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REGULATORY GUIDE

Technical Lead:
Yiu Law

REGULATORY GUIDE RG 1.29

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SEISMIC DESIGN CLASSIFICATION FOR NUCLEAR POWER PLANTS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in identifying and classifying those features of light-water-reactor (LWR) nuclear power plants that must be designed to withstand the effects of the safe-shutdown earthquake (SSE).

Applicability

This guide applies to applicants and reactor licensees subject to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), 10 CFR Part 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2), and 10 CFR Part 100, "Reactor Site Criteria" (Ref. 3).

Applicable Regulations

- 10 CFR Part 50 provides for the licensing of production and utilization facilities.

  - 10 CFR Part 50.48, "Fire protection," contains the requirement for a fire protection plan for nuclear power facilities. 10 CFR 50.48(c) addresses the risk-informed, performance-based standard for fire protection programs. Classification of fire protection equipment is beyond the scope of this regulatory guide.

  - 10 CFR Part 50.55a(h) requires that safety system equipment be designed to meet their functional performance requirements over the range of normal environmental conditions during normal, abnormal and accident circumstances, including SSE.

  - Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," contains general design criteria (GDC) for nuclear power plants. GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that nuclear power plant structures,

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Electronic copies of this regulatory guide, previous versions of this guide, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/. The regulatory guide is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under ADAMS Accession No. MLML16118A148. The regulatory analysis may be found in ADAMS under Accession No. ML15131A397 the staff responses to the public comments on DG-1315 may be found under ADAMS Accession No. ML16118A149.

systems, and components (SSCs) important to safety must be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

- Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," establishes quality assurance requirements for the design, manufacture, construction, and operation of nuclear power plant SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The pertinent requirements of Appendix B apply to all activities affecting the safety-related functions of those SSCs.

- Appendix S to 10 CFR Part 50, "Earthquake Engineering Criteria for Nuclear Power Plants," requires that all nuclear power plants must be designed so that certain SSCs remain functional if the SSE ground motion occurs. These SSCs are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CRF 50.34(a)(1) or 10 CFR 100.11.

- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).

- 10 CFR Part 100 addresses the physical characteristics of a site including seismology and geology in determining the site's acceptability for a nuclear power reactor, as well as guidelines for limiting potential offsite exposure.

Related Guidance

- NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (Ref. 4) provides guidance to the NRC staff for review of Safety Analysis Reports submitted as part of license applications for nuclear power plants.

    - SRP Section 3.2.1, "Seismic Classification," provides guidance to the NRC staff in reviewing seismic classification of SSCs for nuclear power plant applications.

    - SRP Chapter 7, "Instrumentation and Controls" provides guidance on Regulatory Guide 1.97, and seismic requirements for instrumentation and control portions of systems and components in mild environments.

    - SRP Section 19.3, "Regulatory Treatment of Non-safety Systems for Passive Advanced Light Water Reactors," provides guidance to the NRC staff in reviewing the regulatory treatment of non-safety systems for which certain seismic expectations have been established so that they can withstand the SSE.

- RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" (Ref. 5), provides guidance for complying with the agency's regulations with respect to satisfying criteria for accident monitoring instrumentation in nuclear power plants.

- RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants” (Ref. 6), provides guidance on the qualification of electrical and mechanical equipment to withstand the effects of the SSE and remain functional.

- RG 1.189, “Fire Protection for Operating Nuclear Power Plants” (Ref. 7), provides guidance used to establish the design requirements for portions of fire protection SSCs to meet the requirements of GDC 2, as they relate to designing those SSCs to withstand the effects of the SSE.

- RG 1.205, “Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants” (Ref. 8), provides guidance to comply with 10 CFR 50.48(c) to implement a risk-informed, performance-based fire protection program.

- Institute of Electrical and Electronics Engineers (IEEE) Std. 497-2002, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations” (Ref. 9).

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG contains and references information collections covered by 10 CFR Parts 50, 10 CFR 52, and 10 CFR 100, that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget (OMB) control numbers 3150-0011, 3150-0151, and 3150-0093.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

This revision of the regulatory guide (Revision 5) contains minor non-substantive changes that do not present new regulatory requirements, but clarifies content in Section C, "Staff Regulatory Guidance," by (1) addition of a reference to the definition of the reactor coolant pressure boundary in 10 CFR 50.2, and (2) a reorganization of systems and subsystems to add clarity to the staff guidance. In addition, it adds a reference to a related international standard, and it was reformatted to align with the current program guidance for regulatory guides.

Background

The initial issuance of RG 1.29 in 1972 by the Atomic Energy Commission was based on reviews of a number of applications for construction permits and operating licenses for boiling-water and pressurized-water nuclear power plants. In that regulatory guide, the staff developed a seismic design classification system for identifying those plant features that must be designed to withstand the effects of the SSE. In so doing, the staff designated as seismic Category I those SSCs that must be designed to remain functional if the SSE occurs. Subsequent revisions to this regulatory guide have incorporated operating experience.

Harmonization with International Standards

The International Atomic Energy Agency (IAEA) has established a series of safety guides and standards constituting a high level of safety for protecting people and the environment. IAEA safety guides present international good practices and increasingly reflects best practices to help users striving to achieve high levels of safety. Pertinent to this regulatory guide, IAEA Safety Guide NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants" (Ref. 10), includes guidance in Sections 2.11 through 2.26 for establishing seismic categories for various SSCs based on their safety significance. This regulatory guide also addresses classification of SSCs according to their safety significance and is consistent with the basic safety principles provided in NS-G-1.6, though the implementation details are different. As described below in Section D, use of this alternative classification approach instead of the approach presented in this regulatory guide may be deemed acceptable if an applicant or licensee provides sufficient basis and information for the NRC staff to conclude that the proposed alternative demonstrates compliance with GDC 2 and 10 CFR Part 50 Appendices B and S.

C. STAFF REGULATORY GUIDANCE

1. The SSCs of a nuclear power plant that are designated as seismic Category I must be designed to withstand the effects of the SSE and remain functional. The titles and functions of these seismic Category I SSCs for LWR designs are based on existing technology from prior applications. In newer designs certain SSCs which were designated seismic Category I in previous designs may not have the same safety-related functions requiring seismic Category I classification, and certain passive SSCs in new LWR designs may be named differently from SSCs that performed similar functions in previous designs. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 shall apply to all activities affecting the safety-related functions of seismic Category I SSCs. The following SSCs of a nuclear power plant, including their foundations and supports, should be designated as seismic Category I:

   a. the reactor coolant pressure boundary as defined in 10 CFR 50.2;

   b. the reactor core and reactor vessel internals;

   c. systems[1] or portions thereof that are needed for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident containment atmosphere cleanup (e.g., hydrogen removal system);

   d. systems or portions thereof (including but not limited to systems such as residual heat removal and auxiliary feedwater) that are needed to (1) shutdown the reactor and maintain it in a safe shutdown condition, (2) remove residual heat (including heat stored within the spent fuel pool), (3) control the release of radioactive material, or (4) mitigate the consequences of an accident;

   Several key examples of systems included in items 1.c and 1.d are provided below for reference, but do not represent the complete scope of these items. Determining the complete scope of these items is the applicant's or licensee's responsibility.

   - Those portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve up to but ***not*** including the turbine stop valve, and connected piping of a nominal size of 6.35 cm (2.5 inches) or larger, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation (the turbine stop valve should be designed to withstand the SSE and maintain its integrity).

   - Those portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping of a nominal size of 6.35 cm (2.5 inches) or larger, up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation.

   - The spent fuel storage pool structure, including the fuel racks.

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[1] The system boundary includes those portions of the system required to accomplish the specified safety function and connected piping up to and including the second isolation valve or outboard containment isolation valve, such that the effects of an earthquake on non-seismic Category I portions of systems may be isolated from seismic Category I portions. This footnote applies wherever the phrase "systems or portions thereof" appears in this guide.

- The reactivity control systems (e.g., control rods, control rod drives, and boron injection system).

- The control room, including its associated equipment and all equipment needed to maintain the control room within safe habitability limits for personnel and safe environmental limits for vital equipment.

- The primary and secondary reactor containment.

e. support systems that are needed to fulfill the functions described in items 1.c and 1.d above, including but not limited to component cooling, service water, Class 1E electrical systems, fuel systems, seal water systems (such as those for reactor coolant pumps), and supporting subsystems (including the auxiliary systems for the onsite electric power supplies), and diesel fuel support systems;

f. systems or portions thereof that are needed for (1) monitoring and (2) actuating systems, as described further in Regulatory Guide 1.151, "Instrument Sensing Lines" (Ref. 11), including all electrical and mechanical devices and circuitry between the process and the input terminals of the actuator systems involved in generating signals that initiate protective action;

g. systems, other than radioactive waste management systems, not covered by items 1.a through 1.f above that contain or may contain radioactive material and the postulated failure of which would result in conservatively calculated potential offsite doses that are more than 0.005 Sievert (0.5 rem) to the whole body or its equivalent to any part of the body or total effective dose equivalent (TEDE), as applicable. Design guidance for radioactive waste systems can be found in Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 12). Guidance for dose calculations can be found in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors" (Ref. 13), Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors" (Ref. 14), and Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 15);

h. accident monitoring instrumentation - RG 1.97 provides criteria for accident monitoring instrumentation for nuclear power plants through endorsement (with exceptions) of IEEE-497-2002; and

i. those portions of SSCs of which continued function is not required but of which failure could reduce the functioning of any plant feature included in items 1.a through 1.h above to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room should be designed and constructed so that the SSE would not cause such failure. Wherever practical, structures and equipment of which failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

2. At the interface between seismic Category I and non-seismic Category I SSCs, the seismic Category I dynamic analysis requirements should be extended to either the first anchor point in the non-seismic system or a sufficient distance into the non-seismic Category I system so that the seismic Category I analysis remains valid.

3. The pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 should be applied to all activities affecting the safety-related functions of seismic Category I SSCs.

D. IMPLEMENTATION

The purpose of this section is to provide information on how applicants and licensees[2] may use this guide and information regarding the NRC's plans for using this regulatory guide. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting" and any applicable finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Applicants and Licensees

Applicants and licensees may voluntarily[3] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this regulatory guide may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged**.**

Licensees may use the information in this regulatory guide for actions which do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this regulatory guide or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this regulatory guide. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this regulatory guide, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this regulatory guide to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this regulatory guide. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the regulatory guide, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this regulatory guide, generic communication, or promulgation of a rule requiring the use of this regulatory guide without further backfit consideration.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this regulatory guide, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this regulatory guide are part of the licensing basis of the facility. However, unless this regulatory guide is part of the licensing basis for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this regulatory guide constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that

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2 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "applicants" refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

3 In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply to new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this regulatory guide or requesting or requiring the licensee to implement the methods or processes in this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 16) and NUREG-1409, "Backfitting Guidelines," (Ref. 17).

REFERENCES[4]

1. *U.S. Code of Federal Regulations* (CFR), Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," U.S. Nuclear Regulatory Commission, Washington, DC.

2. CFR, Title 10, Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, Washington, DC.

3. CFR, Title 10, Part 100, "Reactor Site Criteria," U.S. Nuclear Regulatory Commission, Washington, DC.

4. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Washington, DC.

5. NRC Regulatory Guide 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Washington, DC.

6. NRC Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," Washington, DC.

7. NRC, Regulatory Guide 1.189, "Fire Protection for Operating Nuclear Power Plants," Washington, DC.

8. NRC, Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Washington, DC.

9. Institute of Electrical and Electronics Engineers (IEEE) Std. 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations." [5]

10. International Atomic Energy Agency (IAEA) NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants," Vienna, Austria, 2003.[6]

11. NRC, Regulatory Guide 1.151, "Instrument Sensing Lines," Washington, DC.

12. NRC, Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Washington, DC.

13. NRC, Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Washington, DC.

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[4] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

[5] Copies of IEEE publications can be obtained from the IEEE Service Center through their Web site: www.IEEE.org, or by writing to IEEE Service Center, 445 Hoes Lane, Piscataway, NJ, 08855, telephone 1-800-678-4333

[6] Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: WWW.IAEA.Org/ or by writing the International Atomic Energy Agency P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria. Telephone (+431) 2600-0, Fax (+431) 2600-7, or E-Mail at Official.Mail@IAEA.Org.

14. NRC, Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Washington, DC.

15. NRC, Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Washington, DC.

16. NRC, Management Directive (MD) 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC.

17. NRC, NUREG-1409, "Backfitting Guidelines," Washington, D.C., July 1990.