U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.29, REVISION 6

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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 50.48, "Fire protection," contains the requirement for a fire protection plan for nuclear power facilities, while 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 RG.
  - 10 CFR 50.55a(h) requires that safety system equipment be designed to meet its functional performance requirements over the range of normal environmental conditions during normal, abnormal, and accident circumstances, including SSEs.
  - 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, systems, and components (SSCs) important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

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Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides. This RG 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 Number (No.) ML21155A003. The regulatory analysis may be found in ADAMS under Accession No. ML21155004.

- 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 public health and safety. 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 an 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 CFR 50.34(a)(1) or 10 CFR 100.11, "Criteria."

- 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 exposures.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Ref. 4), provides guidance to the NRC staff for the 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 the seismic classification of SSCs for nuclear power plant applications.

  - SRP Chapter 7, "Instrumentation and Controls," provides guidance to the NRC staff in reviewing instrumentation and control systems of nuclear power plants.

  - 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 nonsafety 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 Electrical and Active Mechanical Equipment and Functional Qualification of Active 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 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) provides guidance to industry for accident monitoring instrumentation for nuclear power generating stations.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50, 52, and 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), approval numbers 3150-0011, 3150-0151, and 3150-0093, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections Branch ((T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs, NEOB-10202 (3150-0011, 3150-0151,and 3150-0093), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC 20503; e- mail: oira_submission@omb.eop.gov.

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

The NRC has issued Revision 6 of this guide to correct the numbering within Section C, "Staff Regulatory Guidance." What was formerly C.1.i is now C.2. This resulted in C.2 becoming C.3 and what was C.3 became C.4. The staff made other minor edits in RG 1.29 to improve clarity and reformatted the guide to align with the current program guidance for RGs.

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 RG, 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 RG have incorporated operating experience.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from the harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports, reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered IAEA Safety Requirements and Safety Guides pursuant to the Commission's International Policy Statement (Ref. 10) and Management Directive and Handbook 6.6, "Regulatory Guides," dated May 2, 2016 (Ref. 11).

Pertinent to this RG, IAEA Safety Guide NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants," issued 2003 (Ref. 12), includes guidance in Sections 2.11 through 2.26 for establishing seismic categories for various SSCs based on their safety significance. This RG also addresses the classification of SSCs according to their safety significance and is consistent with the basic safety principles provided in NS-G-1.6, although the implementation details are different. As described below in Section D, use of this alternative classification approach instead of the approach presented in this RG 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 that 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, "Definitions";

   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) shut down 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 centimeters (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

[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 nonseismic 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 spent fuel storage pool structure, including the fuel racks
- 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 RG 1.151, "Instrument Sensing Lines" (Ref. 13), 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, noting that design guidance for radioactive waste systems can be found in RG 1.143, "Design Guidance for Radioactive Waste Management Systems Structures and Components Installed in Light-Water-Cooled Nuclear Power Plants" (Ref. 14), and that guidance for dose calculations can be found in RG 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling-Water Reactors" (Ref. 15), RG 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized-Water Reactors" (Ref. 16), and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (Ref. 17); and

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

2. Those portions of SSCs whose continued function is not required but whose 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 whose failure could possibly cause such injuries should be relocated or separated to the extent required to eliminate that possibility.

3. At the interface between seismic Category I and nonseismic Category I SSCs, the seismic Category I dynamic analysis requirements should be extended to either the first anchor point in

the nonseismic system or a sufficient distance into the nonseismic Category I system so that the seismic Category I analysis remains valid.

4. 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 NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, “Backfitting,” and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests” (Ref. 19), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

REFERENCES[2]

1. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."

2. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

3. CFR, "Reactor Site Criteria," Part 100, Chapter 1, Title 10, "Energy."

4. U.S. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

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

6. NRC, RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Washington, DC.

7. NRC, RG 1.189, "Fire Protection for Nuclear Power Plants," Washington, DC.

8. NRC, RG 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." [3]

10. NRC, "Nuclear Regulatory Commission International Policy Statement," *Federal Register*, Vol. 79, No. 132, July 10, 2014, pp. 39415–39418.

11. NRC, Management Directive (MD) 6.6, "Regulatory Guides," Washington, DC, May 2, 2016 (ADAMS Accession No. ML18073A170).

12. International Atomic Energy Agency (IAEA) NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants," Vienna, Austria, 2003.[4]

13. NRC, RG 1.151, "Instrument Sensing Lines," Washington, DC.

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[2] Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at https://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at https://www.nrc.gov/reading-rm/adams.html. The documents can also be viewed on line 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.

[3] Copies of IEEE documents can be purchased from the IEEE Service Center through its Web site: www.IEEE.org, or by writing to IEEE Service Center, 445 Hoes Lane, P.O. Box 1331, Piscataway, NJ, 08855, telephone (800) 678-4333.

[4] Copies of International Atomic Energy Agency (IAEA) documents may be obtained through its 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 by e-mail at Official.Mail@IAEA.Org.

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

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

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

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

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

19. NRC, MD 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, DC.