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DRAFT REGULATORY GUIDE

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(Proposed Revision 1 of Regulatory Guide 1.57, dated June 1973)

Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

A. INTRODUCTION

This draft regulatory guide describes an approach that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in satisfying the requirements of General Design Criteria (GDC) 1, 2, 4, and 16, as specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, of the *Code of Federal Regulations* (10 CFR Part 50), "Domestic Licensing of Production and Utilization Facilities." Specifically, GDC 1, "Quality Standards and Records," requires, in part, that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. To augment those requirements, GDC 2, "Design Bases for Protection Against Natural Phenomena," requires that structures important to safety be designed to withstand the effects of expected natural phenomena when combined with the effects of normal accident conditions without loss of capability to perform their safety function. In addition, to ensure that the containment of a nuclear power plant is designed to withstand natural phenomena, it is necessary to specify the most severe natural phenomena event that may occur as a function of the frequency of occurrence. Similarly, GDC 4, "Environmental and Dynamic Effects, Design Bases," requires that nuclear power plant SSCs important to safety be designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). In addition, GDC 16, "Containment Design," requires that the reactor containment and its associated systems be provided to establish an essentially leaktight barrier against uncontrolled release of radioactivity to the environment and to ensure that design conditions important to safety are not exceeded for as long as required for postulated accident conditions.

10 CFR 50.44 provides the requirements for combustible gas control for currently-licensed reactors and future water-cooled reactor applicants and licensees. This draft regulatory guide describes an approach that the NRC staff considers acceptable for use in considering the structural loads involved and determining the containment response to demonstrate the structural integrity of the containment.

Moreover, leaktightness of the containment structure must be tested at regular intervals during the life of the plant, in accordance with the provisions of 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Finally, certain reactors specified in 10 CFR 50.34(f)(3)(v)(A) and (B) require steel containments to meet specific provisions of the Boiler and Pressure Vessel (B&PV) Code promulgated by the American Society of Mechanical Engineers (ASME), when subjected to loads resulting from fuel damage, metal-water reactions, hydrogen burning, and inerting system actuations.

Meeting these criteria provides assurance that steel containments used for nuclear power plants will be designed to be capable of performing their containment function as long as required to prevent or mitigate the spread of radioactive material, and that they can withstand the effects of natural phenomena and other external events and maintain the plant in a safe condition.

The NRC issues regulatory guides 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 accidents, and to provide guidance to applicants. Regulatory guides are not substitutes for regulations, and compliance with regulatory guides is not required. The NRC issues regulatory guides in draft form to solicit public comment and involve the public in developing the agency's regulatory positions. Draft regulatory guides have not received complete staff review and, therefore, they do not represent official NRC staff positions.

This regulatory guide contains information collections, covered by the requirements of 10 CFR Part 50, that the Office of Management and Budget (OMB) has approved under OMB control number 3150-0011. The NRC may neither conduct nor sponsor, and a person is not required to respond to, an information collection request or requirement unless the requesting document displays a currently valid OMB control number.

B. DISCUSSION

Background

The American Society of Mechanical Engineers (ASME) publishes the “Rules for Construction of Nuclear Facility Components,” as Section III of the ASME B&PV Code.[1] In that section, Division 1, Subsection NE sets forth the rules for Class MC components, which include metal containments and appurtenances, as well as metal portions of concrete containments that are not backed by concrete. (ASME B&PV Code Section III, Division 1, is hereinafter referred to as “the Code.”) However, the existing industry codes and standards are based on the current class of light-water reactors and, as such, may not adequately address design and construction features of the next generation of advanced light-water reactors (ALWRs), including the AP1000 and Combustion Engineering (CE) 80+ designs.

The NRC is committed to the use of industry consensus codes and standards for the design, construction, and licensing of commercial nuclear power reactors facilities. Thus, the recent significant advancement in the technology (both in the nuclear industry and the Code) has prompted a need to revise the regulatory guidance for metal containments. Toward that end, this draft regulatory guide provides guidance on the use of codes and standards for the design of advanced reactors to ensure that SSCs will perform their intended safety functions. While this regulatory guide is only directly applicable to light-water reactor metal containments, the principles may be applied to non-light water reactor containments, subject to review by the NRC.

10 CFR 50.44(b)(2)(i) requires that all currently licensed boiling-water reactors with Mark I or Mark II type containments must have an inerted atmosphere. Also, 10 CFR 50.44(b)(2)(ii) requires that all currently licensed boiling-water reactors with Mark III type containments and all pressurized-water reactors with ice condenser containments must have the capability to control combustible gas generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region, so that there is no loss of containment structural integrity. In addition, 10 CFR 50.44(b)(5)(v)(B) requires that all currently licensed boiling-water reactors with Mark III type containments and all pressurized-water reactors with ice condenser containments, must demonstrate that systems and components necessary to establish and maintain safe shutdown and containment integrity will be capable of performing their functions during and after exposure to the environmental conditions created by burning hydrogen, including local detonations, unless such detonations can be shown to be unlikely to occur.

By contrast, 10 CFR 50.44(c)(3) requires that future water-cooled reactors containments that do not rely upon an inerted atmosphere to control combustible gases must have the capability to control combustible gas generated from a metal-water reaction involving 100 percent of the fuel cladding surrounding the active fuel region, so that there is no loss of containment structural integrity. Also, 10 CFR 50.44(c)(5) requires that for future water-cooled reactors containments, an applicant must perform an analysis that demonstrates containment structural integrity. This demonstration must use an analytical technique that is accepted by the NRC, and must include sufficient supporting justification to show that the technique describes the containment response to the structural loads involved. The analysis must address an accident that releases hydrogen generated from 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.

To address the requirements of 10 CFR 50.34(f) and 10 CFR 50.44(b) and (c), Regulatory Position 1.2.3.3 (in Section C of this guide) provides load combinations for pressure loads that result from a fuel-clad metal-water reaction, an uncontrolled hydrogen burn, and a post-accident environment inerted by carbon dioxide.

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1 ASME Boiler and Pressure Vessel Code, Section III, “Nuclear Components,” Division 1, including that part of the Summer 2003 Addenda.

The design conditions and functional requirements of components that provide a pressure boundary for the primary reactor containment function should be reflected in the application of appropriate design limits (e.g., stress or strain limits) for the most adverse combination of loadings to which these components might be subjected. For components constructed in accordance with Subsection NE (Code Class MC) of the Code, the NRC requires provision of a design specification, which stipulates the design requirements (e.g., the mechanical and operational loadings) for the components.

In Appendix B to the Code, entitled "Owner's Design Specifications," Paragraph B-2125, "Load Combinations," states, "In order to provide a complete definition of service loads, the combination of specific events must be considered. Since these combinations are a function of specific systems which make up a part of a specific type of nuclear facility, this section does not directly address service loads other than to provide different stress limits for various loadings."

To further provide a consistent basis for the design of metal containment system components, this guide delineates acceptable design limits for appropriate combinations of loadings. The intent is to address only the most adverse combinations of loadings resulting from those events or conditions identified herein (e.g., those combinations of loadings that result in the limiting or controlling design condition). These loadings are associated with conditions for which the containment function is required, in combination with specified seismic events producing possible mechanisms for failure that could affect the function and/or integrity of structures, systems, and components important to safety. Included in the latter are the loadings associated with the vibratory motion of the safe-shutdown earthquake (SSE), design external pressure (if applicable), and other loadings that induce compressive stresses. The effects of natural phenomena other than earthquakes, such as tornadoes, hurricanes, and floods, are not considered in this guide, because a Category I concrete shield building typically protects the steel containment from the effects of tornadoes, hurricanes, and floods occurring outside the shield building. In addition to the loading combinations addressed in this guide, primary reactor containment components enclosed within Seismic Category I structures should be designed to withstand the effects of pertinent natural phenomena not otherwise protected against.

The approach set forth in this guide is directly related to Section III of the Code. Design limits as specified in Section III are adopted to provide assurance of maintaining the pressure-retaining integrity of the primary reactor containment. The primary reactor containment system of metal construction includes all components that perform a containment function, such as (1) the containment vessel(s), (2) penetration assemblies and access openings, and (3) piping systems attached to the containment vessel nozzles or penetration assemblies out to and including all pumps and valves required to isolate the containment. The only components that are classified as ASME Code Class MC (i.e., components constructed in accordance with the rules of Subsection NE of the Code) are metal containment vessels, including parts and appurtenances thereof.[2] Such parts and appurtenances may include mechanical, electrical, and piping penetration assemblies,[3] bellows-type expansion joints, and access openings. Piping, pumps, and valves that are defined as components of primary metal containment systems are constructed in accordance with the rules for either Code Class 1 or Code Class 2 components, as required by Article NE-1120. Any piping penetration assemblies or appurtenances that are not a part of the containment vessel should be constructed in accordance with the rules for Code Class 1 or Code Class 2 components, as required by the intended service function.

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2 Refer to NCA-9200 of the Code for definitions of "parts" and "appurtenances."

3 Penetration assemblies are parts or appurtenances that are required to permit piping, mechanical devices, and electrical connections to pass through the containment vessel shell or head and maintain leaktight integrity, while compensating for such things as temperature and pressure fluctuations and earthquake movements.

C. REGULATORY POSITION

1. Code Class MC vessels, electrical and mechanical penetration assemblies, and other penetration assemblies (excluding bellows-type expansion joints) that are parts or appurtenances of the vessel.

For earthquake engineering criteria, 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," would be applicable for the operating-basis earthquake (OBE) and safe-shutdown earthquake (SSE). In this manner, the OBE serves the function as an inspection-level earthquake below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage.

Code Class MC components of primary metal containment systems that are completely enclosed within Seismic Category I structures[4] should be designed to withstand the following loads and loading combinations within the specified design limits.

1.1 <u>Loads</u>

D ------- Dead loads.

L ------- Live loads, including all loads resulting from platform flexibility and deformation, as well as crane loading, if applicable.

P_t ------- Test pressure.

T_t ------- Test temperature.

T_o ------ Thermal effects and loads during startup, normal operating, or shutdown conditions, based on the most critical transient or steady-state condition.

R_o ------ Pipe reactions during startup, normal operating, or shutdown conditions based on the most critical transient or steady-state condition.

P_o------- External pressure loads resulting from pressure variation either inside or outside containment.

E ------- Loads generated by the operating-basis earthquake including sloshing effects, if applicable.

E' ------ Loads generated by the SSE, including sloshing effects.

P_a------- Pressure load generated by the postulated pipe break accident, pool swell, and subsequent hydrodynamic loads.[5]

T_a------- Thermal loads under thermal conditions generated by the postulated pipe break accident, including T_o, pool swell, and subsequent hydrodynamic reaction loads.[5]

R_a ------ Pipe reactions under thermal conditions generated by the postulated pipe break accident, including R_o, pool swell, and subsequent hydrodynamic reaction loads.[5]

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4 Components of primary reactor containment systems are Seismic Category I for seismic design purposes in accordance with Regulatory Guide 1.29, "Seismic Design Classification." Seismic Category I SSCs are designed to remain functional if the SSE occurs.

5 For load combinations 1.2.3.1(5), 1.2.3.3(3), and 1.2.3.4(2), a small or intermediate pipe break is postulated. For all other load combinations involving a loss of coolant accident (LOCA), the design-basis LOCA is postulated.

P_s ------- All pressure loads that are caused by the actuation of safety relief valve (SRV) discharge, including pool swell and subsequent hydrodynamic loads.

T_s ------- All thermal loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic thermal loads.

R_s ------ All pipe reaction loads that are generated by the actuation of SRV discharge, including pool swell and subsequent hydrodynamic reaction loads.

Y_r ------ Equivalent static load on the structure generated by the reaction on the broken pipe during the design-basis accident.

Y_j ------- Jet impingement equivalent static load on the structure generated by the broken pipe during the design-basis accident.

Y_m ----- Missile impact equivalent static load on the structure generated by or during the design-basis accident, such as pipe whipping.

F_L ------ Load generated by the post-LOCA flooding of the containment, if any.

P_{g1} ------ Pressure load generated from fuel clad metal-water reaction.

P_{g2} ------ Pressure loads generated by hydrogen burning.

P_{g3} ------ Pressure load from post-accident inerting, assuming carbon dioxide is the inerting agent.

1.2 <u>Loading Combinations and Design Limits</u>

The specified loads and load combinations are acceptable if found to be in accordance with the following guidance. The following load combinations include all loading combinations for which the containment might be designed for or subjected to during the expected life of the plant.

1.2.1 *Testing Condition*

This includes the testing condition of the containment to verify its leak integrity. In this case, the loading combination includes:

$D + L + T_t + P_t$

1.2.2 *Design Conditions*

These include all design loadings for which the containment vessel or portions thereof might be designed for during the expected life of the plant. Such loads include design pressure, design temperature, and the design mechanical loads generated by the design basis accident. In this case, the loading combination includes:

$D + L + P_a + T_a + R_a$

1.2.3 *Service Conditions*

The load combinations in these cases correspond to and include Level A service limits, Level B service limits, Level C service limits, Level D service limits and the post-flooding condition. The loads may be combined by their actual time history of occurrence, taking into consideration their dynamic effect upon the structure.

1.2.3.1 Level A Service Limits

These service limits are applicable to the service loadings to which the containment is subjected, including the plant or system design-basis accident conditions for which the containment function is required, excepting only those categorized as Level B, C, or D, or Testing Loadings.
The loading combinations corresponding to these limits include the following:

(1) Normal operating plant condition

$D + L + T_o + R_o + P_o$

(2) Operating plant condition in conjunction with multiple SRV actuations

$D + L + T_s + R_s + P_s$

(3) Loss-of-coolant accident

$D + L + T_a + R_a + P_a$

(4) Multiple SRV actuations in combination with a small- or intermediate-break accident

$D + L + T_a + R_a + P_a + T_s + R_s + P_s$

(5) Normal operating plant conditions in combination with inadvertent full actuation of a post-accident inerting hydrogen control system

$D + L + T_o + R_o + P_o + P_{g3}$

(6) Pressure test load to ensure that the containment will safely withstand the pressure calculated to result from carbon-dioxide inerting

$D + 1.10 \times P_{g3}$

1.2.3.2 Level B Service Limits

These service limits include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena during which the plant must remain operational. The loading combinations corresponding to these limits include the following:

(1) Loss-of-coolant accident in combination with the operating-basis earthquake

$D + L + T_a + R_a + P_a + E$

(2) Operating plant condition in combination with the operating-basis earthquake

$D + L + T_o + R_o + P_o + E$

(3) Operating plant condition in combination with the operating-basis earthquake and multiple SRV actuations

$D + L + T_s + R_s + P_s + E$

(4) Loss-of-coolant accident in combination with a single active component failure causing one SRV discharge

$D + L + T_a + P_a + R_a + T_s + R_s + P_s$

1.2.3.3 Level C Service Limits

These service limits include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required. The loading combinations corresponding to these limits include the following:

(1) Loss-of-coolant accident in combination with the SSE

$D + L + T_a + R_a + P_a + E'$

(2) Operating plant condition in combination with the SSE

$D + L + T_o + R_o + P_o + E'$

(3) Multiple SRV actuations in combination with a small- or intermediate-break accident and SSE

$D + L + T_a + R_a + P_a + T_s + R_s + P_s + E'$

(4) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by hydrogen burning

$D + P_{g1} + P_{g2}$

[NOTE: In this load combination, $P_{g1} + P_{g2}$ should not be less than 310 kPa (45 psig).]

(5) Dead load plus pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by the added pressure from post-accident inerting, assuming carbon dioxide as the inerting agent

$D + P_{g1} + P_{g3}$

[NOTE: In this load combination, $P_{g1} + P_{g3}$ should not be less than 310 kPa (45 psig).]

1.2.3.4 Level D Service Limits

These service limits include other applicable service limits and loadings of a local dynamic nature for which the containment function is required. The load combinations corresponding to these limits include the following:

(1) Loss-of-coolant accident in combination with the SSE and local dynamic loadings

$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_m + E'$

(2) Multiple SRV actuations in combination with a small- or intermediate-break accident, SSE, and local dynamic loadings

$D + L + T_a + R_a + P_a + Y_r + Y_j + Y_j + P_s + T_s + R_s + E'$

1.2.3.5 Post-Flooding Condition

This includes post-LOCA flooding of the containment in combination with the operating-basis earthquake

$D + L + F_L + E$

1.3 Design Limits

Total stresses for the combination of loads delineated in Regulatory Position 1.2 (above) are acceptable if found to be within the limits defined by Articles NE-3221.1, NE-3221.2, NE-3221.3 and NE-3221.4 of the Code.

1.4 Treatment of Buckling Effects

Earthquake, thermal, and pressure loads require consideration of buckling of the shell. An acceptable approach to this problem is to perform a nonlinear analysis. Buckling of shells with more complex shell geometries and loading conditions than those covered by Article NE-3133 of the Code should be considered in accordance with the criteria described in ASME Code Case N-284-2, pending endorsement in Regulatory Guide 1.84.[6]

2. Bellows-Type Expansion Joints that are Parts or Appurtenances of ASME Code Class MC Vessels

Bellows-type expansion joints that are parts or appurtenances of Code Class MC components that are completely enclosed within Seismic Category I structures should be designed to withstand the loads and loading combinations within the design limits specified in Regulatory Position 1 (above), as applicable, supplemented by the design limits specified in Article NE-3366.2(b) of the Code.

[6] Code Case N-284, "Metal Containment Shell Buckling Design Methods, Class MC Section III, Division 1," is currently being revised. Revision 1 of N-284 is unacceptable to the NRC, as discussed in Regulatory Guide 1.193. Revision 2 of N-284 is correcting errata, misprints, recommendations, and errors identified by the NRC staff, and is expected to be approved when it is published.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this draft regulatory guide. No backfitting is intended or approved in connection with its issuance.

The NRC has issued this draft guide to encourage public participation in its development. Except in those cases in which an applicant or licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods to be described in the active guide will reflect public comments and will be used in evaluating (1) submittals in connection with applications for construction permits, standard plant design certifications, operating licenses, early site permits, and combined licenses; and (2) submittals from operating reactor licensees who voluntarily propose to initiate system modifications if there is a clear nexus between the proposed modifications and the subject for which guidance is provided herein.

REGULATORY ANALYSIS

1. Statement of the Problem

The U.S. Nuclear Regulatory Commission (NRC) issued Regulatory Guide 1.57 in June 1973 to describe design limits and loading combinations for metal primary reactor containment system components that the NRC staff considered acceptable for use in nuclear power plants to prevent undue risk to the health and safety of the public. Reasons to revise the existing regulatory guide include (1) making it consistent with other similar regulatory guides, (2) including interim or current staff positions regarding new or advanced reactors, (3) reducing unnecessary conservatism, and (4) updating in accordance with the ASME Boiler & Pressure Vessel Code and removing some of the exceptions that the staff had previously taken from the1971 edition and addenda (because more recent editions have included those exceptions). Therefore, revision of this regulatory guidance is necessary to reflect current conditions.

2. Objective

The objective of this regulatory action is to update the NRC's guidance with respect to design limits and loading combinations for metal containments. This will give applicants and licensees the opportunity to take advantage of the Code requirements and current staff positions, which should lead to increased regulatory effectiveness by avoiding unnecessary conservatism that offers little or no safety benefit.

3. Alternative Approaches

The NRC staff considered the following alternative approaches to the problem of outdated guidance regarding metal containments:

(1) Do not revise Regulatory Guide 1.57.
(2) Update Regulatory Guide 1.57.

3.1 <u>Alternative 1: Do Not Revise Regulatory Guide 1.57</u>

Under this alternative, the NRC would not revise this guidance, and licensees would continue to use the 1973 version of this regulatory guide. This alternative is considered the baseline or "no action" alternative and, as such, involves no value/impact considerations.

3.2 Alternative 2: Update Regulatory Guide 1.57

Under this alternative, the NRC would update Regulatory Guide 1.57 with the updated Code requirements and current staff positions.

The benefit of this action would be to include the latest code requirements and current staff position. Therefore, the updated Regulatory Guide 1.57 could improve regulatory effectiveness and reduce unnecessary conservatism, while maintaining or enhancing safety.

The costs to the NRC would be the one-time cost of issuing the revised regulatory guide (that is, relatively small), and applicants and licensees would incur little or no cost. Therefore, any adverse consequences of adopting this alternative are considered extremely remote.

4. Conclusion

Based on this regulatory analysis, the staff recommends that the NRC revise Regulatory Guide 1.57. The staff concludes that the proposed action will reduce unnecessary conservatism in the design limits and loading combinations for metal containments, leading to a more efficient regulatory process, while maintaining or enhancing safety.

BACKFIT ANALYSIS

This draft regulatory guide provides licensees and applicants with new guidance that the NRC staff considers acceptable for use in design and analysis of metal primary reactor containments in nuclear power plants. The application of this guide is voluntary. Licensees may continue to use the original version of this regulatory guide if they so choose. No backfit, as defined in 10 CFR 50.109, is either intended or implied.

REFERENCES

American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components, Division 1, Subsection NE, Class MC Components," 2001 Edition including 2003 Addenda.[7]

Regulatory Guide 1.29, "Seismic Design Classification," Revision 3, U.S. Nuclear Regulatory Commission, Washington, DC, September 1978, available in ADAMS under Accession #ML003739983.[8]

Regulatory Guide 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Revision 33, U.S. Nuclear Regulatory Commission, Washington, DC, August 2005, available in ADAMS under Accession # ML052130562.

Regulatory Guide 1.193, "ASME Code Cases Not Approved for Use," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, August 2005, available in ADAMS under Accession #ML052140501.

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7 Copies of the Code and addenda thereto may be obtained from the American Society of Mechanical Engineers, Three Park Avenue, New York, New York 10016-5990.

8 All regulatory guides listed herein were published by the U.S. Nuclear Regulatory Commission. Where an ADAMS accession number is identified, the specified regulatory guide is available electronically through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html. All other regulatory guides are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. Single copies of regulatory guides may also be obtained free of charge by writing the Reproduction and Distribution Services Section, ADM, USNRC, Washington, DC 20555-0001, or by fax to (301)415-2289, or by email to DISTRIBUTION@nrc.gov. Active guides may also be purchased from the National Technical Information Service (NTIS)on a standing order basis. Details on this service may be obtained by contacting NTIS at 5285 Port Royal Road, Springfield, Virginia 22161, online at http://www.ntis.gov, or by telephone at (703) 487-4650. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room (PDR), which is located at 11555 Rockville Pike, Rockville, Maryland; the PDR's mailing address is USNRC PDR, Washington, DC 20555-0001. The PDR can also be reached by telephone at (301) 415-4737 or (800) 397-4205, by fax at (301) 415-3548, and by email to PDR@nrc.gov.

U.S. Code of Federal Regulations, Title 10, *Energy,* Part 50, "Domestic Licensing of Production and Utilization Facilities."[9]

U.S. Code of Federal Regulations, Title 10, *Energy,* Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

U.S. Code of Federal Regulations, Title 10, *Energy*, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

U.S. Code of Federal Regulations, Title 10, *Energy*, Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors"

U.S. Code of Federal Regulations, Title 10, *Energy*, Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

U.S. Code of Federal Regulations, Title 10, *Energy*, Part 100, Appendix A, "Seismic and Geological Siting Criteria for Nuclear Power Plants."

9 All NRC regulations listed herein are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/reading-rm/doc-collections/cfr/. Copies are also available for inspection or copying for a fee from the NRC's Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301) 415-4737 or (800) 397-4209; fax (301) 415-3548; email PDR@nrc.gov.