomalies have occurred and take emergency steps as necessary Perform round inspections on plant equipment (including special access-controlled areas except for reactor containment vessel interiors) Check leak detection systems, radiation monitors, and tank levels for radiation control inspection inspections Perform performance testing of engineered safety features and the like in accordance with the surveillance test procedure
125 Gal or higher ~ below 250 Gal	Perform class III inspections Perform safety function check I Perform reactor safety assessment Perform radiation control assessment	JMA intensity 5-lower or higher (in the event of reactor scram triggered by seismic sensors)	Check main control room alarms, instruments, and others to see whether plant anomalies have occurred and take emergency steps as necessary Perform round inspections on plant equipment (including reactor containment vessel interiors) Check leak detection systems, radiation monitors, and tank levels for radiation control inspections Perform performance testing of engineered safety features and the like in accordance with the surveillance test procedures Perform functional testing of safety critical equipment in accordance with the surveillance test procedures or the functional test procedures

“7.11. Both post-earthquake operator actions and automatic scram should be based upon a proper set of parameters derived from the recorded data and suitably processed, with two main goals:

(1) To avoid spurious signals;

(2) To provide an indicator of damage for comparison with the assumptions made at the seismic design phase.”

While the SDAS is designed for post-earthquake data analysis, the SAS and the ASTS are characterized in that they need to be capable of processing data in real time and transmitting accurate information during the earthquake.

Matters to pay attention to, with regard to triggers and main control room alarms/indications, are discussed in the following sections.

2.3.2.1. SDAS start-up trigger

Trigger setting level

The trigger signal to start-up the SDAS is also related to the definition of a ‘felt earthquake’. It may also be used as an alarm or annunciator signal to indicate in the main control room that the SDAS has started up.

In general, seismic switches detect acceleration, and the regulatory requirements related with trigger acceleration setting values are different from one Member State to another, as shown below.

United States of America

In the United States, 0.01 g is specified as the maximum setting value for the seismic trigger [6]. The frequency range of the seismic trigger needs to include the range of 1 to 10 Hz.

Germany

In Germany, the regulations on settings are [7]:

– The trigger in the reactor building is to be set to 0.1 m/s^2 or below. The trigger in the free field will be set to 0.2 m/s^2 or below. When the SDAS starts up frequently, the trigger location needs to be altered. (Changing the setting value is the last resort to spurious signal prevention).

– The limit value of a seismic switch will be set to the acceleration corresponding to the maximum acceleration value analysed or specified for the inspection of the setting location.

India

Depending on the following requirement on the trigger value, it is commonly set to 0.01 g at all plants, according to the results of investigations.

Regulations on setting values (Atomic Energy Regulatory Board (AERB) /SG/S-11):

– The trigger level will not exceed 0.02 g (free field ground surface). One trigger will be able to start-up all instruments.

China

As a trigger to start-up the SDAS, the same seismic switch used for sending an alarm to the main control room is used, and the trigger level is set to 0.01-0.02 g. The seismic instrumentation system configuration of a typical nuclear power plant is described in Table II-5 of Annex II.

France

Règle Fondamentale de Sureté I.3.b requires an alarm in every unit control room and time-histories recorded after the onset of a significant earthquake (greater than 0.01 g acceleration). Triggers are implemented [8]:

– At the floor of the reactor building

– At the floor of another building containing systems important to safety and whose foundations are different from those of the reactor building

Japan

Since an ASTS is installed for safety purposes, SDAS is considered as a voluntary system for structural analysis purposes. Although, the start-up (trigger) level is not particularly specified, the setting range of a trigger built in conventional seismic instrumentation systems is variable between approximately 0.1 Gal and 100 Gal. Utilities make decisions based on the intended use and the level of seismicity. As shown in Table 5, it is clear that these setting levels are generally lower than those of other Member States. As mentioned in Section 2.1.4, in Japan the reference seismic motion is defined on a (hypothetical) free surface of baserock, and great importance is attached to measuring points within buildings. Because the SDAS is intended to record the dynamic behaviour of the ground and the structures, it can be seen that SDAS will start-up upon a lower acceleration level when obtaining data from inside the instrumented boreholes in the ground.

Simultaneity of SDAS start-up timing

As shown in Section 2.2.3.1, SDAS sensors are installed at various locations. To examine dynamic behaviours of structures, it is desirable that these sensors are synchronized. The following functions are required from the seismic switch as a trigger of the SDAS:

– The SDAS's horizontal and vertical acceleration time-history recording need to be started up simultaneously. To this end, one or more seismic switches are to be installed [6].

– The SDAS is to be trigged by both the vertical as well as the horizontal seismic excitation. Data recording will start as soon as the data recording threshold is exceeded. In addition, recording need to be continued for at least 30 seconds after the last exceedance of this threshold. A medium capable of storing 30 minutes of data after the trigger start-up is to be used [7].

– The circuit design is to be able to start-up all acceleration sensors and recorders with one trigger. For SDAS acceleration sensors, one alarm-sending seismic switch and one triggering seismic switch is to be designed for each building [7].

TABLE 5. EXAMPLE OF TRIGGER SETTING LEVELS AND OBSERVATION POINTS IN JAPAN

Utilities	Utility L	Utility M	Utility N
Trigger setting level and its position	1 Gal within rock (deep part) 4 Gal on reactor building base mat	3 Gal at reactor building	1 Gal within rock (shallow part) directly under the building 5 Gal on reactor building base mat

Indications in main control rooms

Operators in the main control room are to be announced that the SDAS has started up (start-up signal from the SAS).

– When time-history acceleration data comes from a free ground surface or from a foundation level, the trigger needs to be announced in the control room. When a multi-unit site has more than one main control room, it needs to be announced in every control room [6].

Recording of trigger signals

As mentioned in Section 2.2.3.1, SDAS data is important for the clarification of the plant’s transient phenomena after an earthquake. Thus, the function to clarify the connection between SDAS data and transient phenomena is required.

– The time of trigger occurrence is to be recorded [4].

Dynamic characteristics

In view of the purpose of seismic switch installation, frequency band-pass filter is advisable for protection against noise.

– In the frequency range from 0.1 to 30 Hz no resonances are allowable in the sensors [7]. The amplitude frequency response shall not deviate by more than ± 1 % from the amplitude setpoint. the phase frequency response may not deviate from the set point by more than ± 2 % [7].

– To suppress influences not caused by earthquakes, the seismic trigger needs to incorporate amplitude attenuation above 10 Hz (e.g. a low-pass filter with a cut-off frequency of 10 Hz [7]).

– Bandwidth of triggers may be limited to the range of 0.1 to 10 Hz to avoid spurious tripping by phenomena other than earthquakes [8].

2.3.2.2. *Control room alarms and annunciations*

In IAEA Safety Standard NS-G-1.6, seismic alarms for operators at a plant without ASTS are described as shown below [2].

> "(b) For seismic monitoring: to provide alarms for alerting operators of the potential need for a plant shutdown depending on post-earthquake inspections."

The following description in Kerntechnischer Ausschuss (Germany) 2201-05 is important for the function of SDAS and it is considered applicable not only to nuclear power plant without ASTS, but also with ASTS [7].

> "5. (4) The following alarms shall be documented in the main control room or in a control room annex:
>
> (a) Actuation of data measurement and recording,
>
> (b) Actuation of any of the alarm triggers,
>
> (c) Loss of the external power supply to the instrumentation specified in Section 3.
>
> These alarms shall be interconnected to initiate a group alarm that shall be optically and acoustically announced in the main control room".

To this end, it is recommended that the SDAS panel is placed in the main control room or in its vicinity. Procedures need to prescribe that operators will check the DIPs provided by the SDAS (for instance, CAV or instrumental seismic intensity), when there is a seismic alarm.

Both functions required of SAS, that is, starting up SDAS and giving warning to operators, use the mechanism of a seismic switch relying on acceleration sensors instrumentation. On the other hand, as detailed later in Section 3 and 4, the peak acceleration value is not necessarily related to seismic damage to SSCs. To allow operators determine the 'earthquake intensity' more precisely, alarms/annunciations based on acceleration may be combined with DIP indications. In Japan where JMA instrumental seismic intensity prevails, instrumental seismic intensity meters and the like are used for general industrial purposes.

In the event of an earthquake, an alarm or annunciation will be activated in the main control room upon the three earthquake levels listed below.

- SDAS trigger signal level (See Section 2.3.2.1)
- Earthquake level requiring a decision for manual reactor shutdown (see Section 2.3.1.2 for a plant without ASTS)
- ASTS activation level (at a plant with ASTS)

The second alarm level is particularly applicable to nuclear power plant without ASTS and it is specified as described below in IAEA Safety Standard NS-G-1.6 [2].

> "7.7. The lower trigger level (alert) should be close to SL-1 (usually associated with operational limits), at which significant damage to safety items is not expected. If the

> overall seismic capacity of the plant is lower than SL-1 (e.g. during the seismic re-evaluation), the lower trigger level should be referred to the actual seismic capacity of the plant."

In KTA 2201-05 (Germany), this setting level is specified in connection with walkdown inspections as shown below.

> "5. (3) The threshold values for alarms shall be adjusted to the acceleration limit values that correspond to the maximum accelerations specified or calculated for the inspection levels at the respective placement locations."

There are some cases, for instance in China and France, in which both functions of SAS are performed by treating an SDAS trigger signal as an alarm. On the other hand, Table 6 shows some cases in other Member States in which an alarm is activated at an earthquake level different from a trigger signal.

ASTSs have been installed in Japan. In this regard, when an earthquake motion exceeds a pre-set trigger level, based on the design basis earthquake motion, ASTS activation is indicated as an alarm in the main control rooms. Furthermore, utilities set the main control room alarms and annunciations separately depending on the level of seismicity at their sites. Some instances are shown in Table 7. In Japan, as discussed in Section 2.3.1 (c), on-site inspection classes have been specified according to the level of earthquake motions observed to notify operators that the SDAS has started up and have them recognize the earthquake motion intensity. As shown in the column for Utility Z in Table 7 there is an instance in which earthquake early warnings are used as alarms during earthquakes, which is detailed later in Section 2.3.5.

2.3.3. Important considerations for seismic alarm/annunciation systems

2.3.3.1. Prevention of spurious signals

As mentioned in Section 2.3.2, it is strongly required for a SAS to prevent malfunctions. When both triggering and alarm signals are not transmitted due to malfunction, safety might be compromised.

At nuclear power plants without ASTS, in particular, signals need to be transmitted appropriately because they have the important function of assisting with the decision to continue reactor operation after an earthquake. On the other hand, spurious signals will waste the valuable time of operators in the aftermath of an earthquake. Spurious triggering needs to be avoided [6].

The following measures are suggested to ensure correct information transmission and prevent spurious signals:

– Prevent acceleration noise with band-pass frequency filters;

– Prevent impact noise by installing protective covers;

– Ensure redundancy with multiple sensors.

Spurious signals occur typically due to local vibrations, impact load from surrounding objects, plant transients, and instrument malfunctions. Protection of seismic instrumentation against

TABLE 6. MAIN CONTROL ROOM ALARM LEVELS (INSTANCES IN WHICH ALARM LEVELS ARE DIFFERENT FROM SDAS TRIGGER LEVELS)

Member States	Nuclear power plant	Main control room alarm set levels	Remarks
Hungary	Paks	0.05 g	Sensor on the containment building foundation mat
India	TAPS-1&2	0.05 g	The trigger level of the seismic switch will not exceed 50 % of PGA (SL-1 level) and the maximum value will be 0.1 g and below. (Applicable standard): AERB/SG/S-11
	RAPS-3&4	0.025 g	
	MAPS-1&2	0.04 g	
	KAIGA-1,2,3&4 (with ASTS)	0.05 g	
	KAPS-1&2 (with ASTS)	0.1 g	
United States of America	North Anna	Horizontal: 0.06 g, Vertical: 0.04 g	Sensor on the containment building foundation mat

TABLE 7. EARTHQUAKE-RELATED INFORMATION IN MAIN CONTROL ROOM (INSTANCES IN JAPAN)

Utilities	Utility X	Utility Y	Utility Z
Indication in the main control room	An instrumental seismic intensity meter has been installed in Unit 1 (two-unit site). When acceleration equal to 1 Gal or higher is detected, the seismic intensity and maximum acceleration will be printed out in the main control room (without sounding an alarm). Observed values will be displayed on the data recorder installed in the main control room.	When the earthquake warning system installed on the second floor of the auxiliary building detects acceleration equal to 3 Gal or higher, the JMA instrumental seismic intensity, intensity grade, and maximum acceleration will be printed out and an alarm will go off in the main control room and three other locations, and observed values will be recorded in the data recorder connected with the earthquake warning system.	A seismometer assigned for safety assessment is installed on the representative reactor building foundation mat and the maximum acceleration and JMA seismic intensity will be indicated in the main control rooms. When it is expected that an earthquake of seismic intensity 3 or higher will occur (early warning), the estimated intensity grade will be announced on-site (including the main control rooms) with a paging system.

accidental impact is prescribed by the regulation in some Member States [6]. It is important to prevent the effects of non-seismic events and frequency filters are recommended to be applied (see Section 2.3.2.1).

As mentioned in Section 2.1.2.2, the acceleration response spectrum within the bandwidth of 2 to10 Hz is used to determine operating basis earthquake exceedance. It is considered that mechanical equipment made of ductile metallic materials will not be damaged by accelerations at 10 Hz or higher frequencies.

Protection against external impact is also effective in protecting seismometers from noise. In particular, it is considered as an essential measure when the seismometers are part of a SAS. An example of external impact prevention measures in a Japanese plant is shown in Fig. 6.

In Member States where importance is given to main control room alarms (plants without an ASTS), redundancy is considered for the seismic switch in order to assure the objectives of the SAS. An example of redundancy in China is shown in Table II-5 of Annex II. Two accelerometers are selected as SAS sensors for redundancy.

2.3.3.2. Multi-unit sites

The concept of seismic instrumentation system at a multi-unit site is provided in Section 2.1.5. When seismic instrumentation is installed at one representative unit on a multi-unit site, however, it is necessary to send alarm/annunciate signals from the seismic switch to the main control room of all units, which is prescribed in the quoted texts shown below.

Regulatory Guide 1.12 [6]

> "Triggering of the free-field, downhole, or any foundation-level time-history recorder should be announced in the control room. If there is more than one control room at the site, annunciation should be provided to each control room."

Règle Fondamentale de Sureté (France) I.3.b [8]

> "The starting of the records will be based on triggers set to a threshold corresponding to a significant earthquake. Exceeding this limit will generate an alarm in the control room of every unit of the site."

Information about the observed earthquake motion intensity is needed not only in the main control rooms, but also in administrative buildings. In instance Z of Table 7, for example, this information is announced not only in administration buildings, but also across the plant, which is considered particularly important at multi-unit sites.

2.3.3.3. Other considerations

Testability

Highly reliable maintainability and testability are required for a SAS. It is desirable that functional testing and on-site calibration can be performed. From this perspective, the following description is very suggestive [12].

> "The system should include capability for remote, in place functional testing to verify performance during routine maintenance procedures. The sensors should be designed to allow on-site calibration using simple procedures and tools. Sensors with DC response usually offer simpler calibration procedures (tilt vs. shake table testing) and are recommended for this reason. Procedures for zero offset adjustments of DC sensors should be easily undertaken by plant personnel without specialized training."

Appearance/Seismometer sensor inside the cover

FIG. 6. Example of external impact prevention measures for seismometer sensors in Unit 2 reactor building of Shika nuclear station. (Courtesy of Hokuriku Electric Power Company)

Power supplies

A loss of off-site power is anticipated in the event of an earthquake. It is recommended that uninterruptible power supplies need to be used to fill the window time until the emergency AC power supply system starts up.

On the other hand, especially in low seismicity areas like France, there are no requirements on electric power supply. However, the regulation requires mechanical records of peak acceleration instead, saying that some complementary devices, autonomous, simple, without any requirement of power supply, need to provide approximate maximum values of felt accelerations at various locations of the specified structures [8].

Seismic and quality requirements

See Table 1 in Section 2.1, General Considerations on Seismic Instrumentation for Decision Making.

Arrangements

Depending on the seismic instrumentation system configuration, the SDAS may also function as seismic switch. In those cases, the followings the following considerations apply to the SAS seismic switch:

– The SDAS accelerometer and the SAS seismic switch will normally be oriented such their axes are parallel to the axes of the coordinate system used for the seismic analysis of the building [7].

– (When the SDAS also functions as SAS) One SAS seismic switch need to be specified for each building from among several SDAS acceleration meters.

The reliability of seismic switches is important, and maintainability needs to be taken into account:

– Seismic switches need to be accessible as needed for operation and maintenance [7].

Digital or analogue

For SDAS, digital systems are recommended because of the necessity of prompt data processing. On the other hand, SAS functions as a switch, for which reliability and maintainability are important. Thus, for SAS, mechanical systems (analogue) can be employed as well as digital systems.

Area network information

Information from regional emergency preparedness networks is important as well as on-site earthquake observation. It is required to obtain accurate earthquake information by actively using off-site network information.

2.3.4. Lessons learned and other observations

2.3.4.1. Emergency electric power supply

The SAS has the function of providing alarms and/or announcements in the main control room. It needs to be connected to uninterruptible power supplies. Other panels installed in and around the main control room, for instance, indicator panels, need to have power supply configurations well balanced with the SAS.

An instance occurred, in which no alarms were activated, although the emergency diesel started up upon a loss of off-site power after an earthquake. The reason was that portions of the seismic instrumentation panel in the control room were not connected to an uninterruptible power supply and therefore they were not functioning during the brief power outage (about 8 seconds) while the emergency diesel generators started and loaded following the loss of offsite power. This resulted in an inability of the plant operators to promptly determine if the ground motion levels exceeded the SL-1 earthquake levels, since the about 3 seconds of strong ground motion of the earthquake in each of the three orientations occurred during the 8-sec power outage [19].

It is advisable that alarms and indicating lights on control room panels use uninterruptible power supplies.

2.3.4.2. Instrument calibration

An event occurred at a nuclear power plant with mechanical seismic switches (SDAS triggers), in which the plant's SDAS did not start-up even though the earthquake motion recorded by the seismometer in the vicinity of the power plant significantly exceeded the trigger setting value. As a result of the investigation of causes, it was found that all sensor masses of the triggering unit were misaligned. The masses were locked on their stops in all three orthogonal directions and could not trigger at the 0.01 g setpoint. The calibration of the triggering unit had been performed in the laboratory and the unit was then returned to the field location for re-installation. It seemed that the sensor masses became misaligned sometime between calibration and re-installation. Post installation testing consisted of mechanically agitating the trigger unit, which did not detect the misalignment [19].

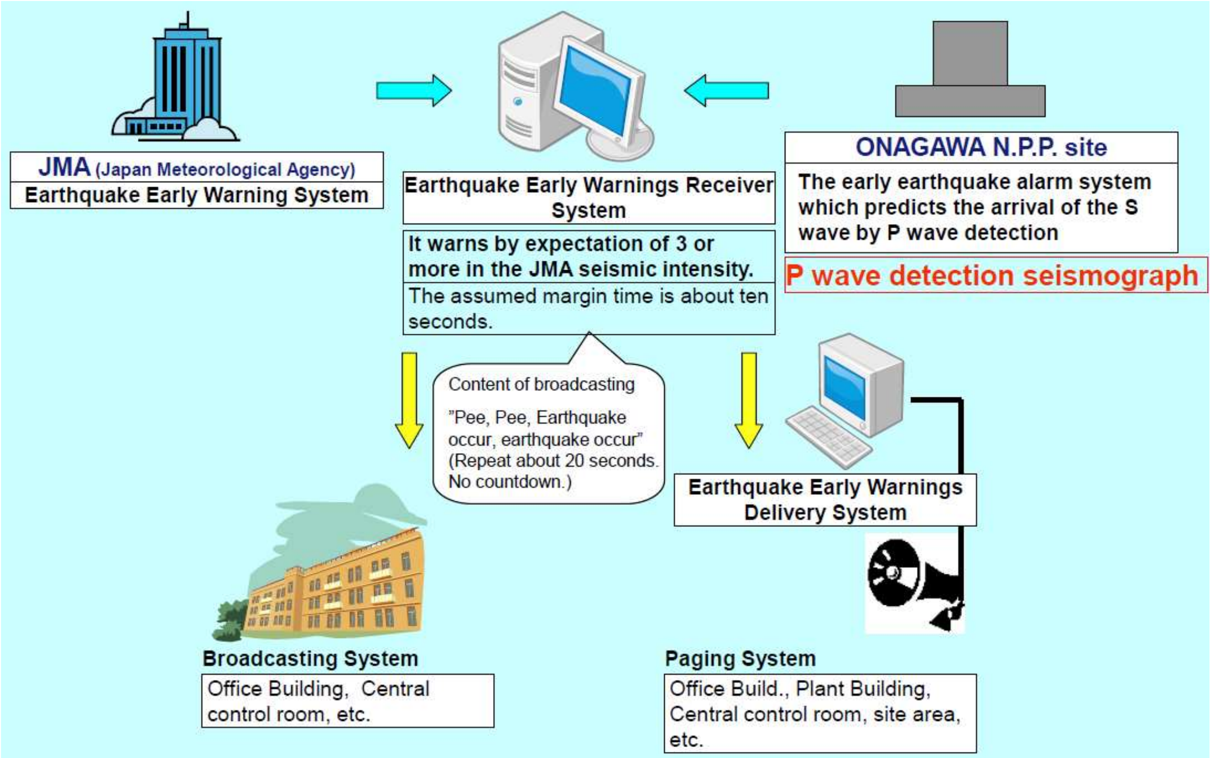


FIG. 7. Schematic view of the configuration of earthquake early warning system at Onagawa nuclear power plant [21].

As discussed in Section 2.3.3.3 Testability, it is desirable that on-site calibration can be performed at the locations in the field where a SAS is installed. In this instance, the functional inspection method after resetting was improved.

2.3.5. Future requirements and recommendations

2.3.5.1. Introducing DIPs

Acceleration is used for the SAS controlling value, for which real time processing is required. To prevent malfunctions and provide operators with an accurate earthquake motion severity, however, it is desirable that DIPs superseding acceleration are introduced (See Section 4.6.1). It has become a challenge to develop appropriate DIPs, calculated in real time and included in the SAS.

In Japan, where JMA instrumental seismic intensity is employed, measures have been taken to indicate both the acceleration peak value and the JMA seismic intensity value. The level (details) of walkdown inspections are decided based on them.

2.3.5.2. Introducing an earthquake early warning system

In Japan, the Meteorological Agency gives earthquake early warnings with a system that capitalizes the difference between the P wave and the S wave arrival times, to prepare the entire society against earthquakes. Similar systems are used for emergency braking systems in trains, as an example of its industrial use. In the area of nuclear power generation, Tohoku Electric Power Company voluntarily installed a similar system to give warnings on-site (See Fig. 7).

The system is activated when an earthquake of JMA seismic intensity of 3 or higher is expected at Onagawa site, based on the information provided by Tohoku Electric Power's own seismometer, which is capable of predicting the arrival of the S-wave, or on earthquake early warnings distributed over the Internet by the Meteorological Agency. When the system is activated, the occurrence of an earthquake is announced across the site over the on-site paging system and in the administrative building. The seismic intensity scale is displayed, and the data is recorded on the computer installed inside the administrative building [20].

2.4. AUTOMATIC SEISMIC TRIP SYSTEMS

2.4.1. Introduction

This section discusses the decision-making considerations and the current status of ASTSs, mainly based on the discussions carried out within the working group during the preparation of this publication.

IAEA Safety Report Series No. 66 comments on the ASTS as follows:

> "The link between the perception of an earthquake (a felt earthquake) and the consequential actions to be taken by the staff in the control room of an operating nuclear power plant may be basically established by using one of the two available approaches:
>
> (1) Manual actions, i.e. shutdown initiated by operator action; or
>
> (2) Fully automatic actions at a certain preset level of recorded motions.
>
> Both approaches present advantages and limitations with regard to the response time, reliability and safety. The experience and regulatory practices of Member States in relation to the selected approach are quite broad, depending on a number of issues."
>
> "In some States, safety regulations or operating procedures mandate that nuclear power plants install an automatic shutdown system that is triggered when earthquake motions at the site exceed a predetermined level. This is the case in Japan, an area of high seismicity. Other areas of high seismicity may also require automatic shutdown systems. In the United States of America, although no specific regulatory requirements impose the installation of automatic shutdown systems, power plant units located in areas of high seismicity, for example, California, have installed and operated them, for example, the Diablo Canyon nuclear power plant. Automatic scram systems are installed in some nuclear power plants of the former Soviet Union design, including those located in zones of low seismicity. There are also States in which such a system is not mandatory, or the safety regulations do not address it. States with less experience in the nuclear power industry generally prefer to follow the practice of the States from which the nuclear steam supply system comes."

IAEA Safety Standard NS-G-1.6 describes the key issues which govern the decision whether to have an automatic scram system or an operator action, as follows [2].

(a) "The level, frequency and duration of earthquake activity at the nuclear power plant site: an automatic system is rarely justifiable for sites in areas of low seismic activity.

(b) The seismic capacity of nuclear power plant systems: automatic systems should be used as an additional protective measure, particularly in the case of upgrading of the seismic design basis.

(c) Safety considerations relating to spurious scrams: an automatic system should not be used for places with high levels of ambient noise, including noise induced by other plant equipment.

(d) Evaluation of the effects of the superposition of earthquake acceleration on the seismic transient induced by an automatic scram. In some cases, such a combination may be more challenging for plant safety than the scenario with an earthquake affecting the plant in full operation.

(e) Broad ranging safety issues relating to the consequences for the State of the shutdown of a plant immediately following an earthquake. In States with a limited electricity grid and few seismically qualified power generation plants, the availability of power in an emergency could be essential, and an automatic scram should therefore be used only if it is ascertained that there is a challenge to the safety of the plant.

(f) Level of operator confidence and reliability: for a non-automatic system, the operator plays an important part in the decisions on post-earthquake actions and therefore should be adequately trained for this contingency."

And IAEA Safety Report Series No. 66 adds two more key issues:

(g) Other reactor trips;

(h) Public acceptance.

Historically, the basic objective of installing an ASTS was the so-called 'prediction' of the necessity to shut down the nuclear reactor, as a safety measure against the earthquake motion, when the reactor scram capability, 'scrammability', was in doubt [22]. However, after experiments verified the 'scrammability' during an earthquake, other objectives and merits/demerits have been discussed for a long time. In 1995, a special consultants meeting on ASTS was held in IAEA, and discussions are summarized in the IAEA Working Material [23].

The issue that needs to be considered first when operators discuss on the necessity of an ASTS is the capability of control rod insertion during strong earthquake motion following a reactor trip signal. The control rod insertion time into the reactor core is delayed due to the relative deformation of the core induced by the earthquake force. Fig. 8 shows the typical trend of control rod insertion time delay experienced in the proof tests and analyses done in Japan.

The specified control rod insertion time limit is decided considering plant transients after the reactor protection system detects any symptom of abnormality.

The time sequences in case of reactor scram originated by an ASTS or by a reactor protection system during an earthquake are compared in Fig. 9. In view of the figure, an ASTS is considered to give some 'margin' for the reactor scram time against the events due to the earthquake, and possibly mitigate the transients. This margin, 'early scram' in other words, allows reducing transient pressure and loads and the heat generation rate in the core.

However, it is usually considered that the advantages of an 'early scram' are not so important to decide the adoption of an ASTS [24–25]. This is the case when the scram time during the earthquake satisfies the safety requirements due to a proven high 'scrammability' of the reactor design, or when expected earthquake motions are relatively small.

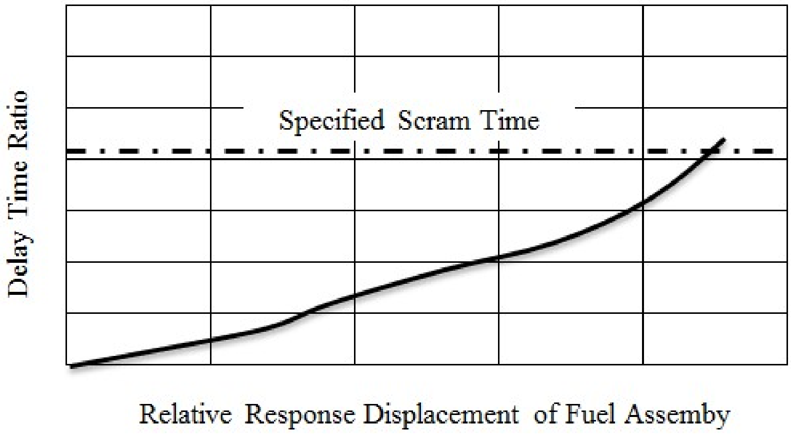


FIG. 8. Trend of delay time of control rod insertion against earthquake motion (delay time ratio = delay time / initial insertion time).

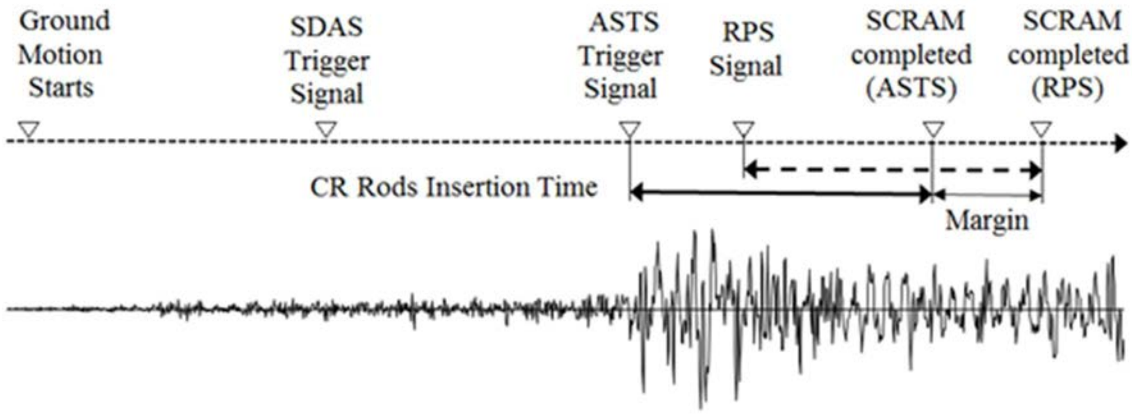


FIG. 9. Seismic scram and reactor protection system (RPS) scram.

2.4.2. Considerations on manual shutdown vs. automatic seismic trip systems

2.4.2.1. General

There are many factors to consider when deciding whether an ASTS is needed or it is appropriate for a nuclear power plant. The following is a summary of such considerations. The advantages and disadvantages are documented in Refs. [1–2] and [23–24], and summarized as follows:

(a) The level, frequency and duration of earthquake activity at the nuclear power plant site.

An automatic system is rarely justifiable for sites in areas of low seismic activity. Moderate to high seismicity areas are more likely to justify an ASTS.

(b) The seismic capacity of nuclear power plant systems and the annual frequency of the design basis earthquake ground motion occurring at the site.

Automatic systems could be used as an additional protective measure, particularly when the seismic hazard at the site or the seismic design basis have been increased.

(c) Safety considerations related to spurious scrams.

An automatic system is not appropriate for sites with high levels of ambient noise, including noise induced by other plant equipment. Spurious scrams may have a negative impact on the perception of the public on the reliability of the plant, especially if it leads to a loss of electricity in the public's daily life.

(d) Effects of the superposition of earthquake acceleration on the seismic transient induced by an automatic scram.

In some cases, such a combination may be more challenging to plant safety than the scenario of an earthquake affecting the plant at full power.

(e) Broader range issues specific to the Member State, if the plant shuts down immediately following an earthquake.

In Member States with a limited electricity grid and few seismically qualified power generation plants, the availability of power in an emergency could be essential, and an automatic scram therefore needs to be used only if it is ascertained that there exists a challenge to the safety of the plant.

(f) Level of operator confidence and reliability.

For a non-automatic system, the operator plays an important role in the decisions on post-earthquake actions and therefore needs to be adequately trained for this contingency.

(g) Operator acceptance and appreciation.

For large ground motions, a time of high stress, with many events occurring on-site and off-site (such as concerns for family and relatives), operators may appreciate the decision to shut down being automatic.

(h) Public acceptance.

Public acceptance is an important aspect, which may influence the decision on the approach to adopt. It needs to be noted that the installation of an automatic trip system may be perceived either positively as an additional safety system or negatively as a lack of confidence in the seismic design level and the seismic safety of the installation. Public opinion depends heavily on the level of experience and education of the population with regard to seismic events. The impact of spurious trips, if perceived directly by the public due to a perturbation in the supply of electricity, will probably impact negatively on the public perception of the reliability of the plant.

2.4.2.2. Approaches to decision-making

The following approaches to decision making are summarized in this section: (1) evaluation of the advantages of an ASTS as a function of demonstrated 'scrammability' of the nuclear reactor during earthquake shaking; (2) overall risk; and (3) lessons learned from actual earthquakes.

An ASTS needs to be carefully discussed especially for nuclear power plants in low seismicity areas because the effectiveness of automatic seismic scram is dependent on the site-specific seismic conditions (e.g. expected strong motion duration time) and on the reactor type (e.g. scram delay time).

Scrammability

The term 'scrammability' refers to the demonstrated ability of the control rods to be inserted during the earthquake shaking of the plant. This phenomenon encompasses two aspects of plant shutdown. The first is the ability of the control rods to be inserted into the core in the required time, taking into account the seismic demand imposed on the core, control rods, and control rod insertion system. This seismic demand includes relative deformations of the core on the control rod channels. The second aspect is the load combination of the reactor scram and loads imposed by transients or other events. Both aspects are important.

An automatic scram system set at trigger levels less than the SL-1 or SL-2 is expected to cause control rod insertion to be completed before strong shaking occurs and before loading conditions from consequential events are imposed on the core, control rods and control rod insertion system.

Overall risk

A pilot study employing a systematic approach to assess the benefits of an ASTS from a risk perspective was performed in 1981 by the Lawrence Livermore National Laboratory for the US NRC [24]. The approach was to use available models and data to assess the change in a risk metric - core damage frequency – when an ASTS is installed at a plant. Existing information on plant accident sequences and plant behaviour was extracted from safety analysis reports and used in conjunction with probabilistic safety assessment systems models to assess the advantages of an ASTS. Specific information on timing of loading conditions induced in systems due to reactor trips was used along with the timing of earthquake induced stresses and deformations to assess the impact. Also, the required time for control rod insertion played a role. A realistic analysis for a specific site and plant requires site specific and plant specific data. Such data was not available to the study and generic data for a hypothetical plant was used.

A decision-tree modelling approach was used to compare the risk (core damage frequency) when employing an ASTS with the risk when not employing an ASTS. For the hypothetical plant, and using data from many sources, the results showed that an ASTS would reduce the frequency of an earthquake induced core damage event by about a factor of three. Partially off setting this advantage was the disadvantage of inadvertent reactor trips.

The study performed in [24] is thirty years old. All aspects of nuclear power plant design have significantly changed in the following three decades. The approach employed in [24] is applicable to decision making, but studies need to be performed on existing sites and plants with site and plant specific data.

Lessons learned from actual earthquakes

Earthquake induced automatic reactor shutdown at a plant without an ASTS

In the Virginia earthquake on August 23, 2011, both units of the North Anna Nuclear Generating Station (a two-unit plant with pressurized water reactors) experienced an emergency automatic reactor shutdown. This was the first experience of automatic pressurized water reactor shutdown due to an earthquake. It was initially thought that the shutdown was a result of the loss of off-site power, but the analysis of plant records found that the shutdown was initiated with a 'High Neutron Flux Rate' signal from the reactor protection system (See Section 4.4.2 and Fig. 47). It remains unclear why there were changes in the neutron flux rate. However, detailed post-earthquake fuel inspections with the reactor pressure vessel open showed no sign of damage in the reactor core internal structures or fuel assemblies. Around three months later, the plant resumed its operation and has since operated with no issues.

This is a good example that even without an ASTS, automatic reactor shutdown can be achieved with safe control rod insertion in the event of earthquake motion, when the insertion is triggered by a reactor protection system signal after a plant abnormality is detected.

ASTS-initiated automatic reactor shutdown and operators' behaviour

In Japan, automatic reactor shutdown by ASTS has been repeatedly experienced since the first such an automatic shutdown of the Onagawa Nuclear power plant, affected by the 2005 Miyagi-Oki earthquake. On the other hand, there has never been any spurious automatic reactor shutdown attributable to the ASTS.

In the main control room of a nuclear power plant struck by a strong earthquake, many alarms sound and indicator lights flash to annunciate anomalies caused by instruments which are sensitive to acceleration or its effects, for example, water level sensors. Operators in the room try hard to manage plant operation in such an upset condition. They are also possibly exposed to events accompanying the initial strong earthquake, such as aftershocks and tsunami. The Onagawa nuclear power plant, at which a tsunami was experienced approximately 40 minutes after the earthquake, was placed in far more severe conditions. As seen from these observations, ASTSs need to be re-evaluated from the perspective of operators' emergency actions. The following paragraphs describe the experienced operators' behaviour at nuclear power plants with an ASTS.

At the main control room of Kashiwazaki-Kariwa nuclear power plant in the 2007 Niigata-ken Chuetsu-Oki earthquake

The first example of a lesson learned from an actual earthquake is the experience of the shift supervisor, Kashiwazaki-Kariwa nuclear power plant, Unit 4, when the Niigata-ken Chuetsu-Oki earthquake occurred on 16 July 2007 [26].

The shift supervisor wrote his experience as follows:

> "I immediately believed that the plant would undergo automatic scram, i.e. emergency automatic reactor shut down. In order to hold myself, I stretched my arms to hold on to my desk as I kept sitting in the chair. Other members of the crew who were standing and talking in front of me crouched down also. A few seconds later we recognized emergency shut down of the plant with 'Automatic scram alarm' signal on the control panel from two independent systems. Without waiting for my instructions, the main equipment operator dashed to the control board to confirm all control rods inserted signal and neutron flux level drop indicating nuclear fission reactivity level drop. He firmly and clearly said, 'Scram succeeded'. As I looked back, thanks to his strong voice, 'Scram succeeded', everyone regained confidence and calmness, and went about his own business for plant recovery operations".

In summary, the shift supervisor praised the ‘Success of the Scram’ as it promptly shutdown the reactor and consequently calmed the operating staff, which led to their being able to perform their other duties in a calm and professional manner.

The shift supervisor and staff were commended by the Japan society of mechanical engineers, for their outstanding contribution to the reactor safety.

<u>At the main control room of the Onagawa nuclear power plant in the 2011 off-the-Pacific-coast-of-Tohoku earthquake</u>

The plant operation division of Tohoku Electric Power Company, whose plant has experienced the largest earthquake ground motions and also post-earthquake tsunami, provided the following responses to questions:

(a) What did the operators in control room feel at this earthquake?

“We could not have presence of mind due to the tremble situation, which we have not experienced before. But we had been carefully and adequately trained, incorporating the lessons learned from the Chuetsu-Oki earthquake, and so there was no panic in the control room personnel. During the earthquake, it was impossible to stand up and all what we could do was to protect ourselves. Tremor was so severe that it seemed to destroy the central control room. Even in this severe earthquake, we thought the reactor was able to stop safely due to counter measures to strengthen earthquake resistance. We had the confidence that nuclear power plants had been proven to withstand the earthquake.”

(b) Was there any sense of fright for the long duration of earthquake? Was there any sense of security with the reactor scram by the ASTS?

“Three nuclear reactors were automatically scrammed by sense of the large seismic acceleration and we confirmed all control rods were inserted in reactor cores. We confirmed a sense of safety that we had been able to stop the reactor safely. By the long duration of huge shaking, we felt the danger in the condition of plaster boards and fluorescent lights falling from the ceiling and, scattered things on the desk.”

(c) Were operators able to focus their attention in the catastrophic condition?

“Since the reactor automatic scram worked fine, we were able to check the status of the plants evenly. We had been carefully trained to incorporate the lessons learned from the Chuetsu-Oki earthquake, we could control and operated Units 2, 3 without panic condition until the tsunami coming. However, at Unit 1, fire occurred at metal clad switch gear, fire-extinguishing correspondence was serious.

After the tsunami coming, operations became more complex because we had to deal with the flooding at parts (non radiation controlled area) of Unit 2 reactor building and seawater pump stopped. At the end, we succeeded in stopping reactors, verifying the plant condition and putting 3 plants in cold shutdown condition.”

(d) After the earthquake, how did you perform the walkdown process in the plant at Onagawa site?

“After the earthquake, operators walked down the plants, and found the flooding in the basement at No. 2 reactor building and smoke emitted from a normal metal clad switch gear in the Unit No. 1. Maintenance personnel were temporarily evacuated to higher level area, when tsunami warning had come out. After that, maintenance personnel dealt with the above-mentioned incidents (flooding and fire). Walkdown by the maintenance personnel after the earthquake occurred was carried out from March 14 in the morning, because it was judged that carrying out walkdown under the aftershocks that have occurred in succession was dangerous."

(e) How the tsunami warning was addressed at the power plant?

"After the tsunami alarm, control room operator sent the evacuation instructions to the site workers with paging. In addition, headquarters personnel, except the emergency treatment ones, were evacuated to a high elevation place (maintenance centre). With intention to withdraw the workers from the controlled area of site as soon as possible, several radiation control personnel were dispatched to controlled area and evacuation was done safely."

As mentioned in these interviews, wide-ranging actions are required immediately by operators after an earthquake and other events that incidentally occur create stress. In addition, as in the case of the 2011 off-the-Pacific-coast-of-Tohoku earthquake, additional actions were required to deal with tsunami, even though the reactors had been appropriately shut down by the ASTS. This means that manual operation was required by the operators after the reactor scram was completed before the arrival of tsunami. So, the ASTS has an advantage in reducing burden on operators.

Note: Consideration on the operators' behaviour in Japan (example)

In Japan, operators' behaviour in the event of a strong earthquake has been discussed for a long time. As a measure thereof, some power stations have installed seismic handrails on the control panels, in the main control room (See Fig. 10). In the event of a strong earthquake, these handrails prevent operators from inadvertently touching the switches on the panels. They are also expected to help operators stabilize their bodies and gain a sense of mental security by holding them. In the event of a strong earthquake, moreover, system monitoring devices, for instance, water level sensors, will be activated and a rather abnormal atmosphere will prevail in the main control room as many buzzers sound and indicating lights go on and off. Operators are trained mentally through drills anticipating conditions created by strong earthquakes.

2.4.3. Types and characteristics of automatic seismic trip systems

As seismometers for ASTS, seismic switches are often used. In particular, when an ASTS is used as reactor protection system, multiple seismic switches are combined for redundancy, and a system is configured to ensure high reliability and seismic capacity.

On the other hand, an ASTS that is combined in a SDAS is also being contemplated, along with the development of digital technology. As a digital ASTS, a seismic instrumentation system featuring the functions of SDAS to SAS and ASTS is proposed. The issues of such a system would be the variation in required reliability and seismic qualification requirements for each of the functions, as shown in Section 2.1.3.

FIG. 10. Handrails installed at the control panel in main control room at Onagawa nuclear station. (Courtesy of Tohoku Electric Power Company)

In order to ensure automatic reactor shutdown at the pre-set level of seismic motions using ASTS as reactor protection system, it is necessary to meet requirements for reactor protection systems, such as redundancy. The system needs to also have an appropriate number of measurement points at appropriate locations and incorporate countermeasures against spurious operation. In general, the following considerations are important for an ASTS:

- Robustness and reliability;
- Safety requirements;
- Measurement type of sensors; electric or mechanical;
- Directions of recorded motions and their combination method;
- Frequency range or bandwidth from the viewpoint of countermeasures to noise;
- Interface with plant operation and safety systems;
- Redundancy and separation;
- Location of sensors.

Instances in Japan

Since the ASTS does not have to record acceleration time-history data, either mechanical or electrical seismometers are used as seismic switches for the ASTS. The characteristics of typical mechanical and electrical seismometers used for ASTSs are shown in Table 8.

Seismic switches utilized for ASTSs require very high reliability. Since the ASTS system is categorized as reactor protection system, it is designed with measures against spurious reactor scram, testability, operability, etc. (see Section 2.3.3). Due to these considerations, the mechanical type switch as shown in Fig. 11 is widely installed in ASTSs. This kind of switch has demonstrated good and safe performance, without any malfunction, for a long time.

These seismic switches are horizontally omnidirectional, and their frequency range is set with a maximum of 10 Hz, to avoid detection of noise. Mechanical ones can be calibrated only by adjusting gaps as shown in the operating principle. This is a significant advantage in terms of reliability, since the instrument can be calibrated right on the spot without transportation.

Instances in the United States of America

At Diablo Canyon nuclear power plant, triaxial seismic switches are used.

Instances in the Russian Federation

In the Russian Federation, a seismic switch is used for reactor automatic shutdown in the analogue seismic instrumentation system named SIAZ, in pressurised water reactors (VVER).

TABLE 8. TYPICAL TYPES OF SEISMIC SWITCH UTILIZED FOR ASTS IN JAPAN

Type	Mechanical	Electrical
Sensitivity direction	Horizontally omnidirectional	Horizontally omnidirectional
Setting range	10-500 Gal	20-2000 Gal
Frequency range	DC-10 Hz	0.1-10 Hz
Analogue output	No	Yes
Outside dimensions (mm)	550W×320D×400H (depending on the model)	550W×320D×340H (depending on the model)

2.4.4. Important considerations for automatic seismic trip systems

2.4.4.1. Earthquake motion level for seismic scram

In principle, there may be two cases of ASTSs trigger motion level for an ASTS, considered in relation with the objectives addressed in Section 2.4.1, and depending on the 'scrammability' of each nuclear power plant.

– Case A: higher trigger level

    - 'Scrammability' of nuclear power plant is proved for the SL-2 earthquake motion level
    - Objectives: to confirm integrity of safety-related SSCs after an earthquake

- Trigger Level: SL-1 or SL-2, according to the design earthquake motion level of the engineered safety features

– Case B: lower trigger level

  - ‘Scrammability’ of nuclear power plant not yet proved for the SL-2 earthquake motion level
  - Objectives: to scram the reactor with larger margins during an earthquake
  - Trigger Level: predicted level of probable damage to SSCs

In Japan, as Case A, a trigger level slightly lower than SL-1, the elastically dynamic design earthquake ground motion (Sd), is recommended based on the following considerations:

(a) The engineered safety features are normally in a stand-by condition; therefore, it is difficult to prove their full functionality after the observed earthquake without functional tests. On the other hand, their integrity can be assured for earthquake motion levels smaller that the design earthquake with elastic design limit.

(b) ‘Sd’ is large enough to postulate some damage in conventional facilities, such as electric transmission lines. Therefore, automatically shutting down the reactor will have a small influence on the electricity supply (demand). In addition, the possibility of a spurious scram is small at the Sd level, considering the reliable system logic.

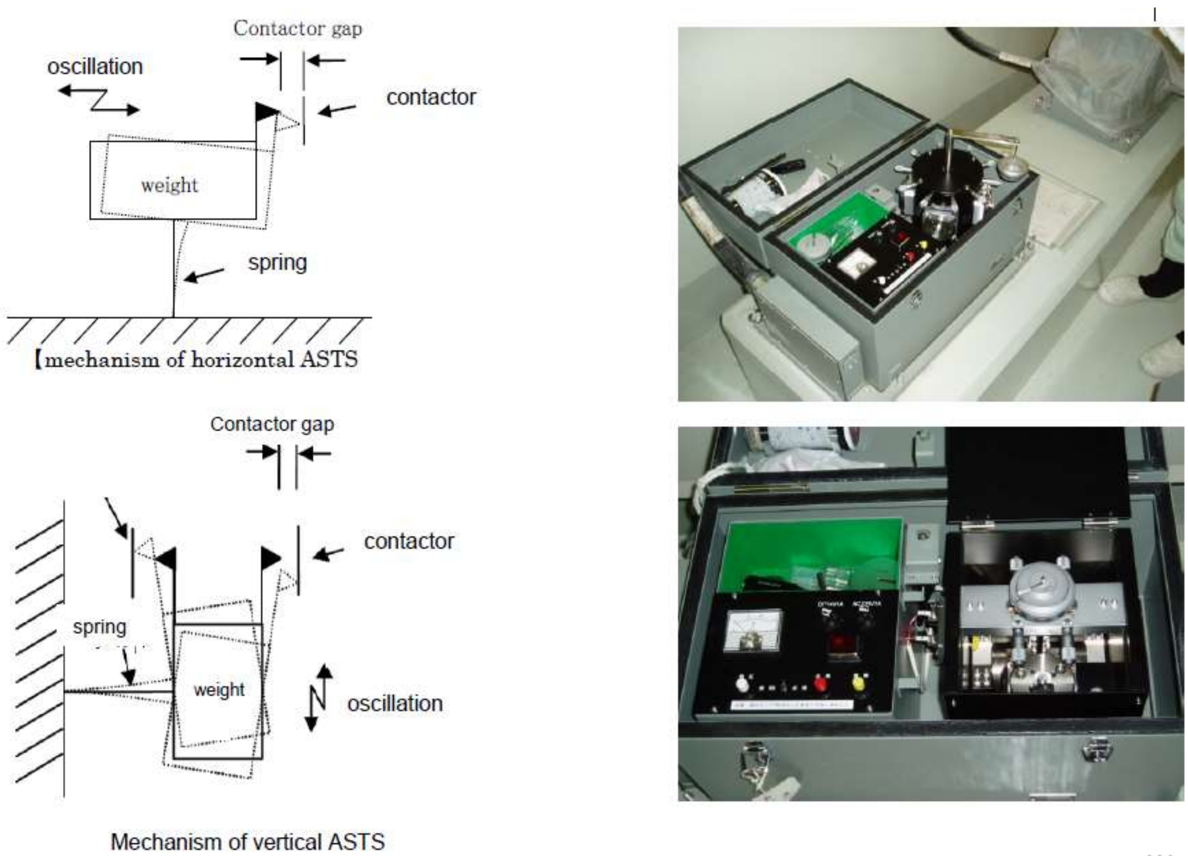


FIG. 11. Principle and exterior view of a mechanical seismic switch within an ASTS (example in Japan).

In Japan, the first commercial-use reactor introduced from overseas (Tokai Unit 1: Gas-cooled reactor) had two types of ASTSs. One of them had the trigger level as low as 50 Gal for vertical motions. This was introduced for ‘prediction’, with the intent of sending the reactor into an early scram with the detection of P-waves. It is therefore an example of Case B.

2.4.4.2. Seismic category and qualification

The decision to shut down a reactor automatically when the observed earthquake level exceeds the predetermined threshold needs to be fulfilled without failure by means of an ASTS. This is the reason why an ASTS is installed as reactor protection system or reactor protection system equivalent, accompanied by relevant requirements for reliability. Consequently, since plants are strictly controlled with the ASTS, the quality of the SAS and the SDAS is left up to the discretion of licensees, giving them freedom in data management.

Japan’s Technical Specifications on Safety Protection Features state that the nuclear power plant needs to incorporate safety protection features that, in the event of an abnormal transient during operation or a disruption of the reactor operation due to an occurrence of earthquake, keep the reactor below the allowable fuel damage limit, in conjunction with the reactor shutdown systems and the engineered safety features.

2.4.4.3. Number of seismic switches installed and scram logic circuit

It is necessary to ensure that the ASTS is activated whenever necessary with minimal errors. This is the reason why the ASTS needs to have redundancy, which determines the number of seismic switches to be installed. The ASTS needs to have a high level of reliability when it is installed as part of the reactor protection system. For this reason, similarly to other reactor protection system devices, its control logic is either ‘two out of four’, ‘double one out of two’ or ‘two out of three’ depending on the plant safety design philosophy. Fig. 12 shows ‘double one out of two’ and ‘two out of three’ examples. This way, at least three or four seismic switches are installed at locations that are perceived to have identical conditions.

2.4.4.4. Installation locations of seismic switches

According to the redundancy requirement discussed in the previous section, seismic switches are installed in at least three or four locations on the same plane (floor), depending on each plant’s safety design logic. The Japanese seismic design guideline recommends the following locations for installing seismic switches [27].

The location for the seismic trigger of the earthquake-detecting equipment needs to be determined by considering the object for which the seismic motion is to be detected; and the selected location(s) need to be easy for maintenance/inspection and need to be able to ensure high reliability.

More specifically, in a building which contains equipment important to safety, the seismic switches are set on the lowest elevation of the building to detect the seismic motion input to the building. In some cases, seismic switches are also set on a typical floor among the upper floors.

In the United States of America, as shown in Fig. 13, three triaxial seismic switches are installed at an equal interval on the foundation mat of a reactor containment structure, which is the same approach as that of Japan (in case of scram logic: ‘two out of three’).

2.4.4.5. *Other topics*

Similarly to the case of SAS systems, it is desirable to take into account the following matters:

– Maintenance and testing.

  Similarly to SAS, it is recommendable to consider accessibility when selecting the installation locations for ASTS, and to facilitate on-site calibration of the seismic switches as much as possible.

– Operability

  In order to ensure their activation when needed, it is advisable to install seismic the switches themselves and their connection cables in locations free of interference from surrounding structures even in case of an earthquake. Also, to install protective covers and implement waterproofing measures.

– Main control room notification

  Appropriate notification of initiation of the ASTS signal needs to be announced in the main control room.

2.4.5. Lessons learned and other observations

In the United States of America there have been reported cases of automatic reactor shutdown caused by spurious signals from ASTS at testing/research reactors and at the Diablo Canyon nuclear power plant in its early stages of operation. In contrast, no inadvertent activation of ASTS has been reported in Japan.

The prevention of spurious trigger signals is a particularly important issue for the ASTSs with relatively low scram settings. Spurious signal prevention measures are the same as those for SAS, with the basic measures shown below (See Section 2.3.3.1 for details):

– Installing band-pass filters to prevent acceleration noise;

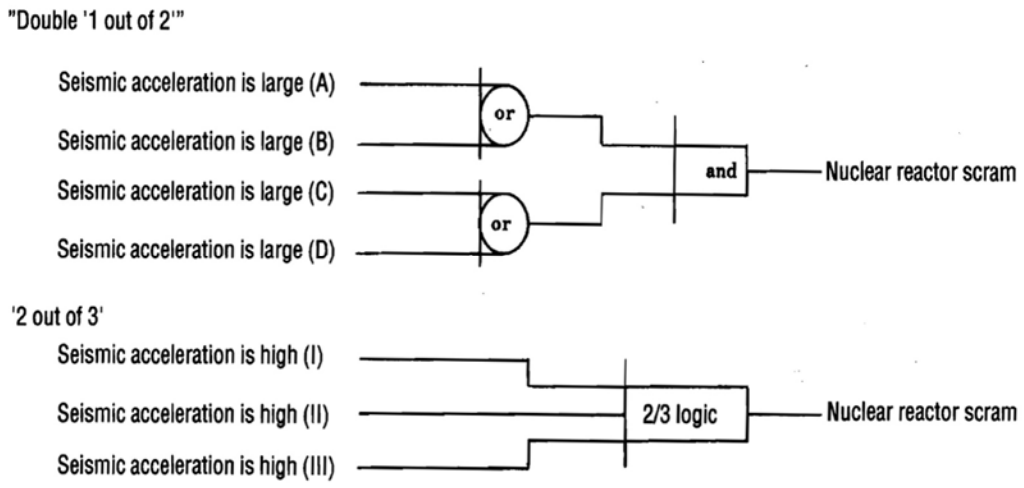


FIG. 12. Logic circuit examples of a seismic switch for an ASTS.

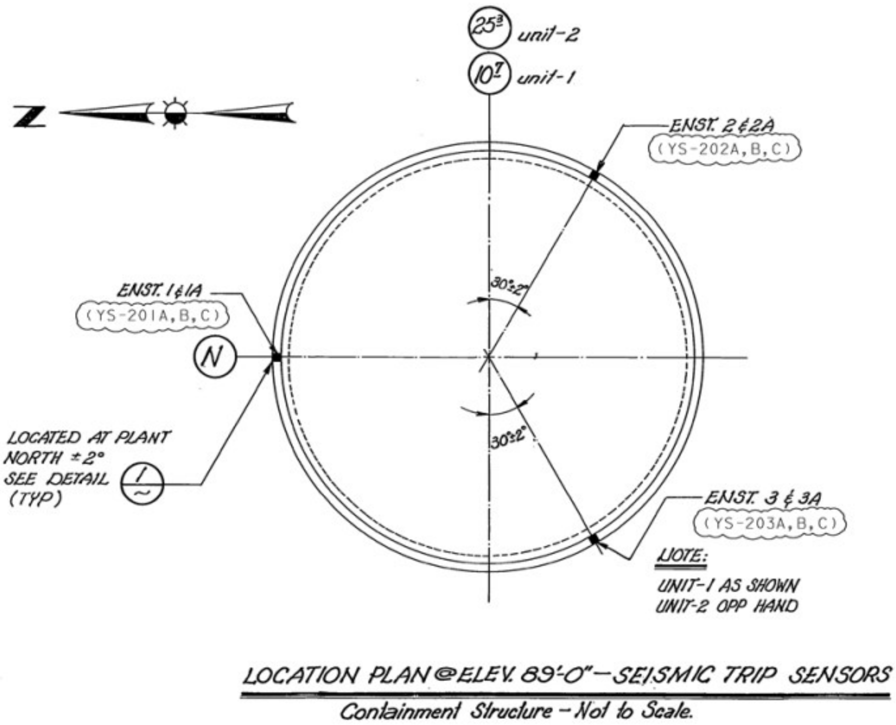


FIG. 13. Layout of seismic switches for ASTS at Diablo Canyon nuclear station Unit 1, United States of America. (Courtesy of Pacific Gas & Electric Company)

- Installing protective covers to prevent external impact noise;
- Multiplexing sensors for redundancy.

In Japan, the seismic switches for ASTS are installed in rugged protective boxes, and mounted with waterproofing and anti-flooding measures, as shown in Fig. 14.

2.4.6. Status of automatic seismic trip systems in nuclear power plants

2.4.6.1. General

The IAEA issued a questionnaire soliciting responses from eighteen Member States on the subject of seismic instrumentation systems (country requirements, types of instruments in place, location of instruments, maintenance and operability, etc.). As part of the questionnaire, recipients were asked about the requirements, implementation and operability experience with

ASTS. In general, the responses showed that plants in high seismicity areas are more likely to have ASTSs than those located in low to moderate seismicity areas.

Generally, the parameter to trigger scram signals is peak acceleration as measured in the free-field, structure foundation (or base mat), in structure locations or in some combination of these locations. Other parameters, such as response spectrum exceedance are also used.

As reported by respondents, normally a redundant logic is applied, that is, [2 x (1 out of 2)], (2 out of 3), etc. Experiences in different Member States are discussed in the following sections.

Appearance

Seismometer inside the cover

FIG. 14. External impact prevention measures for ASTS seismometer in Unit 2 reactor building at Shika nuclear power plant. (Courtesy of Hokuriku Electric Power Company)

2.4.6.2. Japan

Japan has regulatory requirements to install ASTSs in all nuclear power plant units. Table III-1 in Annex III present the data received from the Japanese utilities concerning ASTSs for nuclear power plants in Japan. Table III-1 A shows again the design basis earthquakes unit-by-unit for reference purposes and lists the ASTSs location and trigger levels for which answers were received. In general, scram can be initiated by exceedances of the trigger levels as established for the free field, for the foundation (basemat), for in-structure locations, or for a combination of those. Table III-1 B lists the results of comparison of the trigger levels with the design basis earthquake ground motion S1 and the design basis earthquake ground motion S2 earthquake levels. It is important to emphasize that the S1 earthquake in Japan is only partially comparable to the operating basis earthquake (SL-1) in other Member States.

The data from Japan provides a meaningful insight into the trigger levels of scram compared to the S1 and S2 design basis earthquakes. Trigger levels compared to S1 are about 50 % to

90 % of the S1 (free field, basemat, or in structure responses). Trigger levels compared to S2 are about 40 % to 60 % of the S2 (free field, basemat, or in structure responses). The exception being Hamaoka nuclear power plant, where the trigger levels are 27 % of S1 and they look like less related with design earthquake levels.

2.4.6.3. India

Two nuclear power plants in India have ASTSs: Kakrapar (KAPS 1 and 2) and Narora (NAPS 1 and 2). Table III-2 A lists appropriate information on design basis earthquakes and the trigger levels for ASTSs at these two nuclear power plants.

The trigger levels for KAPS 1 and 2 are the SL-1 level, that is, PGA = 0.1 g. The trigger levels for NAPS 1 and 2 are, also, PGA = 0.1 g, which is less than the SL-1 (PGA = 0.15 g).

2.4.6.4. United States of America

There are no specific regulations requiring ASTSs in nuclear power plants in the United States. Only the two nuclear power plants in high seismicity areas have an ASTS: Diablo Canyon power plant Units 1 and 2 and San Onofre nuclear generating station Units 2 and 3. Table III-2 B lists specific information concerning the design basis earthquakes and the ASTSs at Diablo Canyon and San Onofre plants. San Onofre was permanently shut down in 2013.

The Diablo Canyon power plant has three different levels associated with the seismic design: design basis earthquake, double design basis earthquake and the Hosgri fault earthquake. Horizontal peak accelerations for design basis earthquake, double design basis earthquake and Hosgri earthquakes are 0.2 g, 0.4 g, and 0.75 g, respectively. All three are applicable to the seismic design. Peak acceleration recorders are located on the containment building basemat. The trigger logic is (2 out of 3), that is, two out of three sensors need to exceed the trigger level for the scram to be initiated. The operating basis earthquake level remains as that initially defined for the design, that is, a PGA of 0.1 g. The trigger level for scram is a basemat acceleration of 0.35 g.

The seismic design ground motion for San Onofre Units 2 and 3 was PGA = 0.67 g. In both units, the trigger level for scram was the operating basis earthquake level PGA = 0.335 g.

2.4.6.5. Russian Federation

The questionnaire response from the Russian Federation participants indicates that an ASTS is a requirement for nuclear power plants. However, there was no detail provided for existing plants, which makes questionable whether these are requirements for new nuclear power plants, and not for existing plants.

2.4.6.6. China

At the Tianwan nuclear power plant (SL-2 peak acceleration=0.2 g), which has pressurised water reactors imported from the Russian Federation (VVER type reactors), seismic switches installed with the ‘two out of four’ safety logic concept are used for both activating alarms in the main control room and automatically shutting down the reactors.

2.4.7. Observations

Since the 2011 off-the-Pacific-coast-of-Tohoku earthquake, there have been initiatives to improve the seismic safety of nuclear power plants, and the perception about the necessity of an ASTS seems also to be changing.

In the Republic of Korea, their improvement goals include ‘ensuring that reactors are capable of safe shutdown even in an earthquake greater than the design basis level’. The introduction of the ASTS was recommended as a short-term response to be completed by 2012. More specifically, all the reactors will undergo improvement work so that they will automatically shut down upon detection of an earthquake at the level of 0.18 g or higher.

In India, while at present only a limited number of nuclear power plants (RAPS 1-2 and NAPS 1-2) feature a system that automatically shuts down the reactor in an earthquake event, as discussed in Section 2.4.6.3, it has been recommended to introduce the system to all the nuclear power plants.

While European Union nations have conducted stress tests on their nuclear plants since the 2011 off-the-Pacific-coast-of-Tohoku earthquake, some of the countries are considering the improvement of seismic instrumentation systems as a future task. Future tasks cited are outlined below, as an overview of post stress-test national action plans regarding seismic instrumentation.

As part of the European Nuclear Safety Regulators Group (ENSREG) Action Plan regarding the follow-up of the peer review of the stress tests performed on European nuclear power plants. The ENSREG decided to prepare by September 2012, a consistent compilation of stress test peer review recommendations and suggestions to assist the preparation or review of national action plans by national regulators.

ENSREG summarized the experience of stress tests and peer review of the stress tests performed on the European nuclear power plants [28]. ENSREG recommended installation of seismic monitoring systems with related procedures and training (paragraph 3.1.5 Seismic Monitoring in Ref. [28]).

In the framework of national stress tests and/or in response to ENSREG recommendations the seismic instrumentation of plants has been reviewed and actions have been defined. The actions are of different type as follows:

– Improvement of the seismic instrumentation, ensuring compliance to the national or international requirements (Belgium, Czech Republic, Finland, France);

– Analysis of arguments for automatic seismic scram, its advantages and disadvantages (Hungary, France, Switzerland);

– Considerations for amendment of existing or development of new regulatory requirements related to seismic instrumentation (Finland, Slovenia, Germany);

– Comparative study national versus international practice (France).

Many countries judged the plant seismic instrumentation and procedures for operator post-earthquake actions as appropriate (Bulgaria, Romania, Slovenia, Spain, Sweden and Hungary).

Following the ENSREG recommendation, some countries included into the plant's seismic instrumentation the regional (micro) seismic monitoring network and official institutions made it available to plant operators the national seismic network monitoring records (Slovakia, United Kingdom).

2.4.8. Future requirements and recommendations

2.4.8.1. Decision making process

A key element in the future decision-making process is the performance of systematic evaluations of the potential benefits and drawbacks of implementing an ASTS. Probabilistic safety assessment approaches for generic nuclear power plant types, but treating specific sites/plants, are recommended.

2.4.8.2. DIP application

Many of the ASTSs currently in use employ seismic switches, but there is a possibility that a system integrating a digital SDAS and an ASTS will be adopted in the future, with such a suggestion already being addressed. However, it is not necessarily appropriate to use the acceleration values themselves, as recorded with the SDAS, in the trigger settings for the ASTS, without any other evaluation.

In case that a system recording earthquake acceleration time-history is used for the ASTS, there is a need for measures to ensure the activation of reactor scram and preventing unnecessary reactor scrams. This includes introducing redundancy (number and layout of acceleration sensors), ensuring seismic capacity and enforcing the specifications required for quality assurance, as well as paying special attention to spurious scram. Records taken with accelerometers include acceleration noise or acceleration spikes from earthquake motions, however they possibly have no influence on the damage to SSCs. It is therefore necessary to introduce measures for preventing undesired activation through, for example, installation of frequency filters.

The objective of installing an ASTS (especially in Case A of Section 2.4.4) is to shut down a reactor upon detecting earthquake motion severity greater than the pre-defined level, which could lead to the functional damage of safety-related facilities. As discussed later in sections 3 and 4, DIP is more suitable than acceleration as indicator to be used for this purpose.

If an earthquake motion parameter is used as the ASTS trigger signal, the parameter needs to be observed or calculated in real time during an earthquake. Challenges surrounding such real time calculation of the DIPs proposed by this publication are discussed in Section 4.1.2 for the CAV, and in Section 4.1.3 for the Japanese instrumental intensity A_{JMA} (I_{JMA}). The challenges are also related with the data processing capacity of computers to be used for the SDAS. Through comprehensive considerations on specific system configuration, specifications, etc., The adoption of a DIP-based trigger signal is desirable.

2.4.8.3. Prediction capability

The most desirable concept of an ASTS is that it were possible to predict earthquake motion severity and, in the case significant damage was predicted, to complete reactor scram before the main component of the strong earthquake motions arrived [22].

In Japan, since October 2007, the Meteorological Agency has announced earthquake early warnings for disaster preparedness, recommending that as many possible preparedness measures be implemented before the arrival of the strong motions. This system was developed jointly by the Meteorological Agency and a railway company (Japan Railways East). Japan Railways East's ‘Urgent Earthquake Detection and Alarm System’ successfully initiated the emergency shutdown of Tohoku Shinkansen operations at the time of the 2011 off-the-Pacific-coast-of-Tohoku earthquake, thereby preventing an accident.

In the nuclear field, as described in Section 2.3, there was an example of actually introducing the ASTS from a SAS. However, in these cases, it is necessary to carefully evaluate the balance of the reliability of prediction and that of plant ‘scrammability’, at the anticipated level of earthquake motions, because an unnecessary reactor scram could cause a transient to undermine reactor safety and produce inconvenience to the society.

The earthquake early warning system, currently adopted by Japan Railways East, predicts the earthquake magnitude and hypocentral distance based on the starting gradient of absolute acceleration for P-waves. Discussion needs to be also held on whether experience-based data is applicable to a different region (country) with different seismic characteristics.

From the perspective of prediction reliability, anticipated topics for debate include as follows:

(a) Using international data, as the base data for prediction varies according to seismic characteristics and regions (countries);

(b) Considering the impact of background noise in coastal regions with numerous industrial facilities located, and

(c) Availability of only a small amount of data for countries with a low seismic activity, thereby influencing the accuracy of prediction.

While the time required to recover a train service from the state of shutdown is relatively small, it takes much longer to restore a shutdown reactor. The shutdown could have an extremely large social impact, including the possible discontinuation of electricity supplies. Since the possibility of inadvertent scram undermining reactor safety cannot be denied, actual performance data such as at the Onagawa nuclear power plant, where the system is already used for alarm activation, is expected to be accumulated further for more discussions prior to employing the system in the nuclear field.

3. PARAMETERS INDICATING EARTHQUAKE MOTION INTENSITY AND EXPERIENCED DAMAGE

3.1. EQUIPMENT DAMAGE CAUSED BY EARTHQUAKES

3.1.1. Causes of earthquake damage to structures, systems and components

Unacceptable behaviour associated with malfunction, damage or failure of SSCs caused by earthquakes is primarily from one of two sources. The first is inertia forces which, when applied to the mass of a structure (building), distribution system (piping, raceway, conduit, ductwork) or component (pump, valve, pressure vessel, fan, switchgear, motor control centre, transformer etc.), induce stresses/strains or member and section forces/moments which may result in unacceptable behaviour. These seismically induced stresses/strains or forces/moments, when combined with other applicable load resultant stresses/strains or forces/moments, may result in exceedance of specified allowable stresses/strains or forces/moments in the SSC. The design basis of SSC is required to meet limiting stresses/strains or forces/moments as prescribed by specified Codes and Standards. The design procedures as prescribed by Codes and Standards lead to significant margin with respect to mean estimates of actual failure of SSCs.

The second source of unacceptable behaviour, damage or failure is differential displacements. Configurations that are susceptible to differential displacement concerns are independent structures founded on a common basemat and independent structures on independent basemats.

Four potential failure modes of SSCs due to differential displacements are:

(1) Failure of interconnected secondary structures, such as walkways due to relative displacements;

(2) Failure of distribution systems (piping, ventilation ducts, conduit, cables, etc.) due to relative displacements between supporting points;

(3) Pounding of structures (impact causing additional stress on structure elements);

(4) Soil failure leading to large imposed relative displacements on structural elements (walls, basemat, etc.).

This secondary displacement type of failure is usually precluded at the seismic design levels by site foundation investigations and upgrading of the foundation media as necessary to resist the design basis earthquake load. The existing foundation condition or upgrading would render such a phenomenon as non-credible at design basis load levels. Typically, these failure modes are of concern only for beyond design basis earthquakes.

The structural failure mode that receives the most attention in seismic design or evaluation is usually the first type of inertia (acceleration) induced damage, failure or malfunction. In this case the inertia seismic forces are applied as an equivalent external force to SSC. The resultant limiting stresses/strains are determined by allowable stress design or by forces/moments determined by strength design procedures in structural and mechanical systems or components. Design standards define the acceptance levels of response.

It needs to be noted that most earthquake design standards acceptance criteria for conventional buildings and structures have historically been associated with life safety, not with continued operation or function. As a result, acceptable behaviour associated with life safety, which typically permits response of SSCs well into the inelastic range, may not be acceptable for continued operation or for keeping safety functions.

Electrical power, instrumentation and control devices are typically supported on racks, cabinets or boards and are affected by amplified seismic motion of not only the building structure, but also of their supporting racks, cabinets or board components. However, damage malfunction or failure of such electrical devices, in general, can only be determined by functional testing and not by structural analysis for resultant stresses, strains, forces/moments or displacements.

As mentioned earlier, seismic damage can roughly be divided into two categories: caused by inertia forces, and caused by the inability to absorb relative displacement due to installation ground subsidence and structural support movement. In the 1995 South Hyogo prefecture earthquake (the Great Hanshin-Awaji earthquake) in Japan, it was suggested from the structural specialists' survey that the damage rate to conventional industrial facilities caused by ground subsidence and the inability to absorb relative displacement accounts for nearly 50% [29]. Since the present publication is intended to evaluate and observe earthquake motion levels, differential displacement seismic damage is not included in the scope of this publication.

3.1.2. Physical characteristics of earthquake motions (quakes) and damage modes

Earthquake motions are essentially random vibrations and a wide variety of damage modes have been experienced due to the complexity of seismic loading on equipment. Matters to pay attention to when the relationship between the earthquake severity and the damage mode is considered are specifically shown below.

3.1.2.1. Reversing earthquake loading and damage modes

Two types of macro-perspective failure mechanisms are discussed generally for passive equipment in relation with the earthquake motion severity.

What is important to keep in mind when considering damage caused by vibrations due to earthquake motions is: (1) that the vibrational response of equipment will grow over time due to resonance phenomena with predominant frequency contents; (2) that a loading represented in the form of the product of the absolute response acceleration and the mass will act on the equipment; and (3) that the seismic loading is alternating and its duration is finite. The response of equipment during an earthquake is determined by balancing the kinetic energy due to the inertial force, the energy consumption by the damping mechanism and the equipment's elastic/plastic strain energy with the total energy from the seismic input motion. Thus, the mass of the target equipment (kinetic energy), the damping constant (energy consumption), and the seismic input (total energy) would affect equipment damage (exceedance of the capacity to absorb energy as strain energy). On the other hand, from the perspective of the alternating nature of seismic loading, it is necessary to consider the relationship between the so-called stress category (primary or secondary stresses) and the damage of structures, and the possibility of fatigue fracture or ductility exhaustion, especially in the case of metallic materials.

The duration of earthquake alternating strong motions is short, generally less than one minute and less than a few minutes at the very most, and the vibrational period over which reverse loading works is less than one second in many cases. In this regard, the structures with capacity to absorb a large amount of energy (i.e. with high ductility to tolerate considerable deformations) are hard to lead to seismic damage.

When structures with metallic materials respond to earthquake motion, the potential seismic damage mechanism is due to this complex input earthquake motion characteristics, that is, random reversing dynamic loading. In this regard, based on the damage experiences, it seems generally easier to understand the seismic damage mechanism by categorizing associated damage into 'first excursion' type damage and 'cumulative fatigue' type damage. Fig. 15 shows a conceptual image of this categorization.

Here, 'first excursion' type damage occurs when the seismic response of the target structure initially exceeds a certain level. 'Cumulative fatigue' type damage occurs when the cumulative response value reaches a certain level.

Examples of damage mode of 'first excursion' type include brittle fracture, ductile breaking, plastic collapse, buckling, and so forth. As a damage mode of 'cumulative fatigue' type, an example is what is known as (extremely) low cycle fatigue or ductility exhaustion.

Actual structural damage is complex in that it involves interactions among various factors and is determined by the characteristics of specific structures. When setting the level of damage (threshold value), it needs to be noted that both the momentary response value concerning first excursion damage and the cumulative response value concerning cumulative fatigue damage need to be considered.

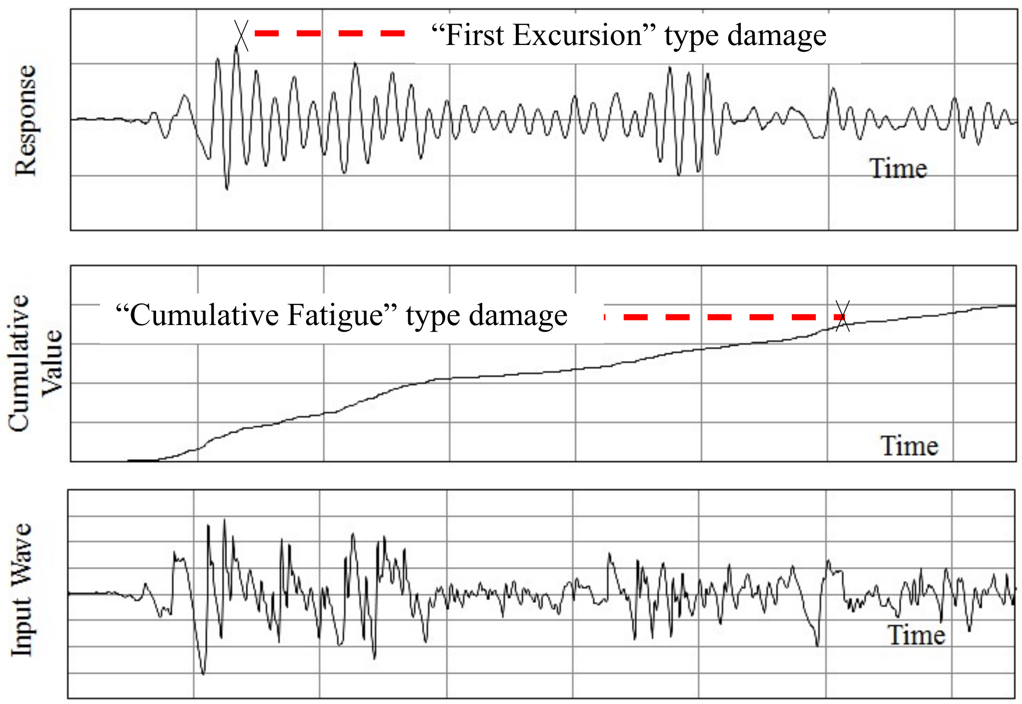


FIG. 15. Scheme of 'first excursion' and 'cumulative fatigue' damage

3.1.2.2. Hidden damage and fatigue fracture

Here, 'hidden damage' means the damage that is difficult to discover during operator walkdowns performed immediately after an earthquake and subsequent inspections. IAEA Safety Report Series No. 66 considers two categories of hidden damage [1]:

(1) "Damage to hidden parts: Damage that can be identified by disassembly but cannot be visually identified externally due to configurations or locations, for example, damage inside structures or components. Examples of degradation that may be hidden are:

  (i) Damage to mechanical couplings of buried piping and degradation of corrosion prevention coatings due to peel-off;

  (ii) Damage to inner components of emergency batteries, transformers, relays, etc., and damage causing malfunctions of float switches;

  (iii) Damage due to wear and deformation of inner parts of rotating equipment are examples of damage to hidden parts, which may be identified from past experience of earthquakes, that is, when performing maintenance, repairs and inspections, and reviewing shaking test data and design information.

(2) Invisible and/or undetectable damage: Damage that is very difficult to identify by visual inspections, such as loss of fracture toughness due to the combined loading conditions of an earthquake and other induced stress states. Examples of undetectable damage are the increase of fatigue usage factors for metal components, plastic deformation and cracks occurring inside concrete (e.g. around embedded anchorages)."

It was mentioned earlier (see Section 2.1.2) that IAEA Safety Report Series No. 66 determines post-earthquake actions, focusing on hidden damage to the nuclear power plant affected by an earthquake and taking into consideration the earthquake level in addition to the damage level identified during visual inspections.

Although hidden damage can be identified during equipment overhauling or functional testing, components installed in areas with difficult access require prior inspection planning, including the inspection of readily accessible typical equipment with the same structural characteristics and subjected to the same or a more severe loading condition instead. For this purpose, the relations between cumulative fatigue fracture and earthquake load will be discussed as follows.

Failure and fatigue damage during an earthquake

It needs to be noted that in civil engineering structures (buildings) usually there is no difference in the acceptance criteria used to address both the inertia (primary) type seismic loads and those induced by displacements (secondary) type loads. In mechanical engineering structures (pressure retaining components, piping, ductwork, etc.) the primary type stress/strains or forces and moments are typically limited to lower values than are the secondary seismic stress and strains since, in general, for secondary type loads it is only necessary to prevent ratcheting or fatigue type failure for such loading phenomena. In general, a single or a small number of load cycles (i.e. less than 10 equivalent full cycles) cannot cause failure from self-limiting secondary loads. This fact forms the basis of not limiting the secondary stress levels in the American

Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III Division1 for Service Level D loads, which are reduced to a single event or to a limited number of cycles per event (i.e. 10).

As seen from the above, it is extremely rare that fatigue damage occurs during an often-experienced earthquake if the frequency of seismic loading is limited. This is thought to be an intrinsic issue concerning the allowable stress system for seismic load as well, that is, whether failure due to fatigue damage would actually be caused by an earthquake. This issue warrants further discussions in connection with the characteristics of design basis or observed seismic motions (in particular, the duration and frequency contents of major motions). As described later (see Section 4.5.1), however, it has been confirmed, for example, in the large-scale vibration test of piping system, that nuclear power generating facilities designed with a particularly conservative allowable stress system have a large seismic safety margin.

On the other hand, it may be necessary to evaluate cumulative damage in post-earthquake evaluations so as to determine integrity throughout the plant life, when a large enough earthquake motion has been observed and supposed to influence the plant safety.

Seismic residual strain and fatigue life

As an example of evaluation for the continued operation or restarting of a nuclear plant affected by a strong ground motion, a peak acceleration more than three times the dynamic seismic design basis of the safety-related equipment was observed at the Kashiwazaki-Kariwa nuclear power plant in Japan during the 2007 Niigata-ken Chuetsu-Oki earthquake. This acceleration time-history is characterized in that it was due to a few pulse waves with relatively long duration (pulse width).

In this earthquake, no functional impact of seismic motions far exceeding the design basis was observed to any safety-related equipment installed in the plant. However, the existence of local residual strains could not be ruled out, although no damage to components and piping was found after careful visual inspections. Seismic response analysis was performed based on the observed earthquake motions and the stress levels were evaluated based on the results thereof. In parallel with these analyses, a series of tests and research work on the effect of seismic residual strain on fatigue life were conducted.

The focus of the fatigue test series was to investigate the influence of initial residual plastic strains on fatigue life. The test methods and the fatigue test results, parametrized as a function of the initial residual plastic strain level, are summarized in IAEA Safety Report No.66, Annex III on Influence of Plasticity on Fatigue Strength [1].

According to results of this study, there are no significant differences in fatigue life after repeated pre-loading with less than approximately 16% of residual (plastic) strain range (i.e. 8% alternating strain, $\Delta\varepsilon = 16\%$). Figures 16 and 17 summarize the test results as a function of material and temperature.

In the end, the suspicion of hidden damage was proved unfounded at the Kashiwazaki-Kariwa nuclear power plant as the results of realistic seismic response and stress analysis did not exceed the elasticity limit, due to a large margin embedded in the ‘seismic design method used during the construction of the nuclear power plant’. However, this series of tests and research work conducted in 2007–2012, in which the loading pattern and test conditions such as temperature were taken into account, will serve as a useful reference for evaluating the effect of residual strain on fatigue strength [30–31].

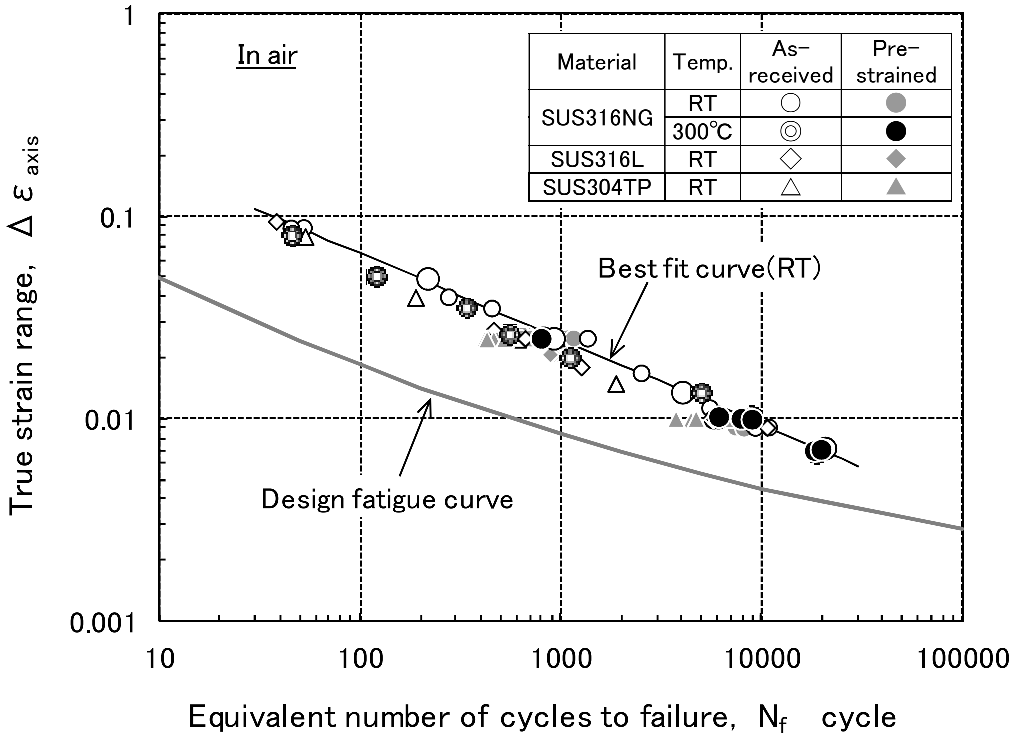


FIG. 16. Results of low cycle fatigue tests after repeated application of pre-strain (stainless steel).

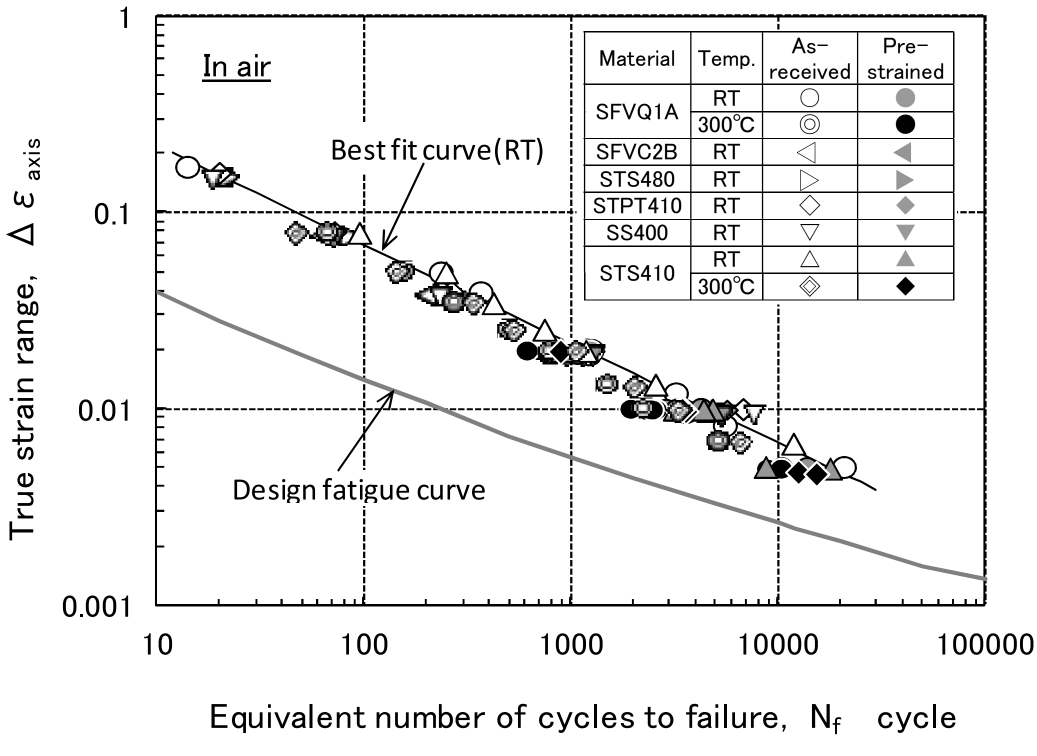


FIG. 17. Results of low cycle fatigue tests after repeated application of pre-strain (carbon steel).

3.2. SEISMIC INTENSITY SCALES

Seismic waves propagate in a heterogeneous medium, from the heterogeneous fault rupture to the ground surface. Therefore, the ground motion at the surface is complex and the effects on SSCs are not uniform. In order to indicate the severity of the ground motion in a concise way, 'seismic intensity scales' are defined using subjective human body sensation and observed damages in buildings and other objects.

3.2.1. Historical seismic intensity scales

Historically, seismic DIPs have been defined primarily by damage intensity scales, such as the Modified Mercalli intensity (MMI) scale [32], the Medvedev-Sponheuer-Karnik (MSK) scale [33], the Japan Meteorological Agency (JMA) seismic intensity scale [34] or the European macro-seismic scale (EMS) [35]. Using these scales, a numerical value is assigned to the observed levels of damage or unacceptable behaviour. Based on the description of a historical earthquake event, a numerical value can be assigned to the intensity of an earthquake at a particular location, based on the damage reported at that location.

Modified Mercalli intensity scale

The MMI scale was developed by Wood and Neumann [36] and modified by Richter [32] in the 1930s. It has 12 intensity degrees expressed as Roman numerals, I to XII. Little work has been done since then to improve this scale. Table IV-1 in Annex IV lists the unabridged MMI scale definitions. A shortened version of the MMI scale has been published and it is used often for intensity assessment, which can lead to errors in borderline cases. The MMI scale has been used as an observation-based DIP intensity scale, primarily in North America.

Medvedev-Sponheuer-Karnik scale

The MSK scale, known as well as the MSK-64 scale, is a macro-seismic intensity scale used to evaluate the severity of ground shaking on the basis of observed effects in an area of the felt earthquake occurrence.

The scale was first proposed by Sergei Medvedev (USSR), Wilhelm Sponheuer (Poland), and Vit Karnik (Czechoslovakia) in 1963 and 1964. It was based on the earthquake damage experience available in the early 1960s from the application of the MMI scale and on the 1953 version of the Medvedev scale, known also as the GEOFIAN scale.

With minor modifications in the mid-1970s and early 1980s, the MSK scale became widely used in Europe and the Russian Federation. In early 1990s, the European Seismological Commission used many of the principles formulated in the MSK scale in the development of the EMS, which is now a *de facto* standard for evaluation of seismic intensity in many European countries. The MSK intensity scale is also currently used in China, India, Israel, Russian Federation, and in Eastern Europe.

The MSK scale is similar to the MMI scale used in the United States of America. The MSK scale also has 12 intensity degrees expressed in Roman numerals**.** At the first working meeting of the Seismicity and Seismo-Tectonic Working Group [37] the judgmental terms used in the MSK intensity scale were quantified as shown in Table IV-2 in Annex IV.

Japan Meteorological Agency seismic intensity scale

The JMA observed intensity scale range is from 0 to 7. Thus, this scale as originally developed had 8 intensity degrees, defined as shown in Table IV-3 in Annex IV.

The original JMA seismic intensity scale of 8 grades was modified in 1996 (Table IV-4 in Annex IV). Two more scale grades where added and the scale values 5 and 6 were subdivided into two more scale levels termed Upper and Lower 5 and 6, respectively.

European macro-seismic scale

The EMS was derived collaboratively in 1998 among researches in Europe, based primarily on the MSK scale, with the same quantification factors as shown in Table IV-2 in Annex IV.

- Five grades of earthquake damage are defined in detail for different building types, ranging from slight damage to total destruction.
- Six vulnerability classes are defined, and building types are subdivided into vulnerability classes according to their level of earthquake resistant design.
- The adjectives ‘few’, ‘many’ and ‘most’ are defined quantitatively by the fraction of structures affected.
- With these features, the intensity levels are defined as the effects on few, many or most people, or as the grade of damage experienced by few, many or most buildings in a vulnerability class. As compared with MMI scale, the EMS allows a more objective evaluation of earthquake effects and damage and reduces the bias that an analyst might introduce.

3.2.2. Seismic intensity scales correspondence

Table 9 shows a general correspondence between the MSK, MMI, JMA (original scale) and EMS scales [38].

It needs to be understood that observed earthquake intensity at a particular location is a function of two phenomena; (1) the physical changes to the ground surface associated with ground motion and in particular ground displacement and, perhaps more importantly, (2) damage to mean mode structures.

In the upper range of earthquake magnitudes, above Magnitude 6, within the epicentral region there is usually a wide range of intensities. To a considerable degree, this wide difference is due to the differences in the quality of manmade construction in that location. In locations with well developed and enforced building codes, which require engineered construction practices for industrial installations, the observed intensity due to damage is much lower than a location where such code or standards and their enforcement for construction have not been followed.

TABLE 9. GENERAL CORRESPONDENCE BETWEEN THE MSK, JMA, MMI AND EMS INTENSITY SCALES (MODIFIED FROM [38])

Description	MSK	JMA (Original scale)	MMI	EMS
Not felt	I	0	-	I
Felt by very few	II	1	I	II
Felt indoors by few	III	2	II	III
Moderate vibration felt	III	2, 3	III	III
Hanging objects swing	IV	3	III	IV
Felt indoors by many	IV	3	IV	IV
Glassware and China clatter	V	3, 4	IV	V
Entire building trembles	V	4	V	V
Small objects shift	VI	4	V	VI
Plaster falls	VI	4	VI	VI
Furniture shifts	VII	4	VI	VII
High damage to weak structures	VII	4, 5	VII	VII
Moderate damage to ordinary structures	VIII	5	VII	VIII
High damage to ordinary structures	VIII	5, 6	VIII	VIII
Moderate damage to well-built structures	IX	6	VIII	IX
General panic	IX	6	IX	IX
Damage to most masonry and frame structures	X	6	IX	X
High damage to well-built structures	X	6, 7	X	X
Most masonry structures destroyed	XI	7	XI	XI
Most buildings destroyed	XII	7	XII	XII

3.2.3. Damage to conventional industrial electric and mechanical equipment and seismic intensity scale

As seen in their definitions, these seismic intensity scales are based on observed damage to conventional building structures. The relationship between seismic damage of conventional industrial electric and mechanical equipment and the JMA seismic intensity scale was investigated based on the seismic damage observed in the South Hyogo prefecture earthquake (the Great Hanshin-Awaji earthquake) experienced in Japan in 1995.

The Japan Society of Mechanical Engineers reported their investigation results on the damage experienced in the Great Hanshin-Awaji earthquake [29]. As addressed in Section 3.2.1, the JMA seismic intensity scale was revised in 1996. Hence, the damage cases listed in the publication are related with the original seismic intensity scale (see Table IV-3 in Annex IV). Annex V shows the damage list of conventional industrial facilities observed and the estimated JMA seismic intensity scale at their locations. Picking up the meaningful data from the viewpoint of damage induced by the inertial earthquake force, and classifying the data into categories of damage modes, Table 10 shows the correlation of typical damage modes and damaged structures with the lowest JMA original seismic intensity scale on which the damages were observed.

Some of the damage modes in Table 10 are not necessarily relevant in the case of nuclear power plants. Those are: (a) Movement, (b) Falling, (d) Contact and (e) Tumble (without anchorage). The reason is that these damage modes are easily identified by walkdown checks during construction and therefore prevented following the fundamental seismic design practice for nuclear power plants in Japan.

Items (g) Pulling out/fracture of anchor bolt (with base isolation support system), (i) Deformation of steel shelve frame, (j) Power boiler support structure, (k) Collapse of harbour crane (unloaded), (m) Water tank panel damaged by sloshing, (u) Collapse of smock stack and (y) Failure of power boiler burner, do not need to be considered, because those structures are not utilized at nuclear power plants.

Item (p) Contact/Hitting of pipe, produces minor damage and it has no influence on the functional capability of the piping systems.

According to the results thereof:

(a) Seismic damage including unanchored equipment damage occurs at earthquake motions equivalent to JMA original seismic intensity scale 4 and above;

(b) Anchoring the conventional equipment will improve its seismic performance to JMA seismic intensity scale 5;

(c) The JMA original seismic intensity scale level 5 is a threshold for significant damage in conventional SSCs.

(d) For nuclear power plants to which structural engineers would give dedicated consideration to improve seismic performance, it is presumed that earthquake motions equivalent to JMA original seismic intensity scale level 6 would be the threshold for initiating significant damage.

TABLE 10. TYPICAL DAMAGE OF INDUSTRIAL EQUIPMENT AND JMA ORIGINAL SEISMIC INTENSITY SCALE

	FAILURE MODE / DAMAGED STRUCTURE	JMA SEISMIC INTENSITY SCALE			
		4	5	6	7
a	MOVEMENT	—	—	—	—→
b	FALLING	- - -	- - - / —	—	—→
c	FALLING (AIR DUCT)		- - -	- - - / —	—→
d	CONTACT (SHOCK)	- - -	- - - / —	—	—→
e	TUMBLE (WITHOUT ANCHORAGE)	- - -	- - - / —	—	—→
f	TUMBLE (WITH WEAK ANCHORAGE)		- - -	- - - / —	—→
g	PULLING OUT / FRACTURE OF ANCHOR BOLT (WITH BASE ISOLATED SUPPORT SYSTEM)	- - -	- - - / —	—	—→
h	PULLING OUT / FRACTURE OF ANCHOR BOLT			- - -	- - - / —→
i	DEFORMATION OF STEEL SHELVE FRAME	- - -	- - - / —	—	—→
j	POWER BOILER SUPPORT STRUCTURE (STEEL FRAME SEISMIC TIE)	- - -	- - - / —	—	—→
k	COLLAPSE OF HARBOR CRANE (UNLOADED)	- - -	- - - / —	—	—→
l	FAILURE OF FOUNDATION / ANCHORAGE		- - -	- - - / —	—→
m	WATER TANK PANEL DAMAGED BY SLOSHING		- - -	- - - / —	—→
n	CIRCULAR STORAGE TANK WALL BUCKLING (ELEPHANT FOOT)			—	—→
o	OVERFLOW (SLOSHING)			- - -	- - - / —→
p	CONTACT / HITTING OF PIPE (INSULATION & GRATING DAMAGED)	- - -	- - - / —	—	—→
q	FAILURE OF MECHANICAL PIPE JOINT			- - -	- - - / —→
r	PIPE SUPPORT STRUCTURE DAMAGED (PULLING OUT OF ANCHOR BOLT)			- - -	- - - / —→
s	PIPE SUPPORT STRUCTURE BUCKLING				—→
t	FAILURE OF TRANSMISSION LINE SUPPORT			- - -	- - - / —→
u	COLLAPSE OF SMOKESTACK			—	—→
v	DERAILMENT OF ELEVATOR COUNTERWEIGHT			- - -	- - - / —→
w	BUCKLING OF CRANE BASE BASEMENT			—	—→
x	FAILURE OF STEEL SUPPORT FRAME				—→
y	FAILURE OF POWER BOILER BURNER				—→

Note : The dotted line indicates the width of the investigator’s estimation of the seismic intensity levels, e.g. ‘5 or 6’.

3.2.4. Seismic intensity scale specialized for nuclear power plant structures, systems and components

The issue associated with the seismic intensity scales is that the extent of damage to a target structure is closely related to the seismic design standards used at its construction stage. For example, the seismic intensity scale observed may change in and after the year in which new standards are enacted.

In addition, these seismic intensity scales cover an extensive range from weak earthquake levels that can be felt by the human body, for which there is hardly any seismic damage, to those strong earthquakes for which devastating damage occurs. In addition, they tend to saturate at higher intensity levels. As a DIP for nuclear power plants which were designed against large earthquakes, it is desirable to establish an intensity scale different from those conventional seismic intensity scales.

3.2.4.1. Seismic capacity of nuclear power plant SSCs and intensity scale

All seismic intensity scales discussed in Section 3.2.3 are based in part on earthquake induced damage to structures, mechanical and electrical distribution systems and components that for the most part have not been designed to resist earthquakes and, much less, designed to respond essentially elastically to earthquakes with an anticipated mean 10^{-4} yr^{-1} annual frequency of occurrence.

National Building Codes typically define design basis earthquakes at the 2×10^{-3} yr^{-1} frequency level[1] (i.e. 10 per cent probability of exceedance in 50 years) and allow the elastically computed earthquake induced stresses, forces and moments to be reduced by dividing by factors which range from 1.5 to 4.0, depending on the assumed inelastic energy absorption capacity of the component being designed. Conventional industrial structures (buildings) constructed in the last 75 years in earthquake prone regions typically have undergone some level of earthquake resistant design. However, in such cases these designs have generally been limited to design for life safety of occupants (i.e. no gross structural collapse of the structure or building), that is, they are intended not to have major injuries to building occupants caused by the collapse of the building structure. Building structure seismic life safety design is also meant to allow building occupants to leave the building during or following the earthquake.

Electrical power instrumentation and control and mechanical pressure retaining process distribution systems and components, in conventional industrial or power plant facilities, have received almost no seismic operational or safety function design consideration. As a result, most of the existing seismic intensity scales are based on behaviour of process SSCs that were not designed to resist strong earthquake ground motions and were located in building structures, which were designed to prevent catastrophic structural collapse. In addition, they have been designed in accordance with National Building Codes, which permit response to design basis earthquake strong motions well into the inelastic region.

It needs to be noted that earthquake damage and failure of positively supported or anchored civil-mechanical industrial or power plant SSCs are primarily a function of the inertial

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[1] In some Member States, such as Canada and the United States of America, National Building Codes define the design basis earthquake at the 4×10^{-4} yr^{-1} frequency level (i.e. 2 per cent probability of exceedance during a facility design life of 50 years), but in such cases they multiply the resultant seismic forces by two-thirds which effectively reduces the earthquake induced stresses, forces and moments to the 2×10^{-3} yr^{-1} frequency level.

acceleration applied during earthquake strong motion in the 2 to 10 Hz frequency range [39]. For malfunction or damage to electrical components and devices having dominant frequencies above the 10 Hz range, malfunction or failure is due mainly to accelerations with frequencies above 10 Hz or due to an impact loading.

In this section damage indicating scales are discussed, which could be applied to nuclear power plant safety-related SSCs that have been designed to operate without loss of safety function and the ability to shut down the plant safely when subjected to a significant earthquake, with acceleration values at a fraction of or larger than the design bases SL-1 or SL-2 earthquake levels.

3.2.4.2. EPRI seismic damage scale

The Electric Power Research Institute (EPRI) in the United States has developed a seismic damage scale for use in nuclear power plants, with the purpose of determining the potential for damage to seismically designed safety-related or important-to-safety SSCs.

Report EPRI NP-6695 [3] explains the necessity of and the approach to the seismic damage scale and proposes the four damage levels described as follows [3].

> "The EPRI seismic damage scale for nuclear power plant facilities has been developed for this report (EPRI NP-6695) because existing damage scales were not considered suitable for the evaluation of equipment and structures constructed to the standards used in nuclear power plants. The MMI scale, developed in 1931, is based on damage assessment of conventional and residential buildings and other effects. Use of the MMI scale may result in an over-estimate of the damage potential of the earthquake in nuclear power plants, because nuclear plants are designed and constructed to much more stringent standards than conventional structures. The EPRI damage scale provides a measure against which knowledgeable personnel can establish a timely, objective, plant specific estimate of the earthquake potential effects on well designed structures and equipment such as those found in nuclear power plants.
>
> The EPRI seismic damage scale is such that Intensity 0 is intended to correspond to earthquake that are slightly below the operating basis earthquake exceedance criterion given in Ref. 1 and summarized in Appendix A of this report. Intensity 1 corresponds to earthquakes that are slightly above the operating basis earthquake exceedance criterion. Intensity 2 is defined as the point at which the first damage to seismically designed (generally safety-related) equipment occurs. Intensity 3 corresponds to significant damage to seismically designed equipment."

A general description of the levels of the Earthquake Damage Intensity Scale proposed by EPRI [3] is as follows. Standard ANSI/ANS-2.23-2002 [14] endorses the scale.

Level 0. No damage or distress to safety-related seismic designed equipment or structures. Some evidence or distress/upset in non-seismic damage indicators.

Level 1. No damage or distress to safety-related, seismic designed equipment or structures. Widespread distress in non-seismic commercial buildings, windows, unreinforced masonry.

Level 2. First evidence of damage/leakage/cracking in safety-related, seismic-designed equipment and structures. Considerable damage to non-seismic civil structures.

Level 3. Clear evidence of permanent deformation, cracking of safety-related equipment, piping supports and structures. Severe damage to civil structures.

Guidelines in Ref. [3] include the pre-earthquake identification of samples SSCs which have well defined fragility levels related to the SL-1 or SL-2 design basis earthquakes. The response of such SSCs to a significant earthquake establishes a baseline where their existing as-is condition is well established and some of these SSCs could be instrumented such that earthquake induced strains or deformations could be measured. These pre-selected SSCs could then be inspected following a significant felt earthquake to determine the potential for damage extended to the full range of safety-related SSCs installed in the nuclear power plant or other nuclear installations.

Table 11 is a modification of the original EPRI intensity scale that ranged from scale values 0 to 3 by adding an intensity level 4 [40]. Prior to the last 10 years, it was assumed that a SL-1 or SL-2 level earthquake exceedance at a nuclear facility would be a very rare event and would be expected to result in failure of safety-related or important to safety SSC and if exceeding could result in loss of required safety function. Recent experience at several Japanese nuclear power plants and the Perry nuclear power plant in the United Sates of America, where the SL-2 level earthquake was exceeded, indicated that the SL-2 accelerations can be exceeded by a significant amount with little or no damage and no loss of safety function. If the analytical design basis acceptance criteria for the SL-1 and SL-2 level earthquake are limited to essentially elastic behaviour, loss of safety function would not be expected until the SL-2 earthquake peak ground spectral accelerations are exceeded by a factor of two or more.

3.2.4.3. Damage level specified in IAEA Safety Report Series No.66

IAEA Safety Report Series No. 66 specifies the post-earthquake actions judging from a combination of two parameters: earthquake level and damage level. The former is based on the comparison between the observed earthquake and the design level earthquakes, SL-1 or SL-2, using the exceedance criteria. The latter is decided by the post-earthquake inspections and functional tests. In Section 3.4.2, IAEA Safety Report Series No. 66 gives the following definition of damage level [1]:

> "Damage levels are numerically designated from 1 to 4 depending on the damage to SSCs important to safety and those not important to safety. Damage levels are defined on the basis of significant damage."

Table 12 shows the description of the damage levels referred to in the quotation above.

Here, the definition of 'significant damage' needs to be explained in detail, especially from the viewpoint of judging about the consequences of the experienced earthquake. Significant damage is defined as a damage (physical or functional), which has the potential to adversely affect the operability, functionality or reliability of SSCs. Examples are shown in the Table 13. In addition, Table 14 provides guidance to assess the significance of cracks in reinforced concrete structures.

TABLE 11. MODIFIED EPRI SEISMIC DAMAGE SCALE FOR NUCLEAR POWER PLANTS [40]

Damage Level	Maximum Damage Description
DL0	Damage that is limited to a wide range of architectural type items that are relatively fragile, common to most industrial and non-industrial facilities (e.g., homes, offices, etc.), and have been shown to be good indicators of a low level of shaking. These items have no significant impact on the safety or operability of the plant. The items of equipment in this category are referred to as non-safety related, non-seismically designed 'damage indicators'. Observed damage that is limited to these items is classified as damage level 0. Examples include damage such as displacement of panels in wire hung suspended ceilings, some tipping, displacement and spilling of contents of book cases and storage containers, and some cracking of plaster and unreinforced masonry walls in buildings built to commercial and/or residential standards such as office buildings, administration buildings and shops.
DL1	No damage to safety related SSCs. Additional damage to non-safety related, non-seismically designed SSCs typically found in commercial, industrial and power plant facilities, but which have been shown to have relatively low seismic ruggedness. Examples of damage to this category of SSCs include widespread falling of panels in suspended ceilings, widespread cracking of windows, plaster, masonry and concrete structures not designed or built to commercial seismic standards. Some evidence of piping insulation deformation/denting and interaction of non-seismically designed piping with nearby structural elements. Slight damage to low pressure storage tanks that does not limit their functionality (e.g. no significant leakage, limited shifting on foundations, limited anchor bolt inelastic deformation, limited buckling). Displacement of unanchored equipment on its foundation. Tripping of vibration-sensing instrumentation. Damage to fragile switchyard components such as high voltage ceramics.
DL2	No damage to safety related SSCs. Additional damage to non-safety related, non-nuclear seismically designed SSCs typically found in commercial, industrial and power plant facilities, and which have shown relatively high seismic ruggedness in past earthquakes. These would include SSCs designed and built to commercial seismic standards such as the uniform building code and the international building code. Examples of damage to this category of SSCs include widespread cracking in concrete and masonry structures, leakage of flanged and threaded joints and evidence of insulation deformation/denting in non-seismically designed piping. Permanent deformation of anchorages and walls of non-seismically designed low pressure storage tanks, including leakage that challenges the continued functionality of the tanks. Damage to less fragile switchyard components such as low voltage ceramics, air-blast circuit breakers and rail-mounted transformers.
DL3	Isolated evidence of damage to safety related SSCs in addition to the kinds of damage referred to in the lesser damage levels above. SSCs in this category include distribution systems (raceways and ductworks) and both seismically designed and non-seismically designed tanks and anchorages of some electrical equipment. Evidence of isolated and limited cracking in safety-related concrete walls and equipment foundations. More severe and widespread damage to non-seismically designed concrete, masonry construction. General over-turning of unanchored equipment and storage containers.
DL4	Clear evidence of permanent deformation, cracking and malfunction of safety related equipment, piping, supports and structures in high demand locations. Severe damage and isolated collapse of non-seismically designed civil structures. Widespread damage to switchyard components and supports. General failures of low-pressure storage tanks leading to loss of contents. Evidence of seismic interactions between distribution systems and nearby equipment and structures. Indications of reactor coolant leakage from detection alarm systems.

TABLE 12. DAMAGE LEVEL IN IAEA SAFETY REPORT No. 66 [1]

Damage Level	Definition
Level 1	No significant damage or malfunction to SSCs important to safety and those not important to safety.
Level 2	No significant damage or malfunction to SSCs important to safety. Significant damage or malfunction to SSCs not important to safety (NRPG: not required for power generation).
Level 3	No significant damage or malfunction to SSCs important to safety. Significant damage to or malfunction of SSCs not important to safety (required for power generation - RPG).
Level 4	Significant damage to or malfunction of SSCs important to safety (it is highly likely that SSCs not important to safety will experience significant damage at this damage level).

TABLE 13. EXAMPLES OF SIGNIFICANT DAMAGE [1]

Concrete structures	New or earthquake induced cracks in concrete greater than a prescribed threshold (e.g. see Table 14), spalling of concrete and visible distortion of frames
Steel structures	New or earthquake induced visible plastic deformation or cracking of joints and visible distortion of bolts, bolt holes or steel members
Piping	Through-wall cracks in pipe resulting in leakage; evidence of new or increased leakage at joints or connections following an earthquake; complete or partial severance of pipes; significant flow reduction due to cross section impairment[a]; or flow control valve malfunction; plastic deformation identifiable through visual inspection[b]
Distribution system supports	When supports are no longer capable of performing their support design safety function[c]
Mechanical or Electrical equipment	Visible distortion of anchorage system, sliding of the base of the component, rupture (leakage) of attached distribution system; general crimping or buckling of the equipment body, shell or housing[d]
Rotating equipment	Excessive noise, vibration or temperatures in running equipment

(a) Damage to insulation and denting or scratching of pipes are not considered to be significant.
(b) A laboratory test demonstrated that plastic deformation of about 8% does not significantly affect the material fatigue strength.
(c) Bent or deformed supports, so long as they are capable of performing their design safety function, are not considered to be significant.
(d) Scratches and localized denting of the equipment body or housing are not considered to be significant.

TABLE 14. GUIDANCE FOR SIGNIFICANT CRACKS OF REINFORCED CONCRETE STRUCTURES [1]

Crack Size	Guidance
≤ 0.5 mm	Insignificant crack unless near expansion anchor in which case anchorage tensile capacity can be reduced.
0.5 - 1.5 mm	Needs to be mapped. Not likely to be significant to structural capacity.
1.5 - 3.0 mm	Indicates yielding of rebar has occurred. Need to assess cause. Unlikely to have significantly degraded structural capacity.
≥ 3.0 mm	Either rebar is absent or has significantly yielded. Need to assess cause. May degrade structural capacity.

3.2.4.4. Damage level based on seismic design categories

Since seismic damage depends on the applied seismic design methods, it is possible to link the seismic design categories, which specify the details of seismic design, to damage levels, especially focusing on the nuclear installations, for which detailed seismic design standards have been established.

Japan's seismic design classification has three levels, namely, S, B, and C. The guideline proposed by the Japan Nuclear Technology Institute, as shown in Table 15, defines damage levels linked with these seismic design classes [41].

TABLE 15. DAMAGE LEVEL IN JAPAN NUCLEAR TECHNOLOGY INSTITUTE (JANTI) GUIDELINE (COURTESY OF JANTI)

Damage level	Definition
Level I	There is no significant damage to the systems, structures and components (hereinafter, equipment) of the power station.
Level II	There is no significant damage to seismic design class S equipment and to equipment required for generating power at seismic design class B and seismic design class C plants, but there is significant damage to other equipment.
Level III	There is no significant damage to seismic design class S equipment, but there is significant damage to equipment required for generating power at seismic design class B and seismic design class C plants.
Level IV	There is significant damage to seismic design class S equipment.

What is important here is that, like in IAEA Safety Report Series No. 66, seismic damage is divided into significant damage and minor damage based on past experience, and damage levels are determined based on significant damage. Although, they are determined by structural experts basically, examples of minor damage based on past seismic experience of nuclear power plants in Japan are given as follows:

– Window cracks and breakage (that do not have an effect on safety-related equipment and radiation control);
– Damage to pipe insulation;
– Damage and deformation due to contact between pipes and gratings;
– Damage, moving, and falling of covers such as cable tray covers (if there are no major effects on surrounding equipment);
– Hairline cracks on concrete;
– Bending or deformation of supports that do not affect support functions;
– Deformation of monorail stoppers;
– Falling of fluorescent lights and lighting fixtures (if there are no major effects on surrounding equipment);
– Minute leaks from liquid-level gauges and flow glass junctions (for which repair is easy);
– Leakage from the transformer pressure discharge tube (pressure discharge device);
– Increased leakage from the rotor shaft seal;
– Books and office supplies falling from desks (if there are no major effects on surrounding equipment);
– Deformation of shelves in warehouses, etc. and falling of stored items;
– Drum cylinders falling over.

3.3. SEISMIC INTENSITY SCALE AND EARTHQUAKE MOTION PHYSICAL PROPERTIES

3.3.1. Historical development of damage indicating parameters

As mentioned earlier, conventional seismic intensity scales are influenced by the seismic design methods applied to the structures and they are fundamentally qualitative, that is, they are not based on quantitative determination of physical parameters of the ground motion. Nevertheless, it has been attempted to link the physical parameters of damaging earthquake motions themselves to seismic intensity scales.

On the other hand, the progress of seismic instrumentation technologies makes it possible to record the acceleration, velocity and displacement corresponding to earthquake motions. Particularly, the maximum or peak value of observed acceleration became a parameter indicative of the earthquake severity, because, the acceleration value is widely used in seismic design and it lends itself to an easy calculation of seismic loads (e.g. for simple SSCs, the seismic load can be obtained by multiplying the acceleration by the mass of the target component). Thus, it has been attempted to correlate seismic intensity scales and PGAs.

Starting in the 1930s, strong motion acceleration recorders began to be installed in earthquake prone regions and started recording ground motion acceleration time-histories from earthquakes. Therefore, correlations could be established between recorded PGAs and the observed response of SSC to the earthquake. An example of early correlation between the MMI intensity scale and PGAs is shown in the Ref. [42].

The recorded PGA was selected as the DIP of interest in the mid 1960s, at the start of the nuclear power plant design era, since it was immediately available following a felt earthquake. Based on this parameter, an assessment could be made about the potential to damage nuclear safety-related SSCs, by comparison with seismic design basis SL-1 or SL-2. It also provided a basis for plant shutdown and post-earthquake damage evaluation, potential or requirements for repairs and upgrades and plant restart.

In the past 30 years, several high PGAs have been recorded at nuclear power plants. Actual plant damage (or lack of damage) and various other analytical predictors of damage have demonstrated that the recorded PGA was a very poor measure or indication of earthquake damage potential for seismically designed SSCs.

Over the past 20 years, algorithms have been developed to utilize the digital time-history records obtained by strong motion recorders, which are capable of analysing various parameters of the recorded time-history and the response spectra in real time. They can provide this information to nuclear power plant operators during and immediately following the earthquake to aid in making decisions about plant shutdown, to identify which components have greater damage potential, and to formulate evaluation requirements for nuclear power plant restart.

With the development of the Standardized CAV [43] over the past 25 years in the United States of America and the development of the computed JMA instrumental intensity scale over the past 20 years in Japan, it is now possible to compute damage intensity parameters better than the PGA, based on the full records of earthquake motions.

3.3.2. Instrumental seismic intensity

Seismic intensity at a given location that results from an earthquake is normally determined after post-earthquake investigations. On the other hand, with the growth of disaster prevention consciousness in recent years, there is an increasing need to facilitate emergency preparedness measures by identifying immediately after an earthquake the region in which damage is anticipated to occur with a high probability. This requires automatization, by avoiding human interpretation. Consequently, the relationship between the observed earthquake motion and the anticipated seismic damage is being studied.

Here, it is important underlining that the seismic intensity scale based on observation is relative, as addressed earlier, to the seismic capacity of the target structures, such as buildings. There is an issue, for example, that when the seismic capacity of the target structures improves as a result of changes in seismic design standards or rules, damage will decrease for the same level of earthquake motion level, resulting in a lower seismic intensity scale. Moreover, since it takes time to survey actual seismic damage, regular intensity scales are not adequate to determine the actions to be taken immediately after an earthquake. It is therefore desirable to establish a relation between seismic intensity scales and a physical parameter of earthquake motion.

Acceleration, velocity, etc, have been pointed out as candidates for physical quantities representing the characteristics of earthquake motions in lieu of seismic intensity scales. However, although these types of physical quantities are easy to determine by means of seismic instrumentation systems, it is difficult for the public to understand what they actually mean. In this respect, the seismic intensity scales that have been used and generally accepted for a long time are more adequate from the viewpoint of preparation against earthquake disaster. Based on this background, instrumental seismic intensities, which estimate the seismic intensity level using seismic instrumentation, have been developed.

3.3.2.1. MMI instrumental intensity scale used in ShakeMaps of USGS

From the perspective of emergency preparedness, the United States Geological Survey (USGS) has published 'ShakeMaps' on which the earthquake motion intensity levels estimated immediately after an earthquake are shown in the MMI scale.

The USGS states in the ShakeMap manual [44] the following:

> "That is not to say that instrumentally derived seismic intensity alone is sufficient for loss estimation. In fact, peak velocity and spectral response provide a more physical basis for such analyses. However, for the majority of users, we expect that the intensity map will be more readily interpreted than other maps of ground motion parameters and will be, therefore, more useful."

The estimated intensity map is derived from ground motions recorded by accelerographs and it represents intensities that are likely to be associated with the recorded ground motions.

The MMI instrumental intensity scale is correlated by means of several equations with the observed peak acceleration or velocity of the earthquake motion, as shown below. Table 16 provides a graphic view of these correlations.

TABLE 16. SHAKEMAP INSTRUMENTAL INTENSITY SCALE TEXT DESCRIPTIONS (COURTESY OF U.S. GEOLOGICAL SURVEY)

PERCEIVED SHAKING	Not felt	Weak	Light	Moderate	Strong	Very strong	Severe	Violent	Extreme
POTENTIAL DAMAGE	none	none	none	Very light	Light	Moderate	Mod./Heavy	Heavy	Very Heavy
PEAK ACC.(%g)	<0.05	0.3	2.8	6.2	12	22	40	75	>139
PEAK VEL.(cm/s)	<0.02	0.1	1.4	4.7	9.6	20	41	86	>178
INSTRUMENTAL INTENSITY	I	II–III	IV	V	VI	VII	VIII	IX	X+

ShakeMap Manual [44]

Converting from PGA and PGV to instrumental intensity:

> "Wald and others (1999b) recently developed regression relationships between Modified Mercalli intensity I_{mm} (Wood and Neumann, 1931, later revised by Richter, 1958) and PGA or PGV specifically for ShakeMap use by comparing the peak ground motions to observed intensities for eight significant California earthquakes. For the limited range of Modified Mercalli intensities $V \leq I_{mm} \leq VIII$, Wald and others (1999a) found that for PGA,
>
> $$I_{mm} = 3.66\log(PGA) - 1.66 \qquad (\text{standard deviation} = \text{sigma} = 1.08) \qquad (2)$$
>
> and for peak velocity (PGV) within the range $V \leq I_{mm} \leq VIII$,
>
> $$I_{mm} = 3.47\log(PGV) + 2.35 \qquad (\text{standard deviation} = \text{sigma} = 0.98) \qquad (3)$$
>
> Because we are also interested in estimating intensity at lower values, and our current collection of data from historical earthquakes does not provide constraints for lower intensity, we have imposed the following relationship between PGA and I_{mm}:"
>
> $$I_{mm} = 2.20\log(PGA) + 1.00 \qquad (4)$$

Basically, in the lower range of the MMI scale, where the intensity is determined by perception of human senses, the maximum acceleration plays a key role; while in the upper range of intensities, where intensity is determined by the extent of seismic damage to structures, the peak velocity becomes more important. The manual of ShakeMap [44] contains the following description, which serves as a useful reference in considering one or the other parameter for indicating earthquake levels from the perspective of damage:

> "In practice, we compute the I_{mm} from the I_{mm} versus PGA relationship (Eq. 2 and Eq.4), and if the intensity value determined from peak acceleration is ≥ VII, we then use the value of I_{mm} derived from the I_{mm} verses PGV relationship (Eq. 3). If the I_{mm} determined from PGA is between V and VII, we weight both the PGA-derived and PGV-derived values, weighted by a factor linearly ramping from 1.0 for PGA at I_{mm} V to 0.0 at I_{mm} VII and vice versa. The switch to PGV for higher intensity insures that spurious high frequency acceleration spikes will not result in high intensities because the corresponding velocity for such a spike will be low. With our procedure, whereas the large acceleration peak would provide an abnormally high intensity, the much smaller velocity amplitude would provide a more appropriate, lower intensity.

Using peak acceleration to estimate low intensities is intuitively consistent with the notion that lower (<VI) intensities are assigned based on felt accounts, and people are more sensitive to ground acceleration than velocity. Higher intensities are defined by the level of damage; the onset of damage at the intensity VI to VII range is usually characterized by brittle-type failures (masonry walls, chimneys, unreinforced masonry, etc.), which are sensitive to higher frequency accelerations. With more substantial damage (VII and greater), failure begins in more flexible structures, for which peak velocity is more indicative of failure."

3.3.2.2. JMA instrumental seismic intensity

The seismic intensity scale used in Japan is based on the measurement with a specific seismometer, that is, with a seismic intensity measuring device. A network of such devices has been set up all over the country. The JMA seismic intensity scale is indicative of the general phenomena due to an earthquake and of the damage situation at each grade of seismic intensity, as shown in Table IV-4 in Annex IV.

Before introducing this instrumental scale, seismic intensity was determined by observation of the extent of damage to buildings, the perception of meteorological observatory personnel and other elements, using a table describing each of the intensity levels. However, from the experience of strong earthquakes, an issue arose in that delays in announcing the seismic intensity were likely to occur (a mobile observation team from the JMA Seismological Division would need to conduct a field investigation). In this regard, from April 1996, seismic intensity has been determined and officially announced by using seismic intensity measuring devices.

JMA instrumental seismic intensity is calculated from the observed earthquake ground motion wave form as follows [34].

Calculation procedure of the JMA instrumental seismic intensity

– Step 1

Calculate Fourier spectra for the three spatial components, two horizontal and one vertical, of the recorded earthquake acceleration time-histories

– Step 2

Correct the influence of frequency (period) contents of earthquake motion using three filters shown in Fig. 18.

– Step 3

Compute the time-histories with inverse Fourier transform

– Step 4

Calculate vector synthesis of time-histories of the three spatial components (absolute value of acceleration)

– Step 5

Calculate 'A_{JMA}' value (acceleration) so that the total time during which the absolute acceleration is higher than 'A_{JMA}' becomes equal to 0.3 seconds (Fig. 19).

Blue line:	low frequency (<0.5Hz) cut
Red line:	high frequency (>10Hz) cut
Green line:	frequency effect (considering acceleration and velocity)
Black line:	total (normalized to 1.0 at 1.0Hz)

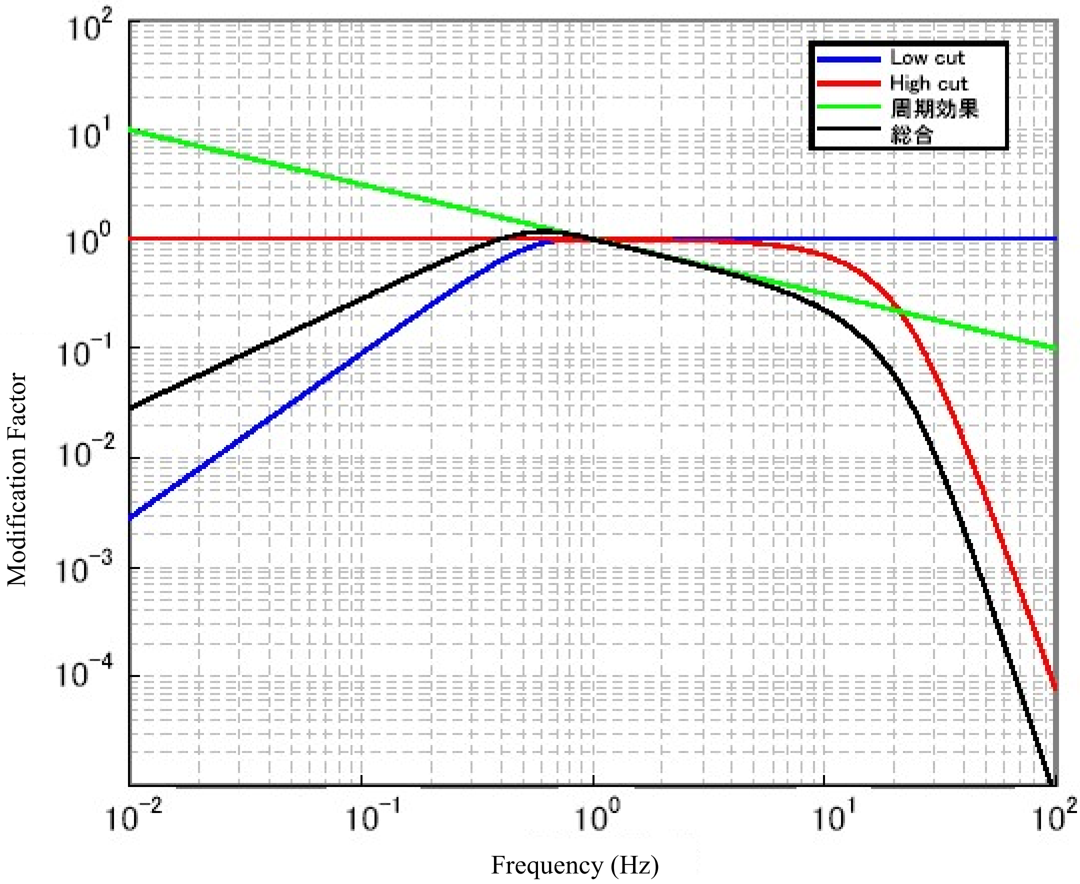


FIG. 18. Filters for calculation of the JMA instrumental seismic intensity [34]. (Courtesy of Japan Meteorological Agency)

– Step 6

Calculate the instrumental seismic intensity 'I_{JMA}', as;

$$I_{JMA} = 2\, log(A_{JMA}) + 0.94 \quad \text{(Round the thousandth digit and round off the hundredth digit)} \qquad (5)$$

Relationship between 'JMA seismic intensity scale' and 'instrumental seismic intensity' I_{JMA} is shown in Table 17.

After long studies on instrumental seismic intensity, the intensity scale calculated from the instrumental records as mentioned above has been used as the official intensity scale since 1996, based on the experience in the 1995 Southern Hyogo prefecture earthquake. Subsequently, no changes have been made although the computation formulas have been reviewed based on the damage found after the occurrence of strong earthquakes.

What needs to be noted here is that JMA specifies seismometers for its monitoring of instrumental seismic intensity (referred to as instrumental seismic intensity meters). Since acceleration data significantly depend on the ground and geological features where the seismometers are installed, the locations of installation are specified as given below.

– Intensity meters need to be installed on flat land that is entirely made of the same type of geology, avoiding for example cliffs.

– Intensity meters need to be installed away from structures, so that they are not affected by the vibrations of such structures.

– Intensity meters need to be installed in a manner that the intensity meters or their foundations are rigidly connected with the supporting ground, so that the intensity meters will shake the same as the ground surface.

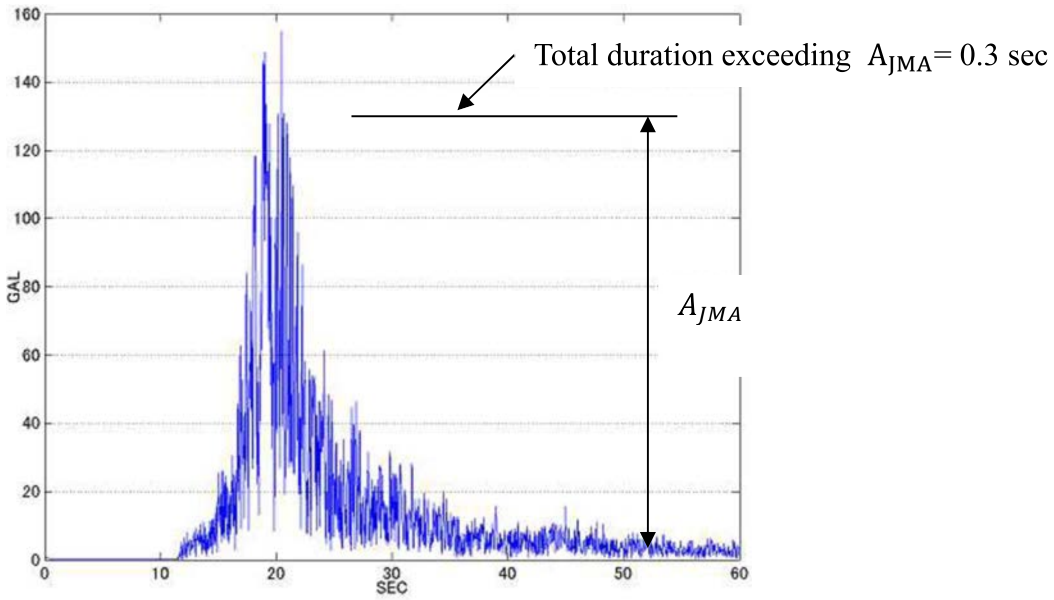


FIG. 19. Vector synthesis of three spatial components time-history. (Courtesy of Japan Meteorological Agency)

TABLE 17. RELATIONSHIP BETWEEN JMA SEISMIC INTENSITY SCALE AND INSTRUMENTAL SEISMIC INTENSITY I_{JMA}

JMA seismic intensity Scale	Instrumental seismic intensity I_{JMA}
0	less than 0.5
1	larger or equal to 0.5 and less than 1.5
2	larger or equal to 1.5 and less than 2.5
3	larger or equal to 2.5 and less than 3.5
4	larger or equal to 3.5 and less than 4.5
5 Lower	larger or equal to 4.5 and less than 5.0
5 Higher	larger or equal to 5.0 and less than 5.5
6 Lower	larger or equal to 5.5 and less than 6.0
6 Higher	larger or equal to 6.0 and less than 6.5
7	larger or equal to 6.5

From the viewpoint of the physical meaning of A_{JMA}, the characteristics of the frequency filter applied in Step 2 needs to be pointed out first. The high-cut filter corresponds to the common thinking that the high frequency component of earthquake motion has less influence on structural damage. The frequency filter with a -1/2 gradient on double logarithmic chart, means that damage likely happens due not only to acceleration but also to velocity. This may correspond to MMI instrumental seismic intensity in the Table 16.

A second insight is that the time used in Step 5 results from consideration of the impulse caused by the earthquake inertial load that is effective to produce structural damage.

4. DAMAGE INDICATING PARAMETERS AND EARTHQUAKE MOTION LEVELS BASED ON CALCULATIONS

As briefly addressed in Section 3, the subject of obtaining reliable DIPs from recorded earthquake time-histories has been studied for a long time. These calculated DIPs could be used to enhance the reliability of inspection and contribute to the safe early restart of nuclear power plants following earthquake. Reliable DIPs can help in (a) judging the level of operator walkdown inspection immediately after the earthquake, (b) enhancing the reliability of the initial focused inspection by estimating possible damaged SSCs failure modes, (c) evaluating the earthquake level in comparison with design earthquake more effectively. For plants utilizing an automatic seismic reactor trip system, DIPs could be used as a trip signal instead of acceleration, because the peak acceleration of an earthquake is recognized as a poor measure of the earthquake damage potential for SSCs.

This section presents earthquake DIPs calculation and the relation between DIPs and real damage caused by actual earthquake motions observed at nuclear power plants, conventional industrial facilities and in vibration tests. The material presented in this section is valuable because it uses data from recently experienced earthquakes in nuclear power plants that have been seismically designed for operational requirements.

4.1. CANDIDATE DAMAGE INDICATING PARAMETERS FOR THE NUCLEAR POWER PLANTS

The historical development of DIPs is provided in Section 3.3.1. This section focuses on the DIPs selected for the present publication.

4.1.1. Characteristics of a damage indicating parameter

Based on the records from observed earthquake motions, a wide variety of parameters indicating ‘earthquake motion intensity’, including the spectrum intensity originally proposed by Housner, have been proposed from the perspective of seismic damage. These can roughly be divided into three categories as follows:

– ‘Peak Parameter’ or ‘Momentary Parameter’

  Example: Peak value of the measured physical quantity associated with ground or in structure seismic motions, such as PGA, PGV or peak ground displacement.

– ‘Integral Parameter’ or ‘Cumulative Parameter’

  Example: Arias intensity

– Response Parameter

  Example: Response spectrum and associated average values (acceleration, velocity, energy).

As a result of a correlation study of observed damage with strong motion earthquake parameters, several DIPs have been identified by the working group during the preparation of this publication. They provide a somewhat better correlation with observed damage than the PGA or floor accelerations.

Peak ground velocity (PGV), which determines the upper levels of the MMI instrumental intensity scale (Section 3.3.2.1), and I_{JMA} value for JMA instrumental intensity scale (Section 3.3.2.2) might be better correlated with damage experience. Both of these DIPs are categorized as 'Peak Parameters'. After discussion within the working group, the equivalent effective maximum acceleration A_{JMA}, a basic intermediate result to calculate I_{JMA}, was selected as a candidate peak parameter for the further study work, mainly because of the extensive use of earthquake experience data from Japan, and also from other Member States through the seismic intensity correspondence table shown as Table 9 in Section 3.2.2.

As for integral-type DIPs to study, Arias intensity and standardized CAV were discussed within the working group, and the latter was selected as a candidate DIP for further study, based on the conclusions from the EPRI operating basis earthquake exceedance studies [39], namely:

– CAV exhibits the most consistent alignment with the threshold of seismic damage for engineered construction (i.e., building of good design and construction);

– Arias intensity is a close second choice.

In the category of 'Response Parameter' DIPs, the average 5.0 per cent damped response spectra was selected [39].

Thus, the three DIPs selected for the correlation study are: the A_{JMA} acceleration value associated with the JMA instrumental seismic intensity scale, the standardized CAV, and the average 5.0 per cent damped response spectra.

When evaluating DIPs, it needs to be carefully considered whether the input earthquake motion has been observed on the ground or at the location of the equipment, within the structures. In general, the former has abundant seismic damage data, but attention needs to be paid to the amplified response of support structures. In many nuclear power plants, seismometers are installed within the buildings. The evaluation of actual input earthquake motion for SSCs makes it easier to link it to the true damage mode or to evaluate the threshold value. This section covers the equipment installed inside and outside the buildings. In this regard, the term 'ZPA' (zero period acceleration) appears in this section hereafter with the meaning of 'maximum or peak acceleration' to cover both possibilities: on the ground and in structure.

4.1.2. Japan Meteorological Agency instrumental seismic intensity

As mentioned in Section 3.2, seismic intensity scales, such as MMI, MSK, EMS, and JMA, although their definitions slightly differ, are widely used and seismic damage data has been gathered in association with the reported intensities. As shown in Table 9, the interrelations between scales have also been studied. Since attempts have been made to calculate seismic intensity scales from the recorded values of seismic motion parameters (see Section 3.3.2), instrumental seismic intensities used for obtaining seismic intensity levels may be considered as a DIP.

In the present publication, the equivalent effective maximum acceleration A_{JMA} used for computing the JMA instrumental intensity has been selected as a representative 'Peak

Parameter', because it can be easily associated with a large amount of documented experience in real earthquakes or with experimental data

A_{JMA} is determined as shown in Section 3.3.2.2. The frequencies below about 0.5 Hz and above 10 Hz are filtered so they do not result in a significant contribution to the A_{JMA} value. The A_{JMA} value is usually measured in Gal. The JMA instrumental intensity is computed from the A_{JMA} value using Eq. (5). Hence, there is a one-to-one correspondence between the A_{JMA} value and the JMA instrumental seismic intensity I_{JMA}. However, due to the logarithmic nature of the I_{JMA}, this parameter will be saturated in the range in which the earthquake severity is high enough to cause damage to equipment with high seismic capacity, such as nuclear power plant equipment. This is not the case of the A_{JMA} value.

In addition, in the case of seismic margin assessments, which is one of the prospective uses of DIPs, a linear relationship is assumed between the seismic response and the physical quantity defining the earthquake motion, at least within an interval. In this respect, the A_{JMA} value can more easily connected with a scale factor for seismic margin than the I_{JMA} value.

The relationship among peak acceleration (ZPA), I_{JMA}, and A_{JMA} can be illustrated as shown in Fig. 20 and Fig. 21, which are based on Japan's earthquake motion data observed at nuclear power plants. Comparison between these two figures shows that A_{JMA} has a wider band in the higher acceleration range, in which seismic damage is frequently observed, and it is more appropriate than I_{JMA} for the evaluation of damage to nuclear power generating equipment because of its linearity.

One of the issues associated with the use of the A_{JMA} value as a DIP is how its relationship with damage in nuclear power plant facilities can be established, since JMA intensity is mainly linked to the damage to conventional buildings and structures. Although many of the experienced damage cases are considered to be due to first excursion damage, it may be necessary to evaluate the appropriateness of A_{JMA} application to the ductile metallic materials used for mechanical equipment in nuclear power plants.

JMA instrumental seismic intensity is essentially determined from recorded ground motions. The guidance for the environment in which seismometers are to be installed has been presented in Section 3.3.2.2. However, the A_{JMA} value itself is calculated from acceleration records, following analytical procedures empirically based on damage mechanisms. Consequently, the parameter can be used for equipment installed inside structures as well as on the ground.

4.1.3. Standardized cumulative absolute velocity

CAV parameter is an integral parameter originally defined as follows:

$$CAV = \int_0^{t_{max}} |a(t)|\, dt \qquad (6)$$

Where $a(t)$ is acceleration time-history, and t_{max} is the duration of the record.

The CAV parameter is the area under the absolute acceleration time-history. If a plot of ground motion velocity time-history is used (Fig. 22), the CAV is simply the sum of the absolute peak to valley velocity changes, that is, the CAV value can be obtained by summing (without regard to sign) the velocity changes for each peak to valley pair sequentially.

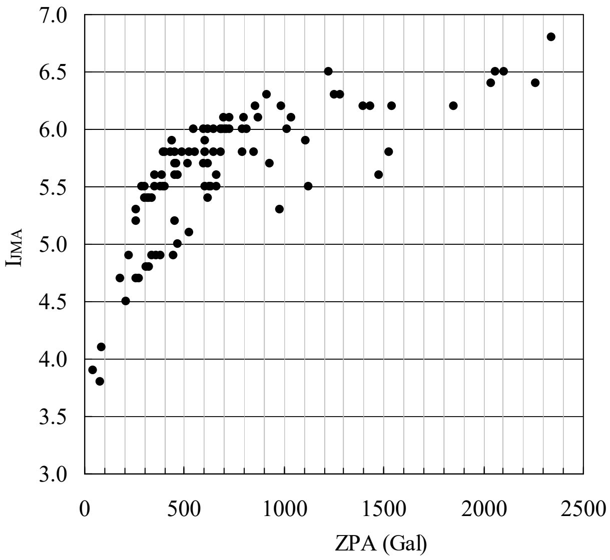


FIG. 20. I_{JMA} vs. ZPA relation typical (observed earthquake wave at nuclear power plants in Japan).

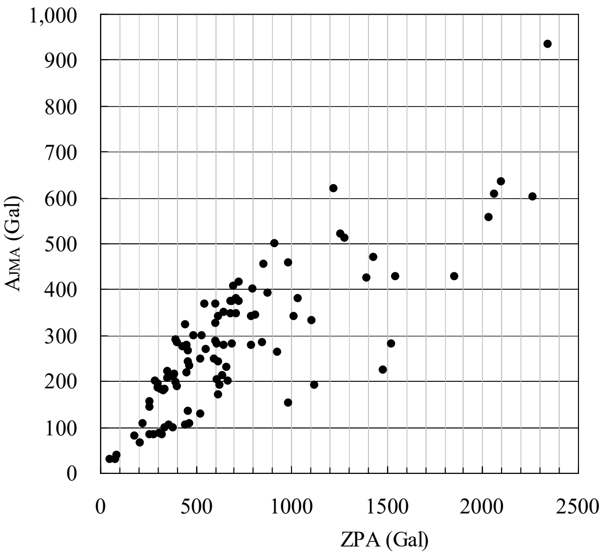


FIG.21. A_{JMA} vs. ZPA relation typical (observed earthquake wave at nuclear power plants in Japan).

The CAV parameter has been standardized by eliminating any one second duration of the recorded acceleration time-history which does not exceed 0.025 g [43]. In the example shown in Fig. 23, the duration of the time-history is 4 seconds. In the first second, the -0.025 g threshold is exceeded. The same is true for seconds 2 and 3 where both ± 0.025 g are exceeded. In second 4 there is no exceedance of either ± 0.025 g. Hence, the areas under the 4th second segment of the time-history are not included in the standardized CAV value.

Thus, the original CAV definition (Eq. 6) is revised to incrementally calculate CAV in one second intervals as follows:

$$CAV_{Total} = CAV_i + \int_{t_{i-1}}^{t_i} |a(t)|\, dt \tag{7}$$

Where $a(t)$ is acceleration value in a one second interval where at least one value exceeds 0.025g, and i goes from 1 to n , with n equal to the record length in seconds. This is termed 'standardized CAV'.

Standardized CAV introduces in the DIP an important parameter associated with damage caused by earthquakes, which is the duration of the earthquake strong motion. This is missing in 'Peak Parameter'-type DIPs.

The original definition of CAV allowed long-duration, low-level, non-damaging acceleration values to contribute to the total CAV. This fact hampered the use of the parameter as a measure of damage threshold. Standardized CAV definition assures that only significant acceleration levels contribute to the CAV measure.

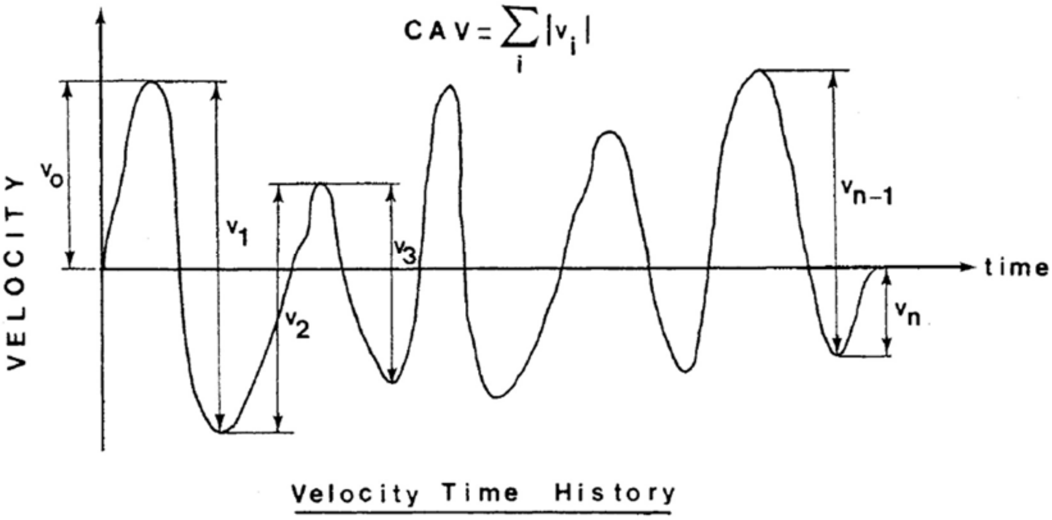


FIG. 22. Cumulative Absolute Velocity (CAV) obtained from velocity time-history [39]. (Courtesy of EPRI)

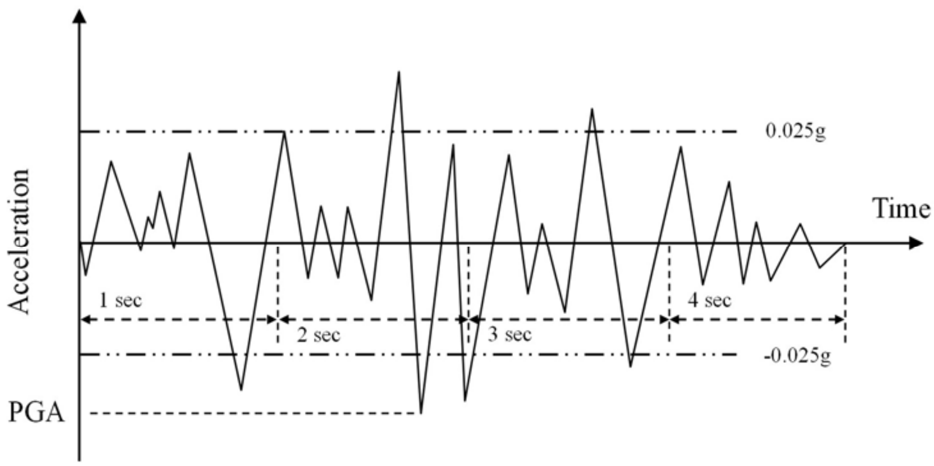


FIG. 23. Standardized CAV obtained from acceleration versus duration curve (time-history).

The Arias intensity (Eq.8) defined without the π/(2g) factor, was also considered as a damage threshold indicator in [39] and it was considered as an alternative to CAV.

$$I_A = \frac{\pi}{2g}\int_0^{T_d} a\,(t)^2\,dt \tag{8}$$

A plot of the integral in Eq. 8 as a function of time (upper limit of the integral varying from 0 to T_d), normalized by the Arias intensity, is referred to as a Husid plot. Such plots allow the strong motion portion (effective strong motion duration) of the earthquake time-history to be identified.

The equations defining both the CAV and the Arias intensity identify their DIP category as an integral parameter and, in general, the CAV and Arias intensity cumulative integral are simply alternative representations of the same information about the strength and duration of an earthquake time-history. The un-normalized Husid plot can be used to determine the effective duration of the strong motion portion of the earthquake time-history.

In the working group, the relation between the damage mode of SSCs due to earthquake motion and the CAV value was also discussed. Damage experiences, especially with earthquakes with a long strong motion duration, suggest that an integral parameter like CAV can be an important DIP. Regarding the physical meaning of CAV, focused on damage of structures subjected to seismic motions, Ref. [45] concludes as follows:

– CAV is proportional to the product of the strong motion duration and the average energy of the strong motion acceleration.

– CAV needs to be an adequate damage indicator since it is correlated with the main parameters controlling damage phenomena, that is, with the number of load cycles and its median frequency, and with the amplitude of the alternating load, which is proportional to the ground motion acceleration amplitude.

This conclusion suggests that CAV is a parameter that is deeply linked to cumulative fatigue damage.

CAV is normally calculated for each of the three directions of recorded earthquake motion with the reported value being the maximum of the three. On the other hand, the JMA instrumental seismic intensity takes all three directions into consideration. Thus, for the purpose of comparison of these parameters, this publication evaluates the resultant CAV value with the SRSS combination method for the three directional calculated CAV values.

4.1.4. Average response spectra as a damage indicating parameter

The average response spectra in a specified frequency range (i.e. 2-10 Hz) that typically cause damage to civil-mechanical SSC have also been used as a DIP. In general, based on studies of both commercial and industrial buildings and equipment, which were constructed in accordance with building code requirements, and were not specifically earthquake resistant, it has been found that the threshold for damage to these SSC for a 5 per cent damped spectrum is 0.2 g [39, 43]. This average 5.0 per cent damped spectral acceleration value in the 2 to 10 Hz range, along with a 5.0 per cent damped spectral velocity value of 15.24 cm/sec (6.0 in/sec) in the 1-2 Hz range have been used as the primary damage indication parameters in the US NRC Regulatory Guide 1.166 [46] criteria for post-earthquake evaluation. The standardized CAV value of 0.16 g-sec is used as an alternate criterion for the threshold of potentially damaging motion to SSCs. Since both the spectral criteria and the CAV value are based on the observed damage threshold of commercial and industrial SSCs, the use of such criteria for nuclear power plant SSCs is judged to be sufficiently conservative.

The criteria of US NRC Regulatory Guide 1.166 [46] are intended to prevent the premature shutdown of a plant due to nearby, small magnitude, seismic events. Based on the review of the lack of damage to fossil plants and industrial facilities when subjected to significant earthquakes, a reference spectrum (see Fig. 19) was developed [47], which represents the seismic input level (in terms of a 5 % damped response spectrum) that equipment has sustained without loss of post-earthquake function. This reference spectrum is common to eight (later revised to 20) equipment classes. It represents the input level to which at least 30 equipment items of a given class were subjected to in several earthquakes at several facilities in different earthquakes without loss of post-earthquake function. The equipment considered in the study is similar to equipment in nuclear power plants.

Associated with each equipment class is a set of caveats, or restrictions [47] associated with the construction and installation of the components which need to be verified. The maximum spectral acceleration between 2.5 and 7.5 Hz in the reference spectrum is 1.2 g (Fig. 24). This level is not a damage threshold, but a lower bound of it, since a significant number of items did not fail at this level. A statistical evaluation can demonstrate that the reference spectrum is a high-confidence-of-a-low probability-of-failure capacity level for the given equipment class.

In the frequency range above 10 Hz, the EPRI NP-5930 report [39] concluded that earthquake damage to civil-mechanical SCCs was not likely, due to the low response levels and resultant material strains. However, for electrical components and devices, malfunction due to shaking at frequencies greater than 10 Hz requires consideration. In recent years, for hard rock sites in intra-plate regions such as Scandinavia, central and eastern North America, much of the Indian subcontinent, and eastern South America, earthquake motions are expected to have spectral content in the 10-20 Hz or 20-30 Hz range. For these sites, the concern associated with electrical component or device malfunction in the greater than 10 Hz range is valid.

The seismic qualification of electrical and instrumentation and controls equipment has been typically carried out by shake table testing. In many instances, fragility testing, up to malfunction or failure, has been conducted in order to determine the margin above required qualification testing levels. Generic equipment ruggedness spectra (GERS) have been prepared [48] for several equipment classes, which document the results of these tests and the attained level of shaking up to failure (Fig. 25).

The averaging method of the response spectra used in this publication follows that of EPRI NP-5930, after a sensitivity study carried out by the working group. EPRI NP-5930 defines the method as follows [39]:

> "The averaging is performed using spectral values at frequencies evenly spaced in the logarithmic domain. This leads to a lower density of points as the frequency increases; therefore, the computed averaged spectral acceleration has a higher contribution from the low frequency range spectral values, which are more damaging."

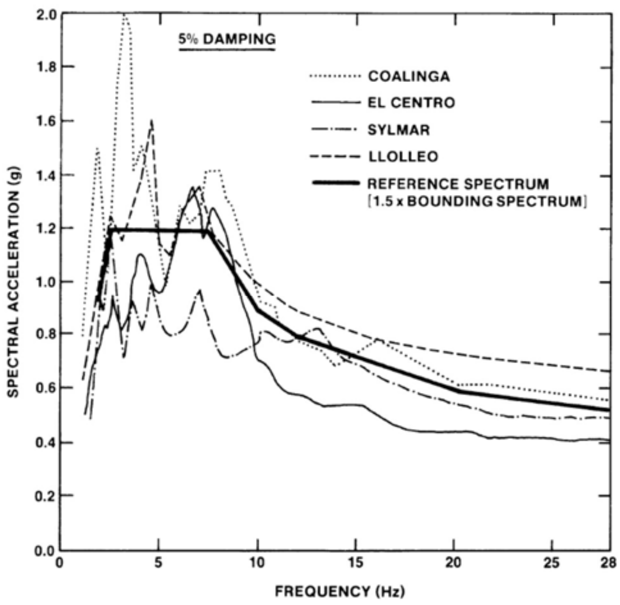
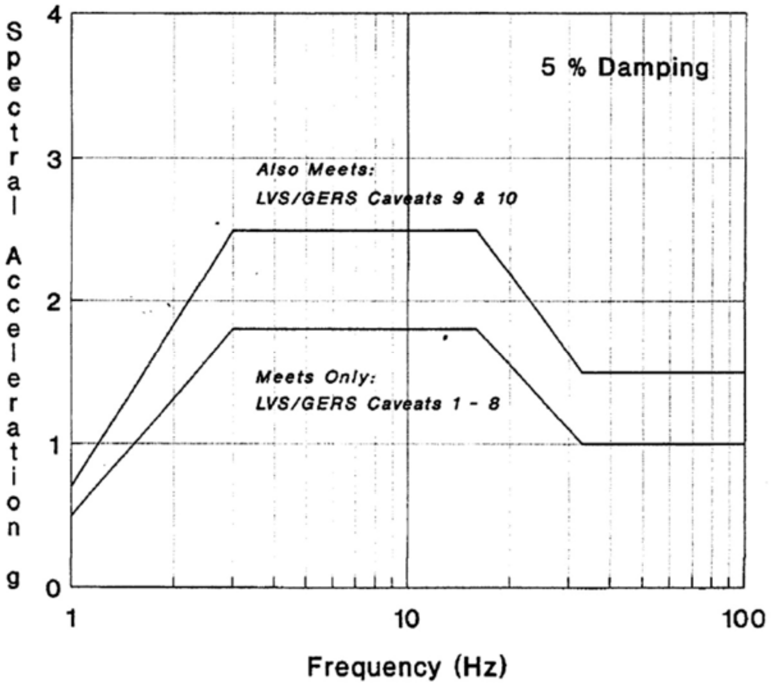


FIG. 24. Seismic Motion Reference Spectrum for lack of damage in power utility and industrial equipment given certain caveats are met for the particular class of equipment [47].

GERS-MVS/LVS.7 (Low Voltage)
2/1/91



Frequency (Hz)	1	3	16	33
Meets LVS/GERS Caveats 1 - 10 (g)	0.7	2.5	2.5	1.5
Meets LVS/GERS Caveats 1 - 8 (g)	0.5	1.8	1.8	1.0

FIG. 25. Example of generic equipment ruggedness spectra (GERS) for low voltage switchgear based on shake table test data [48]. (Courtesy of EPRI)

4.2. SEISMIC INTENSITY SCALES AND DAMAGE INDICATING PARAMETERS

As discussed in Section 3.2, most of the seismic intensity scales used around the world are defined based on seismic damage observations. The seismic intensity at the point at which the earthquake motion time-history is observed indicates the damage level in its vicinity. However, since there are few seismic damage observations of specific SSCs for which actual earthquake motion data is available, the relationship between seismic intensity scales and the DIPs selected during the preparation of this publication, is presented here as an approximate evaluation.

4.2.1. Example of evaluation in the United States: MMI scale

The standardized CAV which was originally developed from a study of over 200 earthquake time-history records, which were correlated with the observed damage to SSCs in the immediate vicinity of the recording instrument [43].

Almost all correlations of damage were for residential and light commercial SSCs that had not been designed to any engineering-based industry standard and, much less, received any seismic resistant design. A value of 0.16 g-sec standardized CAV was associated with the threshold of damage. This value is one of the DIPs listed in US NRC Regulatory Guide 1.166 [46] as a threshold value for damage to a nuclear power plant, in relation with the SL-1 earthquake exceedance.

The relationship between the computed standardized CAV and the MMI determined in the study is plotted in Fig. 26 using the data listed in Ref. [43]. As can be seen in the figure, the Standardized CAV has a wide range of variation for the same MMI intensity, when the MMI intensity is larger than grade V. In addition, it is seen that the threshold of 0.16 g-sec is an extremely conservative value, which corresponds with the minimum computed Standardized CAV at an MMI intensity of VII.

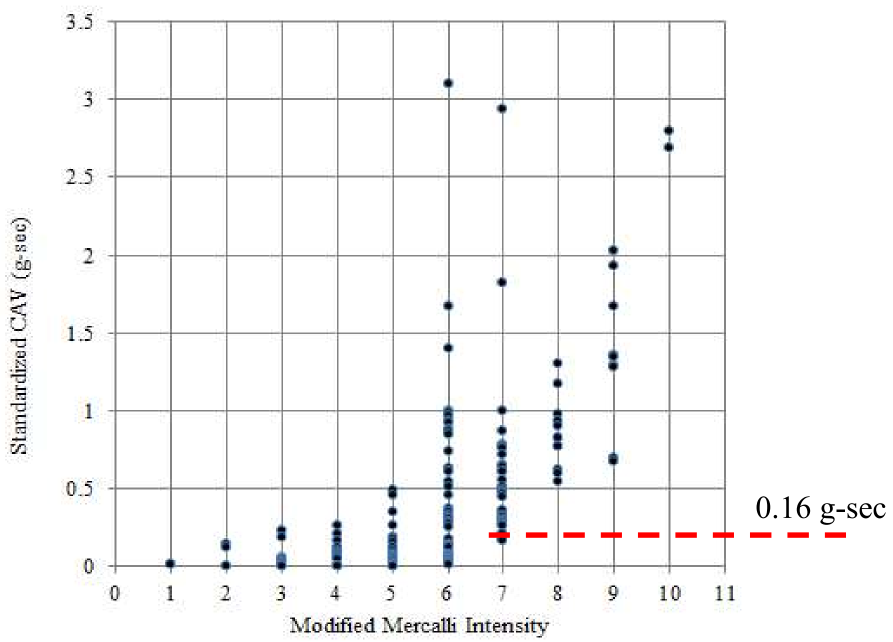


FIG. 26. Standardized CAV vs. MMI intensity scale.

The qualitative definition of level VII in the MMI intensity scale is as follows (see also Table IV-1 in Annex IV):

- Damage negligible in buildings of good design and construction; slight to moderate damage in well-built ordinary buildings; considerable damage in poorly built or badly designed buildings, adobe houses, old walls (especially if laid up without mortar), spires, etc.
- Cracked chimneys to considerable extent, walls to some extent.
- Fall of plaster in considerable to large amount, also some stucco.
- Numerous windows break, furniture to some extent.
- Loosened brickwork and tiles shake down.
- Weak chimneys at the roofline break (sometimes damaging roofs).
- Fall of cornices from towers and high buildings.
- Dislodged bricks and stones.
- Overturned heavy furniture, with damage from breaking.
- Damage is considerable to concrete irrigation ditches.

The damage threshold for buildings of good design and construction is 2.8 times higher than the 0.16 g-sec threshold. For industrial/power generating facilities that actually experienced damage, the smallest computed standardized CAV value was 0.768 g-sec, which is 4.8 times higher than the 0.16 g-sec threshold [43].

4.2.2. Example of evaluation in India: MSK scale

In the working group, the results of DIP computations for seismic motions observed in India were reported [49]. For the four earthquakes shown in Table 18, the DIPs of the seismic motions observed at a total of 69 points were calculated and the seismic damage in the vicinity of these points was identified in relation to the MSK seismic intensity scale.

What is interesting here is the relationship between the MSK seismic intensity and the JMA instrumental seismic intensity. JMA instrumental seismic intensities were calculated using the equations in Section 4.1.2, even though the type of seismometers and the guidance for installation environments (ground and geological features) are specified for the JMA seismic intensity scale and these specifications may have not been followed in the recording stations shown in Table 18. The results are shown in Fig. 27. A large scatter in JMA instrumental intensity is obtained for each value of MSK intensity when the later is larger than grade IV. Hence, the expected correlation between the two intensities, as shown in Table 9, is not found in these cases. Regression lines are shown in Fig. 27, for each of the four earthquakes. The line corresponding to the Chamoli earthquake shows an abnormal behaviour, such as a negative slope.

TABLE 18. EXAMPLE OF EVALUATION IN INDIA - SUMMARY OF EARTHQUAKE DATA

No	EQ	Date	Magnitude	Recording stations	Epicentre		Acceleration Range
					Lat (°)	Long (°)	Gal
1	Sikkim, at India Nepal Border	Sep.18, 2011	Mw = 6.9	13	27.6 N	88.2 E	0.61 – 201.65
2	Chamoli (NW Himalaya)	Mar. 29, 1999	Mb = 6.3/6.8 MS = 6.6/6.5 (USGS/IMD)	10	30.408 N	79.416 E	9.56 – 352. 84
3	Uttarkashi	Oct. 20, 1991	Mb = 6.1 (IMD) MS = 7.1 (USGS)	13	30.780 N	78.774 E	17.4 – 288.80
4	NE India, at Indo Myanmar border region	Aug. 6, 1988	Mb = 6.8 MS = 7.3	33	25.149 N	95.127 E	38.6 – 331.3

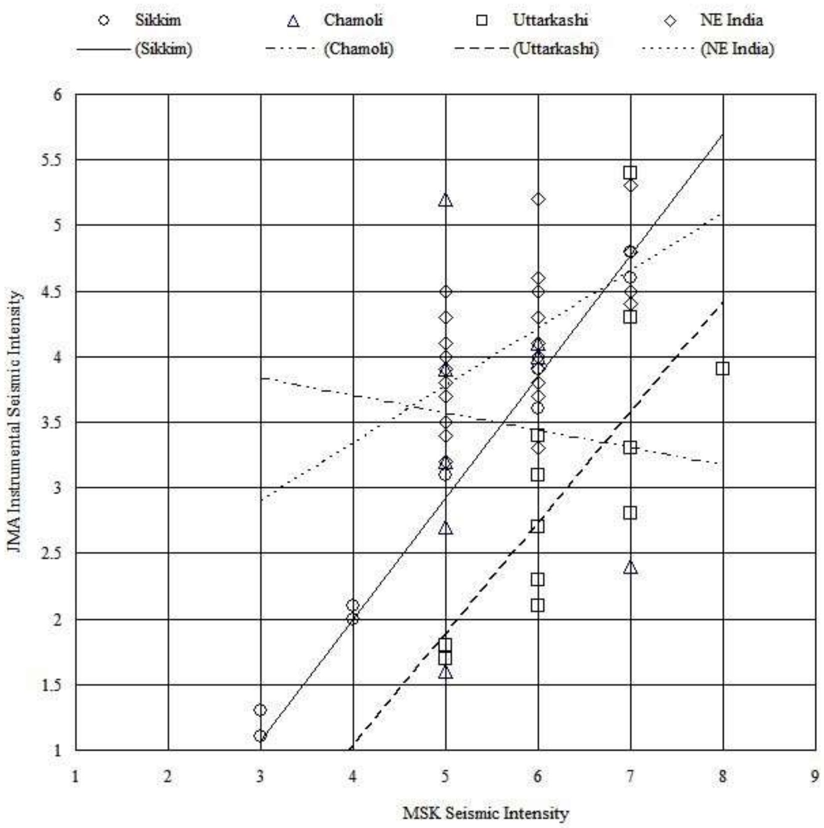


FIG. 27. Example of evaluation in India - MSK intensity vs. JMA instrumental intensity.

The Chamoli earthquake observation data is from a small magnitude earthquake with a shallow hypocentre that occurred in the Garhwat region of the Western Himalayas. According the report [49], significant disparities in damage occurred even in adjacent regions due to the effect of river terraces formed by alluvial deposits contacting sand and boulder. From the fact that earthquake motions are significantly affected by ground and geological features, the disparities at the point at which the acceleration data was observed and at the point at which the seismic damage occurred leads to the result mentioned above. While the JMA seismic intensity scale specifies the environments in which seismometers need to be installed, the measurements taken in Chamoli did not meet these requirements (see Section 3.3.2.2), making it impossible to accurately estimate the JMA instrumental seismic intensity. It can be said that this is a good example that shows the effect of ground and geological features on earthquake motions and also on damages. Seismic intensity scales are affected by the locations, observers and design standards. Thus, the result of intensity is not absolute, but relative to these parameters, as mentioned in Section 3.2.

The results of DIP computation for the four earthquakes shown in Table 18, including Chamoli earthquake are shown in Fig. 28 through Fig. 31.

As shown in Table 9, the MSK intensity grade tends to be larger than the MMI intensity grade for the same seismic damage. Taking an approach similar to the one in Ref [43], the minimum standardized CAV value is calculated for an MSK intensity of VII. This value is 0.15 g-sec, which is a slightly smaller than the 0.16 g-sec value given in Ref [46] as an operating basis earthquake exceedance criterion. However, when excluding the data of Chamori earthquake, the minimum CAV value for MSK VII for the earthquakes considered in India becomes 0.18 g-sec. Consequently, the value in Ref [46] to judge whether an observed earthquake is significant or not, that is, a standardized CAV of 0.16 g-sec, can be judged as reasonable, even for the case of India.

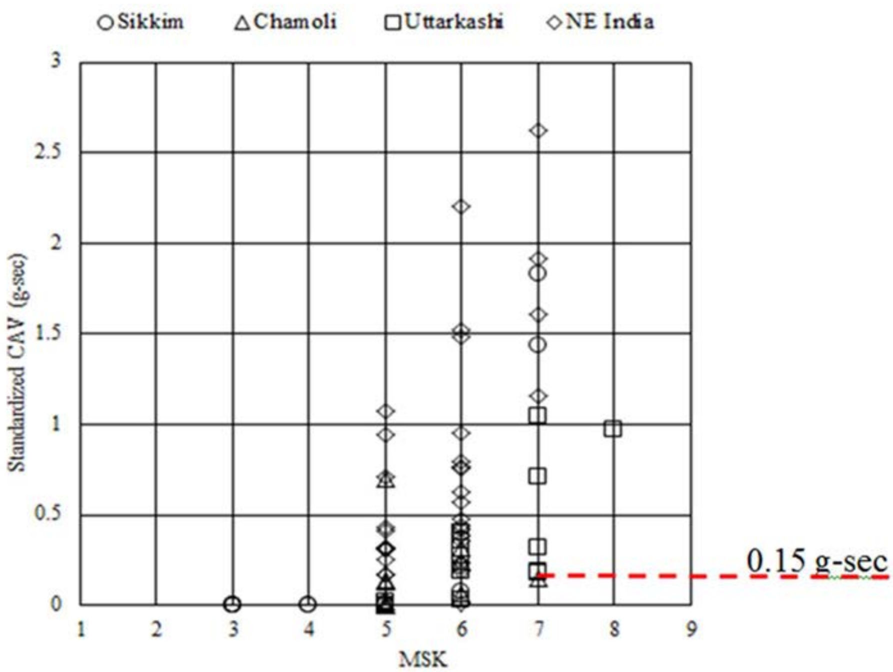


FIG. 28. Example of evaluation in India - Standardized CAV vs. MSK

In addition, some scatter in the equivalent effective maximum acceleration A_{JMA} can be seen in Fig. 29. This may be due to the environments in which the seismometers were installed, the structural seismic capacity realized by seismic design requirements and the locational influence on seismometers. It is important to be careful about the instrumentation environments, for example, to install them in the vicinity of the target facilities and with a rigid foundation.

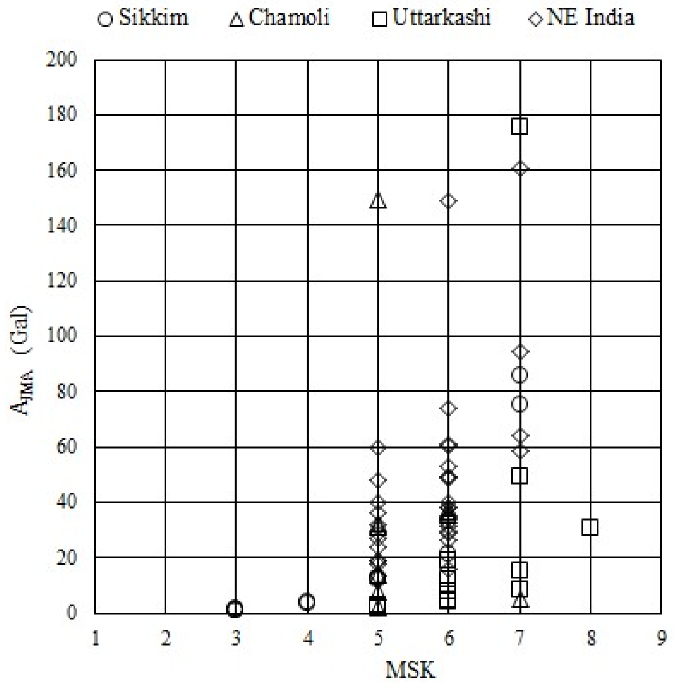


FIG. 29. Example of evaluation in India - A_{JMA} vs. MSK.

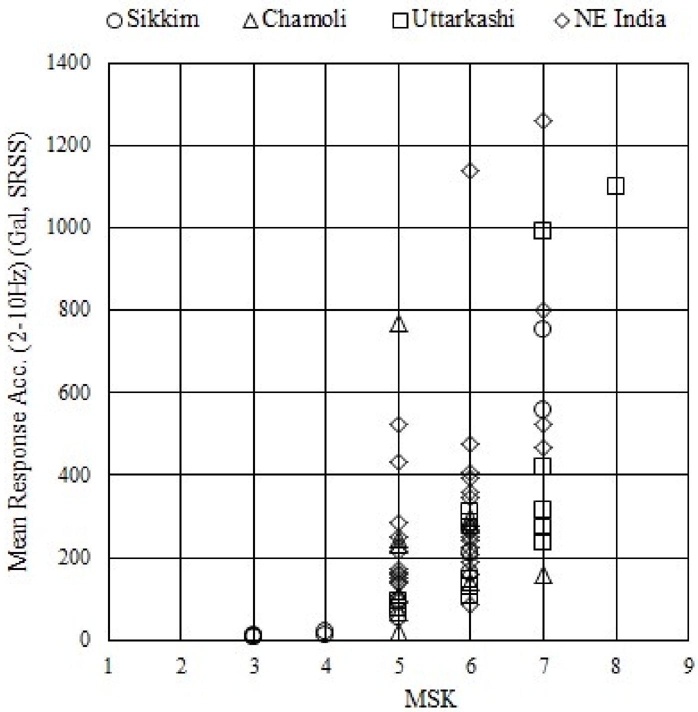


FIG. 30. Example of evaluation in India - Mean response acceleration (2-10 Hz, h=0.05, SRSS) vs. MSK.

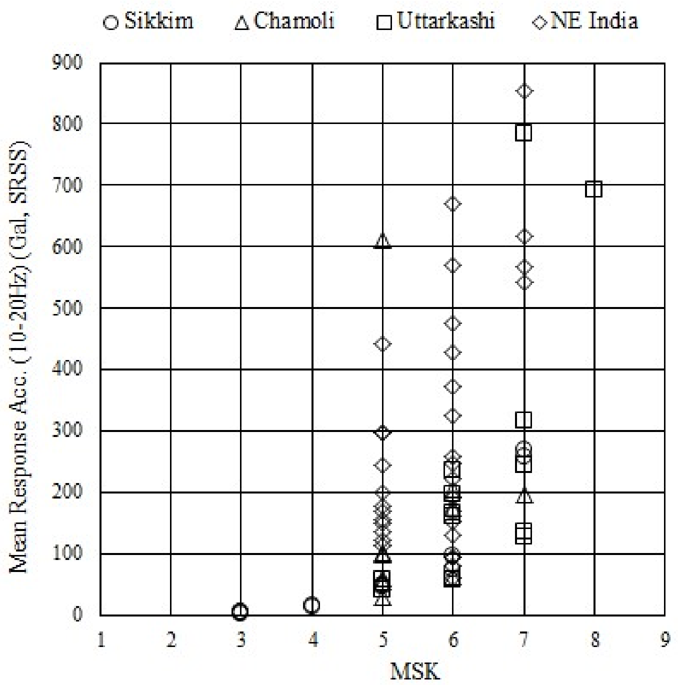


FIG. 31. Example of evaluation in India - Mean response acceleration (10-20 Hz, h=0.05, SRSS) vs. MSK.

4.3. DAMAGE TO CONVENTIONAL INDUSTRIAL FACILITIES AND DAMAGE INDICATING PARAMETERS

4.3.1. Analysis in the United States of America

At least eight fossil fuel power stations in the United States have experienced strong motion earthquakes with PGAs at the site of 0.2 to 0.4 g. The behaviour of these facilities during the earthquakes is described in Ref [50].

Based on the seismic damage data base (EPRI Seismic Qualification Utility Group database) including 29 earthquakes and 176 sites, the EPRI studied the relationship between seismic damage and DIP for the two earthquakes shown below, Imperial Valley and Coalinga earthquakes, in facilities which included many equipment items that are common in nuclear power stations [51].

The seismic motions observed during the 1979 Imperial Valley California earthquake had a maximum acceleration of north-south 0.48 g, east-west 0.35 g and vertical 0.71 g at the El Centro Steam Plant. When the earthquake occurred, Units 1 and 2 were in out of service and Units 3 and 4 were in operation. However, Unit 4 tripped due to the earthquake.

The seismic motions observed during the 1983 Coalinga California earthquake had the maximum acceleration of horizontal 0.59 g, 0.55 g, and vertical 0.35 g (observed on ground slope) and 0.38 g (on the base), at the Pleasant Valley Pumping Plant (founded on alluvial soil).

The DIPs calculated based on these observed earthquake data are shown in Table 19 and all of them are intensity ‘6 lower’ in the JMA seismic intensity scale, at which equipment damage can be anticipated in reference to the Table IV-4 in Annex IV.

TABLE 19. SUMMARY OF DIP VALUES ASSOCIATED WITH THE TWO SELECTED EPRI SQUG DATABASE SITES (COURTESY OF EPRI)

	Earthquake	Facility
*	1979 Imperial Valley California earthquake Date & Time: 16:16, October 15, 1979 Magnitude: 6.6 Duration of earthquake motions: 10 to 15 seconds	El Centro Steam Plant (4 units) Distance from epicentre: 25 km Observation point: 1 km away from the plant
*	1983 Coalinga California earthquake Date & Time: 16:42, May 2, 1983 Magnitude: 6.7	Pleasant Valley Pumping Plant, Distance from epicentre: 9.2 km Observation point: at the switching station Coalinga Water Treatment Plant, Shell Water Treatment Plant, Distance from epicentre: 4 km Getty Oil Pumping Plant, Distance from epicentre: 6 km Union Oil Butane Plant Distance from epicentre: 3 km

Site	Component	PGA (g)	CAV-S (g-sec)	JMA	SA ave2-10 (g)	SA ave10-20 (g)
PVPP yard	45 degrees 135 degrees Vertical	0.59 0.55 0.35	1.52 1.46 0.89	5.93	1.20 1.47 0.96	0.86 0.68 0.87
ECSP DA	270 degrees 360 degrees Vertical	0.35 0.48 0.71	0.91 0.95 0.57	5.71	1.07 1.05 0.75	0.54 0.55 1.83

Notes:

PVPP yard = Pleasant Valley pumping plant switchyard (1983 Coalinga earthquake)
ECSP DA = El Centro steam plant differential array (1979 Imperial Valley earthquake)
PGA = Peak Ground Acceleration
CAV-S = Standardized CAV
SA ave2-10 = Average spectral acceleration 2-10 Hz
SA ave10-20 = Average spectral acceleration 10-20 Hz
JMA = JMA intensity (single value computed given three components)

As major seismic effects, it is reported a three-inch displacement of unanchored motor control centre, a two-inch displacement of substation switching gear, damaged connecting piping due to rocking and elongation of tank anchor bolts, elephant-foot buckling of the oil storage tank at Shell water treatment plant, sheared and displaced switch gear anchor bolts, some loosed anchor bolts (1/2 inch) of main control panels, damaged sampling piping due to relative displacement of racks, and so forth. Excluding the elephant-foot buckling of the oil storage tank, all of these effects are considered to have occurred due to unanchored equipment or vulnerable anchor structures.

The type and quantity of the components registered on the EPRI SQUG database and the observed seismic damage are shown in Table 20. As EPRI has basic requirements for improving seismic performance, the SQUG caveats, the table has categorized the data as to whether these requirements are met. Basically, EPRI’s stance is that the aforementioned damage would not have occurred if the SQUG caveats had been met. The civil structures (i.e. buildings) affected by the earthquakes were designed for a static seismic coefficient of 0.2 g. It needs to be noted that plant equipment damage was mitigated because these structures were not damaged.

4.3.2. Analysis in Japan

In the 1995 Southern Hyogo prefecture earthquake (Great Hanshin-Awaji earthquake) a large number of power stations and substations were damaged within the service area of Kansai Electric Power Co. [52].

Kansai Electric Power Company recorded acceleration time-histories at 18 sites; those are power plants, substations and their technical institutes. In this report, eight of them are selected as typical observed earthquake motions for the calculation of DIPs at power facilities, as shown below:

- Amagasaki No.3 power station (Fossil, 156 MWe, 3 units, epicentral distance: 34.2 km)
- Takasago power station (Fossil, 450 MWe, 2 units, epicentral distance: 34.2 km)
- Nanko generating station (Fossil, 600 MWe, 3 units, epicentral distance: 29.1 km)
- Gobo power station (Fossil, 600 MWe, 3 units, epicentral distance: 34.1 km)
- Akoh power station (Fossil, 600 MWe, 2 units, epicentral distance: 62.3 km)
- Shin-Kobe substation (275 kV, epicentral distance: 24.7 km)
- Kainanko substation (275 kV, epicentral distance: 51.4 km)
- Nishi-Kyoto substation (500 kV, epicentral distance: 67.6 km)

Typical recorded acceleration time-histories are shown in Fig. 32 and Fig. 33. The data in Fig. 33 shows that the nature of the waveform changed during the earthquake. This is thought to be due to the liquefaction of the ground on which the seismometer was installed.

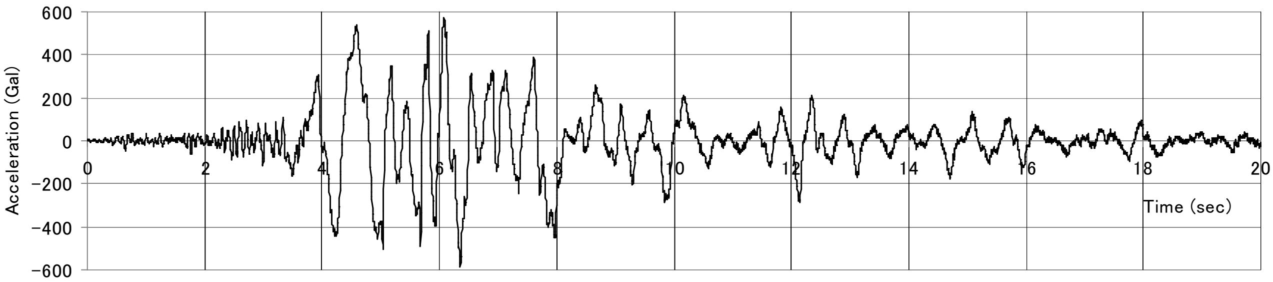


FIG. 32. Acceleration time-history observed at Shin-Kobe substation.

TABLE 20. SUMMARY OF ITEMS LOCATED AT THE TWO SELECTED EPRI SQUG DATABASE SITES (COURTESY OF EPRI)

		Selected EPRI/SQUG Database Earthquakes			Equipment Meeting SQUG Caveats			Equipment Not Meeting SQUG Caveats		
No.	EPRI/SQUG Equipment Class	Imperial Valley Earthquake Items	Coalinga Earthquake Items	Total Items	Success	Damage	Malfunction	Success	Damage	Malfunction
1	Motor control centers	6	18	24	20	0	0	0	4	0
2	Low voltage switchgear	6	1	7	7	0	0	0	0	0
3	Medium voltage switchgear	7	12	19	16	0	0	0	3	0
4	Transformers	12	7	19	16	0	0	1	2	0
5	Horizontal pumps	13	43	56	52	0	0	0	4	0
6	Vertical pumps	3	24	27	27	0	0	0	0	0
7	Air operated valves	25	50	75	74	0	0	0	0	1
8	Motor operated valves	15	72	87	87	0	0	0	0	0
9	Fans	4	4	8	8	0	0	0	0	0
10	Air handlers	0	8	8	8	0	0	0	0	0
11	Chillers	0	2	2	2	0	0	0	0	0
12	Air compressors	4	22	26	26	0	0	0	0	0
13	Motor generators	1	0	1	1	0	0	0	0	0
14	Distribution panels	2	4	6	5	0	0	0	1	0
15	Battery racks	3	9	12	7	0	0	4	0	1
16	Battery chargers and inverters	3	8	11	8	0	0	3	0	0
17	Engine generators	0	1	1	1	0	0	0	0	0
18	Instruments on racks	9	27	36	33	0	0	0	3	0
19	Temperature sensors	1	5	6	6	0	0	0	0	0
20	Control and instrumentation panels and cabinets	6	13	19	15	0	0	0	2	2
	Totals	120	330	450	419	0	0	8	19	4

Note: SQUG Caveat: An equipment installation condition which needs to be satisfied, for instance, (1) item needs to have engineered anchorage or installation; (2) item does not have potential for seismic interaction with adjacent structures, systems, or components; (3) item needs to have line connections (conduit, piping, tubing) which have adequate flexibility, etc.).

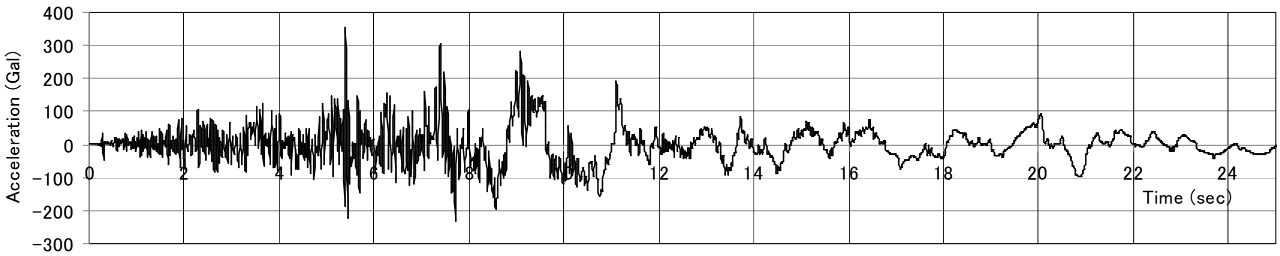


FIG. 33. Acceleration time-history observed at Amagasaki No.3 power station.

TABLE 21. ANALYSIS IN JAPAN FOR INDUSTRIAL FACILITIES - COMPUTED DIP VALUES AT THE SITES

Observation Point		PGA (ZPA)	Standardized CAV	A_{JMA}	JMA Instrumental Seismic Intensity	Average Elastic Spectrum[1] (2-10 Hz)	Average Elastic Spectrum[1] (10-20 Hz)	Average Elastic Spectrum[1] (20-30 Hz)
		Gal	g-sec	Gal	-	Gal	Gal	Gal
Amagasaki 3 Power Station	NS	227	0.99	174	5.4	335	422	336
	EW	354	0.96			387	859	792
	UD	373	0.91			581	1000	6633
Takasago Power Station	NS	191	1.48	178	5.4	376	206	193
	EW	198	1.70			342	205	200
	UD	182	0.94			321	415	207
Nanko Generating Station	NS	107	0.55	94	4.8	206	229	135
	EW	126	0.72			220	314	149
	UD	199	0.61			347	475	283
Gobo Power Station	NS	60	0.11	27	3.8	148	86	63
	EW	74	0.16			180	101	77
	UD	26	0.01			38	84	44
Akoh Power Station	NS	104	0.46	35	4.0	180	215	150
	EW	84	0.37			156	145	122
	UD	122	0.32			142	232	182
Shin-Kobe Substation	NS	511	1.34	416	6.1	1110	675	555
	EW	584	1.76			1433	872	693
	UD	495	1.11			1055	1058	994
Kainanko Substation	NS	98	0.36	69	4.6	277	139	103
	EW	128	0.39			267	163	132
	UD	92	0.17			128	160	116
Nishi-Kyoto Substation	NS	114	0.24	79	4.7	317	155	118
	EW	129	0.24			349	242	140
	UD	83	0.17			163	192	105

Notes: (1) 5% damping; 80 evenly spaced points in the logarithmic frequency domain

DIPs computed from the earthquake motion data recorded at these sites are shown in Table 21.

Due to these earthquake motions, turbine trips occurred at a large number of thermal power plants as a result of excessive turbine shaft vibrations. Those at which no damage was found were restarted, while many power plants had to be shut down for extended periods of time.

At the power stations and substations for which DIPs were computed, damage listed in Table 22 was reported.

TABLE 22. MAJOR DAMAGE OBSERVED AT CONVENTIONAL POWER STATIONS AND SUBSTATIONS

No.	Station	Major Damages
1	Amagasaki No.3 power station	• Damaged boiler anti-vibration devices (16 sections) • Damaged anti-vibration devices for downcomer tube of main steam line (4 sections) • Deformed boiler frames (braces) (13 sections)
2	Takasago power station	• Cracked on-site roads and others
3	Nanko power station	• Deformed boiler cooling spacer tubes (5 tubes) • Deformed anti-vibration devices for low-temperature reheated steam tube of main steam line (2 sections)
4	Gobo power station	No damage
5	Akoh power station	No damage
6	Shin-Kobe substation	• Fractured and moved 275 kV transformer foundation anchor bolts • Dislocated 275 kV breaker bushing • Fractured pressure relief panel of 275 kV transformer • Fractured 77 kV breaker bushing • Fractured frame supporting insulator for 77 kV power capacitor • Fractured radiator piping of 77 kV bypass reactor
7	Kainanko substation	No damage
8	Nishi- Kyoto substation	No damage

In Fig. 34 through Fig. 36 solid markers identify the cases in which some sort of damage was observed. In Fig. 34 the data from the EPRI SQUG database analysis mentioned in the previous section are also plotted (solid markers are used because damage is observed in these examples). The standardized CAV observed at Takasago power station is relatively large, but no significant damage of plant facilities was reported, even though the steam turbine was tripped due to the excessive turbine vibration. It also needs to be noted that cracks on station roads were observed. The Takasago power station restarted successfully around two hours after the earthquake.

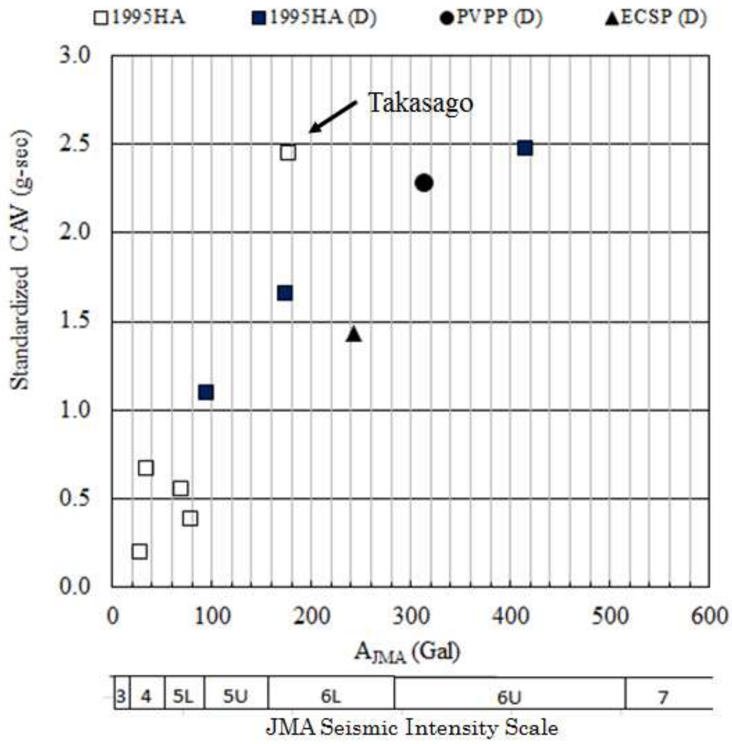


FIG. 34. Analysis in Japan for conventional facilities - Standardized CAV vs. A_{JMA}.

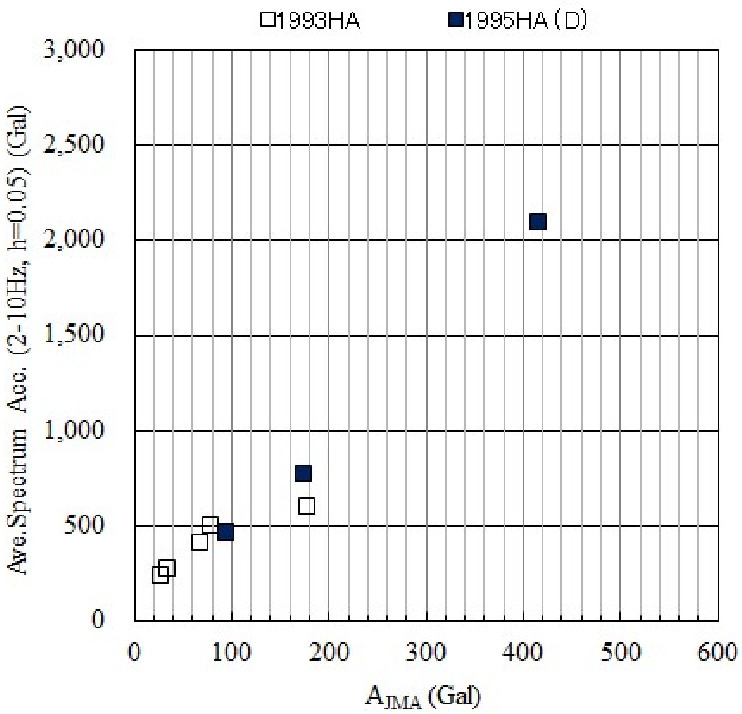


FIG. 35. Analysis in Japan for conventional facilities- Mean response acceleration (2-10 Hz, h=0.05, SRSS) vs. A_{JMA}.

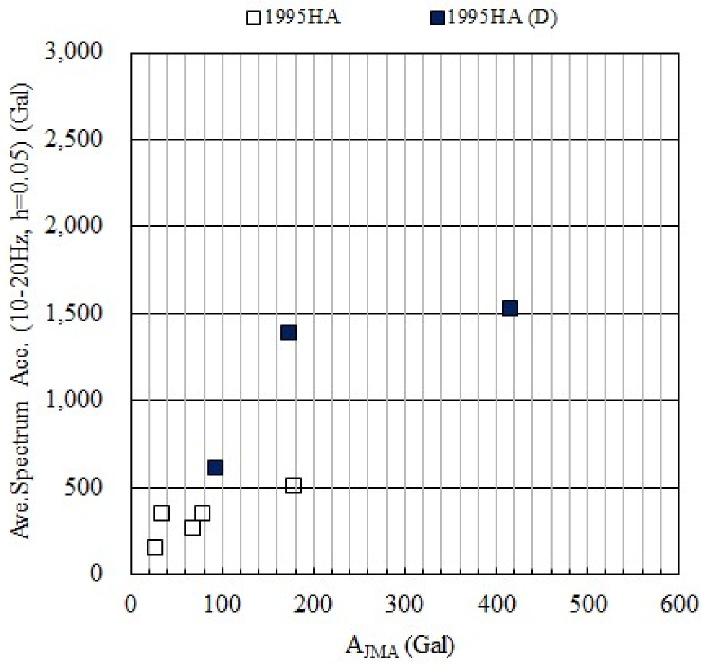


FIG. 36. Analysis in Japan for conventional facilities- Mean response acceleration (10-20 Hz, h=0.05, SRSS) vs. A_{JMA}.

As seen in Fig. 34, the JMA seismic intensity level that serves as a threshold at which some sort of seismic damage will occur to plant equipment is 5U, approximately. This value is close to the value discussed in Section 3.2.3, where the original JMA seismic intensity scale 5 was seen as a threshold of damage to industrial conventional equipment. The good correspondence between calculated and observed JMA value encourages application of I_{JMA} not only to buildings but also to mechanical equipment, and A_{JMA} to be studied further as a promising DIP for nuclear power plant SSCs.

4.4. DAMAGE TO NUCLEAR POWER PLANT FACILITIES AND DAMAGE INDICATING PARAMETRS

4.4.1. Analysis in Japan

4.4.1.1. Strong earthquake motions observed at nuclear power plants in Japan

DIPs are analysed for six earthquakes in Japan. The first three earthquakes are three strong motion, potentially damaging earthquakes, where the reactors were automatically shut down by ASTS signals and some significant damage to Seismic Class B and C SSCs were observed. These earthquakes are as follows:

- 2007 Niigata-ken Chuetsu-Oki earthquake, at the Kashiwazaki-Kariwa nuclear station;
- 2009 Suruga Bay earthquake, at the Hamaoka nuclear station;
- 2011 off-the-Pacific-coast-of-Tohoku earthquake, at the Onagawa, Fukushima Daiichi, Fukushima Daini and Tokai nuclear power plants;

Two other earthquakes also affected nuclear power plants resulting in minor damage to non-safety SSCs. However, maximum acceleration value at the site exceeded the ASTS trigger level:

– 2005 Miyagi Offshore (Miyagi-oki) earthquake, at the Onagawa nuclear station;

– 2007 Noto Hantou earthquake, at the Shika nuclear station.

Other earthquakes have affected a nuclear power plant without any damage or reactor trip by ASTS. For example, the following earthquake has been considered:

– 2009 off Fukushima prefecture earthquake, at the Fukushima Daiichi, Fukushima Daini nuclear station.

Table 23 summarizes the earthquake data and the effects on plant condition.

As for the 2011 off-the-Pacific-coast-of-Tohoku earthquake, due to the huge source area of the earthquake, the characteristics of the observed ground motions are very different from one observation point to the other. Fig. 37 shows the trend of the observed earthquake ground motions in east Japan and the locations of Onagawa, Fukushima dai-ichi, Fukushima dai-ni and Tokai nuclear power plants.

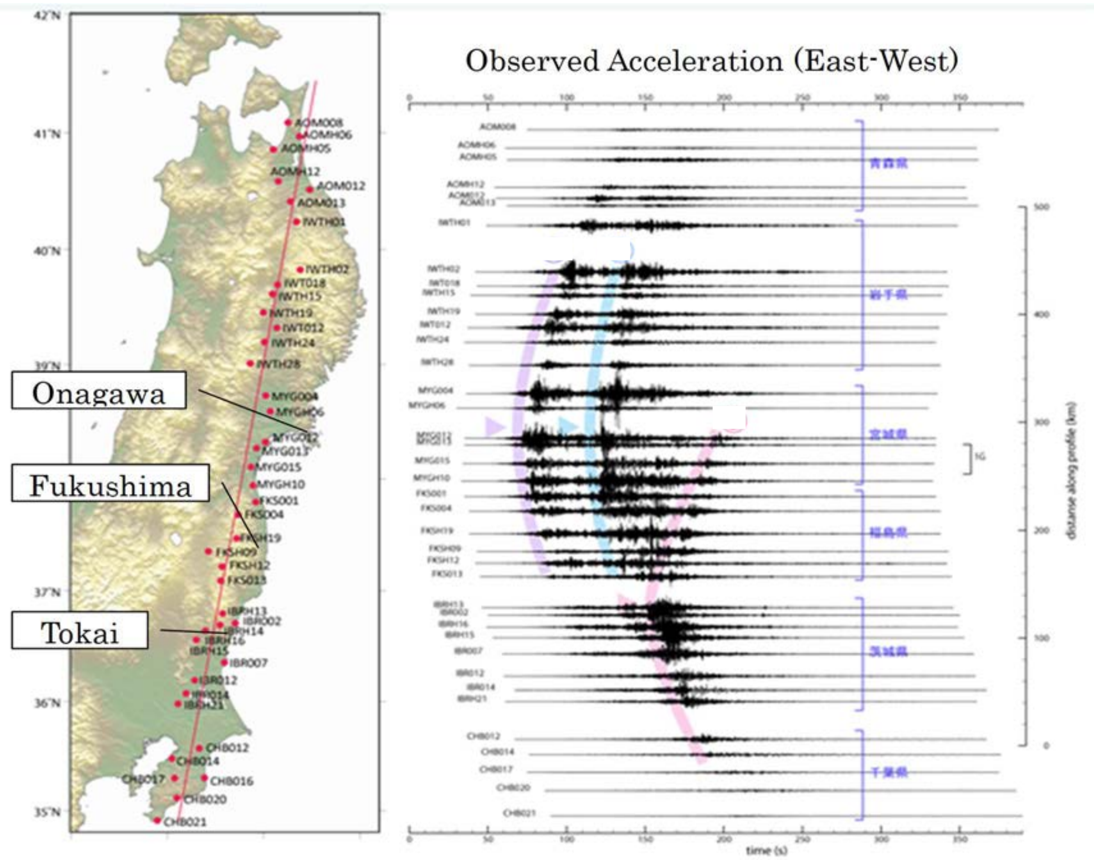


FIG. 37. Nuclear power plant locations and trend of observed earthquake ground motions (from: Earthquake Research Institute of Tokyo University home page: http://outreach.eri.u-tokyo.ac.jp/eqvolc/201103_tohoku/#erismdata).

TABLE 23. ANALYSIS IN JAPAN FOR NUCLEAR PLANTS - EARTHQUAKE AND DAMAGE DATA FOR DIPs STUDY

Earthquake	Earthquake Event Data
Miyagi Offshore Earthquake (11:46 am, 16 August 2005)	Earthquake data
	M_{JMA} = 7.2
	Epicentral distance: 73 km southwest of site
	Focal depth: 42 km
	Condition of Onagawa nuclear power plant
	Units No. 1, 2 and 3 in full power operation
	Automatic shutdown by ASTS activation
	No damage to safety-related structures, systems or components
Noto Hantou Earthquake (09:42 am, 25 March 2007)	Earthquake data
	M_{JMA} = 6.9
	Epicentral distance: 18 km north from site
	Focal depth: 11 km
	Condition of Shika nuclear power plant
	Units No. 1 and 2, in outage for maintenance
	Unit No. 1: water spilled over from spent fuel pool due to sloshing
	Units No. 1 and 2: flashing of over head mercury-vapor lamps
	Unit No. 2: displacement of turbine rotors that were on the floor, in the process of assembling
Niigata-ken Chuetsu-Oki Earthquake (10:13 am, 16 July 2007)	Earthquake data
	M_{JMA} = 6.8
	Epicentral distance: 16 km north from site
	Focal depth: 17 km
	Condition of Kashiwazaki-Kariwa nuclear power plant
	Units No. 1, 5 and 6, in outage for maintenance. Signal from ASTS activated
	Unit No. 2, in start-up operation after scheduled outage, automatic shutdown by ASTS activated
	Units No. 3, 4 and 7 in full power operation, automatic shutdown by ASTS activated
	No damage to safety-related structures, systems or components
	For miscellaneous damage, see Table VII-1 in Annex VII.
Off-Fukushima Prefecture Earthquake (5:08 pm, 14 March 2009)	Earthquake data
	M_{JMA} = 6.7
	Epicentral distance: 77 km east from Fukushima Dai-ichi site
	Epicentral distance: 83 km east from Fukushima Dai-ni site
	Focal depth: 40 km
	Condition of Fukushima Dai-ichi and Fukushima Dai-ni nuclear power plants
	All units: no damage, no automatic shutdown by ASTS was activated
Suruga Bay Earthquake (5:07 am, 11 August 2009)	Earthquake data
	M_{JMA} = 6.5
	Epicentral distance: 37 km northeast of site
	Focal depth: 23 km
	Condition of Hamaoka nuclear power station
	Units No. 1 and 2, in outage for decommissioning
	Units No. 3, 4 and 5, in full power operation, automatic shutdown by ASTS activation
	Units No. 3 and 4, no damage
	Unit No. 5, slight lift up of middle standard bearing box (thrust bearing of main steam turbine rotor) and deformation of the fixing bolts and keys.

TABLE 23. ANALYSIS IN JAPAN FOR NUCLEAR PLANTS - EARTHQUAKE AND DAMAGE DATA FOR DIPs STUDY (cont.)

Earthquake	Earthquake Event Data
Off-the-Pacific-Coast-of-Tohoku Earthquake (2:46 pm, 11 March 2011)	Earthquake data
	$M_W = 9.0$
	Condition of Onagawa nuclear power plant
	Units No. 1 and 3, in full power operation, automatic shutdown by ASTS activation. No damage to safety-related structures, systems or components due to earthquake motion. Unit No. 2, in start-up operation after scheduled outage, automatic shut-down by ASTS activation. No damage to safety-related structures, systems or components due to earthquake motion. Unit No. 1, fire of motor control center (high voltage power supply) in Turbine Building Units No. 2 and 3, slight movement of main steam turbine rotor.
	For miscellaneous damage, see Table VII-3 in Annex VII
	Condition of Fukushima Dai-ichi nuclear power plant
	Units No. 1, 2 and 3 in full power operation, and damaged by tsunami after automatic shutdown by ASTS activation. No damage to safety-related structures, systems or components due to earthquake motion. Units No. 4, 5 and 6, in outage for maintenance and refuelling. Damaged by tsunami. No damage to safety-related structures, systems or components due to earthquake motion. Demineralized water storage tank in yard showed elephant-foot buckling.
	Unit No. 5, break of small pipe connected to drainpipe of turbine moisture separator.
	Condition of Fukushima Dai-ni nuclear power plant
	Units No. 1, 2, 3 and 4, in full power operation, and damaged by tsunami after automatic shutdown by ASTS activation. No damage to safety-related structures, systems or components due to earthquake motion.
	Water storage tank in yard showed elephant-foot buckling.
	Condition of Tokai Dai-ni nuclear power plant
	Unit in full power operation, automatic shut down by ASTS activation.
	Slight movement of the main steam turbine rotor
	Rod break in an oil snubber connected to turbine moisture separator

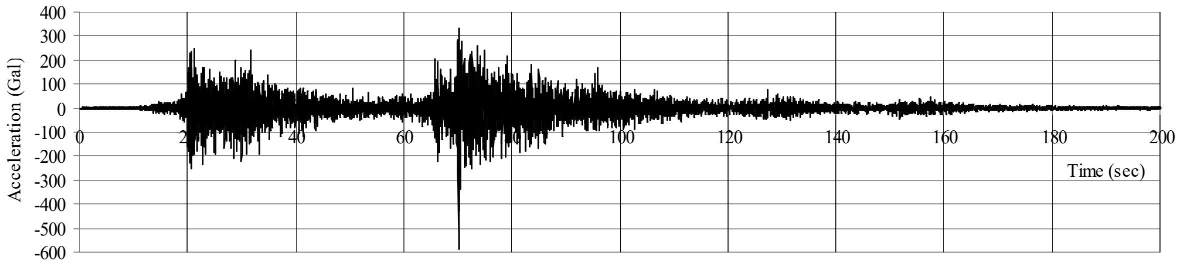
Note: Markers shown in the figures that will be shown hereafter are as follows:

Marker	**Earthquake**	**Nuclear Power Plant**
□O	Miyagi Offshore Earthquake	Onagawa
△S	Noto Hantou Earthquake	Shika
◇KK	Niigata-ken Chuetsu-Oki earthquake	Kashiwazaki-Kariwa
△F	Off Fukushima Prefecture Earthquake	Fukushima-1 & 2
○H	Suruga Bay Earthquake	Hamaoka
□Onagawa	Off the Pacific coast of Tohoku Earthquake	Onagawa
○Fuku-1	(ibid)	Fukushima Daiichi
△Fuku-2	(ibid)	Fukushima Daini
◇Tokai	(ibid)	Tokai Daini

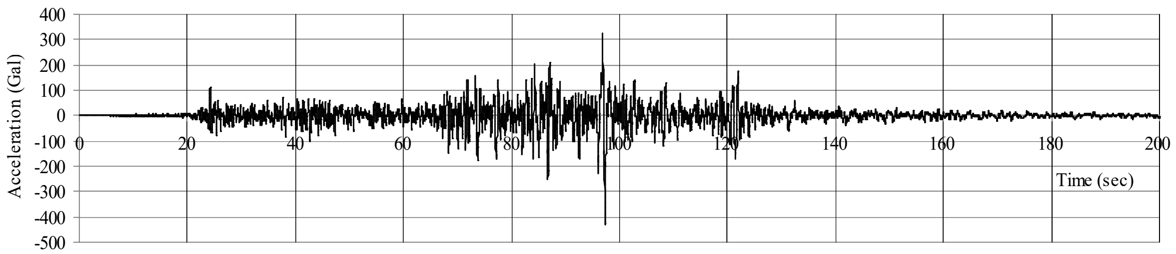
Solid markers with (D) indicate a location where damage was observed

The earthquake motions observed at those four plants during the earthquake are very unique for each location, and the typical in-structure acceleration time-histories are shown in Fig. 38. The acceleration time-history observed at the Kashiwazaki-Kariwa nuclear station during the 2007 Niigata-ken Chuetu-Oki earthquake is also shown in Fig. 38 (e), as a reference to compare the different characteristics of the motion.

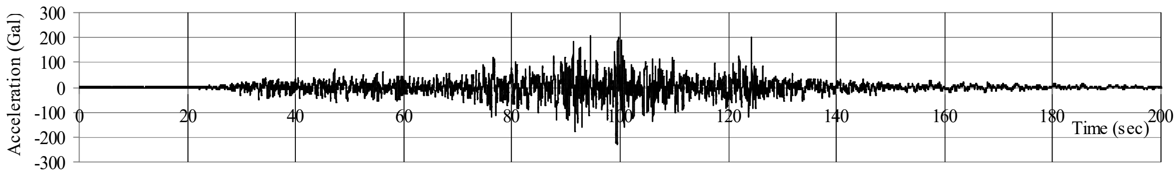
a) Onagawa unit 1 Reactor Building Base Mat (E-W direction)



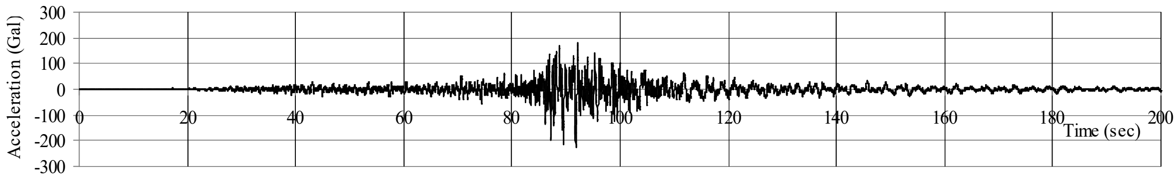
b) Fukushima-daiichi Unit 6 Reactor Building Base Mat (E-W direction)



c) Fukushima-daini Unit 1 Reactor Building Base Mat (E-W direction)



d) Tokai Daini Reactor Building Base Mat (E-W direction)



e) Kashiwazaki-Kariwa Unit 1 Reactor Building Base Mat (E-W direction)

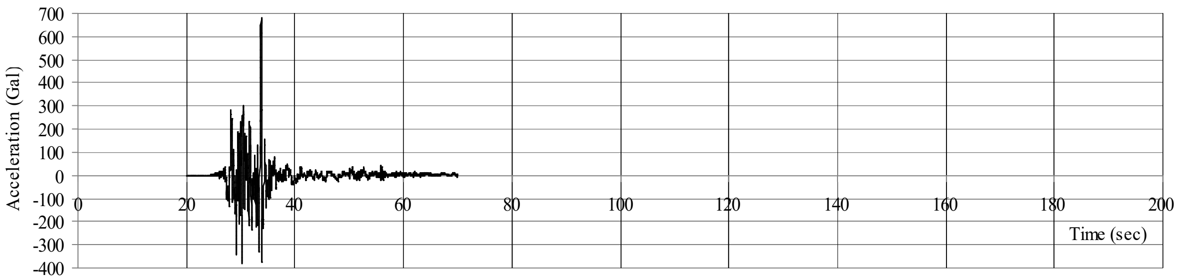


FIG. 38. Analysis in Japan for nuclear plants-Typical in-structure acceleration time-histories

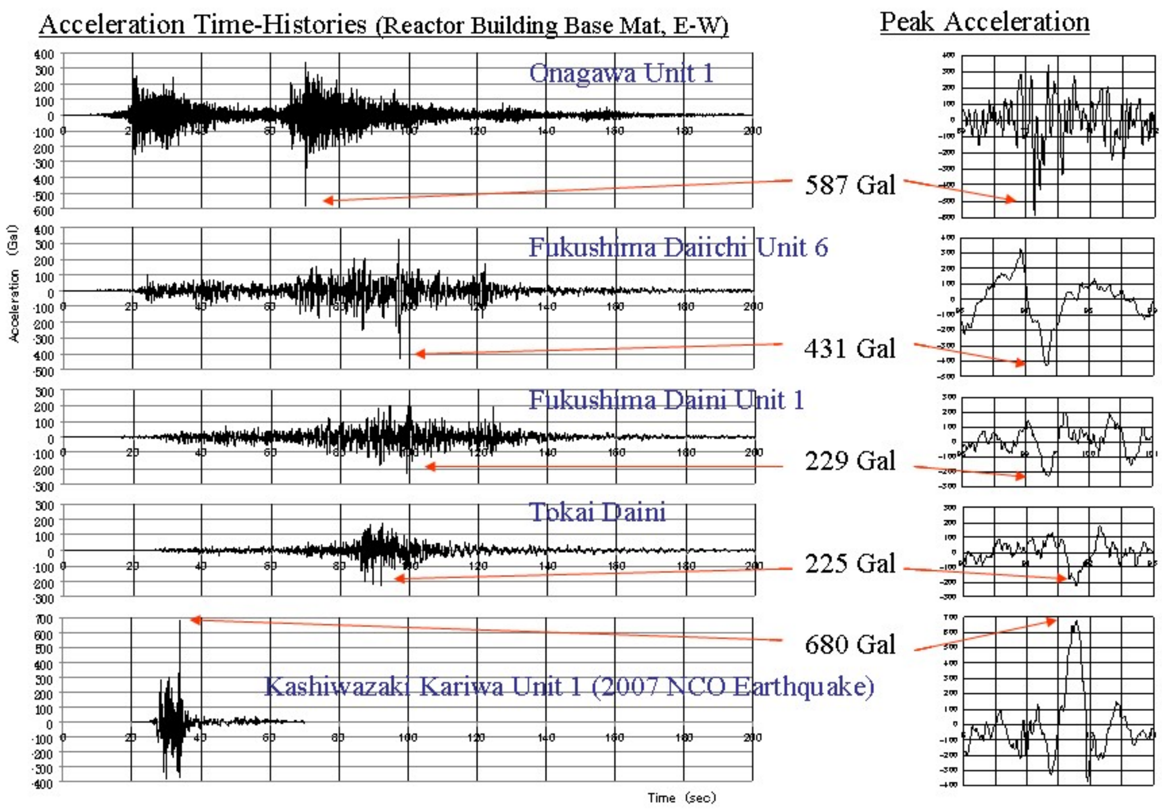


FIG. 39. Analysis in Japan for nuclear plants - Peak acceleration shapes and peak acceleration values.

The uniqueness of the 2011 off-the-Pacific-coast-of-Tohoku earthquake, from the viewpoint of possible damage to the SSCs, is the very long duration of the motions, due to the large magnitude (Mw=9.0), when compared with those earthquakes experienced earlier, such as (e) in Fig. 38.

As addressed in Annex VI, the time-width of the acceleration peak has a strong influence on seismic damage and on DIP values as well. Fig. 39 shows the typical acceleration peak shapes and the associated peak values. From this chart, it is easy to recognize that the peak acceleration value observed at Onagawa plant corresponded to a sharp acceleration spike, which did not produce much damage, even though the peak value was high when compared with other sites.

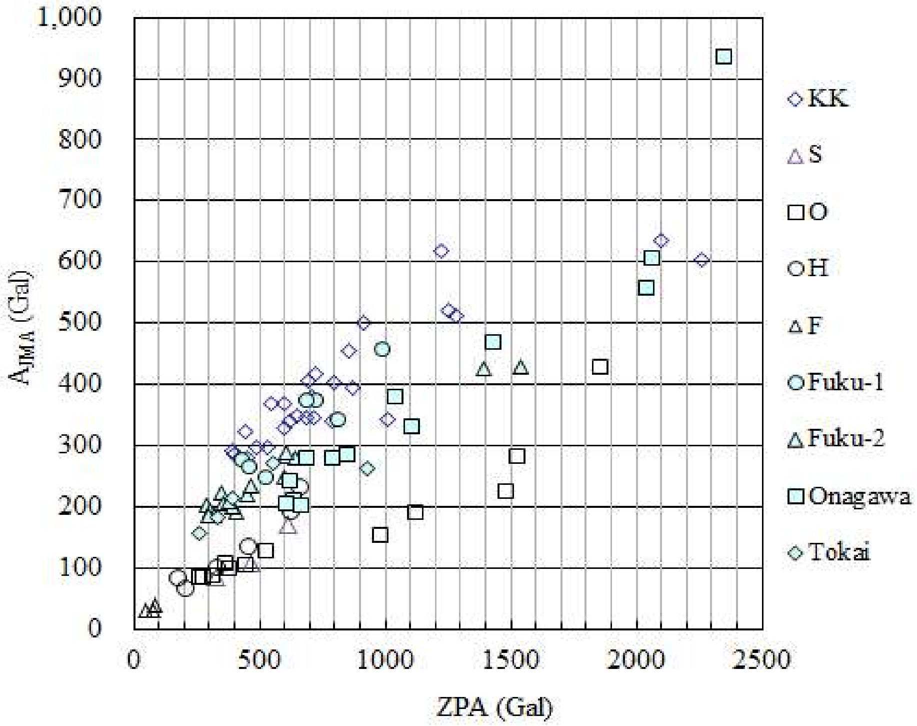
4.4.1.2. DIP calculation results

The DIP calculation results for the five acceleration time-histories shown as (a) to (e) in Fig. 38 are sgiven in Table 24. The peak acceleration observed at Onagawa nuclear power station is higher than the one observed at Fukushima Dai-ichi. However, the effective earthquake load seems to be lower because the duration of that peak acceleration is short, that means ‘spike’. Figures 40 a) through Fig. 40 d) show the plots of calculated DIPs of the six earthquakes given in Section 4.4.1.1.

TABLE 24. ANALYSIS IN JAPAN FOR NUCLEAR PLANTS - CALCULATED DIPS FOR THE TYPICAL EARTHQUAKE TIME-HISTORIES SHOWN IN FIG. 38

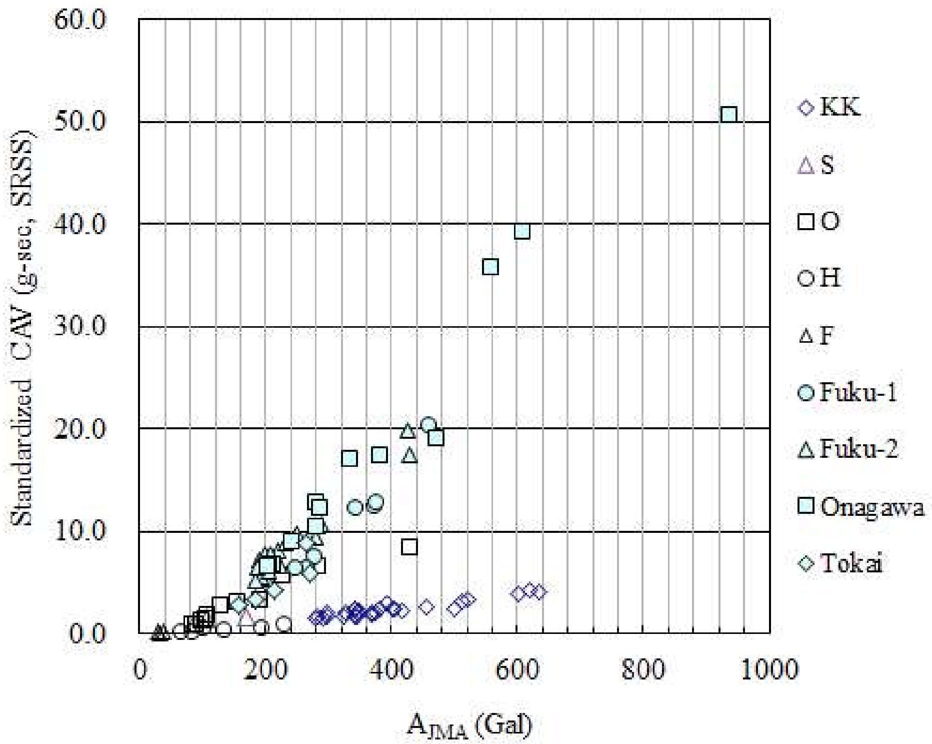
Earthquake Motion	Peak Acceleration (Resultant, Gal)	A_{JMA} (Gal)	Standardized CAV (SRSS, g-sec)	Average Acceleration Response Spectrum (SRSS, Gal, h=0.05)	
				2-10 Hz	10-20 Hz
(a) Onagawa Unit 1	637	225	6.9	925	1189
(b) Fukushima Daiichi Unit 6	460	266	6.4	875	637
(c) Fukushima Daini Unit 1	290	202	5.4	821	617
(d) Tokai Daini	262	157	2.7	735	548
(e) Kashiwazaki Kariwa Unit 1	685	346	1.7	1331	1008

The correlation between standardized CAV and the JMA instrumental intensity I_{JMA} is not well defined because of the saturation of the value of I_{JMA} (See Section 4.1.1). However, when A_{JMA} is used as DIP instead of I_{JMA}, it can be seen that the relation between the standardized CAV and A_{JMA} maintains linearity. Moreover, the slope of the regression lines shown in Fig. 41 vary widely depending on the earthquake wave motions. This observation means that the standardized CAV and the A_{JMA} are indicative of different characteristics of earthquake motions and both of them need to be evaluated as DIP in relation with damage. In other words, it seems unreasonable to replace one with the other for assessing the earthquake motion characteristics in terms of damage potential.



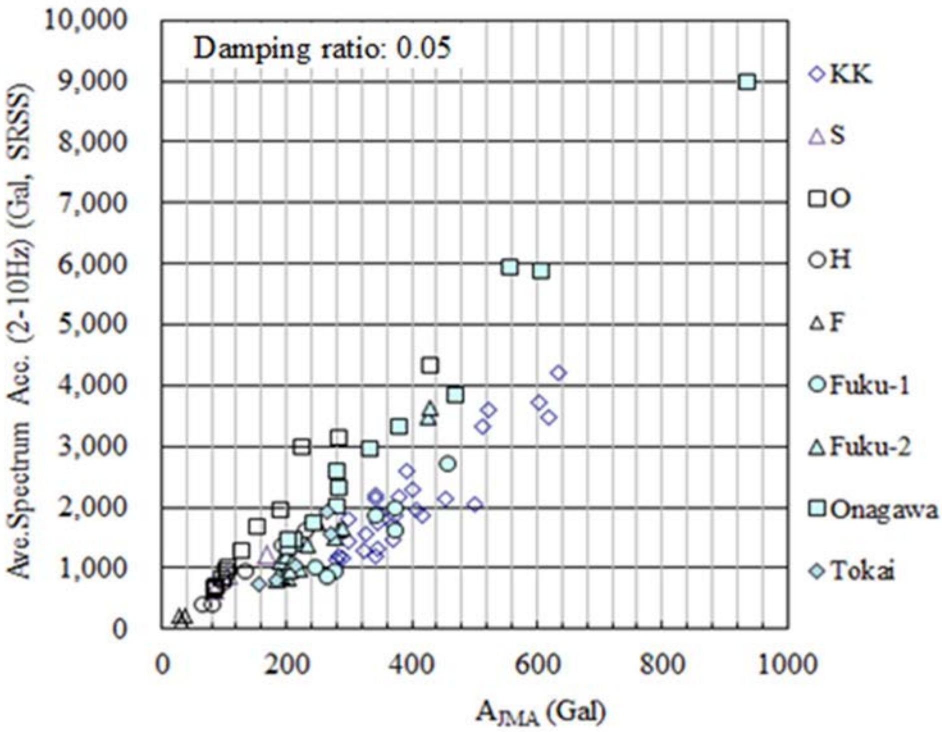
a) A_{JMA} vs. Peak Acceleration

FIG. 40(a). Analysis in Japan for Nuclear Plants - Calculated DIPs.



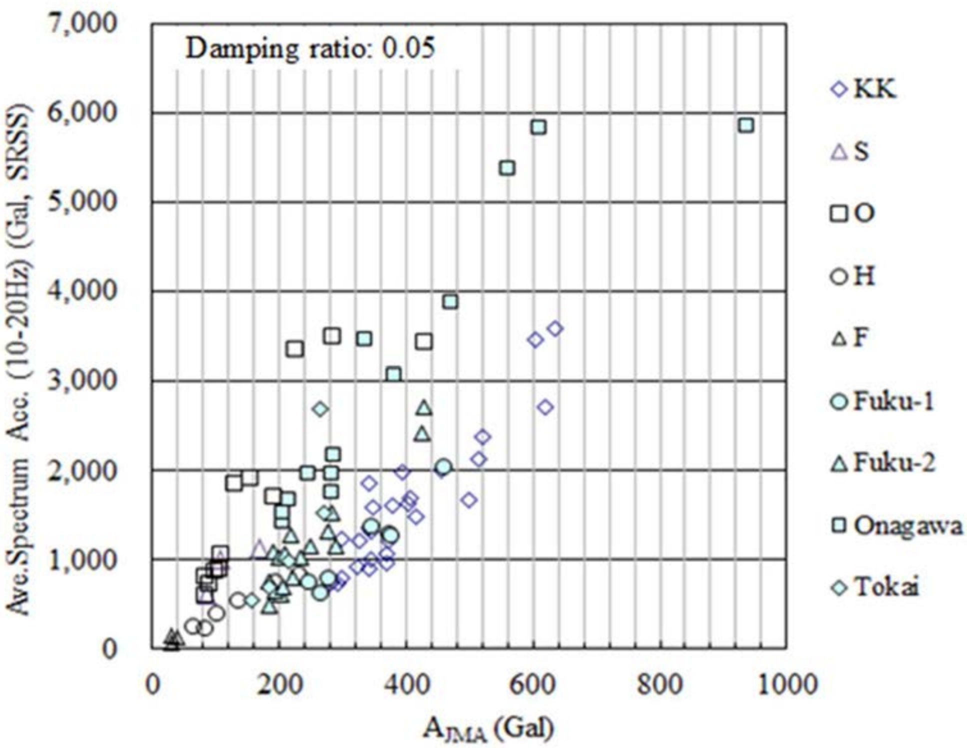
b) Standardized CAV vs. A_{JMA}

FIG. 40(b). Analysis in Japan for Nuclear Plants - Calculated DIPs.



c) 2-10 Hz average spectrum acceleration vs. A_{JMA}

FIG. 40(c). Analysis in Japan for Nuclear Plants - Calculated DIPs.



d) 10-20 Hz average spectrum acceleration vs. A_{JMA}

FIG. 40(d). Analysis in Japan for Nuclear Plants - Calculated DIPs.

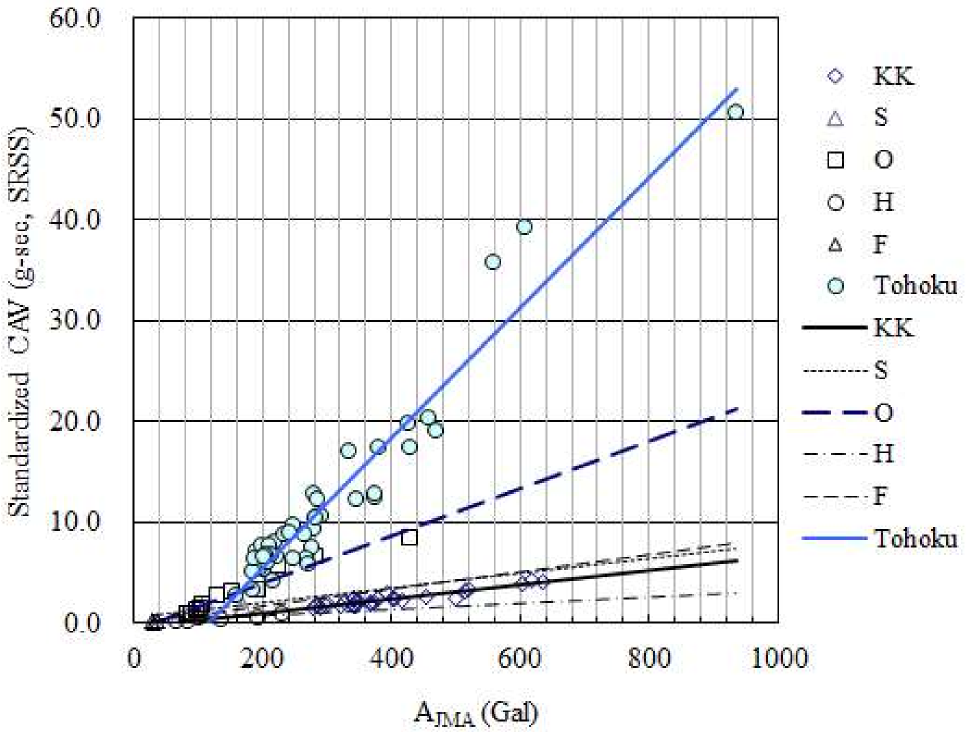


FIG. 41. Analysis in Japan for Nuclear Plants – Correlations between Standardized CAV and A_{JMA}.

4.4.1.3. Calculated DIPs and observed damage

Although nuclear power plants in Japan have experienced strong earthquake motions exceeding the ASTS trigger level five times, as shown in Section 4.4.1.1, there have not been any significant damage to the safety-related SSCs. However, damage data has accumulated for the conventional and non-safety-related SSCs. The data can indicate possible threshold DIP values even for the nuclear power plant SSCs.

Particularly, the Tokyo Electric Power Company (TEPCO) carried out a very detailed evaluation of the damage experienced in the 2007 Niigata-ken Chuetsu-Oki earthquake and the investigation results are useful for the present DIP study. Table 25 lists typical damage and the calculated JMA instrumental seismic intensities. At Kashiwazaki-Kariwa plant, the accelerometers of the SDAS were set at 27 points inside buildings. Typical damage observed at those points is shown in Table VII-1 in Annex VII, and the calculated DIPs are given in Table VII-2. It needs to be noted that the data includes both significant and minor damage, as shown in Table VII-1.

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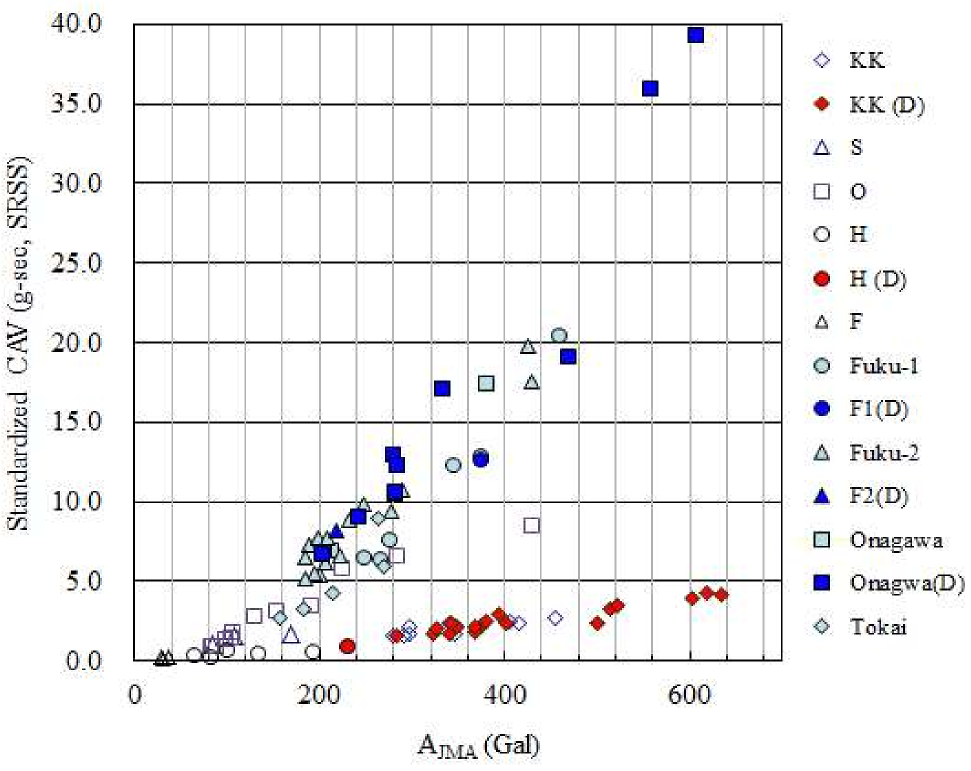
TABLE 25. TYPICAL DAMAGE OBSERVED AT THE KASHIWAZAKI-KARIWA NPS IN NIIGATA-KEN CHUETSU-OKI EARTHQUAKE

Instrumental Seismic Intensity	JMA Seismic Intensity Scale	Classification of Importance in Seismic Design		
		S (As & A)	B	C
6.5	7	None	Deformation of main turbine bearing support key	Buckling of water storage tank
			Bearing metal contact of main turbine (out of service)	Failure of miscellaneous yard facilities
				Bearing metal contact of main generator (out of service)
6.4	6 Upper	None	Deformation of main turbine bearing peripheral equipment	Loosed bolt at main generator alignment key
			Uplift of turbine pedestal gap cover	Deformation of deck of instrument storage box
6.3		None	None	Elephant foot buckling of water storage tank
				Failure of miscellaneous yard facilities
				Main turbine casing cover came off
6.2		None	None	None
6.1		None	None	None
6.0		None	Contact of main turbine blades	Contact of main generator rotor and peripheral equipment
			Rupture of outlet pipe boot at condensate filter	Leakage from connection between pipe and pump/valve
			Concrete crack at base of large equipment	Movement of crane/hoist (damage of stopper, limit switch)
			Turbine building blowout panel came off	Penetration cover plate of cable tray came off
				Deformation of cable duct cover
5.9	6 Lower	None	None	None
5.8		None	None	None
5.2	5 Upper	None	None	None
4.7	5 Lower	None	None	None

In the 2011 off-the-Pacific-coast-of-Tohoku earthquake, some types of damage different from damage found in the other five earthquakes were observed. For example, a fire of high voltage power panel (Seismic Class C) at the Onagawa plant Unit 1 turbine building; a rupture of a snubber rod for the turbine moisture separator (Seismic Class B) at the Tokai plant turbine building and so on. The common characteristic of these damaged components is that they were hanged, that means that they had no rigid anchoring. The elephant-foot buckling of the low pressure water storage tanks (seismic class C) cylindrical shell wall installed in yard, and slight movements of the main steam turbine-generator shaft are commonly observed in both the 2011 off-the-Pacific-coast-of-Tohoku earthquake and the 2007 Niigata-ken Chuetsu-Oki earthquake.

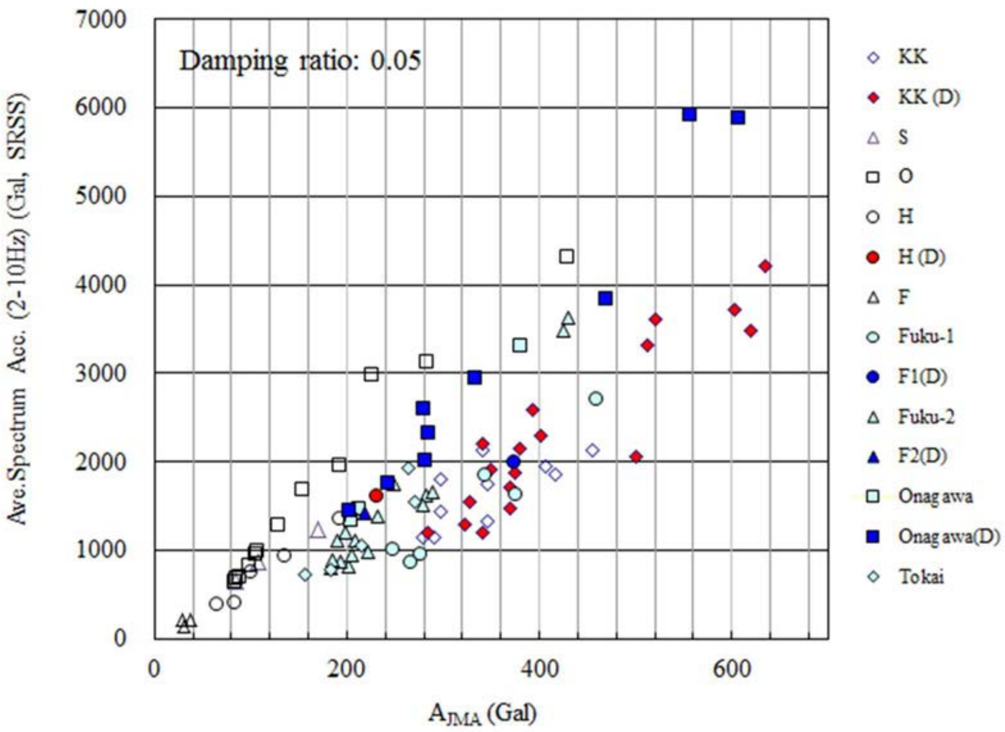
At the Onagawa plant, tri-directional accelerometers which are installed on higher positions in the reactor buildings, such as roof tops and operating floors, recorded very large acceleration time-histories during the 2011 off-the-Pacific-coast-of-Tohoku earthquake. The calculated DIPs for these time-histories are shown in the Table VII-4 and the observed damages at each point are shown in the Table VII-3 in Annex VII. It needs to be also noted that these damage data include both significant and minor damages.

Figures 42(a) through Fig. 42(c) show the distribution of damage at the points in Fig. 40(b) through 40(d), respectively. Damage instances are indicated with solid markers



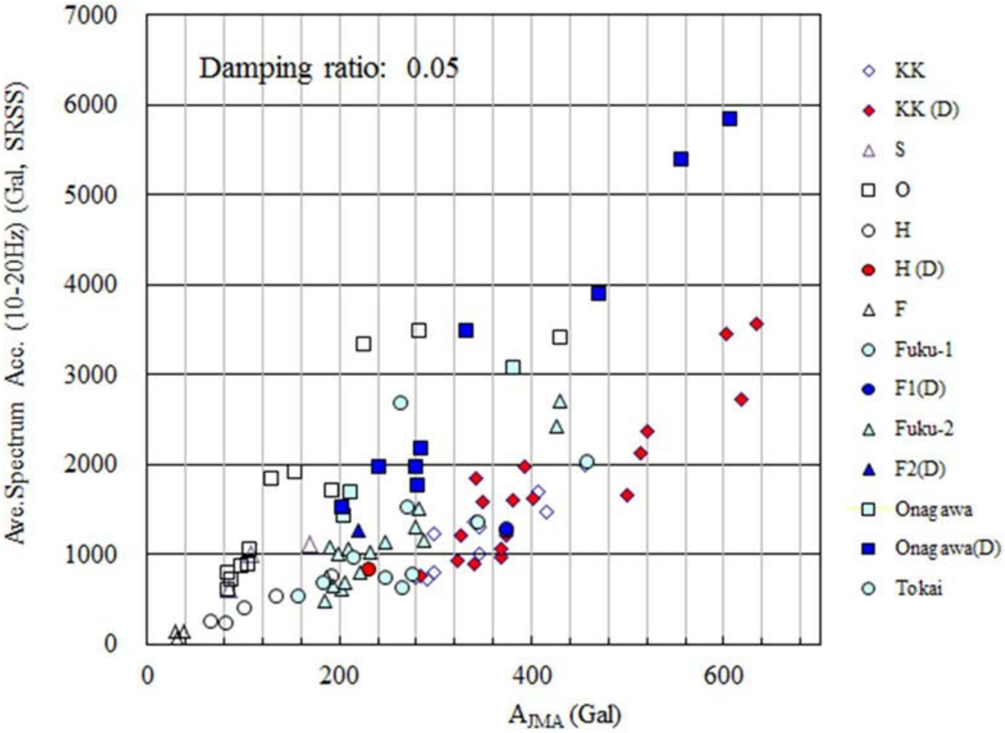
a) Standardized CAV vs. A_{JMA}

FIG. 42(a). Analysis in Japan for Nuclear Plants –Damage distribution (solid markers indicate damage instances).



b) Average spectrum acceleration (2-10 Hz) vs. A_{JMA}

FIG. 42(b). Analysis in Japan for Nuclear Plants –Damage distribution (solid markers indicate damage instances).



c) Average spectrum acceleration (10-20 Hz) vs. A_{JMA}

FIG. 42(c). Analysis in Japan for Nuclear Plants –Damage distribution (solid markers indicate damage instances).

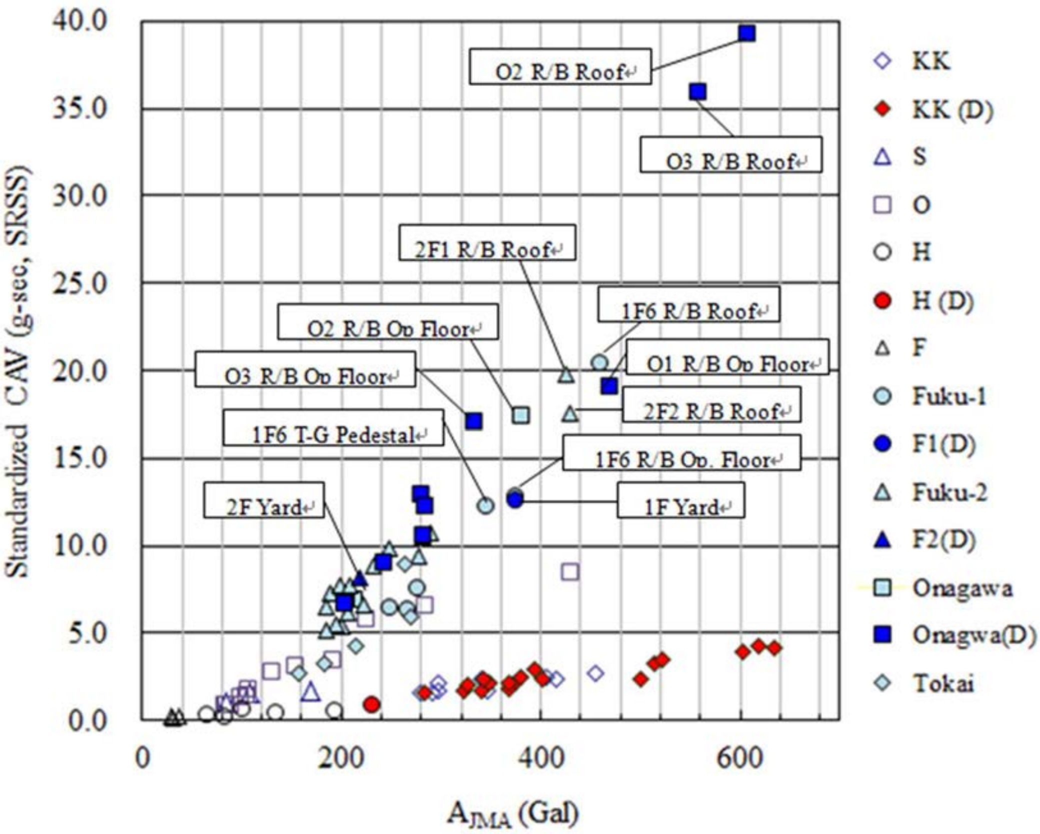


FIG. 43. Analysis in Japan for Nuclear Plants - Typical observation points for the data in the 2011 off-the-Pacific-coast-of-Tohoku earthquake.

Fig. 43 reproduces Fig. 42(a) (Standardized CAV vs. A_{JMA}), adding labels to indicate the position of observation points at the Fukushima Daiichi, Fukushima Daini and Onagawa plants. Note that the A_{JMA} values at these points, especially inside structures, are relatively large.

4.4.1.4. Findings

Finding 1: Comparison between nuclear power plant and industrial facilities

Fig. 44 combines the calculated DIPs for nuclear power plants (Fig. 42(a)) and for industrial facilities, especially thermal power plants and substations (Fig. 34). Fig. 44 shows that there is a clear distinction between them regarding seismic capacity.

Fig. 44 indicates that the threshold A_{JMA} value for any damage, including a minor damage, seems to be around 200 Gal for the nuclear power plants. However, the threshold for significant damage is around 80 Gal for the conventional thermal power plants or substations. That means that seismic capacity of SSCs at nuclear power plants, even though designed as seismic class B or C, seems to be higher than conventional SSCs by one (1.0) seismic intensity scale value in the JMA scale.

Finding 2: Failure modes of main steam turbines and generators

One of the remarkable failure modes observed at both Kashiwazaki-Kariwa plant and Hamaoka plant Unit No. 5 is the slight movement of the main steam turbine shaft in its longitudinal direction, and it was also observed in 2011 off-the-Pacific-coast-of-Tohoku earthquake. This failure mode is considered to be caused by a large inertia force due to the earthquake motion in that direction. Table 26 shows the summary of the damage observed at the Kashiwazaki-Kariwa and the Hamaoka plants regarding this failure mode.

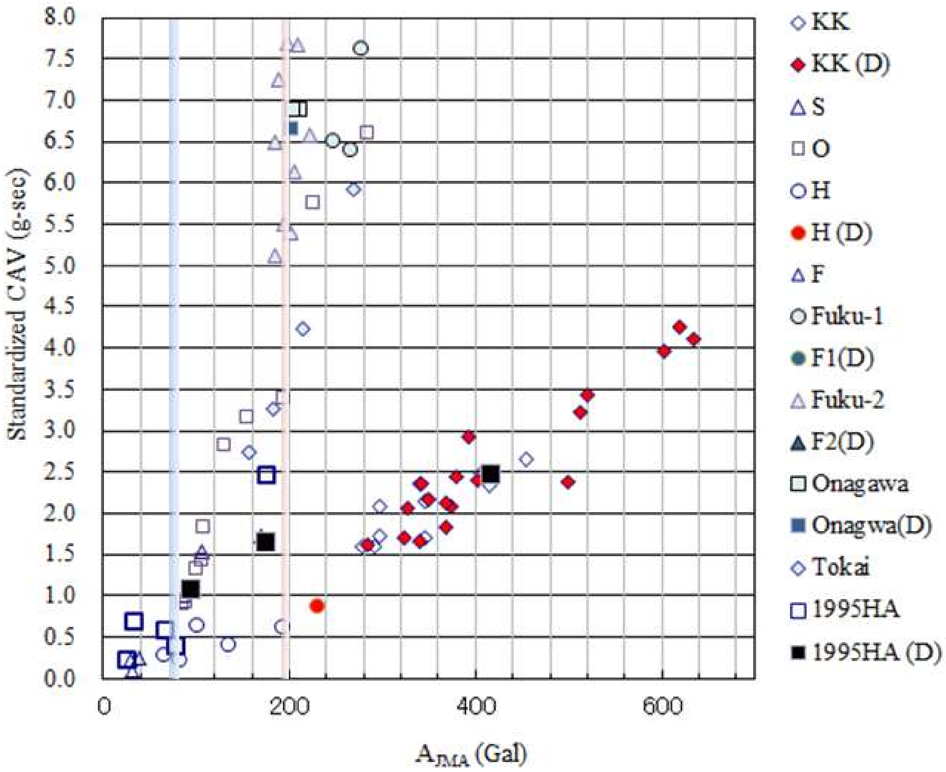


FIG. 44. Comparisons between nuclear power plant and conventional power plant SSCs.

TABLE 26. DAMAGE OBSERVED ON MAIN STEAM TURBINE

Nuclear power plant (typical)	Direction of turbine shaft	Damage
Kashiwazaki-Kariwa Unit No.5	North-South	–Middle standard bearing box (with thrust bearing of turbine rotor): Deformation of the keys –Slight contact between rotating parts and surrounding miscellaneous features
Kashiwazaki-Kariwa Unit No.7	North-South	–Middle standard bearing box (with thrust bearing of turbine rotor): Deformation of the keys –Slight contact between rotating parts and surrounding miscellaneous features
Hamaoka Unit No.5	East -West	–Middle standard bearing box (with thrust bearing of turbine rotor): Slight lift up, and deformation of the fixing bolts and keys –Low pressure turbine inner casing: Deformation of the thrust key –Contact between rotating parts and surrounding miscellaneous features

One of the candidate DIPs to evaluate this failure mode can be A_{JMA} because it captures deformation or break of supporting structures due to momentary earthquake load. However, A_{JMA} corresponds to the combined value in three directions, as shown in Section 3.3.2.2. At the Hamaoka nuclear station, the earthquake load in the turbine longitudinal direction was far larger than in the other two directions. On the contrary, in the Kashiwazaki-Kariwa plant, the load in other direction than the turbine longitudinal direction (north-south) was larger than or equivalent to the turbine longitudinal direction in the Hamaoka plant. A failure mode with strong directivity, such as the steam turbine rotary shaft displacement, can be correlated with the DIP calculated in one direction. Hence, the A_{JMA} in one direction is calculated in this case, even though A_{JMA} is originally defined as a value combining three directions.

Eleven nuclear power plants are evaluated to calculate the DIPs at the top of turbine–generator pedestal in the turbine longitudinal direction. Those nuclear power plants are given as follows:

– Kashiwazaki-Kariwa nuclear power plant, in 2007 Niigata-ken Chuetsu-Oki earthquake (6 units);

– Hamaoka nuclear power plant, in 2009 Suruga Wan earthquake (3 units);

– Fukushima Daiichi nuclear power plant, in 2010 Off Fukushima Prefecture earthquake (1 unit);

– Fukushima Daini nuclear power plant, in 2010 Off Fukushima Prefecture earthquake (1 unit);

The calculated DIPs are shown in Fig. 45, with solid markers when any damage of the steam turbines or generators was reported. It is recognized from this figure that the A_{JMA} value for the Hamaoka plant Unit No. 5 are close to some of the values in the Kashiwazaki-Kariwa plant.

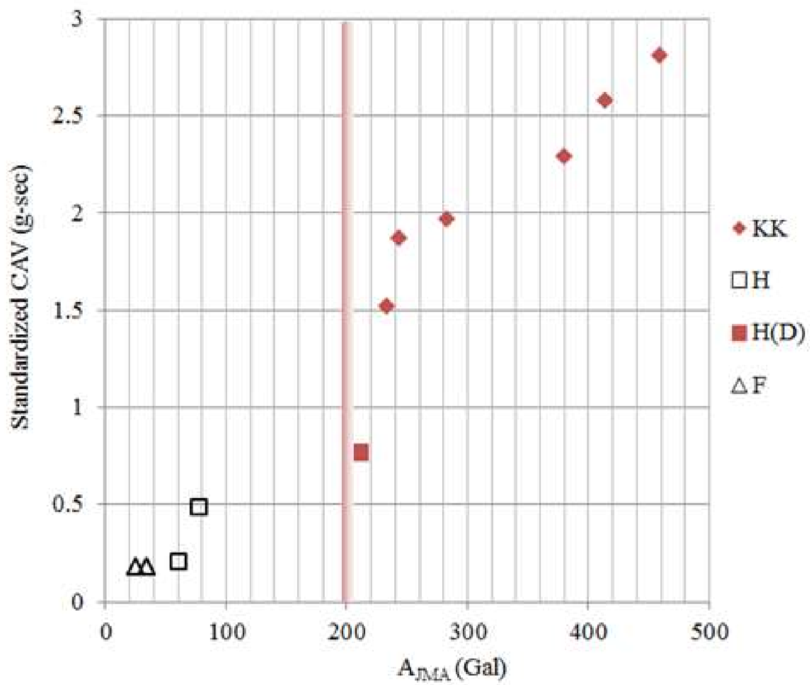


FIG. 45. Calculated DIPs at the top of turbine-generator pedestals.

4.4.2. Analysis in the United States of America

The nuclear power plants in the United States that have experienced potentially damaging earthquakes were the Humbolt Bay nuclear power plant in 1975, the Perry nuclear power plant in 1987 and the North Anna nuclear power plant in 2011. The Humbolt Bay Plant was shut down during the 1975 earthquake and it is no longer in operation. The seismic design procedures used in the design of the Humbolt Bay nuclear power plant are dated well before current seismic design requirements for nuclear power plants were formulated. Hence, they are generally not applicable to currently operating nuclear power plants.

The Virginia Earthquake that occurred at 13:51, August 23, 2011 (Eastern Daylight Time) affected the North Anna Nuclear Generating Station (Westinghouse pressurized water reactor, Unit 1, 971 MWe, and Unit 2, 963 MWe, both of which were in operation with 100% output when struck by the earthquake). The magnitude of the earthquake was 5.8 (Richter scale), the depth of the hypocentre was 6 km, the distance from the power plant to the epicentre was 11 miles (17.7 km) west-south-west, and the MMI intensity in the vicinity of the power plant (at the city of Mineral) was grade VII.

Acceleration time-history of the observed earthquake motion at the foundation of Unit 1 reactor containment building and its acceleration response spectrum (damping ratio 0.05) are shown in Fig. 46 and Fig. 47, respectively. The maximum acceleration in the north-south direction was 0.23 g, which exceeded the design basis earthquake maximum acceleration of 0.12 g [53–54].

The two reactors were automatically shut down and the off-site power for the entire power plant was lost upon the occurrence of the earthquake. Commercial operation was restarted successfully around three months later. The development of the situation after the earthquake is shown below.

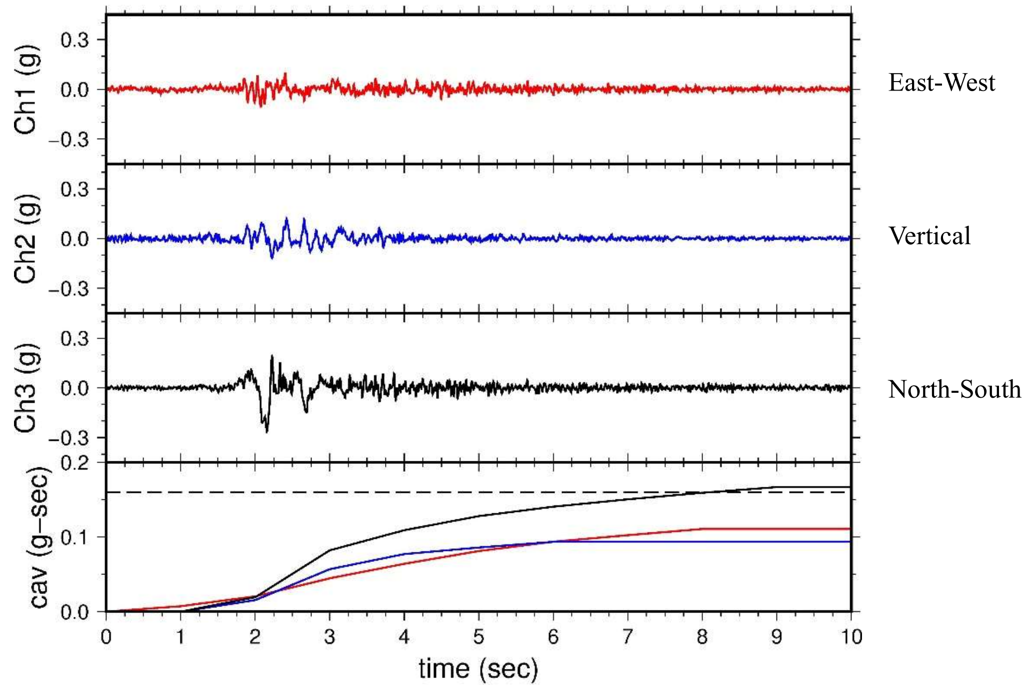


FIG. 46. Virginia 2011 Earthquake - Acceleration time-histories and corresponding CAV in North Anna nuclear power plant, at foundation of Unit 1 reactor containment building [54].

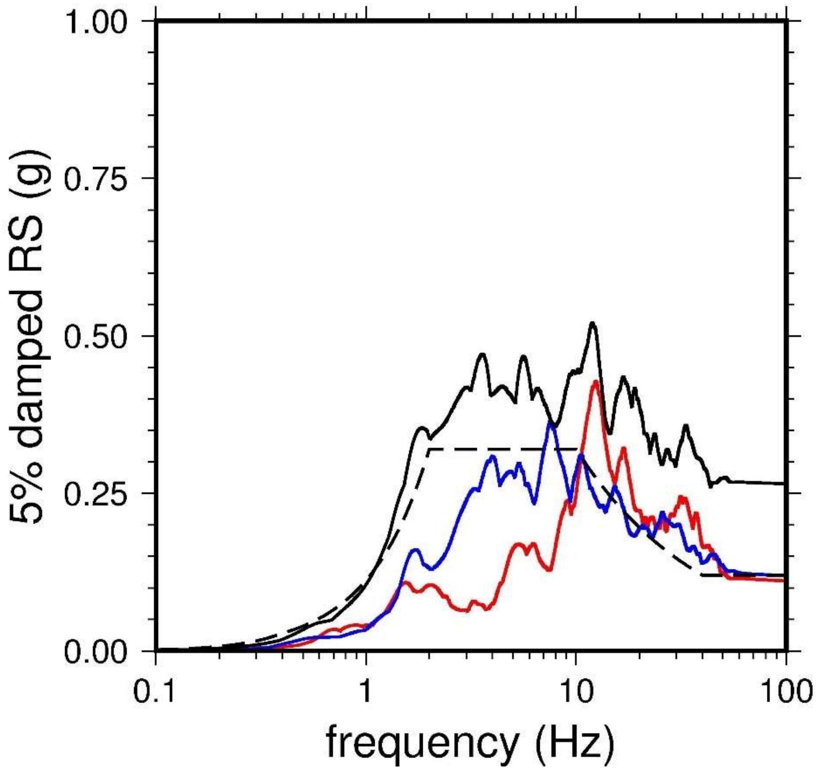


FIG. 47. Virginia 2011 Earthquake – Acceleration response spectra in North Anna nuclear power plant, at foundation of Unit 1 reactor containment building [53]. (Red: East-West direction; Blue: Vertical direction; Black: North-South direction)

August 23

13:51:00	Occurrence of earthquake
13:51:11	Automatic reactor shutdown (upon high flux rate reactor trip)
13:51:12	Loss of off-site power (upon actuation of sudden pressure relay of transformer)
13:51:20	Automatic start-up of emergency diesel generator
22:58	Completion of off-site power restoration

August 24

21:26	Unit 1: Operation Mode 5 (cold shutdown state)

August 26

16:23	US NRC notified possibility of 'non-analytic event'
20:38	Unit 2: Operation Mode 5 (cold shutdown state)

The North Anna nuclear generating station was not equipped with an ASTS using seismic motions as a trigger. Both units were automatically shut down upon a high flux rate reactor trip signal. According to the post-earthquake investigation, neutron flux was fluctuated by the seismic motions (Fig. 48) and a scram signal was triggered when the flux rate reached the lower limit (5%) [55].

The loss of off-site power that occurred immediately after the automatic reactor shutdown was triggered by a trip of the sudden pressure relay in the Unit 1 and 2's generator step-up transformer and standby transformers. Since the acceleration response spectrum exceeded the safe shutdown earthquake in the frequency range of 2 to 10 Hz, both units were put to cold shutdown on August 26, three days after the occurrence of the earthquake, and seismic damage inspections were performed.

4.4.2.1. DIP evaluation results

The results of calculating standardized CAV at the foundation of Unit 1 reactor containment building are shown in Table 27 and Fig. 49. There were no seismometers installed on the ground free surface at North Anna site. Hence, the computed CAV was tentatively compared with the operating basis earthquake exceedance threshold value of 0.16 g-sec, and it was found that the CAV slightly exceeded the threshold only in the north-south direction (Fig. 49). However, the standardized CAV value of the design basis earthquake and the review level earthquake in the Individual Plant Examination of External Events project were not exceeded [54].

4.4.2.2. Seismic damage inspection results

The EPRI seismic damage scale (see Section 3.2.4.2) was determined to be '0' by the operator walkdown performed immediately after the earthquake. However, since the observed earthquake motions exceeded the design basis, an expanded inspection was started for the in-depth inspection of electric and mechanical equipment, including containment internals, buildings, structures and buried pipes.

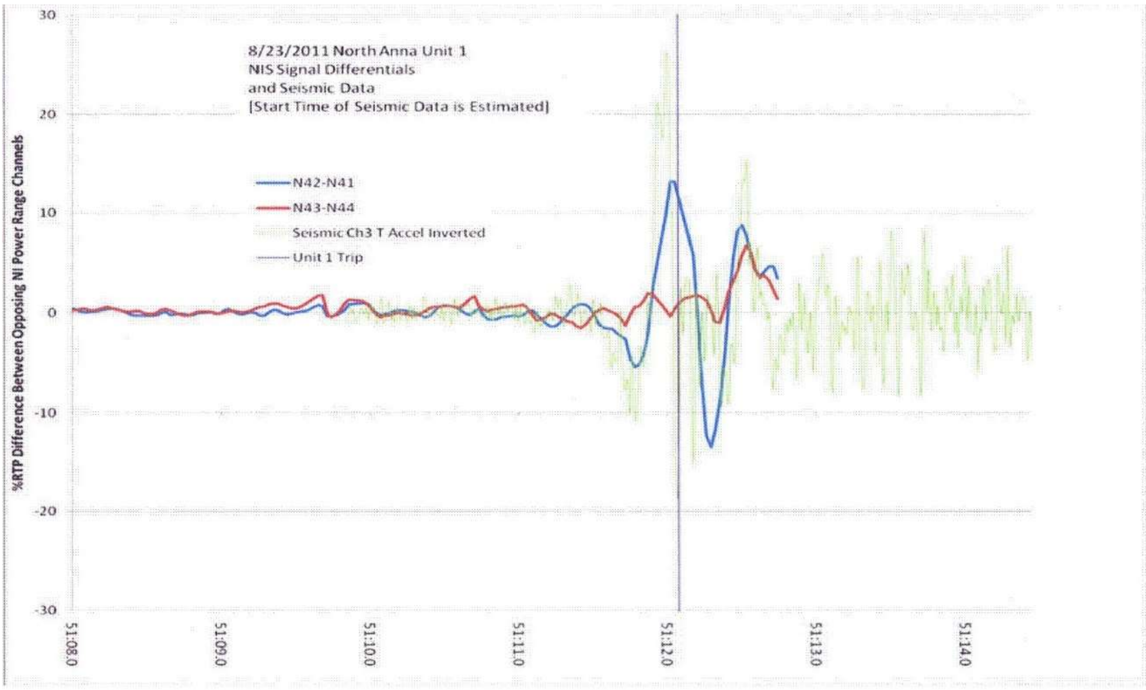


FIG. 48. Virginia 2011 Earthquake –North Anna nuclear power plant Unit 1 reactor - Nuclear instrumentation system signals superimposed to earthquake acceleration signal [55].

TABLE 27. NORTH ANNA NUCLEAR POWER PLANT - STADARDIZED CAV COMPARISONS [54]

Seismic Case	CAV North-South direction (g-sec)	CAV East-West direction (g-sec)	CAV Vertical direction (g-sec)
August 23, 2011 seismic event (Data from containment basemat)	0.172	0.125	0.110
Design Base Earthquake (Rock-founded; synthetic time-history used for containment structure)	0.588	0.580	0.400
IPEEE Review Earthquake (Rock-founded; synthetic time-history used for containment structure)	1.230	1.312	0.875
Operating Basis Earthquake (OBE) exceedance criterion is standardized CAV > 0.16 g-sec (EPRI TR-100082 and US-NRC RG 1.166)			

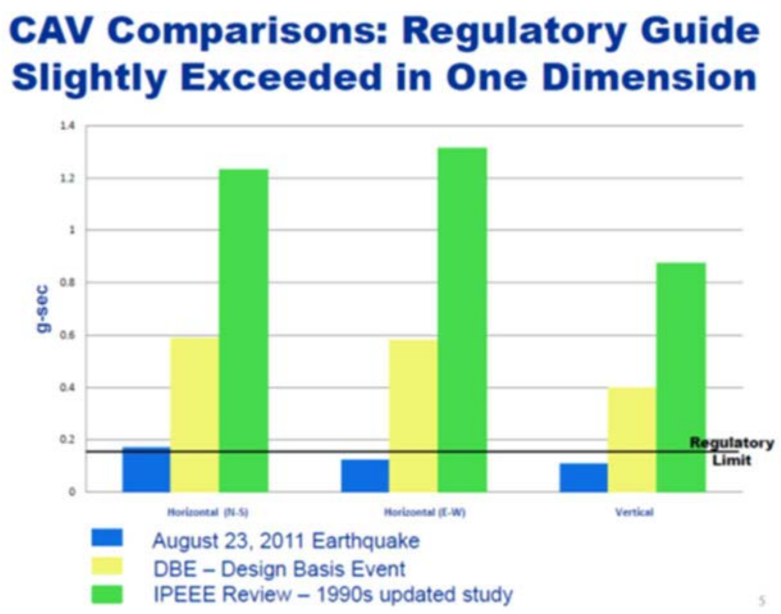


FIG. 49. Virginia 2011 Earthquake –North Anna nuclear power plant Unit 1 -Standardized CAV comparisons [54].

As a result of the expanded inspection, the following damage (including malfunctions) was identified:

– Malfunction of sudden pressure relay of transformer (although it occurred immediately after the earthquake, no anomalies were found in the equipment);

– Cracks in concrete wall and peeled decoration mortar of reactor containment building;

– Cracks in concrete walls of non-safety-related structures;

– Displacement of spent fuel storage casks;

– Damage to foundation and roof vent cover fixtures of spent fuel storage facility (horizontal dry cask storage);

– Slight spalling of concrete from condensate filter tank support (turbine building);

– Bushing leaks from 500 kV generator step-up transformers (8 units, including reserves) (taken back to the factory for repair);

– Damage to lower bellows support arm of Unit 2 500 kV generator current transformer breaker (same as above);

– Cracks in insulators of switch yard system.

As for the nuclear fuel that triggered the automatic reactor shutdown, fuels assemblies, rod guides, and others of Unit 2 were subjected to visual inspection and insertability test, but no anomalies were found.

All the damage found during the inspection can be categorized as minor damage to non-safety-related structures. The standardized CAV exceeded 0.16 g-sec only by 10% in the north-south direction alone. Finally, it was concluded that safety-related facilities had no hidden damage. The reactors were restarted, and they are currently in continuing stable operation, after having been subjected to surveillance tests.

This case is valuable as the first case in which a DIP, the standardized CAV, was used for seismic damage evaluation. In addition, this case offered many lessons and information, including automatic reactor shutdown due to the earthquake and the pitfalls in the seismic instrumentation system (see Section 2.4.2.2).

4.5. ANALYSIS BASED ON THE VIBRATION TEST DATA

The relationship between damage modes and DIPs, which is dependent on the structural characteristics of the particular equipment item, can be understood through a vibration test verifying the seismic capacity of the specific item.

4.5.1. Ultimate strength piping test by NUPEC/JNES

Among the vibration tests done by Nuclear Power Engineering Corporation (NUPEC) / Japan Nuclear Energy Safety Organization (JNES), using the large-scale vibration table on ultimate strength piping test is very significant for the investigation of DIPs.

NUPEC/JNES completed a series of seismic reliability tests on the ultimate strength of piping systems between 1998 and 2003. At the final stage of the experimental program, an ultimate strength test of a carbon steel pipe (nominal outer diameter of 216.3 mm, thickness of 8.2 mm), with a three-dimensional configuration was conducted [56]. The key parameters of the test specimen are shown in Table 28.

Fig. 50 shows the specimen set on the large shaking table. NUREG/CR 6983 describes the test specimen as follows [57]:

> "The ultimate strength test used a similar but modified piping test specimen. An additional mass was added, and a support was removed. The intent of these modifications was to induce failure in the system. The pipe was internally pressurized, and the tests

were performed at room temperature. Horizontal seismic input motion corresponding to a maximum elastically-calculated stress level of 24 S_m was applied. The test was repeated until failure occurred. During the fifth test run, a through-wall crack developed in an elbow. An examination confirmed that the failure was the result of fatigue ratcheting".

TABLE 28. KEY PARAMETERS OF THE ULTIMATE STRENGTH PIPING TEST

Pipe outer diameter	216.3 mm (200A)
Pipe wall thickness	8.2 mm (Sch40)
Material	STS410
Internal pressure	Membrane stress: Sm equivalent
Configuration	Three dimensional (see Fig. 50)
Natural frequency (1st mode)	from 3.8Hz to 3.6Hz
Damping ratio	from 0.9% to 4.5%
Failure mode	Through-wall crack in an elbow
	(Low cycle fatigue with ratcheting)

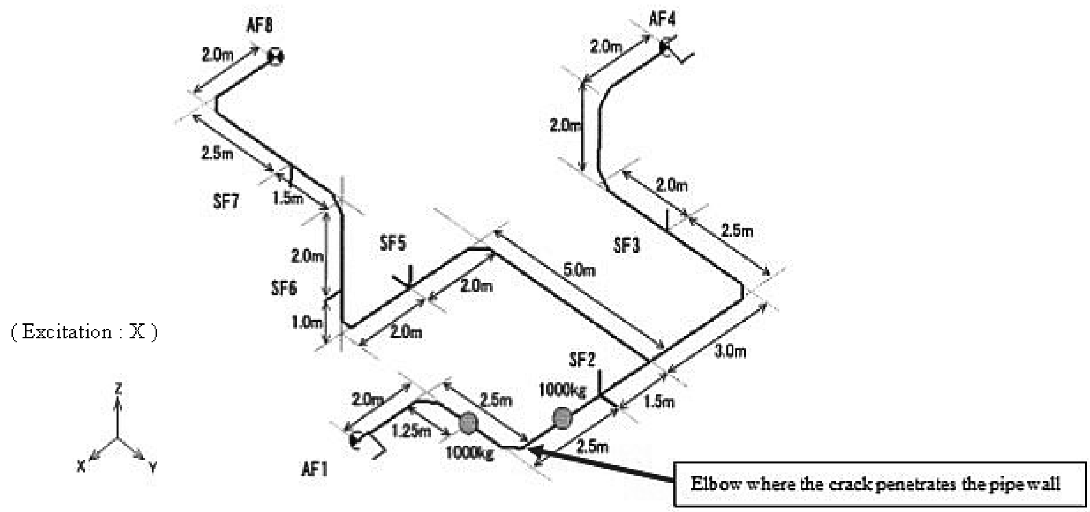


FIG. 50. Three dimensional configuration of the tested piping [57].

NUREG/CR 6983 also describes the test condition as [57]:

> "The ultimate strength test was designed to fail the pipe. As indicated in Table 2-1, this test series included preliminary low-level sine sweep tests (US1) to determine the frequencies and modal damping values as shown in Table 2-3, and ultimate strength seismic tests (US2). The seismic input motion was designed to induce a maximum stress level equal to 8 times the Code limit or 24 S_m. In order to achieve this high stress level, the seismic waves were adjusted so that the dominant input motion frequency was close to the fundamental piping system frequency (on-resonance). In these tests, the seismic table motion was applied only in the horizontal direction. The piping system was internally pressurized to a design stress intensity of S_m and the tests were conducted at room temperature. The seismic input motion was repeated until failure occurred. During the fifth test run, a longitudinal through-wall crack developed in elbow 2. A photograph of the failure is shown in Fig. 2-29. A close-up of the longitudinal crack in the elbow is shown in Fig. 2-30. An examination confirmed that the failure was the result of fatigue ratcheting."

Because the objective of this test was to identify the failure modes of the piping system, the vibration motion on the shaking table was adjusted to be in resonance with the piping, and it consisted of an artificial wave simulated based on the design basis of an actual nuclear power station. The time-history of the acceleration observed on the shaking table is shown in Fig. 51 and the acceleration response spectrum is shown in Fig. 52.

The piping system was excited repeatedly with the simulated earthquake motion mentioned above, and finally, during the fifth repetition a crack penetrated the pipe wall at its elbow. The failure mode was judged to be fatigue ratcheting, so the duration time or the cycles of an earthquake excitation, in addition to acceleration, is one of the dominating factors to be investigated. Table 29 shows the calculation results of the typical DIPs in consideration of the repetition of excitation.

The failure mode observed, 'fatigue ratcheting', is categorized as 'cumulative fatigue damage' and addressed in Section 3.1.2. Table 29 shows the possibility of through-wall cracks in a typical piping system at the standardized CAV value larger than around 90 g-sec.

On the other hand, very large standardized CAV values were observed at the Fukushima Daiichi unit 6 main buildings during the 2011 off-the-Pacific-coast-of-Tohoku earthquake, as shown in Table 30 and Fig. 53. Fig. 53 shows a longitudinal cross-section schematic view of the reactor and turbine building of the Fukushima Dai-ichi Unit 6, marking locations where earthquake motions were recorded during this earthquake.

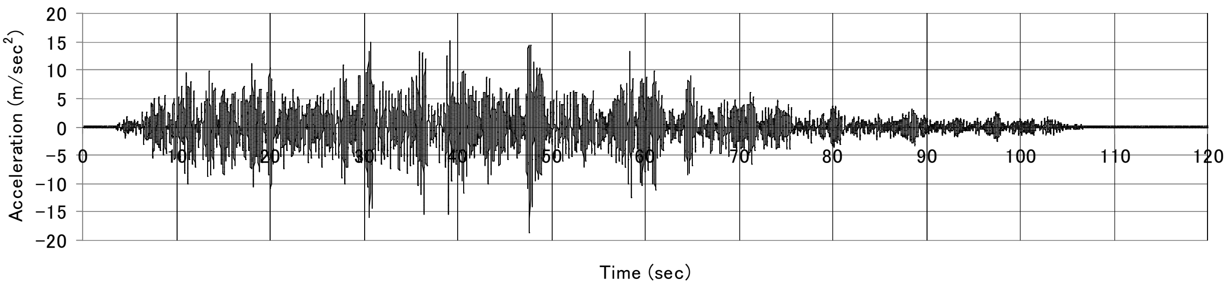


FIG. 51. Acceleration time-history of the input motion during the test (supplied by JNES).

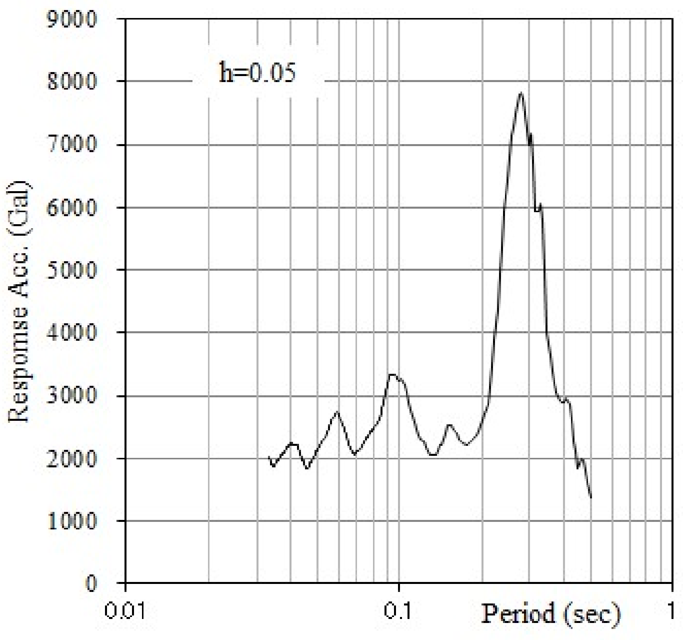


FIG. 52. Acceleration response spectrum corresponding to the input motion during the test.

TABLE 29. ACCUMULATION OF EXCITATION AND CALCULATED DIPS

Repetition	Accrued Duration Time (sec)	ZPA (Gal)	Standardized CAV (g-sec)	Calculated JMA Instrumental Seismic Intensity	Notes
1	120	1877	23.2	6.4	
2	240	1877	46.5	6.5	
3	360	1877	69.7	6.5	
4	480	1877	92.9	6.5	
5	600	1877	116.2	6.5	Crack penetrates the pipe wall

If the threshold value of 90 g-sec is divided by the observed standardized CAV values (Table 30 right-most column), an indication of seismic margin for a typical piping system during that earthquake can be obtained. In this example, it can be concluded that the seismic margin is very large for a piping system, even if it was installed, hypothetically, on the roof of the reactor building.

The seismic margin can be also estimated with a DIP other than standardized CAV. It is the 'Energy Equivalent Velocity', which is related with the elastic energy spectrum. However, the physical meaning of the Energy Equivalent Velocity is somewhat clearer, and the total energy input has been found experimentally to have a good correlation with cumulative fatigue damage by Minagawa and others [58].

TABLE 30. COMPARISONS BETWEEN OBSERVED STANDARDIZED CAV VALUES AND DAMAGE THRESHOLD FOR PIPING (FUKUSHIMA DAIICHI UNIT 6 NUCLEAR POWER PLANT IN THE 2011-OFF-THE- PACIFIC-COAST-OF-TOHOKU EARTHQUAKE)

Building	Floor	Standardized CAV (g-sec): A	90 / A
Reactor Building	Roof	20.4	4.4
	Operating Floor	12.8	7.0
	Second Floor	7.6	11.8
	Basemat	6.4	14.1
Turbine Building	T-G Pedestal Top	12.3	7.3
	Basemat	6.5	13.8

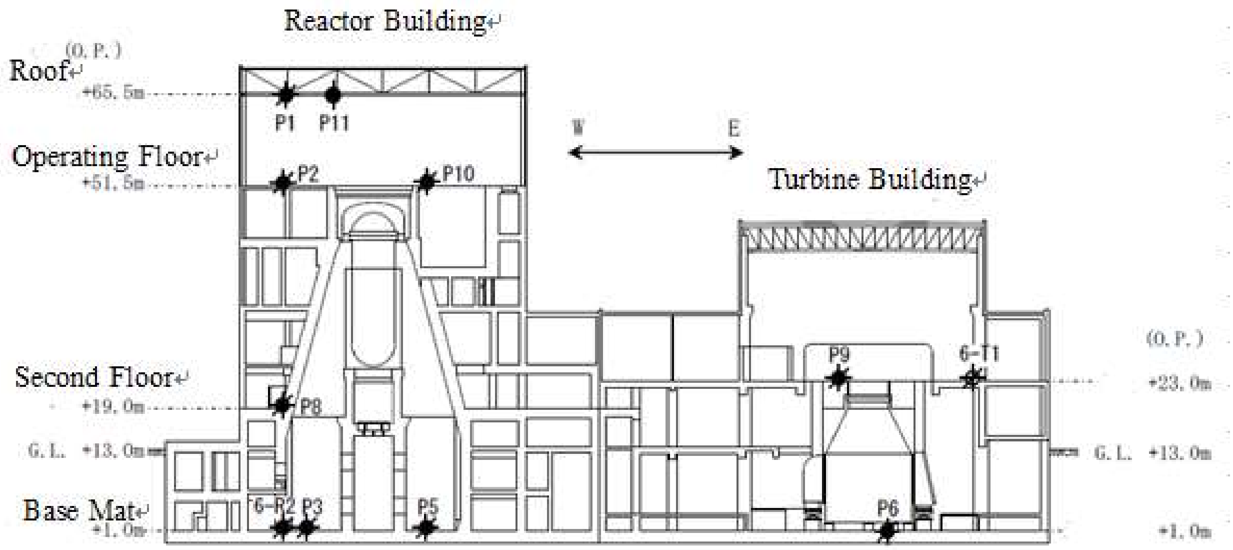


FIG. 53. Observation points of earthquake motions at Fukushima Dai-ichi Unit 6 reactor and turbine building. (Courtesy of Japan Association for Earthquake Engineering)

Total energy input is defined in the following equations;

Equation of motion of single degree of freedom system,

$$m\ddot{x} + c\dot{x} + F(x) = -m\ddot{Z}_H \quad (9)$$

Multiplied by $\dot{x}dt$ to have the work done in dt,

$$m\ddot{x}\dot{x}dt + c\dot{x}^2dt + F(x)\dot{x}dt = -m\ddot{Z}_H\dot{x}dt \quad (10)$$

Energy balance equation becomes,

$$m\int_0^t \ddot{x}\dot{x}\,dt + c\int_0^t \dot{x}^2\,dt + \int_0^t F(x)\dot{x}\,dt = -m\int_0^t \ddot{Z}_H\dot{x}\,dt \quad (11)$$

Where, m is the mass of single degree of freedom system, c is the damping coefficient, $F(x)$ is the restoring force, $\ddot{Z}_H$ is the exciting acceleration, t is time and x is the relative displacement.

Right hand side of the energy balance equation is total energy input to single degree of freedom system, so the total energy input per unit mass is;

$$\frac{E}{m} = -\int_0^t \ddot{Z}_H \dot{x}\, dt \qquad (12)$$

Then, the energy equivalent velocity is defined as;

$$V_E = \sqrt{\frac{2E}{m}} \qquad (13)$$

Note that V_E is dependent on the vibrational characteristics (natural period, damping) of the single degree of freedom system. The calculated elastic energy spectra at the locations inside the reactor building shown in Fig. 53 are given in Fig. 54. In Fig. 54 total energy input per unit mass, E/m, is represented as energy equivalent velocity (a measure of 'cumulative energy') in the vertical axis, and the natural period of the single degree of freedom system is represented in the horizontal axis. The damping ratio is assumed to be 0.05 in these calculations.

Elastic energy spectrum expresses well the vibrational influence of an input earthquake motion to a component. Fig. 54 suggests that this earthquake motion with long duration amplified very much the response of higher floors in the reactor building. The NUPEC/JNES ultimate strength test on the piping system can be applied to the assessment of ruggedness of piping systems installed on those floors using the elastic energy spectrum.

The calculated elastic energy spectrum of the input motion after the fourth excitation to the piping system in the NUPEC/JNES test is shown as dotted line in Fig. 55 and compared with the observed earthquake motion inside Fukushima Dai-ichi Unit 6 reactor building (See Fig. 54).

The seismic margins of piping systems hypothetically located at the different elevations are obtained as shown in Table 31, as ratios of the equivalent velocities provided in Fig. 55 with respect to the NUPEC/JNES test at the natural period of the piping system used in the NUPEC/JNES test. Comparing Table 30 and Table 31, the seismic margin estimated with standardized CAV seems to be more conservative than that obtained with the energy spectrum method.

4.5.2. Verification test for integrity of anchorage

Electric and mechanical equipment has often experienced seismic damage to its foundation anchorage portion. As a result of the 2007 Niigata-ken Chuetsu-Oki earthquake, it was reported that the anchor bolts of the main transformer (seismic design class C) installed outdoors, on the ground, at Kashiwazaki-Kariwa nuclear power plant were broken. In 2008, TEPCO conducted ultimate load testing by exerting dynamic loads on a mechanical foundation embedded in concrete using a large-scale shaking table at the National Research Institute for Earth Science and Disaster Prevention. The purpose was to analyse the seismic safety margin embedded in the design method [59].

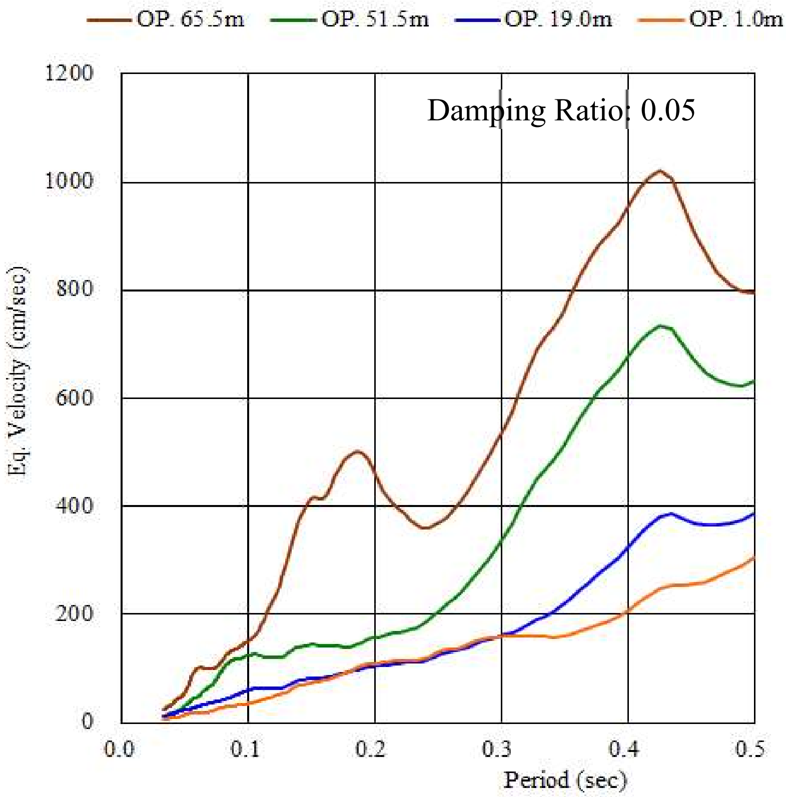


FIG.54. Elastic energy spectra inside reactor building of Fukushima Daiichi Unit 6.

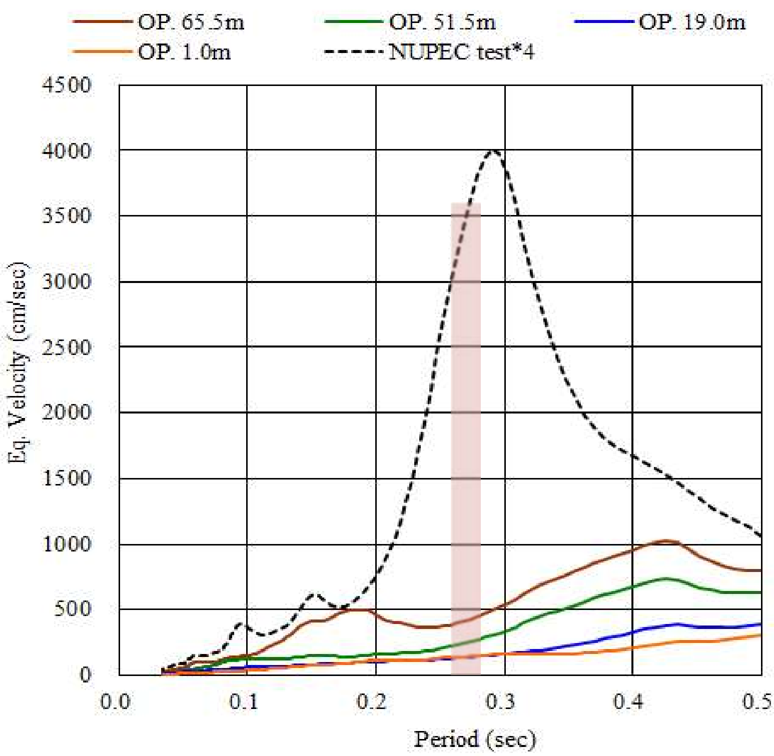


FIG. 55. Comparison of elastic energy spectrum of NUPEC/JNES test and observed at Fukushima Daiichi plant Unit 6.

TABLE 31. SEISMIC MARGIN ESTIMATED FROM ELASTIC ENERGY SPECTRUM

Floor in the reactor building of Fukushima Daiichi plant Unit 6	Seismic margin estimated
Roof (OP. 65.5 m)	8
Operating floor (OP. 51.5 m)	14
Second floor (OP. 19.0 m)	24
Base mat (OP. 1.0 m)	23

In this testing program, three series of tests, namely (a) pull-out loading test, (b) shear loading test, and (c) shaking test with real scale models were conducted. No damage occurred in the tests (c). However, the tests succeeded in verifying damage modes in the pulling test of the concrete anchorage portion, tests (a), and in the shear failure testing of anchor bolts, tests (b). Here, the correlation between DIPs and damage occurrences in the tests in which damage modes due to dynamic loads were obtained, is discussed.

4.5.2.1. Pull-out loading test of anchorage

In these tests, the elements of the anchorage system (e.g. anchor bolt diameter, embedment depth, and loading conditions) were systematically varied and 13 types of test specimens in the shapes shown in Fig. 56 were set on the concrete slab placed on the shaking table and vibrated one-dimensionally in a horizontal direction, as shown in Fig. 57. The acceleration time-history observed on the vibration table is shown in Fig. 58.

This acceleration time-history is an artificial earthquake motion that was calculated to produce a condition of resonance in the test specimens, in order to cause damage in them. Its DIPs are calculated as shown in the Table 32.

As a result of observation of the cross section of the bolt embedment after vibration, shear cone failures, which are peculiar to pull-out loading, were observed in two test specimens (Fig. 59). The pull-out load was calculated from the maximum response acceleration value that acted on each of the test specimens. The results of comparing it with the allowable pull-out load calculated with the design formula are shown in Fig. 60. The design allowable load III_AS shown in Fig. 60 is the allowable load calculated from the anchor bolt design method for the elastically dynamic design earthquake ground motion Sd that is employed in Japan as ‘elastically dynamic design basis earthquake ground motion’.

Test results provided important knowledge on ‘the dependency of the DIP threshold value upon the design method’, regarding the damage to the concrete side of the anchorage as discussed below.

In this testing, shear cone failures occurred in the concrete when the JMA instrumental seismic intensity was 4.3 (JMA seismic intensity grade 4). This value is considerably smaller than about grade 6 in the former JMA seismic intensity scale, experienced at general industrial facilities during the 1995 Southern Hyogo prefecture earthquake (shown in Table 10). It can be seen that both specimens in which shear cone failures were observed had anchor bolts, the embedment

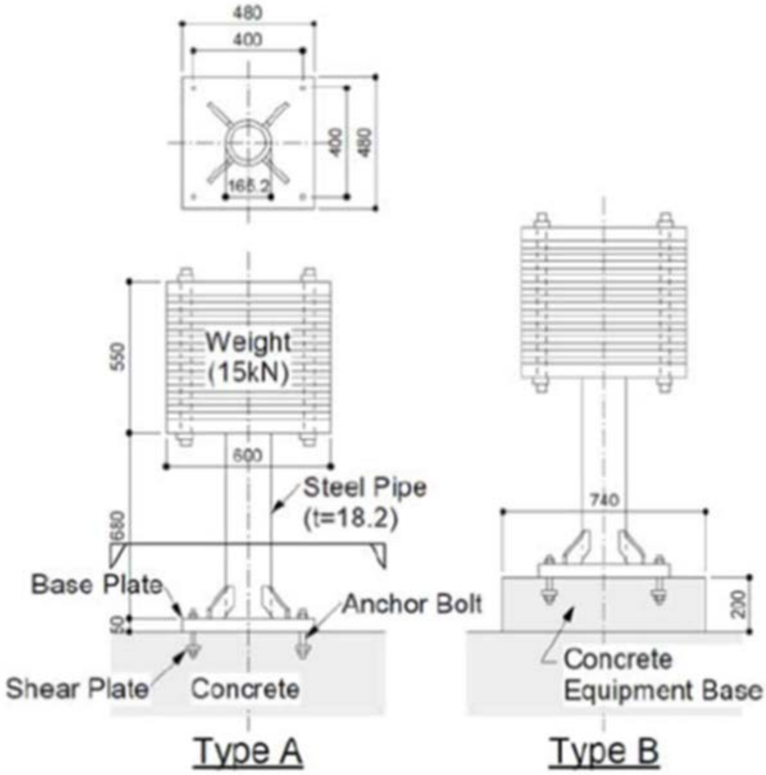


FIG. 56. Test specimen for pull-out loading [59].

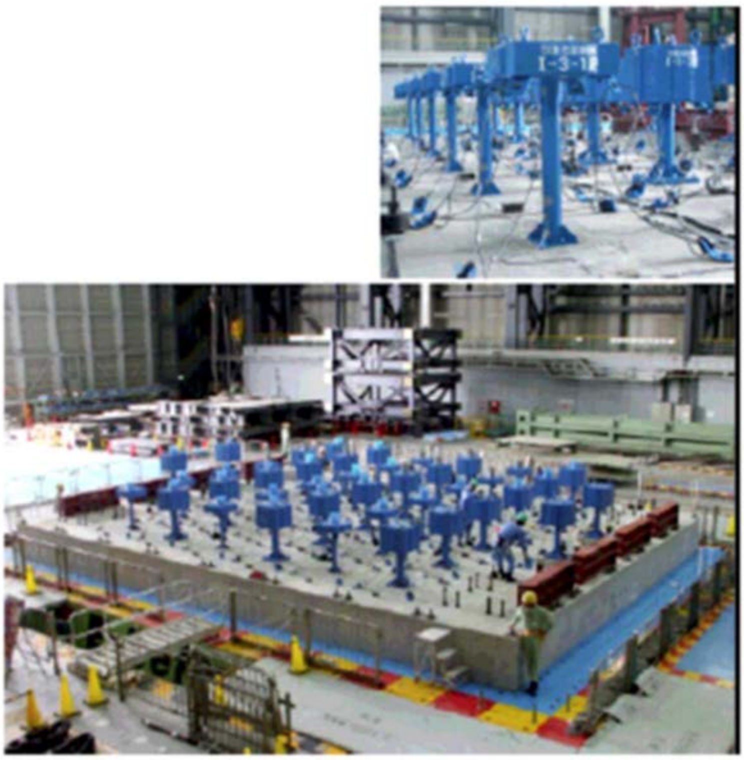


FIG. 57. Pull-out specimens on a concrete slab placed on the shaking table [59].

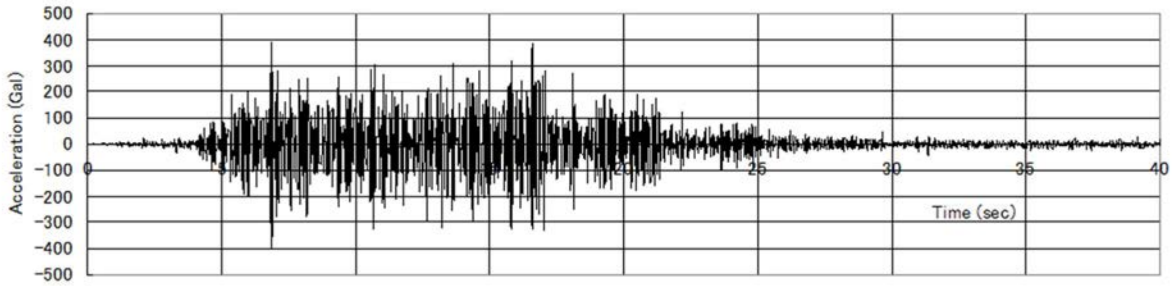


FIG. 58. Acceleration waveform observed on the shaking table in pull-out vibration test.

TABLE 32. CALCULATED DIPS OF THE INPUT MOTION (OBSERVED ON THE SHAKING TABLE)

Type of DIP	DIP Value
ZPA	399 Gal
A_{JMA}	49.7 Gal
I_{JMA}	4.3 (JMA seismic intensity scale: 4)
Standardized CAV	1.71 g-sec

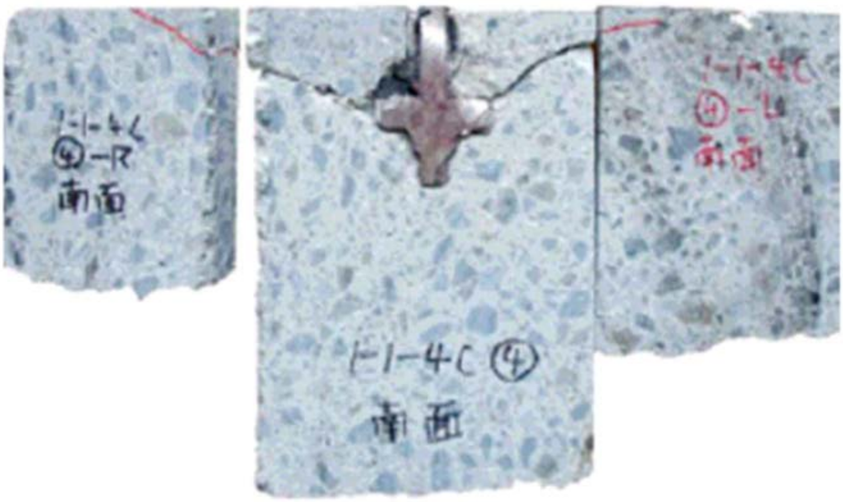
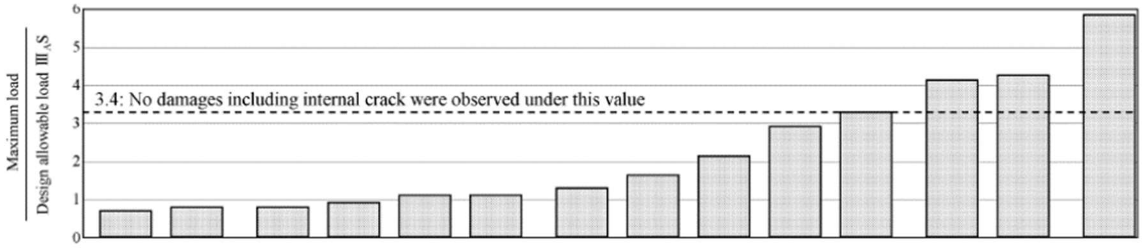


FIG. 59. Internal crack (Specimen I-1-4) [59].



No	I-1-1C	I-1-6C	I-1-5C	I-2-5C	I-2-1C	I-2-6C	I-1-2C	I-1-3C	I-2-2C	I-2-3C	III-1-2C	I-1-4C	III-1-2C	I-2-4C
Bolt	M16	M16	M16	M36	M36	M36	M16	M16	M36	M16	M16	M16	M16	M36
Embedment depth	150	150	150	139	139	139	100	75	90	66	49	50	49	50
Model Type	TypeA	TypeB	TypeA with Sleeve	TypeA with Sleeve	TypeA	TypeB	TypeA	TypeA	TypeA	TypeA	TypeB	TypeA	TypeB	TypeA
Internal crack												X		X

FIG.60. Comparison of maximum pull-out force in vibration test and design allowable value [59].

depth of which was set to be extremely small to let shear cone failures occur (embedment depth/bolt diameter = 1.4 to 1.3). On the other hand, general anchor bolt design normally employs an embedment depth more than ten times larger than the bolt diameter, which is nearly found in test specimens I-1-1C, 5C and 6C. In the anchor bolt design method, the pull-out strength (design allowable) increases almost in proportion to the embedment depth. In reference to the margins of these test specimens, therefore, the A_{JMA} threshold value will be larger than approximately 200 Gal (I_{JMA}=5.6), if the maximum allowable value was employed in the design.

4.5.2.2. Shear loading test of anchor bolts

In this test, weights were added to a steel plate (1600 mm x 1600 mm x 25 mm) simulating a baseplate for mechanical equipment, which was anchored to the concrete slab with four bolts (nominal diameter: 8 mm). Twelve test specimens (see Fig. 61) were placed and vibrated on the shaking table (see Fig. 62). The design was varied so that the load in the test is 0.5, 1.0, and 2.0 times the design allowable load. Other major variations were the presence or absence of a sleeve and the presence or absence of a concrete foundation base. The initial tightening torque for the bolts was reported to be 12 N m. A greased stainless-steel plate was inserted between the steel plate simulating the baseplate and the concrete, to create a very conservative testing condition without friction force between the baseplate and the concrete foundation.

As the waveform of the vibration input to the shaking table, the acceleration time-history observed on the reactor building basemat of the Kashiwazaki-Kariwa nuclear power plant Unit 1 in the east-west direction during 2007 Niigata-ken Chuetsu-Oki earthquake was used. The maximum acceleration observed actually on the concrete slab on the shaking table was reported to be 1270 Gal. Table 33 shows the results of DIP calculations with the maximum acceleration being 1270 Gal.

Due to this excitation, the anchor bolts of two of the test specimens listed in Table 34 broke, as shown in Fig. 63. The test specimens were observed to be moving on the concrete slab.

This result has given valuable information that the margin against the shear failure of anchor bolts under Japan's seismic design method is from one to two, when expressed as the quotient between the maximum load produced by earthquake acceleration and the design allowable load

(III$_A$S), when the friction force between the mechanical foundation and the concrete is ignored. This result is worth analysing from the perspective of DIPs.

According to Japan's anchor bolt seismic design method, test specimens II-1-2, II-1-3 and II-1-4 would reach their design allowable values when the static seismic coefficient is 0.70, 0.41, and 0.25, respectively. Based on this vibration test result, therefore, test pieces with the design static seismic coefficient of 0.41 (400 Gal) could withstand the seismic motion input of A_{JMA}=492 Gal, while those with the design static seismic coefficient of 0.25 (245 Gal) could not. Considering that the strength of actual materials is greater than the design standard value, it can be said that the A_{JMA}, as a DIP for first excursion damage, is a parameter close to the concept of static seismic coefficient.

On the other hand, friction force is working between the mechanical foundation and the concrete due to dead weight. The initial tightening torque is also contributing to the friction force. Nevertheless, it needs to be considered that the frictional force due to dead weight will decrease because of vertical motions.

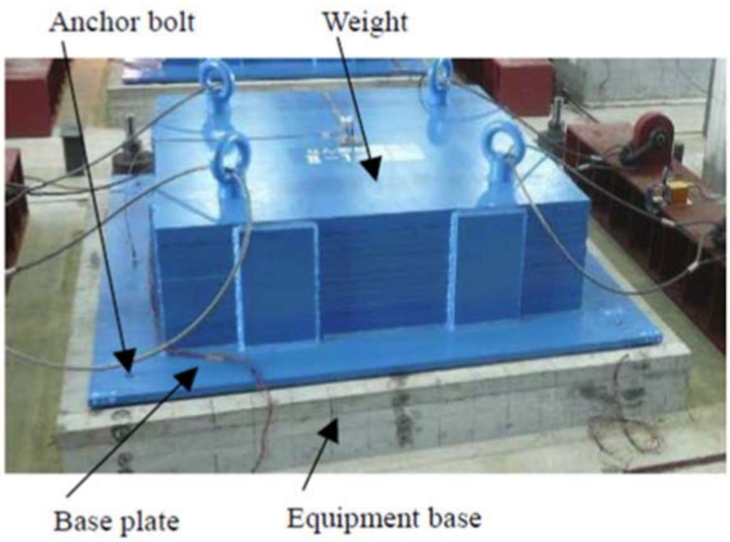


FIG. 61. Test model for shear loading (Type B) [59].

FIG. 62. Shear loading specimens on concrete slab [59].

TABLE 33. DIP ESTIMATION OF THE ACCELERATION INPUT TO THE SHAKING TABLE

DIP	Input to the shaking table (Estimated)	Earthquake motion observed in the east-west direction on the reactor building basemat of Kashiwazaki-Kariwa plant Unit 1 (2007 Niigata-ken Chuetsu-Oki earthquake)
Maximum acceleration (Gal)	1270	680
A_{JMA} (Gal)	492	263
I_{JMA}(JMA seismic intensity scale)	6.3 (6-upper)	5.7 (6-lower)
Standardized CAV (g-sec)	2.44	1.16

FIG. 63. Damage in bolt and surrounding concrete [59].

When JMA seismic intensity grade 6-upper (approximately 340 Gal in A_{JMA}) is observed, it can be concluded that anchor bolts of conventional facilities, including Seismic Class C components at Japanese nuclear power plants, are to be checked, for example, with respect to the design margin (ratio of the design load to the design allowable load) and so on, in order to find potential hidden damage.

4.6. APPLICATION OF DAMAGE INDICATING PARAMETERS TO POST-EARTHQUAKE ACTIONS AND SEISMIC INSTRUMENTATION SYSTEM

This section proposes the application of the DIPs discussed in chapters 3 and 4 to enhance the reliability of during and post-earthquake actions addressed in IAEA Safety Report Series 66, as mentioned in Section 1.

Table 35 summarizes the ideas on how to utilize DIPs and it gives the corresponding sections where the suggested utilization is discussed in detail.

TABLE 34. SPECIFICATION AND RESULTS OF SPECIMENS IN VIBRATION TEST FOR SHEAR LOADING [59]

<table>
<tr><th rowspan="2">Specimen</th><th rowspan="2">Mode Type[1]</th><th rowspan="2">Bolt diameter</th><th rowspan="2">Embedment depth (mm)</th><th rowspan="2">Shear plate (mm)</th><th rowspan="2">Give initial fastening force</th><th colspan="2">Test result</th></tr>
<tr><th>Observed load[2]</th><th>Fracture of anchor portion</th></tr>
<tr><td rowspan="2">II-1-1</td><td rowspan="2">Type A
Design allowable load III$_A$S x 1.0</td><td rowspan="12">M8</td><td rowspan="2">76</td><td rowspan="2">40</td><td rowspan="2"></td><td>1.01</td><td></td></tr>
<tr><td>0.92</td><td></td></tr>
<tr><td rowspan="2">II-1-2</td><td rowspan="2">Type A
Design allowable load III$_A$S x 0.5</td><td rowspan="2">76</td><td rowspan="2">40</td><td rowspan="10">X</td><td>0.59</td><td></td></tr>
<tr><td>0.57</td><td></td></tr>
<tr><td rowspan="2">II-1-3</td><td rowspan="2">Type A
Design allowable load III$_A$S x 1.0</td><td rowspan="2">76</td><td rowspan="2">40</td><td>1.05</td><td></td></tr>
<tr><td>0.94</td><td></td></tr>
<tr><td rowspan="2">II-1-4</td><td rowspan="2">Type A
Design allowable load III$_A$S x 2.0</td><td rowspan="2">76</td><td rowspan="2">40</td><td>1.91</td><td>X</td></tr>
<tr><td>1.86</td><td>X</td></tr>
<tr><td rowspan="2">II-1-5</td><td rowspan="2">Type A
Design allowable load III$_A$S x 1.0</td><td rowspan="2">76</td><td rowspan="2">80</td><td>0.96</td><td></td></tr>
<tr><td>1.07</td><td></td></tr>
<tr><td rowspan="2">II-1-6</td><td rowspan="2">Type B
Design allowable load III$_A$S x 1.0</td><td rowspan="2">150</td><td rowspan="2">40</td><td>0.94</td><td></td></tr>
<tr><td>0.96</td><td></td></tr>
</table>

Notes: (1) Type A: Without concrete equipment base / Type B: With concrete equipment base
(2) Observed Load = (Maximum load by maximum acceleration) / (Design allowable load III$_A$S)

4.6.1. Alarm of significant earthquake

As mentioned so far, the peak acceleration value of observed seismic motions is not always indicative of damage to structurers, systems or components. On the other hand, the improvements in seismometer sensitivity allows for observation of high frequency noise, which could lead to a misinterpretation of the intensity of the felt earthquake, resulting in unnecessary alarms to the reactor operators.

TABLE 35. POSSIBLE ROLES OF DIP FOR DURING AND POST-EARTHQUAKE ACTIONS

<table>
<tr><th colspan="2">Timing</th><th>Role of DIPs</th><th>Action/Instrumentation</th></tr>
<tr><td colspan="2" rowspan="2">During earthquake</td><td>Shutdown criteria exceeded?
ASTS trigger signal
(Section 4.6.6)</td><td>Reactor automatic shutdown</td></tr>
<tr><td>Felt earthquake is significant?
Alarm signal
(Section 4.6.1)</td><td>Alarm for operators</td></tr>
<tr><td rowspan="4">Post-earthquake</td><td rowspan="2">< one day</td><td>Depth of walkdown inspection?
Damage forecast
(Section 4.6.2)</td><td>Walkdown by operators</td></tr>
<tr><td>Design basis exceeded?
Exceedance criteria
(Section 4.6.3)</td><td>Reactor manual shutdown</td></tr>
<tr><td rowspan="2">> one day</td><td>Equipment to inspect?
Selection criteria
(Section 4.6.4)</td><td>Initial focused inspection</td></tr>
<tr><td>Hidden damage?
Seismic margin index
(Section 4.6.5)</td><td>Plant integrity evaluation</td></tr>
</table>

As a DIP suitable for identifying significant earthquake motions, in other words, relevant earthquake motions from the viewpoint of damage, the standardized CAV value is proposed because it indicates the possibility of some sort of damage due to its relationship with the general definition of seismic intensity. The threshold set by the EPRI, 0.16 g-sec, is extremely conservative. This value is almost the lowest threshold. It is correlated with MSK intensity grade VII observed in India, as addressed in Section 4.2. It needs to be noted here that 0.16 g-sec means that there is a possibility that damage will occur at the observed earthquake motion, assuming all possible cases involving the seismic capacity of target SSCs, and the characteristics of earthquake motions and so on. The experience of the North Anna nuclear generating station addressed in Section 4.4.2 shows that even if this value is exceeded, it does not always result in damage to nuclear power plant SSCs. The threshold value at which the occurrence of damage to specific SSCs in a nuclear power plant needs to be anticipated will be discussed in the next section.

When standardized CAV value is utilized as an alarm trigger during the earthquake, its time-history value, that is, the real-time value, needs to be discussed from the standpoint of starting alarm signals. Table 36 shows typical CAV time-history calculation results. Both of CAV and standardized CAV can be used for producing alarm signals about the occurrence of a significant earthquake. However, it needs to be noted that the real-time value of the standardized CAV has a one-second delay, due to the calculation procedure shown in Section 4.1.3.

4.6.2. Prediction of seismic damage immediately after an earthquake

As mentioned in the preceding section, the proposed threshold value of standardized CAV, 0.16 g-sec, can be helpful for operators to judge whether an observed earthquake motion be significant or not from the viewpoint of damage to SSCs. Additionally, the instrumental seismic intensity level is available immediately after an earthquake in some countries, such as the MMI instrumental intensity of ShakeMaps in the United States of America, or the JMA seismic intensity in Japan. As discussed in Section 3.2, the seismic intensity scales utilized around the world are a good parameter to measure earthquake motion level and they can be utilized as DIP, not only for buildings but also for electric and mechanical equipment.

Most of the seismic damage to conventional mechanical equipment, especially due to inertial forces, is categorized as first excursion damage. Based on the experience in Japan (see Table 10), equipment for which no seismic capacity improvement measures have been taken (e.g. unanchored equipment), may be damaged at about original JMA seismic intensity scale grade 4 or higher; when anchored, it may be damaged at about original JMA seismic intensity scale grade 5 or higher; and the equipment for which seismic design has been applied may be damaged at about original JMA seismic intensity scale grade 6 or higher. When the instrumental seismic intensity is provided to operators, they can estimate the level of the damage to be expected. In this regard, a seismic intensity meter is installed in some nuclear power plants in Japan to provide the operators with the instrumental seismic intensity level.

If A_{JMA} is computed in real time at a nuclear power plant, it might be better a DIP than the seismic intensity scales. The results of DIP computation for conventional plant equipment (e.g. thermal power plants and substations) show that equipment damage has occurred when A_{JMA} is about 80 Gal (JMA seismic intensity grade 5-L) or higher, while Japanese nuclear power generating equipment has been damaged when A_{JMA} is 200 Gal (JMA seismic intensity grade 6-L) or higher (See Fig. 44).

When a nuclear power plant is affected by an earthquake, and the earthquake motion intensity is judged to be significant, as mentioned in the foregoing section, an operator walkdown is to be performed. It may be reasonable to set the scope and details of inspection depending on the earthquake motion level specified by DIPs as mentioned above.

4.6.3. Design basis earthquake exceedance criteria

4.6.3.1. Basic approach

It is discussed in Section 4.1.1 that both first excursion damage and cumulative fatigue damage caused by seismic vibrations need to be considered, and A_{JMA} and standardized CAV are proposed as DIPs. The former is a peak parameter for first excursion damage, and the latter is an integral parameter for cumulative fatigue damage.

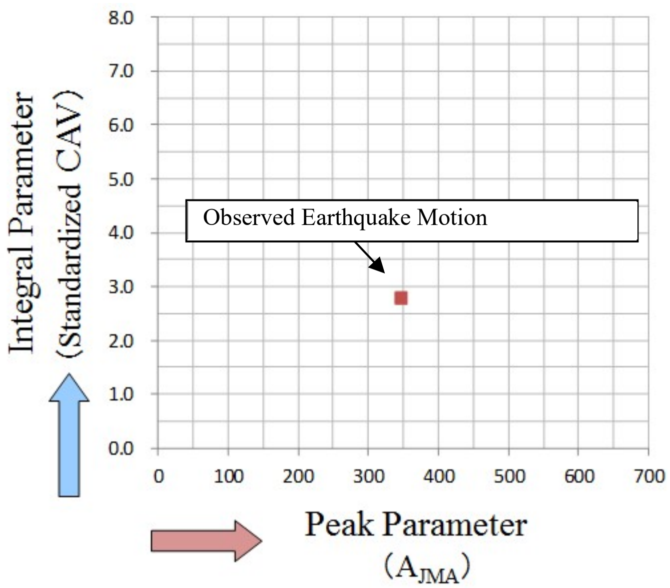
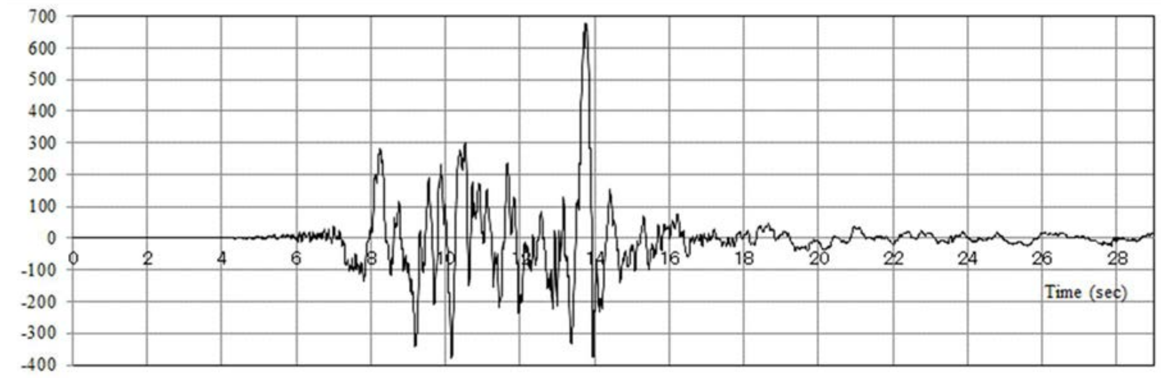
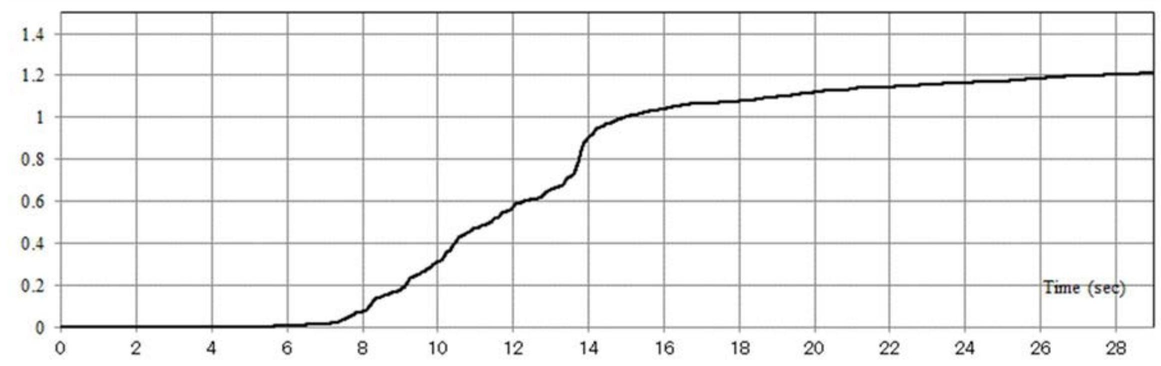
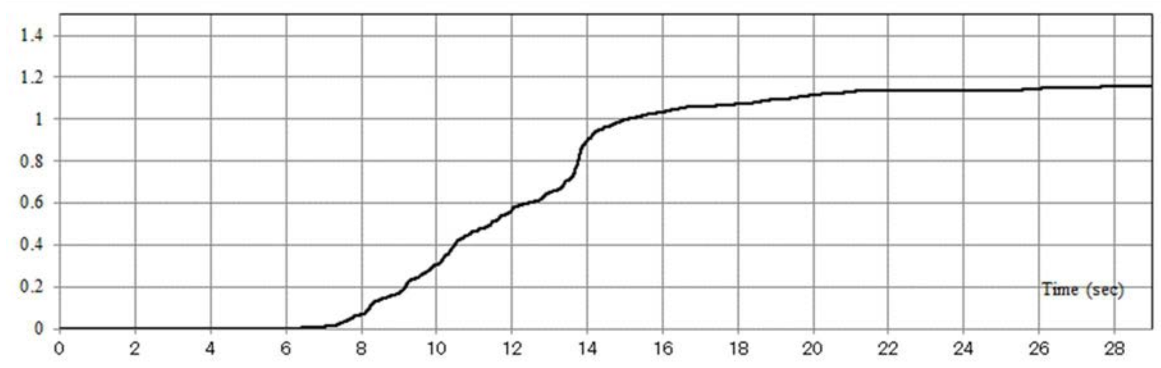


FIG. 64. Peak parameter and integral parameter of the observed earthquake motion.

TABLE 36. TYPICAL TIME-HISTORY OF CAV VALUE DURING EARTHQUAKE (KASHIWAZAKI-KARIWA UNIT 1 REACTOR BUILDING BASEMAT IN 2007 NIIGATA-KEN CHUETU-OKI EARTHQUAKE)

Acceleration Time-History (Gal)

CAV (g-sec)

Standardized CAV (g-sec)


The relation between A_{JMA} and standardized CAV is shown in Fig. 41 and their capacities to characterize earthquake motion seem to be different. In other words, the existence of seismic damage will be judged from the position in a 2D space (plane) defined by two DIPs, one peak parameter (typically A_{JMA}) and one cumulative parameter (typically standardized CAV), as shown in Fig. 64, rather than evaluating seismic damage with single DIP.

A challenge for determining the region within the 2D space where damage is expected is that, doing so, requires an extremely large number of tests and research work. It is possible that such a region is dependent on the type of component, the seismic design methods and even on the judgments made in actual design. Thus, it is considered more realistic to compare the DIPs of observed earthquake motion at the installation points of SSCs with that of design basis earthquake input motion to the SSCs, considering that the margins included in the design methods (seismic response analysis, load and stress calculation, allowable stress, etc.) and in the actual design are on the conservative-side to evaluate threshold DIPs.

4.6.3.2. Equivalent DIP to static seismic coefficient

The threshold DIP values (standardized CAV and A_{JMA}) for the design basis earthquake input motion can be calculated with the equations shown in Section 4.1, if the acceleration time-histories are available. And this is applicable to both design basis ground motions and in-structure motions as to compute the threshold DIPs. However, it may be difficult to apply this method to mechanical equipment, because seismic design conditions for mechanical equipment, in particular, are often given in terms of static seismic coefficients or acceleration response spectrum.

For example, floor acceleration response spectrum is often used for the seismic design of mechanical equipment. Usually, however, artificial processing (e.g. the envelope of several seismic waveforms and smoothing against the natural frequency) has been applied to the floor response spectrum used for design. It takes time and effort to calculate artificial earthquake motion time-histories from the floor response spectrum in order to compute A_{JMA} or Standardized CAV. Depending on the seismic design practice, moreover, non-safety-related equipment of low seismic importance is often designed using a static seismic load defined by a static seismic coefficient.

In the Japan's seismic design practice, dynamic design and static design are used together depending on the seismic importance. Every structure, system and component in a nuclear power plant is classified into three levels, depending on its seismic safety importance. These levels are Class S, Class B and Class C, and the ratio of static seismic coefficient for the design earthquake load calculation is 3.0, 1.5 and 1.0, respectively. The minimum design load value for Class C component is defined as $0.2\times0.8\times1.2=0.192$ (g).

The calculation process of JMA instrumental seismic intensity, shown in Section 3.3.2.2, suggests that the physical meaning of A_{JMA} may be close to that of static seismic coefficient.

The acceleration calculated from static seismic coefficient and acceleration response spectrum is a design value against first excursion damage, and it is considered to be a momentary parameter such as the ZPA or the A_{JMA}. Fig. 65 shows the correlation between A_{JMA} and ZPA with horizontal lines corresponding to the static seismic coefficients used in the Japan's seismic design practice. Although, no seismic damage to safety-related components (seismic design class S) has ever been experienced, damage has occurred to class B and C components, which are designed using static seismic coefficient, and shown with solid markers in Fig. 65. The lines representing static seismic coefficients on the vertical axis (A_{JMA}) look like indicating the threshold of damage occurrence.

As discussed in Section 3.3.2.2, the calculation process of A_{JMA} considers a frequency filter which cuts off the high frequency contents having less influence on damage of ductile mechanical equipment and it also considers the velocity and accumulation effect (energy). In

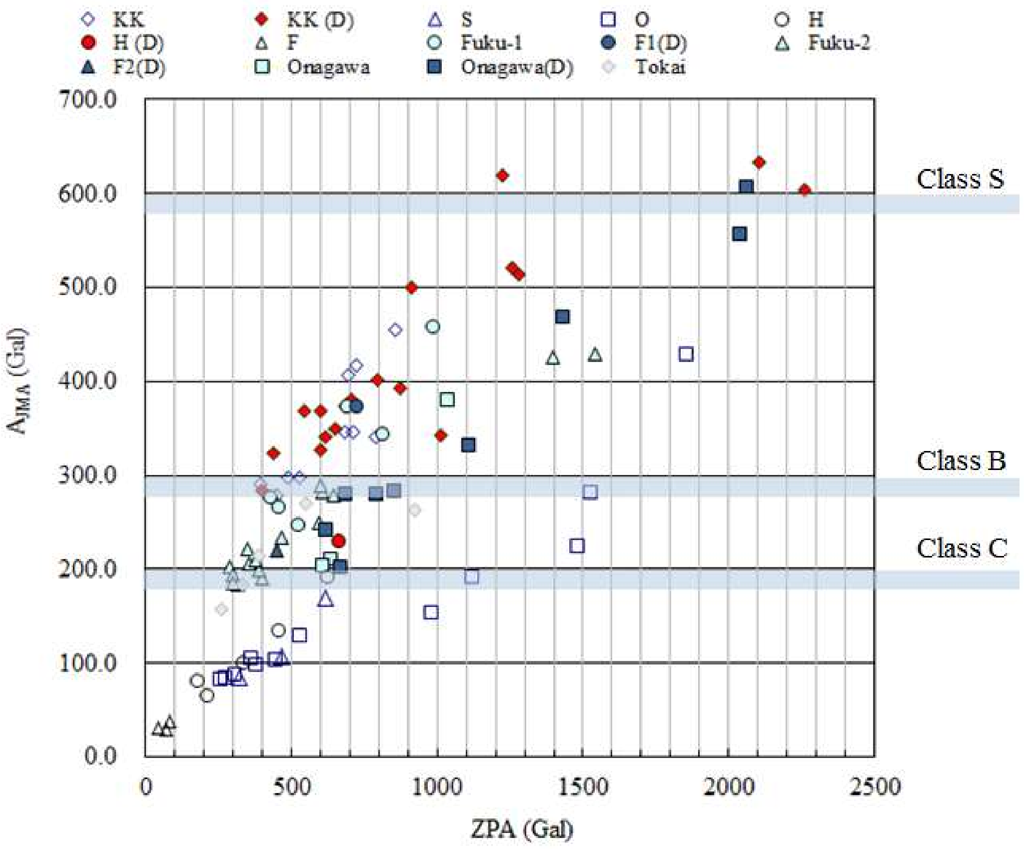


FIG. 65. A_{JMA} and seismic static design load.

other words, A_{JMA} is considered to be one of parameters so-called as 'effective acceleration'. On the other hand, static seismic coefficient corresponds to an acceleration with infinite duration period and affects SSCs as a seismic inertial force.

4.6.3.3. Exceedance evaluation chart

Fig. 66 is proposed as a chart to determine whether the observed earthquake motion exceeds the design basis, in which either acceleration time-histories or static seismic coefficients are specified. The shaded area in this figure corresponds to both the peak parameter and the integral parameter being below the values calculated for the design earthquake. When the point of observed earthquake remains in this area and the calculated A_{JMA} is below the static seismic coefficient line, the observed earthquake motion can be judged as not exceeding the design basis.

It needs to be noted that the actual exceedance determination needs to consider both the assessment by DIPs as mentioned above and the assessment by the response spectrum, because the vibrational effect of SSCs in the seismic response needs to be taken into account as discussed in IAEA Safety Report Series No. 66 [1] and others. In addition, consideration to some kind of margin needs to be given when performing evaluations with DIPs like peak parameters, especially with the relatively flexible SSCs, because of lack of information about their seismic response.

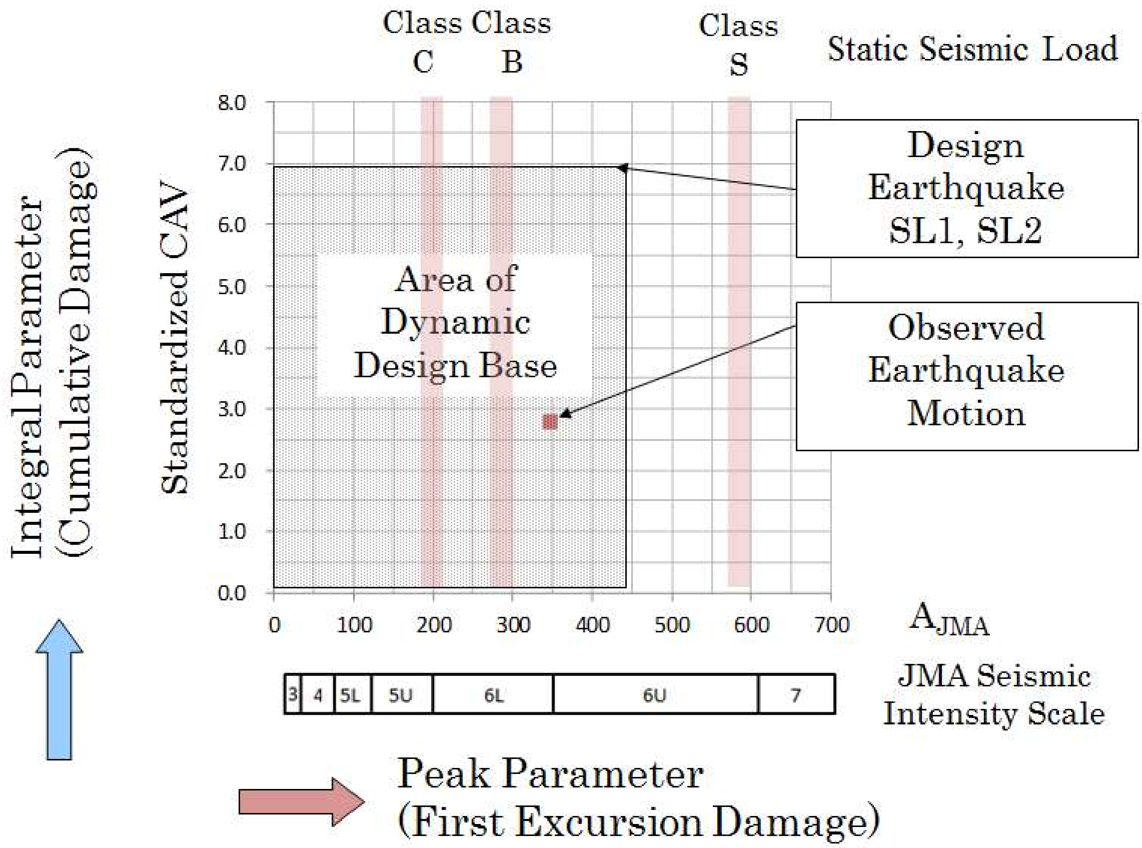


FIG. 66. Diagram to assess exceedance of seismic design bases based on computed DIPs.

4.6.4. Improving reliability of post-earthquake inspections

Nuclear power plant SSCs, which require a high reliability, often employ the same structural design that has had a good performance record in the past or that has been standardized in accordance with Codes and Standards. Thus, existing damage data and test results are useful in evaluating the seismic capacity of SSCs with similar configurations.

In Section 4.3 of this publication is presented, for example, the threshold value for the seismic capacity of conventional industrial facilities. In Section 4.4.1, it is shown from the seismic experience of Japan's nuclear power plants that the main steam turbine shaft of a nuclear power plant might be slightly displaced at an A_{JMA} value equal or greater than 200 Gal. In addition, the ultimate strength test on a piping system described in Section 4.5.1 shows that the occurrence of damage due to crack penetration caused by the seismic loading does not have to be considered unless an extremely large standardized CAV value is produced by the earthquake.

As seen from the above, it is possible to determine the necessity of equipment overhauls or in-depth inspections by comparing the DIPs calculated from the observed earthquake motions around the location where the equipment is installed with the existing damage data of similar structures. This kind of application of DIP is expected to improve the reliability of post-earthquake actions for hidden damage, in particular, and allow for prompt and rational post-earthquake actions.

4.6.5. Index to evaluate seismic margin

The DIP evaluation results of the ultimate strength testing of the piping presented in Section 4.5.1 and of the anchor bolts presented in Section 4.5.2 show that it is very important to select an appropriate physical property as a parameter in evaluating the seismic safety margin of SSCs. As explained in Section 3.1.2, it is clear that there are two seismic failure modes: first excursion failure, and cumulative fatigue failure. The seismic margin against them varies depending on structural characteristics, and the characteristics of earthquake motions affecting these failure modes are so complicated that they cannot be expressed with just the peak acceleration and the shape of an acceleration response spectrum.

Since expensive and difficult vibration testing in relation with ultimate strength cannot be performed so often, it is still difficult to create a fragility curve for each equipment class using DIPs as parameters. However, it is important to consider DIPs when developing vibration motion input for conducting ultimate strength testing.

Many definitions of 'seismic margin' have been developed for nuclear power plants when affected by a strong earthquake. In this section, two types of seismic margin are discussed: (a) margin of the earthquake motion considered in seismic design against the observed earthquake, and (b) margin of the ultimate functional capacity of an SSC against the effects of the observed earthquake.

The seismic margin (a) is shown in Fig. 66, as discussed in Section 4.6.3.3. Because two parameters, peak parameter and integral parameter need to be considered to assess the consequences of an earthquake, the seismic margin is expressed using the two-dimensional space shown in Fig. 67. The smaller value of the ratio of D_P to O_P and of D_I to O_I is the seismic margin (a). Note that the smaller value informs about the dominant damage mode as well.

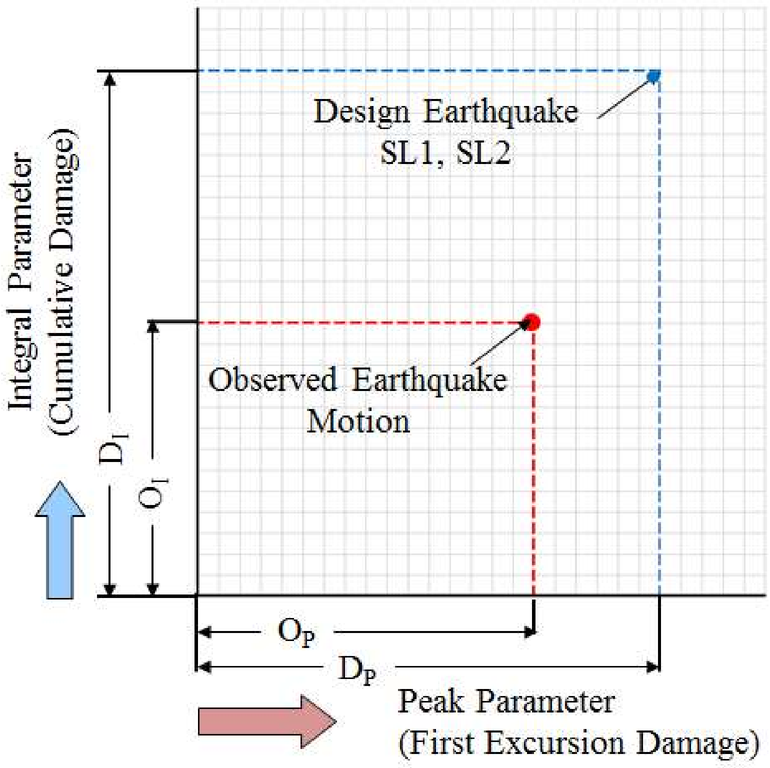


FIG. 67. Scheme of seismic margin evaluation.

As for the discussion on the seismic margin (b), it is necessary to determine the structure, system or component governing the plant seismic capacity. The ultimate functional capability of the element governing the capacity is usually determined by ultimate strength testing (as shown in Section 4.5) or by sophisticated structural analysis. The discussion in sections 3 and 4 helps evaluate the effect of an earthquake motion on structural damage. DIPs are also valuable to decide the input motion to be used in the vibration test. On the other hand, when the characteristics of a target earthquake motion are identified by the use of DIPs, it becomes easier to evaluate the ultimate capacity with analysis or test on the element governing the capacity.

4.6.6. Automatic seismic trip system trigger signal

When a DIP value is utilized as the ASTS trigger signal of a nuclear power plant, it needs to be a real time value that can be processed on a moment-to-moment basis for DIP exceedance determination during an earthquake.

Like in the ASTS of a nuclear power plant, signals during an earthquake are used for general industrial purposes to discontinue operation, thereby minimizing seismic damage to the extent possible. To this end, seismic instruments that are categorized as seismic switches, have been proposed. When the observed acceleration itself is used as a parameter, real time computation (data processing) is not required. However, it may cause malfunctions because acceleration does not always lead to component damage, as mentioned earlier, and depending on the sensitivity of the seismometer, high frequency noise may also be picked up.

To minimize seismically induced damage by early shutdown, it may be reasonable to consider that actions will be taken to prevent first excursion damage, which is more likely to occur in a short interval of time than cumulative damage. The JMA instrumental seismic intensity, which has been discussed as a parameter for first excursion damage, does not lend itself to real time computation during an earthquake because it uses a frequency range filter, as shown in the computation procedure in Section 4.1.2. In Japan, therefore, various methods for computing real time instrumental intensity have been proposed.

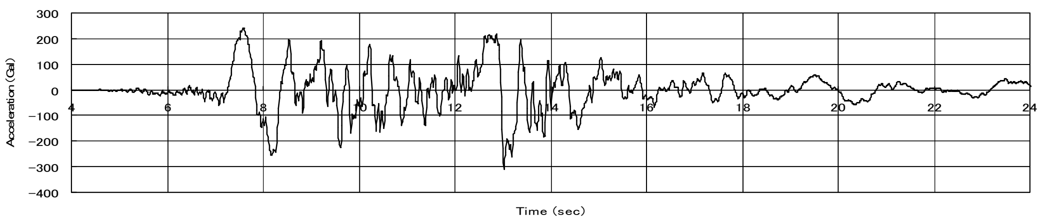
The National Research Institute for Earth Science and Disaster Prevention in Japan has proposed a method called the real time seismic intensity indicator (I_r), in which JMA instrumental seismic intensity is calculated by approximating the frequency filter with a filter in the time domain.

The details and algorithm of Ir computation are shown in Ref. [60]. Fig. 68 shows the correlation between two A_{JMA}, one is from the original calculation procedure shown in Section 3.1.2 and the other, shown as 'Real Time A_{JMA}' in the figure, from the Ir processing method indicated in Ref. [60]. Points in the figure have been calculated using observed earthquake waves at nuclear power plants and conventional industrial facilities. A typical calculated time-history of the Real Time A_{JMA} , and the corresponding Ir, is shown in Table 37. It can be seen that Ir has a promising role as an indicator for ASTS in the future. However, the calculation time of Real Time A_{JMA} increases with the earthquake duration. This is an issue to be solved for the actual application.

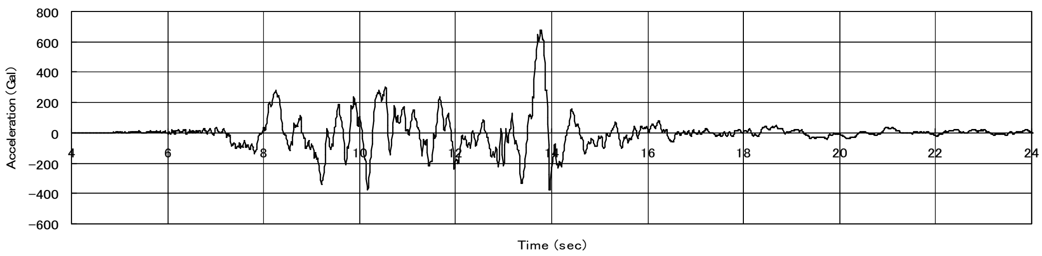
TABLE 37. TYPICAL TIME-HISTORY OF REAL TIME I_{JMA} (IR) AND REAL TIME A_{JMA} (OBSERVED ACCELERATION AT KASHIWAZAKI-KARIWA UNIT 1 REACTOR BUILDING BASE MAT DURING 2007 NIIGATA-KEN CHUETSU-OKI EARTHQUAKE)

Acceleration Time Histories

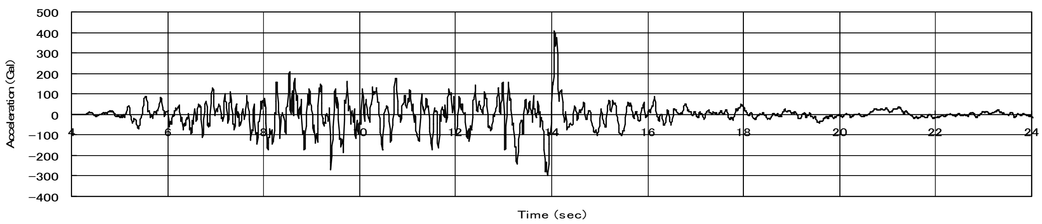
NS



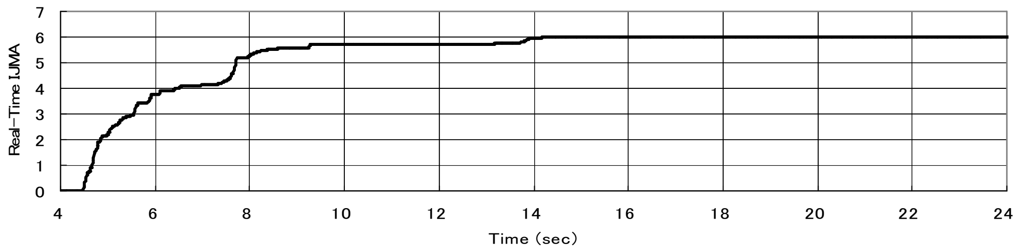
EW



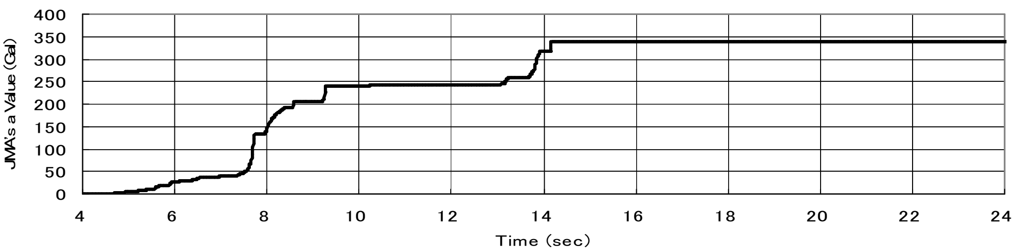
UD



Real-Time I_{JMA} (Ir)



Real-Time A_{JMA}



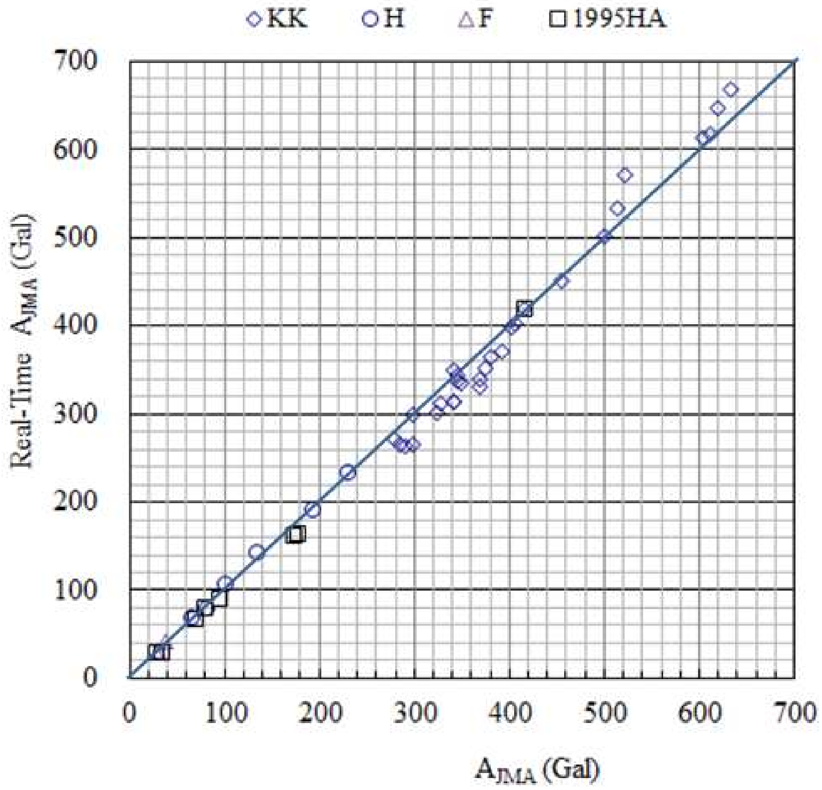


FIG. 68. Correlations between original A_{JMA} and Real Time A_{JMA}

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Earthquake Preparedness and Response for Nuclear Power Plants, IAEA Safety Reports Series No.66, IAEA (2011).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003).

[3] ELECTRIC POWER RESEARCH INSTITUTE, Guidelines for Nuclear Station Response to an Earthquake, EPRI NP-6695 (1989).

[4] AMERICAN NUCLEAR SOCIETY, Criteria for the Handling and Initial Assessment of Records from Nuclear Power Station Seismic Instrumentation, ANSI/ANS-2.10-2003 (2003).

[5] ELECTRIC POWER RESEARCH INSTITUTE, Seismic Evaluation Guidance, Screening Prioritization and Implementation Details (SPID) for the Resolution of Post Fukushima Near Term Task Force Recommendation 2.1: Seismic, EPRI Technical Report 1025287 (2013).

[6] UNITED STATES NUCLEAR REGULATORY COMMISSION, Nuclear Power Plant Instrumentation for Earthquakes, Rev. 3, NRC Regulatory Guide 1.12 (2017).

[7] NUCLEAR SAFETY STANDARDS COMMISSION, Design of Nuclear Power Plants against Seismic Events; Part 5: Seismic Instrumentation, KTA Safety Standards 2201.5 (2015).

[8] AUTORITÉ DE SÛRETÉ NUCLÉAIRE, Règle Fondamentale de Sureté (RFS) n° I.3.b - Instrumentation sismique, ASN (1984) (in French).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-9, IAEA, Vienna (2010).

[10] UNITED STATES GEOLOGICAL SURVEY, Instrumentation Guidelines for the Advanced National Seismic System, Prepared for US Geological Survey and ANSS National Implementation Committee, Prepared by Working Group D of the ANSS Technical Integration Committee (2007).

[11] UNITED STATES GEOLOGICAL SURVEY, Guideline for ANSS Seismic Monitoring of Engineered Civil Systems – Version 1.0, Public Review Draft, Prepared by Advanced National Seismic System (ANSS), Structural Instrumentation Guideline Committee, USGS Open-File Report 2005–1039 (2005).

[12] ELECTRIC POWER RESEARCH INSTITUTE, Seismic Instrumentation in Nuclear Power Plants for Response to OBE Exceedance: Guidance for Implementation, EPRI TR-104239 (1994).

[13] ELECTRIC POWER RESEARCH INSTITUTE, Seismic Instrumentation at Nuclear Power Plants, EPRI White Paper, Product ID 1024889 (2012).

[14] AMERICAN NUCLEAR SOCIETY, Nuclear Station Response to an Earthquake, ANSI/ANS-2.23-2002 (2002).

[15] NUCLEAR SAFETY STANDARDS COMMISSION, Design of Nuclear Power Plants against Seismic Events Part 6: Post-Seismic Measures, KTA Safety Standards 2201.6 (1992).

[16] NUCLEAR SAFETY STANDARDS COMMISSION, Auslegung von Kernkraftwerken gegen Seismische Einwirkungen, Teil 4: Anforderungen an Verfahren zum Nachweis der Erdbebensicherheit für Maschinen- und Elektrotechnische Anlagenteile, KTA 2201.4 (1990).

[17] NUCLEAR SAFETY STANDARDS COMMISSION, Komponenten des Primärkreises von Leichtwasserreaktoren, Teil 2: Auslegung, Konstruktion und Berechnung, KTA 3201.2 (1996).

[18] JAPAN NUCLEAR ENERGY SAFETY ORGANIZATION, Report on the Selection of SSCs for Post-Earthquake Station Inspections from the Viewpoint of Seismic Design, JNES 07-KIKOUHOU-0001 (2008) (in Japanese).

[19] UNITED STATES NUCLEAR REGULATORY COMMISSION, Performance Issues with Seismic Instrumentation and Associated Systems for Operating Reactors, NRC Information notice 2012-25 (2013).

[20] ABE, S., ET AL, “Study on high-reliability of Real-time seismic observation system”, Proc. AIJ Tohoku Chapter Architectural Research Meeting **71** (2008) 175–178 (in Japanese).

[21] HIROTANI, K., The Examples about the Measures of the Earthquakes of ONAGAWA Nuclear Power Plant, 1st Kashiwazaki International Symposium on Seismic Safety of Nuclear Installations (2010).

[22] SHIBATA, H., The Seismic Trigger System is a Kind of Prediction System, IAEA EBP, WA3, 6th SWT Meeting, Mumbai, India (2010).

[23] INTERNATIONAL ATOMIC ENERGY AGENCY, Consultant Meeting on the Advisability of an Automatic Scram Trip System (ASTS) in Nuclear Power Plants, IAEA Consultants Meeting Report, IAEA, Vienna, 3–5 April (1995).

[24] O’CONNELL, W.J., WELLS, J.E., On the Advisability of an Automatic Seismic Scram, Lawrence Livermore National Laboratory, Livermore, CA, Prepared for Office of Nuclear Regulatory Research, US Nuclear Regulatory Commission, Washington, D.C., Report No. NUREG/CR-2513, UCRL-53037, RD, RM, 1.

[25] UNITED STATES NUCLEAR REGULATORY COMMISSION, NRC Resolution of Generic Safety Issues: Item D-1: Advisability of a Seismic Scram (Rev. 1) (NUREG-0933, Main Report with Supplements 1–34) (2011).

[26] IRISAWA, Y., The Recipients Remark for 2009 JSME Power and Energy System Division Award, Adapted from JSME Newsletter, Power and Energy System, No. 38, ISSN 1340-6671 (2009).

[27] JAPAN ELECTRIC ASSOCIATION, Technical Guidelines for Aseismic Design of Nuclear Power Plant, JEA JEAG 4601-1987 (translated into English as NUREG/CR-6241).

[28] EUROPEAN NUCLEAR SAFETY REGULATORS GROUP, Compilation of recommendations and suggestions Peer review of stress tests performed on European nuclear power plants, ENSREG (2012), http://www.ensreg.eu/sites/default/files/Compilation%20of%20Recommendations1.pdf

[29] JAPAN SOCIETY OF MECHANICAL ENGINEERS, Great Hanshin-Awaji Earthquake Investigation Report; Machinery Section; Mechanical equipment damage, Great Hanshin-Awaji Earthquake Investigation Report Compilation Committee, JSME CPD-A22-101 (1999).

[30] NAKANE, M., ET AL., Effect of Pre-strain on Low Cycle Fatigue Life, The Piping Engineering (Haikan Gijutsu) (2011) 42–47 (in Japanese).

[31] JAPAN NUCLEAR TECHNOLOGY INSTITUTE, Structural Integrity Assessment for Nuclear Component Damaged by Earthquake, Interim Report, JANTI, Tokyo (2007–2012).

[32] RICHTER, C.F., Elementary Seismology, W.H. Freeman and Co, San Francisco, CA (1958).

[33] MEDVEDEV, W., SPONHEUER, W., KARNIK, V., Seismic Intensity Scale Version MSK 1964, International Association of Seismology and Physics of the Earth's Interior (1964).

[34] JAPAN METEOROLOGICAL AGENCY, Calculation Method of Measured Seismic Intensity (2017), http://www.data.jma.go.jp/svd/eqev/data/kyoshin/kaisetsu/calc_sindo.htm (in Japanese)

[35] GRÜNTHAL, G., European Macroseismic Scale 1998, Conseil de L'Europe, Cahiers du Centre Européen de Géodynamique et de Séismologie, Luxembourg (1998).

[36] WOOD, H.O., NEUMANN, F., Modified Mercalli Intensity Scale of 1931, Bull. Seis. Soc. Am. **21** 4 (1931) 277–283.

[37] U.N. EDUCATIONAL, SCIENTIFIC AND CULTURAL ORGANIZATION, Consensus Development of Judgmental Terms used in the MSK Intensity Scale Developed at the 1st Meeting of the Seismicity and Seismo-Techtonic Working Meeting Group, UNESCO (1965).

[38] GUIRE, R., MNO-10, Seismic Hazard and Risk Analysis, EERI (2004).

[39] ELECTRIC POWER RESEARCH INSTITUTE, A Criterion for Determining Exceedance of the Operating Basis Earthquake, EPRI NP-5930 (1988).

[40] ELECTRIC POWER RESEARCH INSTITUTE, Guidelines for Nuclear Plant Response to an Earthquake, EPRI Technical Report 3002000720 (2013).

[41] JAPAN NUCLEAR TECHNOLOGY INSTITUTE, Post-Earthquake Equipment Integrity Assessment Guideline - Pre-Earthquake Plan and Post-Earthquake Inspections and Assessments, JANTI-SANE-G1 (2012).

[42] UNITED STATES ATOMIC ENERGY COMMISSION, Nuclear Reactors and Earthquakes, US AECTID 7024 (1963).

[43] ELECTRIC POWER RESEARCH INSTITUTE, Standardization of the Cumulative Absolute Velocity, EPRI TR-100082 (1991).

[44] UNITED STATES GEOLOGICAL SURVEY, Shake Map Manual, Technical Manual, UGSG, Version 1.0, USGS 508TM12-A1 (2006).

[45] KATONA, T.J., Modelling of Fatigue-Type Seismic Damage for Nuclear Power Plants", Open Journal of Safety Science and Technology **2** (2012) 41–46.

[46] UNITED STATES ATOMIC ENERGY COMMISSION, Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Post Earthquake Actions, US NRC Guide 1.166 (2007).

[47] SENIOR SEISMIC REVIEW AND ADVISORY PANEL, Use of Seismic Experiences and Test Data to Show Ruggedness of Equipment in Nuclear Power Plants, Prepared for Seismic Qualification Utility Group and US Nuclear Regulatory Commission (1991).

[48] ELECTRIC POWER RESEARCH INSTITUTE, Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 2 (corrected 14 Feb 1992), EPRI Seismic Qualification Utility Group (1992).

[49] SHYLAMONI, P., Damage Indicating Parameter (Dip) Analysis of Past Indian Earthquakes, Paper presented to WA 4 Meeting, 3–4 July 2012, ISSC EBP, IAEA, Vienna.

[50] STEVENSON, J.D., Survey of Strong Motion Earthquake Effects on Thermal Power Plants in California with Emphasis on Piping Systems, NUREG/CR-6239, Main Report Vol.1, Prepared for US Nuclear Regulatory Commission (1995).

[51] ELECTRIC POWER RESEARCH INSTITUTE, Damage-Indicating Parameter Information, EPRI-SQUG, EPRI-1022682, Technical Update (2011).

[52] KANSAI ELECTRIC POWER CO., Hanshin-Awaji Great Earthquake Disaster Restoration Record (1995) (in Japanese).

[53] VIRGINIA ELECTRIC AND POWER COMPANY, Virginia Electric and Power Company (DOMINION) North Anna Power Station Units 1 and 2, North Anna Independent Spent Fuel Storage Installation, Summary Report of August 23, 2011 Earthquake Response and Restart Readiness Determination Plan, Sep. 17 2011, US NRC ADAMS Acc. No. ML11262A151.

[54] DOMINION, North Anna Power Station Restart Readiness, Oct. 21, 2011 Briefing, Presentation in US NRC Commission Meeting. http://www.nrc.gov/reading-rm/doc-

collections/commission/slides/2011/20111021/dominion-20111021.pdf. ADAMS Acc. No. ML11294A415.

[55] VIRGINIA ELECTRIC AND POWER COMPANY, North Anna Power Station Units 1 and 2, Post-Earthquake Restart Readiness Determination Plan, Status Update", Sep. 27 2011, US NRC ADAMS Acc. No. ML11272A129.

[56] NUCLEAR POWER ENERGY CENTRE, 2003 Status Report on the Seismic Reliability Verification for Nuclear Power Plant Facilities, Part 1, Ultimate Strength of Piping System", NUPEC (2003) (in Japanese).

[57] UNITED STATES NUCLEAR REGULATORY COMMISSION, Seismic Analysis of Large Piping Systems for the JNES-NUPEC Ultimate Strength Piping Test Program, NUREG/CR-6983 (2008).

[58] MINAGAWA, K., et al., Study on Dynamic Strength Evaluation Method of Mechanical Members Based on Energy Balance, J. Press. Vessel. Technol., Transaction of the ASME **131** (2009).

[59] ORITA, S., et al., Verification test for integrity of equipment foundations affected by dynamic load, SMIRT 20 – Division V, Paper 1777 (2008).

[60] KUNUGI, T., et al., A Real Time Processing of Seismic Intensity, Transaction of Seismological Society of Japan, JISIN, **60** 2 (2008) 243–252 (in Japanese).

ANNEX I: EXAMPLES OF EARTHQUAKE DATA COLLECTION SHEETS

It is essential for the earthquake-induced damage study to collect and cumulate earthquake experiences, especially data on damage to structures, systems and components. IAEA EESS/NSNI section extra-budgetary project Working Area 3 team surveyed Member States' experiences at the beginning of their activities in November 2009.

I.1 PURPOSE

The purpose of the survey was to collect experiences and current practices in Member States, in order to explore newly defined DIPs and to better describe actual damage and expand the limits of the DIPs.

I.2. FACILITIES TO BE SURVEYED

Commercial nuclear power plants in operation.

I.3. QUESTIONNAIRES

The survey consisted of four questionnaires as listed below:

- Questionnaire 1: Manual versus automatic seismic shutdown requirements;
- Questionnaire 2: State of the practice on seismic shutdown instrumentation;
- Questionnaire 3: Criteria for plant restart following the seismic shutdown;
- Questionnaire 4: Event data, calculated DIPs, and observed effects for earthquakes affecting nuclear power plants and their facilities.

Results of Questionnaires 1 and 2 are incorporated into the seismic instrumentation system discussion in Section 2. Questionnaire 3 offered no information beyond what had already been discussed in IAEA extra-budgetary project Working Area 3.

In Questionnaire 4, the categorization of seismic structural features of nuclear power plant equipment that are presently used by different institutions was surveyed. As a result of this survey, structures, systems and components were categorized in 52 classes for a boiling water reactor type nuclear power plant case, as shown in Table I-1.

Data collection sheets about observed effects of earthquakes on equipment within these 52 classes were prepared. Typical sheets are presented in this annex. Table I-2 and I-3 show typical sheets for fans and electric transformers.

TABLE I-1. EQUIPMENT CATEGORIES

IAEA WA 3 Category Number	Type of SSC	Final Class Category (IAEA Working Area 3 Project)	Description/Comments
1	Nuclear Plant Generic Items	Fans	Includes the Fan and the Motor Driver
2		Air Compressors	
3		Battery Racks	Includes the Battery and the Support Rack
4		Battery Chargers and Inverters	
5		Air Handlers	
6		Chillers	
7		Transformers	
8		Vertical Pumps	
9		Horizontal Pumps	Horizontal Pumps
			Turbine Driver for Pump
			Reciprocating Pumps
10		Motor Generators	
11		Motor Control Centers	
12		Low Voltage Switchgear	
13		Medium Voltage Switchgear	
14		Distribution Panels	
15		Motor Operated Valves	
16		Air Operated Valves	
17		Engine Generators	Diesel Generators
18		Instrument Racks	
19		Temperature Sensors	
20		Control and Instrumentation Cabinets	Control Panels
21		Low Pressure Storage Tanks	
22		High Pressure Tanks and Heat Exchangers	Includes all heat exchangers, accumulators, filtration demineralizers and pressure retaining tanks.
23		Buried Pipe	
24		Piping	All non-buried piping
25		Cable and Conduit Raceways	
26		HVAC Duct	
27		Damper	
28		Main Turbine	
29		Main Generator	
30		Overhead Cranes	
31		Relays (Includes breakers, switches and contact devices)	
32		Generic Equipment	All other nuclear plant equipment that does not fit the other categories listed.

TABLE I-1. EQUIPMENT CATEGORY (cont.)

IAEA WA 3 Category Number	Type of SSC	Final Class Category (IAEA Working Area 3 Project)	Description/Comments
33	II/I Interaction and Non-Nuclear Structures	Steel Framed Structures	
34		Reinforced Concrete Structures and Masonry Walls	
35		Unreinforced Masonry Walls	
36		Storage Racks	
37		Raised Floors	
38	Safety Related Structures	Reinforced Concrete Structures	Reactor Building, Containment Building, Auxiliary Building, Control Building, Pump House, Emergency Diesel Building, Fuel Building, etc.
39		Steel Framed Structures	Turbine Building
40	BWR Specific Equipment	Control Rod Drive	
41		Internal Pump (Jet Pump)	
42		Fuel Handling Machine	
43		Reactor Pressure Vessel	
44		Reactor Internals	
45		Fuel Rack	
46		Condenser, Feed Water Heater, Moisture Separator & Reheater	
47		Fuel Pool Liner	
48		Primary Containment Vessel	
49		Strainer/Filter	
50		Stream Jet Air Ejector	
51		Dehumidifier	
52		Fuel Assembly	

TABLE I-2. EXAMPLE OF DATA COLLECTION SHEET (FAN)

IAEA EBP on Seismic Safety of Existing NPPS

EARTHQUAKE EFFECT DATA COLLECTION SHEET

Category of Equipment and System	Fans

1. Equipment/System Information

Name and Identified Number		Manufacturer	
Model Type		Model Number	
Building Name		Structure of Building	

Building Height from Foundation Level	
Grade Level	
Foundation level of Building	
Elevation of Equipment/System	

Size	Height		Width		Depth		Weight	

Seismically Designed and, if so, its design base	
Diesel System Component?	

Anchorage/Support			
Anchor Type		Quantity	
Size		Concrete Strength	
Photograph or Sketch of equipment/system including the anchor/support			

2. Facility Information

Name of Facility		Company (Owner)	

(1) Soil Condition

1. Type 1: Vs > 1,100 m/s		2. 1,100 m/s > Vs > 300 m/s		3. 300m/s > Vs		4. Unknown	

Note) Definition above is just beneath the foundation mat.

Vertical Soil Profile	

3. Earthquake or Shaking Test Information

(1) Earthquake or Shaking Test

Event		Date		Moment Magnitude	
Scaling Factor (in case of scale test)					

(2) Effect Data at the nearest point of equipment

Observed data	Value		Observed place	
Time history or Response Spectra, if exists				
Seismic Intensity at site or vicinity				

4. Earthquake Effect

Damaged		Malfunction		Success	
Operability (Continuation of operation)					

Damage		
Damage to fan blades		
Damage to unit anchorage		
Damage to bearing or fretting of bearing		
Damage to shaft seal		
Damage to internal components		
Damage to Motor		
Breaking of attached electrical conduit, cooling water piping or ductwork		
Damage to expansion joint		
Damage caused by falling debris		
Damage caused by soil or foundation settlement		
Others		
Malfunction		
Failure of motor to operate when needed		
Breaking of attached electrical conduit, cooling water piping or ductwork		
Others		
Root cause		
Photograph or Sketch of the Damage		

5. Person filled in the Form

Name		Affiliation		E-mail	
Date filled in the Form					

Category	**Fans**
Earthquake	**0**
Company (Owner)	**0**

Form - Type 01 (Fans)	
Number	

TABLE I-3. EXAMPLE OF DATA COLLECTION SHEET (TRANSFORMER)

IAEA EBP on Seismic Safety of Existing NPPS

EARTHQUAKE EFFECT DATA COLLECTION SHEET

Category of Equipment and System	Transformers

1. Equipment/System Information

Name and Identified Number			Manufacturer		
Model Type			Model Number		
Building Name		Structure of Building			
Building Height from Foundation Level					
Grade Level					
Foundation level of Building					
Elevation of Equipment/System					
Size	Height	Width	Depth	Weight	
Seismically Designed and, if so, its design base					
Diesel System Component?					

Anchorage/Support			
Anchor Type		Quantity	
Size		Concrete Strength	
Photograph or Sketch of equipment/system including the anchor/support			

2. Facility Information

Name of Facility		Company (Owner)	

(1) Soil Condition

1. Type 1: Vs > 1,100 m/s		2. 1,100 m/s > Vs > 300 m/s		3. 300m/s > Vs		4. Unknown	

Note) Definition above is just beneath the foundation mat.

Vertical Soil Profile	

3. Earthquake or Shaking Test Information

(1) Earthquake or Shaking Test

Event		Date		Moment Magnitude	
Scaling Factor (in case of scale test)					

(2) Effect Data at the nearest point of equipment

Observed data	Value		Observed place	
Time history or Response Spectra, if exists				
Seismic Intensity at site or vicinity				

4. Earthquake Effect

Damaged		Malfunction		Success	
Operability (Continuation of operation)					

Damage		
Damage to metal enclosure		
Damage to or failure of equipment anchorage		
Sliding or overturning of equipment		
Damage to internal equipment		
Damage to bushing/silicone cone insulator		
Damage to cooling equipment		
Others		
Malfunction		
Others		
Root cause		
Photograph or Sketch of the Damage		

5. Person filled in the Form

Name		Affiliation		E-mail	
Date filled in the Form					

Category	**Transformers**
Earthquake	**0**
Company (Owner)	**0**

Form - Type 07 (Transformers)	
Number	

ANNEX II: STATUS OF SDAS IN NUCLEAR POWER PLANTS

The IAEA extra-budgetary project team issued a questionnaire soliciting responses from eighteen countries. Responses were received from eleven out of the eighteen countries. The completeness and level of details of the responses, varied considerably. Generally, for countries with multiple nuclear power plant sites, some information was missing for specific plants or units at a specific site. Given these facts, the responses constitute a valid sample of the nuclear power plant populations with respect to existing seismic instrumentation systems.

TABLE II-1 A. SUMMARY COUNTRY/PLANT DATA

	Country	Plant or Other Descriptor	Sites	Units	Notes
1	Armenia	Metsamor	1	2	
2	Finland	Loviisa	1	2	
		Olkiluoto	1	2	Two existing units. One under construction. No data.
3	France	900 MWe	9	34	All standardized structures, systems and components designed to the design basis earthquake of the series. Site specific structures, systems and components, designed to the site design basis earthquake.
		1300 MWe	8	20	
		1450 MWe	2	4	
4	Hungary	Paks	1	4	
5	India				See Table II-1 C
6	Japan				See Table II-1 B
7	Korea	Kori	1	4	Kori units 2-4
		Yonggwang	1	6	
		Ulchin	1	6	Ulchin 5-6 data on seismic instrumentation
		Wolsong	1	4	CANDU reactors. No data.
8	Lithuania				Decommissioned
9	Pakistan	Chasma	1	2	
		Karachi	1	1	CANDU reactor
10	Russian Federation	General	12	31+1?	VVER (16), RBMK (11), FBR (1), LWGR (4)
		Belakova	1	4	
		Beloyarsk	1	1	
		Kalinin	1	3	
		Rostov	1	1?	
		Not listed	8	23	
11	Spain	Almaraz	1	2	In principle, Spain adopts a hierarchy of regulations, with Spanish specific, international criteria and nuclear safety regulations of the country from which the original nuclear technology comes, United States of America and Germany. (In practice, United States criteria, due to its specific and comprehensive nature)
		Asco	1	2	
		Cofrentes	1	1	
		Santa M. Garona	1	1	
		Trillo	1	1	
		Vandellos 2	1	1	

TABLE II-1 B. SUMMARY JAPAN PLANT DATA

Owner	Plant	Sites	Units	Notes
TEPCO	Fukushima Dai-ichi	1	6	
	Fukushima Dai-ni	1	4	
	Kashiwazaki-Kariwa	1	7	
	Higashi-Dori	1	1	
Hokuriku	Shika	1	2	
Chubu	Hamaoka	1	5	
	Hamaoka 1 and 2			Decommissioning
Kansai	Mihama	1	3	
	Takahama	1	4	
	Ohi	1	4	
	Ohi 1 and 2			
	Ohi 3 and 4			
Chugoku	Shimane	1	2	
Shikoku	Ikata	1	3	
Japan Atomic Power Co.	Tokai Dai-ini	1	1	
	Tsuruga	1	2	
Tohoku	Onagawa	1	3	

TABLE II-1 C. SUMMARY INDIA PLANT DATA

Owner	Plant	Sites	Units	Notes
NPCIL	Kaiga	1	6	
	Kaiga-1			PHWR – Operation 2000
	Kaiga-2			PHWR – Operation 2000
	Kaiga-3			PHWR – Operation 2007
	Kaiga-4			PHWR – Under construction
	Kaiga-5, 6			PWR - Planned
	Kalpakkam	1	3	
	Madras-1			PHWR – Operation 1984
	Madras-2			PHWR – Operation 1986
	PFBR			FBR – Under construction
	Kakrapar	1	2	
	KAPS-1			PHWR – Operation 1993
	KAPS-2			PHWR – Operation 1995
	Rajasthan	1	8	
	RAPS-1			PHWR – Operation 1973
	RAPS-2			PHWR – Operation 1981
	RAPS-3			PHWR – Operation 2000
	RAPS-4			PHWR – Operation 2000
	RAPS-5			PHWR – Operation 2010
	RAPS-6			PHWR – Under construction
	RAPS-7, 8			PHWR – Planned
	Tarapur	1	4	
	TAPS-1			BWR – Operation 1969
	TAPS-2			BWR – Operation 1969
	TAPS-3			PHWR – Operation 2006
	TAPS-4			PHWR – Operation 2005
	Narora	1	2	
	NAPS-1			PHWR – Operation 1991
	NAPS-2			PHWR – Operation 1992

TABLE II-2 A. SUMMARY COUNTRY/PLANT DESIGN BASIS EARTHQUAKES

	Country	Plant or Other Descriptor	S2/SSE/DBE (PGA in g)	S2/SSE (Response Spectrum)	S1/OBE (PGA in g)	S1/OBE (Response Spectrum)	Notes
1	Armenia	Metsamor	-	-	-	-	No DBE listed. RLE (PGA) = 0.35g RLE (Ampl.) = 2.15
2	Finland	Loviisa	0.10	Ampl. = 2.3 (10 Hz)	-	-	
		Olkiluoto	-	-	-	-	No data
3	France	900 MWe	0.20 (series) 0.1 or 0.20 (site)	EdF	0.5 × Site SL2	Same shape as SL2	All standardized seismically classified SSCs designed to Series SL2. Site specific seismically classified SSCs designed to Site SL2. In the framework of new periodic safety reviews of EdF NPPs since 2012, the SL1 is redefined as an ‘inspection earthquake' corresponding to a fraction of SL2 set to 0.05 g PGA in the horizontal direction, which is equal or lower than the previous SL1 (1/2 of SL2, according to RFS I.3.b) depending on the site. At the moment of writing (2013), this new SL1 is applicable in three 900 MWe sites.
		1300 MWe	0.15 (series) 0.1 or 0.15 (site)	US-NRC RG 1.60	0.5 × Site SL2	Same shape as SL2	
		1450 MWe	0.15 (series) 0.12 or 0.15 (site)	US-NRC RG 1.60	0.5 × Site SL2	Same shape as SL2	
4	Hungary	Paks	0.25	Derived from UHRS at 10^{-4} yr^{-1}	0.08 or 0.12	Verified against UHRS at 10^{-2} yr^{-1}	OBE derived from OBE exceedance criteria of 0.2 g (2-10 Hz, 5% damping).
5	India	-	-	-	-	-	See Table II-2 C
6	Japan	-	-	-	-	-	See Table II-2 B
7	Korea	Kori	0.20	Ampl.= 3.13 (2.5 Hz)	0.5 x SSE	0.5 × SSE	Kori Units 1-4
		Yonggwang	0.20	Ampl.= 3.13 (2.5 Hz)	0.5 x SSE	0.5 × SSE	
		Ulchin	0.20	Ampl.= 3.13 (2.5 Hz)	0.5 x SSE	0.5 × SSE	Ulchin Units 5-6 data on seismic instrument-tation.
		Wolsong	-	-	-	-	No data
8	Lithuania						Decommissioned

TABLE II-2 A. SUMMARY COUNTRY/PLANT DESIGN BASIS EARTHQUAKES (cont.)

	Country	Plant or Other Descriptor	S2/SSE/DBE (PGA in g)	S2/SSE (Response Spectrum)	S1/OBE (PGA in g)	S1/OBE (Response Spectrum)	Notes
9	Pakistan	Chasma	0.25	US-NRC RG 1.60	0.5 × SSE	0.5 × SSE	
		Karachi	-	-	-	-	No data
10	Russian Federation	General	Basis = 10^{-4} yr^{-1} Confidence 95%	Basis=10^{-4} yr^{-1} Confidence 95%	Basis=10^{-3} yr^{-1}	Basis=10^{-3} yr^{-1}	
		Belakova	-	-	-	-	
		Beloyarsk	-	-	-	-	
		Kalinin	-	-	-	-	
		Rostov	-	-	-	-	
		Not listed	--	-	-	-	
11	Spain	Almaraz	0.10	Newmark	0.05	Newmark	
		Asco	0.13	Modified Newmark	0.07	Modified Newmark	
		Cofrentes	0.17	US-NRC RG 1.60	0.085	US-NRC RG 1.60	
		Santa M. Garona	0.10	US-NRC RG 1.60	0.05	US-NRC RG 1.60	
		Trillo	0.12	US-NRC RG 1.60	0.06	US-NRC RG 1.60	
		Vandellos 2	0.20	US-NRC RG 1.60	0.10	US-NRC RG 1.60	

TABLE II-2 B. SUMMARY JAPAN PLANT DESIGN BASIS EARTHQUAKES

Owner	Plant	S2/SSE (PGA in Gal)	S2/SSE (Response Spectra)	S1/OBE (PGA in Gal)	S1/OBE (Response Spectra	Notes
Hokkaido	Tomari	270 370	-	226	-	
TEPCO	Fukushima Dai-ichi	-	-	-	-	
	Fukushima Dai-ni	-	-	-	-	
	Kashiwazaki-Kariwa					
	KK 1	274 (RB basemat)	Recorded earthquakes			Foundation level (FL): GL-45 m
	KK 2	167 (RB basemat) 450 (rock outcrop)		137 (RB basemat)		Peak accelerations given throughout the height of reactor building (RB). Free field response spectra given at elev. GL-255 m (outcrop bedrock surface). FL= GL-44 m.
	KK 3	193 (RB basemat) 450 (rock outcrop)		151 (RB basemat)		Peak accelerations given throughout the height of the RB. Free field response spectra given at elev. GL-290 m (outcrop bedrock surface). FL=GL-43 m
	KK 4	194 (RB basemat) 450 (rock outcrop)		153 (RB basemat)		Peak accelerations given throughout the height of the RB. Free field response spectra given at elev. GL-290 m (outcrop bedrock surface). FL=GL-43 m
	KK 5	254 (RB basemat) 450 (rock outcrop)		206 (RB basemat)		Peak accelerations given throughout the height of the RB. Free field response spectra given at elev. GL-146 m (outcrop bedrock surface). FL=GL-36 m
	KK 6	263 (RB basemat) 450 (rock outcrop)		195 (RB basemat)		Peak accelerations given throughout the height of the RB. Free field response spectra given at elev. GL-167 m (outcrop bedrock surface). FL=GL-25.7 m
	KK 7	263 (RB basemat) 450 (rock outcrop)		195 (RB basemat)		Peak accelerations given throughout height of RB Free field response spectra given at elev. GL-167 m (outcrop bedrock surface). FL=GL-25.7 m
	Higashi-Dori	-	-	-	-	Instrument locations given

Note: GL=ground level; RB=reactor building; FL=foundation level.

TABLE II-2 B. SUMMARY JAPAN PLANT DESIGN BASIS EARTHQUAKES (cont.)

Owner	Plant	S2/SSE (PGA in Gal)	S2/SSE (Response Spectra)	S1/OBE (PGA in Gal)	S1/OBE (Response Spectra)	Notes
Hokoriku	Shika					
	Shika-1	273 (NS) 256 (EW)	RB basemat (peaks and valleys)	216 (NS) 233(EW)	RB basemat (peaks and valleys)	'Peaks and valleys' = not 'smoothed' spectra
	Shika-2	262 (NS) 332 (EW)	RB basemat (peaks and valleys)	210 (NS) 259 (EW)	RB basemat (peaks and valleys)	
Chubu	Hamaoka					
	Hamaoka 1-2	-	-	-	-	Decommissioning
	Hamaoka 3	608	Peaks and valleys	441	Peaks and valleys	Location not specified. Ground response spectra seem to be from recorded earthquakes or calculated site response (i.e. not 'smoothed' spectra)
	Hamaoka 4	566	Peaks and valleys	438	Peaks and valleys	
	Hamaoka 5	583	Peaks and valleys	445	Peaks and valleys	
Kansai	Mihama					
	Mihama 1-2	400	Ampl. = 2.9 (5-10 Hz, 5% damping)	300	Ampl. = 2.7 (5-10 Hz, 5% damping)	
	Mihama 3	405	Ampl. = 3.1 (5-10 Hz, 5% damping)	270	Ampl. = 3.0 (5-10 Hz, 5% damping)	
	Takahama					
	Takahama 1-2	360	Ampl. = 3.1 (5-10 Hz, 5% damping)	270	Ampl. = 3.0 (5-10 Hz, 5% damping)	
	Takahama 3	360	US-NRC RG 1.60	270	Ampl. = 3.0 (3-8 Hz, 5% damping)	
	Takahama 4	370	US-NRC RG 1.60	270	Ampl. = 3.0 (3-8 Hz, 5% damping)	
	Ohi					
	Ohi 1-2	405	Ampl. = 3.1 (5-10 Hz, 5% damping)	270	Ampl. = 3.0 (5-10 Hz, 5% damping)	
	Ohi 3-4	405	Ampl. = 3.2 (5-10 Hz, 5% damping)	270	Ampl. = 3.0 (3-10 Hz, 5% damping)	
Chugoku	Shimane	456	S2-D1 and S2-D2 Ampl. = 2.6 (5% damp.)	320	Ampl. = 3.1 (3-10 Hz, 5% damping)	
Shikoku	Ikata					
	Ikata 1-2	-	-	-	-	No data
	Ikata 3	Basemat 538 (0.549 g)	Two-peak spectra (5-10 Hz and 20 Hz)	Basemat 250 (0.255 g)	Two-peak spectra (5-10 Hz and 20 Hz)	Basemat response spectra (5% damping)
Japan Atomic Power Co.	Tokai Dai-ni	-	-	-	-	No data
	Tsuruga	532	Ampl. = 2.65 (3-10 Hz, 5% damping)	365	Ampl. = 2.75 (3-8 Hz, 5% damping)	

TABLE II-2 B. SUMMARY JAPAN PLANT DESIGN BASIS EARTHQUAKES (cont.)

Owner	**Plant**	**S2/SSE (PGA in Gal)**	**S2/SSE (Response Spectra)**	**S1/OBE (PGA in Gal)**	**S1/OBE (Response Spectra)**	**Notes**
Tohoku	Onagawa					
	Onagawa 1	-	-	278	-	Reactor building basemat
	Onagawa 2	363	-	265	-	
	Onagawa 3	375	-	260	-	

TABLE II-2 C. SUMMARY INDIA PLANT DESIGN BASIS EARTHQUAKES

Owner	Plant	SL2/SSE/DBE (PGA in g)	SL2/SSE (Response Spectra)	SL1/OBE (PGA in g)	SL1/OBE (Response Spectra)	Notes
NPCIL	Kaiga					
	Kaiga-1 Kaiga-2	0.20	Smoothly increasing to Sa(7 Hz)=0.65 g Smoothly decreasing to 0.20 g at 30 Hz	0.5 × SSE	0.5 × SSE	
	Kaiga-3	-	-	-	-	No data
	Kaiga-4	-	-	-	-	No data
	Kaiga-5, 6	-	-	-	-	No data
	Kalpakkam					
	Madras-1 Madras-2	0.221	Smoothly increasing to Sa(5-20 Hz)= 0.4 g Smoothly decreasing to 0.221 g	-	-	OBE not defined
	PFBR	-	-	-	-	No data
	Kakrapar					
	KAPS-1 KAPS-2	0.20	Smoothly increasing to Sa(5 Hz)=0.6+ g Smoothly decreasing to 0.20 g at 50 Hz	0.5 × SSE	0.5 × SSE	
	Rajasthan					
	RAPS-1	-	-	-	-	No data
	RAPS-2	-	-	--	-	No data
	RAPS-3 RAPS-4	0.10	Smoothly increasing to Sa(5-9 Hz)= 0.3+g Smoothly decreasing to 0.10 g at 35 Hz	0.5 × SSE	0.5 × SSE	
	RAPS-5	-	-	-	-	No data
	RAPS-6	-	-	-	-	No data
	RAPS-7, 8	-	-	-	-	No data
	Tarapur					
	TAPS-1	-	-	-	-	No data
	TAPS-2	-	-	-	-	No data
	TAPS-3 TAPS-4	0.20	Smoothly increasing to Sa(5 Hz)= 0.6+g Smoothly decreasing to 0.20 g at 35 Hz	0.5 × SSE	0.5 × SSE	
	Narora					
	NAPS-1 NAPS-2	0.30	Smoothly increasing to Sa(3-4 Hz)= 0.7g Smoothly decreasing to 0.30 g at 35 Hz	0.5 × SSE	0.5 × SSE	

TABLE II-3 A. SUMMARY PLANT DATA – FREE-FIELD INSTRUMENTS

	Country	Plant or Other Descriptor	Unit	Free Field Instruments	Notes
1	Armenia	Metsamor	2	4	1 on ground surface + 3 in a borehole
2	Finland	Loviisa	1	3	Instruments in place during 1985-1995 Maximum recorded acceleration = 0.015 g
3	France	900 MWe	-	1 or 2	One triaxial accelerometer at free field for homogeneous sites (sites with homogeneous geological and mechanical soil properties and regular topography). For heterogeneous sites, an additional triaxial accelerometer in free field is installed in an area of geological and mechanical characteristics or topography different from that which is already instrumented
		1300 MWe	-	1 or 2	
		1450 MWe	-	1 or 2	
4	Hungary	Paks	1-2	1	Accelerometers / Peak acceleration
			3-4	0	
5	India				See Table II-3 C
6	Japan				See Table II-3 B
7	Korea	Kori	-	-	
		Yonggwang	-	-	
		Ulchin 1-4	-	-	
		Ulchin 5-6	5-6	1	
		Wolsong	-	-	CANDU reactors. No data.
8	Lithuania				Decommissioned
9	Pakistan	Chasma	1	1	
		Karachi	-	-	
10	Russian Federation				Project specific
		Belakova			
		Beloyarsk			
		Kalinin			
		Rostov			
11	Spain	Almaraz	1	1	
			2	0	
		Asco	1	0	
			2	1	
		Cofrentes	1	1	
		Santa M. Garona	1	1	
		Trillo	1	1	
		Vandellos 2	2	1	

TABLE II-3 B. SUMMARY JAPAN PLANT DATA – FREE-FIELD INSTRUMENTS

Owner	Plant	Unit	Free Field Instruments	Notes
Hokkaido	Tomari	1	3	+10.0, -90.0, -250.0 m
		2		
		3	4	+56.0, +2.3, -90.0, -250.0 m
TEPCO	Fukushima Dai-ichi	1, 2, 3, 4, 5	5	+32.3, -5.0, -100.0, -200.0, -300.0 m
		6	2	+13.5, -4.0 m
	Fukushima Dai-ni	1	2	+4.0, -5.5 m
		2		
		3	4	+10.2, -5.5, -50.0, -200.0 m
		4		
	Kashiwazaki-Kariwa	SH	5	+65.1, +16.3, -31.9, -182.0, -333.0 m
		1	4	+5.0, -40.0, -122.0, -400.0 m
		2		
		3		
		4		
		5	6	+12.3, +9.3, -24.0, -100, -180, -300m
		6		
		7		
	Higashi-Dori	1	4	+17.2, +7.2, -82.8, -282.8 m
Hokuriku	Shika	1	4	+19.0, -10.0, -100.0, -200.0 m
		2		
Chubu	Hamaoka	1	1	-2.0 m
		2		
		3	4	-2.0, -25.0, -40.0, -100.0 m
		4	2	-20.0, -100.0 m
		5	3	-2.0, -22.0, -100.0 m
Kansai	Mihama	1	3	-5.0, -15.0, -30.0 m
		2		
		3		
	Takahama	1	2	+1.5, -30.0 m
		2		
		3		
		4		
	Ohi	1	3	+4.0, -24.0, -94.0 m
		2		
		3		
		4		
Chugoku	Shimane	1	12	Not given
		2		
Shikoku	Ikata	1	4	+10.0, -5.0, -80.0, -100.0 m
		2		
		3		
Japan Atomic Power Co	Tokai Dai-ini	2	4	+8.0, -17.0, 192.0, -372.0 m
	Tsuruga	1	3	-17.0, -50.0, -100.0 m
		2		
Tohoku	Onagawa	1	4	-1.7, -27.3, -61.5, -147.1 m
		2		
		3		

TABLE II-3 C. SUMMARY INDIA PLANT DATA – FREE-FIELD INSTRUMENTS

Owner	Plant	Number of Units	Free Field Instruments (TH + PAR + RSR)
NPCIL	Kaiga	6	3 + 0 + 0
	Kalpakkam	3	1 + 0 + 0
	Kakrapar	2	0 + 3 + 0
	Rajasthan	8	1 + 3 + 0
	Tarapur	4	No data
	Narora	2	3 ?

Note: TH=time history recorder; PAR=peak acceleration recorder; RSR=response spectrum recorder

TABLE II-4 A. SUMMARY PLANT DATA – IN STRUCTURE INSTRUMENTS

	Country / Owner	Plant or Other Descriptor	Unit	Buildings TH + PAR + RSR	Notes
1	**Armenia**	Metsamor NPP	1-2	14	Accelerometers and velocity graph recorders
2	**Finland**	Lovissa NPP	1	0	Instruments in place 1985 – 1995, max. = 0.015g
3	**France**	900 MWe		3 or 4 + 4 + 0	900 MWe: For homogeneous sites, Unit 1 is instrumented with: • 1 triaxial accelerometer at the basemat of the reactor building • 1 triaxial accelerometer and 1 PAR at the service floor of the nuclear reactor building • 1 triaxial accelerometer and 1 PAR at the basemat of the nuclear auxiliary building • 1 PAR at the control room • 1 PAR at the top of containment shell For heterogeneous sites, the following devices are added: • 1 triaxial accelerometer at the basemat of the reactor building of each unit 1300 and 1450 MWe: same as 900 MWe + one additional PAR at the top of the shield building
		1300 MWe		3 or 4 + 5 + 0	
		1450 MWe		3 or 4 + 5 + 0	
4	**Hungary**	Paks NPP	1-2	6 + 6 + 0	Accelerometers/Peak acceleration
			3-4	6 + 6 + 0	
5	**India**				See Table II-4 C
6	**Japan**				See Table II-4 B
7	**Korea**	Kori	-	-	
		Yonggwang	-	-	
		Ulchin 1-4	-	-	
		Ulchin 5/6	5/6	7 + (4+2) + 0	Assume in Units 5 and 6 RB/AB/Tank + 2 seismic switches – total for Ulchin site is 1 + 2 x (7 + 6) = 27
		Wolsong	-	-	CANDU reactors – no data
8	**Lithuania**	-	-	-	Decommissioned
9	**Pakistan**	Chasma	1	4 + 0 + 0	Containment Building (basemat, ring girder, oper. floor), ESWP
		Karachi	-	-	CANDU reactor
10	**Russian Federation**				
		Balakova		1 + ?	Unclear
		Beloyarsk		1 + ?	
		Kalinin		1 + ?	
		Rostov		1 + ?	
11	**Spain**	Almaraz	1	5 + 0 + 0	Containment Building (3), Auxiliary Building (2)
			2	0	None
		Asco	1	0	None
			2	5 + 0 + 0	Containment Building (3), Control Building (2)
		Cofrentes	1	5 + 0 + 0	Containment Building (3), Service Building (1), Control Room (1)
		Santa M de Garona	1	5 + 0 + 0	Containment Building (3), Service Building (1), Turbine Building (1)
		Trillo	1	5 + 0 + 0	Containment Building (3), Electric Building (2)
		Vandellos 2	1	5 + 0 + 0	Containment Building (3), Control Building (2)

Note: TH=time history recorder; PAR=peak acceleration recorder; RSR=response spectrum recorder; AB=auxiliary building

TABLE II-4 B. SUMMARY JAPAN PLANT DATA – IN STRUCTURE INSTRUMENTS

Owner	Plant	Unit	Buildings Structures	Notes
Hokkaido	Tomari	1	25	(BF, O/S, C/V, I/C, E/B)
		2	18	(BF, O/S, C/V, I/C, E/B)
		3	18	(BF, O/S, C/V, I/C, E/B)
TEPCO	Fukushima Daiichi	1,2,3,4,5	5	(RB, TB)
		6	10+2	(RB, TB)
	Fukushima Dai-ni	1	11+3	(RB, TB)
		2	3+5	(RB, TB)
		3	5	(RB, TB)
		4	5	(RB, TB)
	Kashiwazaki-Kariwa	SH		
		1	9+3	(RB, TB)
		2	5	(RB, TB)
		3	5	(RB, TB)
		4	5	(RB, TB)
		5	3	(RB, TB)
		6	2	(RB)
		7	5	(RB, TB)
	Higashi-Dori	1		
Hokuriku	Shika	1	22	(RB, TB)
		2	21	(RB, TB)
Chubu	Hamaoka	1 (Decom)	4	(RB)
		2 (Decom)	7	(RB, TB)
		3	25	(Stack, Truss Tower, RB, TB, AB)
		4	15	(Stack, Truss Tower, RB)
		5	17	(Stack, Truss Tower, RB, TB)
Kansai	Mihama	1	8+4	(H + V) (RB)
		2	8+4	(H + V) (RB)
		3	10+3	(H + V) (RB)
	Takahama	1	8+4	(H + V) (RB)
		2		
		3	8+3	(H + V) (RB)
		4	26+7	(H + V) (RB)
	Ohi	1	26+6	(H + V) (RB)
		2		
		3	8+4	(H + V) (RB)
		4	22+9	(H + V) (RB)
Chugoku	Shimane	1	10	(RB)
		2	18	(RB)
Shikoku	Ikata	1	1	(AB – basement)
		2	1	(AB – basement)
		3	1	(AB – basement)
Japan Atomic Power Co.	Tokai Daini	2	6	(RB, TB, CST)
	Tsuruga	1	4	(RB)
		2	10	(RB, AB)

Note: RB=reactor building; TB=turbine building; AB=Auxiliary building; CST=condensate storage tank; BF=basement floor; C/V=containment vessel; I/C=containment inner (backfill) base concrete; EB=environmental building; O/S=outer shield

TABLE II-4 B. SUMMARY JAPAN PLANT DATA – IN STRUCTURE INSTRUMENTS (cont.)

Owner	**Plant**	**Unit**	**Buildings** **Structures**	**Notes**
Tohoku	Onagawa	1		(RB, TB, CB, RWB)
		2		(RB, TB, CB)
		3		(RB, TB, SB, HxB)

Note: RB=reactor building; TB=turbine building; AB=Auxiliary building; CST=condensate storage tank; BF=basement floor; C/V=containment vessel; I/C=containment inner (backfill) concrete; O/S=outer shield; CB=control building; RWB=radwaste building; SB=service building; HxB=heat exchangers building.

TABLE II-4 C. SUMMARY INDIA PLANT DATA – IN STRUCTURE INSTRUMENTS

Owner	Plant	Unit	Buildings
			TH + PAR + RSR
NPCIL	Kaiga		
	Kaiga-1	1	2
	Kaiga-2	2	0
	Kaiga-3	3	Not reported
	Kaiga-4	4	Not reported
	Kaiga-5-6	5-6	Not reported
	Kalpakkam		
	Madras -1 (MAPS-1)	1	0
	Madras- 2 (MAPS-2)	2	2
	PFBR		Not reported
	Kakrapar		
	KAPS-1	1	2 + 1 + 1
	KAPS-2	2	0 + 1+ 0
	Rajasthan		
	RAPS-1	1	Not reported
	RAPS-2	2	Not reported
	RAPS-3	3	Reactor building, 3 + 1 + 0
	RAPS-4	4	Reactor building, 1 + 1 + 0
	RAPS-5	5	Not reported
	RAPS-6	6	Not reported
	RAPS- 7-8	7-8	Not reported
	Tarapur		
	TAPS-1	1	Not reported
	TAPS-2	2	Not reported
	TAPS-3	3	Not reported
	TAPS-4	4	Not reported
	Narora		
	NAPS-1	1	2
	NAPS-2	2	0

Note: TH=time history recorder; PAR=peak acceleration recorder; RSR=response spectrum recorder

TABLE II-5. TYPES OF SEISMIC INSTRUMENTATION IN CHINA

Plant	Type	Purpose	Distribution and Number of Instruments	Logic	Characteristics
Tianwan	Accelerometer	Structural Analysis	Number: 8 Containment foundation (4); Containment building (22.5m) (4)	2 out of 4	Frequency range 0~50 Hz; Dynamic range 96 dB; Temperature: -25°C~85°C; Recording time: 15 m/Mb; Power supply: 12 V DC; Measurement range: ±1g
	Seismic switch	Alarm and automatic trip	Number: 8 Distribution is same as above	2 out of 4	Frequency range 0~50 Hz; Dynamic range 96 dB; Temperature: 10°C~40°C; Storage capacity: 4 Mb; Recording time: 15 m/Mb; Power supply: emergency power supply 12V DC; Measurement range: ±1g
	Additional accelerometer	Used during Maintenance	Number: 6 No. 1 safety train (2); No. 2 safety train (2); No. 3 safety train; No. 4 safety train		Frequency range 0~50 Hz; Dynamic range 96 dB; Temperature: -25°C~85°C; Recording time: 15 m/Mb; Power supply: emergency power supply: 12 V DC; Measurement range: ±1g
	Additional seismic switch	Alarm and automatic trip	Number:6 Distribution is same as above	2 out of 4	Frequency range 0~50 Hz; Dynamic range 96 dB; Temperature: 10°C~40°C; Recording time: 15 m/Mb; Power supply: emergency power supply: 12 V DC; Measurement range: ±1g
	Peak Accelerograph	Recording acceleration	Number: 4 Reactor equipment in containment; No: 1 steam chamber of safety train		
	Seismic control		4		
Qinshan	Seismic switch	Alarm	Number: 4, 2-axis switch	2 out of 4	
	Accelerometer	Structural analysis	Number: 5, 3-axis		Frequency: 0.1 (±0.01)~33 (±1.5) Hz; Measure scope ±1.04g ±0.001g; Dynamic scope: ±1.04g ±0.001g
	Recording and storage instrument	Structural analysis			Dynamic scope: ±1.04g ±0.008g; Sampling speed: 200 d/s; Duration: 2 of 81 s
	Peak Accelerograph	Structural analysis			Frequency: 0.1 (±0.01) ~33 (±1.5) Hz; Measure scope ±1.04g ±0.001g; Dynamic scope: ±1.04g ±0.001g
Qinshan 3rd	Seismic control panel	Alarm	Number: 1		N/A
	3-axis Accelerometer	Structural analysis	Number: 6	2 out of 3	Measure range: 1g
	3-axis Accelerometer	Structural analysis	Number: 4		Measure range: 2g

TABLE II-5. TYPES OF SEISMIC INSTRUMENTATION IN CHINA (cont.)

Plant	Type	Purpose	Distribution and Number of Instruments	Logic	Characteristics
Daya Bay	Seismic control panel	Alarm	Number: 1	2 out of 3	Measure range: ±1g; Statoscope: 2.5 V/g; Frequency: 50 Hz
	3-axis Accelerometer	Structural analysis	Number: 6, Two of them are with switch alarming function	2 out of 3	Frequency: 4 Hz; Acceleration scope: 0.005g~0.05g; Temperature: 20°C~55°C
	3-axis peak accelerograph	Structural analysis	Number: 4		
Lingao	3-axis Accelerometer	Structural analysis	Number: 6 Free field (2); Unit 1 containment foundation; Unit 1 containment structure (20m); NAB corridor; Unit 2 containment foundation	2 out of 3	Full scale range ±1.0g; Sensitivity: 2.5 V/g; Natural frequency: 50 Hz; Bandwidth: DC to 50 Hz (3 dB point)
	Seismic switch	Alarm	2 in 6 sensors are with switch Unit 1 containment foundation; NAB corridor	2 out of 3	
	Peak accelerograph	Structural analysis	Number: 4 Containment structure (8 m); Containment structure (girder); NAB foundation; Electrical building		Full scale range ±2.0g; Dynamic range: 200:1 (46 dB); Natural frequency (±5%): 32 Hz; Damping: 55 to 70% of critical; Bandwidth: 0 to 26 Hz
	KIS computer, printer and signal	Structural analysis	Number: 4		
Qinshan Phase 2	3-axis Accelerometer sensor	Structural analysis	Number: 5 Free field, Unit 1 containment foundation, Unit 1 containment structure (20m); NAB corridor; Unit 2 containment foundation	2 out of 3	Frequency: 40±0.5 Hz; Damping degree: 0.65±0.02; Power constant: >30 V/cm/s; Liner degree better than 2% full extent; Lateral statoscope: <2% full extent.
	Seismic switch	Alarm	Number: 3 On containment foundation		Frequency: 40±0.5 Hz; Damping degree: 0.65±0.02; Power constant: >30 V/cm/s; Liner degree better than 2% full extent; Lateral statoscope: <2% full extent.
	Peak accelerograph	Structural analysis	Number: 4		Frequency: 4 Hz; Acceleration measure scope: ±5g; Frequency: 51 Hz±5%; Damping: 55%-70%; Bandwidth: 0~51Hz; Precision: ±5% full scope.
	KIS computer, printer and signal	Structural analysis	Number: 4		

Note: NAB=nuclear auxiliary building.

ANNEX III: STATUS OF AUTOMATIC SEISMIC TRIP SYSTEMS IN NUCLEAR POWER PLANTS

The IAEA issued a questionnaire soliciting responses from eighteen countries on the subject of seismic instrumentation systems (country requirements, types of instruments in place, location of instruments, maintenance and operability, etc.). In this questionnaire, countries were asked about their requirements, implementation and operability experience regarding ASTS. Responses were received from twelve out of eighteen countries. In general, the responses showed that plants in high seismicity areas are more likely to have ASTSs than those located in low to moderate seismicity areas. Four countries have nuclear power plants with ASTSs: Japan, India, the United States of America and the Russian Federation.

TABLE III-1 A. SUMMARY JAPAN PLANT DATA – ASTS

Owner	Plant	S2/SSE/DBE (PGA)	S2/SSE (Response Spectra)	S1/OBE (PGA)	S1/OBE (Response Spectra)	ASTS
Hokkaido	Tomari	270 Gal 370 Gal		226 Gal		ASTS Trigger H: 180 Gal; V: 90 Gal on basemat H: 340 Gal in-structures
TEPCO	Fukushima Daiichi	-	-	-	-	No data
	Fukushima Dai-ni	-	-	-	-	No data
	Kashiwazaki-Kariwa					
	KK1	274 Gal (RB Basemat)	Recorded Earthquakes			ASTS Trigger H: 120 Gal; V: 100 Gal on basemat H: 185 Gal in-structures
	KK2	167 Gal (RB Basemat) 450 Gal (rock outcrop)		137 Gal (RB Basemat)		Same
	KK3	193 Gal (RB Basemat) 450 Gal (rock outcrop)		151 Gal (RB Basemat)		Same
	KK4	194 Gal (RB Basemat) 450 Gal (rock outcrop)		153 Gal (RB Basemat)		Same
	KK5	254 Gal (RB Basemat) 450 Gal (rock outcrop)		206 Gal (RB Basemat)		Same
	KK6	263 Gal (RB Basemat) 450 Gal (rock outcrop)		195 Gal (RB Basemat)		Same
	KK7	263 Gal (RB Basemat) 450 Gal (rock outcrop)		195 Gal (RB Basemat)		Same
	Higashi-Dori	-	-	-	-	
Hokuriku	Shika					
	Shika-1	273(NS)/256(EW)	RB basemat (peaks and valleys)	216(NS)/233(EW)	RB basemat (peaks and valleys)	ASTS Trigger H: 190 Gal V: 165 Gal; basemat; H: 505 Gal (RB 28.3m)
	Shika-2	262(NS)/332(EW)	RB basemat (peaks and valleys)	210(NS)/259(EW)	RB basemat (peaks and valleys)	ASTS Trigger H: 185 Gal V: 165 Gal; basemat; H: 505 Gal (RB 32.5m)
Chubu	Hamaoka					
	Hamaoka 1, 2					
	Hamaoka 3	608 Gal	Peaks and valleys	441 Gal	Peaks and valleys	ASTS Trigger H: 120 Gal V: 100 Gal; basemat; H: 230 Gal in-structures
	Hamaoka 4	566 Gal	Peaks and valleys	438 Gal	Peaks and valleys	Same
	Hamaoka 5	583 Gal	Peaks and valleys	445 Gal	Peaks and valleys	Same

Note: RB=reactor building; GL=ground level.

TABLE III-1 A. SUMMARY JAPAN PLANT DATA – ASTS (cont.)

Owner	Plant	S2/SSE/DBE (PGA)	S2/SSE (Response Spectra)	S1/OBE (PGA)	S1/OBE (Response Spectra)	ASTS
Kansai	Mihama					
	Mihama 1, 2	400 Gal	Ampl. = 2.9 (assume 5% damping) 5 – 10 Hz	300 Gal	Ampl. = 2.7 (assume 5% damping) 5 – 10 Hz	ASTS Trigger H: 160 Gal V: 80 Gal basemat
	Mihama 3	405 Gal	Ampl. = 3.1 (assume 5% damping) 5 – 10 Hz	270 Gal	Ampl. = 3 (assume 5% damping) 5 – 10 Hz	Same
	Takahama					
	Takahama 1, 2	360 Gal	Ampl. = 3.1 (damping?) 5 – 10 Hz	270 Gal	Ampl. = 3.0 (damping?) 5 – 10 Hz	ASTS Trigger H: 160 Gal V: 80 Gal basemat
	Takahama-3	360 Gal	Like US NRC RG 1.60	270 Gal	Ampl. = 3.0 (damping?) 3 – 8 Hz	Same
	Takahama 4	370 Gal	Like US NRC RG 1.60	270 Gal	Ampl. = 3.0 (damping?) 3 – 8 Hz	Same
	Ohi					
	Ohi 1, 2	405 Gal	Ampl. = 3.1 (assume 5% damping) 5 – 10 Hz	270 Gal	Ampl. = 3.0 (assume 5% damping) 5 – 10 Hz	ASTS Trigger H: 160 Gal V: 80 Gal basemat
	Ohi 3, 4	405 Gal	Ampl. = 3.2 (assume 5% damping) 5 – 10 Hz	270 Gal	Ampl. = 3.0 (assume 5% damping) 3 – 10 Hz	Same
Chugoku	Shimane 1	456 Gal	S2-D1 and S2-D2 Ampl. = 2.6 (assume 5% damping)	320 Gal	Ampl. = 3.1 (assume 5% damping) 3 – 10 Hz	ASTS Trigger H: 140 Gal V: 70 Gal basemat
	Shimane 2					ASTS Trigger H: 140 Gal V: 70 Gal basemat; 350 Gal in-structure
Shikoku	Ikata					
	Ikata 1, 2	-	-	-	-	Unit 1: ASTS Trigger H: 140 Gal V: 70 Gal; basemat; Unit 2: ASTS Trigger H: 180 Gal V: 90 Gal basemat
	Ikata 3	Basemat 0.549g	Two peaked spectra 5–10 Hz and 20 Hz.	Basemat 0.255g	Two peaked spectra 5 – 10 Hz and 20 Hz.	ASTS Trigger H: 190 Gal V: 90 Gal; basemat
Japan Atomic Power Co.	Tokai Daini 2	FF – 380 Gal and 270 Gal	Peak in 1 - 3 Hz range	180 Gal	Peak in 1 - 3 Hz range	Two EQ motions for S2; Unit 2: ASTS Trigger H: 250 Gal V: 120 Gal; basemat; H: 300 Gal (RB 14 m)
	Tsuruga 1, 2	532 Gal	Ampl. = 2.65 (assume 5% damping) (3 – 10 Hz.)	365 Gal	Ampl. = 2.75 (assume 5% damping) (3 – 8 Hz.)	Unit 1: ASTS Trigger H: 160 Gal V: 80 Gal; basemat; H: 300 Gal; (RB 3.2m) Unit 2: ASTS Trigger H: 160 Gal V: 80 Gal basemat; H: 500 Gal (RB 7.3m)

Note: RB=reactor building; GL=ground level.

TABLE III-1 A. SUMMARY JAPAN PLANT DATA – ASTS (cont.)

Owner	Plant	S2/SSE/DBE (PGA)	S2/SSE (Response Spectra)	S1/OBE (PGA)	S1/OBE (Response Spectra)	ASTS
Tohoku	Onagawa					
	Onagawa 1	-	-	278	-	ASTS Trigger H: 200 Gal V: 100 Gal basemat; H: 200 Gal (RB GL+8.7 m)
	Onagawa 2	363	-	265	-	ASTS Trigger H: 200 Gal V: 100 Gal basemat; H: 400 Gal (RB GL-8.8 m)
	Onagawa 3	375	-	260	-	ASTS Trigger H: 200 Gal V: 100 Gal basemat; H: 350 Gal (RB GL-8.8 m)

Note: RB=reactor building; GL=ground level.

TABLE III-1 B. SUMMARY JAPAN PLANT - ASTS PARAMETERS

Owner	**Plant**	**Ratio of Trigger level / (SL1/OBE/S1)** **Free-field or basemat**	**Ratio of Trigger level / (SL1/OBE/S1)** **In-structure**	**Ratio of Trigger level / (SL2/SSE/S2)** **Free-field or basemat**	**Ratio of Trigger level / (SL2/SSE/S2)** **In-structure**	**Notes**
Hokkaido	Tomari	0.80	No data	0.49	No data	Two SSE earthquake response spectra – different shapes – PGA 270 and 370 Gal
TEPCO	Fukushima Daiichi					
	Fukushima Dai-ni					
	Kashiwazaki-Kariwa					
	KK 1	No S1	No S1	0.44	0.40	
	KK 2	0.88	0.89	0.72	0.68	Peak accels. given throughout the height of the Reactor Building (RB). Free-field response spectra given at elev. GL-255 m (outcrop bedrock surface)
	KK 3	0.79	0.80	0.62	0.59	Peak accels. given throughout the height of the RB. Free-field response spectra given at elev. GL-290 m (outcrop bedrock surface)
	KK 4	0.78	0.84	0.62	0.62	
	KK 5	0.58	0.70	0.47	0.52	Peak accels. given throughout the height of the RB. Free-field response spectra given at elev. GL-146 m (outcrop bedrock surface)
	KK 6	0.62	0.59	0.46	0.45	Peak accels. given throughout the height of the RB. Free-field response spectra given at elev. GL-167 m (outcrop bedrock surface)
	KK 7	0.62	0.59	0.46	0.45	
	Higashi-Dori					Instrument locations given
Hokuriku	Shika					
	Shika-1	0.85	No data	0.72	No data	
	Shika-2	0.79	No data	0.62	No data	
Chubu	Hamaoka					
	Hamaoka 1, 2					Decommissioning
	Hamaoka 3	0.27	No data	0.20	No data	Need location – Ground response spectra appear to be from recorded earthqs or calculated site response
	Hamaoka 4	0.27	No data	0.21	No data	Need location – Ground response spectra appear to be from recorded eartqs or calculated site response
	Hamaoka 5	0.27	No data	0.21	No data	Need location – Ground response spectra appear to be from recorded earthqs or calculated site response

Note: GL=ground level

TABLE III-1 B. SUMMARY JAPAN PLANTS - ASTS PARAMETERS (cont.)

Owner	**Plant**	**Ratio of Trigger level / (SL1/OBE/S1)** **Free-field or basemat**	**Ratio of Trigger level / (SL1/OBE/S1)** **In-structure**	**Ratio of Trigger level / (SL2/SSE/S2)** **Free-field or basemat**	**Ratio of Trigger level / (SL2/SSE/S2)** **In-structure**	**Notes**
Kansai	Mihama					
	Mihama 1, 2	0.53	No data	0.40	No data	
	Mihama 3	0.59	No data	0.40	No data	
	Takahama					
	Takahama 1, 2	0.59	No data	0.44	No data	
	Takahama 3	0.59	No data	0.44	No data	
	Takahama 4	0.59	No data	0.43	No data	
	Ohi					
	Ohi 1, 2	0.59	No data	0.40	No data	
	Ohi 3, 4	0.59	No data	0.40	No data	
Chugoku	Shimane 1	FF vs. Basemat	No data	FF vs. Basemat	No data	
	Shimane 2	FF vs. Basemat	No data	FF vs. Basemat	No data	
Shikoku	Ikata					
	Ikata 1, 2	No data	No data	No data	No data	No data
	Ikata 3	0.76	No data	0.35	No data	Basemat response spectra (5% damping)
Japan Atomic	Tokai Daini 2	FF vs. Basemat	No data	FF vs. Basemat	No data	No data
	Tsuruga 1, 2	0.44	No data	0.30	No data	

TABLE III-2 A. SUMMARY INDIA PLANTS - ASTS PARAMETERS

Owner	Plant	S2/SSE/DBE (PGA)	S2/SSE (Response Spectra)	S1/OBE (PGA)	S1/OBE (Response Spectra)	ASTS
NPCIL	Kakrapar					
	KAPS 1	0.2g	Smoothly increasing to Sa(5 Hz) = (0.6+)g – smoothly decreasing to 0.2g at 50 Hz	0.5×0.2g=0.1g	0.5×SSE	Triggers at PGA greater than OBE PGA (0.1g)
	KAPS 2	Assume same	Assume same	Assume same	Assume same	Triggers at PGA greater than OBE PGA (0.1g)
	Narora					
	NAPS 1	0.3g	Smoothly increasing to Sa(3 - 4 Hz) = 0.7g – smoothly decreasing to 0.3g at 35 Hz	0.5×0.3g=0.15g	0.5×SSE	Triggers at PGA = 0.1g OBE PGA = 0.15g
	NAPS 2	Same				

TABLE III-2 B. SUMMARY UNITED STATES PLANTS - ASTS PARAMETERS

Owner	Plant	S2/SSE/DBE (PGA)	S2/SSE (Response Spectra)	S1/OBE (PGA)	S1/OBE (Response Spectra)	ASTS	Notes
PG&E	Diablo Canyon						
	DCPS 1	DBE (PGA = 0.2g) DDBE (PGA = 0.4g) Hosgri (PGA = 0.75g)	Varying shapes	Assume PGA = 0.1g	0.5×DBE ground response spectra	Containment Bldg. basemat – Three PAR (2 of 3 to exceed trigger level to scram reactor); set at 0.35g ZPA	Several Design Earthquakes are considered: DBE (PGA=0.2g) Double DBE (PGA=0.4g) Hosgri (PGA=0.75g)
	DCPS 2	Same					
SCE	San Onofre						
	SONGS 2	PGA = 0.67g	Amplified frequency range Sa = 1.55 g over 1 to 5 Hz smoothly decreasing for frequencies greater than 5 Hz.	PGA 0.335g	0.5×SSE	Four accelerometers installed on Containment Bldg. basemat at 90 degs. – (2 exceed triggers set level to scram reactor) – Setpoints (H: 0.48g; V: 0.60 g)?	All seismic instrumentation is in Unit 2
	SONGS 3						No instrumentation in Unit 3 – All in Unit 2

Note: PAR=peak acceleration recorder.

ANNEX IV: DEFINITION OF SEISMIC INTENSITY SCALES

IV.1. MODIFIED MERCALLI INTENSITY (MMI) SCALE

TABLE IV-1. MMI SCALE OF 1931 [IV-1]

Grade	Description
I	Not felt –or, except rarely under especially favourable circumstances. Under certain conditions, at and outside the boundary of the area in which a great shock is felt: --- Sometimes birds, animals, reported uneasy or disturbed. --- Sometimes dizziness or nausea is experienced. --- Sometimes trees, structures, liquid, bodies of water, may sway; doors may swing, very slowly.
II	Felt indoors by few, especially on upper floors, or by sensitive or nervous persons. Also, as in Grade I but often more noticeably. --- Sometimes hanging objects may swing, especially when delicately suspended. --- Sometimes trees, structures, liquids, bodies of water, may sway; doors may swing, very slowly. --- Sometimes birds, animals, reported uneasy or disturbed. --- Sometimes dizziness or nausea is experienced.
III	Felt indoors by several, motion usually rapid vibration. --- Sometimes not recognized to be an earthquake at first. --- Duration estimated in some cases. --- Vibration like that due to passing of light, or lightly loaded trucks, or heavy trucks some distance away. --- Hanging objects may sway slightly. --- Movements may be appreciable on upper levels of tall structures. --- Rocked standing motorcars slightly.
IV	Felt indoors by many, outdoors by few. --- Awakened few, especially light sleepers. --- Frightened no one, unless apprehensive from previous experience. --- Vibration like that due to passing of heavy, or heavily loaded trucks. --- Sensations like heavy body striking building or falling of heavy objects inside. --- Rattling of dishes, windows, doors; glassware and crockery clink and clash. --- Creaking of walls, frame, especially in the upper range of this Grade. --- Hanging objects swung, in numerous instances. --- Disturbed liquids in open vessels slightly. --- Rocked standing motorcars noticeably.

TABLE IV-1. MMI SCALE OF 1931 [IV-1] (cont.)

Grade	Description
V	Felt indoors by practically all, outdoors by many or most: outdoors direction estimated. --- Awakened many, or most. --- Frightened few – slight excitement, a few ran outdoors. --- Buildings trembled throughout. --- Broke dishes, glassware, to some extent. --- Cracked windows – in some cases, but not generally. --- Overturned vases, small or unstable objects, in many instances, with occasional fall. --- Hanging objects, doors, swing generally or considerably. --- Knocked pictures against walls or swung them out of place. --- Opened or closed doors, shutters, abruptly. --- Pendulum clocks stopped, started, or ran fast or slow. --- Moved small objects, furnishings, and the latter to slight extent. --- Spilled liquids in small amounts from well-filled open containers. --- Trees, bushes shaken slightly.
VI	Felt by all indoors and outdoors. --- Frightened many, excitement general, some alarm, many ran outdoors. --- Awakened all. --- Persons made to move unsteadily. --- Trees, bushes shaken slightly to moderately. --- Liquid set in strong motion. --- Small bells rang – church, chapel, school, etc. --- Damage slight in poorly built buildings. --- Fall of plaster in small amount. --- Cracked plaster somewhat, especially fine cracks, chimneys in some instances. --- Broke dishes, glassware, in conservable quantity, also some windows. --- Fall of knickknacks, books, pictures. --- Overturned furniture in many instances. --- Moved furnishings of moderately heavy kind.

TABLE IV-1. MMI SCALE OF 1931 [IV-1] (cont.)

Grade	Description
VII	Frightened all – general alarm, all ran outdoors. --- Some, or many, found it difficult to stand. --- Notices by persons driving motorcars. --- Trees and bushes shaken moderately to strongly. --- Waves on ponds, lakes, and running water. --- Water turbid from mud stirred up. --- In caving to some extent of sand or gravel stream banks. --- Rang large church bells, etc. --- Suspended objects made to quiver. --- Damage negligible in buildings of good design and construction, slight to moderate in well-built ordinary buildings, considerable in poorly built or badly designed buildings, adobe houses, old walls (especially where laid up without mortar), spires, etc. --- Cracked chimneys to considerable extent, walls to some extent. --- Fall of plaster in considerable to large amount, also some stucco. --- Broke numerous windows, furniture to some extent. --- Shook down loosened brickwork and tiles. --- Broke weak chimneys at the roofline (sometimes damaging roofs). --- Fall of cornices from towers and high buildings. --- Dislodged bricks and stones. --- Overturned heavy furniture, with damage from breaking. --- Damage considerable to concrete irrigation ditches.
VIII	Fright general – alarm approaches panic. --- Disturbed persons driving motorcars. --- Trees shaken strongly – branches, trunks, broken off, especially palm trees. --- Ejected sand and mud in small amounts. --- Changes: temporary permanent; in flow of springs and wells; dry wells renewed flow; in temperature of spring and well waters. --- Damage slight to structures (brick) built specially to withstand earthquakes. --- Considerable in ordinary substantial buildings, partial collapse; racked, tumbled down, wooden houses in some cases; threw out panel walls in frame structures, broke off decaying piling. --- Fall of walls. --- Cracked, broke, solid stone walls seriously. --- Twisting, fall, of chimneys, columns, monuments, also factory stacks, towers.
IX	Panic general. --- Cracked ground conspicuously. --- Damage considerable in (masonry) structures built specially to withstand earthquakes. --- Threw out of plumb some wood-frame houses built specially to withstand earthquakes. --- Great in substantial (masonry) buildings, some collapse in large part; or wholly shifted frame buildings off foundations, racked frames; --- Serous to reservoirs; underground pipes sometimes broken.

TABLE IV-1. MMI SCALE OF 1931 [IV-1] (cont.)

Grade	**Description**
X	Cracked ground, especially when loose and wet, up to widths of server inches; fissures up to a yard in width ran parallel to canal and stream banks. --- Landslides considerable from riverbanks and steep coasts. --- Shifted sand and mud horizontally on beaches and flat land. --- Changed level of water in wells. --- Threw water on banks of canals, lakes, rivers, etc. --- Damage serous to dams, dikes, embankments. --- Damage severe to well-built wooden structures and bridges, some destroyed. --- Developed dangerous cracks in excellent brick walls. --- Destroyed most masonry and frame structures, also their foundations. --- Bent railroad rails slightly. --- Tore apart, or crushed endwise, pipelines buried in earth. --- Open cracks and broad wavy fold sin cement pavements and asphalt road surfaces.
XI	Disturbances in ground many and widespread, varying with ground material. --- Broad fissures, earth slumps, and landslips in soft, wet ground. --- Ejected water in large amount charged with sand and mud. --- Caused sea waves (tidal waves) of significant magnitude. --- Damage severe to wood-frame structures, especially near shock centres. --- Great to dams, dikes, embankments, often for long distances. --- Few, if any (masonry), structures remained standing. --- Destroyed large well-built bridges by the wrecking of supporting piers, or pillars. --- Affected yielding wooden bridges less. --- Bent railroad rails greatly and thrust them endwise. --- Put pipelines buried in earth completely out of service.
XII	Damage total – practically all works of construction damaged greatly or destroyed. --- Disturbances in ground great and varied, numerous fissures. --- Landslides, falls of rock of significant character slumping of riverbanks, etc., numerous and extensive. --- Wrenched loose, tore off, large rock masses. --- Fault slips in form rock, with notable horizontal and vertical offset displacements. --- Water channels, surface and underground, disturbed and modified greatly. --- Dammed lakes, produced waterfalls, deflected rivers, etc. --- Waves seen on ground surfaces. --- Distorted lines of sight and level. --- Threw objects upward into the air.

IV.2. MEDVEDEV-SPONHEUER-KARNIK (MSK) SCALE

TABLE IV-2. QUANTIFICATION OF THE JUDGMENT TERMS USED IN THE MSK SCALE [IV-2]

A. QUANTIFICATION OF THE SCALE		
I.	TYPES OF STRUCTURES (buildings not anti-seismic design)	
	Structure A:	Buildings in fieldstone, rural structures, adobe houses, clay or unreinforced masonry houses.
	Structure B:	Ordinary brick buildings, buildings of the large block and prefabricated type, half timbered structures, buildings in natural hewn stone.
	Structure C:	Reinforced buildings, well-built wooden structures.
II.	DEFINITION OF QUANTITY	
	Single, few: about 5% Many: about 50% Most: about 75%	
III.	CLASSIFICATION OF EARTHQUAKE DAMAGE TO BUILDINGS	
	Grade 1:	Slight Damage: Fine cracks in plaster; fall of small pieces of plaster.
	Grade 2:	Moderate Damage: Small cracks in walls; fall of fairly large pieces of plaster; pantiles slip off; cracks in chimneys; parts of chimney fall down.
	Grade 3:	Heavy Damage: Large and deep cracks in walls; fall of chimneys.
	Grade 4:	Destruction: Gaps in walls; parts of buildings may collapse; separate parts of the building lose their cohesion; inner walls and in-fill walls between frame collapse.
	Grade 5:	Total Damage: Total collapse of buildings.
IV.	ARRANGEMENT OF THE SCALE	
	a)	Persons and surroundings
	b)	Structures of all kinds
	c)	Environment
B. INTENSITY		
I.	NOT NOTICEABLE	
	a)	The intensity of the vibration is below the limit of sensibility; the tremor is detected and recorded by seismographs only.
II.	SCARCELY NOTICEABLE (very slight)	
	a)	Vibration is felt only by individual people at rest in houses, especially in upper floors of buildings.
III.	WEAK, PARTIALLY OBSERVED ONLY	
	a)	The earthquake is felt indoors by a few people, outdoors only in favourable circumstances. The vibration is like that due to the passing of a light truck. Attentive observers notice a slight swinging of hanging objects, somewhat more heavily on upper floors.

TABLE IV-2. QUANTIFICATION OF THE JUDGMENT TERMS USED IN THE MSK SCALE [IV-2] (cont.)

IV.	LARGELY OBSERVED	
	a)	The earthquake is felt indoors by many people, outdoors by few. Here and there people awake, but no one is frightened. The vibration is like that due to the passing of a heavily loaded truck. Windows, doors and dishes rattle. Floors and walls creak. Furniture begins to shake. Hanging objects swing slightly. Liquids in open vessels are slightly disturbed. In standing motor cars the shock is noticeable.
V.	AWAKENING	
	a)	The earthquake is felt indoors by all, outdoors by many. Many sleeping people awake. A few run indoors. Animals become uneasy. Buildings tremble throughout. Hanging objects swing considerably. Pictures knock against walls or swing out of place. Occasionally pendulum clocks stop. Unstable objects may be overturned or shifted. Open doors and windows are thrust open and slam back again. Liquids spill in small amounts from well-filled open containers. The sensation of vibration is like that due to heavy object falling inside the building.
	b)	Slight damages in buildings of Type A are possible.
	c)	Sometimes change in flow springs.
VI.	FRIGHTENING	
	a)	Felt by most indoors and outdoors. Many people in buildings are frightened and run outdoors. A few persons lose their balance. Domestic animals run out of their stalls. In few instances dishes and glassware may break, books fall down. Heavy furniture may possibly move, and small steeple bells may ring.
	b)	Damage of Grade 1 is sustained in single buildings of Type B and in many Type A. Damage in few buildings of Type A is of Grade 2.
	c)	In few cases cracks up to widths of 1 cm possible in wet ground; in mountains occasional landslips; change in flow of springs and in level of well-water are observed.
VII.	DAMAGE TO BUILDINGS	
	a)	Most people are frightened and run outdoors. Many find it difficult to stand. The vibration is noticed by persons driving motor cars. Large bells ring.
	b)	In many buildings of Type C damage of Grade 1 is caused; in many buildings of Type B damage is of Grade 2. Many buildings of Type A suffer damage of Grade 3, few of Grade 5. In single instances landslips of roadway on steep slopes; cracks in roads; seams of pipelines damaged; cracks in stone walls.
	c)	Waves are formed on water, and water is made turbid by mud stirred up. Water levels in wells change, and the flow of springs changes. In few cases dry springs have their flow restored and existing springs stop flowing. In isolated instances parts of sandy or gravelly banks slip off.
VIII.	DESTRUCTION OF BUILDINGS	
	a)	Fright and panic; also, persons driving motor cars are disturbed. Here and there branches of trees break off. Even heavy furniture moves and partly overturns. Hanging lamps are in part damaged.
	b)	Many buildings of Type C suffer damage of Grade 2, and few of Grade 3. Many buildings of Type B suffer damage of Grade 3, and few of Grade 5. Many buildings of Type A suffer damage of Grade 4, and few of Grade 5. Occasional breaking of pipe seams. Memorials and monuments move and twist. Tombstones overturn. Stone walls collapse.
	c)	Small landslips in hollows and on banked roads on steep slopes; cracks in ground up to widths of several centimetres. Water in lakes becomes turbid. New reservoirs come into existence. Dry wells refill and existing wells become dry. In many cases change in flow of level of water.

TABLE IV-2. QUANTIFICATION OF THE JUDGMENT TERMS USED IN THE MSK SCALE [IV-2] (cont.)

IX.	GENERAL DAMAGE TO BUILDINGS	
	a)	General panic; considerable damage to furniture. Animals run to and from in confusion and cry.
	b)	Many buildings of Type C suffer damage of Grade 3, a few of Grade 5. Many buildings of Type B show damage of Grade 4; a few of Grade 5. Many buildings of Type A suffer damage of Grade 5. Monuments and columns fall. Considerable damage to reservoirs; underground pipes partly broken. In individual cases railway lines are bent and roadway damaged.
	c)	On flat land overflow of water, sand and mud is often observed. Ground cracks to widths of up to 10 cm, on slopes and riverbanks more than 10 cm; furthermore, a large number of slight cracks in ground; falls of rock, many landslides and earth flows; large waves in water. Dry wells renew their flow and existing wells dry up.
X.	GENERAL DESTRUCTION OF BUILDINGS	
	a)	Many buildings of Type C suffer damage of Grade 4, a few of Grade 5. Many buildings of Type B show damage of Grade 5; most of Type A have Destruction Category 5; critical damage to dams and dykes and sever damage to bridges. Railway lines are bent slightly. Underground pipes are broken or bent. Road paving and asphalt show waves.
	b)	In ground, cracks up to widths of several decimetres, sometimes up to 1 meter. Parallel to water courses broad fissures occur. Lose ground slides from steep slopes. From riverbanks and steep coasts considerable landslides are possible. In coastal areas displacement of sand and mud; change of water level in wells; water from canals, lakes, rivers, etc. thrown on land. New lakes occur.
XI.	DESTRUCTION	
	a)	Sever damage even to well-built buildings, bridges, water dams and railway lines; highways become useless; underground pipes destroyed.
	b)	Ground considerably destroyed by broad cracks and fissures, as well as by movement in horizontal and vertical directions; numerous landslips and falls of rock. The intensity of the earthquake requires to be investigated specially.
XII.	LANDSCAPE CHANGES	
	a)	Practically all structures above and below ground are greatly damaged or destroyed.
	b)	The topography of the ground is radically changed. Considerable ground cracks with extensive vertical and horizontal movements are observed. Falls of rock and slumping of riverbanks over wide areas; lakes are dammed; waterfalls appear, and rivers are deflected. The intensity of the earthquake requires to be investigated specially.

TABLE IV-2. QUANTIFICATION OF THE JUDGMENT TERMS USED IN THE MSK SCALE [IV-2] (cont.)

C. ESTIMATED NUMERICAL CHARACTERISTICS OF MACRO-SEISMIC MSK INTENSITIES

Intensity	Acceleration, a (cm-sec^{-2})	Velocity, v (cm-sec^{-1})	Displacement, d (mm)
V	12 – 25	1.0 – 2.0	0.5 – 1.0
VI	25 – 50	2.1 – 5.0	1.1 – 2.0
VII	50 – 100	5.1 – 8.0	2.1 – 5.0
VIII	100 – 200	8.1 – 16.0	5.1 – 8.0
IX	200 – 400	16.1 – 32.0	8.1 – 16.0
X	400 – 800	32.1 – 65.0	16.1 – 32.0

a: Ground acceleration in cm/sec^2 for periods between 0.1 sec and 0.5 sec

v: Velocity of ground oscillation in cm/sec for periods between o.5 sec and 2.0 sec

d: Amplitude of displacement in mm of a pendulum mass centre with the natural frequency of 4 Hz and the logarithmic decrement of 0.5 (8% of critical damping)

D. TYPES OF STRUCTURES, QUANTITY AND CLASSIFICATION OF DAMAGE TO BUILDINGS

Intensity	Types of Structures		
	A	B	C
V	Single – 1		
VI	Single – 2	Single – 1	
	Many – 1		
VII	Single – 4		
	Many – 3	Many – 2	Many – 1
VIII	Single – 5	Single – 4	Single – 3
	Many – 4	Many – 3	Many – 2
IX		Single – 5	Single – 4
	Many – 5	Many – 4	Many – 3
X			Single – 5
		Many – 5	Many – 4
	Most – 5		

I, II, III, IV, V = Classification of damage

IV.3. JMA SEISMIC INTENSITY SCALE

TABLE IV-3. ORIGINAL JMA SEISMIC INTENSITY SCALE (UP TO 1996) [IV-3]

Intensity	Degree	Description
0	Not felt	Too weak to be felt by humans; registered only by seismographs.
I	Slight	Felt by some persons at rest or by those who are especially sensitive to earthquakes.
II	Weak	Felt by most persons; slight shaking of windows and Japanese latticed sliding doors.
III	Moderately Strong	Shaking of houses and buildings, heavy rattling of windows and Japanese latticed sliding doors, swinging of hanging objects, stopping of some pendulum clocks, and moving of liquids in vessels; some people are so frightened that they run out of doors.
IV	Strong	Strong shaking of houses and buildings, overturning of unstable objects, and spilling of liquids out of vessels.
V	Very Strong	Cracking brick and plaster walls, overturning stone lanterns and gravestones, and similar objects, damaging chimneys and mud- and- plaster warehouses, and causing landslides in steep mountains.
VI	Disastrous	Causing destruction of 1-30% of Japanese wooden houses; causing large landslides; fissures in flat ground and some in low fields, accompanied by mud and waterspouts
VII	Ruinous	Causing destruction of more than 30 per cent of the houses; causing large landslides, fissures and faults.

TABLE IV-4. JMA SEISMIC INTENSITY SCALE (AFTER 1996) (COURTESY OF JMA [IV-4])

A. Human perception and reaction as well as indoor and outdoor response to earthquake motions			
Seismic Intensity	Human Perception and Reaction	Indoor Response	Outdoor Response
0	Imperceptible to people but recorded by seismometers.	--	--
1	Felt slightly by some people at rest in buildings.	--	--
2	Felt by many people at rest in buildings. Some people may be awoken.	Hanging objects such as lamps swing slightly.	--
3	Felt by most people in buildings. Felt by some people walking. Many people are awoken.	Dishes in cupboards may rattle.	Electric wires swing slightly.
4	Most people are startled. Felt by most people walking. Most people are awoken.	Hanging objects such as lamps swing significantly, and dishes in cupboards rattle. Ornaments with high aspect rations may fall.	Electric wires swing significantly. Those driving vehicles may notice the tremor.
5 Lower	Many people are frightened and feel the need to hold onto something stable.	Hanging objects such as lamps wing violently. Dishes in cupboards and items on bookshelves may fall. Many unstable ornaments fall. Unsecured furniture may move, and with high aspect furniture may topple over.	In some cases, windows may break and fall. People notice electricity poles moving. Roads may sustain cracking damage.
5 Upper	Many people find it hard to move; walking is difficult without holding on to something stable.	Dishes in cupboards and items on bookshelves are more likely to fall. TVs may fall from their stands and unsecured furniture may topple over.	Windows may break and fall, unreinforced concrete block walls may collapse, poorly installed vending machines may topple over, automobiles may stop due to the difficult of continued movement.
6 Lower	It is difficult to remain standing	Many unsecured furniture moves and may topple over. Doors may become wedged shut.	Wall tiles and windows may sustain damage and fall.
6 Upper	It is impossible to remain standing or move without crawling. People may be thrown through the air.	Most unsecured furniture moves and is more like to topple over.	Wall tiles and windows are more likely to break and fall. Most unreinforced concrete block walls collapse.
7		Most unsecured furniture moves and topples over or may even be thrown through the air.	Wall tiles and windows are even more likely to break and fall. Reinforced concrete block walls may collapse.

B. Reinforced concrete buildings		
Seismic Intensity	Reinforced Concrete Buildings	
	High Earthquake Resistance	Low Earthquake Resistance
5 Upper	--	Cracks may form in walls, crossbeams and pillars.
6 Lower	Cracks may form in walls, crossbeams and pillars.	Cracks are more likely to form in walls, crossbeams and pillars.
6 Upper	Cracks are more likely to form in walls, crossbeams and pillars.	Slippage and X-shaped cracks may be seen in walls, crossbeams and pillars. Pillars at ground level or on intermediate floors may disintegrate, and buildings may collapse.

TABLE IV-4. JMA SEISMIC INTENSITY SCALE (AFTER 1996) (Courtesy of JMA [IV-4]) (cont.)

7	Cracks are even more likely to form in walls, crossbeams and pillars. Ground level or intermediate floors may sustain significant damage. Buildings may lean in some cases.	Slippage and X-shaped cracks are more likely to be seen in walls, crossbeams and pillars. Pillars at ground level or on intermediate floors are more likely to disintegrate, and buildings are more likely to collapse.

Note 1: Earthquake resistance tends to be higher for newer foundations. The value tends to be low for structures built up to 1981, and high for those building since 1982. However, to maintain a certain range of earthquake resistance according to differences in structure and 2D/3D arrangement of reinforced walls, resistance is not necessarily determined only by foundation age. The earthquake resistance of existing buildings can be ascertained through analysis.

Note 2: Slight cracks may form in reinforced concrete buildings without their core strength being affected.

C. Response of ground and slopes, etc.

Seismic Intensity	Situation of Ground	Situation of Slopes, etc.
5 Lower	Small cracks may form and liquification[1] may occur	Rock falls and landslips may occur.
5 Upper		
6 Lower	Cracks may form.	Landslips may occur.
6 Upper	Large cracks may form.	Landslips are more likely to occur; large landslides and massif collapses may be seen[2]

[1] Liquefaction may be seen in areas with a high groundwater level and poorly graded and consolidated sand deposits. Damage observed as a result of liquefaction includes spouts of muddy water from the ground, subsidence in riverbanks and quays, ejection of sewage pipes and manholes, and differential displacements or destruction of building foundations.

[2] When large landslides and collapses occur, dams may form depending on geographical features, and debris flow may occur due to large quantities of sediment produced.

D. Influence on utilities and infrastructure, etc.

Suspension of gas supply	In the event of shaking with a seismic intensity of about 5 Lower or more, gas meter with safety devices are tripped, stopping the supply of gas. In the event of stronger shaking, the gas supply may stop for entire local blocks.[1]
Suspension of water supply, electrical blackouts	Suspension of water supply and electrical blackouts may occur in regions experiencing shaking with an intensity of about 5 Lower or more.[1]
Suspension of railroad services, regulation of highways, etc.	In the event of shaking with a seismic intensity of about 4 or more, services on railroads or highways may be stopped for safety confirmation. Speed control and traffic regulations are performed according the judgment of the relevant bodies. (Standards for safety confirmation differ by organization and area).
Disruption of lines of communication such as telephones	In the event of an earthquake, telephone line congestion may occur as a result of increased use related to safety confirmation around regions of strong shaking. To combat this, telecommunications providers offer message boards and message dial services for use in disasters resulting from earthquakes with a seismic intensity of about 6 Lower or more.
Suspension of elevator services	In the event of shaking with a seismic intensity of about 5 Lower or more, elevators with earthquake control devices will stop automatically for safety reasons. Resumption of service may be delayed until safety is confirmed.

[1] For shaking with a seismic intensity of 6 Upper or more, gas water and electric supplies may stop over wide areas.

TABLE IV-4. JMA SEISMIC INTENSITY SCALE (AFTER 1996) (Courtesy of JMA [IV-4]) (cont.)

E. Effects on specific structures	
Shaking of skyscrapers from long period ground motion[1]	Due to their longer characteristic period, skyscrapers react less to earthquakes than general reinforced concrete buildings, which have a shorter characteristic period. However, they exhibit slow shaking over a long time in response to long period ground motion. If motion is strong, poorly fixed office appliances may move significantly, and people may have to hold onto stable objects to maintain their position.
Sloshing of oil tanks	Sloshing of oil tanks occurs in response to long period ground motion. As a result, oil outflows or fires may occur
Damage or collapse of ceilings etc. at institutions covering large spaces	Institutions covering large spaces such as gymnasiums or indoor pools, ceilings may shake significantly and sustain damage or collapse, even in cases where ground motion is not severe enough to cause other structural damage.
[1] Occasionally, when a large earthquake occurs, long period seismic waves reach locations several 100 km from the hypocentre; such waves may be amplified depending on the characteristic period of the ground, thus extending their duration.	
General notes:	
(1) As a rule, seismic intensities announced by JMA are values observed using seismic intensity meters (strong motion accelerometer installed on the ground or on the first floor of low-rise buildings). This document describes the phenomena and damage that may be observed for individual seismic intensity levels. Seismic intensities are not determined from the observed phenomena described here. (2) Seismic ground motion is significantly influenced by underground conditions and topography. Seismic intensity is the value observed at a site where a seismic intensity meter is installed and may vary even within the same city or local region. In addition, the amplitude of seismic motion generally differs by floor and location within the same building, as shaking on upper floors may be considerably amplified. (3) Sites with the same level of seismic motion will not necessarily suffer the same degree of damage, as the effect of tremors depends on the nature of the seismic motion (such as amplitude, period and duration), the type of construction and underground conditions. (4) This document describes typical phenomena that may be seen at individual levels of seismic intensity. In some cases, the level of damage may be greater or less than specified. Not all phenomena described for each intensity level may necessarily occur. (5) The information outlined here is regularly checked at intervals of about five years and is updated in line with actual phenomena observed in new application or improvements in the earthquake resistance of buildings and structures.	

REFERENCES TO ANNEX IV

[IV-1] WOOD, H.O., NEUMANN, F., Modified Mercalli Intensity Scale of 1931, Bull. Seis. Soc. Am. **21** 4 (1931) 277–283.

[IV-2] U.N. EDUCATIONAL, SCIENTIFIC AND CULTURAL ORGANIZATION, Consensus Development of Judgmental Terms used in the MSK Intensity Scale Developed at the 1st Meeting of the Seismicity and Seismo-Tectonic Working Meeting Group, UNESCO (1965).

[IV-3] ELECTRIC POWER RESEARCH INSTITUTE, The Great Hanshin-Awaji Earthquake of January 17, 1995: A Report on Electric Power and Other Impacts, EPRI TR-107083 (1998).

[IV-4] JAPAN METEOROLOGICAL AGENCY, Tables Explaining the JMA Seismic Intensity Scale, http://www.jma.go.jp/jma/en/Activities/inttable.html

ANNEX V: EXPERIENCE OF DAMAGE IN CONVENTIONAL STRUCTURES, SYSTEMS AND COMPONENTS IN THE 1995 SOUTH HYOGO PREFECTURE EARTHQUAKE (HANSHIN-AWAJI EARTHQUAKE DISASTER) IN JAPAN

V.1. MACHINE FOUNDATION AND INSTALLATION

V.1.1. Floor Subsidence

Damage:

	Original JMA Seismic Intensity
- Subsided factory floor	5
- Damaged legs of cold drawbench (rolling machine)	5
- Subsided measurement room floor	6 or 7
- Inclined web press	5 or 6
- Inclined roller storage	5 or 6
- Starter motor and foundation	5 or 6
- Damaged piping system supports	4 or 5

Description:

As listed above, numerous cases of unequal foundation and floor subsidence due to ground settlement and liquefaction were observed as damage to mechanical equipment. Most of these were observed in waterfront areas of reclaimed ground and riverside areas with a seismic intensity of 4 or higher. Cracks and bumps occurred to foundation concrete. These caused damage to mechanical equipment, e.g. core displacement, inclination, uplift from foundation, and sinking.

V.1.2. Movement without Anchor Bolts

Damage:

	Original JMA Seismic Intensity
- Damaged legs of numerically controlled milling machine - 1	7
- Damaged legs of numerically controlled milling machine - 2	7
- Moved grinding machine	7
- Moved jig platform of five-facet processing machine	6 or 7
- Moved stools	5 or 6

-	Processing machine and steel platform	7
-	Moved machine tool	7
-	Moved injection machine	7
-	Moved processing machine with vibration-proof support	7
-	Moved resin processing machine - 1	7
-	Moved machine tool platform	7
-	Damaged legs of container washer	5 or 6
-	Moved filter legs	6
-	Unfixed bottler conveyer	4
-	Moved resin processing machine - 2	4 or 5
-	Moved oven	5 or 6

Description:

As listed above, damage to mechanical equipment without anchor bolts were mostly movement. Although the extent of movement depended on the weight of the equipment and/or the seismic intensity, displacement over several hundred millimetres up to 1500 mm was observed in some serious cases. In areas with a seismic intensity of 7, some jumped over the 75 mm-high stoppers. Some equipment supported by base isolation rubber jumped up and down due to the earthquake. As for mechanical equipment with legs, some legs were damaged or deformed due to lack of strength.

V.1.3. Movement with Anchor Bolts

Damage:

		Original JMA Seismic Intensity
-	Moved machining centre - 1	7
-	Tumbled horizontal boring machine	6 or 7
-	Moved press machine	7
-	Pull-out Legs of machine anchor bolt	7
-	Damaged base isolated equipment support frame	4 or 5
-	Tumbled operating status display panel	5 or 6
-	Damaged legs of overhead fibre-reinforced plastics water tank	7

Description:

As listed above, damage to mechanical equipment fixed with anchor bolts were mainly pull-out or breakage of anchor bolts and nuts. As a result, some anchored equipment items were moved or tumbled. In some cases, expansion anchor bolts were pulled out of concrete foundations. Some foundations were damaged when the anchor bolts were pulled out.

V.2. MACHINES

V.2.1. Machining and Moulding Processing Equipment

Damage:

Damage	Original JMA Seismic Intensity
- Tumbled processing machine with heavy head	7
- Tumbled universal tool grinding machine	6 or 7
- Tumbled machine tool	6 or 7
- Tumbled drill press	6 or 7
- Tumbled general-purpose lathe	6 or 7
- Tumbled grinding machine	7
- Moved machining centre - 2	6
- Moved precision lathe	6
- Damaged lathe	7
- Failure of Milling machine	7
- Damaged resin processing machine	7
- Moved laced heavy mould	7
- Damaged ball bearing in equipment supporting section	4 or 5

Description:

As listed above, machining and moulding processing equipment were often found to have been moved or tumbled. This is due to the fact that these machines were often installed on levelling blocks without anchorage. Many of the tumbled machines had a high barycentre. Many of the displaced machines were damaged by collision against the building and others.

V.2.2. Rolling Equipment

Damage:

		Original JMA Seismic Intensity
-	Misalignment of cold rolling facilities	5
-	Rolling process motor damaged by collapsed stack	6
-	Flooded rolling motor	6

Description:

As listed above, damage to rolling equipment include horizontal and vertical displacements between the ground floor and the pit, due to liquefaction-induced unequal subsidence in waterfront reclaimed areas. Since this equipment was relatively large, core displacements occurred. As they were located in waterfront areas, motors were flooded.

V.2.3. Electric and Electronic Production Facilities

Damage:

		Original JMA Seismic Intensity
-	Displaced optical sensor head for the positioning of the automatic electronic component inspection line	4 or 5

Description:

As shown above, one case of damaged electric and electronic production facilities was observed. The optical sensor head for the positioning of the automatic electronic component inspection line was displaced.

V.2.4. Casting Facilities

Damage:

		Original JMA Seismic Intensity
-	Overflow of aluminium alloy from aluminium oil feeder	7
-	Damaged die casting machine - 1	7
-	Damaged die casting machine - 2	7

Description:

As listed above, damage to casting equipment include an overflow of molten aluminium alloy into the periphery due to sloshing caused by seismic motion. Similarly, molten solder sloshed in the solder tank installed in the automated line and flew apart into the periphery at an electric and electronic manufacturing plant at which electronic parts were mounted on substrates. Moreover, equipment incidental to a die casting machine fell down.

V.2.5. Food Production Facilities

Damage:

	Original JMA Seismic Intensity
- Tumbled small-capacity vertical mixer	5 or 6
- Moved large-capacity vertical mixer	5 or 6
- Moved filter	6
- Moved confectionery equipment	6 or 7
- Moved confectionery conveyor	5 or 6

Description:

As listed above, damage to food production facilities were represented by moved or tumbled equipment. This is because these machines were installed without anchorage just like machine tools. In particular, small-capacity vertical mixers fell down because their heads were heavy.

V.2.6. Printing Press Machines

Damage:

	Original JMA Seismic Intensity
- Fallen printing roll hanger from chain	4 or 5
- Damaged rod with crooked ends to which printing roll hanger sling is attached	4 or 5
- Newspaper conveyor crashed into a wall	4 or 5
- Damaged bolts connecting rotary press with building joists	4 or 5

Description:

As listed above, damage to printing press equipment include those related to printing roll hangers. The rotary press for newspapers is a tall machine that is approximately 12 m high. Its connecting bolts were damaged due to a relative displacement of approximately 5 mm between the top of the rotary press and the building. Since this machine requires a high level of levelness for precision paper feeding, post-earthquake minute deformations were considered problematic.

V.2.7. Electronic Equipment

Damage:

	Original JMA Seismic Intensity
- Tumbled materials testing machine - 1	7
- Damaged and tilted three-dimensional measuring equipment - 1	7
- Damaged and tilted three-dimensional measuring equipment - 2	6 or 7
- Tumbled materials testing machine - 2	7
- Damaged numerically controlled machine tool control panel	7
- Tumbled turning centre control panel	6 or 7
- Moved control panel	5 or 6
- Tumbled magnetic resonance imaging (MRI) control device	7
- Tumbled MRI main storage device	7
- Damaged Monitor TV in classroom due to collision	6
- Damaged LCD projector due to collision	6
- Damaged experiment equipment at accelerator facility - 1	Unknown
- Damaged experiment equipment at accelerator facility - 2	Unknown
- Damaged radioisotope measuring equipment	Unknown

Description:

As listed above, damaged electronic equipment includes many cases of lost functions as testing machines, measuring equipment, control panels, experiment equipment, and so forth. This indicates that electronic equipment is precision machinery and it is vulnerable to seismic shocks and vibrations, and shocks associated with tumbling. Some heavy equipment moved or jumped out of the radio isotope measuring equipment container box.

V.2.8. Medical Equipment

Damage:

		Original JMA Seismic Intensity
-	Tumbled and damaged radioisotope scintillation camera - 1	7
-	Tumbled and damaged radioisotope scintillation camera - 2	7
-	Damaged ceiling boards around overhead traveling X-ray tube supporting section	5 or 6
-	Damaged bed for X-ray photographing	7
-	Moved MRI	7
-	Moved radiotherapy machine - 1	Unknown
-	Moved radiotherapy machine - 2	5 or 6

Description:

As listed above, damage to medical equipment include tumbled and damaged radioisotope scintillation cameras, damaged radiographic heads, and moved MRI and radiotherapy machines in areas with a seismic intensity of 7.

V.2.9. Others

Damage:

		Original JMA Seismic Intensity
-	Damaged articles during processing	5 or 6
-	Damaged mends of vacuum furnace	5 or 6
-	Damaged bearings of oil fence hoist	4 or 5

Description:

As listed above, damage to other equipment include tumbled and damaged articles during processing, reduced airtightness of vacuum furnace due to unequal ground subsidence, damaged bearings of oil fence hoist due to seismic shocks.

V.3. TANKS

V.3.1. Water Tanks

Damage:

		Original JMA Seismic Intensity
-	Failure of fibre-reinforced plastics (FRP) water tank side wall	Unknown
-	Failure of FRP water tank side wall attached to the pipe due to impact	6 or 7
-	Failure of ceiling and side wall of FRP water tank to impact and sloshing action	6 or 7
-	Failure of FRP water tank ceiling due to sloshing action - 1	7
-	Failure of FRP water tank ceiling due to sloshing action - 2	6 or 7
-	Failure of elevated FRP water tank steel stand	7
-	Inclination of elevated FRP water tank due to failure of its steel stand	7
-	Anchor bolt breakage and failure of water tank on the roof of a hospital under dismantlement	7
-	Failure of water tank	5 or 6
-	Failure of heat pump facilities ceiling	6 or 7
-	Failure of FRP water tank due to sloshing action	6 or 7
-	Failure of stainless steel water tank	5
-	Buckled water tank	6
-	Inclination of water tank	5 or 6

Description:

As shown above, prominent damage to water tanks can be described as follows. As for elevated FRP water tanks installed on building roofs in areas with a seismic intensity of 6 or 7, side panels got broken within reinforced frames due to seismic shocks. When pipes were attached to side panels, the panels were damaged and broken around the pipe attaching sections due to the reactive force of the pipes. There were many examples of broken side panels and ceiling panes due to the sloshing of content water. Conversely, there were examples of broken tank supporting frames and anchor bolts due to inertial force occurred to the mass of the tank itself and content water. In some cases, elephant foot buckling of some large circular water tanks appeared due to seismic forces.

V.3.2. Oil Storage Tanks

Damage:

	Original JMA Seismic Intensity
- Damage of a petroleum products tank due to soil liquefaction	Unknown
- A buckled petroleum products tank	Unknown
- Inclined petroleum products tank to soil liquefaction	6

Description:

As described above, some supporting frames got buckled and pipes connected to tanks got broken due to liquefaction-induced tank subsidence.

V.3.3. Liquefied Gas Storage Tanks

Damage:

	Original JMA Seismic Intensity
- Damage to a liquid propane gas depot due to soil liquefaction	6

Description:

Although the tank itself was not damaged, the pipe supporting foundation separate from the tank foundation sank, which may have given rise to a forced displacement to the pipe, resulting in a gas leak.

V.3.4. Tanks for food production facilities

Damage:

	Original JMA Seismic Intensity
- Failure of tank side wall	6
- Movement of tank	6
- Failure of 'sake' tank	6
- Inclination of enamelled 'sake' tank	6
- Overflow of melted raw material from the confectionary raw material tank due to sloshing action	6 or 7

Description:

It seems that seismic measures are not in place for small-sized items. There seem to be many examples of damage associated with displacements due to defective or no anchor bolts.

V.3.5. Others

Damage:

		Original JMA Seismic Intensity
-	Tumble of gas cylinders in the anti-tumbling rack)	5 or 6
-	Tumble of fire extinguishing cylinders with rack)	6 or 7

Description:

It seems generally that seismic measures are not in place. There seems to be many examples of damage associated with tumbling due to defective or no anchor bolts.

V.4. BOILERS, REFRIGERATORS AND AIR CONDITIONERS

V.4.1. Boilers

Damage:

		Original JMA Seismic Intensity
-	Breakage of seismic tie - 1	4 or 5
-	Breakage of seismic tie - 2	4 or 5
-	Deformation of boiler steel frame near the seismic tie - 1	4 or 5
-	Deformation of boiler steel frame near the seismic tie - 2	4 or 5
-	Deformation of boiler steel frame near the seismic tie - 3	4 or 5
-	Failure of spacer pipe in the boiler	4 or 5
-	Smashed boiler drum due to a fallen, collapsed chimney	6
-	Movement of the boiler	7
-	Failure of the boiler burner	7
-	Failure of the boiler casing, cooling water pump and pipes	Unknown
-	Breakage of the canvas-flexible joint and the water tube boiler's pipe	Unknown
-	Ground subsidence around the boiler	6
-	Separation of the outer water valve cover for main steam, without damage to the valve itself	Unknown

Description:

Except for extremely large boilers used in power stations, over 50% of boilers had some damage in the city of Kobe and other areas. 80% of those without foundation bolts were damaged mainly due to inertial displacement.
On the other hand, large-sized boilers were suspended, and damage was observed mainly in their stabilizers (e.g. seismic ties) which were designed to control their vibrations and in their peripheral supporting structures.

V.4.2. Refrigerators and air conditioners

Damage:

		Original JMA Seismic Intensity
-	Movement of the refrigerator and the cake production conveyer	5 or 6
-	Tumble of the cooler	6 or 7
-	Deformation of the cooling tower leg	7
-	Failure of the cooling tower leg	Unknown
-	Tumble of the cooling tower	6 or 7

Description:

There seem to be many examples of damage associated with defective anchoring.

V.5. PUMPS

Damage:

		Original JMA Seismic Intensity
-	Breakage of casing pump casing connected with rigid pipe - 1	5 or 6
-	Breakage of casing pump casing connected with rigid pipe - 2	6
-	Deformation of pump casing connected with rigid pipe	6
-	The pump and its stand come off from the floor	6 or 7
-	Leakage of melted confectionery raw material at a coupling between pump and pipe	6 or 7
-	Movement of pump	6 or 7
-	Submersion of water pump due to liquefaction flooding	6 or 7

Description:

In many cases, pumps are separated from piping support foundations. Pump casings were damaged due to relative displacements between the piping and the pump casings due to liquefaction-induced unequal subsidence.

V.6. PIPING

V.6.1. Steam Piping

Damage:

		Original JMA Seismic Intensity
-	Boiler main steam line (no damage to the main steam pipe, but insulation around the steam pipe was partially deformed)	4 or 5
-	Deformed gratings around boiler steam line	4 or 5
-	Deformed pipe due to ground settlement	5 or 6

V.6.2. Water Piping

Damage:

		Original JMA Seismic Intensity
-	Collapsed piping system due to collapsed building	7
-	Damaged buried pipes	6
-	Damaged repaired pipes due to ground lateral flow	6
-	Damaged firefighting piping system due to ground settlement	5 or 6
-	Displaced outdoor pipes due to ground settlement	5 or 6
-	Damaged rooftop pipes due to impact	7
-	Tumbled flow meter	7
-	Damaged water pump piping due to relative displacement	5 or 6
-	Tumbled water heater	5 or 6
-	Damaged pipe joints due to relative displacement	6 or 7
-	Damaged pipes for firefighting equipment due to relative displacement	6 or 7

V.6.3. Oil Piping

Damage:

Damage	Original JMA Seismic Intensity
- Peeled coating on oil pipe surface	4 or 5
- Loosened U-bolts for fixing oil pipes	4 or 5
- Deformed outdoor fuel pipes and conduits	5 or 6
- Sunken fuel pump	5 or 6
- Outdoor overhead pipes due to ground liquefaction	6
- Damaged trench due to ground deformation	6

V.6.4. Gas Piping

Damage:

Damage	Original JMA Seismic Intensity
- Damaged gas piping due to relative displacement	5 or 6
- Damaged piping of liquid propane (LP) gas storage facility due to ground settlement	6
- Damaged tie rods for piping of LP gas storage facility due to ground liquefaction	6
- Damaged piping supports of LP gas storage facility due to ground liquefaction	6
- Damaged instrumentation piping of LP gas piping control valve due to ground liquefaction	6
- Damaged bellows for piping of LP gas storage facility due to ground liquefaction	6
- Damaged spherical tank piping of LP gas storage facility due to ground liquefaction	6
- Damaged piping due to ground liquefaction – 1	6
- Damaged piping due to ground liquefaction - 2	6

V.6.5. Food Production Facilities

Damage:

		Original JMA Seismic Intensity
-	Moved mixer tank	6 or 7
-	Displaced confectionary material transport pipes	6 or 7

V.6.6. Instrument Piping

Damage:

		Original JMA Seismic Intensity
-	Pipes attached to landing pier due to ground uneven settlement	5 or 6
-	Damaged outdoor conduit due to ground uneven settlement	5 or 6
-	Damaged instrumentation pipes of rotary press due to relative displacement	4 or 5
-	Damaged instrumentation ducts of rotary press due to relative displacement	4 or 5
-	Damaged transmission cable supports	6 or 7
-	Deformed instrumentation pipes within confectionery production line due to relative displacement	6 or 7

V.6.7. Joints and Supports

Damage:

		Original JMA Seismic Intensity
-	Leakage from piping connected to pump due to relative displacement	6
-	Damaged piping supports (pulling out of anchor bolts)	6 or 7
-	Buckled supporting column of piping	7

V.6.8. Air Ducts

Damage:

		Original JMA Seismic Intensity
-	Damaged air ducts (Falling)	5 or 6

V.7. ELECTRIC POWER SUPPLY EQUIPMENT

Damage:

	Original JMA Seismic Intensity
- Damaged insulators of substation	Unknown
- Displaced transformer	Unknown
- Loosed 500k VA transformer bracket	7
- Damaged 500k VA transformer (Collision due to movement)	7
- Tumbled generator and power supply panels	6 or 7
- Tumbled distribution panel	4 or 5
- Tumbled radiographic transformer	7
- Tumbled elevator operation control panel	Unknown
- Damaged fact line	7

V.8. EMERGENCY AND INDEPENDENT POWER SUPPLY SYSTEMS

Damage:

	Original JMA Seismic Intensity
- Tumbled storage battery	6 or 7
- Tumbled storage battery	6 or 7
- Damaged storage battery electrodes due to tumble	6 or 7
- Moved emergency diesel power generating equipment	7
- Damaged emergency diesel power generating equipment	6 or 7
- Damaged exhaust pipe of emergency diesel power generator	6 or 7

V.9. CRANES

V.9.1. Overhead Traveling Crane

Damage:

	Original JMA Seismic Intensity
- Fallen overhead traveling crane - 1	6
- Fallen overhead traveling crane - 2	7

Description:

Seismic damage to overhead traveling cranes reported includes a fall of traveling beams due to a collapsed building shown in the first case above. In areas struck by severe seismic motion, traveling beams fell even though the building did not collapse in the second case above.

V.9.2. Unloaders

Damage:

	Original JMA Seismic Intensity
- Collapsed coal unloading crane - 1	4 or 5
- Collapsed coal unloading crane - 2	4 or 5
- Collapsed coal unloading crane - 3	4 or 5

Description:

Upper structures of unloaders installed at landing piers fell down or collapsed due to seismic motion.

V.9.3. Container Cranes

Damage:

	Original JMA Seismic Intensity
- Buckled legs of container crane - 1	6
- Buckled legs of container crane - 2	6
- Damaged legs of container crane - 1	6
- Buckled legs of container crane - 3	6
- Derailed container crane traveling wheels - 1	6
- Derailed container crane traveling wheels - 2	6

-	Damaged legs of container crane - 2	6
-	Damaged container traveling system	6
-	Damaged container	6

Description:

A typical pattern was observed in damaged container cranes. Damage concentrated on legs and traveling sections as described below.

As for damaged legs, caisson quay moved and subsided due to lateral flow of revetments, which gave rise to a relative displacement of traveling rails between the ocean side and the shore side, forcing the legs to open to buckling. Although buckled legs concentrated on the ocean side (The 2nd and the 3th case), some occurred on the shore side (The 1st case and the 4th case).
On the other hand, damage to traveling sections include derailed traveling wheels due to rail spans expanded by the displacement of traveling rails (The 5th and the 6th case), a fallen traveling motor or a broken deceleration input shaft (The 8th case), and disengaged gear.

V.9.4. Climbing Jib Cranes

Damage:

		Original JMA Seismic Intensity
-	Broken climbing jib crane	4
-	Broken column of climbing jib crane	Unknown

Description:

Due to structurally high barycentres, top sections fell/dropped. This damage more is also noticeable with the next item.

V.9.5. Tower-type Jib Cranes

Damage:

		Original JMA Seismic Intensity
-	Fall down of tower-type jib crane	Unknown
-	Damaged tower-type jib crane	Unknown
-	Fallen jib crane	4

Description:

Swivel frames fell/dropped (The 1st and 3rd case) and the jib itself got broken (The 2nd case)

V.9.6. Others

Damage:

	Original JMA Seismic Intensity
- Petroleum product reception and shipping facility - 1	4 or 5
- Petroleum product reception and shipping facility - 2	4 or 5

Description:

At petroleum product reception and shipping facility, no damage occurred to the equipment with a structure similar to a crane. However, movable arm fixing pins came off due to seismic motion and the arm was displaced from the upright retracted position to the operating position.

V.10. ELEVATORS

Damage:

	Original JMA Seismic Intensity
- Damaged elevator door due to collapse of building	Unknown
- Damaged elevator main rope	Unknown
- Damaged elevator wiring	Unknown
- Damage guide shoe of counterweight	Unknown
- Derailed counterweight of elevator - 1	6 or 7
- Derailed counterweight of elevator - 2	Unknown
- Damaged rope hoist	Unknown

Description:

The above damage cases show damaged regions/modes of elevators. Typical damage include damaged platform associated with damaged building (the 1st case), damaged main rope (the 2nd case), damaged wiring (the 3rd case), damaged guide shoe (the 4th case), derailed counter weights (the 5th and 6th cases), displaced or fallen hoist or motor (the 7th case), damaged internal equipment, and cage-related damage.

V.11. RAILROAD LINE AND ELECTRIC WIRE

Damage:

	Original JMA Seismic Intensity
- Train inspection pit rail	7
- Damaged rail foundation of train inspection plant - 1	5 or 6
- Damaged rail foundation of train inspection plant - 2	5 or 6
- Damaged rail foundation of train depot	7
- Train inspection pit rail	7
- Damaged rail	7
- Damaged in-house railroad tracks - 1	5
- Damaged in-house railroad tracks - 2	5 or 6
- Steam locomotive under repair	7
- Tumbled equipment within interlocking tower	Unknown
- Fallen signaller	7
- Damaged wiring	Unknown
- Damaged overhead wire sling	Unknown

Description:

Damage in train inspection stations include some cases resulting from collisions due to rocking and derailed cars placed in inspection pits (The 1st through 6th case). In-house railroad tracks got broken due to horizontal and vertical displacements of plant floors caused by unequal ground subsidence (The 7th and 8th cases), signalling equipment fell (The 10th case) and signaller itself dropped (The 11th case). Due to shocks caused by seismic motion, overhead wire slings were deformed, broken, or fell (The 12th and 13th cases).

V.12. WAREHOUSE FACILITIES

Damage:

	Original JMA Seismic Intensity
- Collapsed triple-decker base paper within base paper storage	5 or 6
- Damaged steel racks in warehouse	4 or 5
- Damaged steel racks	5 or 6
- Moved iron racks - 1	7
- Moved iron racks - 2	7
- Tumbled tool boxes - 1	7
- Tumbled tool boxes - 2	7
- Damage to automated multi-story warehouse	7
- Product racks and casters without anchoring	4 or 5
- Tumbled drawing storage racks	5 or 6
- Damaged office desks without anchoring	5 or 6
- Tumbled book racks in office room	5 or 6
- Collapsed bookshelves in library - 1	6
- Collapsed bookshelves in library - 2	6
- Tumbled bookshelves in library	6
- Tumbled book card racks	6
- Inclined bookshelves in library	6

Description:

Examples of damage to general in-house storage equipment include collapsed bulk packages (The 1st case), deformed/buckled heavy item storage rack frames (The 2nd and 3rd case), displaced unfixed storage racks (The 4th and 5th case), and tumbled tool boxes (The 6th and 9th case). The examples of typical damage to automated multi-story warehouses include moved/tumbled buckets (pallets) (The 8th case). There were numerous tumbled/collapsed book racks in which documents were stored (The 11th through 17th case).

ANNEX VI: EFFECTS ON DAMAGE INDICATING PARAMETERS OF EARTHQUAKE ACCELERATION SPIKES

One of the key items to be investigated for the selection of DIPs may be the sensitivity to the acceleration spikes, because very high acceleration spikes are often observed and reported sensationally in spite of their small influence on damage to mechanical and building structure components. In order to study this sensitivity, the following four earthquake motions are selected as typical waves with acceleration spikes or pulses observed in Japan.

- 2007 Niigata-ken Chuetsu-Oki earthquake, Kashiwazaki-Kariwa NPS Unit 1 (Kashiwazaki-Kariwa -1) reactor building (R/B) basemat, east-west direction.
- 1995 South Hyogo prefecture earthquake, Shin-Kobe substation, east-west direction.
- 2004 Niigata-ken Chuetsu earthquake observed at Tokamachi, north-south direction (from the K-NET FTP site of the National Research Institute for Earth Science and Disaster Prevention).
- 2008 Iwate-Miyagi Nairiku eartquake observed at Ichinoseki Nishi, UD direction (from the KiK-net FTP site of the National Research Institute for Earth Science and Disaster Prevention).

The pulses observed at Kashiwazaki-Kariwa-1 nuclear power plant and Shin-Kobe substation are of the type so-called long period 'killer pulse', as shown in Fig. VI-1, and they are believed to be damaging to the structures. On the other hand, acceleration spikes with short duration are considered less damaging, although, the peak acceleration is very high.

The calculation results of the DIPs for these four earthquake motions are shown in Table VI-1. The values of I_{JMA} are distributed in the relatively narrow range, i.e. 6.0 – 6.3, even though the ZPA values change greatly, i.e., from 584 to 3845 Gal. This table shows that a sharp acceleration spike has a small influence on the expected first-excursion type damage. On the other hand, the ZPA determines the response absolute acceleration in the high frequency range as shown in Fig. VI-2.

The concern when using the average elastic response spectrum acceleration, especially for earthquake motions with acceleration spikes or pulses, is that it may cause the overestimation of the effective inertia force on SSCs, which is a cause of damage in the high frequency range. Nevertheless, the response spectrum is useful to evaluate the influence on the dynamic behaviour of specific SSCs, which are sensitive to acceleration.

On the other hand, CAV evaluates accumulation of cyclic loads associated with both duration and acceleration, and I_{JMA} evaluates the effective inertia force, which has a direct influence on damage.

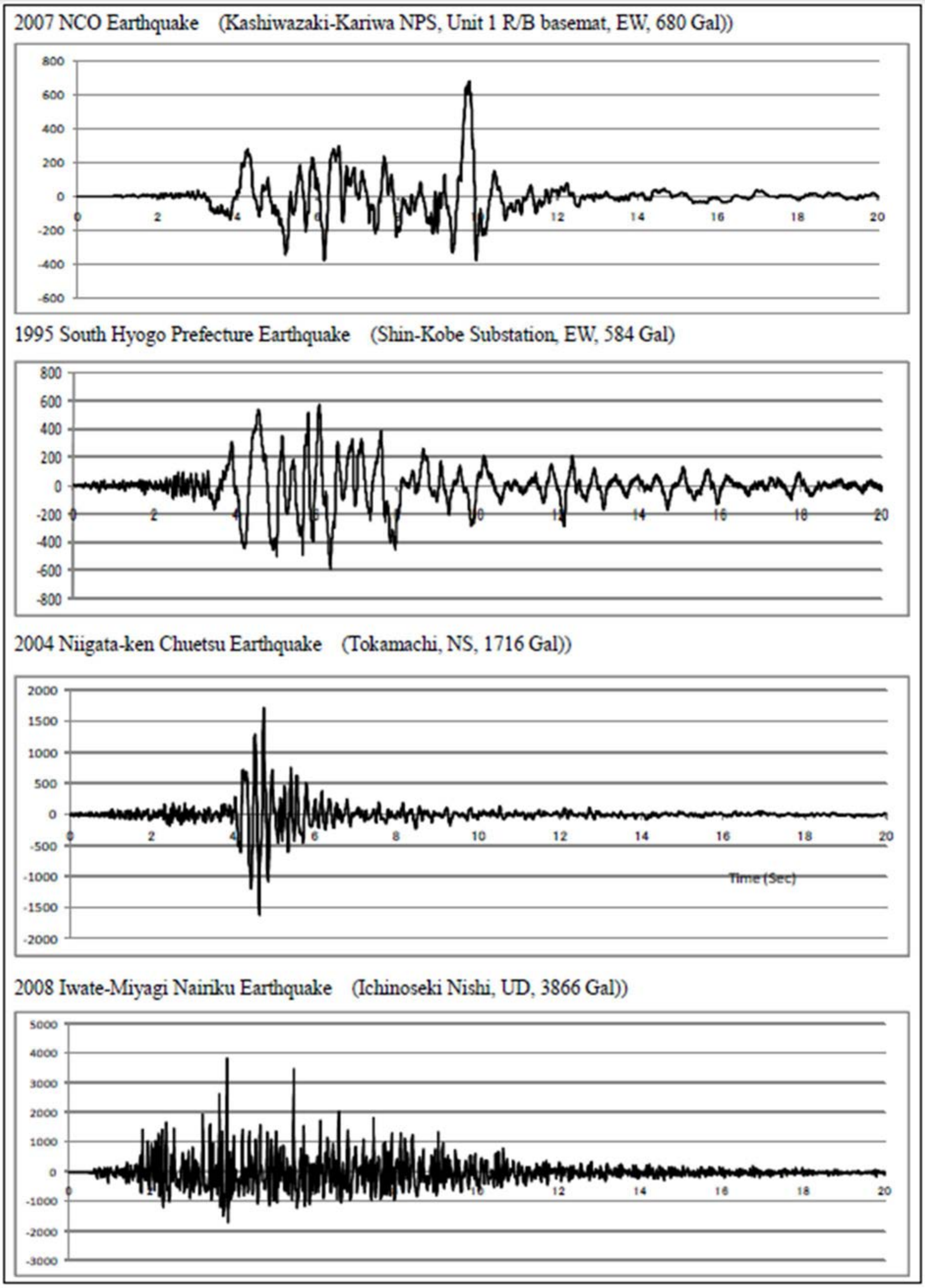


FIG. VI-1. Earthquake acceleration time-histories investigated.

TABLE VI-1. CALCULATED DIPS FOR FOUR TYPICAL EARTHQUAKE MOTIONS

Observed Location	ZPA (Gal)				Calculated JMA		Standardized CAV (g-sec)			
	NS	EW	UD	Resultant	A_{JMA} (Gal)	I_{JMA}	NS	EW	UD	SRSS
KK-1 R/B Basemat	311	680	408	685	346	6	0.97	1.16	0.76	1.70
Shin-Kobe	511	584	495	696	416	6.1	1.34	1.76	1.11	2.48
Tokamachi	1716	850	564	1750	426	6.1	1.62	1.45	0.60	2.25
Ichinoseki Nishi	1143	1432	3866	4022	532	6.3	4.54	4.79	6.18	9.04

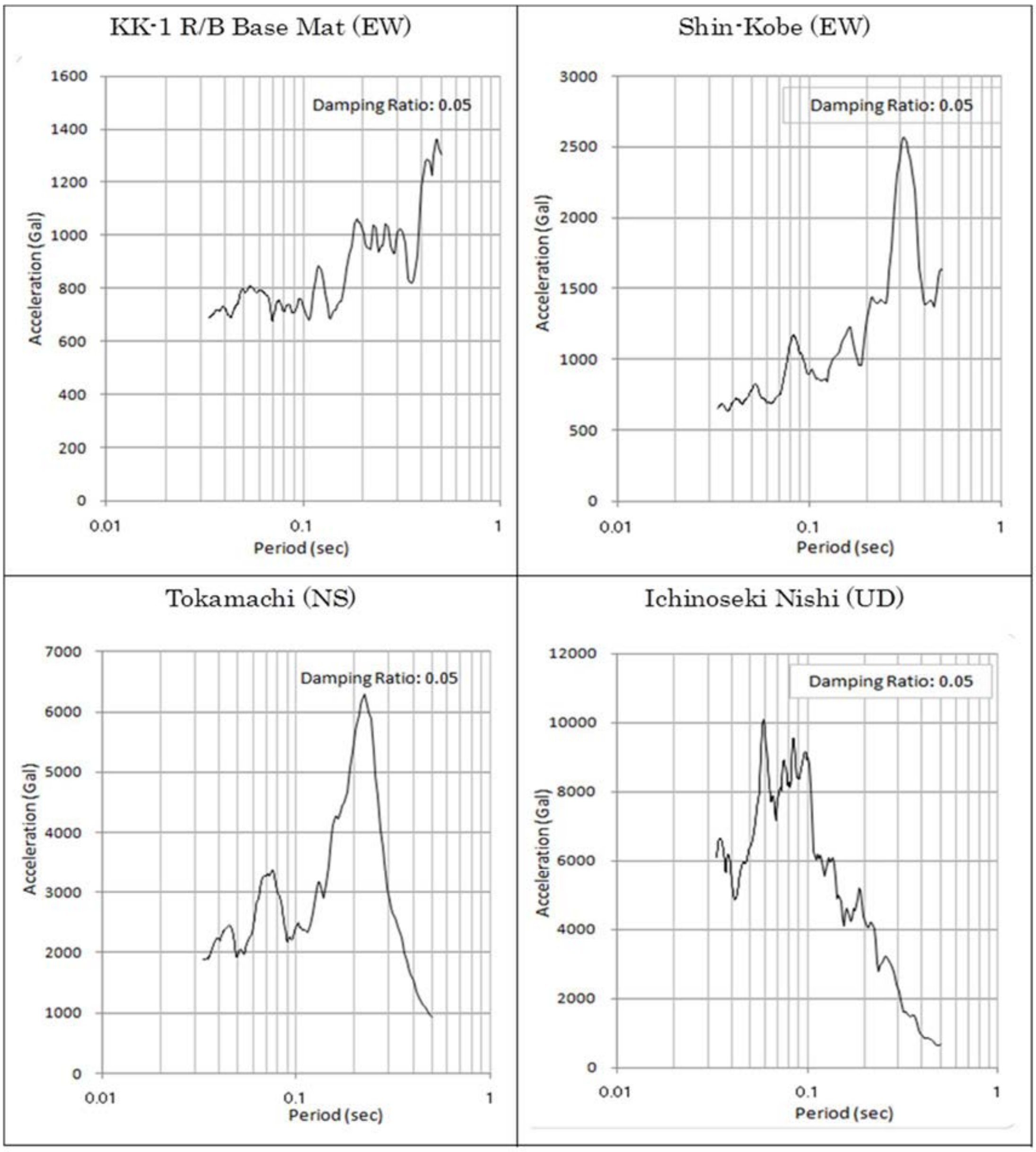


FIG. VI-2. Acceleration response spectra.

ANNEX VII: DAMAGE OBSERVED AT NUCLEAR POWER PLANTS AND CALCULATED DAMAGE INDICATING PARAMETERS

TABLE VII-1. SIGNIFICANT AND MINOR DAMAGES IN STRUCTURE OBSERVED AT THE LOCATIONS OF CALCULATED DIP (KASHIWAZAKI-KARIWA NUCLEAR POWER PLANT DURING NIIGATA-KEN CHUETSU-OKI EARTHQUAKE)

Unit	Building	Floor	Component	Description	Damage Category
1	Turbine	TG Pedestal Top	Main turbine, main generator	Minor axial displacement of rotor	Significant
	Reactor	2F	High pressure nitrogen gas supply system valves	Leakage from gland bush of regular and emergency isolation valves	Minor
		B5F	-	-	-
2	Turbine	1F	Blowout panel	Loosened and damaged blowout panels	Minor
		TG Pedestal	Main turbine, main generator	Minor axial displacement of rotor, loosened casing cover (low-pressure turbine)	Significant
		B3F	Condensate filter	Damaged discharge pipe book on cation cut-out side of condensate filter	Minor
	Reactor	2F	-	-	-
		B5F	High-conductivity effluent system sampling piping	Deformed sampling pump of high-conductivity effluent system standing water pump room	Minor
3	Turbine	TG Pedestal	Main turbine, main generator	Minor axial displacement of rotor	Significant
		B3F	Condenser vacuum pump hoist	Warped limit switch for running, traces of contact with running wheels on rails	Minor
	Reactor	2F	-	-	-
		B5F	Low-pressure core spray system pump crane	Deformed stopper and gear cover of traverser	Minor
4	Turbine	1F	Blowout panel	Loosened blowout panel	Minor
		TG Pedestal	Main turbine, main generator	Minor axial displacement of rotor	Significant
		B3F	Condenser	Leakage from condenser water chamber communication valve flange	Minor
	Reactor	2F	-	-	-
		B5F	Piping Support	Loosened conduit support in residual heat removal system pump (A) room	Minor
5	Turbine	TG Pedestal	Main turbine, main generator	Minor axial displacement of rotor	Significant
	Reactor	3F	-	-	-
		B4F	-	-	-
6	Reactor	3F	-	-	-
		B3F	-	-	-
7	Turbine	2F	-	-	-
		TG Pedestal	Main turbine, main generator	Minor axial displacement of rotor	Significant
		B2F	Condenser	Scratches between internal small-bore pipes and support	Minor
	Reactor	3F	Support	Deformed power collector support of electric chain block for main steam isolation valve wrapping room	Minor
		B3F	-	-	-

TABLE VII-2. CALCULATED DIP IN STRUCTURES (KASHIWAZAKI-KARIWA NUCLEAR POWER PLANT DURING NIIGATA-KEN CHUETSU-OKI EARTHQUAKE)

Unit	Building	Floor	JMA Seismic Intensity Scale	JMA Instrumental Seismic Intensity	A_{JMA} (Gal)	ZPA (Gal)				Standardized CAV (g-sec)				Ave. Elastic Spectra Acc. (2-10 Hz) (Gal)				Ave. Elastic Spectra Acc. (10-20 Hz) (Gal)			
						NS	EW	UD	Resultant	NS	EW	UD	SRSS	NS	EW	UD	SRSS	NS	EW	UD	SRSS
1	Turbine	T-G Pedestal	7	6.5	634	1862	1459	741	2101	2.58	2.94	1.26	4.11	2817	2936	1095	4213	2674	2055	1182	3573
	Reactor	2F	6U	6.3	500	599	884	394	916	1.37	1.69	0.96	2.38	1131	1550	746	2059	792	1159	888	1661
		B5F	6U	6	346	311	680	408	685	0.97	1.16	0.76	1.70	570	951	737	1331	387	754	545	1008
2	Turbine	1F	6U	6.1	402	431	764	594	799	1.38	1.60	1.12	2.39	1082	1538	1316	2295	621	963	1145	1620
		T-G Pedestal	6U	6.3	513	642	1159	650	1281	1.98	2.25	1.21	3.23	1799	2529	1191	3324	1105	1564	946	2136
		B3F	6U	6	374	387	681	470	684	1.19	1.35	1.04	2.08	877	1133	1199	1868	539	819	722	1218
	Reactor	2F	6U	6.1	416	517	718	412	725	1.37	1.68	0.88	2.34	939	1439	699	1855	676	796	1047	1479
		B5F	6U	6	341	304	606	282	619	0.99	1.13	0.70	1.65	619	818	617	1197	386	660	461	893
3	Turbine	T-G Pedestal	6U	6.4	603	1350	2058	619	2260	2.29	2.88	1.49	3.97	2167	2801	1170	3730	1763	2509	1575	3447
		B3F	6U	6	349	581	549	513	650	1.30	1.35	1.10	2.17	1196	1040	1067	1910	950	1025	733	1578
	Reactor	2F	6U	6.1	406	525	650	518	698	1.46	1.67	1.13	2.49	1102	1372	835	1948	673	782	1351	1700
		B5F	6L	5.9	323	308	384	311	443	1.02	1.12	0.80	1.71	732	815	671	1285	470	619	500	924
4	Turbine	1F	6U	6	380	411	560	549	708	1.30	1.55	1.37	2.44	930	1380	1377	2160	765	771	1179	1603
		T-G Pedestal	6U	6.1	393	614	763	526	874	1.87	1.79	1.35	2.92	1562	1681	1199	2589	1055	1069	1292	1982
		B3F	6U	6	369	348	442	443	601	1.16	1.30	1.22	2.12	758	924	1238	1721	566	527	725	1060
	Reactor	2F	6U	6.2	455	606	713	548	857	1.62	1.71	1.22	2.66	1262	1310	1117	2134	1029	862	1482	1999
		B5F	6U	6	369	310	492	337	546	1.08	1.25	0.81	1.84	827	1021	679	1479	403	624	615	965
5	Turbine	T-G Pedestal	7	6.5	619	1166	1157	533	1224	2.81	3.02	1.04	4.25	2391	2358	900	3477	1986	1460	1146	2718
	Reactor	3F	6U	6	346	472	697	331	713	1.33	1.43	0.90	2.15	921	1245	796	1741	534	879	804	1305
		B4F	6L	5.8	279	277	442	205	451	0.96	1.09	0.67	1.60	604	800	564	1150	351	557	343	743
6	Reactor	3F	6U	6	341	554	545	578	791	1.48	1.34	1.25	2.36	1177	1046	1441	2134	696	622	999	1367
		B3F	6L	5.8	298	271	322	488	490	0.96	1.09	0.91	1.72	608	693	1100	1436	358	401	597	804
7	Turbine	2F	6L	5.8	298	418	506	342	529	1.24	1.38	0.94	2.08	978	1221	909	1809	776	655	700	1234
		T-G Pedestal	6U	6	342	673	1007	362	1013	1.52	1.55	0.89	2.35	1460	1450	785	2202	1072	1344	699	1855
		B2F	6L	5.8	284	318	322	336	401	0.96	1.05	0.76	1.61	622	670	779	1201	460	426	438	764
	Reactor	3F	6L	5.9	327	367	435	464	602	1.21	1.28	1.05	2.05	765	911	1003	1555	526	512	977	1222
		B3F	6L	5.8	291	267	356	355	396	0.92	1.02	0.81	1.59	549	604	814	1153	377	408	479	734

TABLE VII-3. SIGNIFICANT AND MINOR DAMAGES IN STRUCTURE OBSERVED AT THE LOCATIONS OF CALCULATED DIP (ONAGAWA NUCLEAR POWER PLANT DURING 2011 OFF-THE-PACIFIC-COAST-OF-TOHOKU EARTHQUAKE)

Unit	Building	Floor	Observed Damage	Damage Category
1	Reactor	(Roof (61.6m))	(- Damaged steel frame supporting operation cab of overhead bridge crane)	Significant
		Op. Fl. (44.7m)	- Failed information control equipment of refuelling machine - Failed boric acid storage tank level indicating circuit	Minor
		1F (15.0m)	- Main steam safety relief valve (C) position switch relocation - Dislocated support bars for control rod drive system housing support fittings - Dislodged locking device of concrete shield plug	Minor
		Basemat (2.3m)	-	-
2	Reactor	Roof (50.7m)	- Broken or loosened bolts and deformed parts of secondary parts (sub-truss, etc.) of roof truss (-Damaged steel frame supporting operation cab of overhead bridge crane)	Significant
		Op. Fl. (33.2m)	-	-
		1F (15.0m)	- Damaged locking devices of motorized step-back shielding door on B1F (two damaged devices found in total) - Deformed locking device for shielding door inside reactor containment vessel - Dislocated support bars for control rod drive system housing support fittings	Minor
		Basemat (-8.1m)	-	-
3	Reactor	Roof (50.5m)	- Broken or loosened bolts and deformed parts of secondary parts (sub-truss, etc.) of roof truss (- Scratch marks on the surface of rails of overhead bridge crane)	Significant
		Op. Fl. (30.6m)	- Dislodged cable holding caterpillar of refuelling machine - Fallen ground console inside refuelling machine room - Indication failure of operation floor radiation monitor - Loosened tightening bolt of pool water gate	Minor
		1F (15.0m)	- Dislodged locking device of concrete shield plug - Dislocated support bars for control rod drive system housing support fittings	Minor
		Basemat (-8.1m)	(- Inability of suction valve to automatically open fully of high-pressure spray system)	Minor

TABLE VII-4. CALCULATED DIPS IN STRUCTURES (ONAGAWA NUCLEAR POWER PLANT DURING 2011 OFF-THE-PACIFIC-COAST-OF-TOHOKU EARTHQUAKE)

Unit	Building	Floor	JMA Seismic Intensity Scale	JMA Instrumental Seismic Intensity	A_{JMA} (Gal)	ZPA (Gal)				Standardized CAV (g-sec)				Ave. Elastic Spectra Acc. (2-10 Hz) (Gal)				Ave. Elastic Spectra Acc. (10-20 Hz) (Gal)			
						NS	EW	UD	Resultant	NS	EW	UD	SRSS	NS	EW	UD	SRSS	NS	EW	UD	SRSS
1	Reactor	Roof (61.6m)	7	6.8	935	2000	1637	1389	2345	41.66	27.05	10.21	50.71	7697	4460	1320	8993	4192	2820	2998	5875
		Operation floor (44.7m)	6U	6.2	469	1303	985	1184	1430	12.28	12.14	8.15	19.09	2827	2288	1276	3854	2414	1562	2634	3899
		1F (15.0m)	6L	5.7	242	573	574	511	620	5.26	5.81	4.54	9.06	1213	997	792	1758	961	1368	1073	1986
		Basemat (2.3 m)	6L	5.5	212	540	587	399	637	4.01	4.06	3.86	6.89	975	875	681	1476	941	1203	735	1695
2	Reactor	Roof (50.7m)	7	6.5	606	1755	1617	1093	2061	29.56	22.47	12.90	39.31	4613	3338	1501	5889	3565	3420	3128	5847
		Operation floor (33.2m)	6U	6.1	380	916	830	659	1038	11.01	10.33	8.72	17.44	2395	1997	1133	3318	1733	1530	2041	3084
		1F (15.0m)	6L	5.8	280	605	569	330	688	6.51	6.93	4.67	10.59	1409	1234	785	2031	1063	1137	870	1783
		Basemat (-8.1 m)	6L	5.5	204	502	443	304	607	3.97	4.14	3.83	6.90	885	799	640	1353	787	985	693	1439
3	Reactor	Roof (50.5 m)	6U	6.4	556	1869	1579	1004	2036	26.82	21.55	10.26	35.90	4708	3257	1551	5931	3584	3276	2352	5396
		Operation floor (30.6 m)	6L	5.9	332	897	652	806	1109	9.74	9.18	10.68	17.12	2218	1643	1066	2959	1430	1471	2820	3487
		1F (15.0 m)	6L	5.8	284	657	692	547	851	6.97	8.33	5.70	12.27	1674	1389	848	2334	1066	1373	1331	2189
		Basemat (-8.1 m)	6L	5.5	202	545	436	321	666	4.23	4.11	3.10	6.66	1077	773	602	1456	1071	902	633	1537

DEFINITIONS

The following definitions apply only for the purposes of this publication:

Automatic Seismic Trip System (ASTS): A seismic instrumentation system which basically consists of a seismic switch that provides a trigger signal for the automatic shutdown of the nuclear reactor. (Section 2.1.1)

Cumulative Absolute Velocity (CAV): Given an acceleration time-history *a(t)*, CAV is given by the following integral, extended over the complete duration of the time-history (Section 4.1.3):

$$\int |a(t)| \ dt$$

Damage Indicating Parameter (DIP): A parameter that suitably predicts potential damage from earthquake motions. It basically connects the 'earthquake motion intensity', as indicative of damage at a particular location, with the physical properties of the earthquake motion at this location. (Section 2.1.2.2)

Standardized CAV: To make the CAV value representative of strong ground shaking rather than coda waves (small amplitudes that can continue for a long time after the strong shaking), Standardize CAV restricts the integration for computing CAV to 1-second time windows that have amplitudes of at least 0.025 g (Section 4.1.3):

$$\sum_{i=1}^{N} H(pga_i - 0.025) \int_{t_i}^{t_{i+1}} |a(t)| \, dt$$

Where N is the number of 1-second time windows in the time history, pga_i is the peak ground acceleration (g) during the time window i, and $H(x)$ is the Heaviside function (= 1 for x>0, and =0 otherwise)

Seismic Alarm/annunciation System (SAS): A seismic instrumentation system which basically consists of a seismic switch to provide alarms for alerting operators of the potential need for a plant shutdown depending on post-earthquake inspections. (Section 2.1.1)

Seismic Data Acquisition System (SDAS): A complete seismic monitoring system consisting of sensor(s), and data acquisition units that acquire, store, and transmit digital data from one or more systems, including communication hardware and software. (Section 2.1.1)

Units of acceleration

- **Gal** (after Galileo) is a unit of acceleration, equal to 1 cm/sec^2.
- **g** is equal to the acceleration of 980.665 cm/sec^2 and is close to the acceleration of gravity, which depends on the geographical location.

ABBREVIATIONS AND ACRONYMS

AC	alternate current
ASME	American society of mechanical engineers
ASTS	automatic seismic trip system
BWR	boiling water reactor
CAV	cumulative absolute velocity
CFR	code of federal regulations (United States of America)
DBE	design basis earthquake
DC	direct current
DIP	damage indicating parameter
EMS	European macro-seismic scale
EPRI	electric power research institute
JMA	Japan Meteorological Agency
KTA	Kerntechnischer Ausschuss (Germany)
MMI	Modified Mercalli intensity scale
MSK	Medvedev-Sponheuer-Karnik intensity scale
NRC	nuclear regulatory commission (United States of America)
PGA	peak ground acceleration
PWR	pressurized water reactor
SAS	seismic alarm/annunciation system
SDAS	seismic data acquisition system
SSC	structures, systems and components
UHRS	uniform hazard response spectrum
USGS	United States geological survey
ZPA	zero period acceleration

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Vienna, Austria: 18–20 August 2010
Toyama, Japan: 29 November –2 December 2010
Vienna, Austria: 3–4 July 2012
Vienna, Austria: 4–7 June 2013
Vienna, Austria: 25–27 November 2013
Vienna, Austria: 30 January – 1 February 2014

GENERAL NOTES / ACKNOWLEDGEMENTS

The strong earthquake motion time-histories used for the computation of DIPs in this publication were provided by the Japan Association of Earthquake Engineering (nuclear power plants data) and the National Research Institute for Earth Science and Disaster Prevention, Japan (K-NET data).

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