Safety Reports Series

No. 66

Earthquake Preparedness and Response for Nuclear Power Plants

IAEA

International Atomic Energy Agency

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available at the IAEA Internet site

http://www-ns.iaea.org/standards/

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

OTHER SAFETY RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

Reports on safety and protection in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

Other safety related IAEA publications are issued as **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications. Security related publications are issued in the **IAEA Nuclear Security Series**.

EARTHQUAKE PREPAREDNESS AND RESPONSE FOR NUCLEAR POWER PLANTS

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN
ALBANIA
ALGERIA
ANGOLA
ARGENTINA
ARMENIA
AUSTRALIA
AUSTRIA
AZERBAIJAN
BAHRAIN
BANGLADESH
BELARUS
BELGIUM
BELIZE
BENIN
BOLIVIA
BOSNIA AND HERZEGOVINA
BOTSWANA
BRAZIL
BULGARIA
BURKINA FASO
BURUNDI
CAMBODIA
CAMEROON
CANADA
CENTRAL AFRICAN REPUBLIC
CHAD
CHILE
CHINA
COLOMBIA
CONGO
COSTA RICA
CÔTE D'IVOIRE
CROATIA
CUBA
CYPRUS
CZECH REPUBLIC
DEMOCRATIC REPUBLIC OF THE CONGO
DENMARK
DOMINICAN REPUBLIC
ECUADOR
EGYPT
EL SALVADOR
ERITREA
ESTONIA
ETHIOPIA
FINLAND
FRANCE
GABON
GEORGIA
GERMANY
GHANA
GREECE
GUATEMALA
HAITI
HOLY SEE
HONDURAS
HUNGARY
ICELAND
INDIA
INDONESIA
IRAN, ISLAMIC REPUBLIC OF
IRAQ
IRELAND
ISRAEL
ITALY
JAMAICA
JAPAN
JORDAN
KAZAKHSTAN
KENYA
KOREA, REPUBLIC OF
KUWAIT
KYRGYZSTAN
LATVIA
LEBANON
LESOTHO
LIBERIA
LIBYAN ARAB JAMAHIRIYA
LIECHTENSTEIN
LITHUANIA
LUXEMBOURG
MADAGASCAR
MALAWI
MALAYSIA
MALI
MALTA
MARSHALL ISLANDS
MAURITANIA
MAURITIUS
MEXICO
MONACO
MONGOLIA
MONTENEGRO
MOROCCO
MOZAMBIQUE
MYANMAR
NAMIBIA
NEPAL
NETHERLANDS
NEW ZEALAND
NICARAGUA
NIGER
NIGERIA
NORWAY
OMAN
PAKISTAN
PALAU
PANAMA
PARAGUAY
PERU
PHILIPPINES
POLAND
PORTUGAL
QATAR
REPUBLIC OF MOLDOVA
ROMANIA
RUSSIAN FEDERATION
SAUDI ARABIA
SENEGAL
SERBIA
SEYCHELLES
SIERRA LEONE
SINGAPORE
SLOVAKIA
SLOVENIA
SOUTH AFRICA
SPAIN
SRI LANKA
SUDAN
SWEDEN
SWITZERLAND
SYRIAN ARAB REPUBLIC
TAJIKISTAN
THAILAND
THE FORMER YUGOSLAV REPUBLIC OF MACEDONIA
TUNISIA
TURKEY
UGANDA
UKRAINE
UNITED ARAB EMIRATES
UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
UNITED REPUBLIC OF TANZANIA
UNITED STATES OF AMERICA
URUGUAY
UZBEKISTAN
VENEZUELA
VIETNAM
YEMEN
ZAMBIA
ZIMBABWE

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

SAFETY REPORTS SERIES No. 66

EARTHQUAKE PREPAREDNESS AND RESPONSE FOR NUCLEAR POWER PLANTS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2011

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
http://www.iaea.org/books

Printed by the IAEA in Austria
May 2011
STI/PUB/1473

IAEA Library Cataloguing in Publication Data

Earthquake preparedness and response for nuclear power plants. — Vienna : International Atomic Energy Agency, 2011.
p. ; 24 cm. — (Safety reports series, ISSN 1020–6450 ; no. 66)
STI/PUB/1473
ISBN 978–92–0–108810–9
Includes bibliographical references.

1. Nuclear power plants — Earthquake effects — Safety measures. 2. Nuclear power plants — Earthquake resistant design. 3. Earthquake hazard analysis. I. International Atomic Energy Agency. II. Series.

IAEAL 11–00675

FOREWORD

Over the past three decades, a few nuclear power plants have experienced earthquake ground motions. In more recent years, a number of nuclear power plants, mainly in Japan, have been affected by strong earthquakes. In some cases, the measured ground motions have exceeded the design or evaluation bases.

The experience from these events shows that operating plants were shut down immediately following the event and remained shut down for extended periods while comprehensive studies, investigations and evaluations were conducted to assess their safety. In most cases, no significant damage was identified in these nuclear power plant units. In a limited number of cases, upgrades were implemented to meet new definitions of the design basis or requirements for beyond design basis earthquakes.

Those recent events demonstrated the need for formulating specific and detailed criteria and procedures for addressing situations where the original seismic design or evaluation bases are exceeded by actual seismic events. Management and operational response to an earthquake should be planned in advance, taking into account the aforementioned elements and extensively relying on damage assessments at the nuclear power plant itself.

Very few national standards have been established that systematically reflect the concepts mentioned herein, particularly for those cases in which the seismic design bases are significantly exceeded. The seismic safety knowledge and experience of Member States from these recent earthquakes needs to be collected and disseminated to the international nuclear community, thereby providing updated guidance for the actions to be taken in preparation for, and following, a felt earthquake at nuclear power plants.

The intention of this report is to provide guidance to operating organizations in the formulation of an earthquake preparedness and response programme. The programme described herein addresses the full range of seismic ground motions at a site from low level motions (less than the SL-1 seismic design basis) to high level motions (exceeding the SL-2 seismic design basis). This programme may also be used as guidance by regulatory authorities responsible for the decision making process of shutting down and restarting a plant after the occurrence of an earthquake.

This report complements the IAEA Safety Standards as a technical supporting publication relative to the seismic safety of new and existing nuclear installations. The report was developed within the framework of the activities of the International Seismic Safety Centre (ISSC) of the IAEA, and it has been thoroughly reviewed by members of the Scientific Committee of the ISSC.

The contributions of all those who were involved in the drafting and review of this report are greatly appreciated. In particular, the contributions to the preparation of this report provided by the members of the ISSC Scientific Committee and by J.J. Johnson (United States of America) and K. Nagasawa (Japan) are acknowledged. The IAEA officers responsible for this publication were A. Godoy, P. Sollogoub and A. Chigama of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

DISCLAIMER

The present publication reflects feedback on and experience in earthquake preparedness and response accumulated until 2010. The accident at the Fukushima Daiichi nuclear power plant in Japan caused by the disastrous earthquake and tsunami of 11 March 2011 and the consequences of the emergency for people and the environment have to be fully investigated. They are already under study in Japan, at the IAEA and elsewhere. Lessons to be learned for nuclear safety and radiation protection and for emergency preparedness and response will be reflected in the relevant IAEA publications as they are issued in the future.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

At a nuclear power plant that could be affected by an earthquake, planning needs to be performed to identify actions to be taken before an earthquake occurs (pre-earthquake) and after the occurrence of the earthquake (post-earthquake). The principal objective of the plan is to ensure that safety is maintained during and after the earthquake. In addition, actions are identified that lead to decisions concerning shutdown, restart and other longer term activities. Many complex factors contribute to the plan for pre-earthquake activities and post-earthquake actions. Three factors important to the assessment of the seismic safety of an installation are:

(1) The original seismic design basis and the results of any seismic evaluations performed;
(2) The earthquake, and its characteristics, that affected the installation;
(3) The existence of adequate earthquake related operational procedures.

Initially, the response by the operators to a felt earthquake is based on the behaviour of the nuclear power plant systems and on the information transmitted to the control room. A felt earthquake is a vibratory ground motion perceived by nuclear power plant operators in the control room as an earthquake and confirmed by seismic instrumentation or other related information. Typically, it has a free field surface peak ground motion acceleration at the nuclear power plant site greater than 0.02 g (where g (cm^2/s) is the acceleration due to gravity). Inspections of the nuclear power plant by the operators are performed after the safety systems of the nuclear power plant are confirmed to be operating as required and fundamental safety functions are assured. In parallel, other personnel from the operating organization evaluate the seismic design and evaluations of the nuclear power plant in light of the characteristics of the earthquake.

A nuclear power plant that is shut down after experiencing earthquake ground motion may not be restarted for some period of time. This may pose an important challenge to the stable supply of electricity to the local or regional community. The need to ensure the safety of the plant in its shut down condition and after restart is the highest priority. However, reasonable approaches to achieving this goal are emphasized in the programme that is set out in this publication. Successful demonstration of plant safety will help with public acceptance of plant restart. The methodology presented here applies to existing and new nuclear power plants.

The IAEA safety standards address site evaluation and design of new nuclear power plants at the level of the following Safety Requirements:

(a) Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1 [1];
(b) Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. NS-R-3 [2].

In addition, the IAEA has published Safety Guides dedicated to aspects of seismic hazard assessment, seismic design and qualification of structures, systems and components (SSCs) of new nuclear power plants, and the seismic safety evaluation of existing nuclear installations. These Safety Guides are:

(a) Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 [3];
(b) Seismic Design and Qualification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.6 [4];
(c) Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13 [5].

To address the issue of potential earthquake ground motion at the site, it is recommended in Ref. [4] that post-earthquake actions be planned. In this regard, general guidance is provided (paras 7.15–7.19). In Ref. [5], it is recommended that as an end product of all seismic safety evaluations performed (para 2.13) the

> "… actions to be taken before, during and after the occurrence of an earthquake that affects the installation, including arrangements for and actions in operational and management response, analysis of the obtained instrumental seismic records and performed inspections, and the integrity evaluations to be performed as a consequence"

should be identified.

In addition to the need for an adequate action programme to be available and operative for dealing with the occurrence of a felt earthquake, the IAEA Safety Guides recommend, in line with the design requirements, that adequate consideration should be given to events that exceed the design basis, i.e. the so-called 'beyond design basis events' (e.g. Ref. [4], paras 2.39 and 2.40; Ref. [5], para. 2.10).

Recent strong motion earthquakes that have affected nuclear power plants, mainly in Japan, have reinforced the observations that:

(a) There may be significant unquantified conservatism in the seismic analysis and design methods and procedures implemented around the world for nuclear power plants.
(b) High frequency ground motions are not damaging to well engineered SSCs.
(c) On-site seismic instrumentation is essential in addressing the issues that can arise from earthquakes experienced at the site.

These aspects are discussed in detail in Section 2.3. The recent events also showed the need for formulating specific and detailed criteria and procedures for dealing with situations where the original seismic design bases are exceeded by actual seismic events, considering that:

(a) Preparations need to be made for a response commensurate with the level of the ground motion.
(b) The impact of the earthquake on fundamental safety functions needs to be properly and promptly identified.

The nomenclature of Ref. [1] is adopted herein, with respect to the classification of SSCs. Reference [1] further defines the class of items important to safety (ITS) to include those items that are a part of a safety group and/or whose malfunction or failure could lead to radiation exposure of the site personnel or members of the public. Thus, the SSCs are classified into two overall groups:

(1) Items important to safety.
(2) Items not important to safety (NITS). In the present publication, the SSCs not important to safety are further classified into two subgroups as:
    (i) Those required for power generation (RPG);
    (ii) Those not required for power generation (NRPG).

Items required for good plant management, including physical protection system items, may be classified into either ITS or RPG. The RPG designation affects the actions required for restart of the nuclear power plant after an earthquake.

Finally, since very few national standards, see, for example, Refs [6–10], have been established that systematically reflect these concepts, particularly for the case in which the seismic level 2 (SL-2) design basis has been exceeded, the IAEA has prepared the present report as a compilation of available references based on related standards and on actual experience gained by some Member States.

This report complements the IAEA safety standards as a technical supporting publication relative to the seismic safety of new and existing nuclear installations. It provides detailed guidance to Safety Guides NS-G-1.6 [4] and NS-G-2.13 [5] regarding the implementation of post-earthquake response actions as recommended in those guides. Furthermore, the report provides a compilation of available references based on related standards and actual recent experiences from some Member States.

1.2. OBJECTIVE

The objective of this report is to provide updated and detailed guidance on the actions to be taken in preparation for, and following, a felt earthquake at a nuclear power plant, taking into account the recently gained seismic safety knowledge and experience of Member States from strong earthquakes that have affected nuclear installations, in some cases beyond the original seismic design basis.

This guidance is intended to assist operating organizations in the preparation and implementation of an overall pre- and post-earthquake action programme for dealing with situations in accordance with the level of seismic ground motion experienced at the site and the level for which SSCs important to safety in the installation were originally designed or, later, seismically re-evaluated or requalified. This report may also be used as a tool for regulatory bodies responsible during the decision making process for continued operations, shutting down and restarting the plant after a felt earthquake. Detailed guidance is provided on the criteria to be applied regarding:

(a) The evaluation of the seismic safety of the plant following a seismic event, including inspection procedures and protocols;
(b) The identification of the phases and tasks to be performed in accordance with specific plant conditions, including their priorities;
(c) A common and integrated technical framework for defining the applicable acceptance criteria.

The guidance provided in this report covers the full range of earthquake levels that can affect a nuclear power plant, including the case in which the SL-2 seismic design basis has been exceeded.

1.3. SCOPE

The scope of the present report covers the pre-earthquake planning and post-earthquake actions that need to be undertaken for dealing with the occurrence of a felt earthquake that affects a new or existing nuclear power plant. The actions described in the present report include the operations, inspections, investigations, tests and evaluations to be conducted prior to and after a felt earthquake.

Existing nuclear power plants are defined as those plants which are either:

(a) In the operational stage, or
(b) In pre-operational stages for which construction of structures, manufacturing, installation and/or assembly of components and systems, and commissioning activities, have significantly progressed or are fully completed.
(c) At temporary or permanent shutdown stage while nuclear fuel is still within the facility (in the core or the pool).

The scope of this report also covers plants in various stages of decommissioning.

This report describes only briefly the methodologies of seismic safety evaluations conducted in advance or the assessment of seismic safety over a long period after plant restoration, since Ref. [5] and its supporting IAEA publications treat these subjects in more detail.

1.4. STRUCTURE

Section 2 presents the general philosophy of the pre-earthquake planning and post-earthquake action programme. Section 3 discusses the preparatory and planning activities to be performed before an earthquake occurs. Section 4 provides details of post-earthquake shutdown inspections and tests to be conducted mainly in the short term. Section 5 discusses the procedures for the restart of nuclear power plants. Section 6 covers the activities to be performed in the long term, including seismic evaluation and backfitting. In Section 7, the management aspects of these activities are discussed. The publication is completed by the following: Annex I provides examples of tests, inspections and evaluations to be performed; Annex II expands on the lessons learned from past earthquakes experienced in nuclear power plants; Annex III discusses the effect of plastic strain on the fatigue strength of components; and Annex IV provides a list of typical surveillance tests performed on systems of boiling water reactors (BWRs) and pressurized water reactors (PWRs).

2. OVERVIEW OF A POST-EARTHQUAKE ACTION PROGRAMME

2.1. GENERAL CONSIDERATIONS

2.1.1. Objectives of a post-earthquake action programme

When an earthquake is felt at an operating nuclear power plant, immediate and appropriate actions need to be taken in line with prescribed procedures. For such purposes, a specific dedicated programme should be in place in advance, providing a combination of pre-earthquake planning and short and long term post-earthquake actions through:

(a) A rational, experience based, approach for determining the real damage potential of felt and significant earthquakes;
(b) A systematic methodology for assessing the need for plant shutdown and the plant's readiness for restart, based on physical inspections and tests (if the plant has been shut down);
(c) Criteria for assuring the long term integrity of the plant.

The programme is comprehensive, addressing:

(a) The effects of vibratory ground motion on the nuclear power plant site;
(b) Concomitant phenomena, such as river flooding, due to dam failure, coastal flooding due to tsunamis, landslides and failure of the lifelines needed for short and long term normal operation of the plant.

In addition, the programme is comprehensive enough to minimize the likelihood of prolonged plant shutdowns following seismic ground motions that do not damage SSCs important to safety. In all cases, primary emphasis is on the physical and functional condition of the plant, as opposed to analytical evaluations. In many cases, confirmatory analytical evaluations may be performed while the plant is in operation after restart.

It is the intent of the present report that the initiation of the recommended actions as part of such a programme be limited to only those earthquakes that, having been felt at the nuclear power plant, are also considered to be 'significant earthquakes'. A significant earthquake is a felt earthquake having free field surface ground motion characteristics approaching the threshold of damage[1] or malfunction[2] of non-seismically designed SSCs. Some typical definitions of a significant earthquake are earthquakes with a free field surface ground motion greater than 0.05 g or a standardized cumulative absolute velocity (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) or an earthquake with spectral velocities in the 1–2 Hz range greater than 15.24 cm/s.

The designation of a significant earthquake is a function of the site and the seismic design basis of the nuclear power plant, since it may determine the 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.

As a basic reference for such a programme, the following are the general recommendations provided in Ref. [4] that are to be taken into consideration:

(a) *Information to and response by the control room operator:* The control room operators are informed of the occurrence of a felt earthquake by means of the installed seismic instrumentation or by their physical sense. Subsequent responses include an evaluation of the recorded earthquake motion in comparison with the specific design of SSCs important to safety, an evaluation of the damage to the plant through a walkdown, and an evaluation to determine the readiness of the plant for the resumption (or continuation) of operation following the occurrence of a felt earthquake.
(b) *Items to be inspected following the occurrence of a seismic event:* The list of items to be inspected in such a walkdown is consistent with the safety class (either important to safety or not important to safety), seismic categorization and importance to power generation of plant items. Ideally, the items are determined and documented in the pre-earthquake planning step of the procedure. After a felt earthquake has been characterized as a significant earthquake, the nature, extent and location of inspections/tests to be carried out are clearly defined and directly related to the damage that can be expected due to a felt earthquake. For practical reasons, the inspections/tests might be limited to the visual inspection of accessible items; the results of these inspections/tests might be extended by similarity

---

[1] Damage is the change in state from the original configuration of an SSC to an altered degraded state due to the earthquake.

[2] Malfunction is the change in state of monitoring, control or power equipment which results in an erroneous action or indication.

to the seismic behaviour of other items that are important to safety but may not be accessible. In addition to the inspections performed by plant operators and engineers, inspections may be performed by authorized inspection agencies or regulatory bodies.

(c) *Level of inspections:* Different levels of inspections/tests can be defined according to the level of earthquake motion or damage experienced or expected (measured in terms of appropriate analytical parameters or empirical observations); different responsibilities are identified accordingly among the operators, the technical support staff at the plant and within the licensee's organization, and external consultants as needed.

(d) *Involvement of the regulatory body:* The notification to the regulatory body and its involvement in the shutdown or restart of the plant are specified in appropriate pre-earthquake planning procedures.

(e) *Operational procedures:* Recommendations and guidance on operational procedures following an earthquake, including the timing of, responsibilities for and tracking of the necessary actions, are provided in Ref. [11].

Given the background described above and the need for dealing with earthquakes that are felt at existing nuclear power plants, a comprehensive **p**ost-**e**arth**q**uake **a**ction **p**rogramme (PEqAP) is established and implemented with the objectives of providing guidance and specific and detailed procedures to the operating organization, at the plant site and at headquarters.

The programme covers the complete range of seismic ground motions ranging from values lower than those corresponding to seismic level 1 (SL-1 earthquake level) to values higher than those corresponding to SL-2[3].

The PEqAP is based on the following basic principles:

(a) The post-earthquake actions to be taken will facilitate timely decision making concerning the present or future state of the nuclear power plant, for example, to shut down, to continue in operating mode or to restart.

(b) Communication to all stakeholders will be timely and transparent with regard to plant status, actions taken and actions to be taken.

[3] SL-1 and SL-2 earthquake levels are defined in Ref. [4].

(c) A tiered approach is to be employed starting with overall evaluations and proceeding to very detailed evaluations only when required by the situation.
(d) Conforming to these principles, the two basic stages of the programme are:
   (i) *Planning:* Pre-earthquake activities with a view to preparing an appropriate response, as described in Section 3; these activities include all tasks to be performed in advance, before an earthquake occurs.
   (ii) *Response:* Post-earthquake action plans defined as a function of the earthquake felt or ground motion recorded at the site and the observed consequences to the plant, as described in Sections 4–6, after an earthquake has occurred.

The PEqAP is prepared and implemented by the operating organization in agreement with the regulatory body and in accordance with specific regulatory requirements and be known by all parties involved in post-earthquake actions.

2.1.2. Seismic design and evaluation information

Information on the original seismic design of the nuclear power plant needs to be available and accessible in a timely manner to support all aspects of decision making concerning the actions to be taken, for example, shutdown, restart and inspections. If the nuclear power plant was subject to a seismic safety evaluation programme, this information is also needed for the PEqAP. Given that assembling this information may be time consuming, it is organized in the preplanning stage of the PEqAP, such that it will be easily accessible following an earthquake.

The information on the original seismic design includes:

(a) Design basis earthquake(s) (DBE(s)) for which the plant was originally designed, i.e. the SL-1 and the SL-2 earthquake levels as defined in Ref. [4]. The SL-2 earthquake level is defined as a ground motion for which design measures are used to satisfy safety requirements. Depending on the Member State, the SL-1 earthquake level may or may not be treated as a safety requirement. In some Member States, the SL-1 level is only related to operational or inspection requirements. This information is of primary interest because actions to be taken after a felt earthquake depend on the earthquake ground motions that occurred and their relationship to the ground motions corresponding to the SL-1 and SL-2[4] earthquake levels.

(b) In-structure responses (e.g. peak displacements and response spectra) at key locations, such as the foundation and important locations in the structures; comparison of design values with measured or calculated values from the felt earthquake aid in decision making concerning shutdown and restart.

(c) Relative displacements between structures and/or buildings for the comparison of design values with measured or calculated values from the actual earthquake.

The same types of information assembled for the original seismic design basis are assembled for any beyond design basis earthquake (BDBE) evaluations. This information may include seismic evaluation worksheets (SEWSs), calculations and reports. If seismic safety evaluations for a seismic input higher than the original design basis had been performed as recommended in Ref. [5], the results would provide very useful information to understand the earthquake level that the plant can withstand with minimal or no damage.

In particular, if the seismic margin assessment (SMA) approach was applied, the high confidence of low probability of failure (HCLPF) of the plant would be available along with individual HCLPF values for the SSCs. Thus, if an earthquake with ground motion exceeding the SL-1 or SL-2 level occurs, the results of such evaluations would be effective tools to use in evaluating the seismic safety of the nuclear power plant just after the earthquake. In addition to the seismic capacities of SSCs important to safety, these BDBE evaluations may, in some cases, provide estimates of the capacities of SSCs not important to safety. Comparison between the calculated capacities of SSCs important to safety and SSCs not important to safety and of observed, measured or calculated values from the actual earthquake provide information for decision making.

[4] The SL-1 and SL-2 earthquake levels may define two levels of earthquake design: SL-1 corresponds very generally to the operating basis earthquake (OBE) level in the United States of America (USA); SL-2 corresponds very generally to the safe shutdown earthquake (SSE) level in the USA and the Ss earthquake level in Japan; for new nuclear power plants licensed in the USA, the OBE will require a separate explicit design basis only if it exceeds one third of the SSE — otherwise, it is assumed to be satisfied by the seismic design considerations of the SSE; in this case, its main purpose is to define inspection requirements after an earthquake occurs at the site. The Sd and Ss earthquake levels in Japan are a recent development and are currently being specified in more detail.

2.1.3. Establishing criteria for decision making

Decision makers include the operating organization (operators at the plant and other responsible individuals at the plant and headquarters) and the regulatory body having jurisdiction over the nuclear power plant. When considering the measures taken in response to an earthquake, decisions on the possibility of continuing plant operation will depend on the assessment of risk to the health and safety of the public caused by the effects of the earthquake on the plant.

The decision making process for the programme may include decisions on shutdown or restart of the nuclear power plant. For example:

(a) If the plant did not trip or scram during the motion arising from the earthquake, the operators will need to assess whether manual shutdown should be initiated.
(b) If the plant did trip or scram but did not go into shutdown mode, the operators will assess whether manual shutdown needs to be initiated.

The criteria for decision making regarding shutdown or restart are dependent on several factors which are introduced here and fully described in later sections of this report, as follows:

(a) The earthquake level. Basically three levels are considered, ranging from 1 to 3, as defined in Sections 3.3.1 and 3.3.2:
    (1) Earthquake level 1 (EL 1): ground motion less than (<) SL-1;
    (2) Earthquake level 2 (EL 2): ground motion greater than (>) or equal to (=) SL-1 and less than (<) or equal to (=) SL-2;
    (3) Earthquake level 3 (EL 3): ground motion greater than (>) SL-2.
(b) The damage level, which ranges from none to severe damage, with designations of DL 1 to DL 4;
(c) The effects on SSCs important to safety and on SSCs not important to safety.

The category of SSCs important to safety may be further subdivided based on the function of the SSCs, with considerations such as those important to reactor safety compared with those SSCs necessary to maintain the safety of stored spent fuel or high/intermediate/low level radioactive waste.

The combination of these factors leads to the definition of action levels (ranging from 1 to 8), which are directly correlated with the decisions to be made and, consequently, with the actions to be taken.

Ground motion parameters, such as peak ground acceleration and response spectral ordinates, are essential to the seismic design process, but may be poor indicators of malfunction and damage to the SSCs of nuclear power plants. Recent earthquake experience has demonstrated this fact again. However, without viable alternatives describing potential malfunction and failure for the purposes of the methodology described in this report, these ground motion parameters determine the exceedance of SL-1 and SL-2. It is hoped that the need for further research towards defining better damage indicating parameters for nuclear power plant SSCs will be recognized and that efforts will be undertaken in this area in the future. Examples of candidate damage indicating parameters are the CAV and the Japan Meteorological Agency (JMA) intensity.

2.1.4. Ageing considerations

Managing ageing for nuclear power plants means ensuring that the required safety functions are available throughout the service life of the plant, taking into account changes that occur with time and use. This requires addressing physical ageing of SSCs. The ageing process of SSCs is often thought of as causing a degradation of their performance characteristics. In some cases, the ageing process may also result in increases in the capacity of SSCs. Both aspects are taken into account.

Ageing management programmes have been implemented in line with various IAEA recommendations (IAEA Technical Reports Series No. 448 [12], IAEA EBP Report [13] and IAEA Safety Standards Series No. NS-G-2.12 [14]). In addition, plant specific programmes have been submitted as part of the required documentation for the regulatory bodies or regular periodic safety reviews. These activities play a valuable role in improving and updating the current operation and management of ageing of plants.

The engineering assessment used to develop ageing management programmes takes into account the following:

(a) The applicable design basis and regulatory requirements;
(b) Information on the materials, service conditions, stressors, degradation sites, and ageing mechanisms and effects of the structure or component;
(c) Appropriate indicators of relevant ageing phenomena;
(d) Quantitative or qualitative models of relevant ageing phenomena.

Ageing management programmes should have the following generic attributes [14]:

(1) A scope based on an understanding of ageing;
(2) Preventive actions being taken to minimize and control ageing degradation;
(3) Detection of ageing effects;
(4) Monitoring and trending of ageing effects;
(5) Mitigation of ageing effects;
(6) Acceptance criteria;
(7) Corrective actions;
(8) Feedback from operating experience and feedback from research and development results;
(9) Quality management;
(10) As low as reasonably achievable (ALARA) consideration associated with repair or replacement of ageing SSCs.

The results of attributes (3) and (4) above are taken into account in the baseline inspections described in Section 3.7.2 and appropriately documented.

The ageing management programme includes information regarding materials, degradation sites, ageing stressors and environment, ageing mechanisms and effects, inspection and monitoring requirements and methods, mitigation methods, regulatory requirements and acceptance criteria.

A demonstration of the functionality of any safety equipment that performs safety functions under harsh conditions[5] is important for the equipment qualification programme as one of the key elements of the ageing management programme. Service conditions following a postulated initiating event can be significantly different from normal operation conditions and only limited confidence may be derivable from performance during normal operation, pre-operational tests and periodic surveillance tests.

Ageing of specific equipment is managed by using the concept of either ‘qualified life’ or ‘qualified condition’[6] established by equipment qualification.

Environmental qualification (EQ) of equipment is a process to generate testing or analytical evidence to ensure that an item of safety related equipment can perform its safety function under accident conditions (loss of coolant accident, high energy line break, etc.) to meet system performance requirements for the design life of the equipment. Environmental qualification has become part

---

[5] Harsh conditions refer to the operating conditions for the equipment as a result of a postulated initiating event.

[6] The qualifying condition of equipment is expressed in terms of a measurable condition indicator(s) for which it has been demonstrated that the equipment will meet its performance requirements.

of regulatory requirements, and in many Member States an EQ programme has been established and implemented.

More detailed information is provided by the IAEA to help operating organizations and regulatory bodies demonstrate that the effects of ageing are being managed and to help them to assess existing plant programmes. The topics discussed in Refs [14] and [15] are closely related to the areas mentioned above.

Three aspects of ageing are important for this effort:

(1) Defining the ‘as is’ condition of the nuclear power plant at the time the earthquake occurs for evaluation purposes;
(2) Defining the change in state of SSCs due to the earthquake loading conditions imposed;
(3) Assessing the effect of the earthquake on the future reliability, service life and seismic capacity of the nuclear power plant.

Each of these items is discussed in the following sections.

2.1.5. Hidden damage

In performing visual inspections associated with the operator walkdowns, pre-shutdown inspections and post-earthquake inspections, it is important to keep in mind the possibility of the existence of ‘hidden damage’, i.e. damage to the SSCs that cannot be identified visually. Approaches need to be prepared (e.g. non-destructive examinations (NDEs)) to address the possibility of hidden damage in case it is suspected or discovered during testing.

Hidden damage due to earthquakes is classified into two types:

(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, i.e. when performing maintenance, repairs and inspections, and reviewing shaking test data and design information.

In general, hidden damage is detected by disassembly or functional tests. Hence, it is important to prepare in advance an inspection plan which assumes that potential hidden damage may be present. In addition, as is the case for the degradation of corrosion prevention coatings by peel-off, the possibility of accelerated future degradation is considered in preparing maintenance, repair and inspection plans after an earthquake.

(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[7] for metal components, plastic deformation and cracks occurring inside concrete (e.g. around embedded anchorages).

The increase of the usage factor may be estimated on the basis of a seismic fatigue analysis of components suspected to have experienced multiple cycles of significant stresses arising from an earthquake. For earthquakes with ground motions less than an SL-2 earthquake level, increases in fatigue life usage factors have not been shown to be significant. Minor plastic deformation is not significant for the seismic safety of passive SSCs (Annex III). In general, these small perturbations do not have a significant impact on the performance of SSCs, i.e. if the imposed seismic loading conditions are within the allowable criteria established by the applicable code or standard.

However, for earthquake ground motions exceeding the SL-2 level, it is recommended to confirm the integrity of SSCs by conducting analytical evaluations for representative SSCs or by comparing the actual responses of the SSCs with past qualifying test results. In this case, fatigue life usage stemming from the earthquake may be estimated by an analytical evaluation, for example, for piping systems. For cracks inside concrete, the evaluation of a realistic conservative case may be used to bound the potential impact of the earthquake on the behaviour of the structural element.

[7] The fatigue usage factor is typically defined as the ratio between the number of estimated cycles and the allowable number of cycles (n/N). Details are provided in the Boiler and Pressure Vessel Codes (BPVCs) for nuclear class components; for example, in the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB.

2.1.6. Seismic instrumentation: Manual or automatic shutdown system

2.1.6.1. Seismic instrumentation

Seismic instrumentation — an array of strong motion accelerographs installed at the plant site — plays a key role in collecting site specific seismic instrumental data during the life cycle of the nuclear power plant from site selection, to site characterization and to the operational stage until decommissioning.

The site specific seismic instrumental data are required for various purposes, ranging from helping in the assessment of the seismic hazard at the site to recording the actual seismic response of SSCs in the event of a felt earthquake, and assisting in the consequential post-earthquake actions.

Reference [3] also recommends that a local network of weak motion sensitive seismographs (of both short period and broadband period types) be installed and operated near the site, i.e. the zone within about 25–40 km around the plant site, in order to acquire detailed information on potential seismogenic sources for seismotectonic interpretation. This local network is usually connected to the regional and national seismological networks.

Regarding the seismic instrumentation to be installed at the nuclear power plant site, in particular a network or array of strong motion accelerographs, Ref. [4] recommends the installation of such instrumentation for the following reasons (Ref. [4], para. 7.1):

“(a) For structural monitoring: to collect data on the dynamic behaviour of 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.”

This report addresses the seismic instrumentation corresponding to strong motion accelerographs installed at the nuclear plant site.

2.1.6.2. Manual or automatic shutdown system

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 (USA), 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 (NSSS) comes.

2.1.6.3. Elements for decision making on which approach to use

In general, the decision on which approach to use, either manual or automatic shutdown as a result of a felt earthquake, will depend on seismological considerations, structural and technological plant features, economic consequences and public acceptance aspects.

In a more specific sense, Ref. [4] recommends that a number of issues govern the decision on whether to have an automatic scram system or to rely on a combination of plant trip mechanisms and operator actions, supported by measurements of the earthquake motion at the site or characteristics of the felt earthquake. These issues include:

(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 may be used as an additional protective measure, particularly in the case where the DBE levels have been increased as a result of the seismic hazard evaluation.
(c) *Safety considerations relating to spurious scrams*. It is not recommended to use an automatic system for nuclear power plants with high levels of ambient noise, including noise induced by other plant equipment.
(d) *Evaluation of the effects of the superposition of earthquake induced loading conditions with other transient loads 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 and continued operation.
(e) *Other reactor trips.* A reactor trip may be initiated for other reasons; for example, turbine vibration detectors, water level monitoring systems in large tanks and damage to outside electrical lines that trigger a load rejection transient.
(f) *Broad ranging safety issues relating to the consequences for the State of the shutdown of a plant immediately following an earthquake.* In Member States with a limited electricity grid and few other types of power generation plants that are seismically qualified, 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 is a challenge to the safety of the plant.
(g) *Level of operator confidence and reliability.* For manual action, the operator plays an important part in the decisions on post-earthquake actions and therefore needs to be adequately trained for this contingency.
(h) *Public acceptance*. This issue is also an important aspect which may influence the decision on the approach to adopt. It should 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.

For a plant in a particular Member State, one or a combination of these factors will lead to the decision as to whether or not to employ an automatic scram system. The relative importance of each of the issues may depend on the particular Member State.

2.1.7. Multi-unit sites

Multiple units located at the same site are common throughout the world; on average, there are more than two units per site worldwide. Moreover, it is expected that the number of units per site will increase in future, considering that planned and/or under construction new units are being located at sites of operating plants. The common cause nature of an earthquake affects all units simultaneously. Examples of multi-unit sites affected by earthquakes are highlighted in Section 2.3, for example, Kashiwazaki-Kariwa nuclear power plant with seven units, Onagawa nuclear power plant with three units, and Hamaoka nuclear power plant with three units, all of them in Japan.

The programme described in this report is designed to be applicable to a single unit. However, in the development and implementation of the overall emergency plan for the single unit, important aspects of the multi-unit site need to be considered. Some of these aspects are:

(a) The infrastructure on-site and off-site is affected simultaneously. Transport routes off-site and on-site may be disrupted, affecting the emergency response plan.
(b) Off-site power may be unavailable. If one switchyard is the interface to the grid, all units will be similarly affected by loss of off-site power due to the earthquake.
(c) Normal on-site power from adjacent nuclear power plant units or from a conventional power generation plant located with the nuclear power plant unit of interest will probably not be available due to plant shutdown because of the earthquake. The expected behaviour is dependent on the size of the ground motion on-site.
(d) The use of shared systems by two units in close proximity may be severely limited. The assumption that redundancy exists in safety or other systems because of systems shared by units may not be applicable.

In general, the same free field ground motion is experienced by all units. Although, in some instances, the area of the site may be large, the soil and rock configuration may vary over the site area, and the units may be sited some distance apart. In this case, the free field ground motion applicable to the individual units may differ. An example of this case is the Kashiwazaki-Kariwa nuclear power plant, where Units 1–4 experienced a different free field motion than Units 5–7 during the Niigata-ken Chuetsu-Oki earthquake (NCOE).

The seismic design bases of units at the same site may be different depending on their vintage. Older units are more likely to have DBE ground motions that are less than those of units of more recent vintage. This reflects the evolution of knowledge about seismic hazards and their modelling around the world. Therefore, the definitions of SL-1 and SL-2 may differ for each unit, and the actions required to be performed for each may differ substantially.

The actual performance of each unit may be different as a function of the seismic design, evaluation and upgrades that may have been implemented. Evaluations and upgrades may be a result of a periodic safety review or other programmes initiated by the regulatory body; for example, programmes to assure that appropriate levels of seismic safety margin with respect to the seismic design/seismic hazard for the unit and the site exist. A note of caution is that the seismic design is a function of an integrated methodology, including definition of ground motion, location at which the ground motion is applied, aspects of soil–structure interaction (SSI) modelling and parameters, modelling of structures including damping values, modelling and analysis of subsystems including methodology and parameter values (e.g. damping) and design code acceptance criteria. Hence, the capacity of one unit relative to another is not defined by the DBE alone. Conclusions concerning relative seismic capacity should not be drawn prematurely, i.e. prior to the completion of the appropriate inspections/evaluations.

Finally, the effort required by the regulatory body to verify that a unit is ready for restart may be influenced by the performance of all the units on a site. This possibility should be recognized.

2.1.8. Overall emergency plan: Summary

The *overall emergency plan* describes the objectives, policy and concept of operations for the response to an emergency in the event of an earthquake. It describes the structure of the response and the roles and responsibilities of the nuclear power plant operating organization and government regulatory bodies for a systematic, coordinated and effective response.

The term *overall emergency plan* denotes a broad plan that encompasses: State and local government interactions; interactions with the public, media and other stakeholders; and operating organization actions both on-site and at headquarters. Specific operational procedures of a plant in the event of the occurrence of an earthquake tier down from the overall emergency plan to operating and emergency procedures.

The key elements of the overall emergency plan to be defined as part of the programme are:

(a) The scope and purpose of the plan;
(b) The composition of the emergency response team for the operating organization, i.e.:

   (i) The management located at headquarters,
   (ii) On-site plant management and operators;

(c) Definitions of roles and responsibilities as a function of the situation and time;
(d) Pre-earthquake planning;
(e) Post-earthquake actions — short term and long term;
(f) Communication with local and national government agencies, including regulatory body and other stakeholders, for example, the media and the public; redundant emergency communications methods need to be considered;
(g) Decision making considerations, for example, regarding shutdown, repairs, evaluations, upgrades and restart;
(h) Education, training and exercises.

2.1.9. Human reliability

The following aspects of human reliability are also considered when establishing the post-earthquake action programme:

(a) The likelihood of human errors or inattention by the operators when faced with the increased stress arising from the occurrence of the earthquake.
(b) The ability of the operators to perform their short term post-earthquake required functions when faced with the potential of earthquake caused injury to off-site or on-site personnel (see below), or earthquake caused damage within the nuclear power plant site and to SSCs. The possibility of not having an adequately staffed team for later shifts due to, for example, personal or family injuries, material damage or failure to access roads should be taken into account.
(c) The earthquake is an external event with a regional impact outside the plant boundary; hence, the concern of the operators for possible injuries to family, relatives and friends, and possible damage to their personal property may adversely affect their performance.

2.1.10. Considerations other than safety for shutdown, restart and upgrading

2.1.10.1. General considerations

Nuclear power plants are considered, in general, more likely than other power plants (fossil and hydroelectric) to be capable of generating power following an earthquake. The seismic design requirements for a high level of earthquake loading and the high standards of construction for nuclear power plants improve the potential for continued operation, i.e. no plant trip or scram, or the potential for quick plant restart once damage assessments have been made. Because of this higher seismic capacity, there is a high probability that in the region affected by the earthquake only the nuclear power plants will remain on-line.

The need for power generation does not take precedence over technical specifications, plant operating procedures or the requirements of the operating licence. In the event that no plant trip or scram occurs and no damage or abnormal plant conditions arise, the control room supervisor may decide, with guidance from the load dispatcher, to remain on-line temporarily until alternative power sources are available to the grid. Similarly, in the event that a plant trip or scram occurs, the decision may be made to restart once the reviews of recorded data and damage (or malfunction) assessments have been completed and no potential safety problems have been identified.

The overall emergency plan provides guidance for these situations. It is expected that the operating organization will interact with local, regional and national regulatory agencies in connection with any such decisions, and specifically with the regulatory body. Continued operation or return to operation of a nuclear power plant will only be carried out when safety is assured.

2.1.10.2. Systems which may trigger a reactor trip

In a nuclear power plant, a reactor trip may be triggered by a number of systems as follows:

(a) Seismic switches. The choice of the seismic trigger system depends on the response time requirements and on the reliability of the trigger with regard to spurious triggering. The redundancy of the triggering channels and the logic of trip actuation are selected as a function of the seismic risk versus the impact of spurious triggering. The seismic trigger system also conforms to the design requirements for all the reactor protection systems. The seismic trigger level is usually set to a fraction of the plant seismic design.

(b) Other trip systems. Several systems of protection are likely to initiate a reactor trip in the case of an earthquake, independently of any seismic switch. Typical systems that may respond to earthquake induced motions include:
   (i) Turbine vibration or shaft deflection detectors, which will trigger a turbine trip.
   (ii) Destruction or damage to the outside electrical grid, leading to a load rejection transient. Some plant designs permit a plant to be switched to isolated operation without reactor trip, by reducing the power to the in-house or self-consumption level. Other plants will trip in the case of full or partial load rejection.

(iii) The water level monitoring systems in liquid containers may be perturbed by the sloshing induced by a seismic event. These perturbations may trigger false alarms, or even a plant trip.
(iv) High neutron flux scram (for BWRs).

2.1.10.3. Potential consequences of shutdown

Conditions may exist, following an earthquake which exceeds the SL-1 level, such that a discontinuation of power generation could result in loss of critical lifeline functions and potential loss of life. Such conditions could include:

— Extreme weather conditions;
— Loss of other generating plants;
— Power blackout;
— Disruption to rescue operations and emergency services (e.g. fire brigade, medical services and civil defence).

2.1.10.4. Restart

The need for power in the region to address situations that have significant consequences for the public will contribute to the decision making for restart, for example, the timing of restart and power level.

2.1.10.5. Upgrades

The timing and scope of evaluations, repairs and upgrades to the plant may be dependent on the need for power in the region, resulting in conditions being set for restart, for example, repairs and upgrades to be completed before restart, timing of restart and power level. Cost–benefit studies may contribute to the decision making process.

2.2. POST-EARTHQUAKE ACTION PROGRAMME

2.2.1. General process

The guidelines provided in this report form the basis of a comprehensive programme (PEqAP) for preparedness and response by a nuclear power plant to an earthquake. Flow charts depicting a high level view of the general process of the programme are shown in Figs 1 and 2.

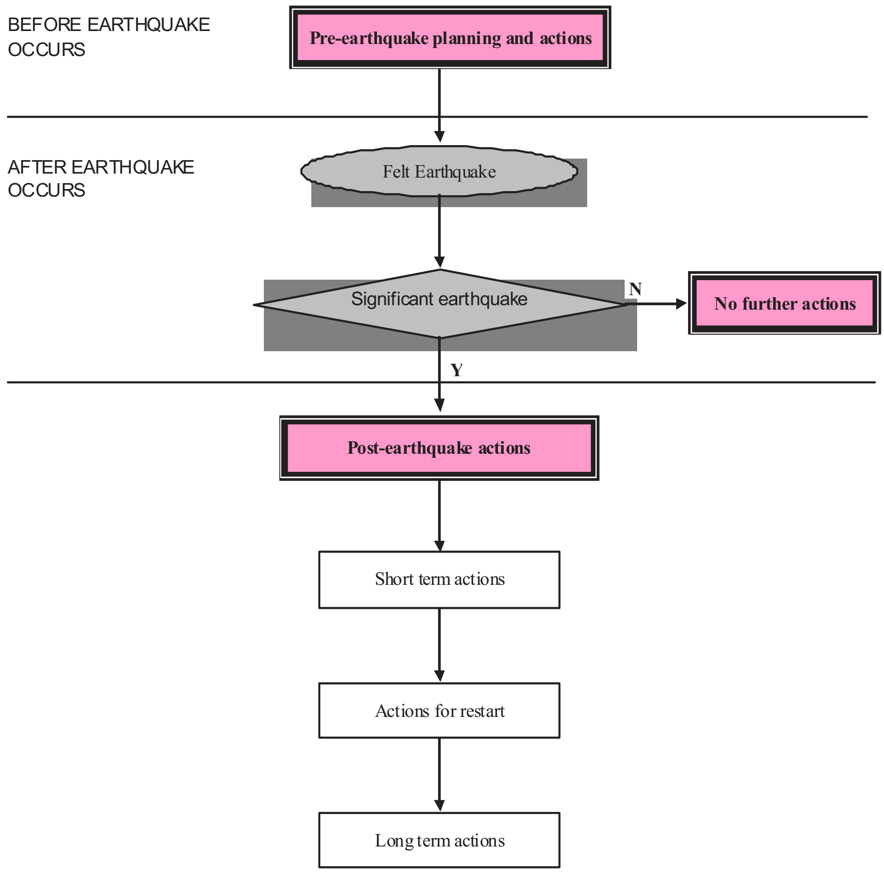


FIG. 1. General process of the PEqAP (Y: yes; N: no).

The recommended PEqAP is divided into two major stages, as follows:

(1) Planning, including all pre-earthquake activities;
(2) Response, i.e. short term actions, post-shutdown inspections and tests, and long term evaluations.

In Sections 4 and 5, flow diagrams outlining the specific actions recommended for the short term and for expanded inspections are included to expand this overall view. Long term actions are described in Section 6. The actions illustrated in these figures and described in this report are intended to be used by nuclear power plant owner/operators in the development of plant specific procedures that specify the appropriate types and level of response to be made in

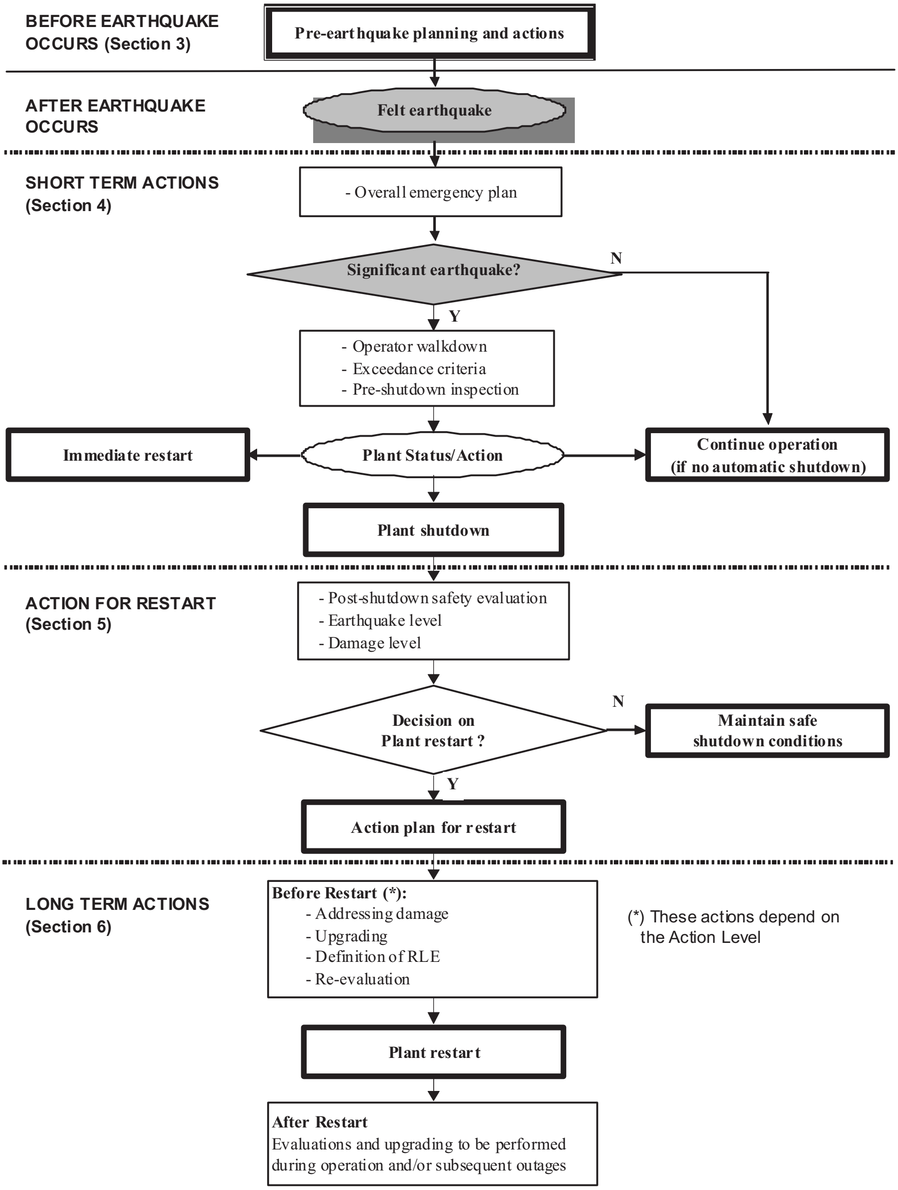


FIG. 2. Flow chart of the general process of the PEqAP (Y: yes; N: no; RLE: review level earthquake).

the event that an earthquake is felt at the plant. A summary of the planning, actions, responsibilities and other considerations which are covered in this report, and which have to be addressed in the plant specific earthquake response procedures, is given in Table 1.

TABLE 1. OUTLINE OF PLANT SPECIFIC EARTHQUAKE RESPONSE PROCEDURE

Purpose

To provide guidance to the nuclear plant owner/operators and the Member State's regulatory body on preparations, responsibilities and response to an earthquake. In particular:

- Activities after automatic shutdown;
- Activities for manual shutdown:
  — Assessment of the need to shut the plant down,
  — Preparation for an orderly shutdown;
- Assessment of readiness for restart.

Pre-earthquake planning (see Section 3)

The procedure needs to describe the equipment, capabilities and actions needed in preparation for (in advance of) an earthquake, as follows:

- Development of the overall emergency plan, including considerations on plant shutdown due to an earthquake;
- Selection, installation, maintenance and monitoring of plant seismic instrumentation to implement the exceedance criterion, or alternative actions if such instrumentation is not installed or does not provide an indication in the control room;
- A method/procedure for processing records from seismic instruments in a timely manner;
- A preselected sample of structures and equipment to be inspected in the event of an earthquake;
- Baseline inspection results for the above structures and equipment;
- Plant SL-1 and SL-2 DBE levels, any BDBE or review level earthquake (RLE) used for seismic safety evaluation and reference information;
- Maintaining and updating the numerical analytical models and computer software.

Post-earthquake responsibilities

Plant operations

- Confirmation of a felt and significant earthquake
- Stabilization of plant in accordance with normal and/or emergency operating procedures
- Activation of the overall emergency plan
- Plant walkdown inspection
- Determination of exceedance of SL-1 and/or SL-2 earthquake levels
- Pre-shutdown evaluation
- Plant shutdown
- Prescribed inspections, surveillance tests and evaluations
- Plant restart

TABLE 1. OUTLINE OF PLANT SPECIFIC EARTHQUAKE RESPONSE PROCEDURE (cont.)

Engineers with earthquake related experience

- Processing of recorded ground motions in a timely manner to determine the earthquake levels
- Detailed inspections of preselected equipment and structures
- Expanded inspections and specification of tests
- Reconciliation of results of inspections with available instrumentation data
- Long term confirmatory evaluations

Action initiators

Earthquake response

- Activation of seismic instruments, or
- Consensus of operators that a felt earthquake has occurred

Decision to shut down plant

- SL-1 and/or SL-2 exceedance
- Damage to SSCs

Readiness for restart

- Physical condition of plant
- Demonstrated functionality of equipment

Long term plant integrity

- Confirmatory long term evaluations
- Supplemental functional tests, inspections and NDEs

Recommended actions

Short term actions

- Fundamental safety functions assured
- Stable operation
- Implementation of overall emergency plan, as required
- Operator walkdown inspections (damage level determined)
- Evaluation or processing of ground motion records (earthquake level determined)
- Determination of action level
- Shutdown decision

Pre-shutdown checks (if warranted)

- Orderly shutdown (if warranted)

TABLE 1. OUTLINE OF PLANT SPECIFIC EARTHQUAKE RESPONSE PROCEDURE (cont.)

Post-shutdown activities • Visual inspections of a preselected sample of equipment • Confirmation of damage level and action level • Expanded visual inspections, non-destructive examinations (NDEs), comparative analyses, etc. (if warranted) • Surveillance tests to meet limiting conditions for operation (if warranted) • Authorization for restart (if required) • Restart
Long term actions • Perform seismic hazard evaluation (define the RLE or probabilistic seismic hazard assessment (PSHA)) • Perform SMA, seismic probabilistic safety assessment (SPSA) or alternative approaches, which may be demonstrated to be acceptable — Member States may prefer a specific method • Upgrade selected SSCs (if necessary) • Decision on which long term actions are to be performed before restart and which after restart

A criterion for determining whether the ground motion generated by an earthquake exceeds the SL-1 or SL-2 level is defined in later sections. Briefly, if the exceedance of the SL-1 level is confirmed in accordance with the established criterion, then shutdown and further inspection of the plant is recommended. If the SL-1 level exceedance criterion is not exceeded and no significant damage is found during operator walkdown inspections, manual shutdown of the plant is not considered necessary.

Actions in the earthquake response plan are based on several important premises and concepts, as follows:

(a) The behaviour of the plant and instrumental information recorded at the plant itself are the best indicators of the intensity of the earthquake at the plant site, rather than damage information from nearby communities or recorded ground motion distant from the site.
(b) Detailed inspections of preselected equipment and structures, which are inspected prior to the earthquake (baseline inspections), together with the use of special seismic damage scales (if developed and available) for nuclear power plants, can be used to quantify the potential damage caused

by the earthquake and to establish the extent of the inspections, tests and evaluations necessary to demonstrate readiness for restart.

(c) In some instances, for example, for earthquakes with ground motion greater than the SL-2 level, additional inspections and tests may be required to demonstrate the integrity and functionality of the SSCs important to safety.

2.2.2. Planning and activities during the pre-earthquake stage

The planning and basic activities to be conducted at the pre-earthquake stage are:

(a) Development of an overall emergency plan defining the roles and responsibilities of on-site and off-site operating organization personnel and the interaction with the regulatory body and other stakeholders in the event of the occurrence of an earthquake.

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

   (i) A *felt earthquake* is any earthquake that produces vibratory ground motion at the site perceived by nuclear power plant operators in the control room as an earthquake and confirmed by seismic instrumentation or other related information. Typically, seismic instrumentation installed at nuclear power plants is triggered at peak ground acceleration values of 0.01 g to 0.02 g.

   (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) structures, systems or components. 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.

(c) Assessment of existing or new seismic instrumentation in terms of its capability to provide the information and data required for implementing the programme development plan for the use of the recorded data based on the programme described in the present report.

(d) Definition of the earthquake levels affecting the nuclear power plant, i.e. Earthquake Levels 1, 2 and 3 as a function of the DBE levels SL-1 and SL-2; or evaluation levels (RLEs).
(e) Definition of the damage levels for the nuclear power plant and the specific SSCs to which the damage levels apply.
(f) Definition of action levels as a function of earthquake levels and damage levels.
(g) (i) Preselection of SSCs that provide the scope of the short term evaluations to be performed;
  (ii) Development of the baseline data of the ‘as is’ configurations of the preselected SSCs, taking into account the ageing management programme;
  (iii) To set up the baseline data in a form that is easily accessible by the post-earthquake evaluation teams, for example, electronic copies of photographs, figures, drawings, etc., preparation in advance of the post-earthquake inspection worksheets.
(h) (i) Performance of exercises and training for potential participants in the implementation of the overall emergency plan;
  (ii) To increase operator awareness of earthquakes and their potential effects on SSCs (e.g., typical malfunction or damage to equipment).
(i) Maintenance, and updating of the as-is conditions if so necessary, of numerical analytical models and corresponding computer software, used for calculating the structural response of buildings and components.

2.2.3. Post-earthquake actions

2.2.3.1. Short term actions

In the short term, after an earthquake has been felt at an operating nuclear power plant, prompt actions should be taken. In this regard, it is important to immediately determine the parameters of the earthquake (its magnitude, location of its epicentre, etc.), and especially the recorded motions on the site, and any damage or malfunction of SSCs (particularly those important to safety). Whether or not the plant can be maintained in a safe and stable condition should be determined.

If the plant does not shut down automatically during an earthquake, it is necessary to decide whether to continue operation or to initiate plant shutdown (emergency or normal shutdown). If an earthquake causes the plant to shut down automatically (automatic shutdown by seismic scram or other means), it will be necessary to maintain stable cold shutdown conditions and to decide on further measures on the basis of consideration of the damage done to SSCs.

As part of the initial post-earthquake response, in the short term, operators and available plant employees need to take the following prompt measures to enable making the decisions mentioned above:

(a) To implement the overall emergency plan (according to the earthquake level, specific regulatory requirements, etc.).
(b) To carry out immediate actions (within a time frame established in conjunction with the regulatory body, for example, within 24 hours, depending on the task to be accomplished and the operability requirements of the safety systems):
   (i) Immediate operator actions:
      — Observation of plant parameters and response to plant alarms,
      — Walkdown inspections (Section 4.3.2),
      — Initiation of emergency response procedures;
   (ii) Immediate actions for engineers (those on-site or at headquarters):
      — Processing of recorded motions (free field and in-structure motions) and comparison with design values,
      — Assessment of exceedance or non-exceedance of SL-1 or SL-2 levels and other damage indicating parameters to determine whether shutdown is required.
(c) To perform pre-shutdown inspections preparatory to a decision on manual shutdown:
   (i) Functional confirmation of safety of shutdown systems (Section 4.3.2),
   (ii) Availability of power sources — on-site or off-site (Section 4.3.3),
   (iii) Availability of on-site emergency power sources (Section 4.3.4),
   (iv) Control and confirmation of other conditions prescribed by procedures and technical specifications.

As a result of the short term actions, decisions are taken regarding: (a) continuation of operation, (b) immediate plant shutdown, or (c) immediate restart of the plant.

2.2.3.2. Actions for restart or extended shutdown: Action levels

If the plant is either automatically shut down following an earthquake or by an operator decision taken as a result of previous short term actions, the next stage is to carry out the activities required for deciding whether the plant can restart in the short term or be maintained in another normal operational state (as described in Section 5.4). A complete and comprehensive set of inspections and tests aimed at assessing the integrity of the installation should be carried out. To ensure continued stable power operation after restart, it is necessary to address issues, such as repairs, inspections and evaluations, some of which may be required before restart and others which may be performed after restart. The purposes of the post-earthquake inspections and tests are to identify the plant damage level on the basis of the physical damage and/or functional damage to SSCs. This report provides guidelines for performing visual inspections and tests of SSCs. As a result, decisions concerning further actions including restart, long term evaluation and upgrading are made.

The following approach is recommended for performing post-shutdown inspections and tests:

(a) Post-trip review.
(b) Post-shutdown SSC safety evaluation (inspection, analysis and/or test):
    (i) Decision on the strategy to be followed;
    (ii) Comparative analysis;
    (iii) Post-shutdown inspection and tests:
        — Formation of inspection teams and briefing teams on inspection plans and procedures (teams to meet on a regular basis to ensure consistency of evaluations (possibly daily as inspections begin and less frequently thereafter)),
        — Initial focused inspections and tests,
        — Expanded inspections and tests;
    (iv) Decision on the damage level;
    (v) Acceptance criteria.
(c) Decision on whether to restart or to maintain safe shutdown.
(d) Plan of actions for restart:
    (i) Definition of the action level for restart;
    (ii) Addressing damage;
    (iii) Surveillance (or inspection) tests;
    (iv) Startup tests;
    (v) Documentation.
(e) Plan of actions for maintaining safe shutdown.

The action levels for restart are defined on the basis of the earthquake level and the damage level. Eight action levels are defined as indicated in Table 2. The action level determines the sequence of pre- and post-restart activities. It is allowable that the repair of damage to SSCs that are not important to safety and not required for power generation be performed after plant restart.

Furthermore, the regulatory body may agree to allow seismic evaluation and upgrading (Section 6), if necessary, to be conducted after plant restart if the appropriate seismic margin of the plant has been preliminarily confirmed according to Ref. [5].

TABLE 2. POST-EARTHQUAKE ACTION LEVELS

Damage level (DL) \ Earthquake level (EL)		EL 1	EL 2	EL 3
		EL < SL-1	SL-1 ≤ EL ≤ SL-2	EL > SL-2
DL 1	• No significant damage to important to safety SSCs • No significant damage to not important to safety SSCs	—	Action level 1	Action level 5
DL 2	• No significant damage to important to safety SSCs • Significant damage to not important to safety SSCs, not required for power generation	—	Action level 2	Action level 6
DL 3	• No significant damage to important to safety SSCs • Significant damage to not important to safety SSCs, required for power generation	Action level 3		Action level 7
DL 4	• Significant damage to important to safety SSCs	Action level 4		Action level 8

This preliminary confirmation may be accomplished through a combination of activities, such as the plant evaluation for the felt earthquake in conjunction with the similarity of the nuclear power plant of interest to others for which SMAs or seismic probabilistic safety assessments (SPSAs) have been performed. Other alternative approaches to establishing preliminary estimates of seismic margin are acceptable. The regulatory body may require concurrence in these activities and their timing relative to restart.

Section 5 of the present report indicates specific and detailed procedures for plant restart in relation to a decision on the action level for restart, inspection, analyses and/or testing required for a comprehensive safety evaluation.

2.2.3.3. *Long term actions*

More extensive seismic hazard evaluations in the short term or in the long term may be required depending on the compatibility of the event with the seismotectonic model used for determining the original seismic design basis motion and the specific characteristics of the earthquake motion (e.g., frequency content) and the consequences for SSCs on the site. Some of the longer term actions may be performed after plant restart. Specifically, some of the following actions may be required depending on the size of the ground motion and the condition of SSCs:

- — Evaluation of the seismic hazard at the site;
- — Evaluation of the seismic safety of the plant;
- — Upgrading of SSCs, if necessary.

2.3. LESSONS LEARNED FROM RECENT STRONG MOTION EARTHQUAKES

2.3.1. Niigataken Chuetsu-oki earthquake (16 July 2007)

2.3.1.1. *Outline of the Kashiwazaki-Kariwa nuclear power plant*

The Kashiwazaki-Kariwa nuclear power plant consists of seven units: Units 1–5 are of the BWR5 type (1100 MW(e)), and Units 6 and 7 are of the ABWR type (1356 MW(e)) and are operated by the Tokyo Electric Power Company (TEPCO).

When the Niigataken Chuetsu-oki earthquake occurred, three units (Units 3, 4 and 7) were in operation at 100% power, one unit (No. 2) was in startup mode and the remaining units (Nos 1, 5 and 6) were shut down for periodic inspection.

2.3.1.2. *Design criteria: DBE (SL-1 and SL-2) ground motion*

Unit 1 was designed to JEAG 4601-1970 guidelines [16]. The dynamic ground motions used for the seismic design of Unit 1 were generated using seismic records registered in the USA (California) and denoted by El Centro, Taft and Golden Gate. Vertical seismic motion was taken into account by applying a static seismic force in the vertical direction.

Units 2 to 7 were built after the publication of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities [17] and the revision to JEAG 4601 published in 1987 [18]. In particular, these units were designed to withstand both the static load and dynamic loads (S1 and S2), which were assumed to be defined at a rock outcrop (V_s = 700 m/s) postulated to exist far below the foundation mat.

Structures, systems and components are classified into three seismic categories (Classes A, B and C) based on their safety function. The SSCs of the Class A category, which are of particular safety importance, such as those constituting pressure boundaries, are further subcategorized as Class As. In general, the S1 and S2 dynamic ground motions are used for Class As and S1 for Class A SSCs.

2.3.1.3. Earthquake characteristics

The Niigataken Chuetsu-oki earthquake (M_w = 6.6 and M_{JMA} = 6.8) occurred at around 10.13 a.m. on 16 July 2007. The Kashiwazaki-Kariwa nuclear power plant (seven units with a total output of 8212 MW(e)) is located about 16 km from the epicentre and the JMA intensity observed at the plant yard ground surface was 7.

2.3.1.4. Comparison of earthquake design parameters with recorded data

A total of approximately 90 accelerographs were installed at outdoor ground surfaces and vertical shafts as well as in the buildings of each unit at different levels. The maximum accelerations observed exceeded the maximum accelerations calculated for the design basis ground motions for all units. Moreover, the in-structure response spectra (ISRS) of observed records significantly exceeded those calculated for the design in almost all frequency ranges.

2.3.1.5. Inspections and evaluations

(a) Inspections

Immediately after the earthquake occurred, a plant walkdown inspection was conducted by plant operators and maintenance staff to determine through visual inspections if damage had occurred to SSCs important to safety. It was confirmed that no malfunction, damage or failure had occurred to SSCs important to safety. The operating units were automatically shut down and achieved cold shutdown.

The initial inspections were followed by extensive additional plant walkdowns by seismic engineers for all units. No malfunction, damage or failure of Class A or As SSCs was found. Only minor damage to non-Class A or As SSCs was found, for example, a house transformer fire of Unit 3 and failure of fire extinguishing piping in the yard.

Following the earthquake, inspections were implemented as planned, including: visual inspections common to each facility; basic inspections such as operational tests; disassembled inspection due to the results of these basic inspections; response analysis results; and NDEs.

(b) Evaluations

From an engineering perspective, modifications were made to the finite element model used for the design of the building, to improve the prediction capability of the model, thereby better reproducing the instrumental recordings. The revised model was benchmarked against the recorded motion and then used to generate the seismic responses of SSCs important to safety for evaluation purposes.

As of September 2009, all SSCs important to safety for Units 6 and 7 were evaluated for the Niigataken Chuetsu-oki earthquake. All SSCs important to safety were confirmed to have experienced load or stress levels within their elastic limits. Eventually, the structural integrity of all SSCs important to safety for Units 6 and 7 was assured in a comprehensive manner on the basis of the inspection and analysis results. The approach to be used in the evaluation of Units 1–5 will be the same.

2.3.1.6. Seismic safety evaluation

As a result of the revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Reactor Facilities (September 2006) [19] and the experience of the Niigataken Chuetsu-oki (NCO) earthquake, TEPCO defined new DBEs for the Kashiwazaki-Kariwa site. Upgrading of SSCs important to safety to the newly defined earthquake standard was conducted and completed.

The IAEA conducted three safety review missions to the plant, immediately after the occurrence of the NCO earthquake, in August 2007, and in January and December 2008. All IAEA mission reports are available on the IAEA web page.

2.3.1.7. Restart status

After completing the above described inspection and evaluations and reporting these to the regulatory body, Units 7, 6, 1 and 5 were restarted. Unit 7 restarted on 9 May 2009. Unit 6 restarted on 26 August 2009, Unit 1 on 31 May 2010 and Unit 5 on 18 November 2010.

2.3.2. Noto Hanto earthquake: (25 March 2007)

2.3.2.1. Outline of the Shika nuclear power plant

The Shika nuclear power plant consists of two units with a total output of 1898 MW(e)): Unit 1 is of the BWR5 type (540 MW(e)) and Unit 2 is of the ABWR type (1358 MW(e)), and they are operated by the Hokuriku Electric Power Company. When the Noto Hanto earthquake occurred, these two units were both shut down for periodic inspection.

2.3.2.2. Design criteria: DBE (SL-1 and SL-2) ground motion

The standards and regulations used for the Shika nuclear power plant are the same as those for Units 2–7 of the Kashiwazaki-Kariwa nuclear power plant.

Units 1 and 2 of the Shika plant were built and designed to conform to Refs [17, 18]. In particular, these units were designed in such a manner as to withstand both the static load and dynamic loads (S1 and S2), which were assumed to be defined at the rock outcrop (V_s = 1500 m/s) postulated to exist below the foundation mat.

2.3.2.3. Earthquake characteristics

The Noto Hanto earthquake (M_{JMA} = 6.9) occurred at around 9.42 a.m. on 25 March 2007. The Shika nuclear power plant is located about 18 km from the epicentre, and the maximum value of the JMA instrumental intensity calculated from the observed data was as high as approximately 7.

2.3.2.4. Comparison of earthquake design parameters with recorded data

A total of 48 accelerographs are installed, including four in outdoor vertical shafts and 22 in the buildings of each unit. The maximum response accelerations observed in the Noto Hanto earthquake were below the maximum response accelerations estimated from the design basis ground motions in both units on the foundation mats. However, the response spectra of observed records were found to be in excess of those of design basis ground motions in some frequency ranges for the S2 earthquake.

2.3.2.5. Inspections and evaluations

(a) Inspections

Immediately after the earthquake occurred, a plant walkdown inspection was conducted by plant operators and maintenance staff to confirm equipment damage, in accordance with the manual of procedures to be followed after an earthquake. No safety related facility malfunction, damage or failure was confirmed to have occurred. Operational tests were then conducted for equipment and systems in a sequential manner: these confirmed that there was no unusual behaviour of SSCs in the plant.

(b) Evaluations

From an engineering perspective, modifications were made to the building model used for the design, to improve the prediction capability of the model, thereby better representing the recorded motions. Once the building response analysis results obtained using this adjusted model reasonably reproduced the actual building responses during the Noto Hanto earthquake, the seismic responses of the SSCs important to safety were also evaluated.

On the basis of these response analyses, the SSCs important to safety (seismic Class A and As) of Units 1 and 2 were demonstrated to have remained in the elastic range of behaviour.

2.3.2.6. Restart status

Additional seismic margin enhancement work was conducted for Unit 2 in addition to the work already undertaken before the occurrence of the earthquake, and Unit 2 restarted operation 12 months after the earthquake. Unit 1 remained shut down for a longer period for reasons independent of the earthquake.

2.3.3. Earthquake off the coast of Miyagiken (16 August 2005)

2.3.3.1. Outline of the Onagawa nuclear power plant

The Onagawa nuclear power plant consists of three units: Unit 1 is of the BWR4 type (524 MW(e)), and Units 2 and 3 are of the BWR5 type (825 MW(e)), with a total output of 2174 MW(e)). It is operated by the Tohoku Electric Power Company. When the earthquake off the coast of Miyagiken (16 August 2005) occurred, all of these units were in operation at 100% output.

2.3.3.2. Design criteria: DBE (SL-1 and SL-2) ground motion

The standards and regulations applied to the Onagawa nuclear power plant are the same as those applied to the Kashiwazaki-Kariwa and Shika nuclear power plants. Units 2 and 3 were built after the publication of the seismic design guidelines [16] and were designed to conform to the guidance in Refs [17, 18]. In particular, these units were designed in such a manner as to withstand both the static load and dynamic loads (S1 and S2), which were assumed to be defined at a rock outcrop (V_s = 1500 m/s) postulated to exist below the foundation mat.

Unit 1 was built before Ref. [17] was published, and therefore, was not designed to conform to the guidelines in Refs [16] or [17]. However, its seismic design was performed in accordance with methods similar to those specified later in Refs [17] and [18]. The dynamic ground motions employed were generated on the basis of the records observed in El Centro, Taft and Onagawa. Additionally, vertical static seismic forces were taken into consideration.

2.3.3.3. Earthquake characteristics

The earthquake off the coast of Miyagiken (M_{JMA} = 7.2) occurred at around 11:46 a.m. on 16 August 2005, and the site of the Onagawa plant is located about 73 km from the epicentre.

2.3.3.4. Comparison between earthquake design parameters and recorded data

A total of approximately 180 accelerographs are installed in vertical shafts as well as in the buildings of each unit. The maximum response accelerations observed in the earthquake off the coast of Miyagiken on 16 August 2005 were below the maximum response accelerations estimated from the design basis ground motions in all units. The response spectra of observed records on the reactor building foundation mat were found to be in excess of those of design basis ground motions in some frequency ranges.

2.3.3.5. Inspections and evaluations

(a) Inspections

Immediately after the earthquake occurred, a plant walkdown was conducted by plant operators and maintenance staff to determine equipment malfunction, damage or failure, in accordance with the manual of procedures to be followed after an earthquake. No safety related facility failure was confirmed to have occurred. Operational tests were then conducted for equipment and systems in a sequential manner and these confirmed that there was nothing unusual in the plant.

(b) Evaluations

From an engineering perspective, modifications were made to the building model used for the design to improve the prediction capability of the model, thereby better representing the recorded motions.

Once the building response analysis results obtained using this adjusted model reasonably reproduced the actual building responses during the earthquake off the coast of Miyagiken, the seismic responses of SSCs important to safety were also evaluated.

This evaluation confirmed that these facilities were all within their elastic limits.

2.3.3.6. Restart status

After the earthquake off the coast of Miyagiken, the seismic safety of the three units was confirmed. Restart of the units occurred as follows: Unit 2 restarted after five months, Unit 3 restarted after seven months and Unit 1 restarted after eleven months.

2.3.4. Summary

Valuable experience has been gained over the last three decades concerning the effects of earthquakes on nuclear power plant SSCs.

Table 3 summarizes key aspects of this experience of nuclear power plants subjected to actual earthquakes, with special attention to the action level (as defined in Section 3.5) versus the time to restart.

Three major lessons, learned from the experience gained through nuclear power plants being subjected to actual earthquakes, are obvious:

(1) There may be significant unquantified conservatism in the seismic analysis and design methods and procedures implemented by the nuclear industry. These conservatisms are difficult to take into account because they are currently unquantified. Efforts to quantify and understand these conservatisms will allow them to be taken into account in the future in the design process and in the evaluation of the design margins for BDBE motions.

TABLE 3. SUMMARY OF EARTHQUAKE EXPERIENCE AT NUCLEAR POWER PLANTS

Nuclear power plant	Earthquake	Action level	Time to restart	Reference
Hamaoka, Japan	Surugawan (2009)	—	Unit 3: 2 months Unit 4: 1 month Unit 5: 18 months	—
Kashiwazaki-Kariwa, Japan	NCOE (2007)	7b	Unit 1: 35 months Unit 5: 40 months Unit 6: 25 months Unit 7: 22 months	Annex II
Shika, Japan	Noto Hanto (2007)	6c	Unit 2: 1 year Unit 1: 1 year + time due to factors other than the earthquake	Annex II
Onagawa, Japan	Miyagi offshore (2005)	6a	Unit 1: 11 months Unit 2: 5 months Unit 3: 7 months	Annex II
Krško, Slovenia	(1989)	5	Manually shut down and soon restarted	Ref. [20]
Metsamor, Armenia	Spitak (1988)	1	Plant shut down by Government decision	Annex II
Perry, Ohio, USA	Leroy (1986)	6a	At pre-operational stage. Startup delayed for over 4 months	Ref. [21]
V.C. Summer, South Carolina, USA	Reservoir induced seismicity (1977–1979)	1 or 5	Not shut down	Ref. [22]

(2) High frequency ground motions are not damaging to engineered SSCs. Approaches need to be developed and accepted by the nuclear industry (operators and regulators), to take into account this repeated observation. Significant resources are ‘wasted’ when extensive seismic capacity evaluations of SSCs are conducted when high frequency exceedances are observed following the occurrence of an earthquake. The methodology described in the present report attempts to address this issue. In addition, there is a need to identify and validate damage-indicating parameters that

will be better descriptors of damage to nuclear power plant SSCs than the observed acceleration values.

(3) Seismic instrumentation is essential in addressing issues that can arise if and when an earthquake occurs and is felt and is significant at a nuclear power plant site. The importance of the availability of seismic records and of numerical analytical models for prompt actions after the occurrence of an earthquake should be emphasized.

3. PRE-EARTHQUAKE PLANNING

3.1. OVERALL EMERGENCY PLAN

Successful management of a situation such as the occurrence of a felt or significant earthquake at a nuclear power plant site hinges on an overall emergency plan being in place. Therefore, it is recommended as part of the PEqAP that an overall emergency plan be developed and implemented in a timely manner.

The key elements of the overall emergency plan are:

(a) Scope and purpose of the plan — this encompasses the time frame between the occurrence of the earthquake and the completion of all the actions required to bring the plant into its appropriate end state.

(b) The composition of the emergency response team for the operating organization:
   (i) Management at headquarters plays a significant role in interacting with regulators, the media, the public and other stakeholders. Procedures for transparent timely communication with all stakeholders are developed and put in place. Lessons learned from recent events, particularly the response to the public concern in the case of the Kashiwazaki-Kariwa nuclear power plant, in Japan, in July 2007, shows the importance of clear, prompt, precise and reliable information in the hours after the event.
   (ii) On-site plant management and operators — the plant manager on the site or the designee — will manage the on-site activities with appropriate communication with headquarters and the regulatory body. Operators play a key role as described later in pre-shutdown and post-shutdown inspections and decision making. Organizational and

individual roles and responsibilities need to be defined as a function of the situation and time.

(c) Pre-earthquake planning.
(d) Post-earthquake actions — short term and long term.
(e) Consideration of earthquake induced hazards, for example, fire, flooding, landslide and subsidence.
(f) Communication with local, regional and national regulators and other stakeholders, for example, the media and public. In communicating information about the situation at the nuclear power plant, it is helpful to present comparative examples that relate situations more familiar to the public with the situation at the nuclear power plant site. One example is that of an aircraft where multiple redundant systems exist: automatic pilot/manual control; multiple engines on an aircraft — multiple safety systems at a plant; non-safety aircraft systems (galley, drinking water, air conditioning, etc.) — non-safety plant systems (transformers, water tanks, on-site roads, etc.).
(g) Decision making considerations, for example, regarding shutdown, repairs, upgrades and restart, to be in accordance with a plan based on action levels. The action levels (described in Section 3.4) dictate many of the considerations for decision making.
(h) A well established management system to be maintained, including careful consideration of documentation and recording of all actions taken.
(i) Training and exercises are essential to the success of the programme, especially as personnel and their responsibilities change.

Pre-earthquake planning is an important first step for the successful implementation of the overall emergency plan following an earthquake. Pre-earthquake planning constitutes the preparatory phase in which all elements for coping with the situation are properly established.

The elements of pre-earthquake planning discussed here are as follows:

(a) Installation and setting up of seismic instrumentation;
(b) Establishment of the criteria for exceedance of the DBEs (SL-1 and SL-2) and definition of the earthquake levels for the purpose of identifying future actions;
(c) Definition of the response terms *malfunction, damage and failure* and *significant damage* as they relate to SSCs important to safety and SSCs not important to safety;
(d) Definition of different response levels of SSCs important to safety and SSCs for the purpose of identifying future actions;

(e) Definition of action levels as a function of earthquake ground motion and damage;
(f) Definition of the criteria for the selection of SSCs for pre- and post-earthquake inspections.

Although it is not discussed here in detail, an important pre-earthquake planning activity is the maintenance, and updating to the as-is conditions if so necessary, of all numerical analytical models used for calculating the structural response of buildings and components, including the computer software.

3.2. SEISMIC INSTRUMENTATION

3.2.1. Recording earthquake motions at the nuclear power plant site

For post-earthquake response, seismic instrumentation can play a significant role in the decision making process, for example, decisions regarding shutdown and restart, including actions to be performed for the reasons indicated in Section 2.1.6.1. The installation, maintenance, upgrading and operability of seismic instrumentation are key elements of pre-earthquake planning for nuclear power plants in areas of significant earthquake potential.

In general, seismic instrumentation consists of a network or array of triaxial time-history strong-motion accelerographs located in the free field on the ground surface or within the soil or rock profile, on structure foundations and/or in the structures of interest. The amount and type of instrumentation may be a function of the perceived seismic activity in the region of the site. However, even in areas of perceived low seismic activity, minimum seismic instrumentation is recommended to be installed and operable.

The term ‘free field motion’ refers to ground motion which is minimally affected by the vibration of nearby structures due to the earthquake. Other seismic instrumentation of value could be instruments that record velocities and displacements and/or produce parameters that are directly correlated to damage, such as CAV or JMA intensity.

The term ‘triaxial’ refers to the ability of one instrument, or a group of instruments, to record motions in three orthogonal directions, one of which is vertical.

Regarding the type of seismic instrumentation to be installed at the site, Ref. [4] recommends the following:

“7.2. The amount of seismic instrumentation to be installed, its safety classification and its seismic categorization should be decided on the basis of the relevance of the postulated seismic initiating event for system design and, in general, on the basis of the instrumentation’s significance for the emergency procedures for the plant. Seismic monitoring and automatic scram systems, when installed, should be properly classified and adequate redundancy should be provided.

“7.3. The seismic instruments installed at the nuclear power plant should be calibrated and maintained in accordance with written maintenance procedures.

“7.4. A minimum amount of seismic instrumentation should 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 should be considered for sites having an SL-2 free field acceleration equal to or greater than 0.25 g.

“7.5. The collection and analysis of data should be carried out on a regular basis to support the periodic safety review of the plant.”

Seismic instruments that record ground motion provide data for comparison with seismic design parameters, such as seismic design ground motion (SL-1 and SL-2 earthquake levels), and other parameters, such as CAV or JMA intensity. Seismic instrumentation that records motion on the foundations of structures or at locations in structures provides data for comparison with seismic design parameters, such as in-structure time histories of motion or response spectra.

These records also provide data for further evaluation of SSCs as described in later sections. In selecting the location of these instruments, consideration has to be given to the end use of the data. For example:

(a) Will the data be used to calculate motions at other locations in the structure? If so, are translational inputs at a single location adequate or should a small array of instruments be placed to permit definition of rotations?
(b) Will the data be used for comparison with in-structure design basis data?

(c) Is the instrument located such that equipment–structure interaction effects are minimal?

Recorded data from seismic instrumentation in the free field, on the foundation and in-structure that are necessary for use by the operations staff are transmitted and annunciated to the control room staff in a timely manner.

The specific and detailed requirements for seismic instrumentation to be installed in a nuclear power plant site are usually specified by the regulatory body. In addition to installation requirements (location, types and capabilities of instruments), requirements for maintenance, upgrading and operability are essential.

Once again, lessons learned from the Kashiwazaki-Kariwa NPP case at the time of the July 2007 event shows the significance of these requirements.

3.2.2. Automatic scram trip system

An automatic scram trip system (ASTS) is generally installed at plants located in high seismicity areas, such as Japan. However, the WWER type reactors of the former Soviet Union design located in the Russian Federation and eastern European countries, many of which are located in low to moderate seismicity areas, were recommended to install an ASTS. The following parameters are important considerations for the selection and implementation of an ASTS.

3.2.2.1. Seismic hazard and seismic characteristics of plant

The seismic capacity of the plant needs to be considered, i.e. as originally designed, or as a result of plant upgrades or requalification, compared with the seismic hazard. The comparison between the DBE (SL-2) and the seismic hazard is important for considering whether to install an automatic trip; for example, if the probability of occurrence of an SL-2 is relatively high, an automatic trip may be desirable. The SL-1 earthquake is important for determination of the triggering level, i.e. the earthquake level at which actions may be required by the operator and, possibly, the regulator.

3.2.2.2. Time to scram

The time to scram the reactor (i.e. the time to insert the control rods or equivalent systems) needs to be compared with the expected duration of the earthquake. Automatic scram is best utilized if it leads to reactor trip before the maximum shaking of the earthquake. If not, the transients that will result from the trip will be superimposed on the seismic transient and may be challenging for the plant equipment. The strong motion shaking during a potentially damaging earthquake typically initiates within 10 s of the felt shaking at the site. The response time of the triggers is also added to the time to scram for the comparison mentioned.

3.2.2.3. Elements to be considered in the design of ASTSs

ASTS sensors and associated circuitry are designed with appropriate logic in order to fulfil the intended purpose of the system, i.e. shutdown of the reactor system within the required time to fulfil the corresponding operational parameters. The design needs to be of proven technology to maximize the reliability of the system and minimize malfunctions such as spurious actuation. Seismic monitoring and automatic scram systems, when installed, are safety classified, and adequate redundancy is provided. The seismic instruments installed at the nuclear power plant are calibrated and maintained in accordance with written maintenance procedures.

The sensors are located at points for which design response spectra and time histories are available. Typically free field and foundation levels are the locations chosen for sensors. The settings chosen on these sensors need to be compatible with the corresponding parameters. These trigger levels are adapted to the locations of the sensors in the plant, in accordance with the seismic dynamic analysis carried out. One of two trigger levels is generally used for ASTSs:

(1) The first trigger level is chosen to be close to the SL-1 level, usually associated with operational limits. Significant SSCs are not expected to malfunction, be damaged or fail at levels lower than the SL-2 earthquake level; however, regulatory requirements and/or operational limits for SSCs may require shutdown for inspections as discussed in later sections. A trigger level less than or equal to the SL-1 level is the most typical case.
(2) The second trigger level may be close to the SL-2 level. This second trigger level is generally considered for plants where only an SL-2 earthquake is specified, or where the decision is made that automatic shutdown is only required at the SL-2 level and only if the plant continues to operate after experiencing ground motion levels higher than SL-1. Earthquakes at the SL-2 level and higher are expected to cause disruption of off-site facilities, such as loss of off-site power and disruption of water supplies.

Settings lower than the SL-1 earthquake level may be selected for interim periods in cases where seismic capacity assessment and upgrading work for the nuclear power plant are under way. However, considering the inherent seismic resistance of the nuclear power plant SSCs, a lower bound trigger of 0.05 g peak ground acceleration is suggested for the seismic scram systems.

The ASTS is considered as an element of the safety systems of the nuclear power plant. Accordingly, the system needs to conform to all relevant requirements of seismic qualification. The control panel of the system needs to be easily accessible by the operator.

Data processing methods, and short term and long term post-earthquake actions taken on the basis of the actuation of scram systems, or on the basis of monitoring, are discussed throughout this report.

Note that response actions by nuclear power plant operators to an earthquake are most likely required, regardless of the decision to install an ASTS (Section 4).

3.3. CRITERIA FOR EXCEEDANCE OF DBE AND EARTHQUAKE LEVELS

3.3.1. Criteria for determining exceedance of DBEs

It is recommended that the plant shutdown criteria consist of multiple elements; they are presented in Section 3.6. Two of these involve the exceedance of the DBEs, i.e. SL-1 and SL-2, and/or the exceedance of a threshold value of a damage indicating parameter.

(a) Exceedance of a DBE (SL-1 or SL-2). A comparison between the DBE characteristics and the felt earthquake needs to be performed. In general, this entails a comparison of the DBE response spectra with comparable response spectra generated for the recorded motions of the actual earthquake. This evaluation to determine whether SL-1 or SL-2 was exceeded needs to be performed using data obtained from the seismic instrumentation of the plant. A tiered approach to judging exceedances of SL-1 or SL-2 should be employed. First, the free field data are considered. The three components of recorded free field ground motion, i.e. two horizontal and the vertical, are processed obtaining response spectra for comparison with the SL-1 and SL-2 design basis ground response spectra. Assuming the recorded motions are acceleration–time histories, pseudo-acceleration or absolute acceleration response spectra can be generated through routine calculations. The response spectra are calculated at appropriate frequency increments. For example, calculate the response spectra over the frequency range from 0.1 Hz to 50 Hz at a frequency interval equivalent to a total of 100 frequencies per decade (0.1–1 Hz, 1–10 Hz, 10–100 Hz) evenly spaced over the logarithmic frequency scale of each decade. The response spectra for each of the three components is calculated and compared at an appropriate damping level; a typical value is 5% critical damping. The SL-1 or SL-2 design ground response spectra have been exceeded if, for any frequency, the calculated response spectral ordinate exceeds the design value by more than 5%. Some regulatory guidance, for example, Ref. [6], specifies that this check be made over the low frequency range only. If free field records are not available at the prescribed location of the design basis ground motion, the response spectra for motions recorded on the foundations or in-structure may be used for purposes of comparison. Alternatively, the free field motions at the specified location may be calculated using the recorded motions and compared with the design spectra. An example of this latter case is the specified SL-1 or SL-2 ground motion at the specified depth in the soil foundation media. The equivalent values for the felt earthquake may then be calculated and compared.

(b) 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 intensity.

A combination of response spectra checks and the damage indicating parameter check is preferred.

3.3.2. Earthquake levels

Earthquake levels are defined on the basis of recorded motions in the free field, on foundations and in-structure compared with the corresponding data for the design basis SL-1 and SL-2 earthquake levels. The principal comparison is with the free field motion. If the free field motion is not available, comparisons with foundation and in-structure motions may be used.

The size of the earthquake motions is a key parameter in relation to decisions as to plant shutdown, subsequent evaluations and restart. For plants not explicitly designed to the SL-1 earthquake level, the assumption may be made that the SL-1 check applies to an earthquake defined as 0.33 times the SL-2 earthquake level. The earthquake levels are:

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.

For applications described later in Sections 4–6, it is recommended that Earthquake Level 3 be further separated into Earthquake Levels 3a, 3b and 3c according to the frequency characteristics of the ground motion, i.e. high frequency, mid-amplified frequency range and low frequency range. If exceedance occurs in multiple frequency ranges, this should be properly taken into account.

The importance of the further subcategorization into Earthquake Levels 3a, 3b and 3c is that the short term and long term actions will differ depending on the ground motion frequency characteristics with respect to the soil–structure system frequencies.

On the basis of several factors, high frequency earthquake ground motions (Earthquake Level 3a) have minimal adverse effects on nuclear power plants and other industrial facilities.

These factors include: (i) observed lack of damage to nuclear power plants subjected to high frequency ground motions (e.g., the Onagawa, Perry and V.C. Summer cases, Section 2.3) and the results of detailed post-earthquake evaluations; (ii) extensive analytical studies performed to date demonstrating a lack of damage to ductile SSCs subjected to high frequency motions; (iii) evaluations of shake table test data.

Earthquake Level 3a may induce malfunctioning of some electrical equipment or instrumentation and control systems, which may require evaluation; low frequency ground motions (Earthquake Level 3c) are observed to have minimal effects on typical nuclear power plant SSCs as evidenced by the Shika case (Section 2.3), which was subjected to low frequency ground motion for which evaluations (analysis and testing) demonstrated lack of damage to SSCs.

Thus, significantly fewer post-earthquake actions are required for Earthquake Level 3a; fewer post-earthquake actions are required for Earthquake Level 3c; but continued extensive evaluations are required for Earthquake Level 3b. In general, the specific frequency ranges defining Earthquake Levels 3a, 3b and 3c are dependent on the dynamic characteristics of the nuclear power plant

SSCs. Earthquake Level 3b requires extensive evaluations. These ranges may require approval from the regulatory body.[8]

3.4. MALFUNCTION AND DAMAGE

3.4.1. Malfunction, damage and significant damage

For the purposes of the present report, the terms 'malfunction', 'damage' and 'significant damage' are defined below.

(a) *Malfunction* is the inability of a structure, system or component to perform its required function. Malfunction may be due to physical damage or to the temporary loading environment of the earthquake, for example, shaking causing 'chatter' of electrical devices.

(b) *Damage* is defined as the change in state from the original configuration of an SSC to an altered degraded state due to the earthquake. Damage can be categorized as minor damage or significant damage. Minor damage (e.g., slight impact deformations, deformation of insulation or hairline cracks in concrete), even if caused by the earthquake, is not considered to be significant and does not initiate actions to be taken.

(c) *Significant damage (physical* or *functional*) is considered to be damage which has the potential to adversely affect the operability, functionality or reliability of SSCs. Examples of significant damage are given in Table 4.
The term 'significant damage' may refer to:
   (i) Significant damage to SSCs important to safety;
   (ii) Significant damage to SSCs not important to safety:
      - NITS — required for power generation (RPG),
      - NITS — not required for power generation (NRPG).

(d) *Physical damage* is damage to the physical characteristics of the SSC, in contrast to functional damage or malfunction. Physical damage may be visually detected or may be hidden, may cause immediate consequences or affect the long term ability or life of the SSC, may be significant or not, and may cause functional failure of items such as structures and pressure boundaries.

---

[8] On the basis of judgment, one example of the frequency ranges associated with Earthquake Levels 3a, 3b and 3c is 3a (greater than 10 Hz), 3b (2–10 Hz) and 3c (less than 2 Hz).

TABLE 4. EXAMPLES OF SIGNIFICANT DAMAGE

Concrete structures	New or earthquake induced cracks in concrete greater than a prescribed threshold (e.g., see Table 8), 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 (Annex III).
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.

3.4.2. Damage levels

Damage levels, as defined here, are one set of parameters, which determine the required actions in response to the earthquake with the end goal being nuclear power plant restart (action levels). 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:

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).

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).

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).

The use of damage indicating parameters, such as CAV, is desirable initially in establishing the damage level. Post-earthquake inspections and tests, when implemented, further determine or confirm whether or not significant damage has occurred. These results may lead to a revision of the damage levels and, consequently, a revision of the action level.

It is recommended to identify SSCs important to safety and SSCs not important to safety specifically or by category as a pre-earthquake planning task. Table 5 contains examples of typical SSCs in the various categories.

3.5. DEFINITION OF ACTION LEVELS

The recommended post-earthquake actions are a function of the earthquake ground motion level at the site and the damage experienced and observed in the nuclear power plant. Table 2 presented the combinations of earthquake and damage levels leading to the definition of action levels. Action levels 1 to 8 are summarized in Table 6 and described in detail in Section 5.2.

For a nuclear power plant that was built without a seismic design, the equivalents to action levels are determined by the operating organization and approved by the regulatory body.

TABLE 5. CATEGORIZATION OF NUCLEAR POWER PLANT SSCs AND DECISION ON DAMAGE LEVEL (EXAMPLE)

	Structures			Systems and components		
	Safety related (ITS)	Non-safety-related (NITS)		Safety related (ITS)	Non-safety-related (NITS)	
		Important (RPG)	Not important (others)		Important (RPG)	Not important (Others)
SSCs	Building (related to safety) Primary containment facilities Control room Ultimate heat sink Emergency control centre	Building Heat sink	Harbour facilities Internal plant roads Building	Safe shutdown system Primary boundary: ∘ Reactor pressure vessel ∘ Primary coolant piping ∘ Steam generator Instrumentation and control: ∘ Reactivity control ∘ Inventory control Residual heat removal system Emergency core cooling system (ECCS) Power supply: ∘ Switch gear ∘ Bus/distribution cables Emergency power supply ∘ Diesel generator ∘ DC battery Fire protection system (in the case that firewall is not installed)	Balance of plant systems: ∘ Feedwater system ∘ Turbine/generator Transformer Power line Power supply: ∘ Switchgear ∘ Bus/distribution cable Switchyard Reactor building crane	Fire protection system Filtered water system Crane Demineralized water system Incinerator Laundry systems

TABLE 5. CATEGORIZATION OF NUCLEAR POWER PLANT SSCs AND DECISION ON DAMAGE LEVEL (EXAMPLE) (cont.)

		Structures			Systems and components		
		Safety related (ITS)	Non-safety-related (NITS)		Safety related (ITS)	Non-safety-related (NITS)	
			Important (RPG)	Not important (others)		Important (RPG)	Not important (Others)
Damage level	1						
	2			X*			X
	3		X	Not concerned		X	Not concerned
	4	X	Not concerned	Not concerned	X	Not concerned	Not concerned

* X: significant damage found.

TABLE 6. ACTION LEVELS AND ACTIONS TO BE TAKEN

Action level	Actions to be taken	Notes
Action Level 1 EL 2 No damage or malfunction	1. Initial focused inspections and tests 2. If successful, restart	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed
Action Level 2 EL 2 Damage to NITS NRPG SSCs	1. Initial focused inspections and tests 2. If successful, restart 3. Repair or replace NITS NRPG after restart	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed
Action Level 3 EL 1 and EL 2 Damage to NITS RPG SSCs	1. Initial focused inspections and tests 2. If successful, repair or replace NITS RPG SSCs 3. Restart	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed
Action Level 4 EL 1 and EL 2 Damage to ITS SSCs Possible damage to NITS SSCs	1. Expanded inspections and tests 2. Evaluate root cause of ITS SSC damage 3. Corrective action based on root cause analysis results: redefine input; repair, upgrade, replace, requalify ITS SSCs; verify SSC capacity 4. Repair or replace damaged NITS RPG SSCs 5. Restart 6. Repair or replace damaged NITS NRPG SSCs 7. Possibly redefine RLE	Expanded inspections and tests include initial focused inspections and tests
Action Level 5 EL 3 No damage or malfunction	1. Initial focused inspections and tests successful 2. Restart 3. Re-evaluate seismic hazard after restart, if deemed necessary	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed

TABLE 6. ACTION LEVELS AND ACTIONS TO BE TAKEN (cont.)

Action level	Actions to be taken	Notes
Action Level 6a EL 3 EL 3a (high frequency) No damage to ITS SSCs Damage to NITS NRPG SSCs	1. Initial focused inspections and tests successful 2. Restart 3. Repair or replace NITS NRPG SSCs after restart 4. Re-evaluate seismic hazard after restart, if deemed necessary 5. Re-evaluate ITS SSCs (intelligently selected sample: tiered approach), if deemed necessary after step 4 6. Upgrade ITS SSCs, if appropriate	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed
Action Level 6b EL 3 EL 3b No damage to ITS SSCs Damage to NITS NRPG SSCs	1. Initial focused inspections and tests successful 2. Comparative analyses: earthquake induced response versus SL-2 3. Re-evaluate seismic hazard before or after restart, as deemed necessary 4. Re-evaluate ITS SSCs (intelligently selected sample: tiered approach), if deemed necessary after step 3 5. Upgrade, if appropriate 6. Restart 7. Repair, replace and upgrade NITS NRPG SSCs	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed

TABLE 6. ACTION LEVELS AND ACTIONS TO BE TAKEN (cont.)

Action level	Actions to be taken	Notes
Action Level 6c EL 3 EL 3c (low frequency) No damage to ITS SSCs Damage to NITS NRPG SSCs	1. Initial focused inspections and tests successful 2. Comparative analyses — earthquake induced response versus SL-2 — evaluate potential consequences of low frequency ground motion exceedance on SSCs; if evaluation shows no expected consequences, proceed with steps 3–7; if evaluation shows potential consequences to specific SSCs, proceed with steps 3–7 with a special focus on these SSCs 3. Re-evaluate seismic hazard before or after restart, as deemed necessary 4. Re-evaluate ITS SSCs (intelligently selected sample: tiered approach), if deemed necessary after step 3 5. Upgrade, if appropriate 6. Restart 7. Repair, replace and upgrade NITS NRPG SSCs	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed
Action Level 7a EL 3 EL 3a (high frequency) No damage to ITS SSCs Damage to NITS RPG SSCs	1. Initial focused inspections and tests successful 2. Repair or replace NITS RPG SSCs 3. Restart 4. Re-evaluate seismic hazard after restart, if deemed necessary 5. Re-evaluate ITS SSCs (intelligently selected sample: tiered approach), if deemed necessary after step 4 6. Upgrade ITS SSCs, if appropriate	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed

TABLE 6. ACTION LEVELS AND ACTIONS TO BE TAKEN (cont.)

Action level	Actions to be taken	Notes
Action Level 7b EL 3 EL 3b No damage to ITS SSCs Damage to NITS RPG SSCs	1. Initial focused inspections and tests successful 2. Comparative analyses: earthquake induced response versus SL-2 3. Re-evaluate seismic hazard before or after restart, as deemed necessary 4. Re-evaluate ITS SSCs (intelligently selected sample: tiered approach), if deemed necessary after step 3 5. Repair, replace and upgrade NITS RPG SSCs 6. Upgrade ITS SSCs, if appropriate 7. Restart	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed
Action Level 7c EL 3 EL 3c (low frequency) No damage to ITS SSCs Damage to NITS RPG SSCs	1. Initial focused inspections and tests successful 2. Comparative analyses — earthquake induced response versus SL-2 — evaluate potential consequences of low frequency ground motion exceedance on SSCs; if evaluation shows no expected consequences, proceed with steps 3–7: if evaluation shows potential consequences to specific SSCs, proceed with steps 3–7 with a special focus on these SSCs 3. Re-evaluate seismic hazard before or after restart, as deemed necessary 4. Re-evaluate ITS SSCs (intelligently selected sample: tiered approach), if deemed necessary after step 3 5. Repair, replace and upgrade NITS RPG SSCs 6. Upgrade ITS SSCs, if appropriate 7. Restart	If anomalies identified during inspections, action level redefined: expanded inspections and tests to be performed

TABLE 6. ACTION LEVELS AND ACTIONS TO BE TAKEN (cont.)

Action level	Actions to be taken	Notes
Action Level 8 EL 3 Damage to ITS SSCs Possible damage to NITS SSCs	1. Expanded inspections and tests 2. Evaluate root cause of ITS SSC damage 3. Corrective action based on root cause analysis results: redefine input; repair, upgrade, replace and requalify ITS SSCs; verify SSC capacity 4. Re-evaluate seismic hazard before restart, as deemed necessary 5. Define seismic hazard for SMA or SPSA 6. Evaluate capacity of SSCs and plant to new seismic hazard 7. Repair or replace damaged NITS RPG SSCs 8. Restart 9. Repair or replace damaged NITS NRPG SSCs	Expanded inspections and tests include initial focused inspections and tests

Notes:

ITS: important to safety.

NITS: not important to safety.

RPG: required for power generation.

NRPG: not required for power generation.

The need for the evaluation of the seismic hazard is dependent on the compatibility of the event with the seismotectonic model used for determining the original seismic design basis motion and the specific characteristics of the recorded earthquake motion (e.g., the frequency content). It may not be directly linked to the action levels defined above.

Review level earthquake (RLE): ground motion bases for SMA and SPSA, as defined in Ref. [5].

3.6. PLANT SHUTDOWN CRITERIA

Plant shutdown criteria address the three possible states of the plant after the earthquake occurs. These three possible states are:

(1) Seismic scram or other conditions have brought the plant to a hot or cold shutdown condition. The plant is shut down and no additional shutdown criteria are activated.

(2) Plant trip has occurred but reactor trip has not occurred. In this case, the removal of heat from the reactor coolant system must be accomplished through other systems. The ability to continue operating the reactor after a turbine trip or another trip, for example, another rotating machinery trip, depends on the capacity of the alternative systems. For current large reactor designs, the capacity of alternative systems is generally 20–40% of full power. For these designs, turbine trip causes reactor trip unless the plant is operating at relatively low power. Smaller plants and some newer designs do have the capacity to dump 100% of reactor power through turbine bypass for long enough to reduce reactor power while avoiding reactor trip. Operator and engineering actions are required to determine the status and need for shutdown.

(3) Plant continues to operate. Whether the felt earthquake is significant has to be determined.

For plant states (2) and (3) above, it is important to establish in advance exceedance criteria based on earthquake records and other relevant information. Elements of the exceedance criteria for determination of plant shutdown for plant states (2) and (3) are:

(a) *Operator actions:*

(i) Responding to control room indicators that anomalies in plant behaviour have occurred, investigate whether the anomalies were due to damage or protective considerations — if reset can be accomplished, reset for potential restart;

(ii) Walkdowns to observe the overall condition of the plant and, specifically, the condition of the preselected sample of SSCs.

(b) *Engineer actions:*

In addition to the operator actions specified in the operating procedures with regard to recorded data enunciated in the control room, the following engineering actions are required:

(i) Processing of recorded data to compare the design basis ground motion with the recorded motion, typically, generating response spectra and derived quantities such as the CAV;

(ii) Definition of the earthquake level.

Initially, the operators define a preliminary estimate of the earthquake level based on the parameters enunciated in the control room, i.e. the peak ground acceleration and possibly others.

At headquarters or on the site, the immediate actions of the engineering staff are to define the earthquake level and related damage indicating parameters, for example, CAV and JMA intensity.

By itself the earthquake level does not determine the actions to be taken. Exceedance of design basis ground response spectra alone is not an adequate measure of the damage potential of an earthquake, especially for an earthquake characterized by relatively high frequency motions, for example, greater than about 10 Hz. Other parameters that are better damage indicating parameters are also part of the decision making process. One example is the CAV, which has been validated as an appropriate indicator of damage to structures and mechanical systems and components. The CAV parameter may not be an appropriate indicator of functional failure of some electrical equipment or instrumentation and control systems. Threshold values of CAV have been recommended for decision making in terms of the potential of a ground motion recorded at the site to cause damage. The CAV has become part of the shutdown/restart criteria for plants in the USA. Other examples to be considered include JMA intensity and spectral intensity.

To illustrate the importance of including parameters more indicative of damage potential than peak ground acceleration, ground response spectra or acceleration–time histories, several examples are cited (in each case the DBE response spectra were exceeded, but no or minimal damage occurred to the nuclear power plant: the plants V.C. Summer in 1978 and 1979, Perry in 1986, Onagawa in 2003 and 2005, and Shika in 2006). Some of these cases are presented in Annex II and summarized in Section 2.3. For these cases, the exceedance of the DBE occurred in the high frequency range of SL-2 earthquakes (for the V.C. Summer, Perry and Onagawa nuclear power plants) or in the low frequency range of the DBE or SL-2 earthquakes (e.g. for the Shika nuclear power plant), or in all frequency ranges (e.g. for the Kashiwazaki-Kariwa nuclear power plant). Detailed inspections showed that the earthquake caused no significant damage in the nuclear power plants. Therefore, it is recommended to supplement design basis ground motion exceedance measures with other damage indicating parameters.

It is important to establish in advance exceedance criteria based on earthquake experience and other relevant information.

It is recommended that calibration standards, computer software and record analysers be prepared in advance to enable the engineering assessments identified above to be performed within a time frame approved by the regulatory body, for example, within eight hours of the occurrence of an earthquake.

It is also recommended that each Member State establish these threshold values of the parameters to be used in decision making for shutdown and restart in advance, taking into account the seismic hazard at the site and the seismic design of the plant, including the original design and significant upgrades implemented during the life of the plant. The combination of damage level and earthquake level, including damage indicating parameters, leads to the initial estimate of action level and, consequently, to the next actions to be taken.

3.7. PRESELECTION OF SSCs FOR INSPECTION

3.7.1. Considerations for selecting SSCs for inspection

In the event of an earthquake greater than the SL-1 level, the initial focus of post-earthquake inspections is on a preselected set of SSCs, chosen to encompass the range of SSCs of interest based on characteristics such as the number of like components, location, vulnerability to damage due to earthquake motion, accessibility after the earthquake and other considerations.

The preselected SSCs are chosen to be representative of SSCs important to safety and include SSCs that experience has shown to most likely be damaged during an earthquake. The SSCs selected also include typical items not important to safety, which experience has shown to be of low seismic capacity: these items may be damage indicators that will assist experienced seismic engineers in evaluating the state of the plant.

In addition, the risk importance of the SSCs should be considered. One way of doing so is to categorize items important to safety and items not important to safety according to their risk importance based on risk informed methodologies. The US Nuclear Regulatory Commission provides an example of guidance in this regard in Regulatory Guide 1.201 [23].

The set of preselected SSCs are baseline inspected as part of the pre-earthquake actions and their 'as is' properties documented. The seismic design/qualification information for the SSCs is assembled so as to be conveniently accessed if an earthquake occurs.

To select the representative SSCs, it is helpful to categorize SSCs into like categories. One such categorization for equipment and components is based on their function:

(a) Active systems and components, which are typified by pumps (that are required to start or stop, operate during or after the shaking, etc.), valves (that have to change position), fans, generators, and electrical equipment such as breakers, switchgears and control and instrumentation devices;

(b) Passive SSCs, which are typified by steel and reinforced concrete structures, tanks, vessels and piping, etc.;
(c) Electrical equipment, typified by transformers, conduit and cable trays;
(d) Other SSCs.

The objective is to select a sample with characteristics that encompass the diverse set of SSCs in the plant and that bound or adequately represent the broader category of SSCs, such that conclusions about the performance of the sample are applicable to the category.

Reactor systems and components are excluded from the preselected SSCs for in-plant inspection. However, as a result of ongoing scheduled testing, baseline data are available when needed for assessing the consequences of the earthquake on the reactor and reactor internals. The evaluation of reactor internals may be made through a combination of testing (surveillance tests and others) and analyses. The need for additional evaluations as a function of the earthquake event is discussed in Section 5.

3.7.2. Baseline inspections

Baseline visual inspections of all SSCs selected for post-earthquake inspections are performed and the results of the inspections documented in written reports, including sketches and photographs of existing abnormalities as appropriate.

In addition, the results of NDEs or condition monitoring of SSCs are made available to the operators and the engineering staff. Any significant cracks in reinforced concrete structures are included and documented in the baseline inspections so that their condition after an earthquake can be properly evaluated.

The purpose of the baseline inspections is to identify and document any pre-existing conditions (e.g., cracks in concrete structures or pipe insulation damage) in order to provide (during the post-shutdown inspections) a basis for differentiating earthquake related damage from pre-existing abnormal conditions.

It is recommended that periodic inspections of the items selected for post-earthquake shutdown inspections also be performed to identify and document any changes in the condition of the preselected items.

3.8. DYNAMIC MODELS FOR RESPONSE CALCULATIONS

An important aspect of the post-earthquake evaluation of the nuclear power plant is the ability to easily calculate the in-structure response of selected SSCs important to safety when they are subjected to the actual earthquake ground motion. These seismic responses can then be compared with those calculated during the seismic analysis and design phases of the project. Such comparisons may be needed to aid in decision making concerning nuclear power plant shutdown and restart.

To perform these analyses in a timely fashion, the dynamic models of the SSCs of interest should reflect ‘as is’ conditions of the nuclear power plant, should be readily available and should be executable in computer software currently available to the responsible engineering staff. The evaluation and restart of the Shika nuclear power plant, after experiencing the Noto Hanto earthquake of 25 March 2007, were significantly enhanced by the availability of dynamic models of structures and computer software on which to analyse them. This permitted analyses of the Shika nuclear power plant structures to be performed quickly, calculating in-structure responses for comparison with the design analyses results.

3.9. PROCEDURES AND TRAINING

Operating procedures are typically symptom driven. It is possible to achieve a prompt response by setting and arranging a procedure for confirming the major parameters recommended in Ref. [24] and other materials.

4. SHORT TERM ACTIONS

4.1. FELT EARTHQUAKES AND SIGNIFICANT EARTHQUAKES

All actions are triggered by the occurrence of an earthquake that is ‘felt’ at the nuclear power plant site.

A *felt earthquake* is any earthquake that produces vibratory ground motion at the site that is perceived by nuclear power plant operators as an earthquake and confirmed by seismic instrumentation or other related information. Typically, seismic instrumentation installed at nuclear power plants is triggered at peak ground acceleration values of 0.01 g to 0.02 g.

The first action to be taken after the earthquake is felt at the site is to consider whether the felt earthquake is significant. If the earthquake ground motion at the site is ‘low’ enough, only minimal actions may be required, for example, notification that a felt earthquake has occurred, ground motion at the site was very low, no adverse consequences to the plant are apparent, and the readings of all plant instrumentation are in the normal range, and no further actions need be taken.

Several factors contribute to the designation of *significant earthquake*: a felt earthquake having a free field surface peak ground acceleration at the threshold of damage or malfunction of non-seismically designed power plant SSCs. Typically, it has a free field surface ground motion of greater than 0.05 g or a standardized CAV greater than 0.16 g·s.

The designation of a *significant earthquake* takes into account the size as described above; however, it is also a function of the site and the seismic design criteria of the nuclear power plant, since it may determine the actions to be taken by the licensee and the regulatory body. The importance of identifying a significant earthquake as a subset of felt earthquakes is that the short term actions described in this section may not need to be performed if the felt earthquake is not a significant one.

Immediately after the earthquake has been felt and classified as a significant earthquake, short term actions should be taken (Section 2.1.6).

However, short term actions by the operating organization may also be required by the regulatory body in some Member States, when seismic instrumentation is not installed or not available, and the earthquake occurs in the site region.

The earthquake may not have been felt at the site, but short term actions may still be required to verify that no adverse effects have occurred within the plant. One example is the requirements for reporting to the regulatory body the plant status when an earthquake of magnitude 5 or higher occurs within 200 km of the site [6].

A flow chart of the short term actions to be taken after the occurrence of a significant earthquake is shown in Fig. 3. Each of the actions shown in Fig. 3 is discussed in detail below.

4.2. IMMEDIATE ACTIONS

4.2.1. Conditions after the occurrence of an earthquake

As described in Section 3.6, three possible conditions may result after the felt earthquake has occurred, as follows:

(1) Seismic scram or other conditions have brought the plant to a condition of hot or cold shutdown.

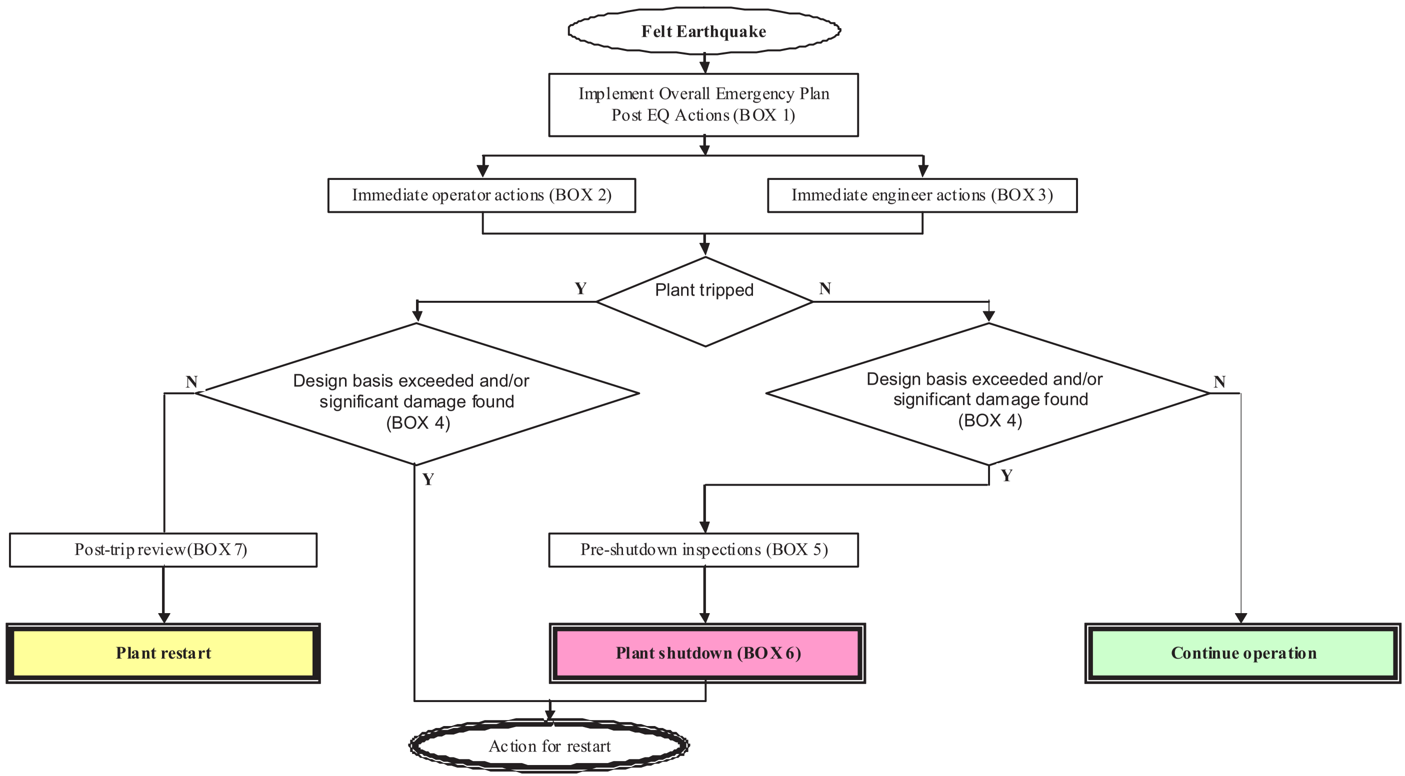


FIG. 3. Flow diagram of short term actions to be taken after the occurrence of a felt earthquake (Y: Yes; N: No).

(2) Plant trip has occurred but reactor trip has not occurred. In this case, the removal of heat from the reactor coolant system must be accomplished through other safety systems.
(3) The plant continues to operate.

In general, performance of these immediate actions is required within a time frame established in conjunction with the regulatory body, for example, within 24 hours after the felt earthquake has occurred.

4.2.2. Implementation of the overall emergency plan: Post-earthquake actions

The post-earthquake actions specified in the overall emergency plan (Box 1 of Fig. 3) are tiered based on the level of the earthquake motion at the site, its effects on the plant and the specific requirements established by regulatory and other national authorities. For very small ground motions, i.e. a felt earthquake of less than 0.05 g peak ground acceleration, the primary actions are usually those performed by the operational staff on-site. For larger earthquakes, more extensive actions are required.

4.2.3. Immediate operator actions

4.2.3.1. Immediate actions

The immediate actions to be conducted by plant operators (Box 2 of Fig. 3) include:

(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.

In the following sections, more details are provided for immediate actions conducted by plant operators.

4.2.3.2. *Confirmation of plant parameters and response to plant alarms*

These actions are taken by operators in accordance with approved procedures in response to operational symptoms identified in the control room. Their purpose is to maintain the plant in a safe stable condition while further assessments are made.

It can be expected that a felt earthquake with sufficient intensity to cause operating system upset, malfunction and/or damage will result in alarms and/or changes in plant parameters which will require action by control room operators according to the plant operational procedures.

Control room instrumentation and alarms provide additional information on the status and performance of components and systems. Together, they provide plant operators with the information needed to determine if the plant can continue to operate or needs to be shut down for additional inspections and evaluations. Reference [24] provides further guidance in this connection.

4.2.3.3. *Walkdown inspections by the operator*

All accessible areas of the nuclear power plant should be walked down and visually inspected by plant operators and available on-site personnel who are familiar with the pre-earthquake physical condition of plant SSCs and the areas being inspected. High radiation areas, the reactor building and other areas with limited access need not be included in these initial walkdown inspections unless plant personnel have reason to suspect that there may be damage in these areas. The purpose of the operator walkdown inspections is to rapidly determine the effects of the earthquake on the physical condition of nuclear plant SSCs. If the plant is not shut down and significant damage is found during the walkdown for SSCs important to safety and those needed for stable operation, operators should immediately initiate pre-shutdown inspections (Section 4.4).

It is considered important that the operator walkdown inspections be performed by plant operators who are familiar with the SSCs (e.g., regarding physical appearance, leak rates, vibration levels and sound of motors), to determine changes from their condition before the earthquake. Plant operators should be assisted in these inspections by structural and mechanical engineers trained in seismic walkdown practices, to the extent possible according to the time available. They may be assisted by available on-site personnel (e.g., engineering, maintenance and quality control personnel).

The inspections are similar to those performed by plant operators during their normal daily rounds, with additional emphasis on visual and audio inspections for evidence of earthquake related damage. In general, the visual and audio inspections fellow specific guidance for earthquake related damage indicators in the plant specific procedures (Table 7).

TABLE 7. SPECIFIC GUIDANCE FOR OPERATOR WALKDOWN INSPECTIONS

(1)	Check for leaks in piping systems, especially at flange or threaded connections and branch lines
(2)	Check for damage to low pressure tanks, particularly ground or floor mounted vertical flat bottom storage tanks
(3)	Check for damage to switchyard equipment
(4)	Check fluid levels in tanks. Level switches may have been activated due to sloshing of the contained fluid (an actual but momentary change in level)
(5)	Check for high vibration, high bearing temperature and unusual noise in rotating equipment and falling objects
(6)	Check for damage to SSCs due to impact between adjacent SSCs or due to anchorage concerns, including deformation or loosening of anchor bolts, pullout or shear of anchor bolts, and rocking, sliding or misalignment of equipment
(7)	Check for damage to attached piping, including hoses, tubing and electrical raceways
(8)	Check for damaged piping, and check pipe and component supports for evidence of excessive displacement
(9)	Check for distortion of electrical and control cabinets, including a brief visual check of a sample of internally mounted components such as relays and circuit breakers
(10)	Check for major cracks, spalling or scabbing in reinforced concrete structures. Hairline cracks in reinforced concrete structures are not considered significant
(11)	Check the operational status of important relays, breakers and other potentially sensitive electrical equipment, in particular, those in protective and seal-in/lockout circuits whose change in state could affect operability of ITS equipment and systems
(12)	Check for portable equipment which may have fallen on ITS equipment
(13)	Check containment penetrations for indications of distress
(14)	Check for settlement and relative motion of buildings and structures
(15)	Check for indications of pounding/impact between structures
(16)	Check for damage indicators for buried piping and other distribution systems (direct indicators, such as pipe breaks, and indirect indicators, such as soil failure)

The preselected SSCs are drawn from this list of examples of SSCs needed for continued operation or for maintaining safe shutdown:

— The reactor building;
— The control building;
— The turbine building;
— The primary reactor coolant system (reactor vessel, pumps, piping, etc.);
— The emergency core cooling system (ECCS) (pump, piping, etc.);
— The reactor control system;
— The emergency power supply (diesel generator, DC battery, etc.);
— External power sources;
— The main power generating turbine;
— The main power generator transformer;
— Switchgear important to safety;
— Motor control centres important to safety.

It is anticipated that the operator walkdown inspections discussed in this section of the report are performed within a time frame established in conjunction with the regulatory body, for example, within eight hours, depending on the number of personnel conducting the inspections. Results of the operator walkdown inspections following an earthquake are documented in written reports, including observation notes, photographs and checklists.

Operator walkdown inspections are performed as discussed above, even if the plant automatically shuts down as a result of the earthquake. The operators determine the initial damage level, which when combined with the earthquake level (Section 4.3.3) determines the initial estimate of the action level. The action level designation determines if additional inspections and tests are warranted prior to restart of the plant.

4.2.3.4. Emergency response

Operators should implement established emergency response procedures to mitigate situations that could arise due to the earthquake. These situations include radioactive material release, disruptive earthquake damage, and fires and floods at the site. In this regard, Ref. [25] is a manual that provides some guidance for first responders within the general scope of IAEA Safety Guide No. GS-G-2.1 [26].

4.2.4. Immediate engineer actions

The immediate engineer actions (Box 3 of Fig. 3) are those to be taken in the short term, for example, within eight hours, or as required by the specific regulations required by the regulatory body — including the identification of the earthquake level, which is determined by comparing seismic instrumentation records with the DBEs (SL-1 and SL-2).

If seismic instrumentation is not installed at the site or is not available, other data or information are used to determine the earthquake level. Other information includes records from seismic instrumentation in the area, observations of malfunction and damage or lack thereof in the area, knowledge of the magnitude of the earthquake and the distance from the site. Specific sources of information may be required by the regulatory authority.

4.2.4.1. Processing of recorded motions

In general, seismic instrumentation includes triaxial time history accelerographs located in the free field, on structure foundations and/or in structures of interest. Other seismic instrumentation of value could be instruments that record velocities and displacements and/or produce parameters that are directly correlated to damage, such as CAV and JMA intensity. The records from these instruments are processed to provide additional information to that available to the operators for decision making.

Design basis earthquake motions (SL-1 and SL-2) are generally defined by ground response spectra in the free field. Typically, instruments record acceleration–time histories from which response spectra are calculated. These response spectra of recorded motions are compared with the design basis response spectra to determine the earthquake level at the site, i.e. Earthquake Level 1, 2 or 3:

(a) For the comparison of free field motions, when feasible, the seismic instruments are located at the same location (in the free field) where the design basis ground motion was defined. This allows a direct comparison between the design basis ground motion and the recorded motion.
(b) If it is not feasible to locate the seismic instrumentation at the same location in the free field where the design basis ground motion was defined, for example, if the design basis ground motion was defined at depth in the soil and on a hypothetical outcrop, then the following options are available:
    (i) To calculate the design basis ground motion at the instrument location and compare the recorded motion with the calculated motion or their derived quantities (e.g., response spectra);

(ii) To compare recorded motions or derived quantities and design basis values at other locations, such as on the foundations of structures important to safety; or
(iii) To compare recorded motions or derived quantities at other locations in structures important to safety.

Other parameters, such as CAV or JMA intensity, may be calculated from the recorded acceleration–time histories or may be obtained directly from instruments recording these parameters. To define Earthquake Levels 1, 2 or 3, any of the above listed options is applicable. To further refine the definition of Earthquake Level 3 into 3a, 3b and 3c, the free field data are the most applicable.

As stated in Section 3.2 and above, seismic instrumentation on the foundations of structures and at locations in structures provides data for comparison with seismic design parameters, such as ISRS. These latter records also provide data for further evaluation of SSCs as described in later sections, for example, in Section 6.

Systems and components of interest are located throughout the buildings and site. It may be helpful to develop models or transfer functions in advance, relating motion at recording locations to locations for input to the systems and components of interest. This would allow relatively quick evaluations to be done for major systems and components.

4.2.4.2. Assessment of the earthquake level

One measure of exceedance or non-exceedance of the SL-1 or SL-2 earthquake levels is the design basis ground motion comparison described in Section 3.3.2. This comparison defines three earthquake levels. One result of the immediate engineer actions is to determine the earthquake level (either Earthquake Level 1, 2 or 3) corresponding to the event.

4.2.5. Assessment of the results of immediate actions

The immediate actions of the operators define the damage level on the basis of the results of the walkdowns and other information from the field and the control room indicators.

The immediate actions of the engineering staff define the earthquake level and related damage indicating parameters, for example, CAV and JMA intensity.

Exceedance of design basis ground response spectra alone is not an adequate measure of the damage potential of an earthquake, especially an earthquake characterized by relatively high frequency motion, i.e. greater than about 10 Hz. Other parameters that are better damage indicating parameters should be part of the decision making process.

With all the information collected from the performed tasks and the determination of the earthquake level and the damage level, as defined in Boxes 1, 2 and 3 of Fig. 3, the process now proceeds to the following tasks according to the real situation in relation to:

(a) Plant status: Was the plant tripped?
(b) Exceedance criteria: Was the design basis exceeded?
(c) Damage level: Was significant damage found?

The answers to these three questions will allow different alternatives as indicated in Box 4 of Fig. 3.

4.3. PRE-SHUTDOWN INSPECTIONS

4.3.1. General conditions

In the case that the plant was not tripped but it was verified by the immediate operator and engineer actions that significant damage was found and/or design basis criteria were exceeded, the plant shutdown should proceed considering the following recommendations. Prior to normal shutdown, pre-shutdown inspections (Box 5 of Fig. 3) are needed to verify that all required systems are available and operable. The purpose of these inspections is to determine the effect of the earthquake on essential safe shutdown equipment which is not normally in use during power operation, and to determine the appropriateness of performing shutdown procedures.

The pre-shutdown inspections focus on functional damage to SSCs that may impair the capability of the damaged SSC to perform its important to safety function. Some physical damage which does not affect SSC functionality or operability is not a major concern in these inspections. The SSCs required for safe shutdown that are identified as inoperable due to the earthquake or which were out of service (‘tagged out’) prior to the earthquake are repaired or an alternative SSC may be placed in service prior to plant shutdown.

Some items of equipment may require resetting at the time of the inspections due to the earthquake (e.g., relays and other switches may have tripped due to chatter, or an isolation valve may have opened or closed). In these situations, the appropriate plant procedures for resetting the equipment item are used.

The operator may decide on a range of inspections according to the size of the earthquake ground motion at the site and its relationship to the SL-1 and SL-2 design basis ground motion. It is anticipated that the pre-shutdown inspections will be performed within eight hours of the earthquake.

4.3.2. Confirmation of safe shutdown and fundamental safety functions

To assure safe shutdown, plant operators have standard operating procedures for pre-shutdown inspections of necessary SSCs. These operating procedures are reviewed and verified to include earthquake related considerations. It is assumed that the operating procedures identify and maintain a list of essential safe shutdown equipment as well as inspection procedures and documentation guidelines, including checklists and forms, to be used in the pre-shutdown inspections.

The essential systems include those required to perform the following fundamental safety functions:

(a) Reactivity control;
(b) Removal of heat from the core (reactor coolant pressure control, reactor coolant inventory control and decay heat removal);
(c) Confinement of radioactive material and control of operational discharges.

It should be confirmed that the above mentioned functions have been maintained on the basis of the result of immediate operator actions. In addition, it is preferable to perform functional tests of equipment (e.g. shutdown cooling pumps) that are necessary for a safe shutdown.

Inspections include all trains of redundant safe shutdown equipment. Components and systems required only for accident mitigation do not need to be inspected as part of the pre-shutdown inspections.

4.3.3. Availability of power sources

Off-site power may be disrupted following an earthquake due to potential damage to fragile ceramic insulating materials and unanchored equipment typically used in non-seismically-qualified high voltage distribution systems, and the potential for relays to chatter or change state. Therefore, the availability of plant power sources is evaluated.

During shutdown and the removal of the turbine generator from the grid, the transfer from in-house power to off-site power utilizes several circuit breakers and transformers. These circuit breakers and transformers and the associated distribution systems are checked. Specifically, this includes:

(a) Determining the availability of off-site power sources. Contacting the dispatcher and determining the status of the grid, switchyards and substations. Determining the number of available off-site power sources. If fewer than two sources of off-site power are available, or if the condition of the off-site power sources is uncertain, board checks of the on-site emergency power systems are recommended.
(b) Visually inspecting the startup/auxiliary transformers and circuit breakers and the associated electrical distribution equipment. Specifically, checking that transformer sudden pressure switches have not been actuated, resulting in isolation of the startup transformers.

4.3.4. On-site emergency power sources

If the availability of off-site power sources is uncertain or is determined to be marginal (i.e. degraded) following the earthquake, the availability of on-site emergency or alternative power should be determined. Specifically, this includes:

(a) Performing a visual and audio inspection of the emergency diesel generators, and inspecting the starting system, cooling system, fuel oil system, lubricating oil system, intake and exhaust structures, and electrical distribution system.
(b) Performing a visual and audio inspection of the plant's DC power system. The inspection includes a visual inspection to determine if the batteries appear undamaged, including the rack-battery system, cables interconnecting the batteries, and cables in and out of the system. Checks of the batteries are made to ensure that the battery parameters, such as electrolyte level and voltage, indicate availability.
(c) Depending on the severity of the earthquake and the condition of the grid, performing any other plant specific inspections or tests considered necessary to assure that on-site emergency power will be available in the event of loss of off-site power.

4.3.5. Decision on shutdown

After satisfying the pre-shutdown conditions as stated above, normal shutdown of the nuclear power plant should proceed (Box 6 of Fig. 3).

4.4. POST-TRIP REVIEW

The causes of the plant trip are determined during the immediate actions of the operations and engineering staff, to assure that all essential systems — safety and non-safety — are available and operable (Box 7 of Fig. 3). If the operator and engineering inspections conclude that plant restart is appropriate, and there are no regulatory conditions to be met, startup is performed in accordance with normal restart procedures.

5. ACTIONS FOR RESTART

5.1. INTRODUCTION AND MAIN ASSUMPTIONS

Section 4 of this report provides guidelines for determining if the SL-1 level has been exceeded, and for performing visual inspections and the tests of essential safe shutdown equipment prior to shutdown (if shutdown is considered necessary).

The present section addresses the restart strategy (Fig. 4). The restart strategy to be employed is dependent on the earthquake level, the initial damage level and malfunction assessment of the plant, and regulatory requirements. The combination of the first two parameters, i.e. the earthquake level and the damage level, defines the initial action level.

Actions are considered in a tiered approach based on the results of examinations, inspections, tests and evaluations, as defined for each of the action levels. In addition, the scope and timing of the actions are dependent on the action level and the results of the examinations, inspections, tests and evaluations.

Guidelines for performing additional long term evaluations of SSCs important to safety are provided in Section 6. In addition to the recommended post-shutdown seismic safety evaluations, if a plant has recorded earthquake data from in-plant instruments, these data are evaluated and reconciled with the results of the physical evaluations as described herein.

This report proposes that the readiness of a nuclear power plant to resume operation (following shutdown due to an earthquake) be based primarily on the results of the visual, audio and physical evaluations and tests described in this section. The long term evaluations described in Section 6 would normally be performed after the plant has returned to power. Sections 5 and 6 identify a series of actions to be taken prior to or after plant restart on the basis of the

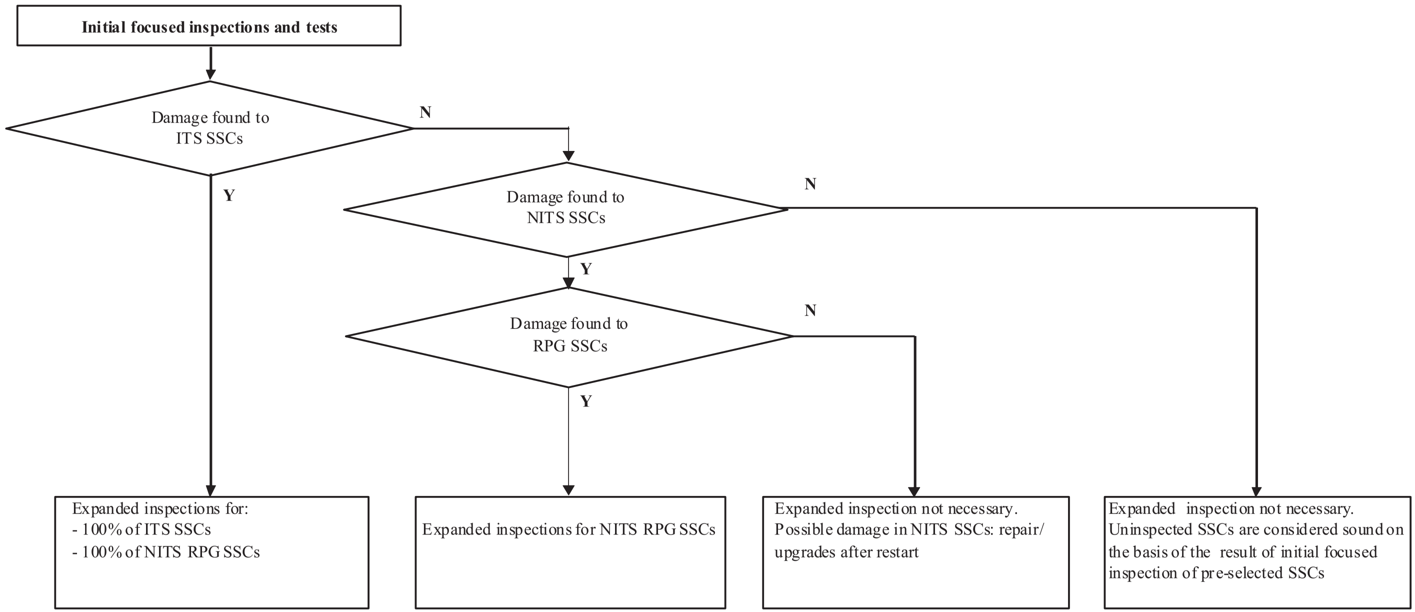


FIG. 4. Expanded inspections (Y: yes; N, no).

characteristics of the earthquake (and its relationship to the SL-1 and SL-2 level DBEs), the seismic performance of nuclear power plant SSCs and the following assumptions:

(a) The plant is shut down (hot or cold shutdown) due to seismic scram, plant trip including reactor trip or normal shutdown by operators after pre-shutdown inspections.
(b) Immediate actions of the operations and engineering staff have been performed as follows:
   (i) Operators have performed walkdowns of all accessible areas and noted signs of damage and have decided the initial damage level due to the earthquake.
   (ii) Engineering staff have addressed or processed the records of the earthquake, determined (in coordination with the operators) whether the criteria for shutdown have been exceeded, and have defined the earthquake level.

5.2. STRATEGY FOR RESTART

The strategy for restart is dependent on the regulatory requirements of the individual Member State and on the earthquake level, the damage level and, consequently, the action level. Initial assessments of these three parameters initiate the restart evaluation process. Information developed during the process can change one or more of these parameters, which may lead to a different strategy. Changes in the damage level are the most likely to occur. Changes in the earthquake level are less likely, since the earthquake level parameter was established through a comparison of recorded motions and the DBEs. The exception is for further categorization of Earthquake Levels 3a, 3b and 3c, which may occur after additional evaluations have been performed.

In general, the short and long term actions taken and the results of these actions are evaluated by the regulatory body for concurrence of subsequent actions.

The progression of inspections, tests and evaluations is as follows:

(a) Initial focused inspections and tests.
(b) Expanded inspections and tests (of SSCs important to safety and those not important to safety).
(c) Comparative analyses of the response of soil, rock, foundations, structure and subsystems:

(i) The response due to actual earthquake motions;
(ii) Comparison of the in-structure responses with the design in-structure responses, to establish exceedances or non-exceedances, actions to be taken, etc.

(d) *Non-destructive examinations*. Non-destructive examinations consist of evaluating an SSC or its material without adversely affecting the serviceability of the SSC. Some examples are: visual examination/inspection, magnetic particle examination/inspection, liquid dye penetrant examination/inspection, X ray examination/inspection, ultrasonic testing, air or water pressure testing, percussion tests, Vickers hardness test and vibration tests.
(e) *Surveillance tests*. Tests performed at regular intervals to demonstrate the availability and operability of components and systems. Surveillance tests are identified in the technical specifications of the plant and consist of checks, tests, calibrations, examinations and inspections to verify availability and performance of the tested component or system.

In general, long term actions (Section 6) are performed after restart and include:

(f) Evaluation of seismic hazard and definition of seismic ground motion for evaluation purposes.
(g) Evaluation analyses of the response of soil, rock, foundation, structure and subsystems:
    (i) The response due to the re-evaluated seismic hazard — definition of ground motion denoted RLE[9];
    (ii) Comparison of the in-structure responses with the design in-structure responses to establish exceedances or non-exceedances;
    (iii) Determination of actions to be taken, upgrades, analytical evaluations, etc.
(h) Upgrades.

[9] RLE: review level earthquake. As defined in Ref. [5], the RLE is the ground motion basis for evaluation of the plant as a result of a seismic hazard assessment evaluated according to Ref. [3] and for conducting the SMA or SPSA methodologies for assessing the seismic safety of the facility.

As discussed in the following sections, not all of these activities are required, depending on the findings in the progression of activities as defined by the action levels. In addition, the timing of the activities is a function of the action level; some activities are required to be performed prior to restart, others may be performed after restart.

Action levels as a function of the earthquake level and damage level are defined and presented in Table 2. Table 6 lists the order of actions to be taken as a function of the action level. These actions are then described in more detail. Details of these actions, including scope and timing, are established, and these may need the approval of the regulatory body. The overall strategy as a function of the action level is:

(a) *Action Levels 1, 2, 3 and 5.* The *initial focused inspections and tests* are performed. Upon successful completion of these examinations, inspections and tests (including surveillance tests), the plant would be considered ready for restart. Startup would be in accordance with normal startup procedures. For Action Level 2, the NRPG SSCs not important to safety may be repaired after restart. For Action Level 3, the RPG SSCs not important to safety are repaired or replaced prior to restart. If anomalies are identified during these initial focused inspections and tests, the action level is redefined and the process extends to the expanded inspections and tests.

(b) *Action Level 4.* The step of *expanded inspections and tests* is performed. The initial focused inspections and tests are assumed to be encompassed by the expanded inspections and tests. In addition, root cause analyses of the malfunctions, damage or failures of SSCs important to safety are required. Repairs, upgrading and/or redesign of SSCs important to safety may be needed depending on the results of the root cause analyses. Not important to safety RPG SSCs are repaired and possibly upgraded or redesigned before restart. Once these actions, the other activities required under the expanded inspections and tests as well as the surveillance tests have been successfully completed, the plant would be considered ready for restart. Startup would be in accordance with normal restart procedures.

(c) *Action Levels 6 and 7.* Action Levels 6 and 7 correspond to no damage to RPG SSCs important to safety and no damage to RPG SSCs not important to safety (Action Level 6) or damage to RPG SSCs not important to safety (Action Level 7), as determined from the initial focused inspections and tests. Action Levels 6 and 7 correspond to Earthquake Level 3, i.e. ground motions greater than the SL-2 design basis motions. However, the steps to be taken before restart are highly dependent on the nature of the exceedances of the recorded motions over the SL-2 design basis motions. Taking into account the frequency characteristics of the recorded ground motion with respect to the soil–structure system frequencies[10], as indicated in Section 3.3.2, Earthquake Level 3 is further categorized into Earthquake Levels 3a, 3b and 3c, i.e. the high frequency range, mid-amplified frequency range and low frequency range, respectively, in order to define further actions:

(i) For Earthquake Level 3a (high frequency exceedances), no damage to SSCs important to safety, as verified by the initial focused inspections and tests, the plant would be considered ready for restart after equipment not important to safety but important to the plant have been repaired (Action Level 7a).

(ii) For Earthquake Level 3b (exceedances exist in the mid-frequency range, which includes important soil–structure frequencies), the step of expanded inspections and tests should be performed. During the period of execution of the expanded inspections and tests, additional activities are required. Analyses of representative soil, rock, foundation and structure subsystems are required and comparisons need to be made between the seismic responses calculated for the observed earthquake and those of the design bases. Evaluation of the results determines the next actions to be taken (restart, upgrading, revisions to the design bases, etc.) (Action Levels 6b and 7b).

(iii) For Earthquake Level 3c (exceedances in the low frequency range, less than the important frequencies of the soil–structure system), analyses of representative soil, rock, foundation and structure subsystems are required and comparisons need to be made between the seismic responses calculated for the observed earthquake and those of the design bases. It is expected in this case that it will be possible to show that the recorded motions only affect low frequency behaviour, for example, displacement controlled SSCs, and do not affect major structures and the systems and components housed within. Evaluation of the results determines the next actions that are taken: restart, upgrading revisions to the design bases, etc. Upon successful completion of these analyses, examinations, inspections and tests (including surveillance tests), and after NRPG SSCs not important to safety have been repaired, the plant would be considered ready for restart (Action Levels 6c and 7c).

[10] As used herein, the term ‘soil–structure’ includes rock founded structures where soil–structure interaction may not be an important phenomenon. In that case, references to ‘soil–structure frequencies’ or ‘soil–structure interaction analyses’ would then mean ‘structure frequencies’ or ‘structure analyses’, respectively.

(iv) Startup would be in accordance with normal startup procedures. The need for the seismic design of SSCs not important to safety and of upgrades is evaluated after restart.

(v) Further evaluations are performed during the evaluation phase and after restart (Section 6):

— Evaluation of the seismic hazard may be performed after and during the evaluation phase and after restart. The result of this evaluation may lead to an RLE to be used in evaluations of the seismic design/qualification of SSCs important to safety.

— An interim definition of the RLE for preliminary evaluation purposes — until the final RLE has been determined — may be the recorded earthquake ground motions multiplied by an agreed factor, an increase above the recorded earthquake ground motions to establish margin.[11]

— A reassessment of in-structure design basis responses used for structure design and the design/qualification of other SSCs.

(d) *Action Level 8.* This case follows the approach for Action Levels 4 and 6/7 with important exceptions:

(i) Evaluation of the seismic hazard leading to the RLE to be used in evaluations of the seismic design/qualification of SSCs important to safety. (Similar to Action Levels 6 and 7, an interim definition of the RLE for preliminary evaluation purposes may be the recorded earthquake ground motions multiplied by an agreed factor, an increase above the recorded earthquake ground motions to establish a margin.)

(ii) A reassessment of in-structure design basis responses used for structure design and for the design and qualification of other SSCs.

In addition to the activities for Action Level 4, which are performed for Action Level 8, Section 6 describes in detail the approaches to address the other issues.

[11] The factor should be greater than 1. It should be defined on a case by case basis. As an example, for nuclear power plant sites in low to moderate seismicity areas, it may be determined as the ratio between the SL-2 and SL-1 original design bases, provided SL-1 and SL-2 were based on probabilistic considerations. Otherwise, for these sites, a factor of between 1.5 and 2.0 would be appropriate.

5.3. POST-SHUTDOWN SSC SAFETY EVALUATION: INSPECTION, ANALYSIS AND/OR TEST

The purpose of the post-shutdown evaluations is to assess the state of the plant (SSCs) and the actions to be taken for restart, re-analysis, upgrading or, in some cases, maintaining safe shutdown.

In general, the progression of inspections, tests and evaluations is:

(a) Initial focused examinations, inspections and tests — these physical evaluations and tests are required for all action levels.
(b) Expanded examinations, inspections and tests — dependent on action level and the results of the initial focused physical evaluations and tests.
(c) Comparative analysis: between earthquake induced response and design response — the purpose is to assess the earthquake environment to which SSCs were subjected and to evaluate the response level of SSCs as appropriate, and to provide guidance as to the type and level of examinations, inspections and testing to be performed, for:
    (i) Soil, rock, foundations and structure;
    (ii) Subsystems.
(d) Non-destructive examinations — these NDEs are tiered progressively, depending on the results of other tests, examinations and inspections.
(e) Surveillance tests.
(f) Startup tests.
(g) Restart.

The approach for the post-shutdown examinations, inspections and tests described above can be implemented following an earthquake which exceeds the SL-1 design basis, even if an operator chooses not to preselect SSCs for focused examinations and inspections, and chooses not to perform baseline inspections of the selected items. However, in this case, the time required to perform the recommended examinations and inspections (focused and expanded) following an earthquake will be much longer, and it will be more difficult to distinguish earthquake related damage from pre-existing conditions (particularly for cracks in reinforced concrete structures). This can add considerably to the total time required to determine the readiness of a nuclear plant to restart following an earthquake. Thus, it is recommended that pre-existing conditions be determined, documented and updated periodically.

Long term actions are described in Section 6:

(a) Seismic hazard evaluation — dependent on action levels;
(b) Seismic evaluation of SSCs.

5.3.1. Initial focused inspections and tests

5.3.1.1. General considerations

The initial focused examinations and inspections are carried out for the selected SSCs as discussed in Section 5.3.1.2. The next steps, based on the results of the initial focused inspections, are discussed in Section 5.3.1.3. If no significant physical or functional damage is found in the SSCs selected for the focused post-shutdown examinations and inspections, including the not important to safety earthquake damage indicators, then it can be concluded that the earthquake was non-damaging and further examinations/inspections are considered unnecessary for earthquake ground motion of less than the SL-2 level.

To further evaluate the effect of the earthquake on the functionality of nuclear plant SSCs, it is recommended that surveillance tests be performed to verify that the limiting conditions for operation as defined in the plant technical specifications are met. In general, integrated containment leak rate tests are not considered necessary if no significant physical or functional damage is found in these focused examinations and inspections.

The rationale for excluding the containment leak rate tests under these conditions is the judgment that the benefits of performing these tests are not considered sufficient to warrant the extensive effort and time (several weeks) associated with these tests for earthquake levels which caused no damage to SSCs important to safety or to earthquake damage indicators not important to safety.

Furthermore, previous evaluations have concluded that earthquake ground motions that have not caused damage to important to safety items induce seismic stress levels in containment structures that are well within these design bases. During surveillance testing, the vibration of rotating components (e.g., fans and pumps) is to be closely monitored. Typical surveillance tests for BWRs and PWRs are listed in Annex IV. The time required to perform these tests is plant specific and could range from about one week for some plants to as long as four weeks for others, depending on the technical specifications.

The recommended focused inspections are detailed in nature, as outlined in Table 8. They are based on observed malfunctions or damage to typical power plant SSCs from actual earthquakes. They are not cursory physical evaluations of a small sample of SSCs. These evaluations include a representative number of samples of virtually all classes of SSCs important to safety, both passive and active. Pre-earthquake identification of the type and classification of SSCs, as well as identification of assumed damage locations and patterns for each SSC type due to earthquake loadings, is very effective in performing the initial focused inspections and tests in an efficient and comprehensive manner. Moreover, it is

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8])

Item	Equipment/structure	Types of inspections
	Equipment	
1	Fans	1. Check equipment anchorage/isolation mounts for damage; e.g., stretching or loosening of anchor bolts or nuts; rocking or sliding of equipment 2. Check for damage to attached conduit and ground straps 3. Check for damage or distortion to fan housing tearing off fabric noise eliminators due to seismic loads imposed by attached ducts 4. Check for evidence of excessive fan vibration and/or noise. This may be an indication of misalignment between the motor fan shafts 5. Check clearance between fan wheel and housing 6. Check for damage due to impact or earthquake induced flooding or spraying 7. Check for belt tightness and/or slippage; e.g., belt smoke/odour 8. Check local alarms, breakers and protective devices for actuation/trips
	Air compressors	1. Check equipment anchorage/isolation mounts for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage due to impact or earthquake induced flooding or spraying 3. Check for excessive noise and/or vibration 4. Check for air leaks if compressor is running continuously rather than cycling on and off 5. Check for belt tightness and/or slippage; e.g. belt smoke/odour 6. Check local alarms, breakers and protective devices for actuation/trips

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Static inverters and battery chargers	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts; rocking or sliding of equipment 2. Check for damage to attached conduit and ground straps 3. Check for distortion of cabinet structure 4. Open cabinet, check to see that internally mounted components are secure and undamaged 5. Check for damage due to impact or earthquake induced flooding or spraying 6. Check local alarms, breakers and protective devices for actuation/trips
	Battery racks	1. Check battery rack anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts, evidence of rocking or sliding of racks 2. Check for distortion of rack structure 3. Check for evidence of rocking or sliding of batteries on the racks, buckling or distortion of busbars and the condition of spacers between batteries 4. Check for damage due to impact or earthquake induced flooding or spraying 5. Check busbars, cables and ground straps for damage, distortion or chaffing 6. Check local alarms, breakers and protective devices for actuation/trips

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Air handlers	1. Check equipment anchorage/isolation mounts for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage to attached conduits and ground straps 3. Check for damage to air handler arising from seismic loads imposed by attached ducts or tearing of fabric noise eliminators 4. Check for damage arising from impact of earthquake induced flooding or spraying 5. Check for belt tightness and/or slippage; e.g. belt smoke/odour 6. Check local alarms, breakers and protective devices for actuation/trips
	Chillers	1. Check equipment anchorage/isolation mounts for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage to attached conduits and ground straps 3. Check for leakage or damage to chiller components arising from seismic loads imposed by attached ducts and piping 4. Check for damage arising from impact of earthquake induced flooding or spraying 5. Check for belt tightness and/or earthquake induced flooding or spraying 6. Check local alarms, breakers and protective devices for actuation/trips 7. Check for refrigerant leakage

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Transformers	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage to attached conduits and ground straps 3. Check oil reservoir level 4. Check the nitrogen blanketing system and fire deluge system for damage 5. Check for damage arising from impact of earthquake induced flooding or spraying
	Vertical pumps	1. Check equipment base plate and anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts and equipment movement 2. Check casing below base plate for damage arising from ground settlement/movement 3. Check for evidence of excessive noise and/or vibration and seal misalignment between the motor and pump shaft 4. Check for damage to pump housing from seismic loads imposed by attached piping 5. Check for damage to shaft housing 6. Check for damage arising from impact or earthquake induced flooding or spraying 7. Check all local alarms, breakers and protective devices for actuation/trips 8. Check pump and motor bearings for overheating or lubrication problems 9. Check for damage to attached conduit and ground straps

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Horizontal pumps	1. Check equipment base plate and anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts and equipment movement 2. Check for evidence of excessive noise and/or an indication of misalignment between motor and pump shaft 3. Check for damage to pump housing arising from the seismic loads imposed by attached piping 4. Check for damage arising from impact of earthquake induced flooding or spraying 5. Check local alarms, breakers and protective devices for actuation/trips 6. Check pump and motor bearings for overheating or lubrication problems 7. Check for damage to attached conduit and ground straps
	Motor generators	1. Check equipment anchorage/isolation mounts for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for noise and/or vibration caused by misalignment between motor and generator shaft, especially if they are not mounted on a common base 3. Check for damage to attached conduits and ground straps 4. Check for damage arising from impact of earthquake induced flooding or spraying 5. Check local alarms, breakers and protective devices for actuation/trips

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Motor control centres	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage to attached conduits and ground straps 3. Check for distortion of cabinet structure 4. Open cabinet, check to see that all internally mounted components, including relays and breakers, are secure and undamaged 5. Check for damage arising from impact of earthquake induced flooding or spraying 6. Check controls, breakers and protective devices for actuation/trips
	Low voltage switchgear	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage to attached conduits and ground straps 3. Check for distortion of cabinet structure 4. Open cabinets, check to see that all internally mounted components, including relays and contacts, are secure and undamaged 5. Check for damage arising from impact of earthquake induced flooding or spraying 6. Check local alarms, breakers and protective devices for actuation/trips 7. Reset any trips. Investigate any retrips after reset

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Medium voltage switchgear	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts; rocking or sliding of equipment 2. Check for damage to attached conduit and ground straps 3. Check for distortion of cabinet structure 4. Open cabinets, check to see that all internally mounted components, including relays and contacts, are secure and undamaged 5. Check for damage arising from impact of earthquake induced flooding or spraying 6. Check local alarms, breakers and protective devices for actuation/trips 7. Reset any trips. Investigate any retrips after reset
	Distribution panels	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts; rocking or sliding of equipment 2. Check for damage to attached conduit and ground straps 3. Check for distortion of cabinet structure 4. Open cabinet, check to see that all internally mounted components are secure and undamaged 5. Check for damage arising from impact of earthquake induced flooding or spraying 6. Reset any tripped breakers. Investigate any retrips after reset
	Fluid, air and motor operated valves	1. Check for damage or distortion at attachment or operator to valve body 2. Check for damage to attached conduit/tubing or ground straps 3. Check for damage arising from impact of earthquake induced flooding or spraying 4. Check local alarms, indicators and protective devices for actuation/trips 5. Stroke valve in both directions to check operation

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Engine generators	1. Check equipment anchorage/isolation mounts for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for damage to attached piping, ducts, conduits and ground straps 3. Check for noise and/or vibration arising from misalignment between engine and generator, especially if they are not mounted on a common base 4. Check for damage arising from impact of earthquake induced flooding or spraying 5. Check local alarms, breakers and protective devices for actuation/trips
	Instrument racks	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of equipment 2. Check for distortion of rack structure 3. Check for damage to attached conduit and ground straps 4. Check to see that instruments mounted to racks are secure and undamaged 5. Check for damage arising from impact of earthquake induced flooding or spraying 6. Check local alarms, breakers and protective devices for actuation/trips 7. Reset any trips. Investigate any retrips after reset
	Sensors	1. Check for damage to attached conduit/tubing and ground straps 2. Check for damage arising from impact of earthquake induced flooding or spraying 3. Verify sensor operation with readout check at local/control room indicators

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
1	Control and instrumentation cabinets	1. Check equipment anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts; rocking or sliding of equipment 2. Check for distortion of panel structure 3. Check for damage to attached conduit and ground straps 4. Check to see that instruments, gauges, controls and other equipment mounted to panels are secure and undamaged 5. Check for damage arising from impact of earthquake induced flooding or spraying 6. Check local alarms, breakers and protective devices for actuation/trips 7. Reset any trips. Investigate any retrips after reset
2	Low pressure storage tanks	1. Check tank anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts, deformation of bolt chairs, rocking or sliding on the base 2. Check for damage to attached piping and ground straps 3. Check for cracking or leakage at the base plate to cylindrical shell connection 4. Check for cracking or leakage at the base plate to cylindrical shell connection 5. Check for damage arising from impact of earthquake induced flooding or spraying
3	High pressure tanks and heat exchangers	1. Check for damage to anchorage; e.g. stretching or loosening of anchor bolts or nuts, rocking or sliding of base plates on concrete 2. Check for damage to attached piping

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
4	Piping	1. Check for snubber damage; e.g. snubbers pulled loose from foundation bolts, evidence of excessive travel, jamming of inertia mechanism/leakage of hydraulic fluid and bent piston rods 2. Check for damage at rigid supports; e.g., deformation of support structure, deformation of pipes arising from impact with support structure 3. Check for damage or leakage of pipes at rigid connections; e.g., at anchor points with other equipment and structures 4. Check for damage or leakage of piping and branch lines 5. Check for damage to pipes at building joints and interfaces between buildings 6. Check for damage arising from impact of earthquake induced flooding or spraying
5	Electric raceways	1. Check for deformation of deadweight supports and sway bracing 2. Check for damage to cables at building joints and interfaces between buildings 3. Check for damage arising from impact of earthquake induced flooding or spraying
6	Air handling ducts	1. Check for deformation of deadweight supports and sway bracing 2. Check for damage to ducts at joints 3. Check for damage to ducts at building joints and interfaces between buildings 4. Check for damage arising from impact of earthquake induced flooding or spraying 5. Check for tearing or fabric transition/noise eliminators 6. Check for damage to internal filters and racks

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

<table>
<tr><th>Item</th><th>Equipment/structure</th><th>Types of inspections</th></tr>
<tr><td>7</td><td>Steel framed structures</td><td>1. Check for damage to bolted or welded connections
2. Check for damage to anchorage; e.g., stretching or loosening of anchor bolts or nuts, rocking or sliding of base plates on concrete
3. Check for distortion or buckling of braces and other compression members</td></tr>
<tr><td>8</td><td>Reinforced concrete structures (buildings, containment, cooling towers, intake structure and masonry walls)</td><td>1. Check for new open cracks and spalling of concrete. Minor cracks, even if caused by the earthquake, are not considered significant unless they are large enough to result in yielding of rebars[a]. Example guidance for the definition of significant cracks is as follows:
<table>
<tr><th>Crack size</th><th>Guidance</th></tr>
<tr><td>≤0.5 mm</td><td>Insignificant crack unless near expansion anchor, in which case anchorage tensile capacity can be reduced</td></tr>
<tr><td>0.5–1.5 mm</td><td>Should be mapped. Not likely to be significant to structural capacity</td></tr>
<tr><td>1.5–3.0 mm</td><td>Indicates yielding of rebars has occurred. Need to assess cause. Unlikely to have significantly degraded structural capacity</td></tr>
<tr><td>≥3.0 mm</td><td>Either rebars are absent or have significantly yielded. Need to assess cause. May degrade structural capacity</td></tr>
</table>
2. Check for evidence of ground settlement
3. Check for evidence of differential horizontal and vertical movements between adjacent and/or interconnecting buildings/structures</td></tr>
</table>

For footnote see end of table.

TABLE 8. VISUAL INSPECTION OF EQUIPMENT AND STRUCTURES AFTER AN EARTHQUAKE (based on Ref. [8]) (cont.)

Item	Equipment/structure	Types of inspections
9	Primary coolant system	1. Check for reactor coolant leakage at flanged joints; e.g., control rod drive mechanisms (CRDMs) 2. Check for condition of supports and snubbers for large components; e.g., main coolant pumps, steam generators and pressurizer 3. Check condition of CRDM support structure (PWRs only)
10	Buried pipes	1. Check for damage or leakage at pipe interface with buildings and tanks 2. Fire main leakage will be evidenced by self-excavation and actuation of backup fire pumps 3. Check fire main, service and circulating water piping, especially dead legs, corrosion and growths which are knocked loose by earthquake motion. These loosened accumulations can clog screens and small diameter pipes such as fire hose hydrants. Checks for clogging and flushing of pipe mains are necessary

[a] Rebars: reinforcing bars.

recommended to adjust the physical evaluation procedure to such damage locations and patterns (see Annex I).

Upon successful completion of these tests, the plant is considered ready for restart. Startup should proceed in accordance with normal startup procedures.

5.3.1.2. Scope of the initial focused inspections and tests

The SSCs included in the focused examinations and inspections are selected to sample all types of SSCs important to safety found in the nuclear power plant, and include items which are considered most likely to be damaged by an earthquake. The focused examinations and inspections also include non-seismic SSCs not important to safety which are known to be of lower seismic capacity to serve as earthquake damage indicators. These inspections are performed by engineers experienced in the observation and evaluation of earthquake related damage to industrial and power facilities. The purpose of these examinations and inspections is to determine the need for expanded examinations and inspections and tests, as well as to verify damage levels, which relate directly to actions to be performed for restart.

Typical SSCs classified by type are as follows:

(a) Active SSCs:
  (i) Pumps (vertical, horizontal and reciprocating pumps);
  (ii) Motors;
  (iii) Valves;
  (iv) Fans;
  (v) Turbines (main turbine and turbine for pump drive);
  (vi) Generators;
  (vii) Diesel generators;
  (viii) Other SSCs (freezer, air compressors, dampers, control rod drive mechanism (CRDM), fuel exchanger and overhead cranes).
(b) Passive SSCs:
  (i) Reactor (pressure) vessel;
  (ii) Steel framed structures;
  (iii) Reinforced concrete structures (building, containment, cooling towers and intake structure) and masonry walls;
  (iv) Core internals, including other in-core components (from a symptomatic standpoint);
  (v) Piping;
  (vi) Heat exchangers;
  (vii) Tanks;
  (viii) Heating, ventilation and air-conditioning ducts;
  (ix) Cranes and hoists;
  (x) Spent fuel pool;
  (xi) Fuel assembly (from a symptomatic standpoint);
  (xii) Other SSCs (air handlers, chillers, accumulators, filtered demineralizers, strainers/filters, air exhausting machines and dehumidification tower).
(c) Electrical SSCs:
  (i) Transformers;
  (ii) Batteries;
  (iii) Breakers, switchgears and motor control centres;
  (iv) Instrumentation equipment and structures;
  (v) Panels and cabinets (control, instrumentation, distribution, etc.).

If the nuclear power plant contains only a small number of items within a particular class of SSCs (e.g., battery racks, diesel generators and buildings), all such items are physically evaluated. However, if the nuclear power plant contains a large number of items within a particular class of SSCs (e.g., valves, piping and cable trays), the physical evaluations are performed on a sampling basis. The items to be examined and inspected are preselected to be a representative sample of the variety of systems and components within the SSC class and include those items considered most likely to be damaged by an earthquake (e.g., items located at higher elevations of the building and flat bottomed vertical tanks). As a guide, for most equipment and structures, it is recommended that the sample include about 20% of the total number of items, but no fewer than two items in each class. For equipment and structures with relatively high seismic capacity (e.g., pumps, piping and cable trays), a sample size of less than 20% is considered reasonable. However, for equipment and structures with relatively low seismic capacity (e.g., low pressure storage tanks), a sample size greater than 20% is recommended. For civil structures (steel and concrete), the extent of the physical evaluations includes all safety related structures, but focuses on a representative sample of the construction details indicated in items 7 and 8 of Table 8. For steel structures, this includes bolted connections, anchor bolts and compression braces. For concrete structures, this would include representative areas of concrete structures that are considered to be susceptible to damage (e.g., areas with high moments or shear).

The size of the sample is decided by the operating organization with the concurrence of the regulatory body as required. The larger the sample size, the better the case that can be made that a non-damaging earthquake was truly non-damaging to nuclear plant SSCs.

5.3.1.3. Decision/next steps

Upon successful completion of these inspections and tests (including surveillance tests), the plant is considered ready for restart. Startup would be in accordance with normal startup procedures. This is applicable to nuclear power plants designated to be in Action Levels 1, 2, 3 or 5. During startup following a shutdown due to an exceedance of the SL-1 design basis, particular attention is paid to the following:

(a) Primary coolant system leakage;
(b) Reactor coolant pump seal leakage (for PWRs);
(c) Vibration of rotating equipment, for example, pumps and fans.

If the initial focused inspections and tests identify significant earthquake damage to SSCs, then proceed to the expanded inspections and tests.

5.3.2. Expanded inspections and tests

5.3.2.1. Important to safety SSCs

If significant physical or functional damage is found in the important to safety SSCs selected for the focused post-shutdown inspections and the previously determined action level was not Action Level 4 or Action Level 8, the action level is changed to Action Level 4 or Action Level 8 depending on the earthquake level. For earthquakes less than the SL-2 basis level, Action Level 4 is assigned; for earthquakes greater than SL-2, Action Level 8 is assigned.

For Action Levels 4 and 8, the inspections are expanded to include a more complete inspection of the plant, including additional physical and functional evaluations. Additional SSCs to be included in the expanded inspections are identified in Section 5.3.2.3. All reported malfunctions and damage need to be evaluated, and repaired or corrected as required.

For Action Levels 4, 6b, 7b and 8, a root cause analysis of the malfunctions and/or damage to SSCs important to safety and to SSCs not important to safety is performed. Root cause analyses may determine that:

(a) The seismic demand to which the SSCs were subjected was greater (perhaps significantly greater) than the design or qualification level, which caused failure;
(b) Weaknesses in the design or manufacture of the SSC existed, which caused malfunction and/or damage;
(c) Inadequate support or anchorage caused malfunction and/or damage.

Each of these, and other possible results, requires subsequent actions:

(1) In the case of the seismic design or qualification level being underestimated, a complete evaluation of the development of these input motions is performed.
(2) For Action Level 4, if the damage to safety related SSCs observed in the focused inspections and the root cause analyses isolate the damage or failure to a specific class (or classes) of SSCs, and the cause of the damage is attributable to a specific design or installation deficiency, for example, lack of anchorage or improper installation of anchor bolts, then the expanded examinations and inspections may be limited to the affected class (or classes) of SSCs. For Action Level 8, additional requirements are given in Section 6.

TABLE 9. METHODS AND FOCUS POINTS OF FUNCTIONAL TESTS

Categorization	Method of functional tests	Focus point
Active SSCs	Test run	Smoothness of drive Seal leakage Noise Vibration Temperature
Passive SSCs	Leak test Pressure test	Leakage (leak rate)
Electrical SSCs	Loop test Isolation test	

A general description of functional tests by category of SSCs is contained in Table 9.

Surveillance tests are performed to verify that the limiting conditions for operation as defined in the plant technical specifications are met. During surveillance testing, the vibration of rotating equipment (e.g., fans and pumps) is closely monitored. See Annex IV for a list of typical surveillance tests for BWR and PWR plants.

Integrated containment leak rate tests are recommended at this level of observed damage. If plant operators have the means of monitoring the integrity of the containment on-line, then performing containment leak rate tests prior to startup may not be considered necessary.

For Action Level 4, upon the successful completion of all evaluations, expanded inspections, physical evaluations and tests, the plant would be considered ready for restart. Startup would be in accordance with normal startup procedures.

For Action Level 8, it is recommended that the reactor vessel head be removed for visual inspections of the reactor vessel internals, fuel and control elements and their support structure, as part of the expanded inspections. For Action Level 8, the long term evaluations of Section 6 are performed prior to startup.

5.3.2.2. Not important to safety SSCs

In the event that no significant physical or functional damage is found in the SSCs important to safety during the focused post-shutdown inspections, the assignment of Action Levels 1–3 and 5–7, with respect to damage to SSCs important to safety, is confirmed. If there is newly discovered damage to SSCs not important to safety, the initial damage and action levels assigned to the plant need to be reviewed and changed if appropriate. The guidelines of Section 5.1 then apply to the new action level.

5.3.2.3. Scope of expanded inspections

Expanded inspections include all safety related equipment and structures as well as non-safety-related SSCs required for normal operation of the plant. As a minimum, the expanded inspections include the following:

(a) All SSCs important to safety (and their supports) not included in the focused inspections. This would include 100% of the items that were inspected on a sampling basis in the limited inspections.
(b) All distribution systems important to safety (and their supports).
(c) SSCs not important to safety but required for power generation (e.g., turbine generator, feedwater system and switchyard equipment).
(d) Primary reactor coolant system (e.g., reactor vessel, main coolant pumps, steam generators, pressurizer, piping, and piping and component supports).
(e) For Earthquake Level 3, it is recommended to include in-core structures and the reactor fuel assembly in the evaluation unless:
    (i) The frequency range of the exceedances of the in-structure earthquake responses differs from the frequency range of the in-core structures and fuel assemblies, for example, Earthquake Level 3a.
    (ii) The integrity of the in-core support structures and fuel assemblies was verified by similarity to tests or earthquake experience, where the earthquake environment exceeds that of the actual felt earthquake.
    (iii) The integrity of the in-core structures and fuel assemblies was previously verified for an RLE that exceeds the earthquake.
    (iv) Comparative analyses verify integrity.
(f) Control rod drive mechanisms and hydraulic control units.
(g) Buildings and structures important to safety, and their penetrations.
(h) Containment including containment penetrations.
(i) Intake structure canals, piping and other SSCs as part of the ultimate heat sink.
(j) Dam and reservoir (if needed to preclude unacceptable flooding or loss of the ultimate heat sink).
(k) Buried pipes important to safety, to include interfaces with buildings and tanks; and buried pipes not important to safety at locations where failure or damage could have an adverse effect on SSCs important to safety, i.e. seismic system interaction issues.

5.3.2.4. Decision/next steps

Upon the successful completion of these inspections and tests (including the surveillance tests), the plant is considered to be ready for restart. Startup would be in accordance with normal startup procedures. This is applicable to nuclear power plants designated to be in Action Levels 1, 2, 3, 5, 6a and 7a. This may also be applicable to Action Levels 6c and 7c, depending on the results of the evaluations of the low frequency ground motion exceedances. During startup following a shutdown due to an exceedance of the SL-1 design basis, particular attention should be paid to the following:

(a) Primary coolant system leakage;
(b) Reactor coolant pump seal leakage (for PWRs);
(c) Vibration of rotating equipment; for example, pumps and fans.

5.3.3. Non-destructive examinations

Non-destructive examinations encompass a large number of possible test procedures to verify the physical integrity and functionality of SSCs. Some of these tests are performed routinely without the occurrence of an earthquake. For these cases, additional testing may not be necessary — the regularly scheduled tests will suffice.

Examples of the NDEs to be employed are:

(a) *All SSCs:*
  (i) Visual/audio inspections as discussed previously;
  (ii) Vibration tests.
(b) *Leaktightness:*
  (i) Air or water pressure tests for leaktightness;
  (ii) Liquid dye penetrant inspections for leaktightness.
(c) *Concrete structures and structural elements — simple:*
  (i) Hammer sounding/chain dragging;
  (ii) Rebound hammer — general concrete soundness/strength.
(d) *Concrete structures and structural elements — moderate:*
  (i) Cover meter/pachometer — locations and sizes of bars;
  (ii) Half-cell — bar corrosion;
  (iii) Concrete thickness gauge;
  (iv) Ultrasonic thickness gauge.

(e) *Concrete structures and structural elements — complex:*
  (i) Ultrasonic pulse velocity — strength, honeycombed or cracked;
  (ii) Corrosion rate;
  (iii) Impact echo impulse response — thickness and support integrity;
  (iv) Ground penetrating radar — location of deeply embedded items inside/below concrete.
(f) *Condition of metallic material:*
  (i) Magnetic particle inspection;
  (ii) X ray inspections for flaw detection;
  (iii) Ultrasonic tests;
  (iv) Vickers hardness test.
(g) *Anchorage:*
  (i) Percussion tests — anchorage;
  (ii) Torque tests — anchorage;
  (iii) Ultrasonic tests — anchorage.
(h) *Hidden damage:* Disassembly.

5.3.4. Comparative analyses

Comparative analyses are seismic analyses of soil and rock, foundations, structures and subsystems in which the input is defined by the motions actually recorded in an earthquake. Seismic safety evaluations of revised seismic hazard definitions are treated in Section 6.

In general, for Earthquake Level 3b and 3c events, seismic response analyses are performed to evaluate the state of the SSCs designed by analysis or for which the seismic design/qualification environment is defined by structure seismic responses, for example, structure loads and ISRS. As in other activities, the seismic analyses are tiered to be efficient and provide the maximum amount of information. For example, if the ISRS arising from the earthquake (as measured or calculated) are less than those specified in the design/qualification of subsystems, it can be assumed that the design stresses were not exceeded.

For Earthquake Level 2 or 3a events, it may be informative to perform analyses to benchmark the design analyses, but these analyses are not specifically required by this programme. Individual Member States may require to make additional efforts in this regard.

5.3.4.1. Input motion to seismic response analyses: soil, rock and structure

If acceleration–time history records are available, they should be used as the starting point for the analyses as follows:

(a) The preference is for free field records at the same location as the definition of the SL-1 or SL-2 design basis to be available and to be used. If this is not possible because SL-1 or SL-2 were defined at locations other than the top of grade free surfaces, for example, at a rock outcrop at depth, analyses may need to be performed to generate the earthquake motion at this control point location. In either case, the input to the soil, rock, foundations and structure is defined by the free field motion and SSI analyses performed with this input motion. Responses are generated at points of interest in the structures.
(b) If free field records are unavailable and records on the foundation of the structures are available, these are used in the structure response analyses. Care must be taken that foundation rotational inputs are developed from the translational records or, if this is not possible, accounted for in a consistent, acceptable manner. One criterion as to the acceptability of these analyses is the comparison of recorded motions in the structure with those generated by the analyses.

5.3.4.2. Seismic response analyses of structures and structural elements and components

Seismic response analyses of structures serve two purposes: generating loads or the equivalent responses for checking the structure design; and generating input to subsystems. In general, these analyses will be time history analyses, as discussed elsewhere in this report. The end products for structural loads may be the loads themselves (the internal forces and moments or stresses in members) or a structure response quantity that can be used in a subsequent or second stage analysis of the structure. These structure response quantities are most often peak accelerations. Acceptance criteria for structure loading conditions are discussed in Section 5.3.8.

5.3.4.3. Input motion to seismic response analyses: sample of subsystems

The input to subsystem seismic analyses may be based on recorded motions at the subsystem support location(s) or on the results of the soil, rock and structure analyses. An important principle is that the responses calculated for the recorded earthquake motions are best estimate values, i.e. best estimate parameters, such as material properties of rock, soil and structures, and best estimate SSI/structure response analysis procedures. Application of conservatism in the seismic design response procedure is minimized for comparison purposes.

In-structure response spectra are generated for all elevations of interest on the basis of actual earthquake records. The preferred analysis method to generate ISRS is time history analysis, including SSI, if appropriate. An alternative approach is to use the direct generation approach, which, if used, needs to be verified as applicable for the modelling being performed, for example, for situations where translations and rotations are important inputs.

If the nuclear power plant does not have SL-2 ISRS, proceed directly to the recommended seismic evaluations discussed in Section 6.

5.3.5. Surveillance tests

Surveillance tests are those performed at regular intervals to demonstrate the availability and operability of components and systems. Surveillance tests are identified in the technical specifications of a plant and consist of checks, tests, calibrations and inspections to verify the availability and performance of the tested component or system.

For all action levels, it is recommended that surveillance tests of either a sample or of all the SSCs important to safety be conducted prior to restart of the plant. For Action Levels 1–3 and 5, a sample is adequate. In general, SSCs important to safety have been designed and/or qualified for the SL-2 design basis and they often have significant margin — the margin to code acceptance criteria and a significant margin to failure. In selecting the sample of SSCs important to safety for surveillance testing, SSCs not known to have significant margin are selected. It is necessary that the size of the sample and its composition be agreed with the regulatory body and determined on a case by case basis.

Since this report recommends implementing functional tests in the course of the post-earthquake inspections and tests, the functionality of many SSCs important to safety has already been confirmed. These items may serve as the sample, with the agreement of the regulatory body.

In general, surveillance tests are conducted for structurally complicated active equipment, for example, the CRDM and ECCS pumps. For passive SSCs, for example, tanks, it is expected that surveillance tests are determined on an as-needed basis, taking into account the earthquake level and the estimated margin with respect to the code allowable values.

5.3.6. Startup tests

A startup test is recommended in addition to the component level inspection in the course of short term actions and post-shutdown inspections and tests, in order to confirm the plant parameters and the functionality of those components for which this can only be verified through these tests. For Earthquake Level 1, the startup tests may be omitted because damage to the SSCs important to safety is highly unlikely. The purposes of the startup tests are:

(a) Confirmation of system level functions, for example, output control and feedwater control;
(b) Confirmation of functions that cannot be checked in the shutdown state, for example, confirmation of the steam system and of the support structure under thermal fluctuations;
(c) Identification of unexpected damage.

It is recommended that a startup test be conducted for a period longer than that needed to acquire the fluctuation data, to verify the operability of the systems under operating conditions.

5.3.7. Inspection teams

Post-shutdown inspections and tests are generally performed by inspection teams composed of personnel from several organizations, for example, plant operating organizations, suppliers, engineering firms and maintenance services. Coordination and training of the inspection teams are required to assure consistent evaluations and judgements by the different teams.

It is desirable to have representatives from each of the various disciplines in each of the teams, especially engineering personnel knowledgeable in earthquake investigations to identify earthquake damage by comparison with damage from other sources. Plant operations and maintenance personnel are important to identify operational issues and past performance of SSCs of interest.

Supplier personnel are important to the teams, to identify the typical operational failure modes or characteristics to consider before the SSC damage or failure occurred. The teams should be knowledgeable in, for example, the design, normal operating conditions and inspection history of the SSCs of interest. In addition, the teams, in conjunction with others, will determine when NDEs should be performed.

A consistent approach to these issues requires training and continuous coordination between the teams and the personnel from other disciplines.

5.3.8. Acceptance criteria

5.3.8.1. Comparison between calculated loads arising from the earthquake and design seismic loads

For structural loads, the calculated loads arising from the earthquake are compared with the DBE values for the SL-1 or SL-2 design basis to determine whether the design conditions have been exceeded. This comparison can be made at intermediate or final stages of the design process. If the design loading conditions exceed the loading condition arising from the earthquake, no further evaluation for exceedance is required.

For subsystems, the calculated ISRS based on the actual earthquake records should be compared with the SL-2 ISRS. If the calculated ISRS for any location of interest are enveloped by the SL-2 ISRS, i.e. are less than the SL-2 ISRS, then the design basis for SSCs important to safety supported at these locations has not been exceeded, and seismic evaluation of SSCs important to safety supported at these locations is not required. However, if the calculated ISRS for any locations exceed the SL-2 ISRS at frequencies important to the SSC, then the design basis for the systems and components important to safety, as well as for the structure itself, may have been exceeded and further evaluations should be performed (see Section 6).

In addition to these checks of the state of stress or generalized forces induced in the SSCs compared with the design values, a requirement is to assess the impact of the earthquake loading on the existing and future condition of the SSCs. Has the earthquake used up some of the life of the SSCs to the extent that it has significantly reduced the fatigue life (see Annex III for a discussion of fatigue life as a function of plastic deformation)? Has the earthquake caused accelerated ageing of SSCs due to phenomena such as displacements introducing misalignment of components, and thereby reducing the operating life of the component? Has the earthquake induced damage in the protective coatings of various SSCs, which could lead to premature or associated material degradation? These assessments should be made and also made available to operations and engineering staff for incorporation into future programmes (for assessments of seismic fragility, see Section 6.3).

Acceptance criteria for seismic evaluations are given below for:

(a) The SSCs typically qualified by analysis;
(b) Equipment typically qualified by methods other than analysis, i.e. from tests or seismic data from experience.

The acceptance criteria may take different forms depending on the earthquake level and the characteristics of the earthquake motions. In some cases, a damage indicating parameter, such as CAV, may provide an overall measure of damage. If the value for the earthquake is less than a threshold value and either no damage has been observed or none is suspected, the criteria are judged to have been met. In any case, if the comparison of seismic responses (calculated or measured) stemming from the earthquake with the design values, for example, ISRS, peak accelerations and stresses, shows that those arising from the earthquake are less than the design values and no damage has been observed, it is considered that the criteria have been met.

Acceptance criteria are dependent on the action level. They do not require compliance with the allowable stress criteria normally used for design purposes because the applied load is known, the equipment is available for inspection and evaluation and, therefore, structural margins need not be as high as in an original design. This is consistent with the approach normally used to evaluate, after the event, the effect of plant upsets (e.g., waterhammer events and overpressures) on SSCs. However, care must be taken to assure that the actual earthquake has not significantly affected the service life of SSCs.

5.3.8.2. Structures, systems and components qualified by analysis

The following acceptance standards are recommended for the SSCs typically qualified for seismic loads by analysis, i.e. passive SSCs, for example, piping, piping and component supports, building structures, pressure vessels and tanks, and some other types of mechanical equipment:

(a) If the calculated stresses from the actual seismic loading conditions are less than those allowable for emergency conditions (e.g., Code Level C service limits of the American Society of Mechanical Engineers, or the equivalent in other Member States, for SSCs designed to the limit state requirements of Standard ASCE 43-05 of the American Society of Civil Engineers [27], or the equivalent in other Member States) or the original design bases, then the item is considered acceptable.
(b) If the calculated stresses exceed the acceptance criteria specified in (a) above or the equivalent, acceptability of the item should be based on the following considerations:
    (i) The results of detailed visual inspections;
    (ii) An engineering evaluation of the effects of the calculated stresses on the functionality of the item, including a fatigue assessment if deemed necessary;
    (iii) The results of SSC operability tests.
(c) If the calculated stresses are greater than those allowable for faulted conditions, then the acceptability of the item should be based on the following considerations:
    (i) The results of a detailed visual inspection;
    (ii) An engineering evaluation of the effects of the calculated stresses on the functionality of the item, including a fatigue assessment if deemed necessary;

(iii) The results of SSC operability tests;
(iv) The results of additional NDEs of the item, for example, examinations of specific areas of the item that are found to be highly stressed, deformed or are a concern on the basis of component specific evaluations;
(v) Repair or replacement of potentially damaged areas.

5.3.8.3. SSCs qualified by methods other than analysis

The following acceptance standards are recommended for equipment typically qualified for seismic loads by methods other than analysis, for example, relays, switches, electrical equipment and some types of mechanical equipment. Such electrical and mechanical equipment is considered acceptable *if one or more* of the following conditions are met:

(1) Test response spectrum (TRS). For the SSCs qualified by this test, the TRS envelopes the calculated response spectrum based on the actual earthquake record.
(2) Generic equipment test response spectra (GETRS) developed to represent families of qualification or fragility tests with a conservatism factor applied. One example of such data is the generic equipment ruggedness spectrum (GERS) developed in the USA. If GETRS divided by a reduction factor to introduce margin, such as 1.3, envelopes the calculated response spectrum based on the actual earthquake record, the component may be assumed to be acceptable. For the USA, GERS for relays, thirteen classes of electrical equipment and four classes of mechanical equipment (valves) are published in Refs [28, 29]. These data can also be used to evaluate equipment qualified by a test that does not meet condition (1) above.
(3) The equipment is considered to be qualified for further operation on the basis of data from experience. Earthquake performance data for twenty classes of nuclear plant mechanical and electrical equipment are contained in Ref. [30]. Bounding spectrum values and limitations or caveats applicable to these data are published in Ref. [30].

If none of the above conditions are met, then the acceptability of the item is evaluated on the basis of a combination of the following considerations:

— The results of a detailed visual inspection;
— The results of equipment operability tests;
— The results of additional NDEs of the item.

5.4. DECISION FOR RESTART OR ANOTHER NORMAL OPERATIONAL STATE

After conducting all relevant activities described in Sections 4–6, a decision is required as to whether:

(a) The plant can be restarted, and what additional conditions may be imposed on the plant as a precondition to restart.
(b) The plant should be maintained in a safe shutdown condition.

The decision as to the next steps requires approval by the regulatory body. In addition to safety issues, the operating organization will focus on economic issues.

6. LONG TERM ACTIONS

6.1. STRATEGY TO BE FOLLOWED

Long term actions are those actions required to demonstrate that significant and adequate seismic margin of the nuclear power plant exists, and they need to be carried out either before or after restart because:

(a) The perception of the seismic hazard at the site may have changed due to occurrence of the earthquake;
(b) The physical state of the nuclear power plant may have changed after being subjected to the earthquake. The physical state of the plant SSCs may have degraded due to the loading environment imposed on the plant. In some cases, short and long term actions may involve repair, upgrading or replacement of SSCs, or portions thereof, which may increase their seismic capacity to equal or exceed that existing before the earthquake occurred. The assessments should take all of these situations into account.

The following elements comprise the long term actions:

(a) Seismic hazard evaluation for input to the evaluation of the nuclear power plant seismic capacity;
(b) Seismic safety evaluation of the nuclear power plant SSCs — applying, usually, SMA or SPSA methodologies;
(c) Upgrades of SSCs.

IAEA Safety Guide NS-G-2.13 [5] provides guidance for the post-earthquake evaluations. Selected portions of Ref. [5] are highlighted in the present report.

Specific requirements for long term actions or evaluations are dependent on the action levels defined in Section 5.2 and displayed in Table 2 of this report:

(a) In general, a seismic hazard evaluation is required for all action levels identified with Earthquake Level 3, i.e. the earthquake is greater than the SL-2 level. These are Action Levels 5–8. Note that, within Earthquake Level 3, a distinction is made as a function of the frequency characteristics of the ground motion. This distinction will help to guide the seismic hazard evaluation.

(b) Seismic capacity evaluations for levels beyond the SL-2 DBE may be required for Action Levels 4–8. For Action Levels 6 and 7, these evaluations can be performed after restart. For Action Level 4, a beyond design basis evaluation may be required prior to restart, depending on the results of the root cause analyses of the malfunction, damage or failure of SSCs important to safety. For Action Level 8, a beyond design basis evaluation is required before restart.

6.2. DEFINITION OF SEISMIC INPUT FOR EVALUATION

An initial step for the evaluation of the seismic safety of the nuclear power plant is to establish the seismic hazard. For this purpose, the site specific seismic hazard is comprised of three main elements and should follow fully the guidance provided in Section 4 of Ref. [5] (the reference numbers in this quotation have been renumbered to agree with those in the present report):

“(a) Evaluation of the geological stability of the site [3, 31] with two main objectives:

(i) To verify the absence of any capable fault that could produce differential ground displacement phenomena underneath or in the close vicinity of buildings and structures important to safety. If new evidence indicates the possibility of a capable fault in the site area or site vicinity, the fault displacement hazard should first be assessed in accordance with the guidance provided in Ref. [3]. If a clear resolution of the matter is still not possible, the fault displacement hazard should be evaluated using probabilistic methods.

(ii) To verify the absence of permanent ground displacement phenomena (i.e. liquefaction, slope instability, subsidence or collapse, etc.).

(b) Determination of the severity of the seismic ground motion at the site, that is, the assessment of the vibratory ground motion parameters, taking into consideration the full scope of the seismotectonic effects at the four scales of investigation and as recommended in Ref. [3].

(c) Evaluation of other concomitant phenomena such as earthquake induced river flooding due to dam failure, coastal flooding due to tsunami, and landslides."

The evaluations recommended in (a) and (c) above should be performed in all cases for a programme of seismic safety evaluation, regardless of the methodology used and in accordance with Refs [3, 31, 32].

With respect to (b) above, the vibratory ground motion for the evaluations can take one of several forms depending on the characteristics of the felt earthquake, the purposes for the evaluation of the seismic capacity of the nuclear power plant, and the methodology to be employed. The seismic hazard assessment should provide an input to address the following issues:

(a) Maintaining a margin beyond the original DBE and demonstrating that there is no cliff edge effect, including consideration of earthquake induced changes to the state of the plant SSCs important to safety;
(b) Confirming or re-establishing the reliability and projected long term life of the plant, including consideration of earthquake induced changes to the state of the plant SSCs important to safety;
(c) Calculating risk metrics for comparison with regulatory targets, for example, core damage frequency (CDF) and large early release frequency (LERF);
(d) Revising the original DBE (SL-2) if deemed necessary, which may depend on the size of the ground motion on the site due to the felt earthquake and other information developed from its characteristics, for example, a newly identified fault.

As a result of the seismic hazard assessment, a new site specific seismic hazard may be defined and designated as the review level earthquake (RLE) to be used for the evaluation of the seismic safety and operability of an installation.

To satisfy the objectives related to risk, a site specific probabilistic seismic hazard assessment (PSHA) should be performed. Typically, these objectives entail:

(a) Calculation of risk metrics, for example, CDF and LERF;
(b) Establishment of a risk management tool for risk informed decision making;
(c) Determination of the relative risk of seismic and other internal and external hazards;
(d) Provision of a basis for cost–benefit analysis for decision making in relation to plant operations and upgrades.

6.2.1. Seismic input for seismic margin assessment

For the SMA methodology, an RLE is established. The RLE is a deterministic definition of the seismic input for which the capacity of the nuclear power plant is assessed. In this case, the RLE should be defined with a sufficient margin over:

(a) The original DBE; and
(b) The ground motion of the experienced earthquake, to ensure plant safety and to find any 'weak links' that may limit the plant's capability to safely withstand a seismic event greater than the original DBE and the felt earthquake.

The RLE is either generally defined by ground response spectra at a location in the free field, such as on the free surface top of grade, or at a hypothetical outcrop at depth in the soil. These ground response spectra may be based on:

(a) The ground motion of the experienced earthquake with an amplification factor to add conservatism, such as a factor applied to the recorded motion and some smoothing of the response spectra shapes;
(b) The ground motion of the original design basis with an amplification factor to add conservatism and envelope the experienced earthquake;
(c) A probabilistic or deterministic seismic hazard assessment;
(d) Performance based modification to the PSHA results;
(e) Site independent or general site dependent aggregations of recorded ground motions.

In addition to the ground response spectra, the description of the input motion should include other relevant parameters, such as peak values and time histories of accelerations, velocities and displacements, and duration of the strong motion and damage indicating parameters, for example, the Arias intensity, CAV and JMA intensity appropriate to a realistic estimate of the damage capacity of the input motion. For non-linear analyses, a set of recorded time histories or modified recorded time histories that in total suitably represent the characteristics of the RLE are needed. A structured procedure should be followed for the selection or development of the set of time histories, with defined criteria for determining their adequacy.

The principles detailed in Ref. [31] may be used to determine site specific response spectra.

6.2.2. Seismic input for seismic probabilistic safety assessment

If an SPSA technique is applied to the overall plant, it is necessary to conduct a site specific PSHA, incorporating the data acquired from the earthquake. General guidelines on conducting site seismic hazard assessments are contained in Ref. [3].

For the SPSA methodology, the review earthquake denotes the site specific probabilistic seismic hazard. In general, the results of the site specific PSHA include seismic hazard curves defining the annual frequency of exceedance (often referred to as the annual probability) of a ground motion parameter, such as the mean, median and various fractiles of the parameter. Typical ground motion parameters are the peak ground acceleration and the spectral accelerations (5% damping) at various frequencies. In addition to the seismic hazard curves, uniform hazard spectra are generally available as a function of probability of exceedance and confidence levels. Other important parameters include the characteristics of the dominant source parameters, such as the magnitude and epicentral distance from the site. If risk consistent response spectra are generated, these are also included.

A mean hazard estimate may be adequate for evaluation purposes. (It should be noted that a mean hazard estimate convolved with a mean plant fragility curve will result in a mean failure probability. When the mean fragility curves are for the SSCs in the accident sequences, then the convolution of the mean seismic hazard curve with the accident sequences yields a point estimate of the failure, not a mean failure probability.)

6.3. SEISMIC CAPACITY EVALUATION OF SSCs

Section 5 described the seismic safety evaluation requirements of the nuclear power plant SSCs subjected to the felt earthquake, i.e. to calculate the loadings imposed and their effects on the SSCs. These effects are determined through inspections, tests and analyses. Section 5.3.4 describes the comparative analyses performed for the felt earthquake.

Section 6.1 described the requirements for the evaluation to be performed as a function of the action level. In general, seismic capacity evaluations for beyond the SL-2 DBE and beyond the felt earthquake may be required for Action Levels 4, 6, 7 and 8. For Action Levels 6 and 7, these evaluations may be performed after restart. For Action Level 4, a beyond design basis evaluation may be required prior to restart, depending on the results of the root cause analyses of damage to, or failure of, SSCs important to safety. For Action Level 8, a beyond design basis evaluation is required before restart.

Regarding the methods for seismic safety evaluation, operating organizations may prefer a specific method, such as SMA or SPSA, or alternative approaches which are demonstrated to be acceptable. IAEA Safety Standards Series No. NS-G-2.13 [5] recommends two methods for evaluation of seismic safety of nuclear power plants: deterministic SMA [33, 34] and SPSA [34]. The differences between them lie in the systems modelling approach and in the capacity evaluation methodology. Systems modelling in the former method is carried out by success paths and in the latter method by event trees or fault trees. Capacity evaluations of SSCs are HCLPF values in the former method; in the latter method, capacity methodologies use probabilistically defined fragility functions.

There are many common elements to the SMA and SPSA methodologies [5, 33, 34]. Among the common elements are:

(a) *Plant walkdown* — one of the most important aspects of the methodologies, in-plant reviews are essential to identifying potential vulnerabilities not discernible from drawings.
(b) *Treatment of building structures, the equipment–structure interface, distribution systems, equipment, etc.* — the key issue is to define the function to be performed during and/or after the felt earthquake, determine the 'as is' condition after the earthquake, define the failure mode and quantify the expected performance in terms of the earthquake loading.
(c) *Evaluation of the primary reactor system* — the evaluation is performed by a combination of inspections, analyses and tests (newly performed tests to define the 'as is' condition after the earthquake and the correlation with existing test results for functionality).
(d) *Review of chatter of contact devices (relay chatter)* — the review is a combination of a circuit review and a capacity assessment.
(e) *Review of seismically induced fires and flooding* — those initiated within the SSCs important to safety and those outside the SSCs important to safety with the potential to affect the SSCs important to safety;

(f) *Review of soil related failures (liquefaction, settlement or foundation)* directly related to the capability of SSCs to perform their required functions.

In addition to these common elements, the key elements of the SMA and SPSA methodologies are presented in the following sections.

6.3.1. Seismic margin assessment

Key assumptions are made in the pre-execution phase of the SMA. These assumptions about the following matters may need the approval of the regulatory body:

(a) *The definition of the safety functions to be ensured when the installation is postulated to experience an earthquake as defined by the RLE.* The fundamental safety functions specified in Ref. [1], para. 4.6, are: "(1) control of reactivity; (2) removal of heat from the core; and (3) confinement of radioactive materials and control of operational discharges, as well as limitation of accidental releases." The selected set of SSCs to be evaluated is defined as the success path. One definition of the success path is safe shutdown of the plant (hot or cold shutdown) and maintaining the plant in this condition after the earthquake occurs.

(b) *Plant initiating conditions at the time of the felt earthquake.* For example, loss of off-site power and unavailability of normal on-site power such as from another nuclear power unit or a conventional power generation plant on the site, given that these would be subject to the common cause nature of the earthquake.

(c) *Systems requirements to mitigate earthquake induced plant conditions,* for example, loss of off-site power, and small break loss of coolant accidents inside the containment (small loss of coolant accidents).

(d) *Redundant success paths to be considered,* including assumptions for the availability of SSCs important to safety.

(e) *Availability of outside assistance.* What kind of outside assistance would be needed and when would it be available? The conditions should be established and agreed with the regulatory body; for example:
   (i) Immediately after earthquake induced shaking has stopped; or
   (ii) After a certain period of time (e.g., 24 hours, 48 hours or 72 hours).

Once these and other assumptions have been agreed with the regulatory body, the SMA programme proceeds.

The key elements of the SMA are:

(a) *Selection of the assessment team* — the assessment team should be a multidisciplinary team made up of systems engineers, operations personnel and seismic engineers with recognized expertise in the subject.
(b) *Selection of the RLE* (Section 6.2.1).
(c) *Plant familiarization and data collection* (Section 2.1.2 and 'as is' state of SSCs).
(d) *Selection of success path(s) and of selected SSCs* (selected by the systems engineers with input from operations personnel and seismic capability engineers[12]).
(e) *Determination of the seismic response of selected SSCs for input to capacity calculations* — seismic analyses should be a best estimate or be median centred.
(f) *Systems walkdown and seismic capability walkdown* — are essential elements of the SMA and SPSA methodologies.
(g) *The screening of SSCs with seismic capacities above the RLE* could be performed using the screening criteria established on the basis of earthquake experience data, seismic qualification test data, and past SMAs and SPSAs. Screening criteria should be confirmed to be applicable to the specific plant design and vintage.
(h) *HCLPF calculations for the installation* — HCLPF is one definition of margin used extensively; the seismic capacities of the selected SSCs may be defined as HCLPF capacities; the HCLPF capacity of an SSC is the earthquake motion level at which there is a high confidence (about 95%) of a low (5%) probability of failure; similarly, the plant HCLPF is the earthquake motion level at which there is a high confidence of a low probability of failure of the plant achieving the success state.
(i) *Enhancements* (e.g., evaluation of containment and containment systems).
(j) *Peer review.*
(k) *Documentation*.

The end products of the SMA are many, including plant and SSC HCLPF values, governing SSCs, candidates for upgrading, which will have the most

[12] The term 'selected SSCs' is used in this report to mean those SSCs that have been selected for the evaluation of their seismic capacity using criteria consistent with the regulatory requirements and ultimate objectives of the seismic safety evaluation programme. This is consistent with the terminology used in Ref. [6]. In IAEA technical guidance developed in the 1990s for the seismic safety evaluation of specific nuclear power plants, the term 'safe shutdown equipment list' (SSEL) was also used, as borrowed from earlier usage in the USA. However, as the SSCs cover more than just 'equipment' and the goals of the programme may exceed 'safe shutdown', the term 'selected SSCs' is preferred.

significant effect on the overall plant safety, etc. These end products contribute to decision making in terms of long term operation and other considerations.

6.3.2. Seismic probabilistic safety assessment

The SPSA methodology has evolved over the past three decades following the development of PSA methodologies for internal events. The SPSA methodology comprises a number of steps. In general, an SPSA includes:

(a) *Selection of the assessment team* — the assessment team should be comprised of members with expertise in seismic hazard analysis, members familiar with the internal events PSA (systems engineers, operations engineers and others involved in the development and exercising of the internal events PSA model), experts in the area of fragility function development, and the engineering staff of the operating organization.
(b) *Probabilistic seismic hazard assessment* (Section 6.2.2).
(c) *Plant familiarization and data collection* (Section 2.1.2 and the ‘as is’ state of SSCs).
(d) *Systems analysis and accident sequence analysis leading to event tree and fault tree modelling and identification of selected SSCs for detailed evaluation* — the internal events PSA serves as the starting point. Modifications to the event trees (initiating events) and fault trees (safety system models) are introduced recognizing the phenomena of the earthquake, such as the common cause nature of the event and potential failure of passive SSCs.
(e) *Determination of the seismic response of structures for input to fragility calculations* — median or best estimate responses are calculated as probability distributions conditional on the earthquake ground motion occurring.
(f) *Walkdowns for seismic capability.*
(g) *Fragility calculations for the selected SSCs* — fragility functions are calculated for SSCs in the fault trees (and the initiating events), relating failure to perform a required function with an earthquake motion parameter. Fragility functions are most often described by log–normal probability distributions.
(h) *Human reliability analysis for seismic events* (Section 2.1.9).
(i) *Risk quantification for the installation* — calculation of probabilities of failure of all modelled SSCs, metrics such as CDF, importance ranking of SSCs in terms of risk, candidates for upgrading based on their impact on risk, etc.
(j) *Enhancements* (Section 6.3.1).

(k) *Peer review.*
(l) *Documentation*.

The end products of the SPSA are numerous, and include: an understanding of accident behaviour, the most likely scenarios for accidents induced by earthquakes, identification of dominant risk contributors, a list of SSC fragilities, identification of the range of earthquakes contributing most significantly to risk, a comparison of risk from other accident scenarios and identification of potential upgrading of SSCs. A select subset or all of these end products form one basis for decision making.

6.3.3. Foundation stability assessment

Foundation stability assessment is performed as part of the SMA or SPSA. The assessments proceed by defining the failure modes to be considered, such as excessive deformations, or relative motion between buildings inducing impact or pounding of structures and/or excessive deformations of distribution systems anchored or supported in each of the buildings. The focus of the assessments is on the ability of the SSCs to perform their required functions. These assessments may be probabilistic.

6.4. UPGRADES

Upgrades to selected SSCs important to safety and to SSCs not important to safety may be required based on the evaluations described in Section 5 or the results of the SMA or the SPSA discussed above.

These selected SSCs might not meet the acceptance criteria for the felt earthquake or for the evaluation described in Section 6.3. In the former case, inspections, tests or analyses may indicate that the SSC of interest exceeded the values allowable by the code or the qualification criteria, even though it performed its required function during and/or after the felt earthquake. For this case, the decision may be made to upgrade the SSC to meet the code requirements when subjected to a postulated repeat of the felt earthquake. For the latter case, upgrades of the selected SSCs may significantly increase the plant HCLPF value or reduce the seismic CDF. Hence, the decision may be made that such upgrades are prudent and cost effective. In general, the prioritization of upgrades is based on a cost–benefit analysis where the utility metric is a risk measure or a physical measure such as the plant HCLPF value.

The seismic design basis for the upgrades is proposed by the operator and approved by the regulatory body. A redefinition of the seismic design basis ground motion is likely only in the consideration of Action Level 8 behaviours, i.e. a felt earthquake level greater than the SL-2 level and failure or significant damage to SSCs important to safety. For upgrade design/qualification, the ground motion specification is selected on the basis of the end objective, i.e. the achieved SSC HCLPF value or the probability of failure.

6.5. MODIFICATION OF THE INSPECTION/SURVEILLANCE TEST PROGRAMME

The results of the extensive evaluations performed after the felt earthquake will provide an insight as to which SSCs need to be subjected to modified inspection and surveillance schedules.

7. MANAGEMENT SYSTEM

7.1. APPLICATION OF THE MANAGEMENT SYSTEM

A management system applicable to all organizations involved in pre-earthquake planning and post-earthquake actions should be established and implemented before the start of the pre-earthquake planning and post-earthquake actions programme [31, 35]. The management system should cover all processes and activities of this programme, in particular, those relating to data collection and data processing, field and laboratory investigations, analyses, evaluations and tests that are within the scope of the present report. It should also cover those processes and activities corresponding to the plant upgrading phase of the programme.

For the team implementing the programme of pre-earthquake planning and post-earthquake actions, owing to the variety of evaluations and tests to be carried out and the need to use engineering judgement, technical procedures that are specific to the project are developed to facilitate the execution and verification of these tasks.

Similarly, a peer review of the implementation of the investigation, analysis and test methodology needs to be performed. In particular, it is advisable that the peer review assess the pre-earthquake planning and post-earthquake actions against the recommendations of the present report and current international good practices used for these evaluations and tests.

The peer review is conducted by experts in the areas of systems engineering, operations (including specialists on fire prevention and protection, and on external and internal flooding), earthquake engineering and relay circuits (if a relay review is performed). Finally, the peer reviews are properly documented.

7.2. DOCUMENTATION AND RECORDS

Documentation of actions taken, including the responsible party and justification, should be performed in a timely manner. Such documentation is essential for reconstructing the events following the earthquake and it serves many purposes, such as lessons learned with future impact on the emergency operating procedures, timing and substance of communication to the governing bodies, the public and other stakeholders.

Therefore, an important component of the management system is the definition of the documentation and records to be developed during the implementation of the different steps of the programme, the execution of the pre-earthquake planning and post-earthquake actions, and of the final report to be produced as a result. Detailed documentation is retained for review and future application. It is advisable that the specific plant procedures prepared for dealing with response actions required before, during and after an earthquake be included as part of the final report.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. NS-R-3, IAEA, Vienna (2003).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-9, IAEA, Vienna (2002).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, IAEA, Vienna (2009).

[6] US NUCLEAR REGULATORY COMMISSION, Pre-earthquake Planning and Immediate Nuclear Power Plant Operator Post-earthquake Actions, Regulatory Guide 1.166, US Govt Printing Office, Washington, DC (1997).
[7] US NUCLEAR REGULATORY COMMISSION, Restart of a Nuclear Power Plant Shut Down by a Seismic Event, Regulatory Guide 1.167, US Govt Printing Office, Washington, DC (1997).
[8] ELECTRIC POWER RESEARCH INSTITUTE, Guidelines for Nuclear Plant Response to an Earthquake, Rep. EPRI-NP-6695, EPRI, Palo Alto, CA (1989).
[9] AMERICAN NUCLEAR SOCIETY, Nuclear Plant Response to an Earthquake, Rep. ANSI/ANS-2.23-2002, ANS, La Grange Park, IL (2002).
[10] AMERICAN NUCLEAR SOCIETY, Criteria for the Handling and Initial Evaluation of Records from Nuclear Power Plant Seismic Instrumentation, Rep. ANSI/ANS-2.10-2003, ANS, La Grange Park, IL (2003).
[11] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-R-3, IAEA, Vienna (2006).
[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Plant Life Management for Long Term Operation of Light Water Reactors, Principles and Guidelines, Technical Reports Series No. 448, IAEA, Vienna (2006).
[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Aspects of Long Term Operation of Water Moderated Reactors, Recommendations on the Scope and Content of Programmes for Safe Long Term Operation, IAEA-EBP-SALTO, IAEA, Vienna (2007).
[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.12, IAEA, Vienna (2009).
[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Safe Long Term Operation of Nuclear Power Plants, Safety Reports Series No. 57, IAEA, Vienna (2008).
[16] JAPAN ELECTRIC ASSOCIATION, Technical Guidelines for Seismic Design of Nuclear Power Plants, Rep. JEAG 4601-1970, JEA, Tokyo (1970) (in Japanese).
[17] NUCLEAR SAFETY COMMISSION OF JAPAN, Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, NSC, Tokyo (1981).
[18] JAPAN ELECTRIC ASSOCIATION, Technical Guidelines for Aseismic Design of Nuclear Power Plants, Rep. JEAG 4601-1987, JEA, Tokyo (1987) (in Japanese).
[19] NUCLEAR SAFETY COMMISSION OF JAPAN, Regulatory Guide for Reviewing Seismic Design of Nuclear Reactor Facilities, NSC, Tokyo (2006).
[20] CHEN, C., "Experience data and analytical results of high frequency earthquakes", in Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping (Proc. 4th Symp. Orlando, 1992), North Carolina State Univ., Raleigh, NC (1992).
[21] ELECTRIC POWER RESEARCH INSTITUTE, The 1986 Leroy, Ohio Earthquake Performance of Power and Industrial Facilities, Rep. EPRI-NP-6558, EPRI, Palo Alto, CA (1989).
[22] WHORTON, R.B., High frequency, high amplitude and low energy earthquake study at V.C. Summer nuclear station, Nucl. Eng. Des. **107** (1988) 109–125.

[23] US NUCLEAR REGULATORY COMMISSION, Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance, Regulatory Guide 1.201, Rev. 1, US Govt Printing Office, Washington, DC (2006).
[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.2, IAEA, Vienna (2000).
[25] INTERNATIONAL ASSOCIATION OF FIRE AND RESCUE SERVICE (CTIF), INTERNATIONAL ATOMIC ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, WORLD HEALTH ORGANIZATION, Manual for First Responders to a Radiological Emergency, Emergency Preparedness and Response, IAEA, Vienna (2006).
[26] INTERNATIONAL ATOMIC ENERGY AGENCY, Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007).
[27] AMERICAN SOCIETY OF CIVIL ENGINEERS, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities, ASCE, Reston, VA (2005).
[28] ELECTRIC POWER RESEARCH INSTITUTE, Generic Seismic Ruggedness of Power Plant Equipment, Rep. EPRI-NP-5223, EPRI, Palo Alto, CA (1987).
[29] ELECTRIC POWER RESEARCH INSTITUTE, Seismic Ruggedness of Relays, Rep. EPRI-NP-7147, EPRI, Palo Alto, CA (1991).
[30] ELECTRIC POWER RESEARCH INSTITUTE, Summary of the Seismic Adequacy of Twenty Classes of Equipment Required for the Safe Shutdown of Nuclear Plants, Rep. EPRI-NP-7149-D, EPRI, Palo Alto, CA (1991).
[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-3.6, IAEA, Vienna (2004).
[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Flood Hazard for Nuclear Power Plants on Coastal and River Sites, IAEA Safety Standards Series No. NS-G-3.5, IAEA, Vienna (2003).[13]
[33] ELECTRIC POWER RESEARCH INSTITUTE, A Methodology for Assessment of Nuclear Power Plant Seismic Margin, Rep. EPRI-NP-6041-SL, Rev. 1, EPRI, Palo Alto, CA (1991).
[34] AMERICAN NUCLEAR SOCIETY, External Events in PRA Methodology, Rep. ANSI/ANS-58.21-2007, ANS, La Grange Park, IL (2007).
[35] INTERNATIONAL ATOMIC ENERGY AGENCY, Periodic Safety Review of Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.10, IAEA, Vienna (2003).

[13] Under revision.

Annex I

EXAMPLES OF DAMAGE ANALYSIS AND EQUIPMENT INSPECTION

The equipment to be inspected should be identified, selected and analysed on the basis of the following general procedure:

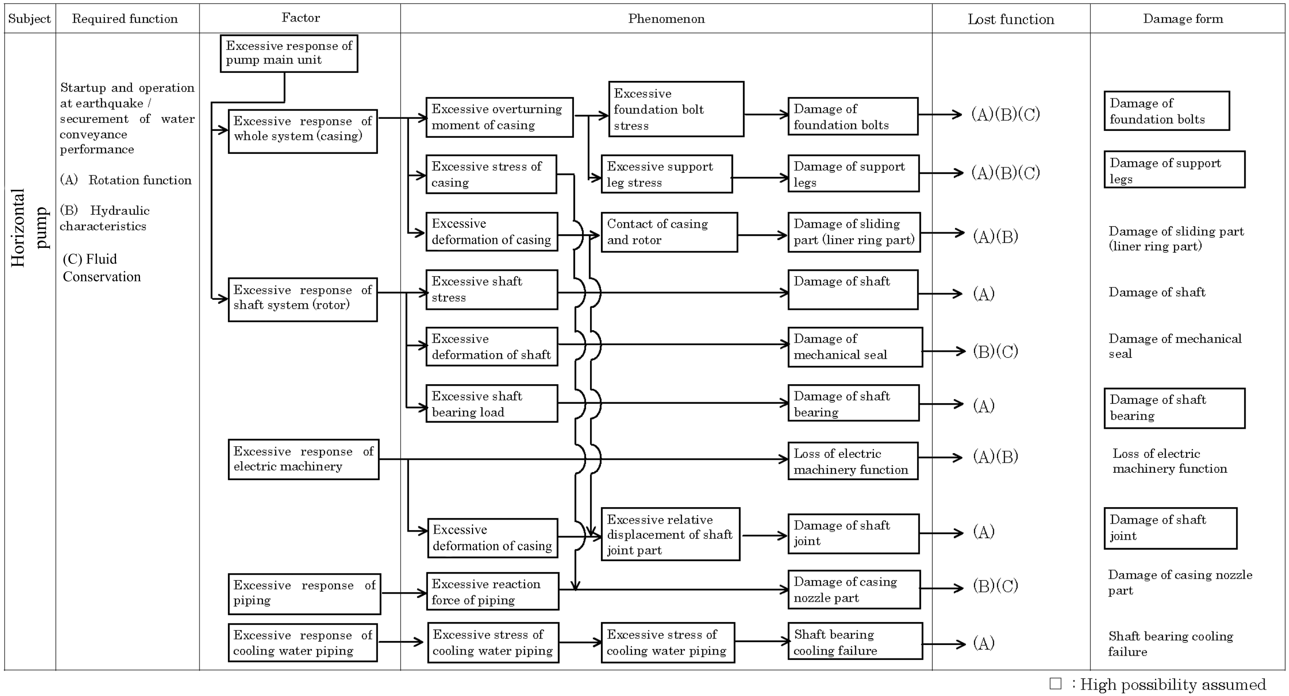
(1) The plant equipment is classified according to its type and similarity of functions.
(2) On the basis of findings and previous experience, the equipment classified by type should be grouped according to:
  (i) Its location in areas or structures where potential damage can occur; and
  (ii) The effects of such damage on equipment functions, as indicated in the examples provided in Tables I–1 to I–4.
(3) A summary on the effects of damage is established for each of the selected items and for each equipment type, in accordance with the criteria indicated in Section 5.3.1.1.

As examples, Tables I–1 to I–4 provide, for three types of equipment (i.e. horizontal pumps, valves, and heat exchangers and piping supports), the following information:

— Type of equipment, for example, valve;
— Required function;
— Cause of malfunction;
— Phenomenon;
— Lost function;
— Form of damage.

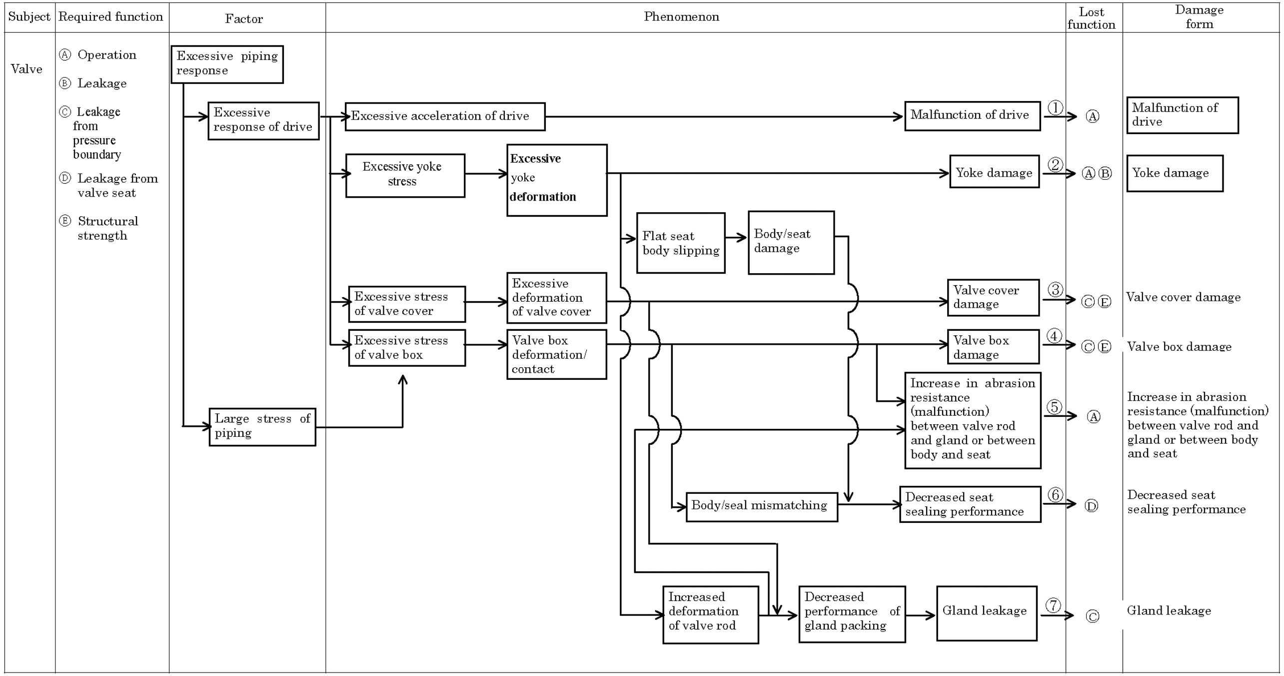
These examples were provided by TEPCO on the basis of the experience gained during the integrity evaluation process conducted at the Kashiwazaki-Kariwa nuclear power plant.

TABLE I–1. HORIZONTAL PUMP: EXAMPLE OF ANALYSIS RESULTS OF SEISMIC DAMAGE MODES *(from Ref. [I–1])*



Source: Special Committee on Earthquake-resistant Design of Nuclear Power Plant, Japan Electric Association
"Study report on inspection of methods of evaluating function maintenance of equipment against horizontal and vertical earthquake motion" (Vol. 36, March 2001)

TABLE I–2. VALVES: EXAMPLE OF ANALYSIS RESULTS OF SEISMIC DAMAGE MODES
(from Ref. [I–1])



□ : High possibility assumed

Source: Special Committee on Earthquake-resistant Design of Nuclear Power Plant, Japan Electric Association "Study report on inspection of methods of evaluating function maintenance of equipment against horizontal and vertical earthquake motion" (Vol. 36, March 2001)

TABLE I–3. HEAT EXCHANGERS: EXAMPLE OF ANALYSIS RESULTS OF SEISMIC DAMAGE MODES *(from Ref. [I–1])*

Subiect	Required function	Factor	Phenomenon		Lost function	Damage form
Heat exchanger	(A) Securing of heat transfer performance (B) Boundary Maintenance (C) Supporting of apparatuses	Excessive response of main body	Excessive stress of main body	Damage of main body	(B)	Damage of main body
			Excessive stress of flange portion	Damage of flange portion	(B)	Damage of flange portion
			Excessive stress of exchanger tube	Damage of exchanger tube	(A)(B)	Damage of exchanger tube
			Excessive stress of supporting legs	Damage of supporting legs	(C)	Damage of supporting legs
			Excessive stress of foundation bolts (or mounting bolts)	Damage of foundation bolts	(C)	Damage of foundation bolts
		Excessive response of piping	**Excessive stress of nozzle**	Damage of nozzle	(B)	Damage of nozzle

□ : High possibility assumed

TABLE I–4. SUPPORT STRUCTURES: EXAMPLE OF ANALYSIS RESULTS OF SEISMIC DAMAGE MODE SUPPORT *(from Ref. [I–1])*

Subject	Required function	Factors	Phenomenon	Lost functions	Damage mode
Support structures	(A) Keep of device support functions	Excessive response of the piping			
		Considerable reactive force of added hardware	Pulling-off of bolts/concrete fixtures, plate deformation, cracks in concrete		①Deformation of plate ②Pulling-off of fixtures
		Considerable reactive force of embedded hardware	Pulling-off of stand/concrete fixtures, plate deformation, cracks in concrete		③Cracks in concrete
		Excessive reactive force of the lug	Considerable stress on welded point and main unit → Damage (deformation, cracks)		④Deformation or cracks in lug
		Considerable reactive force of bridging steel parts	Considerable stress on welded point and main unit → Damage (deformation, cracks)		⑤Deformation or cracks in bridging structures
		Considerable reactive force of mechanical snubbers	Damage (rod deformation, deformation of internal components, damage of ball bearings, and break of pin)		⑥Deformation of mechanical snubber rod ⑦Damage of ball bearing or pin of mechanical snubber
		Considerable reactive force of oil snubbers	Damage (rod deformation, deformation of internal components, damage of ball bearings, and break of pin)		⑧Damage of ball screws ⑨Oil leak
		Large hanger displacement	Rod displacement, deformation of case		⑩Deformation of hanger rod ⑪Deformation of rod of rod restraint
		Considerable reactive force of the rod restraint	Damage (deformation, damage of ball bearing, break of pin)		⑫Rod restraint-damage of ball bearing or pin
		Considerable reactive force of pipe grip	Displacement, damage (cut of wire)		⑬Cut of pipe grip wire
		Considerable reactive force of U-bolts	Considerable stress on U-bolt → Damage (cut, elongation)		⑭U-bolts- cut or elongation
			Considerable stress on structural materials → Damage (deformation, cracks in welded points)		⑮U-bolts-deformation of structural material or crack in welded points
		Considerable reactive force of pipe clamps	Clamp dislocation, considerable stress on clamp body/bolts → Damage (deformation, elongation)		⑯Pipe clamp-deformation of structural material or crack in welded points
		Considerable reactive force of constrained plate	Considerable stress on constrained plate → Damage (deformation, elongation, cut)		⑰Elongation or cut of constrained plate

□ : High possibility assumed

REFERENCE TO ANNEX I

[I–1] SPECIAL COMMITTEE ON EARTHQUAKE RESISTANT DESIGN OF NUCLEAR POWER PLANT, Study Report on Inspection of Methods of Evaluating Function Maintenance of Equipment against Horizontal and Vertical Earthquake Motion, Vol. 36, Tokyo Electric Power Company, Tokyo (2001).

Annex II

LESSONS LEARNED FROM RECENT EVENTS

Recent strong events which exceeded the seismic design basis were experienced at some nuclear power plants and they are illustrated in the following tables:

Table II–1. Kashiwazaki-Kariwa (Japan) nuclear power plant (2007);

Table II–2. Shika (Japan) nuclear power plant (2007);

Table II–3. Onagawa (Japan) nuclear power plant (2005);

Table II–4. Metsamor (Armenia) nuclear power plant (1988).

Note: In Tables II–1 to II–4, Gal (after Galileo) is a unit of acceleration, equal to 1 cm/s^2.

TABLE II–1. KASHIWAZAKI-KARIWA (JAPAN) NUCLEAR POWER PLANT (2007)

II–1.1. SITE AND UNIT INFORMATION

II–1.1.1 Site Data (Name of the Site, Country, Operating Organization)

Kashiwazaki-Kariwa Nuclear Power Station,
Japan, TEPCO

II–1.1.2. Unit Data (Unit Number, Net Capacity (MW(e)), Reactor Type, Reactor Supplier, Date of Commercial Operation, Foundation Level and Type)

Unit #1: 1100 MW(e), BWR-5, Toshiba, 1985, Ground level –45 m, on the 450 m/s bedrock

Unit #2: 1100 MW(e), BWR-5, Toshiba, 1990, Ground level –44 m, on the 450 m/s bedrock

Unit #3: 1100 MW(e), BWR-5, Toshiba, 1993, Ground level –43 m, on the 450 m/s bedrock

Unit #4: 1100 MW(e), BWR-5, Hitachi, 1994, Ground level –43 m, on the 450 m/s bedrock

Unit #5: 1100 MW(e), BWR-5, Hitachi, 1990, Ground level –36 m, on the 450 m/s bedrock

Unit #6: 1356 MW(e), ABWR, Toshiba, Hitachi and GE, 1996, Ground level –25.7 m, on the 450 m/s bedrock

Unit #7: 1356 MW(e), ABWR, Hitachi, Toshiba and GE, 1997, Ground level –25.7 m, on the 450 m/s bedrock

II–1.2. SEISMIC DESIGN DATA

II–1.2.1. Design Basis Earthquake(s) Data

	DBE					
	S1			S2		
	Outcrop Bedrock Surface		Reactor Building Base Mat	Outcrop Bedrock Surface		Reactor Building Base Mat
	Maximum Acceleration	Depth*	Maximum Acceleration	Maximum Acceleration	Depth*	Maximum Acceleration
Unit #1	300 Gal	G.L. –289 m	-	450 Gal	G.L. –289 m	189 Gal
Unit #2	300 Gal	G.L. –255 m	137 Gal	450 Gal	G.L. –255 m	167 Gal
Unit #3	300 Gal	G.L. –290 m	151 Gal	450 Gal	G.L. –290 m	193 Gal
Unit #4	300 Gal	G.L. –290 m	153 Gal	450 Gal	G.L. –290 m	194 Gal
Unit #5	300 Gal	G.L. –146 m	206 Gal	450 Gal	G.L. –146 m	254 Gal
Unit #6	300 Gal	G.L. –167 m	195 Gal	450 Gal	G.L. –167 m	263 Gal
Unit #7	300 Gal	G.L. –167 m	195 Gal	450 Gal	G.L. –167 m	263 Gal

TABLE II–1. KASHIWAZAKI-KARIWA (JAPAN) NUCLEAR POWER PLANT (2007) (cont.)

II–1.2.2. Seismic Instrumentation

II–1.2.2.1. Seismic Instrumentation for SCRAM

Automatic seismic SCRAM system is installed.

Seismic SCRAM sensors: 12/unit (one out of two, twice, horizontal and vertical)

Set values for seismic SCRAM: 120 Gal, 185 Gal (horizontal), 100 Gal (vertical)

Unit	Location*	Number of Instruments		Set Values for seismic SCRAM
		Horizontal	Vertical	
Unit #1	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Second Floor (G.L. +7.8 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)
Unit #2	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Second Floor (G.L. +7.8 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)
Unit #3	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Second Floor (G.L. +7.8 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)
Unit #4	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Second Floor (G.L. +7.8 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)
Unit #5	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Third Floor (G.L. +15.8 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)
Unit #6	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Third Floor (G.L. +11.5 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)
Unit #7	Reactor Building Base Mat	4	0	185 Gal (horizontal)
	R/B, the Third Floor (G.L. +11.5 m)	4	4	120 Gal (horizontal), 100 Gal (vertical)

* G.L.: Ground Level; R/B: Reactor Building.

TABLE II–1. KASHIWAZAKI-KARIWA (JAPAN) NUCLEAR POWER PLANT (2007) (cont.)

II–1.2.2.2. Seismic Instrumentation for Recording

	Location*				Number of Recorders (Operable)	
					Horizontal	Vertical
In-field	Unit #1–#4 side			G.L. +0.0 m	4	2
				G.L. –7.7 m	1	0
				G.L. –45.0 m	2	1
				G.L. –127.0 m	2	1
				G.L. –225 m	2	1
	Unit #5–#7 side			G.L. +0.0 m	2	1
				G.L. –1.4 m	3	2
				G.L. –2.7 m	3	2
				G.L. –13.6 m	4	3
				G.L. –26.0 m	2	1
				G.L. –36.0 m	2	1
				G.L. –37.0 m	5	2
				G.L. –76.0 m	5	1
				G.L. –112.0 m	5	3
				G.L. –192.0 m	2	1
				G.L. –312.0 m	2	1
In-structure	Unit #1	R/B	Roof Truss	G.L. +26.6 m	2	3
			the Third Floor	G.L. +13.0 m	2	1
			the Second Floor	G.L. +7.8 m	2	1
			the First Floor	G.L. +0.3 m	2	1
			Base Mat	G.L. –37.5 m	2	1
		T/B	Roof Truss	G.L. +18.8 m	2	1
			the First Floor	G.L. +0.3 m	2	1
			Base Mat	G.L. –21.7 m	2	1
	Unit #2	R/B	the Second Floor	G.L. +7.8 m	2	1
			Base Mat	G.L. –37.5 m	2	1
		T/B	the First Floor	G.L. +0.3 m	4	2
			Base Mat	G.L. –21.3 m	2	1
	Unit #3	R/B	the Second Floor	G.L. +7.8 m	2	1
			Base Mat	G.L. –37.5 m	2	1
		T/B	the First Floor	G.L. +0.3 m	4	2
			Base Mat	G.L. –21.3 m	2	1
	Unit #4	R/B	the Second Floor	G.L. +7.8 m	2	1
			Base Mat	G.L. –37.5 m	2	1
		T/B	the First Floor	G.L. +0.3 m	4	2
			Base Mat	G.L. –21.3 m	2	1
	Unit #5	R/B	Roof Truss	G.L. +35.2 m	2	2
			the Fourth Floor	G.L. +21.0 m	2	1
			the Third Floor	G.L. +15.8 m	2	1
			the First Floor	G.L. +0.3 m	1	1
			Base Mat	G.L. –29.5 m	4	6
		T/B	the Second Floor	G.L. +10.1 m	4	3
			Base Mat	G.L. –14.75 m	2	1
	Unit #6	R/B	Top of the stack	G.L. +73.0 m	2	0
			Roof Truss	G.L. +33.7 m	4	4
			Stack	G.L. +31.5 m	2	0
			the Fourth Floor	G.L. +19.7 m	4	2
			the Third Floor	G.L. +11.5 m	2	1
			the First Floor	G.L. +0.3 m	2	0
			Base Mat	G.L. –20.2 m	8	5
		T/B	the Second Floor	G.L. +8.4 m	4	2
			Base Mat	G.L. –17.1 m	2	1
	Unit #7	R/B	the Third Floor	G.L. +11.5 m	2	1
			Base Mat	G.L. –20.2 m	2	1
		T/B	the Second Floor	G.L. +8.4 m	4	2
			Base Mat	G.L. –7.1 m	2	1

* G.L.: Ground Level; R/B: Reactor Building; T/B: Turbine Building.

TABLE II–1. KASHIWAZAKI-KARIWA (JAPAN) NUCLEAR POWER PLANT (2007) (cont.)

II–1.2.3. Design Standard

Unit 1: Refer to JEAG 4601-1970 [II–1].
Units 2–7: Refer to NSC Regulatory Guide [II–2] and JEAG 4601-1987 [II–3].

II–1.3. EARTHQUAKE EVENT DATA

II–1.3.1. General Information (Name of the Event, Date and Time, Magnitude, Geological Coordinates of the Epicentre, Focal Depth, Epicentral Intensity, Intensity at the Site)

Niigata-ken Chuetsu-Oki Earthquake (NCOE), 10:13 a.m., 16 July 2007
M_{JMA} = 6.8, Epicentre: 16 km north from the site, Focal Depth: 17 km,
Epicentral intensity: JMA intensity at the site: 7

II–1.3.2. Observed Data at the Site (Accelerations, Time Historic Records, Spectra, etc., if available)

Observation Point			Observed Peak Acceleration (Gal)			Design Value (Gal)	
			NS[a]	EW[b]	UD[c]	NS	EW
Unit #1	R/B	B5F (Base Mat)	311	680	408	274	273
Unit #2	R/B	B5F (Base Mat)	304	606	282	167	167
Unit #3	R/B	B5F (Base Mat)	308	384	311	192	193
Unit #4	R/B	B5F (Base Mat)	310	492	337	193	194
Unit #5	R/B	B5F (Base Mat)	277	442	205	249	254
Unit #6	R/B	B5F (Base Mat)	271	322	488	263	263
Unit #7	R/B	B5F (Base Mat)	267	356	355	263	263

[a] NS: north–south (component).
[b] EW: east–west (component).
[c] UD: up–down (component).
Time history records are shown in Figs II–1.1 to II–1.7.
Spectra of observation records are shown in Figs II–1.8 to II–1.14.

II–1.4. EFFECTS ON NUCLEAR POWER PLANTS

II–1.4.1. Operational States of Nuclear Power Plants (State of the Units, Shutdown Information)

Unit #1: Inspection Period, Signal of Automatic Shutdown by Seismic SCRAM
Unit #2: Startup after a detailed walkdown (D/W) Inspection, Signal of Automatic Shutdown by Seismic SCRAM
Unit #3: Full Power Operation, Automatic Shutdown by Seismic SCRAM
Unit #4: Full Power Operation, Automatic Shutdown by Seismic SCRAM
Unit #5: Inspection Period, Signal of Automatic Shutdown by Seismic SCRAM
Unit #6: Inspection Period, Signal of Automatic Shutdown by Seismic SCRAM
Unit #7: Full Power Operation, Automatic Shutdown by Seismic SCRAM

TABLE II–1. KASHIWAZAKI-KARIWA (JAPAN) NUCLEAR POWER PLANT (2007) (cont.)

II–1.4.2. Fundamental Safety Functions (Control of Reactivity, Cooling, Control of the Release of Radioactive Material)

Control of Reactivity: All Control Rods inserted successfully
— Neutron flux securely confirmed.
Cooling: Shift to the Shutdown Cooling Mode successful
— Reactor temperature, pressure, water level, core flow, feedwater flow and main steam flow securely measured.
Control of the Release of Radioactive Material: No release of radioactive material (minor case indicated in Section 4.3)
— D/W Monitor (temperature and pressure), Stack Radiation Monitor and SGTS Radiation Monitor kept functions.

II–1.4.3. Other Effects (Loss of Off-site Power, Seismically Induced Events)

(1) *Safety-related effects*
— None.
(2) *Non-safety-related damage*
Negligible release of radioactive water by the sloshing of Spent Fuel Pool and unexpected path to uncontrolled area:
— Loss of Off-site Power: two lines out of four not available;
— Fire of Unit #3 House Transformer;
— Internal Flooding at Unit #1 Reactor Building (Rupture of Fire Extinguishment Piping);
— Loss of Service Water (including extinguishing of fire) by rupture of piping and Service Water Tank;
— Damage of Emergency Response Centre.

II–1.4.4. Actions for Restart (as of September 2009)

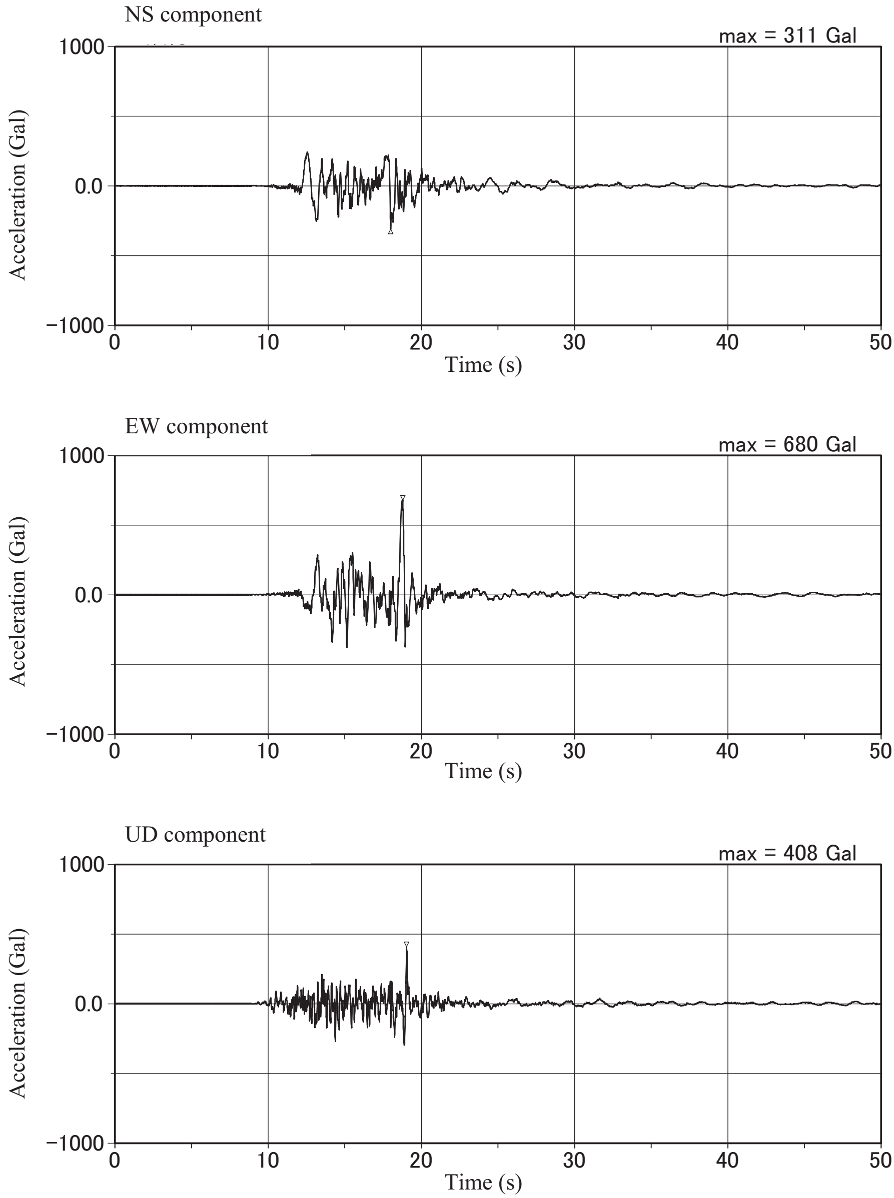
Fully organized inspections, conducted for all facilities including non-seismic safety:
— No significant damage to safety related SSCs.
Seismic response analyses, conducted for all safety related SSCs:
— Response of SSCs under elastically allowable limits (Units #6 and 7).
Evaluation of DBE:
— Re-definitions of DBE and evaluation of safety related SSCs conducted before restart.

Unit	New DBE (Ss)		
	Outcrop bedrock surface		Reactor Building Base mat
	Maximum Acceleration (Gal)	Depth* (m)	Maximum Acceleration (Gal)
Unit #1	2280	G.L. –289	829
Unit #2		G.L. –255	739
Unit #3		G.L. –290	663
Unit #4		G.L. –290	699
Unit #5	1156	G.L. –146	543
Unit #6		G.L. –167	656 (1000)
Unit #7		G.L. –167	642 (1000)

* G.L.: Ground Level
(1000 Gal): horizontal peak acceleration defines motion on R/B base mat for evaluation.

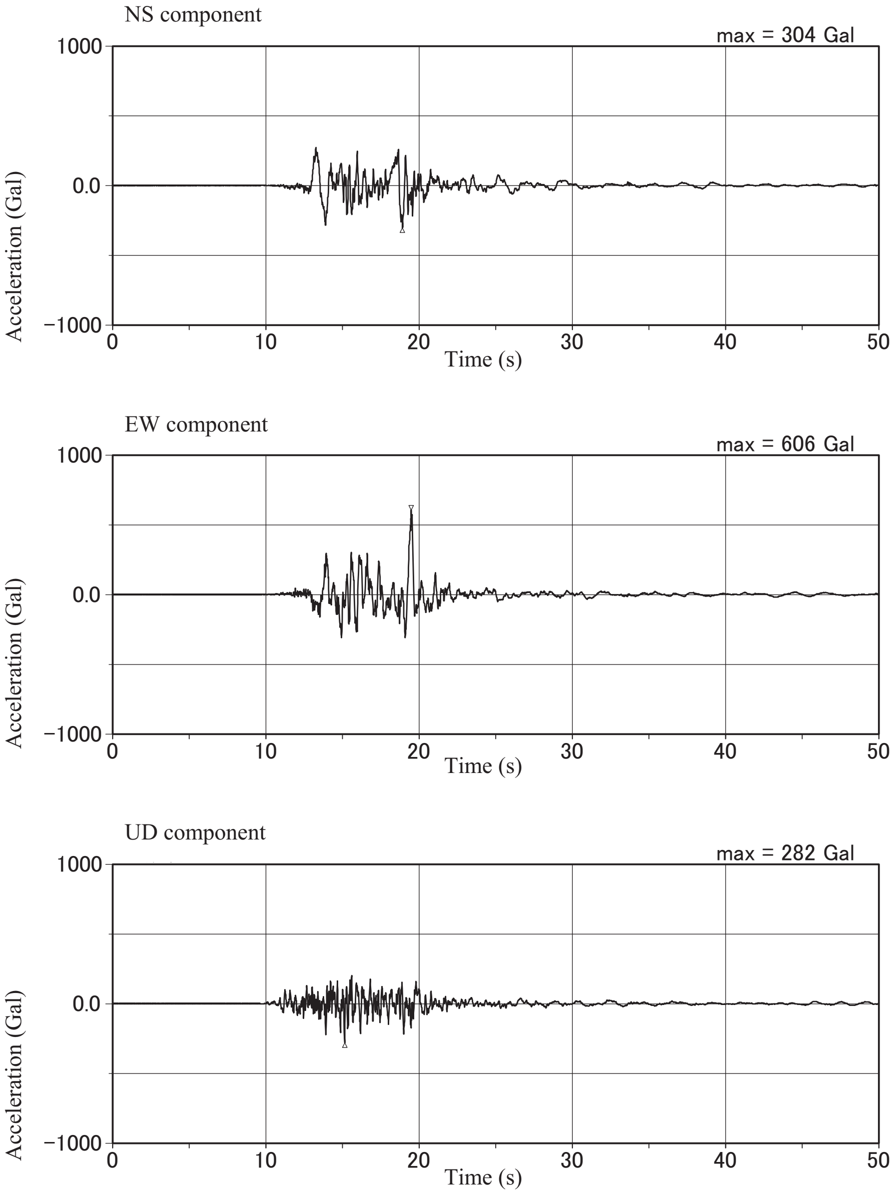
TABLE II–1. KASHIWAZAKI-KARIWA (JAPAN) NUCLEAR POWER PLANT (2007) (cont.)

II–1.4.5. Restart
Unit #7 restarted on 9 May 2009. Unit #6 restarted on 26 August 2009.
II–1.5. OTHER ACTIONS
Engineering Safety Reviews by the IAEA: — First: 6–10 August 2007; — Second: 28 January–1 February 2008; — Third: 1–5 December 2008.



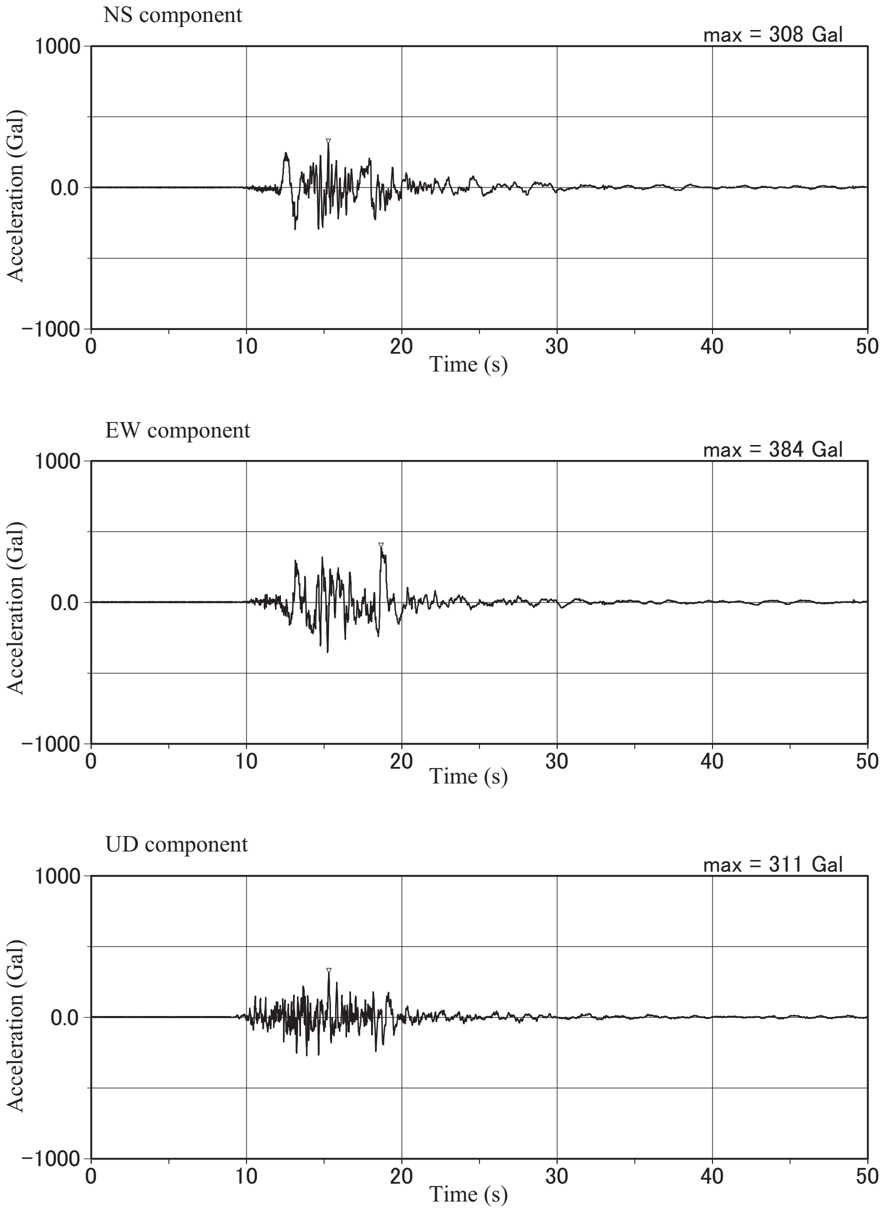
(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.1. Acceleration–time history waveform: Unit 1 reactor building B5F (on foundations).



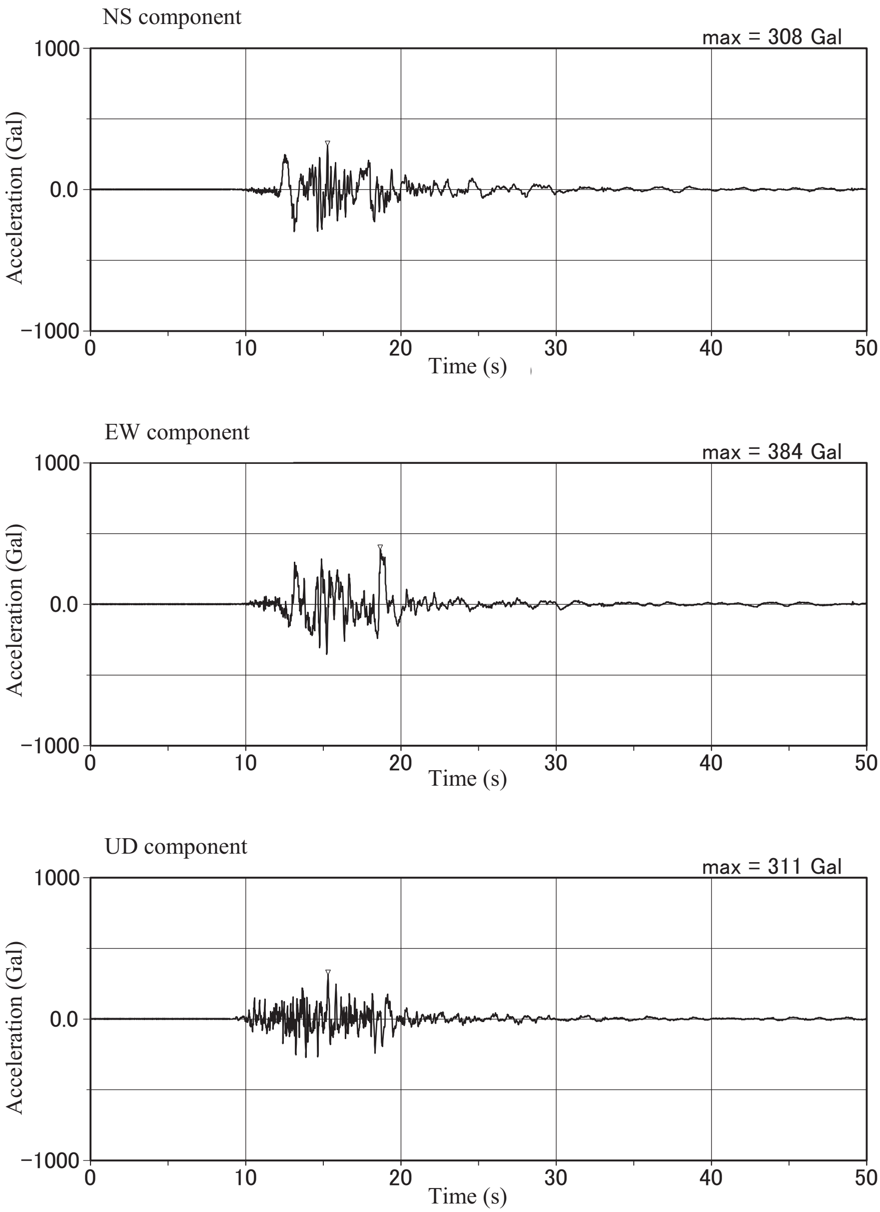
(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.2. Acceleration–time history waveform: Unit 2 reactor building B5F (on foundations).



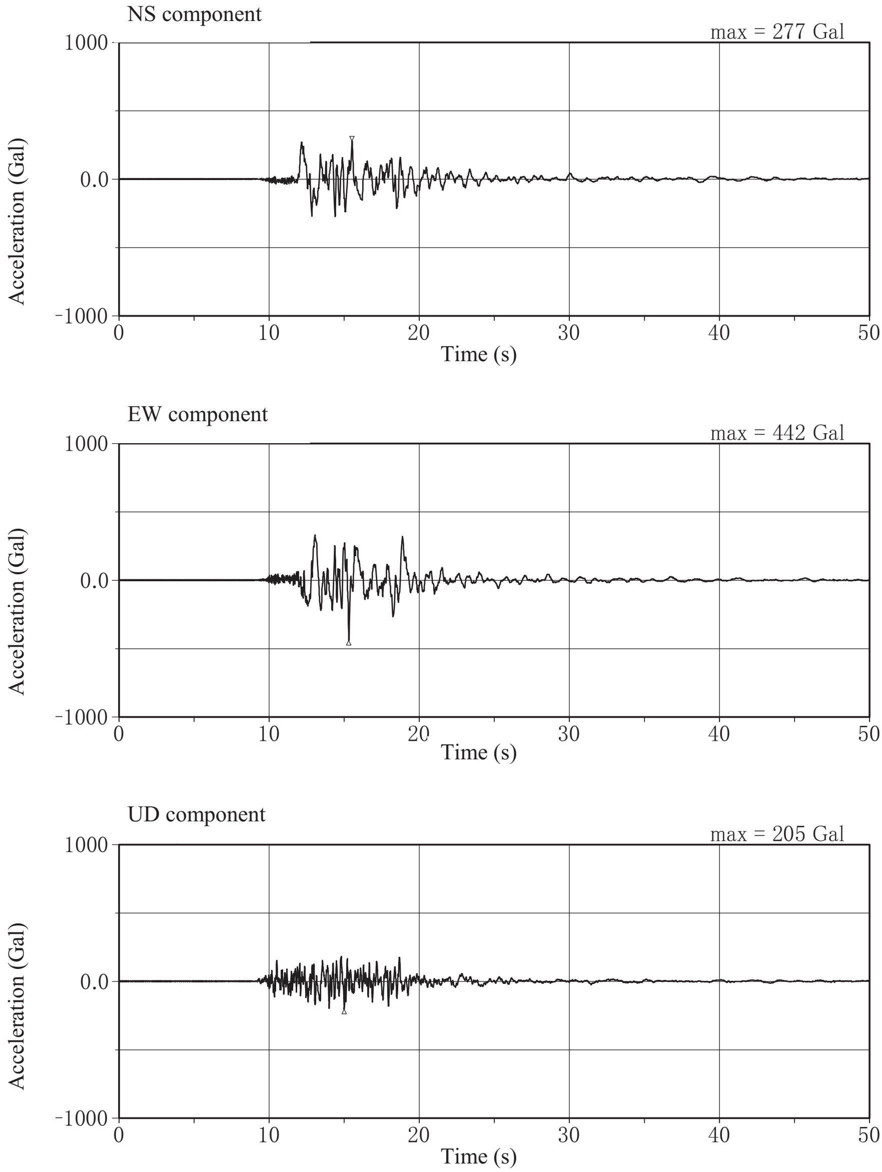
(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.3. Acceleration–time history waveform: Unit 3 reactor building B5F (on foundations).



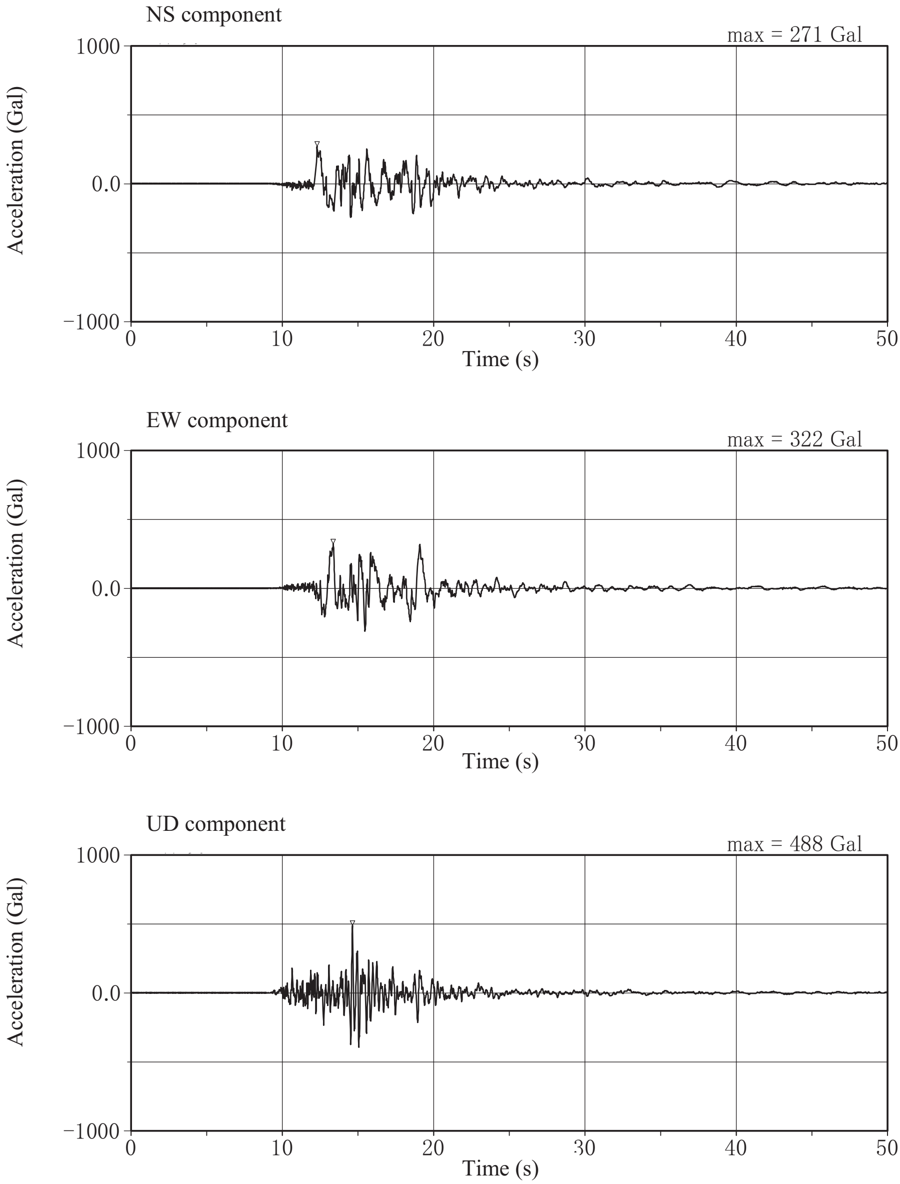
(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.4. Acceleration–time history waveform: Unit 4 reactor building B5F (on foundations).



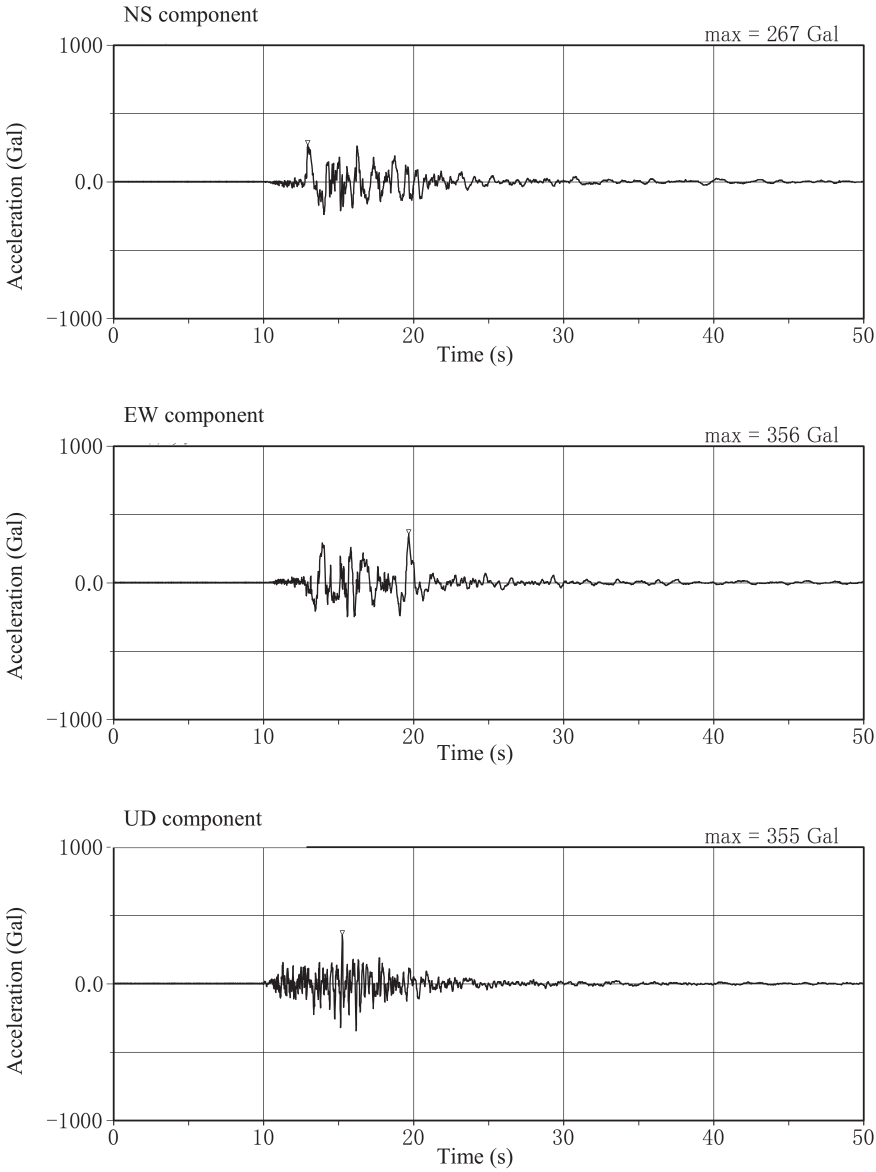
(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.5. Acceleration–time history waveform: Unit 5 reactor building B4F (on foundations).



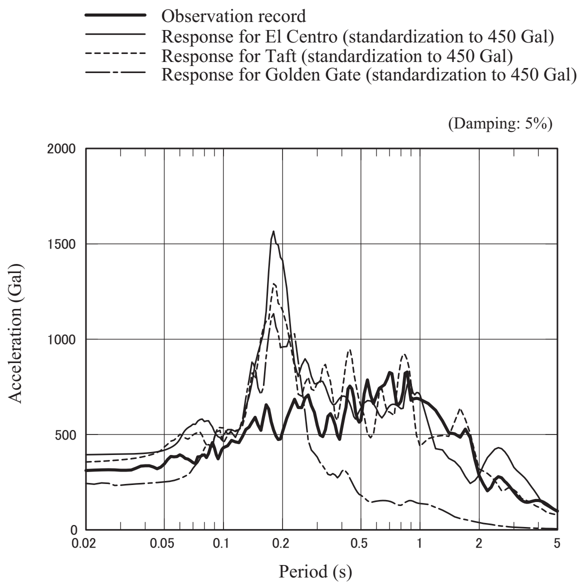
(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.6. Acceleration–time history waveform: Unit 6 reactor building B3F (on foundations).

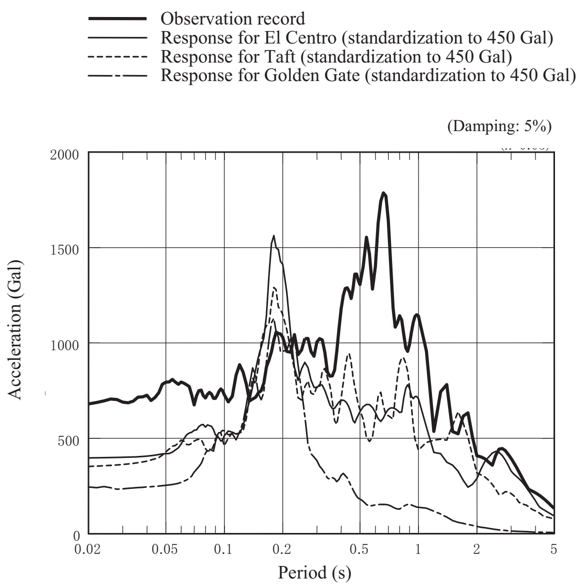


(50 seconds shown from the 20th second to the 70th second of the record)

FIG. II–1.7. Acceleration–time history waveform: Unit 7 reactor building B3F (on foundations).

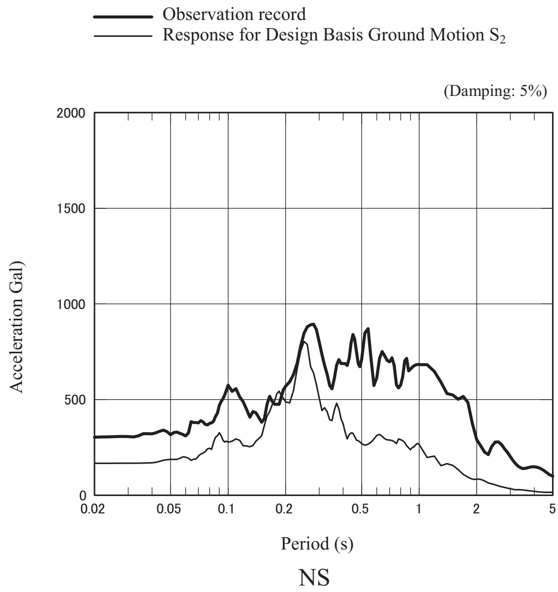


NS



EW

FIG. II–1.8. Acceleration response spectra for basement 5 (on the foundations) of Unit 1 reactor building (observation point 1-R2).



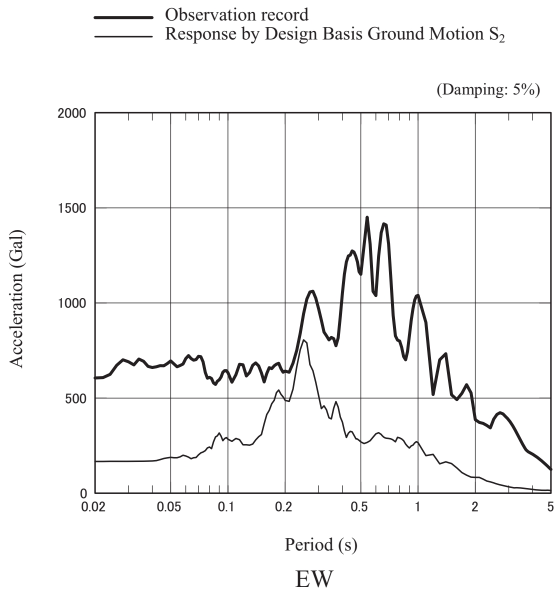
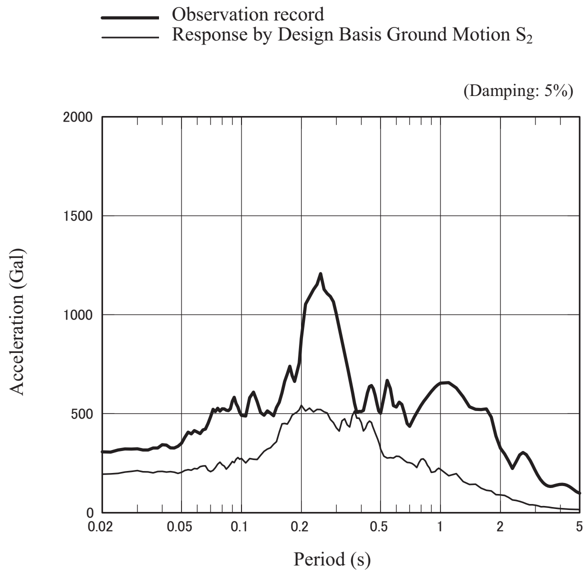
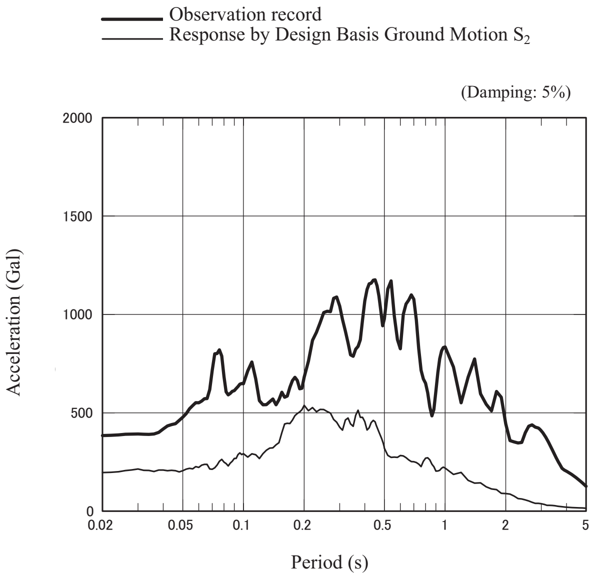


FIG. II–1.9. Acceleration response spectra for basement 5 (on the foundations) of Unit 2 reactor building (observation point 2-R2).

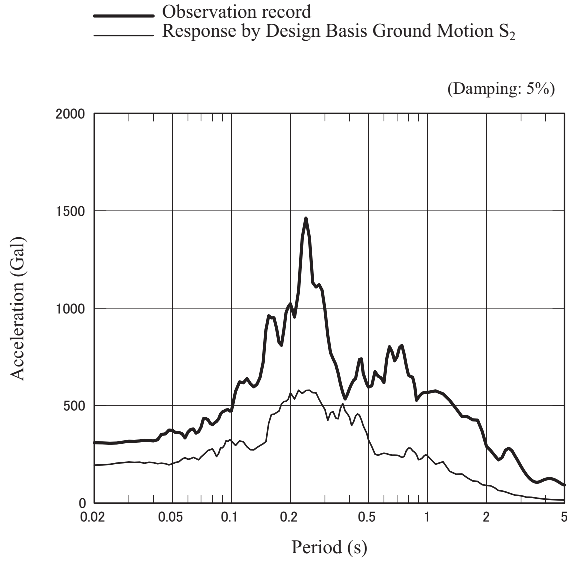


NS

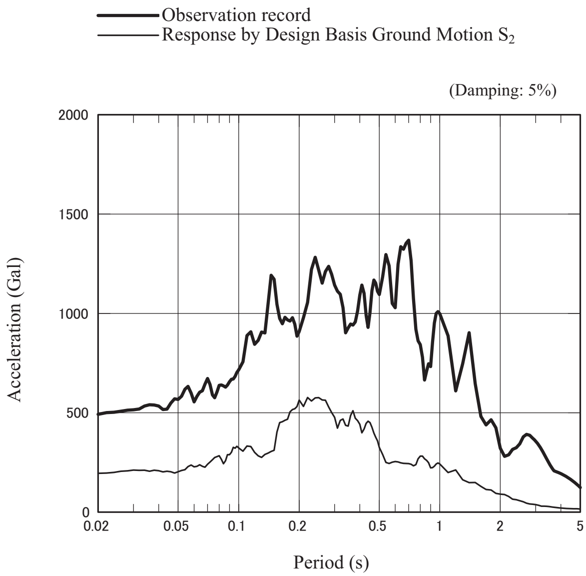


EW

FIG. II–1.10. Acceleration response spectra for basement 5 (on the foundations) of Unit 3 reactor building (observation point 3-R2).

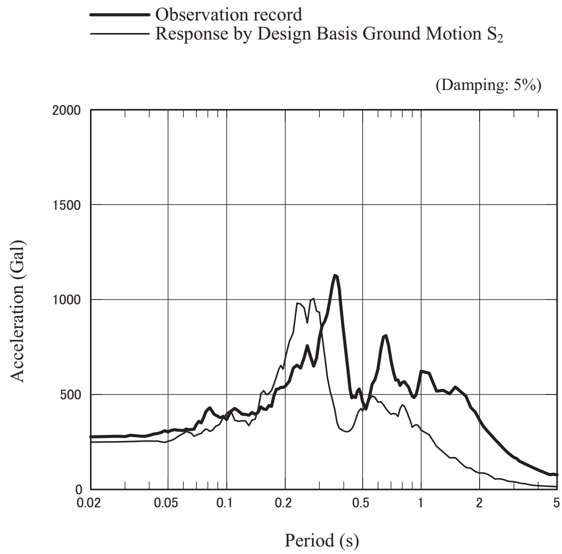


NS

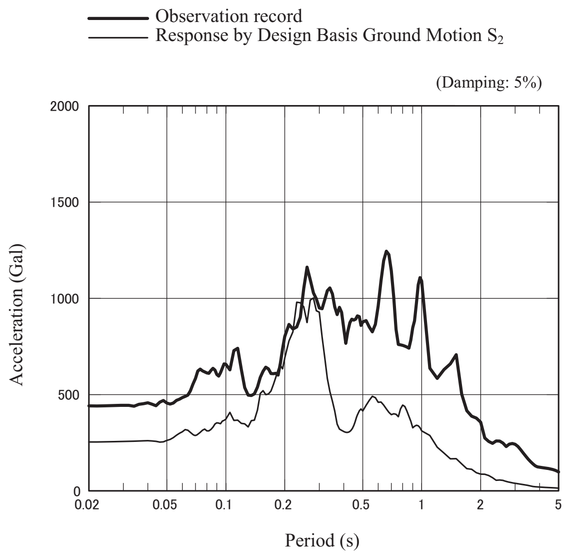


EW

FIG. II–1.11. Acceleration response spectra for basement 5 (on the foundations) of Unit 4 reactor building (observation point 4-R2).

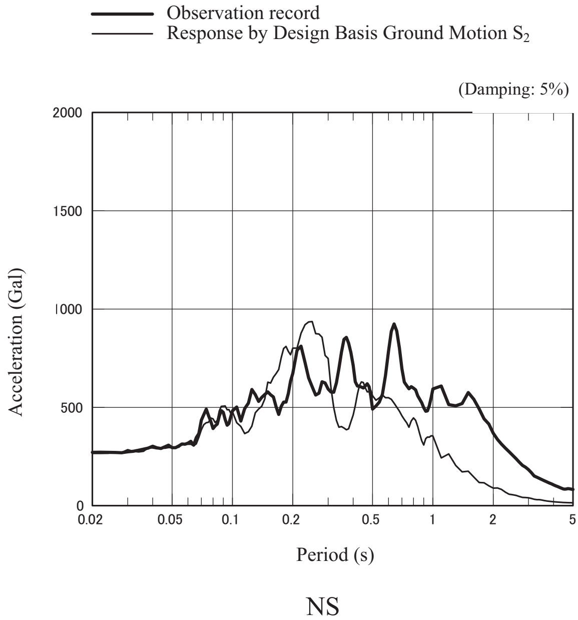


NS



EW

FIG. II–1.12. Acceleration response spectra for basement 4 (on the foundations) of Unit 5 reactor building (observation point 5-R2).



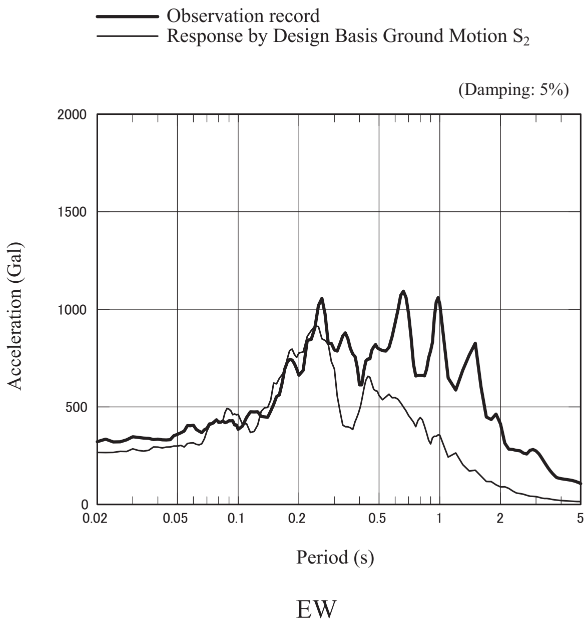
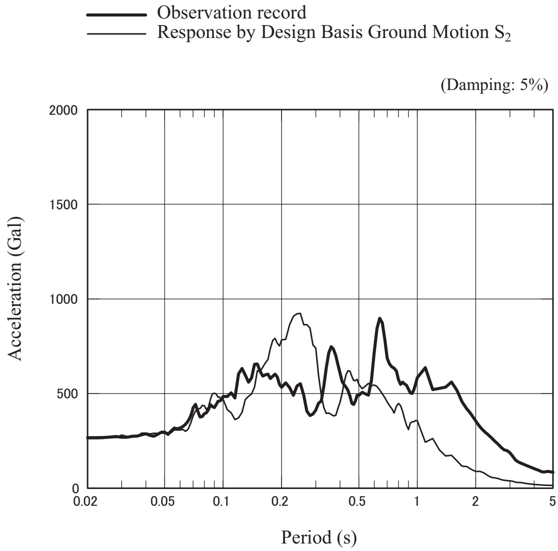
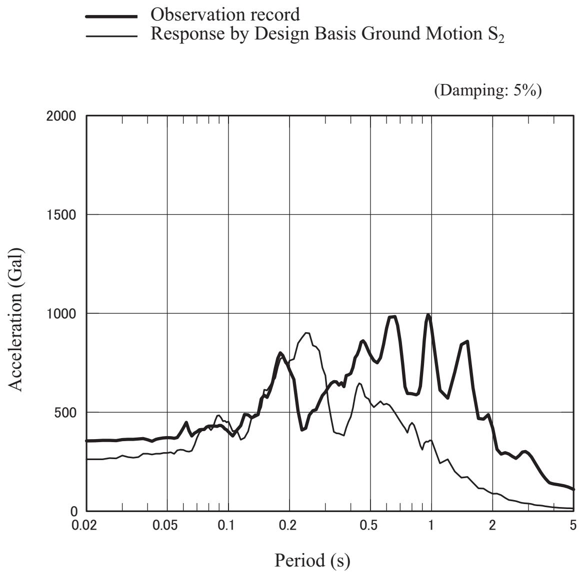


FIG. II–1.13. Acceleration response spectra for basement 3 (on the foundations) of Unit 6 reactor building (observation point 6-R2).



NS



EW

FIG. II–1.14. Acceleration response spectra for basement 3 (on the foundations) of Unit 7 reactor building (observation point 7-R2).

TABLE II–2. SHIKA (JAPAN) NUCLEAR POWER PLANT (2007)

II–2.1. SITE AND UNIT INFORMATION

II–2.1.1. Site Data (Name of the Site, Country, Operating Organization)

Shika Nuclear Power Station,
Japan, Hokuriku Electric Power Company

II–2.1.2. Unit Data (Unit Number, Net Capacity (MW(e)), Reactor Type, Reactor Supplier, Date of Commercial Operation, Foundation Level and Type)

Unit #1: 540 MW(e), BWR-5, Hitachi, 1993, Ground level –28.1 m, on the 1500 m/s bedrock
Unit #2: 1358 MW(e), ABWR, Hitachi, 2006, Ground level –25.7 m, on the 1500 m/s bedrock

II–2.2. SEISMIC DESIGN DATA

II–2.2.1. Design Basis Earthquake(s) Data

	DBE					
	S1			S2		
	Outcrop Bedrock Surface		Reactor Building Base Mat	Outcrop Bedrock Surface		Reactor Building Base Mat
	Maximum Acceleration	Depth*	Maximum Acceleration	Maximum Acceleration	Depth*	Maximum Acceleration
Unit #1	375 Gal	G.L. –31 m	233 Gal	490 Gal	G.L. –31 m	273 Gal
Unit #2	375 Gal	G.L. –31 m	259 Gal	490 Gal	G.L. –31 m	332 Gal

II–2.2.2. Seismic Instrumentation

II–2.2.2.1. Seismic Instrumentation for SCRAM

Automatic seismic SCRAM system is installed.
Seismic SCRAM sensors: Unit #1 12/unit (one out of two, twice, horizontal and vertical);
Unit #2 12/unit (two out of four, horizontal and vertical).
Set Values for seismic SCRAM: Unit #1 190 Gal, 505 Gal (horizontal), 165 Gal (vertical);
Unit #2 185 Gal, 505 Gal (horizontal), 165 Gal (vertical).

	Location*	Number of Instruments		Set values for seismic SCRAM
		Horizontal	Vertical	
Unit #1	R/B, the Third Floor (G.L. +7.3 m)	4	0	505 Gal (horizontal)
	Reactor Building Base Mat	4	4	190 Gal (horizontal), 165 Gal (vertical)
Unit #2	R/B, the Fourth Floor (G.L. +11.5 m)	4	0	505 Gal (horizontal)
	Reactor Building Base Mat	4	4	185 Gal (horizontal), 165 Gal (vertical)

* G.L.: Ground Level; R/B: Reactor Building.

II–2.2.2.2. Seismic Instrumentation for Recording

<table>
<tr><th rowspan="2"></th><th rowspan="2" colspan="4">Location*</th><th colspan="2">Number of Recorders (Operable)</th></tr>
<tr><th>Horizontal</th><th>Vertical</th></tr>
<tr><td rowspan="4">In-field</td><td rowspan="4" colspan="3">Near Unit #2 R/B</td><td>G.L. –1.5 m</td><td>2</td><td>1</td></tr>
<tr><td>G.L. –31 m</td><td>2</td><td>1</td></tr>
<tr><td>G.L. –121 m</td><td>2</td><td>1</td></tr>
<tr><td>G.L. –221 m</td><td>2</td><td>1</td></tr>
<tr><td rowspan="15">In-structure</td><td rowspan="7">Unit #1</td><td rowspan="4">R/B</td><td>Roof Truss</td><td>G.L. +33.83 m</td><td>2</td><td>2</td></tr>
<tr><td>the Fourth Floor</td><td>G.L. +16.63 m</td><td>4</td><td>4</td></tr>
<tr><td>the Second Floor</td><td>G.L. +0.3 m</td><td>8</td><td>0</td></tr>
<tr><td>Base Mat</td><td>G.L. –22.6 m</td><td>4</td><td>7</td></tr>
<tr><td rowspan="3">T/B</td><td>Roof Truss</td><td>G.L. +27.7 m</td><td>2</td><td>1</td></tr>
<tr><td>the Second Floor</td><td>G.L. +7.5 m</td><td>2</td><td>1</td></tr>
<tr><td>Base Mat</td><td>G.L. –11 m</td><td>2</td><td>1</td></tr>
<tr><td rowspan="8">Unit #2</td><td rowspan="5">R/B</td><td>Roof Truss</td><td>G.L. +37.85 m</td><td>4</td><td>2</td></tr>
<tr><td>the Fifth Floor</td><td>G.L. +19.7 m</td><td>4</td><td>1</td></tr>
<tr><td>the Fourth Floor</td><td>G.L. +11.5 m</td><td>6</td><td>2</td></tr>
<tr><td>the Second Floor</td><td>G.L. +0.3 m</td><td>5</td><td>0</td></tr>
<tr><td>Base Mat</td><td>G.L. –20.2 m</td><td>9</td><td>7</td></tr>
<tr><td rowspan="3">T/B</td><td>Roof Truss</td><td>G.L. +34.2 m</td><td>2</td><td>1</td></tr>
<tr><td>the Second Floor</td><td>G.L. +10.1 m</td><td>2</td><td>1</td></tr>
<tr><td>Base Mat</td><td>G.L. –15.4 m</td><td>2</td><td>1</td></tr>
</table>

* G.L.: Ground Level; R/B: Reactor Building; T/B: Turbine Building.

II–2.2.3. Design Standard

NSC Regulatory Guide [II–2] and JEAG 4601-1987 [II–3].

II–2.3. EARTHQUAKE EVENT DATA

II–2.3.1. General Information (Name of the Event, Date and Time, Magnitude, Geological Coordinates of the Epicentre, Focal Depth, Epicentral Intensity, Intensity at the Site)

Noto Hantou Earthquake, 9:42 a.m., 25 March 2007,
M_{JMA} = 6.9, Epicentre: 18 km north from the site, Focal Depth: 11 km,
Epicentral Intensity: JMA Intensity at the Site: 4.8

II–2.3.2. Observed Data at the Site (Accelerations, Time Historic Records, Spectra, etc., if available)

<table>
<tr><th rowspan="2" colspan="3">Observation Point</th><th colspan="3">Observed Peak Acceleration (Gal)</th><th colspan="2">Design Value (Gal)</th></tr>
<tr><th>NS[a]</th><th>EW[b]</th><th>UD[c]</th><th>NS</th><th>EW</th></tr>
<tr><td>Unit #1</td><td>R/B</td><td>B2F (Base Mat)</td><td>160</td><td>246</td><td>No data</td><td>273</td><td>256</td></tr>
<tr><td>Unit #2</td><td>R/B</td><td>B2F (Base Mat)</td><td>193</td><td>264</td><td>No data</td><td>262</td><td>332</td></tr>
</table>

[a] NS: north–south.
[b] EW: east–west.
[c] UD: up–down.
Time history records are shown in Figs II–2.1 to II–2.4.
Spectra of observation records are shown in Figs II–2.5 to II–2.8.

TABLE II–2. SHIKA (JAPAN) NUCLEAR POWER PLANT (2007) (cont.)

II–2.4. EFFECTS ON NUCLEAR POWER PLANTS

II–2.4.1. Operational States of Nuclear Power Plants (State of the Units, Shutdown Information)

Unit #1: Outage.
Unit #2: Outage.

II–2.4.2. Fundamental Safety Functions (Control of Reactivity, Cooling, Control of the Release of Radioactive Material)

Control of Reactivity: Control Rods already inserted
— Neutron flux securely confirmed.

Cooling: Continuous Shutdown Cooling successful
— Reactor temperature, pressure, water level, core flow, feedwater flow and main steam flow securely measured.

Control of the Release of Radioactive Material: No release of radioactive material
— D/W Monitor (temperature and pressure), Stack Radiation Monitor and SGTS Radiation Monitor kept functions.

II–2.4.3. Other Effects (Loss of Off-site Power, Seismically Induced Events)

(1) *Safety-related effects*
None

(2) *Non-safety-related damage*
Loss of off-site power: —
Seismically induced events:
Water spilled (45 L) from spent fuel pool over Unit #1;
Mercury vapour lamps fallen down at Units #1 and #2;
Displacement of turbine rotors of Unit #2 in the process of being assembled;
Rupture discs actuated in transformers;
Evidence of impact between structural elements in turbine building.

II–2.4.4. Actions for Restart

Fully organized inspections, conducted for all facilities including non-seismic safety:
— No significant damage on the safety related SSCs.
Seismic response analyses, conducted for all safety related SSCs:
— Response of SSCs under elastically allowable limits (Units #1 and #2).

II–2.4.5. Restart

Unit #2 restarted after one year.
Unit #1 remained shut down for a longer period for reasons other than the earthquake.

II–2.5. Other Actions

None

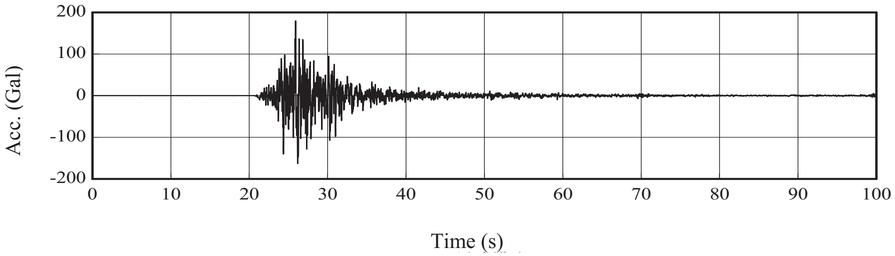


FIG. II–2.1. Acceleration–time history record (Unit #1, reactor building base mat elevation: –1.6 m). Direction: NS; maximum acceleration: 163 Gal.

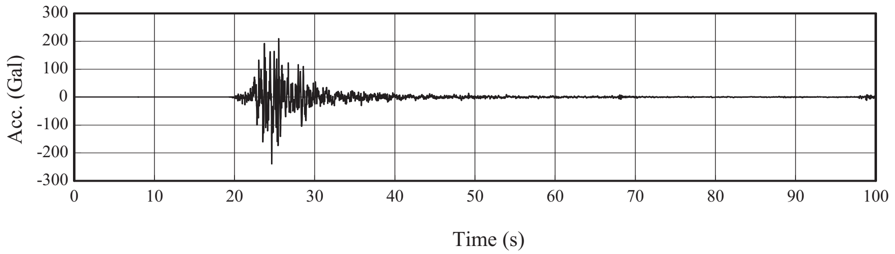


FIG. II–2.2. Acceleration–time history record (Unit #1, reactor building base mat elevation: –1.6 m). Direction: EW; maximum acceleration: 239 Gal.

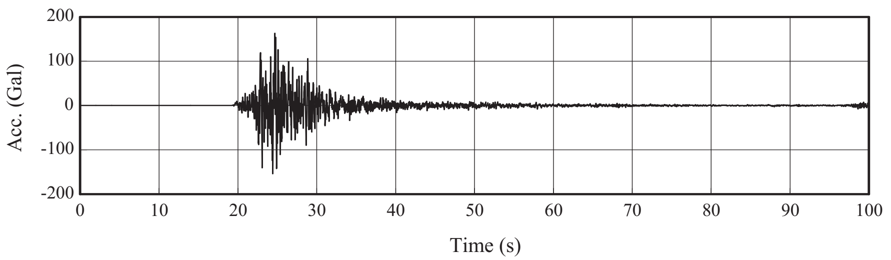


FIG. II–2.3. Acceleration–time history record (Unit #2, reactor building base mat elevation: +0.8 m). Direction: NS; maximum acceleration: 179 Gal.

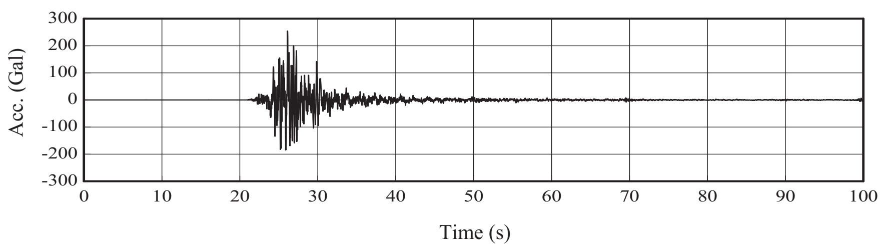


FIG. II–2.4. Acceleration–time history record (Unit #2, reactor building base mat elevation: +0.8 m). Direction: EW; maximum acceleration: 254 Gal.

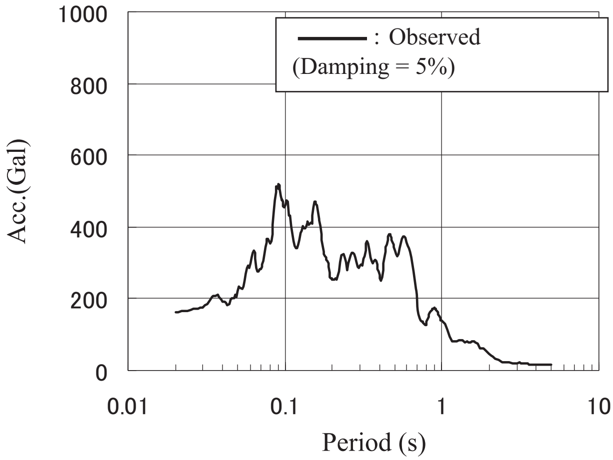


FIG. II–2.5. Acceleration response spectrum (Unit #1, reactor building base mat elevation: –1.6 m). Direction: NS.

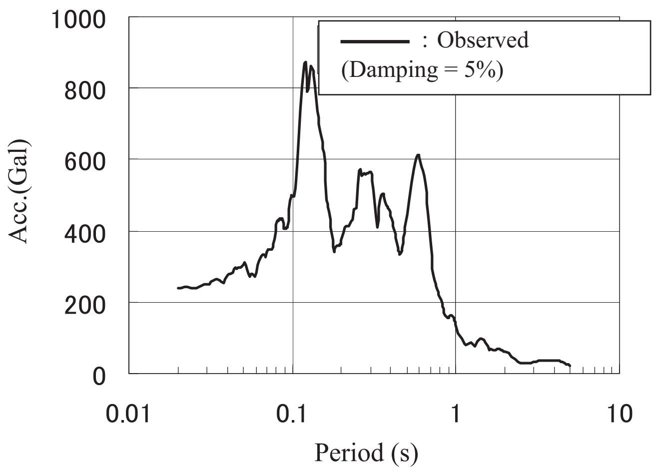


FIG. II–2.6. Acceleration response spectrum (Unit #1, reactor building base mat elevation: –1.6 m). Direction: EW.

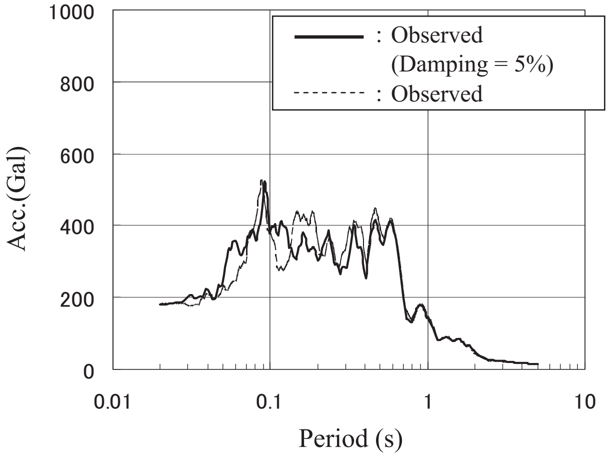


FIG. II–2.7. Acceleration response spectra (Unit #2, reactor building base mat elevation: +0.8 m). Direction: NS.

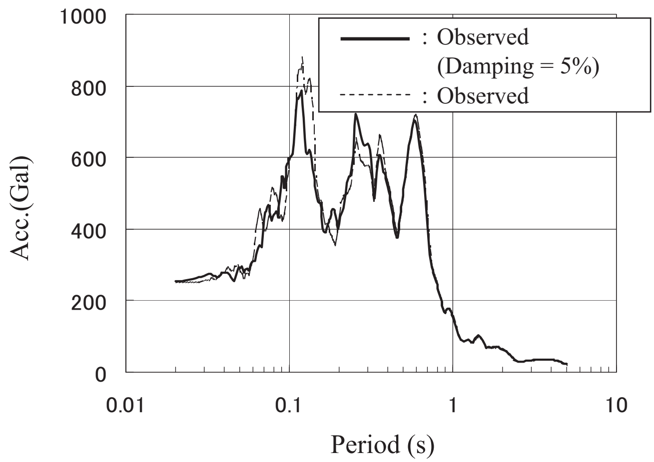


FIG. II–2.8. Acceleration response spectra (Unit #2, reactor building base mat elevation: +0.8 m). Direction: EW.

TABLE II–3. ONAGAWA (JAPAN) NUCLEAR POWER PLANT (2005)

II–3.1. SITE AND UNIT INFORMATION

II–3.1.1. Site Data (Name of the Site, Country, Operating Organization)

Onagawa Nuclear Power Station,
Japan, Tohoku Electric Power Company

II–3.1.2. Unit Data (Unit Number, Net Capacity (MW(e)), Reactor Type, Reactor Supplier, Date of Commercial Operation, Foundation Level and Type)

Unit #1, 524 MW(e), BWR-4, Toshiba, 1984, Ground level –16.0 m, on the 1490 m/s bedrock
Unit #2, 825 MW(e), BWR-5, Toshiba, 1995, Ground level –28.9 m, on the 1300 m/s bedrock
Unit #3, 825 MW(e), BWR-5, Toshiba, 2002, Ground level –28.9 m, on the 1360 m/s bedrock

II–3.2. SEISMIC DESIGN DATA

II–3.2.1. Design Basis Earthquake(s) Data

	DBE					
	S1			S2		
	Outcrop Bedrock Surface		Reactor Building Base mat	Outcrop Bedrock Surface		Reactor Building Base Mat
	Maximum Acceleration	Depth*	Maximum Acceleration	Maximum Acceleration	Depth*	Maximum Acceleration
Unit #1	250 Gal	G.L. –23.4 m	278 Gal	–	–	–
Unit #2	250 Gal	G.L. –23.4 m	265 Gal	375 Gal	G.L. –23.4 m	363 Gal
Unit #3	250 Gal	G.L. –23.4 m	260 Gal	375 Gal	G.L. –23.4 m	375 Gal

II–3.2.2. Seismic Instrumentation

II–3.2.2.1. Seismic Instrumentation for SCRAM
Seismometers: 180 seismometers at Onagawa nuclear power plant.
Automatic seismic SCRAM system is installed.
Seismic SCRAM sensors: 12/unit (One out of two, twice, horizontal and vertical).
Set Values for seismic SCRAM: Unit #1 200 Gal (horizontal), 100 Gal (vertical);
Unit #2 200 Gal, 400 Gal (horizontal), 100 Gal (vertical);
Unit #3 200 Gal, 350 Gal (horizontal), 100 Gal (vertical).

	Location*	Number of Instruments		Set Values for seismic SCRAM
		Horizontal	Vertical	
Unit #1	R/B, the First Floor (G.L. +8.7 m)	4	4	200 Gal (horizontal)
	Reactor Building Base Mat	4	0	200 Gal (horizontal), 100 Gal (vertical)
Unit #2	R/B, the First Basement Floor (G.L. –8.8 m)	4	4	400 Gal (horizontal)
	Reactor Building Base Mat	4	0	200 Gal (horizontal), 100 Gal (vertical)
Unit #3	R/B, the First Basement Floor (G.L. –8.8 m)	4	4	350 Gal (horizontal)
	Reactor Building Base Mat	4	0	200 Gal (horizontal), 100 Gal (vertical)

* G.L.: Ground Level; R/B: Reactor Building.

TABLE II–3. ONAGAWA (JAPAN) NUCLEAR POWER PLANT (2005) (cont.)

II–3.2.2.2. Seismic Instrumentation for Recording

Location*					Number of Recorders (Operable)	
					Horizontal	Vertical
In-field				G.L. –1.7 m	2	1
				G.L. –27.3 m	2	1
				G.L. –61.5 m	2	1
				G.L. –147.1 m	2	1
In-structure	Unit #1	R/B	Rooftop	G.L. +46.8 m	2	1
			Crane Floor	G.L. +37.8 m	2	0
			the Fifth Floor	G.L. +29.9 m	6	1
			the Third Floor	G.L. +16.8 m	2	1
			the Second Floor	G.L. +8.7 m	2	0
			the First Floor	G.L. +0.2 m	2	1
			Base Mat	G.L. –12.5 m	4	9
			Under Base Mat	G.L. –23.4 m	2	1
				G.L. –36.2 m	2	0
		T/B	Rooftop	G.L. +18.65 m	2	1
			the First Floor	G.L. +0.2 m	2	0
			Base Mat	G.L. –14.8 m	2	1
		C/B	Rooftop	G.L. +14.4 m	2	1
			the Third Floor	G.L. +8.7 m	2	0
			Base Mat	G.L. –13.3 m	2	1
		RW/B	Base Mat	G.L. –9.8 m	1	0
	Unit #2	R/B	Rooftop	G.L. +35.9 m	2	1
			Roof Truss	G.L. +32.2 m	0	1
			the Third Floor	G.L. +24.9 m	6	0
				G.L. +18.4 m	2	1
			the Second Floor	G.L. +7.7 m	2	1
			the First Floor	G.L. +0.2 m	2	1
			the First Basement Floor	G.L. –8.8 m	2	1
			the Second Basement Floor	G.L. –15.6 m	2	1
			Base Mat	G.L. –22.9 m	2	8
			Under Base Mat	G.L. –29.8 m	2	1
				G.L. –36.2 m	2	1
				G.L. –57.6 m	2	1
				G.L. –143.2 m	2	1
		T/B	Rooftop	G.L. +32.7 m	2	1
			the Second Floor	G.L. +10.0 m	2	1
			Base Mat	G.L. –14.0 m	2	1
		C/B	Rooftop	G.L. +14.4 m	2	1
			the Third Floor	G.L. +8.7 m	2	1
			Base Mat	G.L. –13.3 m	2	1
	Unit #3	R/B	Rooftop	G.L. +35.7 m	2	1
			the Third Floor	G.L. +15.8 m	5	1
			the First Floor	G.L. +0.2 m	2	1
			Base Mat	G.L. –22.9 m	5	1
			Under Base Mat	G.L. –29.8 m	2	1
				G.L. –143.2m	2	1
		T/B	the First Floor	G.L. +0.2 m	4	2
			Base Mat	G.L. –23.8 m	4	2
		S/B	Rooftop	G.L. +19.2 m	2	1
			Base Mat	G.L. –25.8 m	2	1
		Hx/B	the First Floor	G.L. +0.2 m	2	1
			Base Mat	G.L. –24.3 m	2	1

* G.L.: Ground Level; R/B: Reactor Building; T/B: Turbine building; C/B: Control Building; RW/B: Radwaste Building.

TABLE II–3. ONAGAWA (JAPAN) NUCLEAR POWER PLANT (2005) (cont.)

II–3.2.3. Design Standard

Unit #1: Design similar to procedures given in JEAG 4601-1970 [II–1].
Units #2 and 3: Refer to NSC Regulatory Guide [II–2] and JEAG 4601-1987 [II–3].

II–3.3. EARTHQUAKE EVENT DATA

II–3.3.1. General Information (Name of the Event, Date and Time, Magnitude, Geological Coordinates of the Epicentre, Focal Depth, Epicentral Intensity, Intensity of the Site)

Miyagi Offshore Earthquake, 11.46 a.m., 16 August 2005, M_{JMA} = 7.2, Epicentre: 73 km south-west from the site, Focal Depth: 42 km, Epicentral Intensity: JMA Intensity of the Site: <5.

II–3.3.2. Observed Data in the Site (Accelerations, Time History Records, Spectra, etc., if available)

Observation Point			Observed Peak Acceleration (Gal)			Design Value (Gal)	
			NS	EW	UD	NS	EW
Unit #1	R/B	B2F (Base Mat)	263	194	164	—	—
Unit #2	R/B	B3F (Base Mat)	230	206	186	357	363
Unit #3	R/B	B3F (Base Mat)	238	76	201	375	366

Time Historic Records are shown in Figs II–3.1 to II–3.6.
Spectra of observation records are shown in Figs II–3.7 to II–3.12.

II–3.4. EFFECTS ON NUCLEAR POWER PLANTS

II–3.4.1. Operational States of Nuclear Power Plants (State of the Units, Shutdown Information)

Unit #1: Full Power Operation, Automatic Shutdown by Seismic SCRAM
Unit #2: Full Power Operation, Automatic Shutdown by Seismic SCRAM
Unit #3: Full Power Operation, Automatic Shutdown by Seismic SCRAM

II–3.4.2. Fundamental Safety Functions (Control of Reactivity, Cooling, Control of the Release of Radioactive Material)

Control of Reactivity: All Control Rods inserted successfully
— Neutron flux securely confirmed.

Cooling: Shift to the Shutdown Cooling Mode successful
— Reactor temperature, pressure, water level, core flow, feedwater flow, main steam flow securely measured.

Control of the Release of Radioactive Material: No release of radioactive material
— D/W Monitor (temperature and pressure), Stack Radiation Monitor and SGTS Radiation Monitor kept functions.

II–3.4.3. Other Effects (Loss of Off-site Power, Seismically Induced Events)

(1) Safety related effects:
None.
(2) Non-safety-related damage:
- Crack in window glass at visitor's room in reactor building of Unit 3;
- Fall-down of a lump in the waste storage building;
- Crack in the road surface in the station yard.

TABLE II–3. ONAGAWA (JAPAN) NUCLEAR POWER PLANT (2005) (cont.)

II–3.4.4. Actions for Restart

Fully organized inspections, conducted for all facilities including non-seismic safety:
— No significant damage on the SSCs.
Seismic response analyses, conducted for all safety related SSCs:
— Response of SSCs under elastic allowable limits (Unit #1, #2 and #3).
Evaluation of DBE:
— Re-definitions of DBE and evaluation of safety related SSCs conducted before restart.

	New DBE (Ss)		
	Outcrop bedrock surface		Reactor Building Base Mat
	Maximum Acceleration (Gal)	Depth* (m)	Maximum Acceleration (Gal)
Unit #1	580	G.L. –23.4	528
Unit #2			597
Unit #3			513

* G.L.: Ground Level.

II–3.4.5. Restart

Unit #2 restarted after five months.
Unit #3 restarted after seven months.
Unit #1 restarted after 11 months.

II–3.5. OTHER ACTIONS

None.

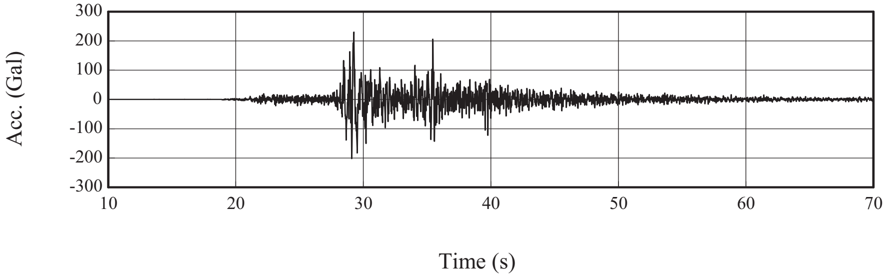


FIG. II–3.1. Acceleration–time history record (Unit #1, reactor building base mat). Direction: NS; maximum acceleration: 241 Gal.

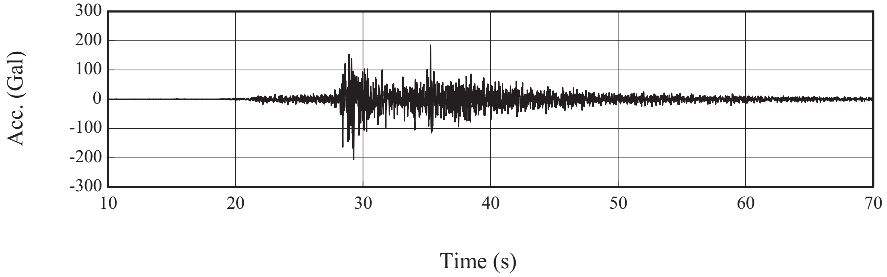


FIG. II–3.2. Acceleration–time history record (Unit #1, reactor building base mat). Direction: EW; maximum acceleration: 175 Gal.

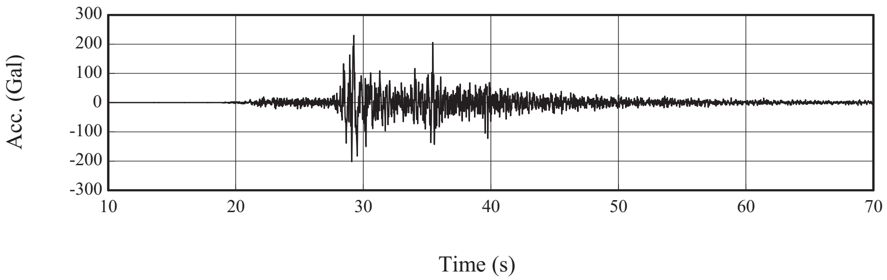


FIG. II–3.3. Acceleration–time history record (Unit #2, reactor building base mat); Direction: NS; maximum acceleration: 230 Gal.

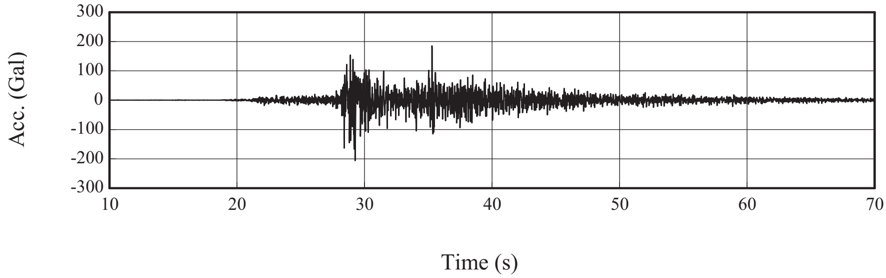


FIG. II–3.4. Acceleration–time history record (Unit #2, reactor building base mat). Direction: EW; maximum acceleration: 206 Gal.

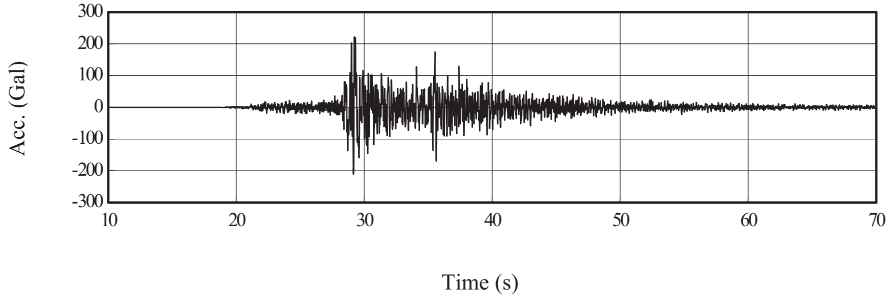


FIG. II–3.5. Acceleration–time history record (Unit #3, reactor building base mat). Direction: NS; maximum acceleration: 222 Gal.

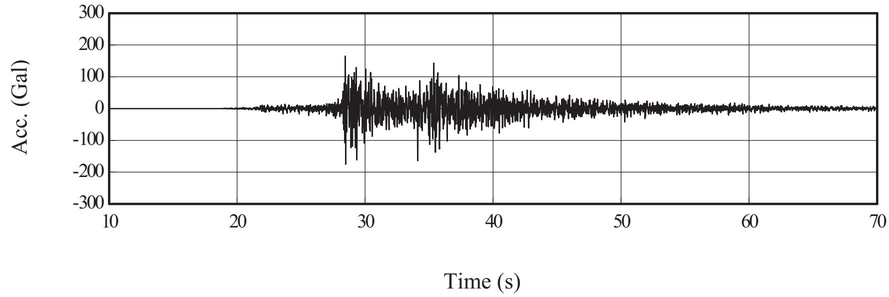


FIG. II–3.6. Acceleration–time history records (Unit #3, reactor building base mat). Direction: EW; maximum acceleration: 175 Gal.

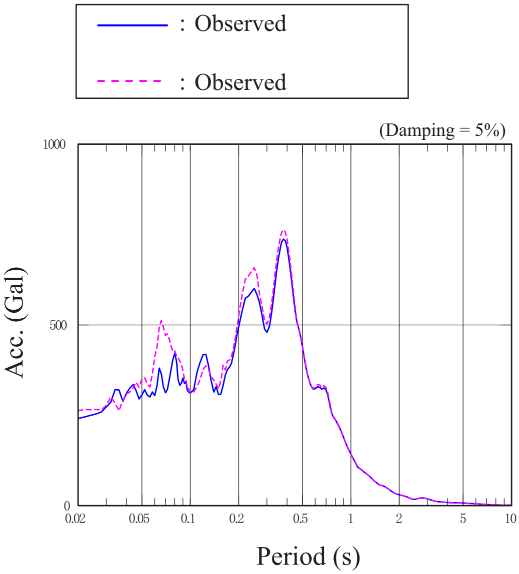


FIG. II–3.7. Acceleration response spectra (Unit #1, reactor building base mat). Direction: NS.

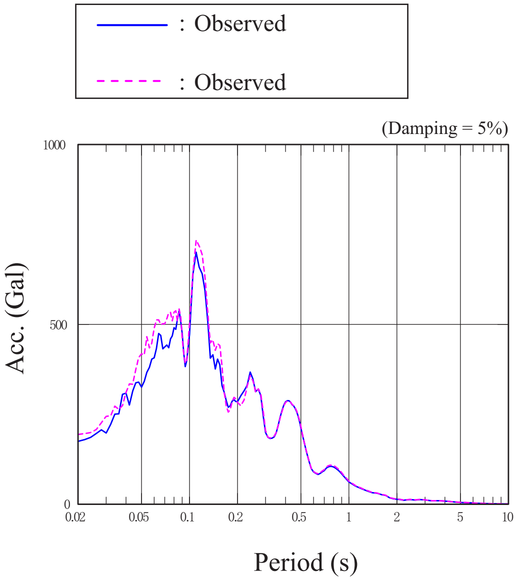


FIG. II–3.8. Acceleration response spectra (Unit #1, reactor building base mat). Direction: EW.

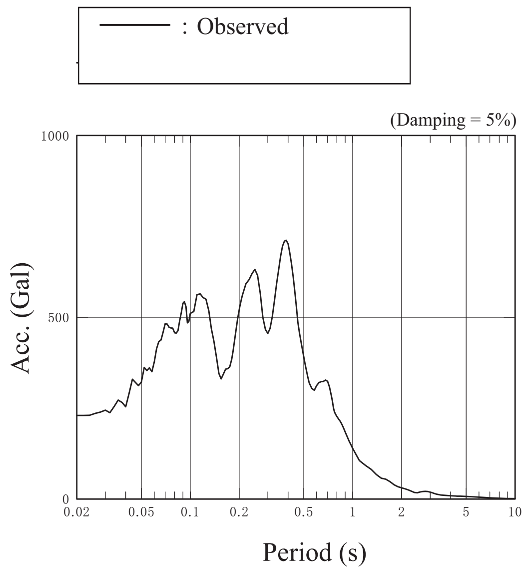


FIG. II–3.9. Acceleration response spectrum (Unit #2, reactor building base mat). Direction: NS.

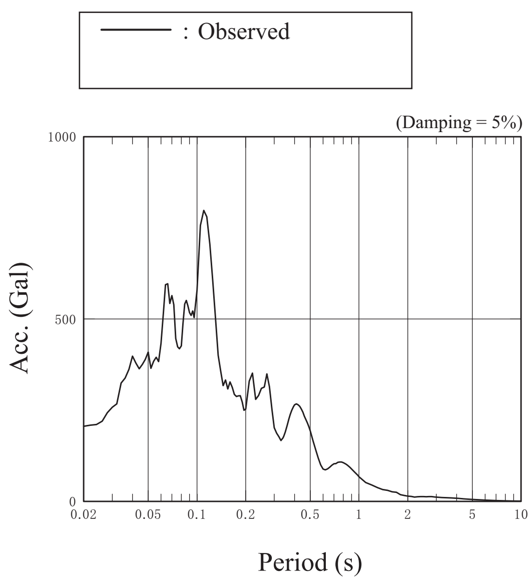


FIG. II–3.10. Acceleration response spectrum (Unit #2, reactor building base mat). Direction: EW.

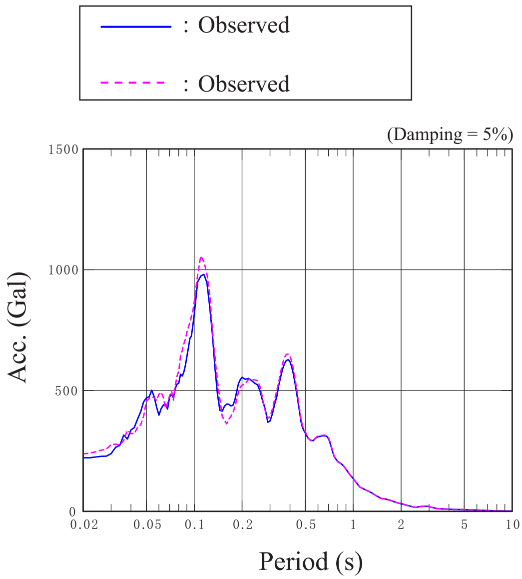


FIG. II–3.11. Acceleration response spectra (Unit #3, reactor building base mat). Direction: NS.

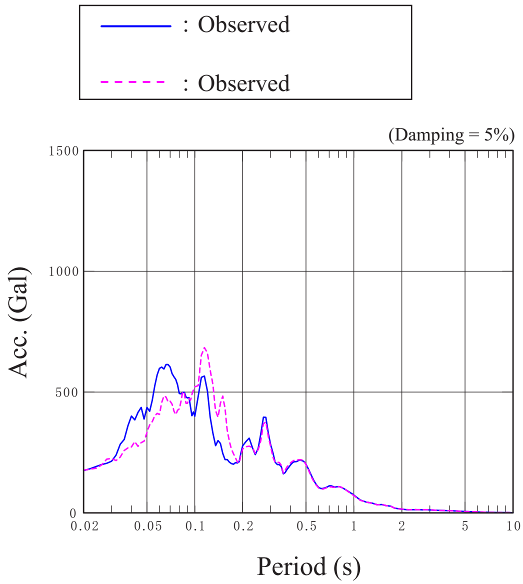


FIG. II–3.12. Acceleration response spectra (Unit #3, reactor building base mat). Direction: EW.

TABLE II-4. METSAMOR (ARMENIA) NUCLEAR POWER PLANT (1988)

II–4.1. SITE AND UNIT INFORMATION

II–4.1.1. Site Data (Name of the Site, Country, Operating Organization)

Armenian Nuclear Power Plant (ANPP), Armenia, Ministry of Energy

II–4.1.2. Unit Data (Unit Number, Net Capacity (MW(e)), Reactor Type, Reactor Supplier, Date of Commercial Operation, Foundation Level and Type)

Unit #1: 440 MW(e), WWER-440/270, 1976, Ground level: –3.8 m on the rock 1800 m/s
Unit #2: 440 MW(e), WWER-440/270, 1980, Ground level: –3.8 m on the rock 1800 m/s

II–4.2. SEISMIC DESIGN DATA

II–4.2.1. Seismic Hazard, DBE(s), SPSA, SMA Data

	DBE, Intensity MSK-64		RLE, ZPGA[a]	
	DE[b]	MDE[c]	50% confidence	84% confidence
Unit #1	7	8	–	–
Unit #2	7	8	0.21 g	0.35 g

[a] ZPGA: zero period ground acceleration.
[b] DE: design earthquake.
[c] MDE: maximum design earthquake.

TABLE II-4. METSAMOR (ARMENIA) NUCLEAR POWER PLANT (1988) (cont.)

II–4.2.2. Seismic Instrumentation

II–4.2.2.1. Seismic Instrumentation for SCRAM

An automatic seismic SCRAM system is installed and three seismic SCRAM sensors with double blocks. I: vent stack foundation, II: administrative building basement, III: switchyard. Three spatial components (two horizontal and one vertical).

The set value for seismic SCRAM is more than 50 Gal (6 bal), where bal is an MKS intensity measure.

II–4.2.2.2. Seismic Instrumentation for Recording

No. of post	Post location	Layout		Level (m)	Components
		Line	Axis		
0	Shelter No. 1			–4.6	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
1	Cable tunnel	B	15–16	–3 6	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
	Operating floor			10.5	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
3	Vent centre	B	18–19 (18)	21.9	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
4	Roof frame	B	15–16	25.7	Accelerometer OSP x; y; z
5	Distribution device	B	16	9.6	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
6	Control room	B	10	9.6	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
7	Operating floor		12	10.5	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
8	Borehole No. 2			–8	Accelerometer OSP x; y; z
9	Borehole No. 2			–25	Accelerometer OSP x; y; z
10	Borehole No. 2			–44	Accelerometer OSP x; y; z
11	Pedestal turbine 2	A and B 12 or elevation A	19–2 (20)	–4.2	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
12	Switchyard			–2.9	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
13	Administrative building			–0.9	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
14	Vent stack			142	Velocity graph VBP x; y
15	Vent stack			82	Velocity graph VBP x; y
16	Vent stack foundations				Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z
17	Free field			Territory	Accelerometer OSP x; y; z
					Velocity graph S5S x; y; z

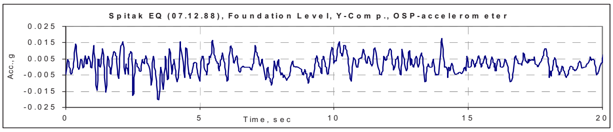
TABLE II-4. METSAMOR (ARMENIA) NUCLEAR POWER PLANT (1988) (cont.)

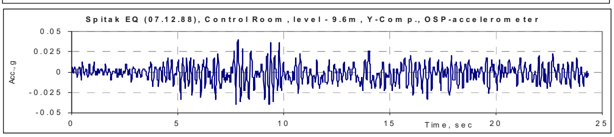
II–4.2.3. Design Standard
For the original design refer to PNAEG G-7-002-86 [II–4] and for the evaluation in 1999 refer to the IAEA Technical Guidelines given in Ref. [II–5].
II–4.3. EARTHQUAKE EVENT DATA
II–4.3.1. General Information (Name of the Event, Date and Time, Magnitude, Geological Coordinates of the Epicentre, Focal Depth, Epicentral Intensity, Intensity of the Site)
Spitak earthquake, 7 December 1988, $M = 6.9$, epicentre, 80 km north of the site, 5.5 (MSK-64) at the site.

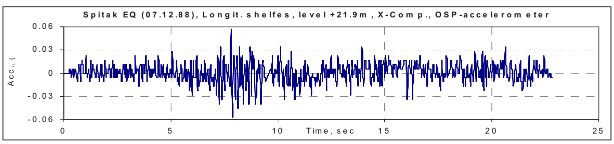
TABLE II-4. METSAMOR (ARMENIA) NUCLEAR POWER PLANT (1988) (cont.)

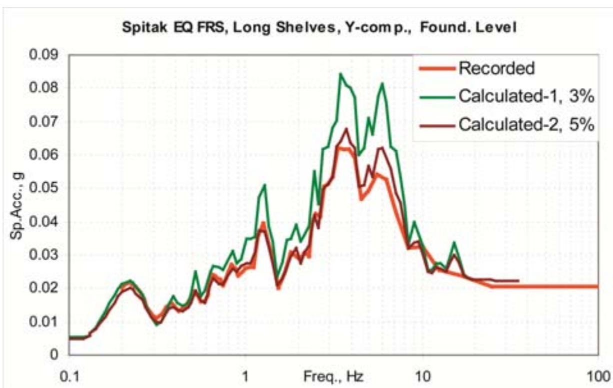
II–4.3.2. Observed Data at the Site (Accelerations, Time History Records, Spectra, etc.)

Observation point			Observed peak acceleration (g)		Design value RLE (g)	
			NS	EW	NS	EW
Unit #1	RB	Cable tunnel	0.02		0.35	0.35
Unit #1	RB	Control room	0.04		0.55	0.6
Unit #1	RB	Intake camera		0.05	0.8	0.9









II–4.4. EFFECTS ON NUCLEAR POWER PLANTS

II–4.4.1. Operational States of Nuclear Power Plants (State of the Units, Shutdown Information)

No automatic shutdown by seismic SCRAM, signals were lower than the trigger level.
Unit #1: detailed walkdown and inspection, no damage revealed.
Unit #2: detailed walkdown and inspection, no damage revealed.
Both units were shut down by a Government decree.

TABLE II-4. METSAMOR (ARMENIA) NUCLEAR POWER PLANT (1988) (cont.)

II–4.4.2. Fundamental Safety Functions (Control of Reactivity, Cooling, Control of the Release of Radioactive Material)
No release of radioactive material.
II–4.4.3. Other Effects (Loss of Off-site Power, Seismically Induced Events)
No safety related damage and no non-safety-related damage.
II–4.4.4. Actions for Restart
Realization of complex reconstruction and seismic upgrading programme. Implementation of additional studies on seismic hazards at the site. Establishment of the Seismic Evaluation Programme [II–5].
II–4.4.5. Restart
Unit #1 remains shut down. Unit #2: operation restarted.
II–4.5. OTHER ACTIONS
IAEA follow-up Seismic Safety Review Missions dedicated to different tasks of the Seismic Evaluation Programme [II–5]: the Programme is currently in the final stage of implementation.

REFERENCES TO ANNEX II

[II–1] JAPAN ELECTRIC ASSOCIATION, Technical Guidelines for Seismic Design of Nuclear Power Plants, Rep. JEAG 4601-1970, JEA, Tokyo (1970) (in Japanese).

[II–2] NUCLEAR SAFETY COMMISSION OF JAPAN, Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, NSC, Tokyo (1981).

[II–3] JAPAN ELECTRIC ASSOCIATION, Technical Guidelines for a Seismic Design of Nuclear Power Plants, JEA, Tokyo (1987) (in Japanese).

[II–4] PNAEG, Conventional Building Code with Max. Spectral Acceleration Input for RV Shaft Reconstruction, Rep. PNAEG-7-002-86, Energoatomizdat, Moscow (1989) (in Russian).

[II–5] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Technical Guidelines for the Seismic Evaluation and Upgrading Programme of the Armenian Nuclear Power Plant — Unit 2, IAEA, Vienna (1997).

Annex III

INFLUENCE OF PLASTICITY STRAIN ON FATIGUE STRENGTH

III–1. INTRODUCTION

The Structural Integrity Assessment Committee for Nuclear Components Damaged by Earthquake (SANE), established under the Japan Nuclear Technology Institute (JANTI), conducted research into the effect of plasticity strain on fatigue strength in order to evaluate the structural integrity of equipment after an earthquake, because an earthquake may cause local plasticity strain. The committee carried out fatigue tests to confirm this by using pre-strained specimens made of two materials used for important equipment: austenitic stainless steel and low alloy steel. As a result of the tests, it was confirmed that 8% cyclic strain ($\Delta\varepsilon$ = 16%) caused no notable change of fatigue strength. The committee reported this result in their interim report [III–1] in April 2008. This Annex is a summary of the result of the fatigue tests carried out.

III–2. FATIGUE TESTS

III–2.1. Materials under test

Hourglass specimens of 8 mm diameter (K_t = 1.05) made of two types of materials, austenitic stainless steel (SUS316NG) and low alloy steel (SFVQ1A), were used for the tests.

III–2.2. Test methods

A fatigue testing machine is shown in Fig. III–1. The tests were conducted controlling specimen diameters by use of displacement gauges.

III–2.3. Fatigue tests after pre-strain

The constant amplitude cyclic pre-strain shown in Fig. III–2 was applied to specimens in consideration of the effect of first passage seismic load and equivalent repeated seismic loads. After this, fatigue tests were carried out and the effect of plasticity strain was researched.

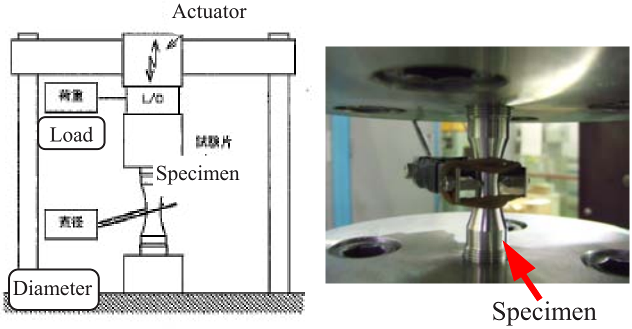


FIG. III–1. Fatigue testing machine and specimen.

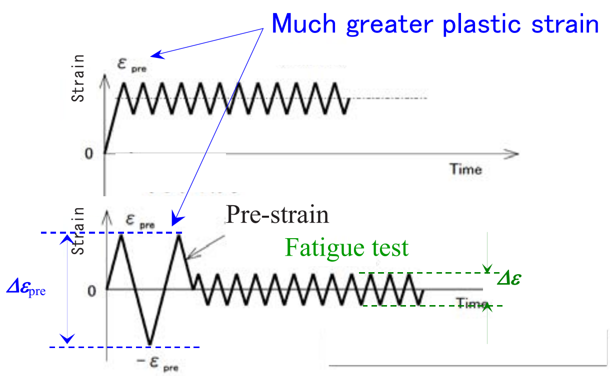


FIG. III–2. Fatigue test conditions.

In the case of pre-strain in the 16% cyclic range, the numbers of cycles were ten patterns: 0.25, 0.5, 0.75, 1.0, 1.25, 1.5, 1.75, 2.0, 2.25, 2.5. On the other hand, in the case of the 4% and 8% cyclic pre-strain ranges, the numbers varied from 4 to 30.

After repeated pre-straining, fatigue tests were carried out and fatigue lives were evaluated. In the tests, the 2.5% (SUS316NG, SUS316L) and 2% (SFVQ1A) strain ranges were used. The ranges corresponded to the condition that the rupture lives were 1000 cycles.

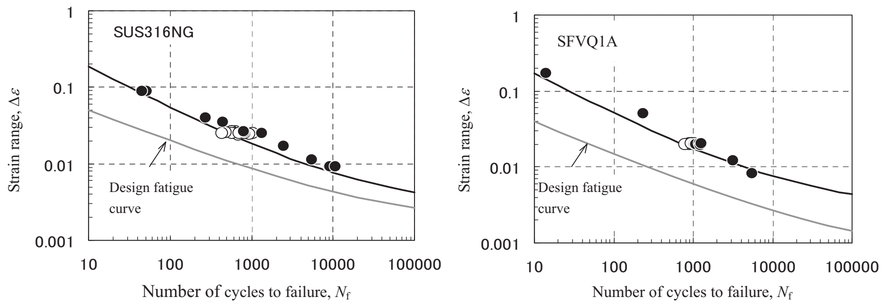


FIG. III–3. Results of low cycle fatigue tests after repeated applications of pre-strain ($\Delta\varepsilon = 16\%$).

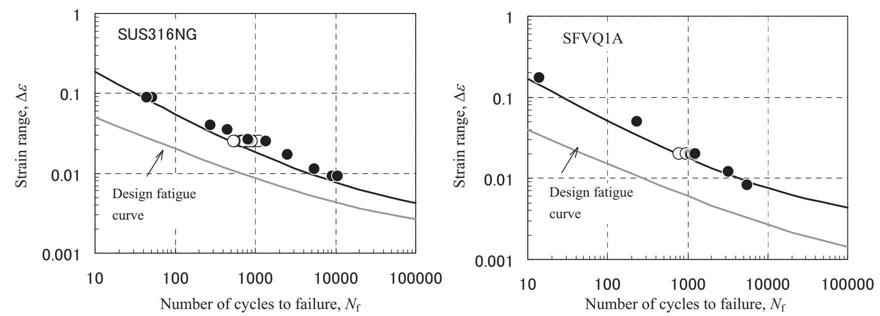


FIG. III–4. Results of low cycle fatigue tests after repeated applications of pre-strain ($\Delta\varepsilon = 8\%$).

III–3. RESULTS

The results of low cycle fatigue tests after cyclic pre-strain was applied to specimens are shown in Figs III–3 and III–4. In these figures, closed circles show the results of fatigue life without pre-strain and open circles show the results after constant amplitude and cyclic pre-strain. As a result of these tests, it is confirmed that both austenitic stainless steel (SUS316NG) and low alloy steel (SFVQ1A) caused no notable reduction of fatigue life, and a sufficient margin of fatigue life remained compared with design fatigue curves.

III–4. CONCLUSIONS

It was confirmed that if the maximum seismic load caused 8% cyclic strain, which is much greater than that caused by the Niigata-ken Chuetsu-oki earthquake in 2007, both austenitic stainless steel and low alloy steel indicated no notable change of fatigue life.

In addition, it was confirmed that if alternate seismic loads caused 30 cycles of 2% strain, greater than that caused by the Niigata-ken Chuetsu-oki earthquake, austenitic stainless steel indicated no notable change of fatigue life of austenitic stainless steel.

REFERENCE TO ANNEX III

[III–1] JAPAN NUCLEAR TECHNOLOGY INSTITUTE, Structural Integrity Assessment for Nuclear Components Damaged by Earthquake, Interim Report, JANTI, Tokyo (2008).

Annex IV

TYPICAL SURVEILLANCE TESTS FOR BWRs AND PWRs

(from Ref. [IV–1])

Surveillance tests include those tests performed at regular intervals to demonstrate the availability and operability of components and systems important to nuclear safety, or required to mitigate the consequences of accidents. Surveillance tests are identified in the plant technical specifications. They are performed by plant personnel and consist of checks, tests, calibrations and inspections to verify the availability and performance of the tested component and system. Typical surveillance tests of components and systems include the following:

— Measurement of the opening and closing times of motor operated valves;
— Measurement of the closing time and leak rate of containment isolation valves;
— Measurement of the flow and discharge pressure of pumps and fans;
— Measurement of the concentration, pressure, temperature, and fluid level of tanks and heat exchangers;
— Verification of automatic startup of standby components and systems (e.g., emergency core cooling pumps and diesel generators);
— Testing and calibration of instrumentation;
— Monitoring of reactor coolant system leakage;
— Visual inspection and disassembly of components;
— Verification of the control logic in reactor protection systems and engineered safety systems;
— Measurement of scram insertion times of control rods.

Systems for which surveillance tests are normally provided for in the technical specifications of nuclear power plants of BWR and PWR types, respectively, include the following:

(a) For BWRs:

— The reactor protection system;
— The control rod system;
— The liquid poison system;
— The core spray system;

— The containment spray system;
— Safety and solenoid-activated relief valves;
— Reactor recirculation pumps;
— Reactor coolant system isolation valves;
— The automatic depressurization system;
— The high pressure coolant injection system;
— The low pressure coolant injection system;
— The residual heat removal system;
— The reactor core isolation cooling system;
— The emergency cooling system;
— Containment (primary and secondary) isolation valves;
— Shock suppressors (snubbers);
— The emergency ventilation system;
— The control room ventilation system;
— Suppression chamber instrumentation;
— Emergency AC and DC power supplies;
— The auxiliary feedwater;
— The service water;
— The component cooling water;
— Diesel generators;
— Fire detection and suppression equipment;
— The remote shutdown panel;
— Radioactive effluent treatment equipment and instrumentation;
— Accident monitoring instrumentation.

(b) For PWRs:

— The reactor protection system;
— The control rod system;
— Protective instrumentation;
— The containment spray system;
— Safety valves and power operated relief valves;
— Reactor coolant system isolation valves;
— The high pressure injection system;
— The low pressure injection system;
— The shutdown cooling system;
— Containment isolation valves;
— Containment vacuum relief valves;
— Shock suppressors (snubbers);
— The emergency ventilation system;
— The control room ventilation system;

— Alarms;
— Emergency AC and DC power supplies;
— Diesel generators;
— Fire detection and suppression equipment;
— The remote shutdown panel;
— Radioactive effluent treatment and instrumentation;
— Accident monitoring instrumentation;
— The auxiliary feedwater;
— The service water;
— The component cooling water.

REFERENCE TO ANNEX IV

[VI–1] ELECTRIC POWER RESEARCH INSTITUTE, Guidelines for Nuclear Plant Response to an Earthquake, Rep. EPRI-NP-6695, EPRI, Palo Alto, CA (1989).

DEFINITIONS

The following definitions apply for the purposes of this specific safety report only.

generic equipment ruggedness spectra (GERS). A set of qualification test or fragility test response spectra for many varieties of equipment that documents the moderately high earthquake motions for which the equipment performs its designated functions. The GERS are generated by reviewing past shake table test results. The applicability of the GERS is a function of provisos to be applied to the equipment under consideration, including make, model and supporting conditions. GERS are one example of a GETRS.

generic equipment test response spectrum (GETRS). A response spectrum developed to represent families of qualification tests or fragility tests.

robust design. A design of a structure, system or component with a capacity that is well above that required for a postulated event, as a result of redundancy or excessive conservatism (i.e. in the definitions of loads or acceptance criteria).

test response spectrum (TRS). Actual response spectrum achieved on the table in shake table testing.

ABBREVIATIONS

ASTS	automatic scram trip system
BDBE	beyond design basis earthquake
BWR	boiling water reactor
CAV	cumulative absolute velocity
CDF	cord damage frequency
DBE	design basis earthquake
EQ	equipment qualification
GERS	generic equipment response spectrum
GETRS	generic equipment test response spectrum
HCLPF	high confidence of low probability of failure
ISRS	in-structure response spectra
ITS	important to safety
LERF	large early release frequency
NCOE	Niigataken Chuetsu-oki earthquake
NDE	non-destructive examination
NITS	not important to safety
NRPG	not required for power generation
OBE	operating basis earthquake
PEqAP	post-earthquake action programme

PWR	pressurized water reactor
RLE	review level earthquake
RPG	required for power generation
SMA	seismic margin assessment
SPSA	seismic probabilistic safety assessment
SSCs	structures, systems and components
SSE	safe shutdown earthquake
TEPCO	Tokyo Electric Power Company
TRS	test response spectrum

CONTRIBUTORS TO DRAFTING AND REVIEW

Abe, H.	Japan Nuclear Energy Safety Organization, Japan
Basu, P.	Atomic Energy Regulatory Board, India
Blahoianu, A.	Canadian Nuclear Safety Commission, Canada
Chigama, A.	International Atomic Energy Agency
Cinat, E.	Comisión Nacional de Energía Atómica, Argentina
Contri, P.	Joint Research Centre, European Commission
Godoy, A.	International Atomic Energy Agency
Guerpinar, A.	Private Consultant, Turkey
Inoue, Y.	International Atomic Energy Agency
Jiménez Juan, A.	Consejo de Seguridad Nuclear, Spain
Johnson, J.J.	James J. Johnson & Associates, Inc., United States of America
Katona, T.	Paks Nuclear Power Plant, Hungary
Kennedy, R.	RPK Structural Mechanics Consulting, United States of America
Kobayashi, K.	Tokyo Electric Power Company, Japan
Kostarev, V.	CKTI-Vibroseism Ltd, Russian Federation
Labbé, P.	Électricité de France, France
Lee, Y.	Korea Power Engineering Company, Inc., Republic of Korea
Liszka, E.	International Atomic Energy Agency
Moreno A.	Ingeniería IDOM Internacional, Spain

Murphy, A.	United States Nuclear Regulatory Commission, United States of America
Nagasawa, K.	Tokyo Electric Power Company, Japan
Ochiai, K.	Japan Nuclear Technology Institute, Japan
Orita, S.	Tokyo Electric Power Company, Japan
Pan, R.	National Nuclear Safety Administration, China
Ravindra, M.K.	Private Consultant, United States of America
Shibata, H.	Association for the Development of Earthquake Prediction, Japan
Shima, A.	Tokyo Electric Power Company, Japan
Sollogoub, P.	International Atomic Energy Agency
Stevenson, J.	Private Consultant, United States of America
Yoshimura, S.	The University of Tokyo, Japan
Yun, K.	Korea Electric Power Research Institute, Republic of Korea
Zadoyan, P.	Armenian Nuclear Regulatory Authority, Armenia

IAEA
International Atomic Energy Agency

No. 22

Where to order IAEA publications

In the following countries IAEA publications may be purchased from the sources listed below, or from major local booksellers. Payment may be made in local currency or with UNESCO coupons.

AUSTRALIA
DA Information Services, 648 Whitehorse Road, MITCHAM 3132
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: service@dadirect.com.au • Web site: http://www.dadirect.com.au

BELGIUM
Jean de Lannoy, avenue du Roi 202, B-1190 Brussels
Telephone: +32 2 538 43 08 • Fax: +32 2 538 08 41
Email: jean.de.lannoy@infoboard.be • Web site: http://www.jean-de-lannoy.be

CANADA
Bernan Associates, 4501 Forbes Blvd, Suite 200, Lanham, MD 20706-4346, USA
Telephone: 1-800-865-3457 • Fax: 1-800-865-3450
Email: customercare@bernan.com • Web site: http://www.bernan.com

Renouf Publishing Company Ltd., 1-5369 Canotek Rd., Ottawa, Ontario, K1J 9J3
Telephone: +613 745 2665 • Fax: +613 745 7660
Email: order.dept@renoufbooks.com • Web site: http://www.renoufbooks.com

CHINA
IAEA Publications in Chinese: China Nuclear Energy Industry Corporation, Translation Section, P.O. Box 2103, Beijing

CZECH REPUBLIC
Suweco CZ, S.R.O., Klecakova 347, 180 21 Praha 9
Telephone: +420 26603 5364 • Fax: +420 28482 1646
Email: nakup@suweco.cz • Web site: http://www.suweco.cz

FINLAND
Akateeminen Kirjakauppa, PO BOX 128 (Keskuskatu 1), FIN-00101 Helsinki
Telephone: +358 9 121 41 • Fax: +358 9 121 4450
Email: akatilaus@akateeminen.com • Web site: http://www.akateeminen.com

FRANCE
Form-Edit, 5, rue Janssen, P.O. Box 25, F-75921 Paris Cedex 19
Telephone: +33 1 42 01 49 49 • Fax: +33 1 42 01 90 90
Email: formedit@formedit.fr • Web site: http://www. formedit.fr

Lavoisier SAS, 145 rue de Provigny, 94236 Cachan Cedex
Telephone: + 33 1 47 40 67 02 • Fax +33 1 47 40 67 02
Email: romuald.verrier@lavoisier.fr • Web site: http://www.lavoisier.fr

GERMANY
UNO-Verlag, Vertriebs- und Verlags GmbH, Am Hofgarten 10, D-53113 Bonn
Telephone: + 49 228 94 90 20 • Fax: +49 228 94 90 20 or +49 228 94 90 222
Email: bestellung@uno-verlag.de • Web site: http://www.uno-verlag.de

HUNGARY
Librotrade Ltd., Book Import, P.O. Box 126, H-1656 Budapest
Telephone: +36 1 257 7777 • Fax: +36 1 257 7472 • Email: books@librotrade.hu

INDIA
Allied Publishers Group, 1st Floor, Dubash House, 15, J. N. Heredia Marg, Ballard Estate, Mumbai 400 001,
Telephone: +91 22 22617926/27 • Fax: +91 22 22617928
Email: alliedpl@vsnl.com • Web site: http://www.alliedpublishers.com

Bookwell, 2/72, Nirankari Colony, Delhi 110009
Telephone: +91 11 23268786, +91 11 23257264 • Fax: +91 11 23281315
Email: bookwell@vsnl.net

ITALY
Libreria Scientifica Dott. Lucio di Biasio “AEIOU”, Via Coronelli 6, I-20146 Milan
Telephone: +39 02 48 95 45 52 or 48 95 45 62 • Fax: +39 02 48 95 45 48
Email: info@libreriaaeiou.eu • Website: www.libreriaaeiou.eu

JAPAN
Maruzen Company, Ltd., 13-6 Nihonbashi, 3 chome, Chuo-ku, Tokyo 103-0027
Telephone: +81 3 3275 8582 • Fax: +81 3 3275 9072
Email: journal@maruzen.co.jp • Web site: http://www.maruzen.co.jp

REPUBLIC OF KOREA
KINS Inc., Information Business Dept. Samho Bldg. 2nd Floor, 275-1 Yang Jae-dong SeoCho-G, Seoul 137-130
Telephone: +02 589 1740 • Fax: +02 589 1746 • Web site: http://www.kins.re.kr

NETHERLANDS
De Lindeboom Internationale Publicaties B.V., M.A. de Ruyterstraat 20A, NL-7482 BZ Haaksbergen
Telephone: +31 (0) 53 5740004 • Fax: +31 (0) 53 5729296
Email: books@delindeboom.com • Web site: http://www.delindeboom.com

Martinus Nijhoff International, Koraalrood 50, P.O. Box 1853, 2700 CZ Zoetermeer
Telephone: +31 793 684 400 • Fax: +31 793 615 698
Email: info@nijhoff.nl • Web site: http://www.nijhoff.nl

Swets and Zeitlinger b.v., P.O. Box 830, 2160 SZ Lisse
Telephone: +31 252 435 111 • Fax: +31 252 415 888
Email: infoho@swets.nl • Web site: http://www.swets.nl

NEW ZEALAND
DA Information Services, 648 Whitehorse Road, MITCHAM 3132, Australia
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: service@dadirect.com.au • Web site: http://www.dadirect.com.au

SLOVENIA
Cankarjeva Zalozba d.d., Kopitarjeva 2, SI-1512 Ljubljana
Telephone: +386 1 432 31 44 • Fax: +386 1 230 14 35
Email: import.books@cankarjeva-z.si • Web site: http://www.cankarjeva-z.si/uvoz

SPAIN
Díaz de Santos, S.A., c/ Juan Bravo, 3A, E-28006 Madrid
Telephone: +34 91 781 94 80 • Fax: +34 91 575 55 63
Email: compras@diazdesantos.es, carmela@diazdesantos.es, barcelona@diazdesantos.es, julio@diazdesantos.es
Web site: http://www.diazdesantos.es

UNITED KINGDOM
The Stationery Office Ltd, International Sales Agency, PO Box 29, Norwich, NR3 1 GN
Telephone (orders): +44 870 600 5552 • (enquiries): +44 207 873 8372 • Fax: +44 207 873 8203
Email (orders): book.orders@tso.co.uk • (enquiries): book.enquiries@tso.co.uk • Web site: http://www.tso.co.uk

On-line orders
DELTA Int. Book Wholesalers Ltd., 39 Alexandra Road, Addlestone, Surrey, KT15 2PQ
Email: info@profbooks.com • Web site: http://www.profbooks.com

Books on the Environment
Earthprint Ltd., P.O. Box 119, Stevenage SG1 4TP
Telephone: +44 1438748111 • Fax: +44 1438748844
Email: orders@earthprint.com • Web site: http://www.earthprint.com

UNITED NATIONS
Dept. I004, Room DC2-0853, First Avenue at 46th Street, New York, N.Y. 10017, USA
(UN) Telephone: +800 253-9646 or +212 963-8302 • Fax: +212 963-3489
Email: publications@un.org • Web site: http://www.un.org

UNITED STATES OF AMERICA
Bernan Associates, 4501 Forbes Blvd., Suite 200, Lanham, MD 20706-4346
Telephone: 1-800-865-3457 • Fax: 1-800-865-3450
Email: customercare@bernan.com · Web site: http://www.bernan.com

Renouf Publishing Company Ltd., 812 Proctor Ave., Ogdensburg, NY, 13669
Telephone: +888 551 7470 (toll-free) • Fax: +888 568 8546 (toll-free)
Email: order.dept@renoufbooks.com • Web site: http://www.renoufbooks.com

Orders and requests for information may also be addressed directly to:

Marketing and Sales Unit, International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 (or 22530) • Fax: +43 1 2600 29302
Email: sales.publications@iaea.org • Web site: http://www.iaea.org/books

10-37601

SEISMIC HAZARDS IN SITE EVALUATION FOR NUCLEAR INSTALLATIONS
IAEA Safety Standards Series No. SSG-9
STI/PUB/1448 (56 pp.; 2010)
ISBN 978–92–0–102910–2 Price: €29.00

EVALUATION OF SEISMIC SAFETY FOR EXISTING NUCLEAR INSTALLATIONS
IAEA Safety Standards Series No. NS-G-2.13
STI/PUB/1379 (84 pp.; 2009)
ISBN 978–92–0–100409–3 Price: €20.00

ADVANCED NUCLEAR POWER PLANT DESIGN OPTIONS TO COPE WITH EXTERNAL EVENTS
IAEA-TECDOC-1487 (222 pp.; 2006)
ISBN 92–0–100506–7 Price: €15.00

SAFETY OF NEW AND EXISTING RESEARCH REACTOR FACILITIES IN RELATION TO EXTERNAL EVENTS
Safety Reports Series No. 41
STI/PUB/1209 (99 pp.; 5 figures; 2005)
ISBN 92–0–111704–3 Price: €25.00

GEOTECHNICAL ASPECTS OF SITE EVALUATION AND FOUNDATIONS FOR NUCLEAR POWER PLANTS
IAEA Safety Standards Series No. NS-G-3.6
STI/PUB/1195 (53 pp.; 2004)
ISBN 92–0–107204–X Price: €19.00

SITE EVALUATION FOR NUCLEAR INSTALLATIONS
IAEA Safety Standards Series No. NS-R-3
STI/PUB/1177 (28 pp.; 2003)
ISBN 92–0–112403–1 Price: €15.00

SEISMIC EVALUATION OF EXISTING NUCLEAR POWER PLANTS
Safety Reports Series No. 28
STI/PUB/1149 (60 pp.; 3 figures; 2003)
ISBN 92–0–101803–7 Price: €18.00

FLOOD HAZARD FOR NUCLEAR POWER PLANTS ON COASTAL AND RIVER SITES
IAEA Safety Standards Series No. NS-G-3.5
STI/PUB/1170 (83 pp.; 2003)
ISBN 92–0–112803–7 Price: €20.00

This Safety Report provides detailed guidance on actions to be taken in preparation for, and following, a felt earthquake at a nuclear power plant, taking into account recently gained knowledge (up to 2010), mainly from Japan. The detailed guidance covers: (a) the evaluation of the seismic safety of the plant following a seismic event, including inspection procedures and protocols; (b) the identification of the phases and of tasks to be performed in accordance with specific plant conditions, including their priorities; and (c) a common and integrated technical framework for defining the applicable acceptance criteria.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978–92–0–108810–9
ISSN 1020–6450