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Guidelines for Nuclear Power Plant Response to an Earthquake

2015 TECHNICAL REPORT

Guidelines for Nuclear Power Plant Response to an Earthquake

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Product Description

Guidelines for nuclear power plant response to earthquakes enable utilities to evaluate in a timely manner the need for post-earthquake plant shutdown and to provide procedures for evaluation of earthquake effects on the plant, as well as criteria for plant restart. The procedures enable the responding team of operators and engineers to identify and assess any earthquake effects and, if shutdown is necessary, to return the plant to safe operation as rapidly as possible. The guidelines presented herein represent an update of guidance in EPRI report 3002000720, "Guidelines for Nuclear Plant Response to an Earthquake", that was issued in October, 2013. The 2013 EPRI technical report is a major re-write of EPRI report NP-6695, issued in 1989, to incorporate lessons learned and experience gained from major earthquakes at nuclear power plants world-wide since the 1990s and to expand the scope and applicability of the guidelines to recent vintage plants. The current EPRI report further revises and updates the 2013 report to incorporate important detailed changes and additions that resulted from the development and review of a pending revision of ANS-2.23-2002, "Nuclear Power Plant Response to an Earthquake ". An ANS Working Group chaired by EPRI is responsible for the ANS-2.23 standard revision.

Objective

To present updated guidelines for nuclear power plant response to an earthquake.

Approach

The updated guidelines were developed by a team with expertise in nuclear system performance, plant operations, licensing and seismic structural engineering disciplines, as well as direct experience in the response of nuclear plants and other industrial facilities to large earthquakes. Based on this expertise and experience, the team formulated comprehensive guidelines for utilities to develop plant-specific procedures for response to an earthquake.

Results

The EPRI guidelines provide for pre-earthquake planning and a progressive response to an earthquake that is felt at a nuclear power plant. The responses include recommended immediate actions, as well as post-shutdown and longer-term actions. The findings at each stage indicate the need for and the level of any additional effort. The guidelines recommend that plant personnel perform initial inspections, gather seismic recordings, and reach decisions on the need for plant shutdown and on plant readiness for shutdown. If the plant is shut down, the guidelines define procedures for near-term actions by plant operators to determine the earthquake's effects, with engineers performing focused inspections to determine if structures, systems and components (SSCs) have sustained significant damage or if operating systems are in any way impaired. The guidelines then define actions necessary to establish the readiness of the plant to restart. Finally, the guidelines provide for long-term evaluations which, in most cases, can be performed after plant restart.

EPRI Perspective

The guidelines given in this report recognize the existence of extensive emergency operating procedures used by operators to maintain nuclear power plants in a safe and stable condition. These procedures include requirements of US Nuclear Regulatory Commission (USNRC) regulations and plant Technical Specifications that must be complied with in the operators' decisions to shut down and to restart a plant, including the need to confer with USNRC representatives. These guidelines are not intended to infringe on or to change these requirements.

In addition, the guidelines presented herein are based on the knowledge that the operators are intimately familiar with the day-to-day conditions of the plant and can best perform the first assessment of the plant's condition following an earthquake. If shutdown is necessary, or the plant is shut down by or before an earthquake and shut down is required, seismic/structural engineers are called in to perform more thorough evaluations. EPRI's guidelines will be especially useful for cases in which earthquakes occur and cause little or no damage to important equipment and structures, as well as for cases involving damage to important equipment. In both instances, use of the guidelines will assist the utility in determining in a systematic, timely manner if the plant should be shut down for in-depth evaluations or can continue or resume operation. In the unlikely event that potentially damaging ground motions should occur at a site, implementation of EPRI's procedures will minimize the time needed to assess the impact on plant SSCs and provide assurance that the plant can safely operate.

Keywords

Earthquakes	Seismic effects
Seismic qualification	Mechanical equipment
Electrical equipment	Equipment anchorage
Structures	Seismic instrumentation

Definitions

For the purpose of this report, the following words and phrases are defined:

BWR. Boiling water reactor type nuclear power plant.

Cumulative Absolute Velocity (CAV) and Standardized CAV. The time integral of absolute acceleration over the duration of the strong shaking. The "Standardized CAV" algorithm in EPRI report TR-100082, *Standardization of the Cumulative Absolute Velocity*, ignores small-amplitude shaking and is therefore more stable. This quantity has been shown to be a good indicator of the damage potential of an earthquake ground motion. In this standard, CAV means Standardized CAV. The CAV is described in detail in Appendix A.

Felt Earthquake. An earthquake of sufficient size such that: (a) the vibratory ground motion is felt at the nuclear plant site and recognized as an earthquake based on a consensus of the control room operators on duty at the time, and (b) for plants with operable seismic instrumentation, the seismic triggers installed at the plant are activated.

Functional Damage. Significant damage to plant SSCs, either physical or other, which impairs the operability or reliability of the damaged item to perform its intended function. Minor damage such as slight or hairline cracking of concrete elements in structures does not constitute functional damage.

Limiting Conditions for Operation. Those conditions which must be satisfied during specific modes of operation of the nuclear power plant. Limiting conditions for operation are defined in 10CFR50.36(c)(2) as the "lowest functional capability or performance levels of equipment required for safe operation of the facility." They are listed individually in the plant Technical Specifications and provide the basis for operation of the plant within the conditions of the operating license.

Malfunction. Inability of a structure, system or component to perform its required function. Malfunction may be due to physical damage or to the temporary loading caused by the earthquake; for example, shaking causing "chatter" of electrical devices.

Non-SR. Non-safety-related, as in non-safety-related structures, systems and components.

Operating Basis Earthquake (OBE) exceedance. The OBE is considered to have been exceeded if the damage parameters based on the vibratory motion due to an earthquake exceed the limit values specified in Section 3.4.

Operable. A system, subsystem, train, component, or device is considered operable when it is capable of performing its specified function(s) in accordance with plant Technical Specifications. Implicit in this definition is the assumption that all necessary instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

Operating Basis Earthquake (OBE) ground motion. An earthquake ground motion that could reasonably be expected to occur at the plant site during the operating life of the plant considering the regional and local geology, seismology, and specific characteristics of local subsurface material. It is that earthquake ground motion for which those features of the nuclear power plant, necessary for continued operation without undue risk to the health and safety of the public, are designed to remain functional. The OBE level earthquake ground motion is referred to in most international applications as the Seismic Level 1 (SL1) earthquake level.

Physical Damage. Damage to plant SSCs that can be detected by visual inspections, nondestructive examinations, and/or tests (e.g., broken parts, cracks, plastic deformation, misalignment of joining components, excessive wear, etc.). The damaged item may or may not be capable of performing its intended function.

PSA. Probabilistic Safety Analysis.

PWR. Pressurized water reactor type nuclear power plant.

Safe Shutdown Earthquake (SSE) ground motion. Earthquake ground motion for which certain SSCs are designed to remain functional. These SSCs are those necessary to ensure:

- The integrity of the reactor coolant pressure boundary,
- The capability to shut down the reactor and maintain it in a safe shutdown condition, or

- The capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures of nuclear radiation exceeding allowable amounts.

The SSE level earthquake ground motion is referred to in most international applications as the Seismic Level 2 (SL2) earthquake level.

SR. Safety-related, as in safety-related structures, systems and components.

SSCs. Structures, systems and components.

Safe Shutdown Earthquake (SSE) exceedance. The SSE is considered to have been exceeded if the damage parameters based on the vibratory motion due to an earthquake exceed the limit values specified in Section 3.4.

Significant Damage. Significant damage (physical or functional) is considered to be damage which has the potential to adversely affect the functionality or reliability of structures, systems or components required for the safe operation of the nuclear power plant. Damage may be indicated by visual inspections, nondestructive examinations, and/or tests. Significant damage may be indicated by each or a combination of any or all of the following indicators:

Concrete Structures	New earthquake-induced cracks in concrete >0.06-inch in width[1] and those that extend through the thickness of the member, spalling of concrete, visible distortion of frames.
Steel Structures	New earthquake-induced visible plastic deformation or cracking of joints, 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 pipe, significant (>10%) flow reduction due to cross-section impairment or flow control valve function. Plastic deformation of piping or supports identifiable by visual inspection.[2]

[1] Concrete cracks 0.06"or greater may be indicative of yielding of reinforcement.

[2] Damage to insulation and denting or scratching of pipe are not considered significant

System Supports	Significant damage is identified when a support is no longer capable of performing its support design function.[3] Examples of supports to be considered are distribution system supports (piping, cable trays, HVAC ducts), pressure vessel supports (reactor, pressurizer, steam generator, heat exchangers, torus, condensers, etc.), and water storage tank supports.
Mechanical or Electrical Equipment	Visible distortion of anchorage system, sliding of the base of anchored components, rupture (leakage) of attached distribution system, general crimping or buckling of the equipment body, shell, or housing restricting the component from performing its function[4], cracking of battery jars and loose or broken electrical connections.
Rotating Equipment	Excessive noise, vibration, or temperatures in operating equipment.

Surveillance. Surveillance is that process whereby systems and components which are essential to plant nuclear safety during all modes of operation or which are necessary to prevent or mitigate the consequences of accidents are checked, tested, calibrated, and/or inspected as necessary to verify performance and availability.

Surveillance Tests. Those tests performed at regular intervals to demonstrate the availability and operability of components and systems. Surveillance tests are identified in the plant Technical Specifications and consist of checks, tests, calibrations and inspections to verify availability and performance of the tested component or system.

Time-History Recorder. An instrument capable of sensing and recording acceleration versus time. The resulting recorded time-histories may be stored locally and/or transmitted to other storage devices for processing and permanent storage. The components of the time-history recorder (acceleration sensor, recorder, seismic trigger) may be assembled in a self-contained unit or may be separately located.

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[3] Bent or deformed supports so long as they are capable of performing their design function are not considered significant.

[4] Scratches and localized denting of the equipment body or housing are not considered significant.

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Section 1: Introduction

1.1 Purpose

This report is an update of EPRI Report 3002000720, "Guidelines for Nuclear Plant Response to an Earthquake"[1][5] that was issued in October, 2013. The October, 2013 EPRI report is a major re-write of EPRI Report NP-6695[2] that was originally issued in 1989 to provide guidelines for the actions to be taken in preparation for, and following, a felt earthquake at a nuclear power plant. These early guidelines were subsequently accepted conditionally by the USNRC in Regulatory Guides 1.166[3] and 1.167[4].

The EPRI 3002000720 re-write of the NP-6695 report was based primarily on the experience gained and lessons learned in partial applications of the guidelines to several nuclear power plants world-wide that have experienced relatively strong earthquakes in the years since issuance of the NP-6695 report. Many of these earthquakes exceeded the plants' design Safe Shutdown Earthquake (SSE) levels. As a result, extensive inspections, tests and analyses were performed to assess the effects of these earthquakes on typical nuclear power plant SSCs in the affected plants. The results of these evaluations have also been included in this update of the 3002000720 report. The resulting main changes and additions in the guidelines since publication of the NP-6695 report are as follows and are included in this report:

- The levels of felt earthquakes that are addressed have been increased to include those earthquakes that exceed the plant's design SSE.
- The scope of SSCs covered in recommended response actions has been expanded to include reactor plant systems, internals and fuel, and both high-frequency- and low-frequency-sensitive devices and components.
- The felt earthquake's damage potential has been re-defined in prescribed Damage Levels (DLs) that are used to define the recommended actions to assess plant damage and readiness for restart. The measured earthquake ground acceleration level, referred to as Earthquake Level (EL), is also considered in assigning recommended responses to the earthquake. The resulting Action Levels (ALs) are presented in a new matrix form.

[5] Numbers in brackets refer to References given in Section 7.

- A new and important section has been added to cover necessary pre-earthquake preparations and planning, the lack of which has hampered the progress of restart actions in some instances.
- Guidance has been included for cases where the SSE is exceeded to address the need for reevaluation of the plant's seismic hazard and impact on previously performed Seismic Margin Assessments (SMAs) and/or Seismic Probabilistic Safety Analyses (SPSAs). The need for a comprehensive long-term seismic implementation plan in this case that outlines how the new, larger earthquake will be considered in future plant modifications and replacements has also been addressed.
- Consideration has been given to guidance developed by the International Atomic Energy Agency (IAEA)[6] in recently published guidelines for earthquake response.

This current update of EPRI Report 3002000720 also incorporates results of industry and USNRC reviews of a pending re-issue of ANS Standard 2.23-2002[5] on this subject that was prepared by an ANS Working Group chaired by EPRI. .

The intent of the guidelines incorporated herein remains the same as in the predecessor reports – namely, to assist nuclear plant personnel in the preparation of detailed plant-specific earthquake response procedures. The objectives of the earthquake response procedures are to determine:

- The effects of the earthquake on the physical condition of the nuclear power plant,
- If shutdown of the plant is warranted, based on observed damage to the plant or because the OBE has been exceeded,
- The readiness of the plant to shut down, if shutdown is required due to an earthquake, and
- The readiness of the plant to resume operation from a shutdown condition following an earthquake.

The guidelines provided in this report do not cover those operator actions performed in connection with the operation and control of the nuclear power plant following an earthquake. These actions are specified in plant-specific Operating Procedures, Emergency Operating Procedures, Alarm Response Procedures and other conditions of the plant's license and are not within the scope of this report.

1.2 Need for Guidelines and Plant-Specific Earthquake Response Procedures

Reviews have been performed of existing earthquake response procedures from a number of US and foreign nuclear power plants, including several that have experienced strong motion earthquakes. The procedures for most plants were found to be general in nature. They typically require that the nuclear power plant be shut down in the event that an earthquake occurs which exceeds the OBE, as required by USNRC regulations, but in many cases, the criterion for determining if the OBE has been exceeded is not clearly defined. Because of this, an OBE exceedance criterion has since been defined and accepted by the USNRC; it is described in Section 2.2 and in detail in Section 3.4.1.

A further consideration is that many plants do not have procedures that define 1) pre-earthquake preparations such as the selection of SSCs that should be inspected and tested following a significant earthquake and the essential pre-earthquake inspections of these SSCs that need to be performed and documented to provide the base-line for future post-earthquake evaluations, 2) the short-term actions required to assess damage and to process and evaluate seismic recordings, and 3) the post-shutdown actions which are appropriate to determine the readiness of the plant to re-start. The absence of clear, detailed, and graded procedures for nuclear plant response to an earthquake, and their implementation, may not only result in unnecessary shutdown, their absence can and has resulted in unnecessary inspections, tests and analyses of important plant SSCs and extensive delays in plant restart. Specific examples are cited in the IAEA's recent guidelines[6] for nuclear plant response to an earthquake. A summary of experience for the most recent earthquakes at nuclear power plants is given below. None of these plants had comprehensive earthquake response procedures such as those described herein, or had implemented pre-earthquake preparations such as recommended in this report.

- Onagawa Plants, Japan, 2005 – Base mat accelerations exceeded SSE ground motion. No damage to safety-related (SR) SSCs. Time to restart – 5 to 7 months for three units. [6]
- Shika Plant, Japan, 2007 – In-structure response spectra (ISRS) exceeded SSE-based ISRS. No damage to SR SSCs. Time to restart – 1 year. [6]
- Kashiwazaki-Kariwa Plants, Japan, 2007 – All ground spectra exceeded SSE, ISRS significantly exceeded ISRS for SSE. No damage to SR SSCs. Time to restart – 22 to 40 months for seven units. [6]
- North Anna Plants, VA, USA, 2011 – Base mat spectra exceeded SSE above and below 10 Hz. No damage to SR SSCs. Time to restart - 2 to 3 months for two units. [Information provided by Dominion Energy]

It is likely that implementation of pre-approved response procedures with defined pre-earthquake preparations and graded action levels could have saved many months of down-time for these plants.

An outline of a plant procedure for response to an earthquake is given in Table 2-1.

1.3 OBE Exceedance Criterion

Earthquake experience before and after issuance of the original NP-6695 report has shown that a plant's design seismic response spectra is not a meaningful measure of seismic capacity alone. As a result, an extensive study of experience in commercial plants that experienced strong-motion earthquakes was undertaken by EPRI to determine the earthquake parameter that best correlated with observed damage in non-nuclear, commercial plants (that is, plants with little or no seismic design and with commercial construction, the types of plants for which this type of experience data were available). The results of this study showed that the best indicator of damage potential is a parameter computed from measured time-history records called the standardized Cumulative Absolute Velocity, or CAV. The background and definition of the CAV are given in Appendix A. Based on this work, a recommended criterion for determining OBE exceedance was developed and accepted by the USNRC[3]. This industry/USNRC consensus OBE exceedance criterion is described in Section 3.4.1.

1.4 Need for Seismic Instrumentation

Seismic instrumentation and data acquisition systems capable of recording and saving ground motion acceleration time histories of significant earthquakes are required to fully implement the guidelines in this report. Software to compute seismic response spectra and the CAV within 4 hours (preferably in real-time) is also required. Considerations in specifying requirements for this system are provided in Appendix B. It is recommended that this seismic instrumentation system be described in the plant response procedure together with procedures for operation, data recovery and storage, system calibration and maintenance. Additional guidance on the use of seismic instrumentation for determining exceedance of the OBE is given in Section 2.3.

Conditions may exist following an earthquake that require plant shutdown that could result in loss of critical lifeline functions in the local service area. Such conditions could include:

- Extreme cold or hot weather
- General power blackout in the service area
- Rescue operations
- Emergency services (e.g, fire, medical, civil defense, etc.)

While the need for power generation does not take precedence over plant Operating and Emergency Procedures or the requirements of the Operating License, the need for power in the service area should be considered. It is recommended that plant-specific earthquake response procedures clearly define 1) plant licensing requirements and conditions for a controlled plant shutdown and 2) the responsibilities for making the ultimate shutdown decision, including the need for consultation with the USNRC.

1.5 Organization of Report

This report is organized such that the report sections follow in the same sequence as the recommended actions; namely:

- Section 2 provides guidance on pre-earthquake planning activities, including preparation of a plant-specific response procedure that covers selection of a broad scope of SR SSCs that are representative of essentially all types of structures, electrical and mechanical equipment, piping and commodities whose function is required during and/or after an earthquake. Typically, this list of SSCs would include a "smart" sample of each category of SSC that is considered most vulnerable to seismic motions. This list of SSCs will require pre-earthquake baseline inspection and will be used to prioritize post-shutdown inspections and tests.
- Section 3 discusses recommended short-term actions of plant operations personnel and others to make the evaluations necessary to support a decision to shut down the plant following occurrence of a felt earthquake. These include an immediate plant walkdown by plant operators and the assimilation and evaluation of earthquake recordings.
- Section 4 presents definitions of damage levels (DLs) that serve to grade the level of actions required to assess the readiness of the plant to restart. The assessment of the DL applicable to the felt earthquake is to be based on inspections and tests by operators and experienced seismic engineers. Measured earthquake levels (ELs) relative to design OBE and SSE exceedance criteria levels are also defined and considered in this process. The resulting Action Levels (ALs) are presented in matrix form (Table 4-1) based on the observed damage levels and measured earthquake levels.
- Section 5 provides detailed guidance on the various post-earthquake inspections and tests that are recommended in the Action Levels presented in the Section 4 Action Level Matrix.
- Section 6 discusses longer term actions, primarily seismic analyses of selected SSCs to confirm the long-term functionality of any SR SSCs or families of SSC types that have shown evidence of damage and/or significant exceedance of their design SSE exceedance criterion level. A key determination discussed in this section is whether these long-term evaluations need to be completed before plant restart, or can be completed after plant restart. Actions recommended to re-assess the seismic qualification status of active equipment normally qualified seismically by test are also provided.
- Section 7 is a list of references.
- Appendix A is a description of the basis and use of the CAV.
- Appendix B describes considerations in specifying seismic measurement and data acquisition systems.

Section 2: Pre-Earthquake Preparations and Planning

Over the past several years there have been a number of significant earthquakes at nuclear power plants in which design earthquake levels have been exceeded and yet no significant damage has occurred to SR SSCs. Despite these observations, most of these plants have spent many months performing extensive analyses and tests to confirm the functionality of important SSCs prior to restart. In each of these cases, the existence and implementation of a clear and thorough response procedure in advance could have eliminated a significant amount of unnecessary time and effort. As a consequence, pre-earthquake preparations involved in developing and implementing such a procedure are key elements in an efficient, cost-effective earthquake response plan.

The primary elements of such a procedure are described in the initial issue of NP-6695[2] and expanded in the more recent IAEA guidelines[6]. They include the following main subjects:

2.1 Responsibilities

Responsibilities of utility management, plant management and operations personnel, and experienced engineering representatives should be defined.

2.2 Definition of Plant Shutdown Criteria

Since 1973, USNRC regulations for nuclear power plants require that plants shut down (or remain shut down) if a felt earthquake exceeds the plant's OBE, although the specific ground motion and other response parameters that define the OBE (and therefore exceedance of the OBE) may not be specified. These would normally be defined in plant-specific licensing documents, but for some US plants, depending on their vintage, this may not be the case. In response to this situation, the US nuclear industry, with consultation and input from the USNRC, developed a standardized OBE exceedance criterion[7 and 8] that has been conditionally accepted by the USNRC in Regulatory Guide 1.166[3] and is considered acceptable for all US licensed nuclear power plants. This "consensus" OBE exceedance criterion (and also the SSE exceedance criterion used herein to define response actions) includes a response spectrum check and the calculation and evaluation of the parameter Cumulative Absolute Velocity (CAV). The OBE and SSE exceedance criteria are described in detail in Section 3.4. They are the criteria that are intended to be used in applying this report for plant shutdown decisions and for subsequent response actions. However, since all US plants are obligated to comply with their plant-specific licensing commitments, plant licensees that elect to implement this report should review their plant-specific licensing requirements to determine if clarifications and/or amendments to their license are required to apply the OBE exceedance criteria described herein for the purpose of complying with applicable USNRC requirements for plant shutdown following a felt earthquake. For example, it may be necessary for some plants to formally adopt the consensus OBE exceedance criterion described in this report as part of their licensing documents. Guidance on this subject is provided in EPRI report 1024889, "*Seismic Instrumentation at Nuclear Power Plants*"[9] and USNRC Final Policy Statement on Technical Specifications for Nuclear Power Reactors[10].

2.3 Seismic Instrumentation

USNRC regulations for nuclear power plants require that suitable instrumentation be provided so that the seismic response of nuclear power plant features important to safety can be evaluated promptly after an earthquake. Seismic instrumentation and data acquisition systems capable of recording and saving acceleration time histories of significant earthquakes are required to fully implement the requirements of this report. Software to compute seismic response spectra and the damage parameter Standardized CAV is also required, as described in Section 3.4 and Appendix A. The system should be capable of computing these parameters within four hours of the earthquake. Guidance for new applications of seismic instrumentation and data acquisition systems that reflect the current state-of-practice is contained in Appendix B and in EPRI report 1024889[9].

Free-field measurements are required for determining the CAV, unless otherwise justified. Free-field measurements are also intended for calculating the observed earthquake's seismic response spectra to be compared to plant design OBE and SSE seismic response spectra. However, if a plant's design OBE and SSE seismic response spectra are defined at a location (control point) other than in the free-field (e.g., at a location on the plant structures such as the containment base mat), seismic instrumentation should also be installed at or near this location unless otherwise justified by engineering evaluation. It is important and is the intent of these guidelines that the location(s) of the seismic instrumentation employed to generate the seismic response specra of the felt earthquake used to determine the need for plant shutdown be consistent with the location(s) at which the plant design OBE and SSE response spectra are defined.

The installation of seismic monitoring instrumentation at other locations within the plant may also be advantageous in post-earthquake evaluations of the effects of the earthquake on installed SSCs throughout the plant. EPRI report 1024889[9] provides guidance that could be used to voluntarily upgrade existing or install additional seismic instrumentation (e.g., digital time-history recorders) in operating nuclear power plants.

2.4 Pre-Earthquake Planning

2.4.1 Selection of Scope of SSCs to be Evaluated

Under the approach recommended herein for responding to a significant earthquake, it is important to pre-select and document a cross-section of nuclear plant equipment (mechanical and electrical, including distribution systems; e.g., piping, raceways, ducts) and structures to be inspected in the event that an earthquake occurs which requires shutdown. The items selected for pre-shutdown baseline inspections and post-shutdown inspections should be representative of SR and non-SR SSCs important to safe plant operation. Section 5 of this report lists the classes of SSCs that should be considered in the pre-selected sample of items for baseline and subsequent "focused" inspections. Examples of the types of damage that should be covered in these inspections are given in Table 5-1. The items selected should also include typical non-SR equipment which experience has shown to be of low seismic capacity to serve as earthquake damage indicators (e.g., architectural features, fragile switchyard equipment such as ceramic insulators, etc.).

The pre-selected SSCs are intended to represent a "smart" sample of SSCs in each of the classes of SSCs to be covered in the initial post-earthquake "focused" inspections in order to provide a broad indication of plant-wide damage. For example, the items pre-selected for the focused inspections should include a representative sample of the items within each equipment class or structure, and should include those items which are considered most likely to be damaged by an earthquake (e.g., items located on higher floors of the building, flat bottomed vertical tanks, etc.). This "smart" sample of SSCs is intended to provide conservative surrogates for all classes of important SSEs in assessing the overall level of damage caused by a damaging earthquake. Where the nuclear power plant contains only a small number of items within a particular equipment class or structure (e.g., one or two items), all such items should be inspected. However, where the nuclear power plant contains a large number of potentially vulnerable items within a particular equipment class, the inspections should be performed on a sampling basis. Experienced engineering judgment should be used in developing a reasonable, conservative sample. For civil structures (steel and concrete}, the scope of the inspections should include all SR structures, but should focus on a representative sample of the construction details described in Items 7 and 8 of Table 5-l. For steel structures this would include bolted connections, anchor bolts, and lateral bracing. For concrete structures, this would include representative areas of concrete structures which are considered susceptible to damage. In the final analysis, the size of the sample is to be decided by the utility. 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.

2.4.2 Baseline Inspections

Baseline visual inspections of all equipment and structures pre-selected for post-shutdown focused inspections, should be performed and documented in written reports that include sketches and photographs of abnormalities, as appropriate.

The purpose of the baseline inspections is to identify and document any pre-existing conditions (e.g., cracks in concrete structures, slight shifting of components on their foundations) in order to provide a basis for differentiating earthquake- related damage from pre-existing abnormal conditions during subsequent post-shutdown inspections. It is recommended that periodic inspections of the items selected for post-shutdown inspections also be performed to identify and document any changes in the condition of the pre-selected items during normal operation. As an example, an inspection interval for Maintenance Rule activities is typically 5 years.

It should be noted that many SR SSCs that are identified for post-earthquake inspection are part of formal plant periodic inspection and/or surveillance programs (e.g., ASME Section XI in-service inspection and test programs, snubber surveillance programs, BWR piping and internals IGSCC monitoring programs, Maintenance Rule inspections, etc.). Where the condition of these SSCs are already tracked and documented and include the appropriate information, they would not need to be included in the pre-earthquake baseline inspections, but the information should be referenced and available.

2.5 Response Action Plan

The main part of the earthquake response procedure is the delineation of required post-earthquake actions. These actions include short-term actions by operators and plant staff to determine the need for plant shutdown (including the case when the plant is tripped as a result of the earthquake), the steps required to identify and classify the level of damage incurred, if any, the level of the felt earthquake as measured by plant instrumentation and the action levels that are appropriate for the observed damage and earthquake levels.

A suggested outline of a procedure for plant response to an earthquake is given in Table 2-1.

2.6 Seismic Design Basis Records

Because a significant earthquake may require comparison of observed seismic loads with the original design basis loads used to qualify important SSCs, it is recommended that a data base of design basis seismic analyses and qualification tests be gathered, reviewed for completeness and made readily available for comparative analyses of design and observed seismic motions and loads. This should include analytical models, where available. The results of current documented base-line inspections discussed in Section 2.4.2, above, should also be included in this data package.

Table 2-1
Outline of Plant-Specific Earthquake Response Procedure

PURPOSE

To provide guidance to nuclear plant owner/operators on preparations, responsibilities, and response to an earthquake. In particular,

- Need to shut down plant
- Preparation for an orderly shutdown
- Assessment of readiness for restart

PREPARATIONS/PREREQUISITES

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

- Plant seismic instrumentation to implement OBE Exceedance Criterion (See Section 3.4 and Appendix A) or alternative actions if such instrumentation is not installed
- Method/procedure for processing records from seismic instruments in a timely manner (within about 4 hours). For plants that will utilize the special considerations described in Section 3.4.3, procedures for performing any required calculations should be described in the plant's pre-earthquake preparations and planning
- Pre-selected sample of structures and equipment to be inspected after an earthquake
- Baseline inspection results for above structures and equipment
- Plant OBE and SSE design basis and reference information

RESPONSIBILITIES

<u>Plant Operations</u>

- Confirmation of felt earthquake
- Stabilization of plant per normal and emergency operating procedures
- Implementation of earthquake response procedure
- Plant walkdown inspection
- Determination of OBE Exceedance
- Pre-shutdown evaluation
- Plant shutdown
- Prescribed surveillance tests
- Operability evaluations (in conjunction with engineering)
- Plant Restart

<u>Engineers with Earthquake-Related Experience</u>

- Detailed inspections of pre-selected equipment/structures
- Determination of earthquake Damage Levels, Earthquake Levels and appropriate Action Levels
- Performance of focused and expanded inspections; specification of tests
- Evaluation of results of inspections and tests, including root cause assessments and operability evaluations
- Long-term confirmatory evaluations

Table 2-1 (continued)
Outline of Plant-Specific Earthquake Response Procedure

ACTION INITIATORS Earthquake Response ▪ Activation of seismic instruments, or ▪ Consensus of operators that earthquake has occurred Plant Shutdown (or Remain Shutdown) Decision ▪ OBE Exceedance ▪ Physical damage to plant Readiness for Restart ▪ Implementation of recommended Action Level(s) ▪ Physical condition of plant ▪ Demonstrated functionality of equipment Long-Term Plant Integrity ▪ Confirmatory, long-term evaluations ▪ Supplemental functional tests, inspections, and non-destructive examinations
RECOMMENDED ACTIONS Short-Term Actions ▪ Safe, stable operation ▪ Implementation of earthquake response procedure ▪ Operator walkdown inspections ▪ Processing and evaluation of seismic records ▪ Shutdown decision ▪ Pre-shutdown checks (if warranted) ▪ Orderly shutdown (if required) Post-Shutdown Actions ▪ Visual inspections of pre-selected sample of equipment ▪ Determination of Damage Level, Earthquake Level and recommended Action Level ▪ Focused visual inspections and tests ▪ Expanded visual inspections and tests ▪ Specific surveillance and other tests to verify equipment and system functionality ▪ Restart Long-Term Evaluations ▪ Obtain in-structure response spectra (ISRS) for actual earthquake ▪ Comparison with SSE design ISRS ▪ Specific evaluation where SSE design loads may have been exceeded ▪ Evaluate need for re-assessment of site seismic hazard and related plant evaluations ▪ Development of long-term seismic implementation plan, when required.

Section 3: Short-Term Actions

This section provides guidelines for short-term actions to determine the immediate effects of an earthquake on a nuclear power plant, and to determine if the OBE has been exceeded. If it is determined that shutdown of the plant is required based on observed damage to nuclear plant SSCs, or that the OBE has been exceeded, then a normal shutdown of the nuclear power plant for inspections and tests prior to a return to power is necessary. Guidelines for visual inspections and tests of essential safe shutdown equipment prior to initiation of shutdown activities are also provided for the case where the plant is not shut down by the earthquake.

Guidelines for determining appropriate post-shutdown responses to the earthquake based on its damage potential and severity are given in Section 4. These actions include graded inspections and tests as well as longer term evaluations. Detailed descriptions of specific post-shutdown inspections and tests to determine the readiness of the plant to resume operation are provided in Section 5 of this report. Guidelines for evaluations of the effects of the earthquake on the long-term functionality of essential SR and non-SR SSCs are provided in Section 6.

It is anticipated that the short-term actions described in this section would be completed by plant operators and other on-site personnel within about eight hours after the earthquake. These actions are in addition to the operator actions that would be taken in response to a plant upset such as an earthquake in accordance with existing plant Operating Procedures, Emergency Operating Procedures, Alarm Response Procedures, the Emergency Plan, etc.

Short-term actions recommended in response to an earthquake include the following:

- Immediate operator actions to control the plant and to identify any abnormalities suspected to have been caused by the earthquake, including concomitant events such as earthquake-caused flooding, tsunamis, fire, off-site and on-site power failures, etc.
- Operator walkdown inspections of accessible areas of the nuclear power plant
- Evaluation of ground motion records from installed seismic instruments, and determination of whether or not the ground motion exceeded the OBE exceedance criterion
- Pre-shutdown inspections of essential safe shutdown equipment (to be performed prior to normal shutdown if normal shutdown is required)

The purpose of these actions is: (1) to obtain a preliminary assessment of the effect of the earthquake on the physical condition of nuclear plant equipment and structures, and (2) to determine if shutdown of the plant is required based on observed damage to nuclear plant SSCs, or because the OBE exceedance criterion has been exceeded.

3.1 Immediate Operator Actions

It can be expected that a felt earthquake with sufficient size to cause operating system upset and/or damage will result in alarms and/or changes in plant parameters which will require control room operators to respond to plant alarms and other immediate effects of the earthquake in accordance with approved plant operating and emergency operating procedures. Operator response to maintain the safe, stable condition of the plant would take precedence over the inspections and tests proposed herein. As part of the operator response to the earthquake, it is recommended that the following specific control room board checks be made:

- Primary coolant and secondary system radiation, temperature, pressure, and flow parameters for changes and excursions coincident with the earthquake. This includes sampling and analysis of primary and secondary coolant.
- Primary coolant loose parts monitoring system for changes in noise levels/signatures.
- Control and/or instrumentation trips/upsets to SR and non-SR SSCs, and any evidence of equipment and system malfunctions.
- Spurious relay actuations.
- Rotating equipment vibration monitoring sensors for changes.
- Indications of fluid levels in important low pressure storage tanks.

In addition to these checks, pre-planned operator walkdown inspections should be made following any felt earthquake as described below.

3.2 Operator Walkdown Inspections

If a felt earthquake, as described in the definitions above, occurs, 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 equipment and structures and the areas being inspected. High radiation areas, the primary containment 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 these operator walkdown inspections is to determine the effects of the earthquake on the physical condition of nuclear plant equipment and structures. Control room instrumentation and alarms provide additional information on the status and performance of components and systems. Together, they provide plant operators with information needed to determine if the plant should be shut down for additional inspections and evaluations or can continue to operate (or restart if automatically tripped as a result of the earthquake), assuming the OBE exceedance criterion is not exceeded. If the OBE exceedance criterion is exceeded and/or any significant damage to SR and non-SR SSCs important to safe plant operation is observed, an orderly plant shutdown is required for US plants. (Incidental upsets of non-power plant architectural features such as bookcases and other unanchored cabinets, etc. , would not be considered as significant). The results of the operator walkdowns also provide a basis for establishing a preliminary Damage Level of the plant, as described in Section 4 of this report.

It is considered important that the operator walkdown inspections be performed by plant operators who are familiar with the equipment to be inspected. These persons are considered to be the most likely to know if the condition of equipment and structures (e.g., physical appearance, leak rates, vibration levels, sound of motors, etc.) has changed from its condition before the earthquake. Plant operators may be assisted in these inspections by available on-site personnel (e.g., engineering, maintenance, quality control, etc.). The inspections should be similar to those performed by plant operators during their normal daily rounds, with additional emphasis on visual inspections for evidence of earthquake-related damage. In general, the visual inspections should include the following in addition to those inspections performed during normal operator rounds. Specific guidance for the operators, based on the following, should be included in the plant-specific response procedure.

- Check for leaks in piping systems, especially at flanged or threaded connections and branch lines.
- Check for damage to low pressure tanks, particularly ground or floor mounted vertical storage tanks.
- Check for damage to switchyard equipment.
- Check of fluid levels in tanks. Level switches may have been activated due to sloshing of the contained fluid (an actual but momentary change in level).
- Check for high vibration, high bearing temperature, and unusual noise in rotating equipment such as pumps and fans.
- Check for damage to equipment and structures due to impact with adjacent equipment and falling objects.
- Check of the condition of a sampling of equipment anchorages including deformation or loosening of anchor bolts, pullout or shear of anchor bolts, rocking, sliding, or misalignment of equipment.
- Check for damage to attached piping including hoses, tubing, and electrical conduit.
- Check for damage to piping, and check of piping and component supports for evidence of excessive displacement or permanent deformation.
- Check for distortion of electrical and control cabinets including a brief visual check of a sampling of internally mounted components such as relays and circuit breakers.

- Check for major cracks or spalling in reinforced concrete structures. Hairline cracks in reinforced concrete structures are not considered significant.
- Check of the operational status of important relays, breakers, and other potentially sensitive electric gear (in particular, those in protective and seal-in/lockout circuits whose change in state could affect operability of equipment and systems).
- Check for portable equipment which may have fallen on safe shutdown equipment.
- Check for signs of obvious settlement of foundations of structures.
- Check for loose electrical connections
- Check for leaks/cracks in station batteries.

In performing these inspections, consideration should be given to the specific list of equipment pre-selected for focused inspections and described in Section 2 of this report. If there are any areas of concern in the minds of plant operators, then additional engineering assessments should be performed during plant operation (assuming the plant is not shut down automatically by the earthquake). Guidance on what is considered to be significant damage is given in the Definitions section of this report.

It is anticipated that the operator walkdown inspections discussed in this section of the report could be performed within about eight hours depending on the number of personnel conducting the inspections. (If it appears that more than about 8 hours will be required to complete these walkdown inspections and to evaluate seismic ground motion recordings as required to make a plant shutdown decision, it is expected that plant personnel will confer with the USNRC. Results of the operator walkdown inspections following the felt earthquake should be documented.

Operator walkdown inspections should be performed under the conditions discussed above, even if the plant automatically shuts down as a result of the earthquake, to determine if the additional post-shutdown inspections and tests described in subsequent sections of this report are needed prior to restart of the plant.

3.3 Evaluation of Ground Motion Records

Should a felt earthquake occur, available seismic ground and structure motion records should be gathered, processed, and evaluated in parallel with operator walkdown inspections to determine if the OBE and SSE have been exceeded. Procedures should be established for removing and storing the records from each seismic instrument. All data should be identifiable and traceable with respect to the date and time of collection, and the location and orientation of the instrument (sensor) from which the record was collected.

3.3.1 Procedure to Determine if the OBE has been Exceeded

The following procedure should be followed to determine if the OBE has been exceeded:

- Gather and process the records from the installed seismic instruments to determine the pertinent ground and structure motion parameters (i.e., acceleration and velocity response spectra, Standardized CAV, and peak ground motion parameters required by the OBE exceedance criterion). The calibration standards, computer software, record analyzers, etc., required to process the records from the seismic instruments should be on hand at the site or available remotely so that the records can be processed within a time period of four hours following the earthquake (a specific procedure may be required to do this).
- The evaluation may be performed on uncorrected earthquake records. It was found in a study of uncorrected versus corrected earthquake records that the use of uncorrected records is conservative.
- Compare the computed ground/structure motion parameters with limit values specified in the OBE exceedance criterion (Section 3.4.1). If the computed values exceed the limit values, then the OBE has been exceeded and the nuclear power plant shall be shut down for additional inspections and tests, consistent with USNRC regulations and plant-specific commitments as discussed in Section 2.2.

The determination of whether the OBE has been exceeded should be performed in accordance with the above procedure even if the plant automatically trips off-line as a result of the earthquake or is in a normal shutdown condition during the earthquake. Determination of an OBE exceedance or significant damage would require the initiation of the post-earthquake evaluations provided in Section 4.

3.3.2 Procedure to Determine if the SSE Has Been Exceeded

The procedure to determine if the SSE has been exceeded is the same as the OBE exceedance procedure with the following exceptions:

- The records do not have to be processed within a time period of four hours. The determination of SSE exceedance is needed to support the post-earthquake evaluations prescribed in Sections 4 through 6.
- Compare the computed ground/structure motion parameters with limit values specified in the SSE exceedance criterion (Section 3.4.2). If the computed values exceed the limit values, then the SSE has been exceeded.

3.4 OBE and SSE Exceedance Criteria

3.4.1 OBE Exceedance Criterion

The OBE shall be considered to have been exceeded if:

- Response Spectrum Check: 1) The 5% damped acceleration response spectrum for any directional component (two horizontal and one vertical) of the earthquake motion at the site at frequencies between 2 and 10 Hz exceeds the corresponding OBE design response spectrum or 0.20 g, whichever is greater, or 2) the corresponding OBE design spectral velocity or a spectral velocity of six inches per second, whichever is greater, is exceeded between 1 and 2 Hz, **AND**
- CAV Check: The computed Standardized CAV value from any component of the free-field earthquake record is greater than 0.16 g-sec. (See 3.4.3, first bullet).

For each directional component of the free-field ground motion, the CAV shall be calculated as follows:

- For each acceleration component time-history, the absolute acceleration (g units) time-history is divided into 1-second intervals.
- For each acceleration component time-history, each 1-second interval that has at least one exceedance of 0.025 g is integrated over time.
- For each acceleration component time-history, all the integrated values are summed together to arrive at the CAV.

3.4.2 SSE Exceedance Criterion

The SSE shall be considered to have been exceeded if:

- Response Spectrum Check: The 5% damped acceleration response spectrum for any directional component (two horizontal and one vertical) of the earthquake motion at the site at frequencies between 2 and 10 Hz exceeds the corresponding SSE design response spectrum or 0.20 g, whichever is greater, **AND**
- CAV Check: The computed Standardized CAV value from any component of the free-field earthquake record is greater than 0.16 g-sec. (See 3.4.3, first bullet). The procedure to calculate the CAV is given in Section 3.4.1, above.

Note that the threshold CAV value used in the SSE exceedance criterion is the same as for the OBE case. This conservative value is selected because a higher value more consistent with an SSE design level higher than the OBE was not determined in the referenced studies.

It should be noted that there are two important types of earthquake exceedances not included in the above definitions that require special consideration. These are observed earthquakes whose ground motion response spectra exceed the design response spectra only above 10 Hz or only below 2 Hz. Earthquake ground motion at the site with exceedances in these frequency ranges have little impact on most power plant SSCs, but can be important for specific components that are sensitive to high-frequency or low-frequency excitations. These special cases are discussed in Section 4.3.

3.4.3 Special Considerations – Definitions of Design OBE and SSE Response Spectra

The previous discussions of measured earthquake parameters indicate that such measurements and the calculated value of the CAV are to be based on free-field records at appropriate locations near the plant and also at other locations in those cases where the plant-specific design OBE and SSE response spectra are defined at locations not in the free-field. It is the intent of this standard that the comparison of plant design and measured/calculated OBE and SSE response spectra be performed at equivalent locations. Specific guidelines are as follows:

- Because the threshold value of the CAV is based on correlations of plant damage with ground motion accelerations, the recorded measurements used in computing the CAV should be based on free-field instrumentation. If free-field measurements are not available, the limiting, threshold value of the CAV should be assumed to have been exceeded, unless otherwise justified.
- Free-field measurements should be used for all cases where the plant-specific design OBE and SSE response spectra are defined in the free-field and/or used as free-field for plant analyses and design.
- In those cases where the design OBE/SSE response spectra input motions are defined as input motions at plant structures' foundations and/or were used as such for plant analyses and design, the recorded measurements at these structure locations can be used to determine OBE/SSE spectral exceedance. (An example would the containment base mat). However, in these cases, it is also considered acceptable to compare the spectra measured by free-field instruments (if available) to the OBE/SSE spectra that may be developed at the location of the free-field instrumentation considering the soil/rock characteristics.
- In those cases where the design OBE/SSE response spectra input motions are defined at other locations not in the free-field or on the structure foundation (e.g., at a rock outcrop), the available recorded motions can be used to calculate the resulting response spectra at the base of the structure foundation (e.g., the base mat), which would then be compared with the plant-specific response spectra calculated for the same location and used for plant analyses and design. In those special cases where it can be shown that soil-structure interaction (SSI) effects of the soil/rock between the foundation base (e.g., base mat) and the location of the design OBE/SSE input motions are not significant, measurements from instruments installed on the foundation base can be directly used for comparison to the OBE and SSE spectra.
- Some plants may have more than one design OBE/SSE in their licensing bases. An example is when some Seismic Category I structures at a plant are

designed to the Certified Seismic Design Response Spectrum (CSDRS) but others to a site-specific ground motion response spectrum (GMRS). In such cases, the response spectrum with the lowest spectral ordinates should be used in the OBE/SSE exceedance and response action determinations. It is noted that this consideration does not apply to situations when a plant has different design basis OBE/SSE spectra specified for rock founded structures vs. soil founded structures. In this case, the applicable design OBE/SSE spectra for comparison to the recorded measurements should correspond to site condition (i.e., rock or soil) where the seismic instrumentation is located. Other unique plant design basis conditions should be evaluated on a case-by-case basis.

For plants that will utilize these special considerations, procedures for performing any required calculations should be described in the plant's pre-earthquake preparations and planning.

3.5 Determination of Need for Plant Shutdown

If the operator walkdown inspections indicate no damage to the nuclear power plant which would require shutdown, and the evaluation of the earthquake motion records indicates that the OBE has not been exceeded, then shutdown of the plant is not required and the plant may continue to operate (or restart following a post-trip review, if it tripped off-line due to the earthquake), consistent with existing plant procedures, Technical Specifications and regulations, including the need for notifications and consultation with the USNRC.

If the OBE exceedance criterion has been exceeded or if significant damage is found during the operator walkdowns, the plant should be shut down in an orderly manner for further evaluations recommended in Section 4. These post-earthquake evaluations are described in detail in Sections 5 and 6. If the plant has already tripped under conditions that would warrant shutdown, it should remain shut down for the prescribed inspections, tests and other evaluations.

Damage to the plant that would require shutdown would include damage to SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable. In the event that significant damage is observed, operators, in collaboration with seismic engineers, should make a preliminary assessment of DL in accordance with Section 4.

3.6 Pre-Shutdown Inspections

If it is determined that shutdown of the plant is required based on the results of the above evaluations, then a normal, controlled shutdown of the plant is recommended, consistent with Plant Operating and Emergency Operating Procedures and the need to consult with the USNRC.

Prior to initiating plant shutdown following an earthquake, visual inspections and control board checks of safe shutdown systems should be performed by plant operations personnel, and the availability of off-site and emergency on-site power sources should be determined. 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 so that any resets or repairs required as a result of the earthquake can be performed, or alternate equipment can be readied, prior to initiating shutdown activities. In order to ascertain possible fuel and reactor internals damage, the following checks should be made, if possible, before plant shutdown is initiated:

- Check control rod drive mechanisms for operability.
- Check in-core instrumentation readouts for changes.
- Check primary coolant radiation monitors for changes.
- Check primary coolant flow, temperature, and pressure for changes.
- Check loose parts monitoring equipment for changes in noise signatures.
- Compare primary and secondary coolant sample chemistry with pre-earthquake samples.

In the event of a plant trip, all records pertaining to the items listed above should be compared to the data which is recorded during a normal shutdown and/or previous plant trips.

3.6.1 Safe Shutdown Equipment

Plant operators should identify and maintain a list of essential safe shutdown equipment to be included in the pre-shutdown inspections. The safe shutdown systems include those required to perform the following functions:

- Reactivity control
- Reactor coolant pressure control
- Reactor coolant inventory control
- Decay heat removal

In identifying safe shutdown equipment, it is assumed that a cold shutdown will be required for the post-shutdown inspections and tests described in Section 5 of this report. Both SR and normal shutdown equipment should be included (standby and running). Components and systems required only for accident mitigation may not need to be inspected as part of the pre-shutdown inspections. Equipment used for safe shutdown but which is also used during normal operation (e.g., service water system) need not be included. However, equipment used for shutdown but not used during normal operation (e.g., residual heat removal system) should be included in the inspections. Examples of equipment and systems that should be inspected include the following:

- Decay heat removal system, including pumps and exchangers
- Major sources of water (Ultimate heat sink)
- Borated water storage tank (PWRs only)

- Refueling water storage tank (PWRs only)
- Condensate storage tank
- Delivery systems
  - Makeup water system
  - Auxiliary feedwater system (PWRs)
- Station emergency electrical system, including the diesel generators, station batteries, AC and DC buses, and associated breakers and relays
- Instrumentation and control systems needed to regulate and monitor essential safe shutdown systems

The following approach is recommended for performing the pre-shutdown inspections and tests of essential safe shutdown equipment in the event that the decision is made to shut down a nuclear power plant following an earthquake.

Perform visual inspections of the equipment included on the pre-shutdown inspection list.

The pre-shutdown inspections should focus on functional damage to equipment that may impair the capability of the damaged item to perform its safe-shutdown function. Physical damage which does not affect equipment operability is not a major concern in these inspections. Equipment or systems 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 may be repaired or an alternate device or system may need to be placed in service prior to plant shutdown.

Some pieces of equipment may require resetting at the time of the inspections due to the earthquake (for example, relays and other switches may be tripped, due to chatter, or an isolation valve may have shut). In these situations, the appropriate plant procedures for resetting the equipment should be used.

3.6.2 Availability of Power Sources

The availability of off-site power following an earthquake may be disrupted 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 should be 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 should be checked.

The availability and stability of off-site power sources should be checked. Contact the power grid dispatcher and determine the status of the grid, switchyards, and sub-stations. Visually inspect in-coming power lines and switchyard components.

Determine the number of available off-site power sources. If less than two sources of off-site power are available, or the condition of the off-site power sources is uncertain, check availablitiy of required on-site power distribution systems, including the following:

1. Board checks should include verification that all circuit breakers and control power indicating lights on the power supply board are showing that conditions conform to normal operating procedure requirements.
2. Visually inspect the startup/auxiliary transformers and circuit breakers and the associated electrical distribution equipment. Specifically, check that transformer sudden pressure switches have not been actuated resulting in isolation of the startup transformers.

3.6.3 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 alternate power should be determined. Specifically:

1. Perform a visual inspection of the emergency diesel generators. Inspect the starting system, cooling system, fuel oil system, lubricating oil system, intake and exhaust structures, and electrical distribution system. Startup of emergency power sources (e.g., diesel generators) may be appropriate in the case of significant, potentially damaging earthquakes).
2. Perform a visual inspection of the station DC power system. The inspection should include a visual inspection to determine if the batteries are in their racks and upright. Checks of the batteries should be made to ensure the battery parameters, such as electrolyte level, voltage, and absence of ground fault indications, indicate availability.
3. Depending on the severity of the earthquake and the condition of the grid, perform 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.

If the above inspections verify availability of all required safe shutdown systems and power sources, then a normal, controlled shutdown of the plant is recommended. If the inspections indicate degradation of safe shutdown systems, actions should be taken in accordance with existing operating procedures and Technical Specifications, including consultation with the USNRC, prior to shutdown.

The above short-term actions that lead to the decision to shut down the plant are shown schematically in Figure 3-1. It is expected that these actions can be completed within about 8 hours.

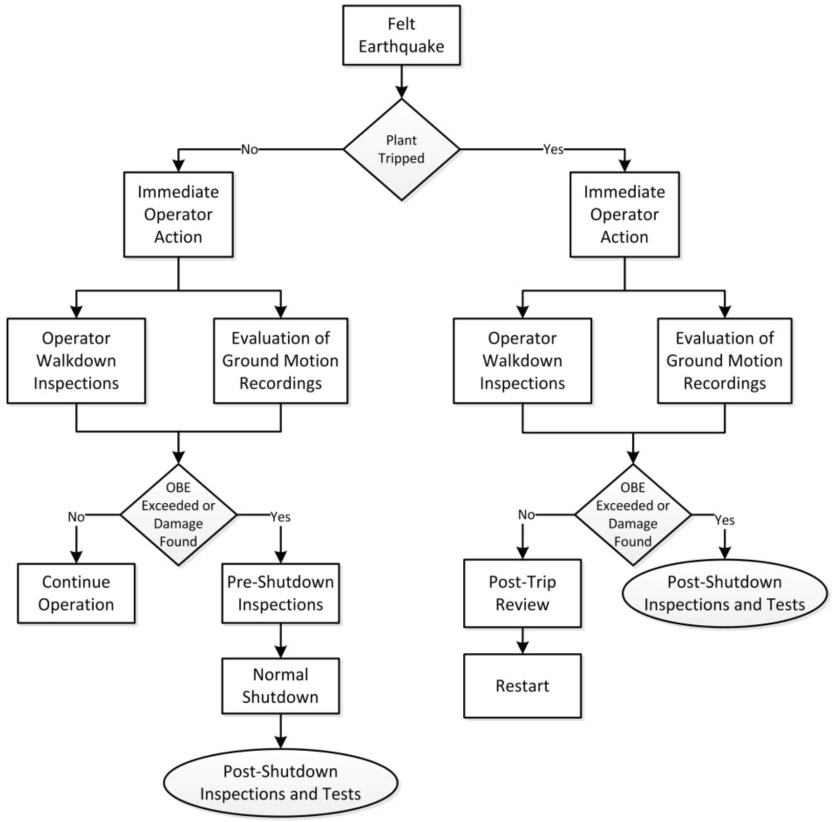


Figure 3-1
Flow Diagram of Short-Term Actions

Section 4: Earthquake Response Action Levels

The short-term actions described in Section 3 that are recommended in immediate response to a felt earthquake lead to a decision on the need for plant shutdown (or continued shutdown if the plant has tripped as a result of the earthquake) and for further focused and expanded inspections and tests intended to characterize the severity and damage potential of the earthquake. This section recommends those graded post-shutdown inspection, test and evaluation actions graded based on the observed damage level and size of the earthquake.

The post-earthquake actions recommended below are intended to provide a comprehensive and balanced response to a felt earthquake at a nuclear power plant. They are based primarily on the following premises and concepts enumerated in EPRI reports NP-6695[2] and 3002000720[1]:

- The plant itself, not damage information from nearby communities or recorded ground motion, is the best indicator of the severity of the earthquake at the plant site.
- Detailed inspections of pre-selected equipment and structures which are baseline inspected prior to the earthquake, together with the use of a defined seismic Damage Level scale for nuclear plant facilities, can be used to quantify the damage caused by the earthquake and to establish the extent of inspections, tests and evaluations necessary to demonstrate readiness for restart.
- Prescribed inspections and tests, keyed to both the level of observed damage, if any, and the level of the earthquake, can best demonstrate the integrity and functionality of SR SSCs.

Experience gained since the issuance of EPRI NP-6695 in the evaluation of nuclear plants' response to earthquakes whose measured seismic motions exceeded the plant design response spectra is discussed in Section 1, and has generally confirmed these premises. Accordingly, the post-earthquake actions that are recommended in this update report continue to be based primarily on results of inspections and tests of a pre-planned, baseline inspected set of SSCs that represent the various structural, mechanical and electrical items in the plant, and include a sample of the most vulnerable of these items to earthquake damage. The SSC types are graded in accordance with their importance to safety (i.e., SR vs. non-SR) and their location in the plant relative to the predicted seismic demand, as well as their seismic ruggedness based on past earthquake experience. The damage observed in each category serves as a pseudo-quantitative measure of the damage potential, or damage level (DL), of a felt earthquake. In addition, because the ruggedness of SR SSCs is related to the level of each nuclear power plant's design earthquake levels (OBE and SSE), the measured level of a felt earthquake (EL) at a given plant is also compared to the plant design levels (although the correlation of design earthquake exceedance with observed damage in past earthquakes has not been particularly meaningful).

The required actions to demonstrate restart readiness increase in scope and detail based on the extent of observed damage (DL) and the measured earthquake level (EL) relative to the design levels. The actions are categorized in six Action Levels (ALs) as shown in matrix form in Table 4-1. The specific actions included under each of the six ALs listed in this table are described below and in more detail in Sections 5 and 6. These actions are intended to provide a technical basis and framework to assist nuclear power plant licensees in meeting USNRC regulations that govern compliance with operability criteria and USNRC restart approval requirements following earthquakes that exceed a plant's OBE and requires shutdown of the plant.

4.1 Damage Levels

The five categories of damage levels are described below in order of increasing damage. It is intended that the overall Damage Level (DL) selected be based on the aggregate of plant inspection results when compared to the five damage levels described below. Individual indicators of observed damage do not immediately place the overall DL to a higher level, nor does an individual component's survivability warrant downgrading the DL.

- **DL 0** - Damage that is limited to a wide range of architectural type items that are relatively fragile, common to most industrial and non-industrial facilities (e.g., homes, offices, etc.), and have been shown to be good indicators of a low level of shaking. These items have no significant impact on the safety or operability of the plant. The items of equipment in this category are referred to as non-SR "damage indicators". Observed damage that is limited to these items is classified as DL 0. Examples include damage such as displacement of panels in wire hung suspended ceilings, some tipping, displacement and spilling of contents of book cases and storage containers, and some cracking of plaster and un-reinforced masonry walls in buildings built to commercial and/or residential standards such as office buildings, administration buildings and shops.
- **DL 1** - No damage to SR SSCs or non-SR SSCs important to safe plant operation. No damage to rugged, industrial-type non-SR SSCs. Damage to non-SR SSCs typically found in commercial, industrial and power plant facilities, but which have been shown to have relatively low seismic ruggedness. Examples of damage to this category of SSCs include wide-spread falling of panels in suspended ceilings, widespread cracking of windows, plaster, masonry and concrete structures not designed or built to commercial seismic standards. Some evidence of new piping insulation

deformation caused by interaction of non-seismically designed piping with nearby structural elements. Slight damage to low pressure storage tanks that does not limit their functionality (e.g., no significant leakage, limited shifting on foundations, limited anchor bolt inelastic deformation, limited buckling). Displacement of un-anchored equipment on its foundation. Tripping of vibration-sensing instrumentation. Damage to fragile switchyard components such as high voltage ceramics.

- **DL 2 –** No damage to SR SSCs. Damage to non-SR SSCs typically found in commercial, industrial and power plant facilities, and which have shown relatively high seismic ruggedness in past earthquakes. These would include SSCs designed and built to commercial seismic standards such as the Uniform Building Code (UBC) and the International Building Code (IBC). Examples of damage to this category of SSCs include wide-spread cracking in concrete and masonry structures, leakage of flanged and threaded joints and evidence of new insulation deformation in non-seismically designed piping. Permanent deformation of anchorages and walls of non-seismically designed low pressure storage tanks, including leakage that challenges the continued functionality of the tanks. Damage to less fragile switchyard components such as low voltage ceramics, air-blast circuit breakers and rail-mounted transformers.
- **DL 3 –** Isolated evidence of damage to SR SSCs in addition to the kinds of damage referred to in the lesser damage levels above. SSCs in this category include distribution systems (raceways and ductworks) and both seismically designed and non-seismically designed tanks and anchorages of some electrical equipment. Evidence of isolated and limited cracking in safety-related concrete walls and equipment foundations. More severe and widespread damage to non-seismically designed concrete, masonry construction. General over-turning of un-anchored equipment and storage containers.
- **DL 4 –** Clear evidence of permanent deformation, cracking and malfunction of SR equipment, piping, supports and structures in high demand locations. Severe damage and isolated collapse of non-seismically designed civil structures. Wide-spread damage to switchyard components and supports. General failures of low pressure storage tanks leading to loss of contents. Evidence of seismic interactions between distribution systems and nearby equipment and structures. Indications of reactor coolant leakage from leak detection alarm systems.

4.2 Earthquake Levels

The earthquake levels (ELs) shown in the matrix in Table 4-1 to which the felt earthquake is to be compared include three levels:

- EL 1 - Measured levels less than or equal to the plant-specific OBE exceedance criterion,
- EL 2 – Levels in excess of the OBE exceedance criterion but less than or equal to the SSE exceedance criterion, and
- EL 3 – Levels that are greater than the defined SSE exceedance criterion.

For the purpose of the AL definitions in 4.3, below, the design OBE exceedance and SSE exceedance levels to be compared with measured/calculated parameters of the observed earthquake are based on the industry/USNRC consensus exceedance criteria discussed in Section 3.4.

4.3 Recommended Post-Earthquake Action Levels

Recommended *post-shutdown* actions include 1) focused inspections of a pre-selected set of SSCs that are representative of a broad cross-section of equipment and structures in nuclear and conventional power plants, 2) expanded inspections if damage is found in the focused inspections, and 3) further graded inspections, tests and analyses that are guided by the damage level and earthquake level assigned based on the definitions given above. These action levels are identified by number in the action level matrix of Table 4-1, and are described below. As in the application of any generic guidelines to specific plants and circumstances, it is important that the guidelines be applied by experienced seismic engineers and plant operations personnel on a case-by-case basis. In particular, it should be recognized that the examples of SSC damage that are intended to define damage levels are to be interpreted as broad indicators of damage potential, *not* as hard and fast decision points based on isolated damage to individual items. Experienced judgment is required.

- Action Level 1
  1. <u>Perform Focused Inspections and Tests per Section 5.1</u>. If no damage is found and the EL is less than or equal to the SSE, plant is considered ready for restart/continued operation and no further post-earthquake actions are recommended. If damage is found, perform Expanded Inspections and Tests as prescribed in Step 2, below. If the EL is greater than the SSE, perform actions in Steps 4 and 5.
  2. <u>Perform Expanded Inspections and Tests per Section 5.2</u>. If no additional damage is found, repair/replace non-SR SSCs necessary for safe plant operation or considered by operators to be prudent to have available and operable. Plant is considered ready for restart. If additional damage is found, proceed to Step 3.
  3. <u>Re-assess DL</u>. If DL is changed (i.e., increased), implement AL corresponding to the new DL and ELs in Table 4-1.

4. If the EL is greater than the SSE exceedance criterion, and the observed DL exceeds DL 1, perform seismic re-evaluations in accordance with ALs 5 or 6 as required in Table 4-1 to assess the need for short-term and long-term remedial actions in accordance with plant licensing requirements.

5. If the SSE is exceeded, develop and implement Seismic Evaluation and Verification Plan that considers the need to evaluate new and replacement SSCs for both the design and measured earthquake response spectra. (See Section 6.5).

- Action Level 2

  1. Perform Focused Inspections and Tests per Section 5.1 (unless already performed under previous AL 1) and Expanded Inspections and Tests per Section 5.2. In addition, check indicators of possible reactor coolant system (RCS) SSC damage. If a concern is indicated, perform inspections and tests per Section 5.3 of in-containment RCS SSCs. If there is no damage to SR and non-SR SSCs important to safe plant operation, repair/replace affected SSCs as required. Plant is considered ready for restart. Otherwise, proceed to Step 2.

  2. Re-assess DL. If DL is changed, implement AL corresponding to the new DL and ELs in Table 4-1.

- Action Level 3

  1. Perform Focused Inspections and Tests per Section 5.1 (unless already performed under previous AL) and Expanded Inspections and Tests per Section 5.2 (unless already performed under previous AL). In addition, check indicators of possible RCS SSC damage. If a concern is indicated, perform inspections and tests per Section 5.3 of in-containment reactor coolant system (RCS) components, piping and supports.

  2. Perform root cause/extent of condition evaluations of any damage to SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable.

  3. Repair/replace any damaged SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable based on the results of root cause evaluation. If damage mode applies to entire equipment class, assess need for repair/upgrade and implement, as required.

  4. Re-assess DL. If DL is changed, implement AL corresponding to the new DL and ELs in Table 4-1. If not, plant is considered ready for restart.

- Action Level 4

  1. Perform Focused Inspections and Tests per Section 5.1 (unless already performed under previous AL) and Expanded Inspections and Tests per Section 5.2 (unless already performed under previous AL). In addition, perform inspections and tests per Section 5.3 of in-containment RCS components, piping and supports.

2. <u>Perform root cause/extent of condition evaluations of any damage to SR and/or non-SR SSCs</u> important to safe plant operation or considered by operators to be prudent to have available and operable. Because evidence of significant damage to SR SSCs under EL 2 is highly unlikely and unexpected, the root cause/extent of condition evaluations should be performed based on applicable methodology described in Section 6 and considering the broader implications of the observed damage level at ELs less that the SSE exceedance criterion. Specifically, in addition to comparison of loads due to the measured earthquake with original qualification design loads, these evaluations should include review of initial plant design, design analyses, previous seismic qualification testing and other areas that could explain the damage observations. The results of these root cause evaluations shall be documented and subject to review and approval as part of the plant's Corrective Action Program (CAP).

3. <u>Develop Corrective Action Plan</u> to address the causal factors (based on the root cause analysis) as well as the broader implications of the causal factors on the seismic design and qualification of SR SSCs. Document justification if the implementation schedule for any corrective action is beyond the re-start schedule. Implement Corrective Action Plan after consultation with USNRC and approval of plan and schedule, as may be required.

4. <u>Repair/replace any damaged SR and/or non-SR SSCs</u> important to safe plant operation or considered by operators to be prudent to have available and operable based on the results of root cause evaluation. If damage mode applies to an entire equipment class, assess the need for repair/upgrade and implement, as required.

5. <u>Perform surveillance tests per Section 5.5</u> and any tests considered necessary to verify no additional hidden damage for SSC types that were identified as being damaged or malfunctioned.

6. <u>Open reactor vessel and inspect reactor vessel internals and fuel</u> if required based on the results of the root cause/extent of condition evaluation in accordance with the approved Corrective Action Plan.

7. <u>Perform integrated containment leak rate tests if required</u> based on the results of the root cause/extent of condition evaluation in accordance with the approved Corrective Action Plan.

8. <u>Re-assess DL</u>. If DL is changed, implement AL corresponding to the new DL and ELs in Table 4-1. When complete and successful, plant is considered ready for restart.

9. If the root cause/extent of condition evaluations identify cases where any upgrades or modifications required in the Corrective Action Plan are a result of observed or calculated earthquake seismic loads that exceed original design loads, develop and implement long-term Seismic Evaluation and Verification Plan for these cases, as appropriate. (See Section 6.5).

- Action Level 5

  1. Perform Focused Inspections and Tests per Section 5.1 (unless already performed under previous Action Level) and Expanded Inspections and Tests per Section 5.2 (unless already performed under previous Action Level). In addition, perform inspections and tests per Section 5.3 of in-containment reactor coolant system (RCS) components, piping and supports.
  2. Perform root cause/extent of condition evaluations of any damage to SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable.
  3. Repair/replace any damaged SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable based on the results of root cause evaluation. If damage mode applies to entire equipment class, assess need for repair/upgrade and implement, as required.
  4. Perform surveillance tests per Section 5.5 and any tests considered necessary to verify no hidden damage for SSC types that were damaged or malfunctioned.
  5. Open reactor vessel and inspect reactor vessel internals and fuel if anomalies occurred during or after the earthquake (e.g., flux perturbations, fuel leakage, control rod malfunction, etc.).
  6. Re-assess Damage Level based on above inspections and tests. If re-assessed DL is DL 3 or less, plant is considered ready for restart; proceed to Step 7, below. If DL is greater than DL 3, implement Action Level 6.
  7. Perform Long-Term Evaluations described in Section 6 of this report to verify operability of SR SSCs for measured earthquake level. Address any prior flaw growth and leak-before-break analyses, as applicable, of existing and/or assumed flaws in reactor coolant system and internals as part of Long-Term Evaluations. (See Section 6.2).
  8. Re-evaluate seismic hazard for plant and evaluate plant for revised seismic hazard. (See Section 6.3).
  9. Develop Seismic Evaluation and Verification Plan to implement any upgrades or modifications identified in 8, above, and to address both the original and observed earthquake levels in the seismic qualification of new and replacement equipment. (See Section 6.5).
  10. Consider need to update Seismic Margin Assessment or Seismic PSA in the event that the re-assessed seismic hazard represents a significant increase in the hazard. (See Section 6.4).

- Action Level 6

  1. Perform Focused Inspections and Tests per Section 5.1 (unless already performed under previous Action Level) and Expanded Inspections and Tests per Section 5.2 (unless already performed under previous Action Level). In addition, perform inspections and tests per Section 5.3 of in-

containment reactor coolant system (RCS) components, piping and supports.

2. Perform root cause/extent of condition evaluations of any damage to SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable.
3. Repair/replace any damaged SR and/or non-SR SSCs important to safe plant operation or considered by operators to be prudent to have available and operable based on the results of root cause evaluation. If damage mode applies to entire equipment class, assess need for repair/upgrade and implement, as required.
4. Perform surveillance tests per Section 5.5 and any tests considered necessary to verify no hidden damage for SSC types that were damaged or malfunctioned.
5. Open reactor vessel and inspect reactor vessel internals and fuel.
6. Perform integrated containment leak rate tests.
7. Perform Long-Term Evaluations described in Section 6 of this report to verify operability of SR SSCs for measured earthquake level. Address any prior flaw growth and leak-before-break analyses, as applicable, of existing and/or assumed flaws in reactor coolant system and internals as part of Long-Term Evaluations. (See Section 6.2).
8. Re-evaluate seismic hazard for plant and evaluate plant for revised seismic hazard. (See Section 6.3).
9. When complete and successful, plant is considered ready for restart.
10. Develop Seismic Evaluation and Verification Plan to implement any upgrades or modifications identified in 8, above, and to address both the original and observed earthquake levels in the seismic qualification of new and replacement equipment. (See Section 6.5).
11. Consider need to update Seismic Margin Assessment or Seismic PSA in the event that the re-assessed seismic hazard represents a significant increase in the hazard. (See Section 6.4)

4.3.1 Actions for Earthquakes with High-Frequency Exceedances

Experience has shown that earthquakes whose measured ground motion spectra exceed the design basis SSE spectra only at high frequencies (i.e., above about 10 Hz) have little effect on engineered power plant equipment and structures. The main concern is with malfunction (rather than damage) of high-frequency-sensitive (HF-sensitive) devices such as relays, contactors and certain switches. As a consequence, actions should be taken in this case 1) to review system operating and alarm records to determine if any unusual system performance or alarm history during the observed earthquake indicated any detectable or suspected malfunctions of HF-sensitive devices, particularly essential, SR relays and 2) to verify that any relays that tripped are re-set as required and that the SR circuits are functional as required by plant Technical Specifications prior to restart.

4.3.2 Actions for Earthquakes with Low-Frequency Exceedances

Earthquakes having very low frequencies (below the fundamental frequency of the soil-structure response of the buildings, typically under about 1 to 2 Hz) affect mainly those items whose response is displacement-controlled. These SSCs are not normally SR components, but include such items as low pressure storage tanks, un-anchored components such as free-standing spent fuel racks, and reactor fuel. Liquid sloshing in open tanks and spent fuel storage pools may also be observed. Accordingly, actions should be taken in this case to investigate the potential for unacceptable low frequency response of such items prior to restart.

Table 4-1
Action Level Matrix

DAMAGE LEVEL	EARTHQUAKE LEVEL		
	EL 1: < or = OBE	**EL 2: >OBE, <or = SSE**	**EL 3: > SSE**
DL 0: No damage to safety-related (SR) SSCs or non-SR SSCs important to safe plant operation. Damage limited to non-SR, Damage Indicators that have no significant impact on plant operation, and typically found in residences, office buildings, etc.	No Actions Recommended	Action Level 1	Action Level 1
DL 1: No damage to SR SSCs; no damage to rugged industrial type non-SR SSCs. Damage to non-SR SSCs not important to safe plant operation.	Action Level 1	Action Level 1	Action Level 1
DL 2: No damage to SR SSCs; damage to rugged industrial type non-SR SSCs. Damage to non-SR SSCs important to safe plant operation.	Action Level 2	Action Level 2	Action Level 5
DL 3: Damage to many non-SR SSCs; Slight/isolated damage to less rugged SR SSCs that does not affect equipment functionality	Note 1	Action Level 3	Action Level 5
DL 4: Damage to SR and non-SR SSCs	Note 1	Action Level 4	Action Level 6

Notes:

1. These combinations of DL and EL are highly unlikely; if they should occur, it suggests more significant problems than are addressed in the scope of this report.
2. In addition to the cases covered in this table, attention should be given to earthquake scenarios involving high-frequency and low-frequency exceedances of SSE design spectra. See Section 4.3, above.

Section 5: Post-Shutdown Inspections and Tests

This section provides descriptions of recommended inspections and tests of nuclear plant equipment and structures prior to restart of a nuclear power plant which has been shut down due to an earthquake. The recommended Action Levels (ALs) are shown in Table 4-1. The post-shutdown inspections and tests listed in the ALs are described below.

5.1 Focused Inspections and Tests

The Focused Inspections and Tests are detailed, visual inspections and tests of a pre-selected sample of representative structures and equipment, as discussed in Section 2. The equipment and structures included in the focused inspections are selected to sample all types of SR and non-SR SSCs important to safe plant operation, and include equipment and types of structures which are considered most vulnerable to damage due to earthquake shaking. The focused inspections also include non-SR SSCs which experience has shown to be of low seismic capacity to serve as earthquake damage indicators. These inspections should be performed by experienced engineers (e.g., experienced and/or trained in seismic design and qualification, experienced in the observation of earthquake damage to commercial SSCs, or equivalent). The purpose of these inspections is to determine the need for expanded inspections and tests and to provide data to establish the earthquake DL defined in Section 4.

A "smart" sample of the following nuclear plant SSCs and their supports should be included in the scope of the focused post-earthquake inspections; guidelines on selection of the "smart " sample are discussed in Section 2.4.1, "Selection of Scope of SSCs to be Evaluated", as part of the intended pre-earthquake planning. The types of SSCs that are intended to be included in the smart sample are as follows:

- Equipment. The twenty classes of safe shutdown equipment identified in the Seismic Qualification Utility Group "Generic Implementation Procedure"[11] , to include:
  - Fans
  - Air compressors
  - Batteries and racks
  - Static inverters and battery chargers

- Air handlers
- Chillers
- Transformers
- Vertical pumps
- Horizontal Pumps
- Motor generators
- Motor control centers
- Low voltage switchgear
- Medium voltage switchgear
- Distribution panels
- Fluid/Air/Motor/Solenoid-operated valves
- Engine generators
- Instrument racks
- Sensors
- Control and instrumentation cabinets

▪ Flat-bottomed, Low Pressure Storage Tanks

▪ Emergency Generator Fuel Oil Tanks

▪ High Pressure Tanks and Heat Exchangers

▪ Piping. Include at least three sizes as indicated below and at least one cold (design temperature less than or equal to 150˚F) and one hot (design temperature greater than 150˚F) system.

  - Less than 6 inches
  - 6 inches to 12 inches
  - Greater than 12 inches

▪ Electrical Raceways. Include at least two each of each major support type (e.g., cantilever, trapeze, etc.).

  - Conduit
  - Cable trays

▪ Air Handling Ducts. Include at least two each of each size.

  - Less than 12 inches
  - Greater than 12 inches

▪ Steel Framed Structures

▪ Reinforced Concrete Structures and Masonry Walls

  - Major buildings (i.e., reactor building, auxiliary building, pump house)
  - Spent fuel pool
  - Ventilation stack (BWRs)

▪ Damage Indicators

  - Switchyard equipment (transformers, ceramic insulators, switchgear)
  - Suspended ceilings, plaster walls, unreinforced masonry walls, and glass windows
  - Unanchored cabinets and storage racks
  - Cranes (off-track) or elevators (out-of-alignment)

- Non-SR building and equipment foundations (settlement or ground failure)
- Fire protection system mains
- Natural gas supply piping

5.2 Expanded Inspections and Tests

Expanded inspections and tests are recommended in the AL matrix if significant physical or functional damage is found in the SR or important non-SR SSCs selected for the focused post-shutdown inspections. The expanded inspections should include all accessible[6] SR equipment and structure types as well as non-SR balance-of- plant equipment important to safe operation of the plant. As a minimum, the expanded inspections should include the following.

- All accessible SR equipment and supports not included in the focused inspections. This would include 100 percent of the items which were inspected on a sampling basis in the focused inspections.
- All accessible SR distribution systems (i.e., piping, raceways and ducting) and supports.
- Non-SR equipment important to safe plant operation (e.g., turbine-generator, feedwater system, switchyard equipment, etc.).
- Seismic Category I buildings (and their penetrations) and structures.
- Containment, including containment penetrations.
- Intake structure canals, piping, and other equipment required for ultimate heat-sink.
- Dam/reservoir (if needed to preclude unacceptable flooding or loss of ultimate heat sink).
- Buried pipe at accessible interfaces with buildings and components. In the event of significant ground failure, buried piping in the failure zone should be evaluated.

An exception to the need to expand the focused inspections and tests applies if the damage to SSCs included as part of the focused inspections is isolated to a specific class (or classes) of equipment or structures, and the cause of the damage is attributable to a specific design or installation deficiency (e.g., lack of equipment anchorage, improper installation of expansion bolts, etc.). In this case, the design/installation deficiency should be corrected for all of the SSCs in the class(es) involved, and inspections of other undamaged classes would not need to be expanded.

[6] Accessible is intended to indicate that the item can be inspected without disassembly, excavation, extensive scaffolding, etc.

5.3 Reactor Coolant System (RCS) Inspections and Tests

Visual inspections of RCS components, piping and supports are recommended for ALs 4 and higher and under ALs 2 and 3 if concerns are identified. The need for these inspections needs to be weighed against the considerable time and effort required to de-inert and open the containment vessel. Concerns that would suggest the need for such inspections include abnormal neutron flux readings, excess leakage from coolant pump seals or abnormal leakage indicated by the RCS leakage monitoring system, increased radiation levels inside containment, abnormal coolant chemistry (including high levels of fission products), abnormal vibration of pumps (where vibration monitoring instruments are installed) and unusual sound from a loose-parts monitoring system (if installed). Components to be inspected would include reactor vessel penetrations, main coolant pumps, steam generators, pressurizer, piping and supports.

Where recommended under ALs 5 and 6, the reactor vessel should be opened and reactor vessel internals and fuel inspected using methods normally employed for in-service inspections.

5.4 Guidelines for Visual Inspections and Operability Tests

General guidelines for visual inspections of SSCs during the post-shutdown inspections (focused and expanded) are provided in Table 5-1. This table lists inspections recommended for each class of equipment or structure. In general, the inspections consist of: (a) a visual observation of the condition of the equipment and its anchorage, (b) a visual observation of the condition of attached piping and conduit, and (c) checks for evidence of functional damage. Several operational tests (e.g., for vibration) are also included. These criteria are based on experience with damage in commercial facilities that have experienced strong-motion earthquakes and on experience from seismic qualification testing of equipment.

Special attention is recommended in the case of electrical equipment qualified by test. Post-earthquake inspections of such equipment need to be very detailed, involving opening of cabinets for access to internal components and attachments, and should focus on known seismic failure modes of electrical equipment such as those that have been observed in seismic qualification tests. These would include physical/structural damage that is common under loads that could affect equipment functionality (e.g., deformed cabinet housings and internal bracing, loose internal attachments and wiring, anchorage damage or loosening, cracks in fillet welded anchorages, etc.). Persons experienced in seismic qualification of electrical equipment should be involved, if possible. If damage is noted in these examinations that suggest equipment functionality could be challenged, consideration should be given to evaluating the equipment qualification records of the affected equipment using seismic inputs equivalent to the observed seismic inputs. Guidelines for such evaluations are discussed in Section 6.1.4.2.

5.5 Surveillance Tests

ALs 4, 5 and 6 recommend surveillance tests as part of the post-earthquake tests. The intent of these guidelines is that the specific surveillance tests performed be based on the damage observed in the various operator walkdowns, focused and expanded inspections and tests, and checks for equipment and system malfunctions. It is not intended that *all* surveillance tests required by plant Technical Specifications be performed as part of the plant restart plan. Instead, surveillance tests selected to be performed should be related to, and in support of evaluations of equipment and systems that have been damaged or are the subject of an operability concern by operators or seismic engineers.

Pre-startup testing required by plant Technical Specifications for all plant startups, and periodic In-Service Inspections required by US Standards (e.g., Section XI of the ASME Boiler and Pressure Vessel Code) are not specifically covered in the guidelines presented in this report. They should be performed as required by plant-specific licensing requirements.

During surveillance testing, the vibration of rotating equipment (e.g., fans and pumps) should be closely monitored.

5.6 High-Frequency-Sensitive Devices

Inadvertent actuations of SR essential relays and other HF-sensitive devices as a result of an earthquake, especially one with significant energy content at frequencies in excess of about 10 Hz, could result in un-wanted actuations of systems and alarms, and could also trip important systems and components. Re-setting tripped circuits would be required in some cases. As a consequence, plant response procedures should provide guidance 1) to plant operators on duty during and after the earthquake for them to be prepared for and to record any unusual, unexpected actuations and trips, and 2) to plant operators and engineers charged with verifying plant readiness to restart to confirm the functionality of essential relays, contactors, vibration monitoring instruments, etc., and to ensure that tripped circuits have been re-set as part of the restart program.

5.7 Inspection Personnel

The post-shutdown inspections (focused and expanded) should be performed by a team (or teams) of utility and/or contractor engineers in addition to plant operations personnel. The post-shutdown inspection teams should include members with expertise in civil/structural, mechanical, electrical and systems engineering in addition to experienced seismic engineers (e.g., experienced and/or trained in seismic design and qualification, experienced in the observation of earthquake damage to commercial SSCs, or equivalent). Personnel who performed the baseline inspections should be included on the post-shutdown inspection teams whenever possible.

5.8 Documentation

Results of pre- and post-shutdown inspections and tests should be documented. The reports should identify each item of equipment or structure inspected and the results of the inspections. Results of inspections should be compared with results of previous baseline inspections. Use of previously prepared checklists that include baseline inspection results and photos is encouraged.

Table 5-1
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
1.	**Equipment** –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 or tearing of fabric noise eliminators due to seismic loads imposed by attached ducts.
		4.	Check for evidence of excessive fan vibration and/or noise. May be an indication of misalignment between the motor and 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/odor.
		8.	Check local alarms, breakers and protective devices for actuation/trips.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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 to attached conduit and ground straps.
		3.	Check for damage due to impact or earthquake-induced flooding or spraying.
		4.	Check for excessive noise and/or vibration.
		5.	Check for air leaks if compressor is running continuously rather than cycling on and off.
		6.	Check for belt tightness and/or slippage; e.g., belt smoke/odor.
		7.	Check local alarms, breakers and protective devices for actuation/trips.
	– Batteries and Battery Racks	1.	Check batteries for damage, leakage and loose terminal connections. Use of thermography is suggested to look for hot spots.
		2.	Check battery rack anchorage for damage; e.g., stretching or loosening of anchor bolts or nuts; evidence of rocking or sliding of racks.
		3.	Check for distortion of rack structure.
		4.	Check for evidence of rocking or sliding of batteries on the racks, buckling or distortion of the bus bars, condition of the spacers between batteries.
		5.	Check for damage due to impact or earthquake-induced flooding or spraying.
		6.	Check buses/cables/ground straps for damage, distortion or chafing.
		7.	Check local alarms, breakers and protective devices for actuation/trips.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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 accessible 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.
	– 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 conduit and ground straps.
		3.	Check for damage to air handler due to seismic loads imposed by attached ducts or tearing of fabric noise eliminators.
		4.	Check for damage due to impact or earthquake-induced flooding or spraying.
		5.	Check for belt tightness and/or slippage; e.g., belt smoke/odor.
		6.	Check local alarms, breakers and protective devices for actuation/trips.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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 conduit and ground straps.
		3.	Check for leakage or damage to chiller components due to seismic loads imposed by attached ducts and piping.
		4.	Check for damage due to impact or earthquake-induced flooding or spraying.
		5.	Check for belt tightness and/or slippage; e.g., belt smoke/odor.
		6.	Check local alarms, breakers and protective devices for actuation/trips.
		7.	Check for refrigerant leakage.
	– Transformers	1.	Check equipment anchorage for damage, 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 oil reservoir level.
		4.	Check the nitrogen blanketing system and fire deluge system for damage.
		5.	Check for damage due to impact or earthquake-induced flooding or spraying.
		6.	Check for damage (e.g., cracking) of ceramic insulators. Consider thermography for detection of hot spots.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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 due to ground settlement/movement.
		3.	Check for evidence of excessive noise and/or vibration and seal leakage. May be an indication of 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 due to impact or earthquake-induced flooding or spraying.
		7.	Check local alarms, breakers and protective devices for actuation/trips.
		8.	Check pump and motor bearings for overheating/lubrication.
		9.	Check for damage to attached conduit and ground straps.
	– 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 vibration and seal leakage. May be an indication of misalignment between motor and pump shaft.
		3.	Check for damage to pump housing due to seismic loads imposed by attached piping.
		4.	Check for damage due to impact or earthquake-induced flooding or spraying.
		5.	Check local alarms, breakers and protective devices for actuation/trips.
		6.	Check pump and motor bearings for overheating/lubrication.
		7.	Check for damage to attached conduit and ground straps.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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 to a common base.
		3.	Check for damage to attached conduits and ground straps.
		4.	Check for damage due to impact or earthquake-induced flooding or spraying.
		5.	Consider use of thermography to detect hot spots.
		6.	Check local alarms, breakers and protective devices for actuation/trips.
	– Motor Control Centers	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, if de-energized, check to see that all internally mounted accessible components, including relays and breakers, are secure and undamaged.
		5.	Check for damage due to impact or earthquake-induced flooding or spraying.
		6.	Check controls, breakers and protective devices for actuations/trips.
		7.	Consider the use of thermography to detect hot spots

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	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 conduit and ground straps.
		3.	Check for distortion of cabinet structure.
		4.	Open cabinets, check to see that all internally mounted accessible components, including relays and contacts, 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.
		7	Reset any trips. Investigate any re-trips after reset.
		8.	Consider the use of thermography to detect hot spots
	– 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 accessible components, including relays and contacts, 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.
		7.	Reset any trips. Investigate any re-trips after reset.
		8.	Consider the use of thermography to detect hot spots

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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, if de-energized, check to see that all internally mounted accessible components are secure and undamaged.
		5.	Check for damage due to impact or earthquake-induced flooding or spraying.
		6.	Reset any tripped breakers. Investigate any re-trips after reset.
	– Fluid/Air/Motor/Solenoid-Operated Valves	1.	Check for damage or distortion at attachment of operator to valve body.
		2.	Check for damage to attached conduit/tubing, ground straps.
		3.	Check for damage due to impact or earthquake- induced flooding or spraying.
		4.	Check local alarms/indicators/protective devices for actuations/trips.
		5.	Stroke valve in both directions to check operation when permitted by plant operational conditions

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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, conduit and ground straps.
		3.	Check for noise and/or vibration due to misalignment between engine and generator, especially if not mounted to a common base.
		4.	Check for damage due to impact or 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 the rack 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.
		7.	Reset any trips. Investigate any re-trips after reset.
	– Sensors	1.	Check for damage to attached conduit/tubing and ground straps.
		2.	Check for damage due to impact or earthquake-induced flooding or spraying.
		3.	Verify sensor operation with readout check at local/control room indicators.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
	– 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 accessible instruments, gages, controls, and other equipment mounted to panels 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.
		7.	Reset any trips. Investigate re-trips after reset.
2.	**Flat-bottomed, 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 buckling of tank walls; e.g., "elephant foot" buckling.
		4.	Check for cracking or leakage at the base plate to cylindrical shell connection.
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 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
4.	**Piping**	1.	Check for snubber damage; e.g., snubbers pulled loose from foundation bolts, evidence of excessive travel, jam up 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 pipe due to impact with support structure.
		3.	Check for damage or leakage of pipe at rigid connections; e.g., anchor points with other equipment and structures.
		4.	Check for damage or leakage of piping and branch lines.
		5.	Check for damage to pipe at building joints and interfaces between buildings.
		7.	Check piping-to-support clearances
		8.	Check pipe hanger spring settings
5.	**Cable and Conduit Raceways**	1.	Check for deformation of dead weight supports and sway bracing.
		2.	Check for damage to cables at building joints and interfaces between buildings.
		3.	Check for damage due to impact or earthquake-induced flooding or spraying.
6.	**Air Handling Ducts**	1.	Check for deformation of dead weight 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 due to impact
		5.	Check for tearing of fabric transitions/noise eliminators.
		6.	Check for damage to internal filters and racks.

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
7.	**Steel Framed Structures**	1.	Check for damage at 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.
		4.	Check for evidence of ground or foundation settlement.
8.	**Reinforced Concrete Structures (Buildings, Containment, Cooling Towers, Intake Structure) and Masonry Walls**	1.	Check for new open (>0.06 inches) cracks, spalling of concrete. [Note: Minor cracks, even if caused by the earthquake, are not considered significant unless they are large enough to result in yielding of re- bar.].
		2.	Check for evidence of ground or foundation settlement.
		3.	Check for evidence of differential horizontal and vertical movement between adjacent and/or interconnecting buildings/structures
9.	**Primary Coolant System (when specified in recommended Action Levels)**	1.	Check for reactor coolant leakage at flanged joints; e.g., CRD mechanisms.
		2.	Check for condition of supports and snubbers for large components and piping; e.g., main coolant pumps, steam generators, pressurizer, reactor coolant and recirculation piping, main steam isolation valves, etc.
		3.	Check condition of CRDM support structure (PWRs only).

Table 5-1 (continued)
Visual Inspection of Equipment and Structures after Earthquake

Equipment/Structure		Types of Inspections	
10.	**Buried Pipe**	1.	Check for damage or leakage at accessible pipe interfaces with buildings and components mounted on separate foundations. In the event of significant ground failure, buried piping in the failure zone should be evaluated.
		2.	Fire main leakage will be evidenced by self excavation and actuation of back up fire pumps.
		3.	Fire mains, service and circulating water piping, especially dead legs, are susceptible to buildups of 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.

Section 6: Long-Term Evaluations

This section of the report provides guidelines for evaluations of SSCs to determine the effects of an earthquake on their long-term integrity and functionality. These evaluations are recommended under ALs 4, 5 and 6 in Table 4-1.

The long-term evaluations are performed when the observed earthquake level is greater than the SSE exceedance criterion (EL 3) and the damage level has been determined to be DL 2 or higher. They would normally be performed after the plant has returned to power for DLs 2 or 3. However, if the level of observed damage based on the post-shutdown inspections is determined to be greater than DL 3 and the EL is 3 (i.e., exceeds the plant SSE exceedance criterion), then it is recommended that the long-term evaluations be performed and completed prior to restart. Long-term evaluations are also recommended when the observed earthquake is less than the plant design SSE (EL 2) and the damage level is DL 4.

The evaluations described in this section provide assurance that the nuclear power plant can operate safely in the long-term and is capable of withstanding a second earthquake. Any corrective measures which are considered necessary as a result of these additional evaluations would be implemented on a case-by-case basis.

It is intended that the evaluations performed under this section, and any resulting repairs/replacements of SR SSCs, would be performed, tracked and documented under the plant's approved Corrective Action Program (CAP).

The long-term evaluations and related actions recommended under ALs 4, 5 and 6 are described below.

6.1 Comparative Analyses of Observed vs. Design Earthquakes

6.1.1 Calculation of Seismic Loads

In-structure seismic response spectra (ISRS) should be generated for all plant elevations of interest based on the actual earthquake motion records using realistic, median-centered methods (e.g., best estimate modeling and damping).

ISRS based on measured time-history data on the floors of interest should be used in place of calculated ISRS for those floors and elevations where suitable measurements are available.

6.1.2 Comparison of Actual and Design Seismic Loads

The calculated or measured ISRS based on the actual earthquake record should be compared with the applicable design SSE ISRS. If the calculated ISRS for any floor elevation are enveloped by the design SSE ISRS, then the design basis for floor mounted equipment and piping on that floor has not been exceeded, and seismic re-evaluation of equipment and piping on that floor is not considered necessary. However, if the calculated ISRS for the applicable floor elevation exceeds the corresponding design SSE ISRS, then the design basis for floor mounted equipment and piping, as well as the structure itself, may have been exceeded and further evaluations should be performed.

6.1.3 Seismic Re-Evaluations

If it is determined that the calculated seismic loads based on the actual earthquake records exceed the SSE design seismic loads, then typical floor mounted equipment, piping, and structures should be re-evaluated based on the actual seismic loading conditions. Considerations in selecting items for seismic re-evaluation are as follows.

- Select items with natural frequencies in the range where the SSE has been exceeded.
- Select items with the highest calculated stresses based on previous stress analysis results.
- Select items which are representative of equipment, piping, and sub-structures located on floors with exceedances. In those cases where measured ISRS are available and exceed the design ISRS, component-specific loads should be determined using modern analytical methods and compared to the design loads used for initial seismic qualification.

6.1.4 Acceptance Standards

Acceptance standards for seismic re-evaluations are provided below for: (a) SSCs typically qualified by analysis, and (b) equipment typically qualified by methods other than analysis (e.g., by test or seismic experience data). The recommended acceptance criteria are intended as guidelines for determining an SSC's structural adequacy and operability following a potentially damaging earthquake that exceeds the plant's design SSE, and also for determining the long-term integrity of the SSC, including its adequacy for a subsequent, like-size earthquake. Initial evaluations include comparisons of earthquake-induced loads/stresses with seismic design criteria typical of or equivalent to original design criteria. Alternative, less restrictive evaluation criteria are recommended for case-specific technical evaluations that include supplemental inspections and/or tests when generally accepted design criteria are not met. These beyond-design-basis evaluations are typical of the type of evaluations performed in nuclear plants to assess the need to repair or replace potentially non-conforming SSCs versus accepting such SSCs as-is for continued service, with or without additional conditions such as enhanced inspections, etc. This approach is basically the same as that routinely used in the nuclear industry to evaluate, after the fact, the effects of all types of plant upsets and beyond design basis events (e.g., water-hammer events, over-pressures, degraded piping and components, IGSC cracking, etc.). Such assessments are controlled, tracked and reviewed under a plant's required Corrective Action Program (CAP). It is also consistent with the NP-6695[2] guidelines accepted in Reg. Guide 1.167[4] and the IAEA guidelines[6].

Implementation of the recommended acceptance standards for equipment and structures qualified by different methods is discussed below.

6.1.4.1 Equipment and Structures Qualified by Analysis.

The following acceptance standards are recommended for SSCs typically qualified for seismic loads by analysis (e.g., piping, piping and component supports, building structures, pressure vessels and tanks, etc.).

1. If the calculated stresses from the actual seismic loading conditions are less than allowables for emergency conditions (e.g., ASME Code, Section III, Level C Service Limits, or equivalent) or original design bases, then the item is considered acceptable.
2. If the calculated stresses are greater than allowables for emergency conditions but less than allowables for faulted conditions (e.g., ASME Code, Section III, Level D Service Limits, or equivalent), then the acceptability of the item should be based on the following considerations.
   - Results of a detailed visual inspection
   - An engineering evaluation of the effects of the calculated stresses on the functionality of the item
   - Results of equipment operability tests (for active components)
3. If the calculated stresses are greater than allowables for faulted conditions, then the acceptability of the item should be based on the following considerations.
   - Results of a detailed visual inspection
   - An engineering evaluation of the effects of the calculated stresses on the functionality of the item
   - Results of equipment operability tests (for active components)
   - Results of additional nondestructive examinations of the item (i.e., examinations of specific areas of the item which were found to be highly stressed or are a concern based on component-specific evaluations)
   - Repair or replacement of potentially damaged areas

Equipment anchorages should be evaluated for the actual seismic loads in accordance with the acceptance standards given above.

For piping, seismic reanalysis should be limited to ASME Code, Section III, Class 1 piping and/or any piping which shows visual evidence of large displacement or distress. Complete seismic reanalysis of all piping is not considered necessary. Experience has shown that piping systems designed to the rules of the ASME Code, Section III, do not show damage due to the inertia loads resulting from an earthquake. If damage occurs, it will most likely show as damage to the piping supports, or damage to the pipe at fixed supports due to relative support movements. These types of damage would be detected by the plant walkdown inspections and post-shutdown inspections described in Section 5 of this report. In general, piping reanalysis should be performed on a sampling basis to verify adequacy of piping and to assess the need for supplemental nondestructive examination of potential high strain areas.

Technical bases for the recommended acceptance standards include the following:

- The Level D type allowables limit inelastic strains to very low values for all normal structural materials
- The applied loading environment is known, the equipment is available for inspection and evaluation and, therefore, seismic margins need not be as high as in an original design.
- Seismic events have been shown to have insufficient numbers of significant stress cycles to pose a low-cycle fatigue concern. This would apply to several postulated subsequent earthquakes
- Data presented in the IAEA report[6], Annex III, demonstrate that limited amounts of initial plastic strain (e.g., up to 8% in these tests) have essentially no detrimental effect on the remaining fatigue life of typical power plant construction materials.

6.1.4.2 Equipment Qualified by Methods Other Than Analysis.

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

1. For equipment qualified by test, the original test response spectrum (TRS) envelopes the calculated or measured response spectrum based on the actual earthquake record.
2. TRS available through test agencies and vendors for equipment determined to be similar to the equipment being evaluated envelope the calculated or measured response spectrum based on the actual earthquake record.
3. Available Generic Equipment Ruggedness Spectra (GERS) for the equipment divided by a "knock-down" factor of 1.3 envelope the calculated

or measured response spectrum based on the actual earthquake record. GERS for relays, thirteen classes of electrical equipment, and four classes of mechanical equipment (valves) are published in EPRI report NP-5223[12] and NP-7147{13]. These data may be used to evaluate equipment qualified by test that does not meet Conditions 1 and 2, above.

4. The equipment is considered to be qualified for further operation on the basis of experience data. Earthquake performance data for twenty classes of nuclear plant electrical and mechanical equipment are contained in the Generic Implementation Procedure[11]. Bounding spectra that define conservative seismic capacities of typical power plant equipment and limitations or caveats applicable to these data are published in the Senior Seismic Review and Advisory Panel Report[14].

Additional considerations and cautions on identification and evaluation of high frequency sensitive devices (and their settings) and low frequency sensitive SSCs are given in Section 4.3.

Equipment that is not found to be acceptable by these means would have to be dispositioned under the plant's approved Corrective Action Program (CAP), as discussed above.

It should be noted that US Seismic Equipment Qualification (SEQ) Standards require tests that include six simulated earthquakes, one at SSE level and five at ½ SSE level. In addition, test spectra are typically conservative, either because of test table limitations in matching required spectra or vendor use of conservative test spectra to qualify his equipment for multiple sites. These factors and the inherent margins in the inspected and undamaged equipment provide reasonable assurance that equipment determined to be acceptable in the evaluations recommended above will be adequate for a subsequent earthquake.

6.2 Re-evaluation of Special RCS Piping and Internals Analyses

A number of operating nuclear power plants have performed special analyses to estimate growth and stability of known and/or assumed flaws in reactor vessel internals and in reactor coolant piping. In BWRs, this is commonly done to track growth of intergranular stress corrosion cracks in piping and vessel internal components. In addition, many plants have performed Leak-Before-Break (LBB) analyses to demonstrate that assumed flaw sizes will not grow unstably before detectable leakage is observed. Under ALs 5 and 6, and in cases where the design seismic loads on SSEs are exceeded, the assumptions and results of these and similar analyses need to be reviewed to verify that important conclusions are not changed by the imposition of higher seismic loads than used in the original analyses.

6.3 Re-evaluation of Site Seismic Hazard

If the observed earthquake level exceeds the plant's SSE exceedance level and damage to SR and/or non-SR rugged, industrial SSCs is found, consideration should be given to re-assessing the seismic hazard for the plant site using modern methods and data. This has been done following several strong-motion earthquakes in Japan. In this case, an evaluation should be performed to confirm that the plant is adequate for the increased seismic hazard. How such a re-evaluation should be implemented for both installed plant SSCs and future additions and modifications should be covered in the Seismic Evaluation and Verification Plan discussed in Section 6.5, below.

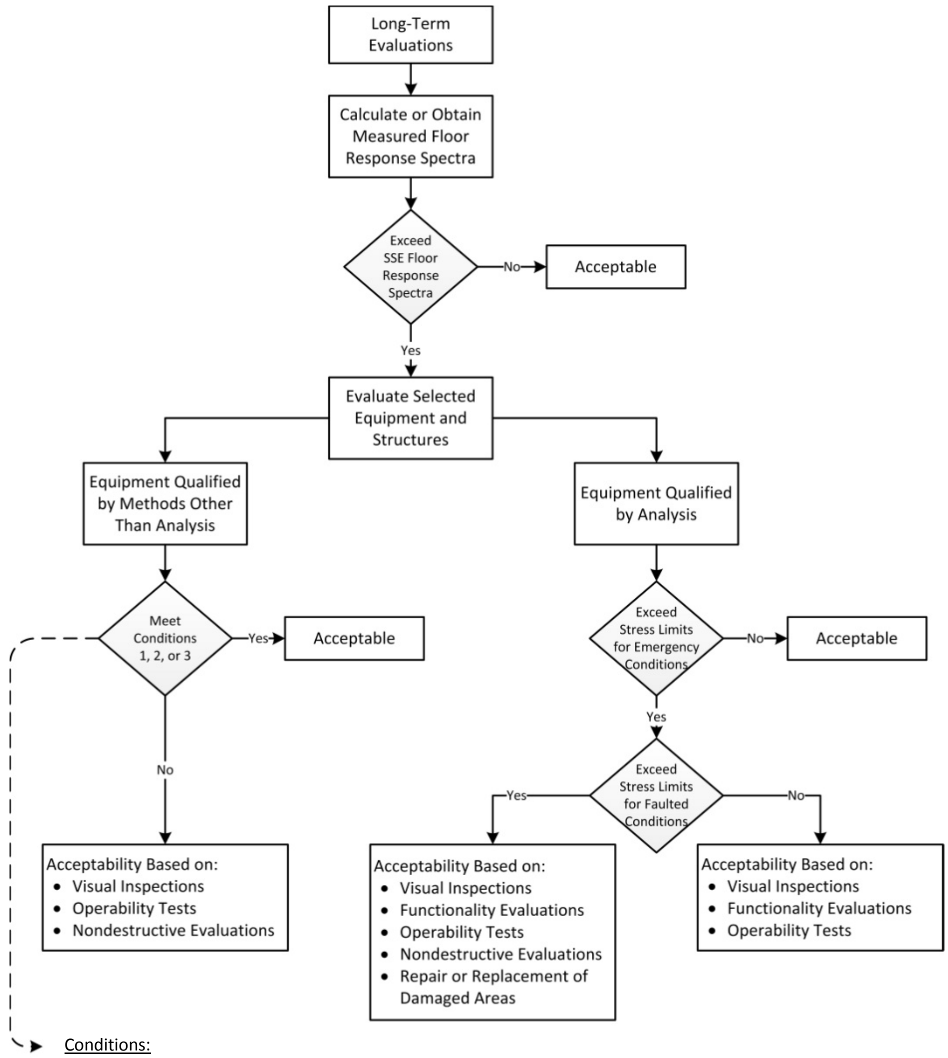
6.4 Up-Date of SMA/SPSA Analyses

If the site seismic hazard is re-assessed and increased significantly (e.g., by 20 to 25%), those plants that have "living" SMA/SPSA evaluations should consider the need to up-date these evaluations to reflect the increased seismic hazard.

6.5. Development of Seismic Evaluation and Verification Plan

In the event that the SSE exceedance level is exceeded, a seismic Evaluation and Verification Plan (or similar management plan) should be developed and implemented to prescribe how the measured and existing design basis response spectra will be implemented for future plant additions and modifications. It is recommended that this plan require that new and replacement SR SSCs be seismically qualified to both the licensing basis design spectra and the observed spectra unless it can be demonstrated that the SSC(s) involved does not pose a significant seismic risk (e.g., by results of a "living" SPSA).

A flow diagram of the recommended long-term evaluations is shown in Figure 6-1.



<u>Conditions:</u>

- TRS, or alternative, applicable seismic capacity qualification data, exceed ISRS based on actual record.
- GERS divided by 1.3 exceeds floor response spectra based on actual earthquake record.
- Qualified on basis of seismic experience data

Figure 6-1
F1ow Diagram of Long-Term Evaluations and Acceptance Criteria

Section 7: References

1. EPRI Report 30020007220, "Guidelines for Nuclear Plant Response to an Earthquake", Electric Power Research Institute, Palo Alto, CA, October, 2013.
2. EPRI NP-6695, "Guidelines for Nuclear Plant Response to an Earthquake", Electric Power Research Institute, Palo Alto, CA, 1989.
3. USNRC Regulatory Guide 1.166, "Pre-earthquake Planning and Immediate Nuclear Plant Operator Post-earthquake Actions", 1997.
4. USNRC Regulatory Guide 1.167, "Restart of a Nuclear Shut down by a Seismic Event", 1997.
5. ANSI/ANS Standard 2.23, "Nuclear Plant Response to an Earthquake", 2002.
6. International Atomic Energy Agency (IAEA) Report "Earthquake Preparedness and Response for Nuclear Power Plants", Safety Reports Series No. 66, 2011.
7. EPRI NP-5930, "A Criterion for Determining Exceedance of the Operating Basis Earthquake", Electric Power Research Institute, Palo Alto, CA, July 1988.
8. EPRI TR 10082, "Standardization of the Cumulative Absolute Velocity", December, 1991.
9. EPRI Report 1024889, "Seismic Instrumentation at Nuclear Plants", Electric Power Research Institute, Palo Alto, CA, 2012.
10. Federal Register Notice 58FR39132, *USNRC Final Policy Statement on Technical Specifications for Nuclear Power Reactors*. July 22, 1993.
11. Seismic Qualification Utility Group (SQUG) "Generic Implementation Procedure for Nuclear Power Plant Equipment", Revision 3A, December 2001.
12. EPRI NP-5223, "Generic Seismic Ruggedness of Power Plant Equipment", Electric Power Research Institute, Palo Alto, CA, May 1987.
13. EPRI Report NP- 7147, "Seismic Ruggedness of Relays", Electric Power Research Institute, Palo Alto, CA, 1991, and addenda, 1993 and 1995.
14. Senior Seismic Review and Advisory Panel (SSRAP) Report, "Use of Seismic Experience and Test Data to Show Ruggedness of Equipment in Nuclear Power Plants", August 26, 1988.

Appendix A: Design Basis Earthquake Exceedance Criteria

Background

In 1987, the Electric Power Research Institute (EPRI) formed the OBE Exceedance Panel (the Panel) to develop a criterion for determining whether the ground motion due to an earthquake at a nuclear plant site exceeded that of the OBE. The objective of the criterion is to provide a technical basis for determining if the OBE has been exceeded based on both exceedance of the OBE seismic response spectrum and the damage potential of the earthquake, and to avoid unnecessary shutdown of operating nuclear power plants following non-damaging earthquakes.

The Panel subsequently performed an extensive study of damage in commercial power plants and industrial facilities caused by earthquakes. This study[7] concluded that the best predictor of earthquake damage to such commercial plants (the only plants available for the study that had experienced strong motion earthquakes) is a parameter referred to as the Cumulative Absolute Velocity, or CAV. The CAV is described below. The OBE Exceedance Criterion includes both the CAV and the plant design OBE seismic response spectra parameters.

Cumulative Absolute Velocity (CAV)

The CAV is defined as follows:

$$CAV = \int_0^{t_{max}} \left|a(t)\right| dt \qquad \textit{Eq. A-1}$$

where: a(t) = acceleration time history
t_{max} = duration of strong motion

This parameter is the absolute area under the ground motion acceleration time history accelerogram over the duration of the strong motion of an earthquake. This duration is defined based on subsequent work[8] and is used in the calculation of the "standardized CAV". Correlation of the values of CAV with the onset of damage in the commercial plants in the study demonstrated that damage was not observed at standardized CAV values less than 0.16 g-seconds. Thus, the threshold of damage was set at this value. Ground motion measurements used for determination of the CAV are intended to be based on free-field instruments unless otherwise justified by an appropriate engineering evaluation.

For each directional component of the free-field ground motion, the CAV is to be calculated as follows:

- For each acceleration component time-history, the absolute acceleration (g units) time-history is divided into 1-second intervals.
- For each acceleration component time-history, each 1-second interval that has at least one exceedance of 0.025 g is integrated over time.
- For each acceleration component time-history, all the integrated values are summed together to arrive at the CAV.

It is important to emphasize that this threshold value is based on plants that are generally not designed for earthquakes (and none for nuclear seismic design plants) and that were constructed to commercial, not nuclear quality assurance requirements. As a consequence, the threshold CAV value of 0.16 g-seconds is considered very conservative.

Appendix B: Characteristics of Seismic Data Acquisition Systems

A Seismic Data Acquisition System is a complete seismic monitoring system consisting of sensor(s) and Data Acquisition Units (DAUs) that acquire, store, and transmit digital data from one or more systems, including communication hardware and software.

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

Preferred installation, operability, and other characteristics of Seismic Data Acquisition Systems:

- Power. The sensor and recorder power source should have sufficient capacity for sensing and recording a minimum of 60 minutes of motion. This may be accomplished by providing battery capacity for a minimum of 60 minutes of system operation without recharging in combination with a battery charger connected to an uninterruptible power supply or line source.
- Storage. To avoid data lost due to overwriting main shock data with aftershock data, Seismic Data Acquisition Systems should have the provision to store tens of individual recorded earthquake scenarios without accidentally overwriting the data from any individual event with that of another. One example of such a provision was described at the Diablo Canyon Power Plant (DCPP), San Luis Obispo, CA, USA. The installed system at DCPP has the ability to store up to 60 events without switching out the storage device for the system. When the storage device approaches capacity, the control room is signaled that a new storage device is required. Redundant storage should be in place, i.e., data may be transmitted to an off-site location, but a redundant set of data should always reside at the plant or another independent location.
- Data transmission capability. If the expectations or requirements are to transmit the recorded data from the recording device to another location on-site or off-site for review, processing, decision-making, etc., a seismically qualified transmission mechanism should be available. This could be hard technology, such as cables, or soft technology, such as wireless transmission. In either case, the transmission mechanism should be verified to be operable if an earthquake occurs.

- Seismic qualification. The Seismic Data Acquisition System should be seismically qualified to perform its functions during and after the earthquake ground motion. The minimum level of seismic qualification should be to the SSE level. It is preferable for the seismic qualification to be at a level greater than the SSE to assure operability if an earthquake produces ground motion greater than the SSE at the site. Seismic qualification should include any support systems necessary for system operation, e.g., emergency power, battery backup.
- Maintenance and testing. To reasonably assure operability of the Seismic Data Acquisition System, appropriate periodic maintenance should be performed. In many instances, maintenance may be sub-contracted to a third party (often the system supplier) for an extended period of time, e.g., 10 years. The operating organization needs to verify that maintenance is performed even after such maintenance agreements have expired. In addition, testing of the operability of the system should be performed on a regular schedule, for example, quarterly. Schedule maintenance to keep maximum number of instruments in service.
- Installation and configuration control. Installation of all elements of the Seismic Data Acquisition System should be such that all seismic systems interaction concerns are resolved. Seismic systems interaction concerns are the consequences of failure of non-SR SSCs causing failure or malfunction of SR SSCs, which would be the Seismic Data Acquisition System if seismically qualified (commonly referred to as II/I). The phenomena associated with seismic systems interaction are: ***falling*** of items impacting the SR item, ***proximity*** meaning impact of adjacent non-SR SSCs causing malfunction or damage to SR SSCs, and ***spray/flooding***. Installation and on-going configuration control procedures should assure that potential II/I issues do not prevent the Seismic Data Acquisition System from performing its functions. In some cases, enclosures may be used to prevent inadvertent damage to portions of the system (or accidental impact by plant personnel).
- Operability of Seismic Data Acquisition System in all operational modes of the nuclear installation. Operational modes of the installation include low power states, and plant shutdown (scheduled and unscheduled outages).
- Multi-unit sites. Provisions should be made to coordinate the Seismic Data Acquisition Systems for multi-unit sites. In some cases, one set of sensors may serve multiple purposes on-site, e.g., free-field sensors. For sites where there are common systems for more than one unit, these common systems may be instrumented and the recorded data may be annunciated in multiple control rooms. Immediate post-earthquake actions for multi-unit sites should be closely coordinated between units.

Specific characteristics of Seismic Data Acquisition Systems:

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

- Measurement type. Acceleration, displacement, deformation, strain, and Standard Cumulative Absolute Velocity (CAV) should be considered. Time varying quantities should be recorded as time histories. Peak values of time varying quantities may also be recorded for specific applications, such as an Automatic Seismic Trip System (ASTS), or manual shutdown. Derived quantities, such as Standard CAV, may be useful in determining the expected level of damage in the nuclear installation and may define whether an operating plant may continue operating or should be shutdown (Appendix A).

- Directions of recorded motions. In general, for nuclear installations, three directions of motion (two horizontal and the vertical) should be recorded. These triaxial sensors should be aligned in the principle directions of the installation for ease of use in subsequent evaluations of the SSCs. It is most convenient if these directions coincide with the principle directions of analytical models of the SSCs.

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

- Frequency range or bandwidth. The frequency range is the range of frequencies that can be accurately reproduced by the recorded data. The overall bandwidth is a function of the system, i.e., sensors, cabling, and digitizer bandwidth. Minimum frequency range is 0.02 – 50 Hz. Typically, the low frequency range is at 0.01 Hz and the high frequency range is 100 Hz. The minimum sampling rate should be 200 samples per second.

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

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

- Pre-event memory. Pre-event memory times should be sufficient to capture the P-wave motions, when the sensor is triggered to save data by the S-wave motions. A minimum of 30 seconds is recommended.
- Recording capacity. Recording capacity should be adequate to capture the entire free-field record and the free vibration response of the structure after the strong shaking has reached a minimum level.
- Multiple event recordings. There should be adequate provisions to permit recording and data capture of multiple events that may occur within a short time interval, such as a few hours.

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