U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.136, REVISION 4

Issue Date: February 2021
Technical Lead: Dr. George Thomas

DESIGN LIMITS, LOADING COMBINATIONS, MATERIALS, CONSTRUCTION, AND TESTING OF CONCRETE CONTAINMENTS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) to meet regulatory requirements for materials, design, construction, fabrication, examination, and testing of concrete (reinforced or prestressed) containments in nuclear power plants.

Applicability

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

Applicable Regulations

- 10 CFR 50.34(f), "Additional TMI-related requirements," requires, for certain reactors specified therein, that plant designs must accommodate loadings associated with hydrogen generation that result from metal-water reaction of the fuel cladding accompanied by hydrogen burning or the added pressure of inerting system actuation.

- 10 CFR 50.44, "Combustible gas control for nuclear power reactors," provides the requirements for combustible gas control for currently licensed reactors (with an operating license on October 16, 2003), for future water-cooled reactor applicants and licensees (comparable to light-water reactor (LWR) designs licensed as of October 16, 2003), and for future nonwater-cooled

---

Written suggestions regarding this guide or development of new guides may be submitted 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, at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

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.) ML20301A167. The regulatory analysis may be found in ADAMS under Accession No. ML20105A216. The associated draft guide DG-1372 may be found in ADAMS under Accession No. ML20105A215, and the staff responses to the public comments on DG-1372 may be found under ADAMS Accession No. ML20301A168.

or other water-cooled (not comparable to those licensed as of October 16, 2003) reactor applicants and licensees.

- 10 CFR 50.55a(g)(4) requires that inservice inspection (ISI) of Class CC concrete containments and metallic shell and penetration liners of concrete containments shall be performed in accordance with the applicable edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," Subsections IWE and IWL (Ref. 3), as incorporated by reference and subject to conditions stated in this regulation.

- Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, provides minimum requirements for the principal design criteria that establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The general design criteria (GDC) applicable to this RG include the following:

    - GDC 1, "Quality Standards and Records," requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated for applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

    - GDC 2, "Design Bases for Protection against Natural Phenomena," requires, in part, that SSCs important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches, with the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena without loss of capability to perform their safety function.

    - GDC 4, "Environmental and Dynamic Effects Design Bases," requires, in part, that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions and dynamic effects (of missiles, pipe whipping, and discharging fluids) associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs).

    - GDC 16, "Containment Design," requires, in part, that a reactor containment shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for as long as postulated accident conditions require.

    - GDC 50, "Containment Design Basis," requires, in part, that the reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

- Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, provides quality assurance (QA) requirements that apply to all activities

(e.g., design, procurement, fabrication, erection, inspection, testing, modification) affecting the safety-related functions of SSCs.

- Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to 10 CFR Part 50, describes required leakage testing of the primary reactor containment for water-cooled power reactors. Leaktightness of the containment structure must be tested at regular intervals during the life of the plant.

- Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50 provides, in part, criteria for the implementation of GDC 2 with respect to earthquakes. This regulation defines requirements for the operating-basis earthquake (OBE) and the safe-shutdown earthquake (SSE).

- 10 CFR Part 52 governs the issuance of, among others, standard design certifications, combined licenses, and standard design approvals for nuclear power facilities.

  - 10 CFR 52.47, “Contents of applications; technical information,” provides requirements for the technical information for standard design certifications submitted under 10 CFR Part 52.

  - 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” and 10 CFR 52.80, “Contents of Applications; Additional Technical Information,” provide requirements for the technical information needed in combined license applications.

  - 10 CFR 52.137, “Contents of applications; technical information,” provides requirements for the technical information needed for standard design approval applications.

Related Guidance

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Chapter 3, “Design of Structures, Components, Equipment, and Systems,” Section 3.8.1, “Concrete Containment” (Ref. 4), provides guidance that the NRC staff uses for performing structural safety reviews relating to concrete containments or to concrete portions of steel/concrete containments as part of a license application.

- RG 1.7, “Control of Combustible Gas Concentrations in Containment” (Ref. 5), describes methods that are acceptable to the NRC staff for implementing the requirements of 10 CFR 50.44 for reactors subject to the provisions of 10 CFR 50.44(b) or (c).

- RG 1.28, “Quality Assurance Program Criteria (Design and Construction)” (Ref. 6), describes acceptable methods for establishing and implementing a QA program for the design and construction of nuclear power plants and fuel reprocessing plants to comply with the requirements of Appendix B to 10 CFR Part 50.

- RG 1.29, “Seismic Design Classification for Nuclear Power Plants” (Ref. 7), describes a method that the NRC staff considers acceptable for use in identifying and classifying those features of LWR nuclear power plants that must be designed to withstand the effects of the SSE.

- RG 1.35.1, “Determining Prestressing Forces for Inspection of Prestressed Concrete Containments” (Ref. 8), provides guidance on determining prestressing forces in containment tendons to be used for ISI of prestressed concrete containment structures.

- RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants” (Ref. 9), provides guidance on damping values that the NRC staff considers acceptable for use in the seismic response analysis of seismic Category I nuclear power plant structures.

- RG 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants” (Ref. 10), describes a method acceptable to the NRC staff for determining the minimum thickness, based on radiation shielding requirements (only), of concrete radiation shields in nuclear power plants. Note that the approach for determining minimum thickness of concrete radiation shields, based on structural requirements, in RG 1.69 is not applicable to concrete containments. For concrete containments, the minimum concrete thickness based on structural requirements is determined using this RG (1.136).

- RG 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants” (Ref. 11), provides guidance that the NRC staff considers acceptable for use in selecting the design-basis tornado and design-basis tornado-generated missiles that a nuclear power plant should be designed to withstand to prevent undue risk to the health and safety of the public.

- RG 1.90, “Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons” (Ref. 12), describes methods that the NRC staff considers acceptable for use in developing an appropriate ISI program for prestressed concrete containment structures with grouted tendons.

- RG 1.92, “Combining Modal Response and Spatial Components in Seismic Response Analysis” (Ref. 13), provides guidance on methods that the NRC staff considers acceptable for combining modal responses and spatial components in seismic response analysis of nuclear power plant SSCs that are important to safety.

- RG 1.107, “Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures” (Ref. 14), describes a method and quality standards that the NRC staff considers acceptable for the use of Portland cement grout for protecting prestressing tendons from corrosion in prestressed concrete containment structures with grouted tendons.

- RG 1.115, “Protection Against Turbine Missiles” (Ref. 15), describes methods acceptable to the NRC staff for protecting important-to-safety SSCs against missiles resulting from turbine failure to satisfy GDC 4.

- RG 1.117, “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants” (Ref. 16), describes an acceptable approach for identifying those SSCs of LWRs that should be protected from the effects of the worst case extreme winds (tornados and hurricanes) and wind-generated missiles, so that they remain functional.

- RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)” (Ref. 17), describes methods and procedures that the NRC considers acceptable for compliance with NRC regulations in the analysis, design, construction, testing, and

evaluation of safety-related nuclear concrete structures, excluding concrete reactor vessels and concrete containments.

- RG 1.199, "Anchoring Components and Structural Supports in Concrete" (Ref. 18), describes a method acceptable to the NRC staff for the design, installation, testing, evaluation, and QA of anchors (steel embedments) used for component and structural supports in concrete.

- RG 1.208, "A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion" (Ref. 19), provides guidance on the development of the site-specific ground motion response spectrum, which is part of the development of the SSE for a site as a characterization of the regional and local seismic hazard.

- RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings above Design-Basis Pressure" (Ref. 20), describes methods that the NRC staff considers acceptable for satisfying applicable regulations for the structural integrity of containments under internal pressurization that pertain to the containment structural capacity above design-basis pressures, to combustible gas control, and to the prevention and mitigation of severe accidents.

- RG 1.217, "Guidance for Assessment of Beyond-Design-Basis Aircraft Impacts" (Ref. 21), describes considerations for aircraft impacts on new nuclear power reactors.

- RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants" (Ref. 22), provides guidance that the NRC staff considers acceptable for use in selecting the design-basis hurricane windspeed and hurricane-generated missiles that a new nuclear power plant should be designed to withstand to prevent undue risk to public health and safety.

- RG 1.231, "Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants" (Ref. 23), describes methods that the NRC staff considers acceptable in meeting regulatory requirements for acceptance and dedication of commercial-grade design and analysis computer programs used in safety-related applications for nuclear power plants.

- American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 37-14, "Design Loads on Structures during Construction" (Ref. 24), describes the minimum design requirements for construction loads and load combinations affecting buildings and other structures that are under construction. The loads are applicable to all conventional construction methods.

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 and 52 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 and 3150-0151. Send comments regarding this information collection to the Information Services 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 (3150-0011 and 3150-0151), 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 collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

This revision (Revision 4) updates the guidance to endorse, with exceptions, the 2019 edition of the ASME B&PV Code, Section III, Division 2 (American Concrete Institute (ACI) Standard 359-19), "Code for Concrete Containments" (Ref. 25), hereafter referred to as "the Code" or "the ASME Code." The significant changes include updated concrete material and mix design provisions for resistance to alkali-silica reaction and improved durability; incorporation of more recent material standards for reinforcing steel; new design provisions requiring radial tension reinforcement in portions of prestressed concrete containment susceptible to delamination; provisions allowing use of mechanically headed deformed bar anchorages, additional types of mechanical splices for reinforcement; and the addition of slip test and slip acceptance criteria for mechanical splices which improve constructability and safety.

This revision of the guide also addresses the acceptability of the Section III Code Cases related to Division 2 of the ASME B&PV Code, Section III. The Division 2 Code Cases addressed in this RG include those listed in RG 1.84, Revision 38, "Design, Fabrication and Materials Code Case Acceptability, ASME Section III," issued October 2019 (Ref. 26); RG 1.193, Revision 6, "ASME Code Cases Not Approved for Use," issued October 2019 (Ref. 27); as well as any later Division 2 Code Cases published up to the release of the 2019 Edition of the ASME B&PV Code, "Code Cases—Nuclear Components," dated July 1, 2019 (Ref. 28). Of the above RGs, RG 1.84 is incorporated by reference in 10 CFR 50.55a. The staff plans to remove the Division 2 Code Cases from these RGs in the next revision to them and the Division 2 Code Cases will be located only in this guide (RG 1.136).

Background

ASME and ACI have jointly published the ASME Section III, Division 2 (ACI 359), "Code for Concrete Containments." This code establishes rules for materials, design, fabrication, construction, examination, testing, and preparation of reports for prestressed and reinforced concrete containments for nuclear power reactors. The containments covered by the Code include (1) structural concrete pressure-resisting shells and shell components, (2) shell metallic liners, and (3) penetration liners extending the containment liner through the surrounding shell concrete. Since the last revision of RG 1.136 in 2007, the Code has been updated several times, most recently in 2019. The 2019 Edition of the Code includes significant technical changes and additions to the Code provisions, along with editorial improvements and clarifications that have occurred over several editions and addenda since 2004.

Significant changes or additions to the Code provisions (indicated within parentheses in bullets below) since the publication of RG 1.136, Revision 3, and incorporated in the 2019 Edition include the following:

- Concrete material and mix design requirements were specified for improving the ability of concrete to resist alkali-silica reaction and sulfate attack. (CC-2231.3, CC-2231.7.4)

- Concrete exposure categories and classes for durability were revised to be consistent with ACI 318-14, "Building Code Requirements for Structural Concrete" (Ref. 29). (CC-2231.7, Table CC-2231.7.1-1, and Table CC-2231.7.2-1)

- Provisions were added requiring radial tension reinforcement in portions of prestressed concrete containments with single curvature and double curvature. (CC-3545)

- The 2016 edition of material specifications ASTM A615, "Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete" (Ref. 30), and ASTM A706, "Standard Specification for Deformed and Plain Low-Alloy Steel Bars for Concrete Reinforcement" (Ref. 31) were incorporated for reinforcing steel allowing use of Grade 75 and Grade 80 high-strength reinforcement bars. Additional conditions, included in ASTM A706-16, are also specified for ASTM A615 reinforcing steel material to ensure ductility: (1) actual yield strength shall not exceed specified minimum yield strength by more than 18,000 pounds per square inch (psi) (125 megapascals (MPa)), (2) the ratio of actual tensile strength to actual yield strength shall not be less than 1.25, and (3) for Grade 60, Grade 75, and Grade 80 bars, new minimum elongations consistent with those for ASTM A706 material are specified for different bar sizes. (Table NCA-7100-3, CC-2300, CC-2331.2)

- Footnote (2) was added to Table CC-3230-1, "Load Combinations and Load Factors," regarding treatment of load combinations involving the OBE when the OBE is set equal to one-third or less of the SSE, consistent with 10 CFR Part 50, Appendix S.

- An upper limit on design yield strength of reinforcement was increased from 60,000 psi (420 MPa) to 80,000 psi (550 MPa). (CC-3422.1, CC-3521.1.3)

- An additional multiplication factor of 1.2 specified for development length of Grade 75 to Grade 80 reinforcement. (CC-3532.1.7)

- Material, design, and qualification provisions were added for use of mechanical anchorage devices for reinforcement end anchorage (mechanically headed deformed bars in tension) for reinforcement with specified yield strength not exceeding 60,000 psi (420 MPa) and bar size not larger than #11. The addition provides an equation for calculating development length for such bars in tension and defines the critical section for development length. (CC-2311, CC-3532.1.2, CC-3532.4, CC-4333)

- Specific conditions are stated under which intentional twisting of prestressing tendons composed of multiple elements stressed simultaneously as a group may be waived. (CC-4432.5)

- Slip test and slip acceptance criteria are added as part of the performance tests required to qualify mechanical splices and mechanically headed deformed bar systems for use. (CC-4433.2.3(c))

Consistent with the NRC's commitment to the use of industry consensus codes and standards for the design, construction, and licensing of commercial nuclear power reactor facilities, this guide describes methods that the NRC staff considers acceptable for the materials, design (including loads and load combinations), fabrication, construction, examination, and testing of reinforced and prestressed concrete containments in the context of the 2019 Edition of the Code.

The NRC staff has evaluated the provisions contained in Articles CC-1000 through CC-6000 of the Code. This RG endorses Articles CC-1000 through CC-6000 of the Code with the exceptions and clarifications noted below. This RG also provides additional guidance on loads and load combinations including combustible gas loading conditions, design and analysis, and a method for determining the ultimate capacity of a concrete containment.

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 guidance positions presented in

Section C of this guide. The reasons and basis for the supplementary guidance in the regulatory guidance positions that are not self-explanatory are as follows.

Discussion of Staff Regulatory Guidance Positions

Regulatory Guidance Position 1

CC-2232.4[1]: Basis for Selection of Self-Consolidating Concrete (SCC)

In Regulatory Guidance Position 1 the staff does not endorse the general use of SCC for containment construction. It further clarifies the limited use of alternative concrete mixture provisions (reproportioning or SCC) provided in the text of the Code provision to localized sites and conditions where consolidation of concrete is unusually difficult. The regulatory guidance position also refers to ASTM standards C1610/C1610M, C1611/C1611M, C1712/C1712M, C1758/C1758M (Refs. 33, 34, 35, 35) and industry best practices in ACI 237R (Ref. 36) if SCC is used for this purpose.

Regulatory Guidance Position 2

CC-2243: Grout for Grouted Tendon System

Regulatory Guidance Position 2 provides guidance to supplement paragraph CC-2243 with respect to grouting of prestressing tendons. This is intended to provide assurance of the integrity of grouted tendons that cannot be directly inspected during the life of the containment.

Regulatory Guidance Position 3

CC-2311: Mechanical Anchorage Devices

Regulatory Guidance Position 3 addresses dimensions of obstructions for headed bars such that the obstruction is considered not to detract from the net bearing area of the head calculated using the code provision in subparagraph CC-2311(b). This regulatory guidance position is consistent with the related requirements in Annex A1 to ASTM A970/A970M-18, "Standard Specification for Headed Steel Bars for Concrete Reinforcement" (Ref. 37).

Regulatory Guidance Position 4

CC-2436.3: Acceptance Standards

Experience with the use of alloy steel materials for anchor blocks and wedge blocks (such as Grade 4140 material of ASTM A29/A29M, "Standard Specification for General Requirements for Steel Bars, Carbon and Alloy, Hot-Wrought" (Ref. 38)) indicates that a high degree of hardness of these materials is a factor in causing cracking (presumably stress-corrosion cracking) under certain 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 Guidance Position 4 provides guidance in addition to the Code

[1] This alphanumeric citing identifies the subarticle, subsubarticle, and paragraph or subparagraph if applicable, of the "Code for Concrete Reactor Vessels and Containments" being discussed.

specifications for the acceptance standards for mechanical tests of prestressing anchorage components.

Regulatory Guidance Position 5

CC-2437: Hardness of Wedge Plates, Wedges and Anchor Nuts

The testing of prestressing materials is needed to qualify them against loss of ductility in cold temperatures; therefore, the guidance in Regulatory Guidance Position 5 is recommended.

Regulatory Guidance Position 6

CC-2438.4: Permanent Coating Material

Permanent corrosion prevention coating applied to the tendons is important to protect against corrosion degradation and maintain the structural integrity of tendons as a load-bearing component. Regulatory Guidance Position 6 addresses the certification of test data verifying properties, qualification test data, and test reporting specified in subparagraph CC-2438.4 that is provided by the coating manufacturer for the permanent coating material.

Regulatory Guidance Position 7

CC-2463: Type and Number of Performance Tests

Various prestressing systems may require different numbers of tests for tendon systems to establish their adequacy for qualification and use. Variations within the tolerance limits of the construction specification for material properties and geometry of anchorages and tendons need to be realistically and adequately represented in the system testing. This is addressed in Regulatory Guidance Position 7.

Regulatory Guidance Position 8

***CC-3000:* DESIGN**

To facilitate design and analysis procedures, some specific guidance in addition to the Code specification are included in Regulatory Guidance Positions 8.a through 8.e to be consistent with the current staff position. That is, Regulatory Guidance Position 8 includes additional load considerations and loading combination guidelines for the design and analysis of concrete containments. Regulatory Guidance Position 8.a(1) addresses hydrodynamic loads resulting from LOCA or safety/relief valve (SRV) actuation for boiling-water reactor containments.

To address the requirements of 10 CFR 50.34(f)(3)(v), 10 CFR 50.44(b) or (c), as applicable, concerning the demonstration of containment structural integrity for accidents involving hydrogen release and combustible gas release, Regulatory Guidance Position 8.b provides loads and load combinations for pressure loads that result from a fuel cladding metal-water reaction, accompanied by an uncontrolled hydrogen burn or accompanied by added pressure from post-accident inerting assuming carbon dioxide is the inerting agent (as applicable). The above requirements and guidance apply only to currently licensed reactors and future water-cooled reactor designs with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to that of LWR designs licensed as of October 16, 2003. For non-water-cooled reactors and water-cooled reactors that have different

characteristics for the production of combustible gases from current LWRs (licensed as of October 16, 2003), the general performance-based requirement in 10 CFR 50.44(d) needs to be addressed on a design-specific basis considering information on whether accidents involving combustible gases are technically relevant for their design depending on construction materials used.

Regulatory Guidance Position 8.c indicates the staff position that use of Grade 75 and Grade 80 high-strength reinforcement allowed by the Code is not endorsed for general use for concrete containments. Research and development that integrates implications, including ductility, development length, and crack control, for the generic use of high-strength reinforcement is limited and still progressing. Furthermore, there is a lack of operating experience with high-strength reinforcement in the U.S. nuclear power industry and a lack of a precedent setting review in a licensing action for establishing a firm regulatory guidance position for its generic use. However, the staff will review a proposed specific use of high-strength reinforcement and its supporting justification in a licensing action on a case-by-case basis.

Regulatory Guidance Position 9

CC-3542: Loss of Prestress

Regulatory Guidance Position 9 provides guidance to supplement considerations in paragraph CC-3542 with respect to loss of prestress in tendons. This ensures that additional considerations related to prestress losses and variations in the factors that affect time-dependent characteristics of tendon forces are adequately factored in when determining prestress losses.

Regulatory Guidance Position 10

CC-4240: Curing

The Code provision does not provide explicit guidance for curing concrete at temperatures higher than 4.4 degrees Celsius (C) (40 degrees Fahrenheit (F)). Consequently, Regulatory Guidance Position 10 gives guidance for curing concrete by referring to standard specifications [ACI 305.1, ACI 306.1, ACI 306R, ACI 308.1 (Refs. 41, 42, 41, 42)] and industry best practices [ACI 305R, ACI 308R (Refs. 43, 44)] for hot weather and cold weather concreting.

Regulatory Guidance Position 11

CC-4333: Mechanical Splices and Mechanically Headed Deformed Bar Systems and CC-4334: Arc Welded Joints

Regulatory Guidance Positions 8.d and 11 recommend that mechanical splices and mechanically headed deformed bar anchorage, as well as welded splices, should be capable of developing at least the specified minimum tensile strength of the spliced or anchored reinforcing bar to assure ductile behavior. This position is consistent with the strength requirements for a Type 2 splice and a headed deformed bar required by Section 18.2.7 and associated commentary of ACI 318-19, "Building Code Requirements for Structural Concrete, Commentary on Building Code Requirements for Structural Concrete (ACI 318R-19) " (Ref. 45) for reinforcement subjected to expected stress levels in yielding regions, especially under earthquake loading. This is also consistent with the requirements for mechanical splices in Sections 12.14.3.2 and 21.1.6 of ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary" (Ref. 46). As stated in the commentary for ACI 318-19 and ACI 349-13, the requirements for Type 2 mechanical splices are intended to avoid a splice failure when the reinforcement is subjected to expected stress levels in yielding regions in beyond-design-basis earthquake shaking. Type 1

mechanical splices (capable of developing 125 percent of the specified minimum yield strength of the bar, as specified in the ASME Code) on any grade of reinforcement and Type 2 mechanical splices on Grade 80 and Grade 100 reinforcement may not be capable of resisting the stress levels expected in yielding regions, and therefore restrictions were placed on their use. ACI 349-13 structures are detailed for ductile response in the event of an earthquake larger than the SSE. Consistent with this philosophy, Type 1 splices are inadequate and are not permitted in nuclear safety-related concrete structures.

Concrete containments in nuclear power plants are important safety-related structures; therefore, their criteria for mechanical or welded splices and mechanical end anchorage should not be less stringent than that of other seismic Category I structures, as defined in ACI 349-13. The containment structure could experience stresses in the yielding regions under design-basis accident and earthquake conditions. Also, the required analysis to calculate the ultimate capacity of reinforced concrete containment structures against the internal pressure as a measure of the safety margin above the design-basis accident pressure can be based on attaining a global hoop strain of 1 percent in the inside reinforcement (Ref. 20), which is significantly higher than the specified minimum yield strength. Further, containment structures are expected to be evaluated for beyond-design-basis accidents, earthquakes (at least 1.67 times the SSE), and other extreme events (e.g., aircraft impact). Consequently, Type 2 mechanical or welded splices and mechanical headed bar systems should be used in concrete containment structures. Therefore, the criterion in the ASME Code, which is the equivalent to the criterion for Type 1 mechanical or welded splices in ACI 318-14 and ACI 318-19, is not an adequate criterion for qualifying mechanical splices for use in concrete containment structures. Thus, the regulatory guidance position provides a more stringent strength criterion for qualification of mechanical splices, welded splices and headed deformed bar systems, and the same criterion should also be used for continuing splice or headed bar performance tests for production in the field. For consistency across splicing/anchoring methods, this more stringent strength criterion is also recommended for lap splices and end anchorage by standard hooks or bends of tension reinforcement (except nominal temperature reinforcement) if their use cannot be avoided in regions of maximum tensile stress, potential yielding regions, or in a region where tension is predicted in a direction perpendicular to the bar to be spliced or anchored.

Regulatory Guidance Position 13

CC-5210: General [Concrete Construction Examination]

The locations of all major embedments, such as plates, embedded piping penetration sleeves, major structural framings, and anchor bolts, need to be preplanned, identified on the construction drawings, and documented on field changes to the drawings. 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. Regulatory Guidance Position 13 provides guidance on this matter, in addition to the Code provisions.

Regulatory Guidance Position 14

Ultimate Capacity of Concrete Containment

GDC 50 in Appendix A to 10 CFR Part 50 requires a determination of the internal pressure capacity for containment structures, as a measure of the safety margin above the design-basis accident pressure. Regulatory Guidance Position 14 describes an acceptable methodology for determining the ultimate pressure capacity of the concrete containment. RG 1.216 describes a deterministic methodology for estimating the ultimate pressure capacity of reinforced concrete and prestressed concrete containments.

Regulatory Guidance Positions 15 and 16

Code Cases Related to ASME Code, Section III, Division 2

Regulatory Guidance Positions 15 and 16 include active Section III Code Cases related to Division 2 that are acceptable or conditionally acceptable, respectively, to the NRC staff. These positions include Division 2 Code Cases that were inadvertently listed in RG 1.84, Revision 38, and any later Division 2 Code Cases published up to the release of the 2019 Edition of the ASME B&PV Code "Code Cases—Nuclear Components." The 2019 Edition of the Code Cases includes Code Case actions issued through Supplement 7 to the 2017 Edition.

Regulatory Guidance Position 17

Regulatory **Guidance** Position 17 includes active Section III Code Cases related to Division 2 that are unacceptable to the NRC staff. This position includes Division 2 Code Cases that were inadvertently included in RG 1.193, Revision 6, and any later Division 2 Code Cases issued up to the publication of the 2019 Edition of the ASME B&PV Code, "Code Cases—Nuclear Components."

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 Standards and Safety Guides for protecting people and the environment from 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.[1] Although the NRC does not endorse the following IAEA safety standard(s) and/or guide(s), in general this RG incorporates similar guidelines and is generally consistent with the basic safety principles provided in them.

- IAEA Safety Standards Series No. SSG-53, "Design of Reactor Containment and Associated Systems of Nuclear Power Plants," issued 2019 (Ref. 47), provides guidance relevant to reactor containment and associated systems at a high level. It addresses the containment structure and the systems with the functions of isolation, control and management of mass and energy releases, control and limitation of radioactive releases, and control and management of combustible gases.

- IAEA Safety Standards Series No. NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants," issued 2003 (Ref. 48), provides recommendations at a high level on a generally accepted way to design a nuclear power plant so that an earthquake motion at the site will not jeopardize the safety of the plant. It also gives guidance on a consistent application of methods and procedures for analysis, testing and qualification of structures and equipment so that they meet the safety requirements covering the design of nuclear power plants, safety assessments for the design and the regulatory issues concerned with the licensing of plants.

---

[1] IAEA Safety Requirements and Guides may be found at 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 Official.Mail@IAEA.Org. It should be noted that some of the international recommendations do not correspond to the requirements specified in the NRC's regulations, and the NRC's requirements take precedence over the international guidance.

- IAEA Safety Standards Series No. NS-G-2.6, "Maintenance, Surveillance and In-Service Inspections in Nuclear Power Plants," issued 2002 (Ref. 49), covers the organizational and procedural aspects of maintenance, surveillance and in-service inspection. The guidance does not give detailed technical advice in relation to particular items of plant equipment, nor does it cover inspections made for and/or by the regulatory body.

- IAEA Safety Standards Series No. NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants," issued 2004 (Ref. 50), provides generalized guidance on the geotechnical engineering aspects of a site that affect the foundation system.

- IAEA Safety Standards Series No. SSG-48, "Aging Management and Development of a Programme for Long Term Operations of Nuclear Power Plants," issued 2018 (Ref. 51), provides generalized guidance for operating organizations on implementing and improving ageing management and, obsolescence management and on developing a program for safe long term operation for nuclear power plants.

- IAEA Nuclear Energy Series No. NP-T-2.5, "Construction Technologies for Nuclear Power Plants," issued 2011 (Ref. 52), provides generalized guidance on the tools and steps that support plans and techniques for constructing nuclear power plants

- IAEA Specific Safety Guide SSG-38, "Construction for Nuclear Installations," issued 2015 (Ref. 53), provides generalized guidance applicable to the construction stage of a new nuclear installation and to major modifications and refurbishments of an existing nuclear installation.

Documents Discussed in Staff Regulatory Guidance

This RG endorses, in part, the use of one or more codes or standards developed by external organizations, and other third-party guidance documents. These codes, standards, and third-party guidance documents may contain references to other codes, standards, or third-party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in an RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in an RG, then the secondary reference is neither a legally binding requirement nor a "generic" NRC-approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

C. STAFF REGULATORY GUIDANCE

For the materials, design, fabrication, construction, examination, testing, and ISI of concrete containments, this RG endorses, with exceptions, the following industry codes and standards:

- ASME, Section III, Division 2, 2019 Edition, "Code for Concrete Containments"
- ASME, Section III, Subsection NCA, 2019 Edition, "General Requirements for Division 1 and Division 2"

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 guidance positions. Unless otherwise stated, the regulatory guidance positions indicated are in addition to the Code provisions.

1. CC-2232.4 Basis for Selection of Self-Consolidating Concrete (SCC)

The staff does not endorse general use of SCC for containment construction. The use of alternative mixture provisions (reproportioning or SCC) provided in the text of the Code provision should be limited to only localized areas of specific difficulty where conditions (e.g., reinforcement congestion) make consolidation of concrete unusually difficult. If SCC is considered for such limited application, the SCC mix should be subject to tests using standard test methods in ASTM C1611/C1611M for slump flow, ASTM C1610/C1610M (column test) and ASTM C1712/C1712M (rapid penetration test) for static segregation, ASTM C1621/1621M for passability, and ASTM C1758/C1758M as standard practice for sample fabrication. The use of SCC should follow applicable guidance for known best practices and processes in ACI 237R-07, "Self-Consolidating Concrete" (reapproved 2019).

2. CC-2243 Grout for Grouted Tendon Systems

In addition to CC-2243 provisions, RG 1.107 should be used for guidance on qualifying grout for grouted tendon systems. In case of conflict between CC-2243 provisions and RG 1.107 provisions, the more stringent provision governs. The licensing application to the NRC should identify and explain such conflicts.

3. CC-2311 Mechanical Anchorage Devices

For the net bearing area to be calculated by taking the gross cross-sectional areas of the head minus the nominal cross-sectional area of the bar, an obstruction should not detract from the net bearing area of the head. For this purpose, the dimensional requirements in Figure A1.3 of ASTM A970/A970M-18 should be met for an obstruction to be considered not to detract from the net bearing area of the head. Accordingly, obstructions should be considered to not detract from the net bearing area of the head if (i) they do not extend from the bearing face of the head more than 0.6 nominal bar diameters (for No. 8 and larger bars), or smaller of 15 millimeters (0.6 inch) and the nominal bar diameter (for bars smaller than No. 8); and (ii) they do not have a diameter greater than 1.5 nominal bar diameters. Otherwise, consistent with Annex A1 to ASTM A970/A970M-18, the net bearing area of a bar with an obstruction meeting the requirements of Figure CC-2310-1 is the gross area of the head minus the maximum area of the obstruction and should not be less than four times the nominal cross-sectional area of the bar.

4. CC-2436.3 Acceptance Standards [Mechanical Tests]

In addition to following CC-2436.3, "Acceptance Standards," 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.

5. CC-2437 Hardness of Wedge Plates, Wedges and Anchor Nuts

In addition to CC-2437, "Hardness of Wedge Plates, Wedges and Anchor Nuts," the staff provides the guidance for protection of 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 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 button heads, wedge plates, wedges, and anchor nuts), the specific components will exhibit the required strength and ductility under the lowest anticipated temperatures.

6. CC-2438.4 Permanent Coating Material

The coating manufacturer should provide certified test data verifying the properties listed in CC-2438.4.1, "Properties of Permanent Coatings," for corrosion prevention of prestressing systems. Additionally, the coating manufacturer should provide certified qualification test data for the tests specified in CC-2438.4.2, "Analysis of Permanent Coatings." The coating manufacturer should also supply with each shipment a certified test report with an analysis of the properties specified in CC-2438.4.2.

7. CC-2463 Type and Number of Performance Tests [Tendon System]

In addition to following CC-2463, any system of prestressing should be subjected to a sufficient number of tests to establish adequacy for its qualification and use. Variations within the tolerance limits of the construction specification for material properties and geometry of anchorages and tendons should be realistically and adequately represented in the system testing. The applicant or licensee should include a description of the test program for qualification of the tendon system, including a justification of the adequacy of number of tests performed.

8. 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 additional regulatory guidance positions:

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 to Table CC-3230-1.

   (1) Hydrodynamic loads resulting from LOCA and/or SRV actuation for boiling-water reactor concrete containments should be combined according to the approach contained in Appendix A of this RG, "Staff Position on Boiling-Water Reactor Containment Pool Dynamics." Fluid-structure interaction associated with these hydrodynamic loads and those from earthquakes should be included.

(2) Where post-LOCA flooding is a design consideration, the design evaluation should be performed using the load combinations in the Code containing internal flooding along with OBE (except where OBE is set at one-third or less of the SSE) and internal flooding along with design wind.

(3) Loads to be protected against during extreme environmental conditions include those induced by the design-basis tornado, hurricane, and the SSE specified for the plant site. Therefore, the W_t load in CC-3222.2, in CC-3240, and in the load combinations in Table CC-3230-1 should include worst case extreme winds (tornados and hurricanes) and corresponding wind-generated missiles.

(4) Other site-related or plant-related loads applicable to containment such as floods, explosive hazards in proximity to the site, potential aircraft crashes (nonterrorism-related incidents), and missiles generated from activities of nearby military installations or turbine failures should be evaluated and included, as applicable, in the appropriate factored load combinations.

(5) Loads encountered during construction of the containment should include dead loads, live loads, prestress loads, temperature, wind, earth pressure, snow, rain, and ice, and other construction loads that may be applicable such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and form pressure. ASCE/SEI Standard 37 gives additional guidance on construction loads for use in the load combination for construction given in Table CC-3230-1 of the ASME Code. Where SEI/ASCE Standard 37 and the ASME Code provide conflicting criteria, then the ASME Code should govern.

b. The integrity of the containment structure should additionally be demonstrated for the combustible gas loading conditions listed below. The acceptance criteria are limited to demonstrating that liner strains satisfy the service and factored load category requirements in CC-3720, "Liner," of the Code. These loading conditions should include the effects of temperature. For prestressed concrete containment the effects of prestress should also be included. The analysis to demonstrate containment structural integrity should account for applicable additional considerations in RG 1.216.

(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-percent fuel cladding-coolant reaction

P_{g2} = pressure resulting from uncontrolled hydrogen burning (for an accident that is accompanied by hydrogen burning)

P_{g3} = pressure resulting from post-accident inerting, assuming carbon dioxide is the inerting agent (for an accident that is accompanied by post-accident inerting)

RG 1.7 provides additional guidance on the pressure load P_{g3} due to combustible gas concentration for satisfying the requirements in 10 CFR 50.44(c).

(2) For the service load category, the strains in the containment liner should not exceed the limits given in CC-3720 for the loading condition $D + P_{g3}$ resulting from an inadvertent

full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide inerting).

(3) As a minimum design loading condition for either condition in 1 above, the following load combination should be satisfied for water-cooled reactor designs, with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to LWR designs licensed as of October 16, 2003, to which the requirements in 10 CFR 50.44(c) apply:

D + 310 kPa (45 psig)

(4) For nonwater-cooled reactors and water-cooled reactors that have characteristics related to the production of combustible gases different from those of current LWRs (licensed as of October 16, 2003), information should be included to address whether accidents involving combustible gases are technically relevant for their design. If found to be technically relevant, the general performance-based requirement in 10 CFR 50.44(d)(2) should be satisfied (note that 10 CFR 50.44(c) requirements do not apply to such reactors). Future applicants of such reactors should submit design-specific information to the NRC indicating how the safety impacts of combustible gases generated during design-basis and significant beyond-design-basis accidents are addressed in the design to ensure adequate protection of public health and safety and common defense and security. This information should be based in part on a design-specific probabilistic risk assessment.

c. CC-3422 Reinforcing Steel (and related Code requirements that allow use of Grade 75 and Grade 80 High-Strength Reinforcement)

In subparagraph CC-3422.1, "Design Strength," design yield strength of reinforcement should not exceed 60,000 psi (420 MPa). Use of Grade 75 and Grade 80 high-strength reinforcement (specified minimum yield strength greater than 60 ksi (420 MPa)) allowed by the Code, in Subarticle CC-2300 and paragraphs CC-3422, CC-3520, and CC-3530, is not generically endorsed. If an applicant or licensee chooses to use such high-strength reinforcement in its specific design application, the NRC staff will evaluate it on a case-by-case basis during the licensing action review process. The applicant should include in the application results of tests, analytical studies, and structural performance evaluations to sufficiently support the adequacy of its use of high-strength reinforcement and inputs to design in the context of Division 2 code provisions. Combination of different grades of reinforcement (e.g., normal- (or low-) strength and high-strength rebar) in design for a given limit state in any structural section is not allowed, and installation of high-strength reinforcement should be confirmed independently during the construction phases.

d. CC-3532 Reinforcing Steel Splicing and Development

Mechanical devices for end anchorage, mechanical splices, and welded splices of reinforcing bars should be qualified by testing to be capable of developing at least the specified minimum tensile strength of the anchored or spliced reinforcing bar. If their use cannot be avoided, lap splices and end anchorage by standard hooks or bends of tension reinforcement (except nominal temperature reinforcement) that must be used in regions of maximum tensile stress, potential yielding regions, or in a region where tension is predicted in a direction perpendicular to the bar to be spliced or anchored should be capable of developing at least the specified minimum tensile strength of the reinforcing bar.

e. CC-3900 Design Criteria for Impulse Loadings and Missile Impact

(1) Whereas CC-3900 applies to the containment structure and liner directly affected by the impactive and impulsive loadings, shock and vibratory effects on the functionality of attached safety-related components at or at points away from the location of impact should also be evaluated.

(2) For cases where impact could occur at any location on the containment structure, impact locations evaluated should be such that the shear demand and flexural demand are maximized.

(3) Where rigorous nonlinear dynamic time history analysis is used in the design for impulse or impact loadings, the input parameters and stress and strain limits used as acceptance criteria for section strength should be specified and the rationale adequately justified.

9. CC-3542 Loss of Prestress

In addition to provisions in CC-3542, RG 1.35.1 should be used for guidance in determining loss of prestress in tendons.

10. CC-4240 Curing

In addition to the specifications for curing concrete in CC-4240, curing and protection against physical and thermal damage from time of placement until end of minimum curing period should be in accordance with ACI 308.1-11, ACI 308R-16, and ACI 305.1-14, along with ACI 305R-10 or ACI 306.1-90 (R2002) along with 306R-16 as applicable.

11. CC-4333 Mechanical Splices and Mechanically Headed Deformed Bar Systems and CC-4334 Arc Welded Joints

The acceptance criteria of paragraph CC-4333 for qualification testing and continuing performance testing of mechanical splices and mechanically headed deformed bar systems, and paragraph CC-4334 for qualification and continuing performance testing of arc welded joints, should be additionally supplemented as follows consistent with Regulatory Guidance Position 8.d:

a. In CC-4333.2.3(a), “Static Tensile Tests”:

(1) The actual tensile strength of the reinforcing bar required by subparagraph CC-4333.2.3(a) of the Code should be determined in accordance with testing standards, definitions and methods specified for tension test in ASTM A370-20 “Standard Test Methods and Definitions for Mechanical Testing of Steel Products” (Ref. 54).

(2) The tensile strength of an individual splice and mechanically headed deformed bar system should not be less than the specified minimum tensile strength (f_t) of the reinforcing bar spliced or anchored.

(3) Each individual test report on the splice and mechanically headed deformed bar system specimen and the unspliced bar specimen should include at least the following information: (a) tensile strength, (b) total elongation, and (c) load-extension curve to

the strain corresponding to the specified minimum tensile strength.

b. In CC-4333.4, "Initial Qualification Tests," CC-4333.5.4, "Tensile Testing Requirements," and CC-4334, "Arc Welded Joints" (Mandatory Appendix D2-VIII, paragraph D2-VIII-1520, D2-VIII-1530, and D2-VIII-1540), the tensile strength of each individual sample shall equal or exceed the specified minimum tensile strength (f_t) of the reinforcing bar as shown in Table CC-4333-1 column titled "Minimum Tensile Strength."

c. In CC-4333.5.5, "Substandard Tensile Test Results," and CC-4334, "Arc Welded Joints" (Mandatory Appendix D2-VIII, paragraph D2-VIII-1555), the strength criteria for failure of tests of any individual sample used for testing should be the specified minimum tensile strength of the reinforcing bar as in item (b) above.

d. Additional testing and evaluation should be performed for impactive and impulsive loading conditions, where the mechanical splices or mechanical headed bar systems are subjected to loading that causes high strain rate.

12. CC-4480 Protection of Post-Tensioning Anchorages and Prestressing Steel

ISI of Class CC reinforced concrete containments and prestressed concrete containments with ungrouted tendons should meet the requirements of Subsection IWL of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME B&PV Code, as incorporated by reference in 10 CFR 50.55a, "Codes and Standards." ISI of prestressed concrete containment with grouted tendons should be in accordance with RG 1.90.

13. CC-5210 General [Concrete Construction Examination]

The provisions of CC-5210 should be supplemented by an inspection to ensure that only those embedments shown on the construction 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.

14. Ultimate Capacity of Concrete Containment

A nonlinear finite element analysis should be performed to determine the ultimate capacity of the containment. The design and analysis procedures described in Regulatory Guidance Position C.1 in RG 1.216 are acceptable.

15. Acceptable Section III, Division 2, Code Cases

The Code Cases listed in Table 1 below are acceptable to the NRC for application in the design and construction of concrete containment structures for water-cooled nuclear power plants.

Table 1. Acceptable Section III, Division 2, Code Cases

CODE CASE NUMBER	CODE CASE TITLE	DATE
N-171	*Postweld Heat Treatment of P-No. 1 Material, Section III, Division 2*	4/30/1993
N-213-1	*Welded Radial Shear Bar Assemblies, Section III, Division 2*	2/11/2016
N-258-2	*Design of Interaction Zones for Concrete Containments, Section III, Division 2*	7/30/1986
N-312	*Use of Stud Welds To Anchor Section III, Division 1, Class MC Vessels to Section III, Division 2, Class CC Concrete Containment Structures, Section III, Division 2*	5/25/1983
N-632	*Use of ASTM A 572, Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2*	12/3/1999
N-633	*Use of SA-533/SA-533M, Type B, Class 2, Plate for Structural Attachments to Class CC Containment Liners, Section III, Division 2*	6/14/2002
N-763	*ASTM A 709-06, Grade HPS 70W (HPS 485W) Plate Material without Postweld Heat Treatment as Containment Liner Material or Structural Attachments to the Containment Liner, Section III, Division 2*	8/28/2008
N-777	*Calibration of C_v Impact Test Machines, Section III, Divisions 1, 2, and 3*	10/10/2008
N-811	*Alternative Qualification Requirements for Concrete Level III Inspection Personnel, Section III, Division 2*	8/5/2011
N-820	*Twisting of Horizontal Prestressing Tendons, Section III, Division 2*	12/29/2011
N-822-4	*Application of the ASME Certification Mark, Section III, Divisions 1, 2, 3, and 5*	8/2/2016
N-833	*Minimum Nonprestressed Reinforcement in the Containment Base Mat or Slab Required for Concrete Crack Control, Section III, Division 2*	1/2/2013

16. Conditionally Acceptable Section III, Division 2, Code Cases

The Code Cases listed in Table 2 below are acceptable for application in the design and construction of concrete containments for water-cooled nuclear power plants subject to the regulatory conditions stated in Table 2. Unless otherwise stated, regulatory conditions indicated are in addition to the conditions specified in the Code Case.

Table 2. Conditionally Acceptable Section III, Division 2, Code Cases

CODE CASE NUMBER	CODE CASE TITLE CONDITION(S) FOR ACCEPTABILITY	DATE
N-850	*Equivalent Rectangular Compressive Stress Block for Concrete Containment, Section III, Division 2* Code Case N-850 is acceptable conditioned on correction of errors as follows: (i) In subparagraph (a)(1), replace $\alpha = \beta_{1c}$ with $\alpha = \beta_1$ c (ii) In subparagraph (a)(3), replace the first sentence "For factored primary loads, α_1 shall be taken as 0.06 and β_1 shall be taken as 0.70." with "For factored primary loads, α_1 shall be taken as 0.6 and β_1 shall be taken as 0.70."	10/20/2014
N-852	*Application of the ASME NPT Stamp, Section III, Divisions 1, 2, 3, and 5* Licensees may use the NPT Code Symbol Stamp with the letters arranged horizontally as specified in ASME BPV Code Case N–852 for the service life of a component that had the NPT Code Symbol Stamp applied during the time period from January 1, 2005, through December 31, 2015.	2/9/2015

17. Unacceptable Section III, Division 2, Code Cases

The Section III, Division 2, Code Cases listed in Table 3 below are unacceptable for use by licensees in their Section III, Division 2, design and construction programs.

Table 3. Unacceptable Section III, Division 2, Code Cases

CODE CASE NUMBER	CODE CASE TITLE REASON(S) FOR UNACCEPTABILITY	DATE
N-791	*Shear Screw and Sleeve Splice, Section III, Division 2* There is no slip criterion for this code case. The staff believe that ASTM A1034/A1034M-2010a, "Standard Test Methods for Testing Mechanical Splices for Steel Reinforcing Bars" (Ref. 55), could be used as a good model to develop definition and test methods for slip. Concrete containments in nuclear power plants are important structures; therefore, their criteria for mechanical splices should not be less stringent than those of other seismic Category I structures, as defined in ACI 349-13. The design criterion for concrete containment structures is based on allowable strains for the steel reinforcing bars. The purpose of this strain criterion is partially to prevent the tearing of steel liner plates, which are attached to the inside face of the containment and serve as a leaktight pressure boundary by limiting strains in both concrete and steel reinforcing bars in containment. The mechanical splices should not be allowed to have a significant slip that would cause the strain from the steel reinforcing bars to be transferred to the steel liner plates. Therefore, the Code Case needs to develop a slip criterion for mechanical splices.	9/20/10

CODE CASE NUMBER	CODE CASE TITLE REASON(S) FOR UNACCEPTABILITY	DATE
	ACI 349-13, Section 21.1.6.1, classifies mechanical splices as Type 1 or Type 2. The criterion for Type 1 mechanical splices is that a mechanical splice shall develop no less than 125 percent of the specified minimum yield strength of the spliced bar, as stated in Section 12.14.3.2 of the Code Case. Type 1 mechanical splices are not allowed to be used in regions that may experience steel yielding. The criterion for Type 2 mechanical splices is that a mechanical splice shall develop the specified tensile strength of the spliced bar, as stated in Section 21.1.6.1 of the Code. Concrete containments in nuclear power plants are important safety-related structures; therefore, their criteria for mechanical or welded splices should not be less stringent than that of other seismic Category I structures, as defined in ACI 349-13. The containment structure could experience stresses in the yielding regions under design-basis accident and design basis earthquake conditions. The analysis to calculate the ultimate capacity of reinforced concrete containment structures against the internal pressure as a measure of the safety margin above the design-basis accident pressure could be based on attaining a global hoop strain of 1 percent in the inside reinforcement (Ref. 20), which is significantly higher than the specified minimum yield strength. Further, containment structures are expected to be evaluated for capability to withstand beyond-design-basis accidents, earthquakes (at least 1.67 times the SSE), and other extreme events (e.g., aircraft impact). Consequently, Type 2 mechanical splices must be used in concrete containment structures. Therefore, the criterion in Section 2.3 of Code Case N-791 is the equivalent criterion for Type 1 mechanical splices of ACI Standard 318, which is not an adequate criterion for qualifying mechanical splices for use in concrete containment structures. Therefore, the Code Case should develop a more stringent strength criterion consistent with Type 2 splice for splice qualification, and the same criterion should also be used for continuing splice performance tests in the field, as stated in Section 5 of the Code Case.	
N-793	*Extruded Steel Sleeves with Parallel Threaded Ends, Section III, Division 2* See comments for Code Case N-791.	9/20/10
N-794	*Swaged Splice with Threaded Ends, Section III, Division 2* See comments for Code Case N-791.	9/20/10
N-796	*Threaded Sleeves with Parallel Threads Cut on Upsized Bar Ends, Section III, Division 2* See comments for Code Case N-791.	10/18/10

CODE CASE NUMBER	CODE CASE TITLE REASON(S) FOR UNACCEPTABILITY	DATE
N-807	*Use of Grades 75 and 80 Reinforcement in Concrete Containments, Section III, Division 2* The NRC does not endorse the generic use of higher grades of steel for the reinforcement of concrete containment construction. Among other potential issues, higher grades may reduce the ductility of the steel reinforcement and thereby reduce the overall ductility of the containment structure, which is undesirable. Also, research and development related to use of high-strength reinforcement is still progressing, and there is a lack of operating experience with the use of high-strength reinforcement in the U.S. nuclear power industry.	4/20/2011
N-837	*Alternative to the Registered Professional Engineer Requirements, Section III, Divisions 1, 2, 3, and 5* This Code Case is only for non-U.S. nuclear facilities; therefore, it does not apply to U.S. nuclear facilities regulated by the NRC.	10/22/2013

D. IMPLEMENTATION

The NRC staff may use this RG 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 RG 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. 56), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52. 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 RG 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

1. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."[2]

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

3. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," Subsection IWE and Subsection IWL, New York, NY.[3]

4. U.S. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 3, "Design of Structures, Components, Equipment, and Systems," Section 3.8.1, "Concrete Containment," Revision 4, Washington, DC, September 2013.[4] (ADAMS Accession No. ML13198A245)

5. NRC, Regulatory Guide (RG) 1.7, "Control of Combustible Gas Concentrations in Containment," Washington, DC.

6. NRC, RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," Washington, DC.

7. NRC, RG 1.29, "Seismic Design Classification," Washington, DC.

8. NRC, RG 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," Washington, DC.

9. NRC, RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," Washington, DC.

10. NRC, RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants," Washington, DC.

---

[2] Copies of the *Code of Federal Regulations* are available electronically through the U.S. Government Publishing Office Web site at https://www.ecfr.gov/cgi-bin/ECFR?page=browse.

[3] Copies of American Society of Mechanical Engineers (ASME) standards may be purchased from ASME, Three Park Avenue, New York, NY 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web site store at http://www.asme.org/Codes/Publications/.

[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.

11. NRC, RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Washington, DC.

12. NRC, RG 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," Washington, DC.

13. NRC, RG 1.92, "Combining Modal Response and Spatial Components in Seismic Response Analysis," Washington, DC.

14. NRC, RG 1.107, "Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures," Washington, DC.

15. NRC, RG 1.115, "Protection Against Turbine Missiles," Washington, DC.

16. RG 1.117, "Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants," Washington, DC.

17. NRC, RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," Washington, DC.

18. NRC, RG 1.199, "Anchoring Components and Structural Supports in Concrete," Washington, DC.

19. NRC, RG 1.208, "A Performance-Based Approach To Define the Site-Specific Earthquake Ground Motion," Washington, DC.

20. NRC, RG 1.216, "Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure," Washington, DC.

21. NRC, RG 1.217, "Guidance for Assessment of Beyond-Design-Basis Aircraft Impacts," Washington, DC.

22. NRC, RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," Washington, DC.

23. NRC, RG 1.231, "Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants," Washington, DC.

24. American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI), ASCE/SEI 37-14, "Design Loads on Structures during Construction," Reston, VA, 2014.[5]

25. ASME B&PV Code, Section III, "Rules for Construction of Nuclear Facility Components," Division 2 (ACI Standard 359-19), "Code for Concrete Containments," and Subsection NCA, "General Requirements for Division 1 and Division 2," 2019 Edition, New York, NY, 2019.

---

[5] Copies of reports from the American Society of Civil Engineers (ASCE) are available through their Web site (https://www.asce.org), or by contacting their home office at American Society of Civil Engineers, 1801 Alexander Bell Drive, Reston, VA, 20191; telephone (800) 548-2723.

26. NRC, RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” Revision 38, Washington, DC, October 2019 (ADAMS Accession No. ML19128A276).

27. NRC, RG 1.193, “ASME Code Cases Not Approved for Use,” Revision 6, Washington, DC, October 2019 (ADAMS Accession No. ML19128A269).

28. ASME, B&PV Code, “Code Cases—Nuclear Components,” 2019 Edition, New York, NY, July 1, 2019.

29. American Concrete Institute (ACI) Standard 318-14, “Building Code Requirements for Structural Concrete (ACI 318-14), Commentary on Building Code Requirements for Structural Concrete (ACI 318R-14),” Farmington Hills, MI, 2014).[6]

30. American Society for Testing and Materials (ASTM) A615, “Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete,” West Conshohocken, PA, 2016.[7]

31. ASTM A706, “Standard Specification for Deformed and Plain Low-Alloy Steel Bars for Concrete Reinforcement,” West Conshohocken, PA, 2016.

32. ASTM C1610/C1610M–19, “Standard Test Method for Static Segregation of Self-Consolidating Concrete Using Column Technique,” West Conshohocken, PA, 2019.

33. ASTM C1611/C1611M–18, “Standard Test Method for Slump Flow of Self-Consolidating Concrete,” West Conshohocken, PA, 2018.

34. ASTM C1712/C1712M–17, “Standard Test Method for Rapid Assessment of Static Segregation Resistance of Self-Consolidating Concrete Using Penetration Test,” West Conshohocken, PA, 2017.

35. ASTM C1758/C1758M–15, “Standard Practice for Fabricating Test Specimens with Self-Consolidating Concrete,” West Conshohocken, PA, 2015.

36. ACI 237R-07, “Self-Consolidating Concrete,” Farmington Hills, MI, 2007.

37. ASTM A970/A970M, “Standard Specification for Headed Steel Bars for Concrete Reinforcement,” West Conshohocken, PA, 2018.

[6] Copies of ACI publications may be purchased from ACI, 38800 Country Club Dr., Farmington Hills, MI 48331-3439, telephone (248) 848-3700. Purchase information is available through the ACI Web site at https://www.concrete.org

[7] Copies of ASTM publications may be purchased from ASTM, 100 Barr Harbor Drive, P.O. Box C700, West Conshohocken, PA 19428-2959; telephone (877) 909-2786. Purchase information is available through the ASTM Web site at http://www.astm.org/.

38. ASTM A29/A29M, "Standard Specification for General Requirements for Steel Bars, Carbon and Alloy, Hot-Wrought," West Conshohocken, PA, 2016.

39. ACI 305.1-14, "Specification for Hot Weather Concreting," Farmington Hills, MI, 2014.

40. ACI 306.1-90, "Standard Specification for Cold Weather Concreting," Farmington Hills, MI, 1990 (R2002).

41. ACI 306R-16, "Standard Specification for Cold Weather Concreting," Farmington Hills, MI, 2016.

42. ACI 308.1-11, "Standard Specification for Curing Concrete," Farmington Hills, MI, 2011.

43. ACI 305R-10, "Guide to Hot Weather Concreting," Farmington Hills, MI, 2010.

44. ACI 308R-16, "Guide for External Curing Concrete," Farmington Hills, MI, 2016.

45. ACI 318-19, "Building Code Requirements for Structural Concrete (ACI 318-19), Commentary on Building Code Requirements for Structural Concrete (ACI 318R-19)," Farmington Hills, MI, 2019.

46. ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," Farmington Hills, MI, 2013.

47. International Atomic Energy Agency (IAEA) Specific Safety Guide No. SSG-53, "Design of Reactor Containment and Associated Systems for Nuclear Power Plants," Vienna, Austria, 2019.[8]

48. IAEA, Safety Guide No. NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants," Vienna, Austria, 2003.

49. IAEA, Safety Guide No. NS-G-2.6, "Maintenance, Surveillance and In-Service Inspections in Nuclear Power Plants," Vienna, Austria, 2002.

50. IAEA, Safety Guide No. NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants," Vienna, Austria, 2004.

51. IAEA, Specific Safety Guide No. SSG-48, "Aging Management and Development of a Programme for Long Term Operations of Nuclear Power Plants," Vienna, Austria, 2018.

52. IAEA, Nuclear Energy Series No. NP-T-2.5, "Construction Technologies for Nuclear Power Plants," Vienna, Austria, 2011.

53. IAEA Specific Safety Guide SSG-38, "Construction for Nuclear Installations," Vienna, Austria, 2015.

54. ASTM 370-18, "Standard Test Methods and Definitions for Mechanical Testing of Steel Products," West Conshohocken, PA, 2018.

[8] Copies of 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.

55. ASTM A1034/A1034M, "Standard Test Methods for Testing Mechanical Splices for Reinforcing Bars," West Conshohocken, PA, 2010a (Reapproved 2015).

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

APPENDIX A

STAFF POSITION ON BOILING-WATER REACTOR CONTAINMENT POOL DYNAMICS

A-1. LOSS-OF-COOLANT ACCIDENTS

a. Loads associated with loss-of-coolant accidents (LOCAs) (such as drywell pressure; annulus pressurization; suppression pool pressures including chugging, condensation oscillation, vent clearing, and pool swell; and jet loads and drag loads acting on submerged structures/components and those above the water surface) should be treated as abnormal pressure loads, P_a. Appropriate load combinations and load factors should be applied accordingly.

b. All of the various LOCA-induced loads may be combined in accordance with their actual time-dependent mutual occurrence.

A-2. SAFETY/RELIEF VALVE DISCHARGE

a. Safety/relief valve (SRV) loads (such as single valve, single valve plus adjacent valve, automatic depressurization system valves, and all valves) should be included in all load combinations as defined in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subsection CC. However, for the load combination that contains 1.5 P_a, a load factor of 1.25 should be applied to the appropriate SRV loads. Also, for the severe environmental load combination, the load factor should be 1.3, which is consistent with the practice of treating SRV as a live load.

b. A single active failure causing one SRV discharge should be considered in combination with the design-basis accident.

c. Appropriate multiple SRV discharges should be considered in combination with the small-break accident and intermediate-break accident.

d. Thermal loads resulting from SRV discharge should be treated as T_o for normal operation and T_a for accident conditions.

A-3. COMBINATION OF DYNAMIC LOADS

Revision 1 of NUREG-0484, "Methodology for Combining Dynamic Responses" (Ref. A1), presents guidance on the acceptable methods for combining LOCA, SRV, and the safe-shutdown earthquake (SSE). This report states that the square root of sum of the squares method is appropriate for the following:

a. Combination of SSE and LOCA dynamic responses for all ASME Class 1, 2, or 3 systems, components, or supports. For dynamic responses resulting from the same initiating event, when the time-phase relationship between the responses cannot be established, the absolute summation of these dynamic responses should be used.

b. Combining responses of dynamic loads other than LOCA and SSE, if a non-exceedance probability of 84% or higher is achieved for the combined SRSS [square root of sum of the squares] response.

Although the above criteria were prepared for ASME systems, components, and supports, they are applicable to seismic Category I structures as well.

A-4. REFERENCES

A1. NUREG-0484, "Methodology for Combining Dynamic Responses," Revision 1, Washington, DC, May 1980. (ADAMS Accession No. ML13260A310)