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DRAFT REGULATORY GUIDE

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DRAFT REGULATORY GUIDE DG-1159

(Proposed Revision 3 of Regulatory Guide 1.136, dated June 1981)

Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments

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, 16, and 50, 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 SSCs important to safety be designed to withstand the effects of expected natural phenomena when combined with the effects of normal and accident conditions. 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 a 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. Finally, GDC 50, "Containment Design Basis," requires that the reactor containment structure (including access openings, penetrations, and containment heat removal systems) be designed so that the structure and its internal compartments will have the capability to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions caused by a LOCA.

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.

In addition, for certain reactors specified in 10 CFR 50.34, "Contents of Applications: Technical Information," requires that plant designs must accommodate loadings associated with hydrogen generation that results from metal-water reaction of the fuel cladding accompanied by hydrogen burning or the added pressure of inerting system actuation. In addition, 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," requires, in part, that measures must be established to ensure materials control and control of special processes, such as welding, and proper testing must be performed.

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

The American Society of Mechanical Engineers (ASME) and the American Concrete Institute (ACI) have jointly published the "Code for Concrete Containments," also known as either the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, Division 2, or ACI Standard 359-01, which this guide refers to as "the Code." This regulatory guide endorses the 2001 Edition of the Code with the 2003 addenda, with the exceptions discussed herein.

Significant advancement in technology, both in the nuclear industry and the Code, has prompted a need to revise this regulatory guide for concrete containments. 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 and high-temperature gas-cooled reactors. Nonetheless, the NRC remains committed to the use of industry consensus codes and standards for the design, construction, and licensing of commercial nuclear power reactors facilities. Toward that end, this guide describes methods that the NRC staff considers acceptable with regard to the materials, design, construction, and testing of reinforced and prestressed concrete containments. As such, the provisions of this guide may be used for the current light-water reactors, as well as future advanced reactors, such as the Advanced Pressurized-Water Reactor (AP1000) and the Economic Simplified Boiling-Water Reactor (ESBWR).

The NRC staff has evaluated the provisions contained in the Articles CC-1000 through CC-6000 of the Code, and is in the process of coordinating related updates to other regulatory guides, as well as the "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800). As a result, this draft regulatory guide endorses Articles CC-1000 through CC-6000 of the Code, with the exceptions noted herein. This draft regulatory guide also provides guidance on loads and load combinations, design and analysis, and a method for determining the ultimate capacity of a concrete containment. In addition, this guide reviews the quality control program proposed for the fabrication and construction of the containment, with emphasis on the extent of compliance with Articles CC-4000 and CC-5000 of the Code, including the following:

> Examination of the materials, including tests to determine the physical properties of concrete, reinforcing steel, mechanical splices, the liner plate and its anchors, and the prestressing system, if any; placement of concrete; and erection tolerances of the liner plate, reinforcement, and prestressing systems.

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 5 (in Section C of this guide) provides loads and 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.

In addition to the above discussion 10 CFR 50.55a also imposes the examination requirements established in Section XI, Subsection IWL, of the Boiler and Pressure Vessel (B&PV) Code promulgated by the American Society of Mechanical Engineers (ASME), as they relate to reinforced and prestressed concrete (Class CC) containments.

In those areas where the provisions of the referenced Code are insufficient for licensing purposes, the staff has provided supplementary guidelines, as part of the regulatory position presented in Section C of this guide.

The reasons for the supplementary guidance for each regulatory position are as follows.

CC-2243: Cement Grout for Grouted Tendon Systems[1]

Regulatory Position 1 recommends using the guidance in Regulatory Guide 1.107, "Qualification for Cement Grouting for Prestressing Tendons in Containment Structures," rather than Article CC-2243, with respect to grouting of prestressing tendons. The staff believes that the recommendations presented herein provide needed assurance regarding the integrity of grouted tendons that cannot be directly inspected during the life of the containment.

CC-2433.2.3: Acceptance Standards

Experience with the use of alloy steel materials for anchor blocks and wedge blocks (such as AISI 4140) indicates that a high degree of hardness of these materials is a factor in causing cracking (presumably stress-corrosion) under certain inevitable environments. Also, it is necessary to control the uniformity of the hardness of these materials. A thorough surface examination and proper protection before and after installation of these materials, together with close control of the amount and uniformity of hardness in these materials, may eliminate cracking. Regulatory Position 2 provides guidance in addition to the Code specifications.

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1 This alphanumeric citing identifies the article, and paragraph if applicable, of the "Code for Concrete Reactor Vessels and Containments" being discussed.

CC-2434: Wedges and Anchor Nuts

The testing of prestressing materials to qualify them against loss of ductility in cold temperatures is needed; therefore, the guidance in Regulatory Position 3 is recommended.

CC-2463.1: Static Tensile Test

Various prestressing systems may require different numbers of tests for tendon systems to establish their adequacy for use. Variations within the tolerance limits of the construction specification for material properties and geometry of anchorages and tendons must be realistically and adequately represented in the system testing. Therefore, Regulatory Position 4 recommends that any system of prestressing should be subjected to a sufficient number of tests to establish its adequacy before it is adopted for use.

CC-3000: Design

To facilitate design and analysis procedures, Regulatory Position 5 provides some specific guidance in addition to the Code specification in order to be consistent with the current staff position. That is, this revision adds load and loading combination guidelines for design and analysis of concrete containments. This method has been used by the NRC staff in reviewing new reactor applications, and is being considered for inclusion in the update to SRP 3.8.1, "Concrete Containment."

CC-3542: Loss of Prestress

Regulatory Position 6 recommends using the guidance in Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," rather than Article CC-3542 with respect to loss of prestress in tendons. The staff believes that the recommendations presented herein provide needed assurance regarding the loss of tendon prestress.

CC-4240: Curing

The Code does not have any provision for curing concrete at temperatures higher than 4.4 °C (40 °F). Consequently, Regulatory Position 7 is in accordance with ACI 308.1-98, "Standard Specification for Curing Concrete."

CC-4352: Splices

Welded splices and other mechanical connections are allowed as long as they conform to ACI-349-01, Section 12.14.3. (See Regulatory Position 8.)

CC-4470: Permanent Corrosion Protection

See the discussion concerning CC-3542, "Loss of Prestress," and Regulatory Position 9.

CC-5210: General

The locations of all major embedments, such as plates, embedded piping penetration sleeves, major structural framings, and anchor bolts, should be preplanned, identified on the design drawings, and documented on field changes thereto. This would permit verification that embedments have been placed with full consideration given to the resulting reduction in structural strengths, radiation shielding effectiveness, and hindrance to the placement and consolidation of concrete. In this regard, Regulatory Position 10 provides guidance in addition to the Code.

CC-6430: Retest

The third sentence of Article CC-6430 provides two options if conditions of Article CC-6410 are not met and further studies still indicate that requirements of Article CC-6410 are not fulfilled. Regulatory Position 11 discusses the need to select one of those two options.

Ultimate Capacity of Concrete Containment

This revised regulatory guide also adds new guidelines for the ultimate capacity of concrete containments, in order to be consistent with the current staff position. This method has been used by the NRC staff in reviewing new reactor applications, and is being considered for inclusion in the update to SRP 3.8.1, "Concrete Containment."

C. REGULATORY POSITION

The design, materials, fabrication, erection, inspection, testing, and inservice surveillance of concrete containments are covered by codes, standards, specifications, and guides that are applicable either in their entirety or in part. In addition to this regulatory guide, the following codes and guides are acceptable to the staff.

Code	Title
ASME, Section III, Division 2	Code for Concrete Containments
ASME, Section III, Subsection NCA	General Requirements for Division 1 and Division 2

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.

The requirements specified in Articles CC-1000 through CC-6000 of the Code are acceptable to the NRC staff for the scope, material, design, construction, examination, and testing of concrete containments of nuclear power plants subject to the following regulatory positions.

1. CC-2243: Cement Grout for Grouted Tendon Systems

Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures," should be used for guidance on qualifying grout for grouted tendon systems.

2. CC-2433.2.3: Acceptance Standards

In addition to the requirements in CC-2433.2.3, "Acceptance Standards," the following guidance should be used:

> The maximum hardness for material of anchor head assemblies and wedge blocks shall not exceed that of Rockwell C40. To maintain uniformity in hardness, the tolerance on a designated hardness number shall not exceed ±2.

3. CC-2434: Wedges and Anchor Nuts

In addition to the requirements in CC-2434, "Wedges and Anchor Nuts," the following guidance should be used to protect prestressing materials from low-temperature effects:

> Materials for all load-bearing components of prestressing systems should be selected so that they can withstand the anticipated low-temperature effects without loss in their ductility. Methods and procedures similar to those used for materials of liners in CC-2520, "Fracture Toughness Requirements for Materials," are acceptable for qualifying the materials. Additionally, suitable tests should be conducted to demonstrate that with the maximum allowable flaw size (cracked buttonheads, wedges, and anchor nuts), the specific components will exhibit the required strength and ductility under the lowest anticipated temperatures.

4. CC-2463.1: Static Tensile Test

In addition to the requirements of CC-2463.1, “Static Tensile Test,” the following guidance should be used:

> Any system of prestressing should be subjected to a sufficient number of tests to establish its adequacy. Justification that a sufficient number of tests have been performed, as well as a description of the test program, should be submitted to the NRC for review.

5. CC-3000: Design

Design and analysis procedures for structural portions of the containment, and specified allowable limits for stresses and strains, should be in accordance with Article CC-3000 of the Code, with the following considerations:

A. The specified loads and load combinations are acceptable if found to be in accordance with Article CC-3000 of the Code, with the exceptions listed below taken to the requirements specified in Table CC-3230-1:

(1) Hydrodynamic loads resulting from LOCA and/or safety/relief valve (SRV) actuation should be combined as indicated in the appendix to SRP Section 3.8.1.

(2) Where post-LOCA flooding is a design consideration, the following combination should also be considered in the factored load category:

$D + 1.0\ L + 1.0\ F + 1.0\ E_0$, where D, L, and E_0 are as defined in the Code, and F is the load generated by the post-LOCA flooding of the containment.

The requirements of Subarticle CC-3720 of the Code should be met when the containment structure is exposed to the following loading conditions:

(1) For the Factored Load Category:

$D + P_{g1} + [P_{g2} \text{ or } P_{g3}]$ where

D = Dead load

P_{g1} = Pressure resulting from an accident that releases hydrogen generated from 100% fuel clad metal-water reaction

P_{g2} = Pressure resulting uncontrolled hydrogen burning

P_{g3} = Pressure resulting from postaccident inerting assuming carbon dioxide is the inerting agent

(2) For the Service Load Category, the strains in the containment liner shall not exceed the limits set forth in Subarticle CC-3720 when exposed to pressure P_{g3}.

(3) As a minimum design condition for either condition 1 or 2 above, the following load combination must be satisfied:

D + 310 kPa (45 psig)

B. For the structural portions of the containment, the specified allowable limits for stresses and strains should be in accordance with Subarticle CC-3400 of the Code, but with the following exceptions:

CC-3421.5

For existing plants, the tangential shear stress carried by the concrete in reinforced concrete containments should be limited to 276 kPa (40 psi) and 414 kPa (60 psi) for the load combinations of Table CC-3230-1, representing abnormal/severe environmental conditions and abnormal/extreme environmental conditions, respectively.

For new plants, tangential shear stress carried by the concrete in reinforced concrete containments should be zero in accordance with the Code. This position is also supported by the ASME Code and the test data contained in NUREG/CR-5209, "Design Provisions for Tangential Shear in Containment Walls."

The allowable upper limit of the tangential shear strength provided by orthogonal reinforcement should be limited to the following value:

$69 \sqrt{f'c}$ (kPa) [$10 \sqrt{f'c}$ (psi)] in accordance with ACI-318-05

For existing prestressed concrete containments, the principal tensile stress should not exceed the following value:

$27.6 \sqrt{f'c}$ (kPa) [$4 \sqrt{f'c}$ (psi)] in accordance with ACI-318-05

For new prestressed concrete containments, the principal tensile stress should be in accordance with the Code.

6. CC-3542: Loss of Prestress

Regulatory Guide, 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," should be used for guidance in determining loss of prestress in tendons.

7. CC-4240: Curing

In addition to the specifications for curing concrete in Subarticle CC- 4240(d), the following guidance should be used:

When the mean daily outdoor temperature is 4.4 °C (40 °F) or higher, the minimum period of curing should be 7 days after placing concrete.

8. CC-4352: Splices

In addition to the specifications in Subsection CC-4352, the following guidance should be used:

Mechanical splices located in areas of high stresses (maximum computed tensile stress $\geq 0.5 F_y$) should have alternate bars spliced or adjacent splices staggered. If tests for slip (or internal plastic deformation) of the splice demonstrate that the slip is low (i.e., not to exceed 50% of the elongation of the unspliced bar along the spliced length), at $0.9 F_y$, the adjacent splices need not be staggered.

9. CC-4470: Permanent Corrosion Protection

See the discussion in Regulatory Position 6, "CC-3542: Loss of Prestress."

10. CC-5210: General

The provisions of Article CC-5210 should be supplemented by an inspection to ensure that only those embedments shown on the design drawings (except minor embedments such as rebar supports and form ties), or covered by documented field changes and later placed on the as-built drawings, remain in the form after the concrete is placed. Additionally, the inspection should ensure that hollow tubes and pipe sections used as support systems or for other construction convenience, if left embedded in the concrete, are filled with concrete or grout as appropriate.

11. CC-6430: Retest

There are two options permitted by the Code in the phrase in the third sentence of Article CC-6430, "…remedial measures shall be undertaken or a retest shall be conducted…"; one should be selected if the conditions of CC-6410 (c) and (d) are not met.

Ultimate Capacity of Concrete Containment

A non-linear finite element analysis should be performed to determine the ultimate capacity of the containment. Additional guidance provided in Appendix A to NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories: An Overview," may be used to determine the ultimate capacity of the concrete containment.

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 Revision 2 of Regulatory Guide 1.136 in June 1981 to describe materials, construction, and testing of concrete containments that the NRC staff considered acceptable for use in nuclear power plants to prevent undue risk to the health and safety of the public. There are four reasons to revise the existing regulatory guide: (1) to make it consistent with other similar regulatory guides, (2) to include interim or current staff positions regarding new or advanced reactors, (3) to reduce unnecessary conservatism, and (4) to update the guidance in accordance with the ASME Boiler & Pressure Vessel Code, removing some of the exceptions that the staff had previously taken from the then-current version of the Code, because the newer Code has included those exceptions. Therefore, a revision to this regulatory guidance is necessary to include updated information.

2. Objective

The objective of this regulatory action is to update the NRC's guidance with respect to the design limits, loading combinations, materials, construction, and testing of concrete containments. This will give applicants and licensees the opportunity to take advantage of the code requirements and the current staff positions, which should lead to increased regulatory effectiveness by avoiding unnecessary conservatism that offers little safety benefit.

3. Alternative Approaches

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

(1) Do not revise Regulatory Guide 1.136.
(2) Update Regulatory Guide 1.136.

3.1 Alternative 1: Do Not Revise Regulatory Guide 1.136

Under this alternative, the NRC would not revise this guidance, and licensees would continue to use Revision 2 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 1.136

Under this alternative, the NRC would update Regulatory Guide 1.136 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 1.136 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); 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 should revise Regulatory Guide 1.136. The staff concludes that the proposed action will reduce unnecessary conservatism in the design limits, loading combinations, materials, construction, and testing of concrete 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

ACI-308.1, "Standard Specification for Curing Concrete," American Concrete Institute, Farmington Hills, Michigan, 1998.[2]

ACI-318, "Building Code Requirements for Reinforced Concrete," American Concrete Institute, Farmington Hills, Michigan, 2005.

ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute, Farmington Hills, Michigan, 2001.

ASME Boiler & Pressure Vessel Code, Section III, Division 2, "Code for Concrete Containments," 2001 Edition with 2003 Addenda, also known as ACI Standard 359-01, American Society of Mechanical Engineers, New York, New York.[3]

ASME Boiler & Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2001 Edition with 2003 Addenda, American Society of Mechanical Engineers, New York, New York.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 9.5.1, "Fire Protection Program," U.S. Nuclear Regulatory Commission, Washington, DC, October 2003.[4]

NUREG/CR-5209, "Design Provisions for Tangential Shear in Containment Walls," U.S. Nuclear Regulatory Commission, Washington, DC, August 1988.

NUREG/CR-6906, "Containment Integrity Research at Sandia National Laboratories: An Overview," U.S. Nuclear Regulatory Commission, Washington, DC, June 2006.

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[2] Copies of the Code may be obtained from the American Concrete Institute, 38800 Country Club Drive, Farmington Hills, MI 48331.

[3] 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, or the American Concrete Institute, 38800 Country Club Drive, Farmington Hills, MI 48331.

[4] All NUREG-series reports listed herein were published by the U.S. Nuclear Regulatory Commission, and are available electronically through the Electronic Reading Room on the NRC's public Web site, at http://www.nrc.gov/ reading-rm/doc-collections/nuregs/. 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. In addition, copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328, telephone (202) 512-1800; or from the National Technical Information Service (NTIS), 5285 Port Royal Road, Springfield, VA 22161, http://www.ntis.gov, telephone (703) 487-4650.

Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," Revision 3, U.S. Nuclear Regulatory Commission, Washington, DC, July 1990, available in ADAMS under Accession #ML003740007.[5]

Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," U.S. Nuclear Regulatory Commission, Washington, DC, July 1990, available in ADAMS under Accession #ML003740040.

Regulatory Guide 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, August 1977, available in ADAMS under Accession #ML003740281.

Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, April 1976, available in ADAMS under Accession #ML003740305.

Regulatory Guide 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures," Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, February 1977, available in ADAMS under Accession #ML003740374.

U.S. Code of Federal Regulations, Title 10, *Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities."[6]

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

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

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5 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.

6 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.

U.S. Code of Federal Regulations, Title 10, *Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants."

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

U.S. Nuclear Regulatory Commission, "NRC Enforcement Policy," Policy Statement Revision, *Federal Register*, Vol. 69, No. 115, June 16, 2004, pp. 33684–33685.[7]

U.S. Nuclear Regulatory Commission, "NRC Enforcement Policy; Extension of Enforcement Discretion of Interim Policy," Policy Statement Revision, *Federal Register*, Vol. 70, No. 10, January 14, 2005, pp. 2662–2664.

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[7] All *Federal Register* notices listed herein were issued by the U.S. Nuclear Regulatory Commission, and are available electronically through the Federal Register Main Page of the public GPOAccess Web site, which the U.S. Government Printing Office maintains at http://www.gpoaccess.gov/fr/index.html. 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.