U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.142, REVISION 3

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SAFETY-RELATED CONCRETE STRUCTURES FOR NUCLEAR POWER PLANTS (OTHER THAN REACTOR VESSELS AND CONTAINMENTS)

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable to demonstrate compliance with NRC regulations for the analysis, design, construction, testing, and evaluation of safety related nuclear concrete structures, excluding concrete reactor vessels and concrete containments.

Applicability

This RG applies to applicants and licensees 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 Part 50, Appendix A, provides the "General Design Criteria for Nuclear Power Plants," establishes design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety through general design criteria (GDC). GDC applicable to this RG include the following:

  - General Design Criterion (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,

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reflecting the appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

   - 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 associated with normal operation, maintenance, testing, and postulated accidents.

- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," provides quality assurance (QA) requirements that apply to all activities (e.g., designing, fabricating, erecting, inspecting, testing, modifying) affecting the safety-related functions of SSCs.

- 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," provides, in part, criteria for the implementation of GDC 2 with respect to earthquakes.

- 10 CFR Part 52 (Ref. 2) governs the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities licensed under Section 103 of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).

   - 10 CFR 52.47 provides requirements on the content of technical information for standard design certifications submitted under Part 52.

   - 10 CFR 52.77 and 52.79 provide requirements on the technical content of combined operating license applications.

Related Guidance

- RG 1.28, "Quality Assurance Program Criteria (Design and Construction)" (Ref. 3), describes acceptable methods for establishing and implementing a quality assurance (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. 4), provides guidance for identifying and classifying features of light-water-reactor (LWR) nuclear power plants that must be designed to withstand the effects of a safe shutdown earthquake (SSE).

- RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants" (Ref. 5), describes a method acceptable to NRC staff for determining the minimum thickness, based on radiation shielding requirements (only), of concrete radiation shields in nuclear power plants.

- RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," (Ref. 6) 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 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments" (Ref. 7), provides guidance on containment design, construction and testing.

- RG 1.199, "Anchoring Components and Structural Supports in Concrete" (Ref. 8), provides guidance for the design, testing, evaluation, and QA, including installation and inspection of anchors (steel embedment) used for anchoring component and structural supports on concrete structures.

- RG 1.217, "Guidance for Assessment of Beyond-Design-Basis Aircraft Impacts" (Ref. 9), describes considerations for aircrafts impacts for new nuclear power reactors.

- RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," (Ref. 10) provides new 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.

- NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 11), provides guidance to the NRC staff in the review of safety analysis reports submitted as part of a license application.

- American Society of Civil Engineers/Structural Engineering Institute, ASCE/SEI 37-14, "Design Loads on Structures during Construction," (Ref. 12), describes the minimum design requirements for construction loads, load combinations, and load factors affecting buildings and other structures that are under construction. It addresses partially completed structures as well as temporary support and access structures used during construction. The loads specified are suitable for use either with strength design criteria, such as ultimate strength design and load and resistance factor design, or with allowable stress design criteria. The loads are applicable to all conventional construction methods.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the NRC's regulations, techniques that the staff uses in evaluating specific issues or postulated events, and information that the staff needs in its review of applications for permits and licenses. RGs are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG 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, DC20503; e- mail: oira_submission@omb.eop.gov.

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B. DISCUSSION

Reason for Revision

This revision (Revision 3) of RG 1.142 endorses, with certain exceptions, American Concrete Institute (ACI) 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary" (Ref. 13), except for Appendix D, "Anchoring to Concrete." Appendix D to ACI 349-13 is addressed by RG 1.199, "Anchoring Components and Structural Supports in Concrete."

Background

ACI 349-13 is based on ACI 318-08, "Building Code Requirements for Structural Concrete and Commentary" (Ref. 14), with modifications in ACI 349-13 to accommodate the requirements specific to nuclear safety-related concrete structures.

Discussion of Regulatory Positions

The following provides background technical information for a selection of regulatory positions in Section C of this guide to provide clarification for those provisions.

Regulatory Position 1

Commentary Section R1.5 of ACI 349-13, "Quality assurance program," references ASME NQA-1-2000 for detailed requirements for development and implementation of a quality assurance program. However, NQA-1-2000 has been superseded by more recent versions that are endorsed with exceptions and clarifications in RG 1.28, "Quality Assurance Program Criteria (Design and Construction) (Ref. 12).

Regulatory Position 2

This position emphasizes the need to evaluate concrete structures for their effectiveness as radiation shields, when they are so intended, and RG 1.69, "Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants", provides staff guidance for the design, construction, and testing of concrete radiation shields in nuclear power plants.

Regulatory Position 3

This position addresses acceptable standards for design of pressure resisting portions of concrete structures within the reactor containment. To ensure that the pressure retaining functions and leak tightness of those structures are not compromised during the loss-of-coolant accident (LOCA), Regulatory Position 3 provides conditions that complement the ultimate strength design (USD) approach in ACI 349-13.

Regulatory Position 4

4.1 In complex structural systems, the definitions of structural components such as walls, slabs, and foundations provided in ACI 349-13 may not be completely adequate for nuclear safety-related structures. This position alerts the designer to consider whether structural components are acting as parts of flexural frames. Because structural components generally experience flexural effects, they are designed as a frame when the flexural moment from seismic loads equals or exceeds a large percentage of the flexural capacity. By setting this limiting ratio at two-thirds, the flexure from seismic loads alone would be within the design capacity even with a seismic event equal to 150 percent of the SSE.

4.2 The qualification and testing of mechanical splices, mechanically headed deformed bar systems and welded splices are not included in ACI 349-13. However, CC-4333 and CC-4334 of American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section III, Division 2 (ACI 359-19), "Code for Concrete Containments," (Ref.11) provides such guidance. For this reason, it was incorporated into the guidance for Regulatory Position 4.2.

4.3 Use of high-strength (HS) reinforcement (Grade 75 and 80) as used in ACI 349-13 is not endorsed for general use. Research and development that integrates implications for the general use of, for example, crack control, material and component ductility, deflection limits, over-strength factors, and strength-reduction factors, is ongoing and, therefore, its use is not generically endorsed.

4.4 This position endorses the provisions in paragraphs (a) and (b) of Section 21.1.5.1 in ACI 349-13 for American Society of Testing and Materials (ASTM) A615 (Ref. 16) Grade 60 reinforcement and provides conditions for minimum acceptable elongation. The conditions for minimum acceptable elongation are those in ASTM A706 for Grade 60 (Ref. 17) deformed reinforcement.

4.5 For deep beams, ACI 349-13 incorporates Section 11.7 of ACI 318-08 by reference. For deep beams, the minimum area for shear reinforcement (A_v), perpendicular to the flexural tension reinforcement, in Section 11.7.4 of ACI 318-08 has a lower limit of not less than $0.0025b_w s$. The minimum area for shear reinforcement (A_{vh}), parallel to the flexural tension reinforcement for deep beams in Section 11.7.5 of ACI 318-08 has a lower limit of not less than $0.0015b_w s_2$. For better control of the growth and width of diagonal cracks, Regulatory Position 4.5 increases the area of shear reinforcement parallel to the flexural tension reinforcement to $0.0025b_w s_2$.

4.6 On the tension face of a structural slab, wall, or shell, the NRC staff does not endorse the "one-third greater than that required by analysis" rule for minimum reinforcement in Section 7.12.4 of ACI 349-13. This provision is not supported by sufficient justification in the commentary of ACI 349-13.

4.7 This position specifies the minimum reinforcement requirement for a foundation base mat. Section 7.12.4 of ACI 349-13 does not specifically address the minimum reinforcement for a foundation base mat.

4.8 Section 11.9 of ACI 349-13 incorporates by reference Section 11.9.1 of ACI 318-08 for the design of shear forces perpendicular to the face of the wall. Section 11.9.1 of ACI 318-08 refers to Section 11.11 for the design of shear forces perpendicular to face of wall. Section 11.11.1.1 of ACI 318-08 states that for the beam action each critical section to be investigated extends in a plane across the entire width of the wall or slab, which is not a conservative approach. Thus, the staff does not endorse that approach. Instead, the effective width of the critical section should be determined according to the zone of influence induced by the concentrated loads.

Regulatory Position 5

5.1 The NRC staff endorses the new set of load combinations contained in ACI 349-13, Section 9.2, with the adjustments described below. With those adjustments, the NRC staff position is to use load factors that follow, to a large extent, those in Section 9.2 of ACI 349-13, with added conservatism for the design of safety-related nuclear power plant concrete structures.

Operational temperature. In the design of nuclear power plant concrete structures, the operational temperature loading, T_o, is considered as a live load. Though extremes of anticipated temperatures are considered for this purpose, the computational methods of cracked section analysis and the extent of cracking do not lend themselves to the same degree of confidence in assessing its effect as those for a dead load computation. The NRC staff position is to use a load factor of 1.6 for T_o in Regulatory Position 5.1.

Pressure. To a certain degree, the structural systems required to withstand pressures are related to the release of radioactivity to the atmosphere. In this regard, the structural systems could function as a direct barrier or as a support for a direct barrier. Also, the characteristics of the pressure transients would depend, in most cases, on the appropriate functioning of various engineered safety features and other backup systems. Considering a band of uncertainty in the magnitude and duration of the energy levels associated with a pipe rupture, the NRC staff position is to use a comparatively conservative load factor for P_a in Regulatory Position 5.1.

Piping and equipment reaction. According to Section 2.1 of ACI 349-13, R_o does not include dead load and earthquake reactions (those components of R_o are included in D and E_o/E_{ss}). Assessing R_o is associated with larger uncertainty than dead load. ACI 349-13, Appendix C, applies a load factor of 1.7 to R_o, which is similar to a live load. In Regulatory Position 5.1, the staff modified the load factor for R_o to treat it as a live load in load combinations (9-2), (9-3), and (9-4).

Live load. To be consistent with ACI 318-08 and Appendix C to ACI 349-13, the NRC staff position is to use a load factor of 1.0 in place of 0.8 for the live load in load combinations (9-6), (9-7), and (9-8).

Stress, strain, and deformation limits for the operating-basis earthquake. Consistent with Appendix S to 10 CFR Part 50, SSCs important to safety must remain functional and within applicable stress, strain, and deformation limits for operating-basis earthquake E_o. Hence, when the effects of E_o are considered in load combination (9-4), the NRC staff position is to include C_{cr} as an operating load with a load factor of 1.4.

5.2 Structural integrity is achieved during construction by implementing appropriate construction sequences to avoid creating vulnerabilities in partially completed structures, and by considering in the design the construction loads induced in partially completed structures and temporary structures and supports, as well as applicable load combinations. Structural Engineering Institute/American Society of Civil Engineers (ASCE) Standard 37, "Design Loads on Structures during Construction" (ASCE 37), gives additional guidance on construction loads. The NRC staff does not endorse ASCE 37; however, the staff recognizes that this standard can provide additional information. To the extent licensees or applicants rely on ASCE 37, they should provide sufficient basis and information to the staff to verify consistency applicable to NRC requirements. And, in cases where the criteria in ASCE 37 conflict with this RG, licensees and applicants should follow the positions in this RG.

5.6 The commentary of ACI 349-13 does not provide sufficient justification for the 10 percent reduction in the load factor for the effects of the seismic load E_{ss} in Section 9.2.10 of ACI 349-13. Therefore, in load combinations (9-6) and (9-9), the NRC does not endorse reducing the load effects of the seismic load E_{ss} by 10 percent provided in Section 9.2.10 of ACI 349-13.

5.7 This regulatory position refers to strength reduction factor (φ) for shear critical members such as low-rise shear walls, portions of walls between openings or diaphragms. The regulatory position reflects the lower ductility and ability to redistribute loads of these shear critical members as compared to flexure-critical members. In Section 9.3.4 and R9.3.4 of ACI 318, φ = 0.60 is used for shear critical members. Commentary section R9.3.1 of ACI 349-13 recommends using φ = 0.75 for such members but the justification provided for using higher φ value is not defensible for general use. The NRC Staff position is to provide margin to account for uncertainties and variabilities in the shear capacity and the consequences of brittle failure. The value for φ should be used depending on the ductile or non-ductile (brittle) limit state. This discussion also applies to Appendix C, Section C.9.3 and RC.9.3.4 of ACI 349-13.

5.8 ACI 349-13, Section 21.9.9, discusses provisions for columns supporting discontinuous shear walls. The use of this type of structural configuration causes irregularities and adversely affects stiffness of the structure and induces additional shear and torsional forces. Therefore, the NRC staff does not endorse the structural configuration described in Section 21.9.9 because it is inappropriate for use in nuclear safety-related structures.

Regulatory Position 6

This position endorses ACI 349-13, Appendices A, C, and F, with certain clarification and conditions that reflect the existing NRC staff review practices. Regulatory Guide 1.199 endorses, with certain exceptions and conditions, Appendix D of ACI 349-13.

6.2.3 The technical background information for this regulatory position is identical to that for Regulatory Position 5.7.

Regulatory Position 7

This position supplements the jurisdictional boundaries between ACI 349-13 and ACI 359-19. ACI 349-13 provides the minimum requirements for design and construction of nuclear safety-related concrete structures (other than containments) and structural members for nuclear facilities, whereas ACI 359-19 covers the design and construction for concrete containments.

Harmonization with 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 for protecting people and the environment from harmful effects of ionizing radiation. These standards provide a system of safety fundamentals, safety requirements, and safety guides reflecting an international consensus on what constitutes a high level of safety. The following IAEA safety guides or standards contain guidance and safety principles similar to those in this RG:

- Safety Guide NS-G-1.5, "External Events Excluding Earthquakes in the Design of Nuclear Power Plants" (Ref. 18)
- Safety Guide NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants" (Ref. 19)
- Safety Guide NS-G-2.6, "Maintenance, Surveillance and In-Service Inspections in Nuclear Power Plants" (Ref. 20)

- Safety Guide NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants" (Ref. 21)

- Safety Report 70, Management System Standards "Comparison between IAEA GS-R-3 and ASME NQA-1-2008 and NQA-1a-2009 Addenda" (Ref. 22)

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 a 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 a 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

This revision (Revision 3) of RG 1.142 endorses, with certain exceptions, ACI 349-13. It provides an adequate basis for complying with the NRC's regulations with regard to the analysis, design, construction, testing, and evaluation of nuclear safety-related concrete structures (other than reactor vessels and containment structures), subject to the positions listed below.

Regulatory Position 1

RG 1.28 should be used for quality assurance in design and construction and for inspection and testing of concrete structures covered by this RG.

Regulatory Position 2

RG 1.69 should be used when concrete is used as a radiation shield for concrete structures covered by this RG.

Regulatory Position 3

ACI 349-13 provides acceptable standards for the design of pressure-resisting portions of concrete structures within the reactor containment. The structures include special structures, such as the pressure-resisting portion of the drywell of Mark III containments (e.g., General Electric BWRs), the divider barrier of ice-condenser containments, or the dividing slab (drywell floor) between the drywell and the wetwell (suppression chamber) of Mark II containments (e.g., General Electric BWRs). GDC 16, "Containment Design," requires these structures to maintain a certain degree of leak tightness during a LOCA. If ACI 349-13 is used to design these types of special structures that function as pressure-resisting and leakage barriers then add the following provisions to ACI 349-13 or provide equivalent provisions:

a. provision for crack control under design load combinations that include LOCA loads;

b. provisions to deal with the transition from the concrete portion of the drywell to the steel portion of the drywell; and

c. provisions for preoperational testing and inservice inspections.

Regulatory Position 4

4.1 Where interconnected structural components, such as walls, slabs, and foundations, exhibit a structural response consistent with the response of structural frames, such components should conform to the requirements of Chapters 10, 11, and 21 of ACI 349-13, in addition to Chapters 13, 14, and 15 as appropriate. Treat the response of structural components in a manner consistent with the response of structural frames if the flexural moment from seismic loads is equal to or exceeds two-thirds of the design flexural capacity of the section in the absence of axial forces.

4.2 Mechanical splices, mechanically headed deformed bar systems and welded splices

4.2.1 For performance criteria of mechanical splices and mechanically headed deformed bar systems, follow the additional qualification requirements in CC-4333 from ASME B&PV Code, Section III, Division 2 (ACI 359-19), "Code for Concrete Containments," with the addition that the strength of an individual mechanical splice and of an individual mechanically headed deformed bar system should be no less than the specified tensile strength of the reinforcement for consistency with Section 12.14.3.2 of ACI 349-13. Any exception to compliance should be supported with adequate justification based on documented testing.

Test and evaluate the mechanical splices in accordance with ASTM A1034, "Standard Test Methods for Testing Mechanical Splices for Steel Reinforcing Bars" (Ref. 23).

Perform additional testing and evaluation for impactive and impulsive loading conditions, where the mechanical splices are subjected to such loading that causes high strain rate.

4.2.2 For qualification of welding of reinforcing bars, follow the requirements of CC-4334 of American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code Section III, Division 2 (ACI 359-19, "Code for Concrete Containments," with the addition that the strength of an individual full welded splice should be no less than the specified tensile strength of the reinforcement for consistency with Section 12.14.3.4 of ACI 349-13. Any exception to compliance should be supported with adequate justification based on documented testing.

4.3 This RG does not endorse, in general, the use of HS reinforcement (Grade 75 and 80) as used in ACI 349-13. Applicants should demonstrate the adequacy of HS reinforcement for specific use in the design by testing, analysis, or performance evaluation.

4.4 In addition to Sections 21.1.5.1(a) and (b) and Sections F4.1(a) and (b) of ACI 349-13, for ASTM A615 Grade 60 reinforcement, the minimum elongation in 8-inch-long reinforcing bar shall be at least the following:

a. 14 percent for bar sizes No. 3 through No. 6,

b. 12 percent for bar sizes No. 7 through No. 11, or

c. 10 percent for bar sizes No. 14 and No. 18.

4.5 For deep beams, the staff does not endorse the area of minimum shear reinforcement (A_{vh}) parallel to the longitudinal axis of the beam in Section 11.7.5 of ACI 318-08. The area of minimum shear reinforcement (A_{vh}) parallel to the longitudinal axis of the beam should be $0.0025 b_w s_2$.

4.6 The staff does not endorse the application of "one-third greater than that required by analysis" provision to the minimum reinforcement provided in Section 7.12.4 of ACI 349-13.

4.7 Section 7.12.4 of ACI 349-13 does not address minimum reinforcement for the foundation base mat. The ratio of nonprestressed reinforcement area to gross concrete area, on a tension face of a foundation base mat, where a calculated reinforcement requirement exists, should not be less than 0.0018 in the direction of the span under consideration.

4.8 To determine the shear strength of slabs and walls for concentrated loads or reactions perpendicular to the plane of the slabs and walls, the effective width of the critical section for the beam action condition is the zone of influence induced by the concentrated loads instead of the entire width of the slab as specified in Section 11.11.1.1 of ACI 318-08, which is incorporated by reference in Section 11.9 of ACI 349-13. The effective zone of influence may be determined, for example, by analysis.

Regulatory Position 5

5.1 The staff endorses the load combinations in Section 9.2 of ACI 349-13 as modified in Table 1 (modifications indicated in ***bold italic*** font).

Table 1. Loads, Load Factors and Load Combinations

Required Strength (U) Load Combinations	ACI Equations
$U = 1.4(D + F) + \boldsymbol{R_o} + T_o$	(9-1) (NL)
$U = 1.2(D + F) + 1.6(L + H \boldsymbol{+ R_o + T_o}) + 1.4C_{cr} + 0.5(L_r \text{ or } S \text{ or } R)$	(9-2) (NL)
$U = 1.2(D + F) + 0.8(L + H \boldsymbol{+ R_o + T_o}) + 1.4C_{cr} + 1.6(L_r \text{ or } S \text{ or } R)$	(9-3) (NL)
$U = 1.2(D + F) + 1.6(L + H \boldsymbol{+ R_o} + E_o) \boldsymbol{+ 1.4C_{cr}}$	(9-4) (SEL)
$U = 1.2(D + F) + 1.6(L + H \boldsymbol{+ R_o} + W)$	(9-5) (SEL)
$U = D + F + \boldsymbol{1.0L} + C_{cr} + H + T_o + R_o + E_{ss}$	(9-6) (EEL)
$U = D + F + \boldsymbol{1.0L} + H + T_o + R_o + W_t$	(9-7) (EEL)
$U = D + F + \boldsymbol{1.0L} + C_{cr} + H + T_a + R_a + \boldsymbol{1.4P_a}$	(9-8) (AL)
$U = D + F + \boldsymbol{1.0L} + H + T_a + R_a + P_a + Y_r + Y_j + Y_m + E_{ss}$	(9-9) (AL)
NL = Normal Loading condition SEL = Severe Environmental Loading condition EEL = Extreme Environmental Loading condition AL = Abnormal Loading condition	

5.2 Construction loads should be considered in the design of concrete structures. These loads consist of dead loads, live loads, temperature, wind, snow, rain, ice, and applicable construction loads, such as material loads, personnel and equipment loads, horizontal construction loads, erection and fitting forces, equipment reactions, and loads from pressure.

5.3 Hydrodynamic loads associated with seismic loads (i.e., the impulsive and sloshing loads for fluids in tanks) are to be considered as part of E_{ss} in load combinations (9-6) and (9-9). Hydrodynamic loads associated with seismic loads are to be considered as part of E_O in load combination (9-4). All other hydrodynamic loads should be taken as Y_j in load combination (9-9).

5.4 Loads resulting from pool dynamics for the concrete structures in pressure-suppression containments should be analyzed. Licensees and applicants should provide justification for the methods of analysis used.

5.5 Design for loads due to accidental explosions, or accidental vehicle impacts, or small aircraft impacts should use load combination (9-7) with those loads in lieu of W_t, and further guidance provided in ACI 349-13, Appendix F, "Special Provisions for Impulsive and Impactive Effects." RG 1.217 covers effects from beyond-design-basis large aircraft impacts. The W_t load in the load combination (9-7) should be evaluated for both tornado and hurricane loads applicable to the site. Related regulatory guidance is covered in RG 1.76 and RG 1.221.

5.6 The NRC staff does not endorse Section 9.2.10 of ACI 349-13. Therefore, for load combinations (9-6) and (9-9), the staff does not endorse reducing the load effects of the seismic load (E_{ss}) by 10 percent for the conditions in Section 9.2.10 of ACI 349-13.

5.7 The use of a higher value of strength reduction factor $\varphi = 0.75$ for shear walls in Section 9.3.4 and R9.3.4 of ACI 349-13 as compared to the value of $\varphi = 0.60$ for shear-critical shear walls in ACI 318-08 should be justified on a case-by-case basis. The applicant should establish whether specific shear walls in their structure are governed by ductile limit states (such as flexural yielding) or non-ductile limit states (such as concrete crushing or shear failure). Given the complex load combinations for the design of safety-related nuclear structures, and the complex geometry of nuclear structures, the applicant should establish and evaluate the potential limit states for shear walls and establish the governing limit state for different load combinations and scenarios. A higher value of $\varphi = 0.75$ may be used for design when the ductile limit states (such as rebar yielding) govern behavior, and a lower value of $\varphi = 0.60$ should be used when non-ductile limit states (such as concrete crushing or shear failure) govern. Similarly, for Appendix C, Section C.9.3 and RC.9.3.4, a higher value of $\varphi = 0.85$ may be used for design when the ductile limit states (such as rebar yielding) govern behavior, and a lower value of $\varphi = 0.75$ should be used when non-ductile limit states (such as concrete crushing or shear failure) govern.

5.8 ACI 349-13, Section 21.9.9, discusses provisions for columns supporting discontinued shear walls. The staff does not endorse the generic use of this type of structural configuration in nuclear safety-related structures.

Regulatory Position 6

The NRC makes the following exceptions, enhancements, and supplements to ACI 349-13, Appendices A, C, D, and F:

6.1 Appendix A, "Strut-and-Tie Models"

The application of strut-and-tie models in design needs justification by licensees or applicants each time that it is used.

6.2 Appendix C, "Alternative Load and Strength-Reduction Factors"

The load combinations of Appendix C are acceptable alternatives to those of ACI 349-13, Chapter 9, with the exceptions noted below. Table 2 shows all load definitions (e.g., D, L, T_o) as given in ACI 349-13 and ASCE 7-05, "Minimum Design Loads for Buildings and Other Structures," issued 2005 (Ref. 24).

Table 2 Load Definitions

Required Strength (U) Load Combination	ACI Equations
$U = 1.4D + 1.4F + 1.7L + 1.7H + 1.7E_o + 1.7R_o + \boldsymbol{1.4C_{cr}}$	(C9-2) NL + SEL
$U = D + F + L + H + T_a + R_a + \boldsymbol{1.4}P_a + C_{cr}$	(C9-6) NL + AL
$U = 1.05D + 1.05F + 1.3L + 1.3H + \boldsymbol{1.2}T_o + 1.3R_o$	(C9-9) 75% NL + TL
$U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3E_o + \boldsymbol{1.2}T_o + 1.3R_o$	(C9-10) 75% (NL + SEL) + TL
$U = 1.05D + 1.05F + 1.3L + 1.3H + 1.3W + \boldsymbol{1.2}T_o + 1.3R_o$	(C9-11) 75% (NL + SEL) + TL

6.2.1 Hydrodynamic loads associated with seismic loads (i.e., the impulsive and sloshing loads for fluids in tanks) are to be considered as E_{ss} in load combinations (C9-4) and (C9-8). Hydrodynamic loads associated with seismic loads are to be considered as E_o in load combinations (C9-2), (C9-7), and (C9-10). All other hydrodynamic loads should be taken as Y_j in load combinations (C9-7) and (C9-8).

6.2.2 Design for loads due to accidental explosions, or accidental vehicle impacts, or accidental small aircraft impacts should use load combination (C9-5), with those loads used in lieu of W_t, with the further guidance provided in ACI 349-13, Appendix F. RG 1.217 covers the effects from beyond-design-basis large aircraft impacts.

6.2.3 Use $\varphi = 0.60$ in C.9.3 and RC.9.3 of ACI 349-13 for shear critical members, reflecting the lower ductility of shear critical members than flexure-critical members. This is identical to regulatory position 5.7.

6.3 Appendix D, "Anchoring to Concrete"

RG 1.199 endorses Appendix D with exceptions.

6.4 Appendix F, "Special Provisions for Impulsive and Impactive Effects"

The NRC staff generally finds acceptable the local exceedance of section strengths in accordance with Appendix F under the impactive and impulsive loadings described in Sections F1.4 and F1.5 (e.g., accidental explosions and vehicle or small aircraft impacts, turbine missiles, and a localized pressure transient during an explosion). The permissible ductility in Appendix F is acceptable with the following exceptions:

6.4.1 In Section F.3.5, the maximum permissible ductility ratios (μ) when a concrete structure is subjected to a pressure pulse caused by compartment pressurization or blast loading should be the following:

a. For the structure as a whole, (μ) = 1.0.

b. For a localized area in the structure, (μ) = 3.0.

6.4.2 In Section F.3.7(a) and (b), when shear controls the design, the maximum permissible ductility ratios should be the following:

a. When shear is carried by concrete alone, (μ) = 1.0.

b. When shear is carried by a combination of concrete and stirrups, headed bars, or bent bars, (μ) = 1.3.

6.4.3 In Section F.3.8(a) and (c), the maximum permissible ductility ratio in flexure should be as follows:

a. When compression controls the design, as defined by a load-moment interaction diagram, (μ) = 1.0;

b. When the compression load is less than $0.1f'_c A_g$ or one-third of that which would produce balanced conditions, whichever is smaller, the permissible ductility ratio should be same as that given in Sections F.3.3 or F.3.4 of ACI 349-13;

c. Vary linearly from (μ) = 1.0 to that given in Sections F.3.3 or F.3.4 of ACI 349-13 for conditions between those described in Section F.3.8(a) and (b).

6.4.4 The dynamic increase factors (DIFs) for flexure are acceptable as given in Section F.2.1 of ACI 349-13. However, DIFs for diagonal tension, direct shear, and bond strength should be as follows:

a. DIF for diagonal tension and direct shear for reinforcing steel (stirrups) should be taken as 1.0, as specified in Table 5.4 of the ASCE "Report of the Committee on Impactive and Impulsive Loads," Volume V, issued September 1980 (Ref. 25).

b. DIF for diagonal tension and direct shear (punchout) and bond for reinforced and prestressed concrete should also be taken as 1.0, as specified in Table 5.4 of Ref. 25.

6.4.5 Section F.4.1 of ACI 349-13 allows the use of reinforcing steel with yield strength greater than 60 kilopounds per square inch. Regulatory Position 4.3 gives the NRC staff's position on this subject.

6.4.6 The NRC staff will review the use of Section F.1.3 on a case-by-case basis. For example, the staff will review the use of Equations (RF-7) or (RF-8) instead of the shear provisions of ACI 349-13, as suggested in Section RF.5 with reference to Section F.1.3 of ACI 349-13.

6.4.7 In the case of the reaction shear (beam action) at the supports in section F.5 of ACI 349-13, effective width of the critical section for the shear capacity at the supports is to be determined according to the zone of influence induced by the local loads instead of the entire width of the support (see Regulatory Position 4.8). The zone of influence induced by the concentrated loads may be determined, for example, by an analysis.

Regulatory Position 7

The applicant or licensee should carefully examine and establish the applicability of the standards in ACI 349-13 and ACI 359-19 for other-than-containment safety-related concrete structures that frame into concrete containment structures, and for a foundation base mat that supports containment structures and other nuclear safety-related concrete structures. Those other-than-containment safety-related concrete structures and common foundation base mat should meet the provisions and load combinations in ACI 349-13 and ACI 359-19, as appropriate, to capture structure-to-structure interaction effects such as the effects on one structure resulting from loadings applied to a separate but monolithically connected structure.

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, “Backfitting,” and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests” (Ref. 26), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

REFERENCES[1]

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

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

3. U.S. Nuclear Regulatory Commission (NRC), RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," Washington, DC.

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

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

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

7. NRC, RG 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," Washington, DC.

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

9. NRC, RG 1.217, "Guidance for the assessment of Beyond-Design-Basis Aircraft Impacts," Washington, DC.

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

11. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Washington, DC.

12. American Society of Civil Engineers/Structural Engineering Institute, ASCE/SEI 37-14, "Design Loads on Structures during Construction," Reston, VA,[2] 2014.

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[1] 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.

[2] 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.

13. American Concrete Institute (ACI) 349-13, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary," Farmington Hills, MI,[3] 2013.

14. ACI 318-08, "Building Code Requirements for Structural Concrete and Commentary," Farmington Hills, MI, 2008.

15. American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section III, Division 2 (ACI 359-19), "Code for Concrete Containments," New York, NY,[4] 2019.

16. American Society for Testing and Materials (ASTM) A615, "Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete," West Conshohocken, PA, 2019.[5]

17. ASTM A706, "Standard Specification for Deformed and Plain Low-Alloy Steel Bars for Concrete Reinforcement," West Conshohocken, PA, 2019.

18. International Atomic Energy Agency (IAEA), Safety Guide NS-G-1.5, "External Events Excluding Earthquakes in the Design of Nuclear Power Plants," Vienna, Austria.[6]

19. IAEA, Safety Guide NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants," Vienna, Austria.

20. IAEA, Safety Guide NS-G-2.6, "Maintenance, Surveillance and In-Service Inspections in Nuclear Power Plants," Vienna, Austria.

21. IAEA, Safety Guide NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants," Vienna, Austria.

22. IAEA, Safety Report 70, Management System Standards, "Comparison between IAEA GS-R-3 and ASME NQA-1-2008 and NQA-1a-2009 Addenda," Vienna, Austria.

23. ASTM, A1034, "Standard Test Methods for Testing Mechanical Splices for Steel Reinforcing Bars," West Conshohocken, PA, 2015.

24. ASCE 7-05, "Minimum Design Loads for Buildings and Other Structures," Reston, VA, 2005

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[3] Copies of American Concrete Institute (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

[4] 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-based store at http://www.asme.org

[5] 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/.

[6] Copies of International Atomic Energy Agency (IAEA) publications may be purchased from IAEA, Vienna International Centre, PO Box 100 A-1400 Vienna, Austria, telephone: (+431) 2600-0, or are available through the IAEA Web site https://www.iaea.org/publications.

25. ASCE, "Report of the ASCE Committee on Impactive and Impulsive Loads – Volume V," Second ASCE Conference on Civil Engineering and Nuclear Power, ISBN 0872622487, 1980[7].

26. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC.

[7] Libraries with copies of the Second ASCE Conference on Civil Engineering and Nuclear Power can be located via the OCLC World Catalogue at https://www.worldcat.org/