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Comment On: NRC-2019-0100-0001
Safety Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)

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General Comment

There are seven comments provided in the attached file named NRC_PC_DG-1283_NRC-2019-0100_v2. These represent the comments from the leadership of ACI 349. These are private opinions of these ladies and gentlemen. However, due to time constraints, we are providing preliminary comments and request the permission to issue a final more complete set of comments in a week's time.

Attachments

NRC_PC_DG-1283_NRC-2019-0100_v2

DG-1283 - Proposed RG 1.142 R3 - ML16172A240

ID	Commenter Name	Page #	Section #	A/C, N, Abs.	Comment
1.	ACI 349 Committee Officers	4	2.3	N	Crack control in nuclear safety-related structures is primarily a concern associated with shear, not flexure. ACI 349-13 does not allow grade 75 or grade 80 steel to resist shear or torsion. ACI 349 permits high strength reinforcing to be used only for flexure, where the issue of crack control for these types of structures is generally much less pronounced. The adoption of grade 75/80 reinforcement by the code committee was solve the recurring problem of rebar congestion and constructability issues in safety-related structures. High strength reinforcing has been permitted in European nuclear construction for over a decade and there is an extensive body of research on its use. ACI 318 has recently adopted the use of A706 Grade 80 reinforcing for earthquake-resisting construction, having addressed concerns related to bond and development. These concerns are also addressed in ACI 349-13 by the introduction of the 1.2 factor on development length above that which is ACI 318-08 development length equation. ACI report ACI ITG-6R-10, “Design Guide for the use of ASTM A1035/A1035M Grade 100 (690) Steel Bars for Structural Concrete”, 2010, Paul Zia Chair, addressed the issue of cracking associated with the use of 100 ksi rebar. While noting that some cracking issues may occur for the 100 ksi rebar in a commercial application, it is not likely it would occur for an 80 ksi rebar in the nuclear industry. This is because the largest loads in nuclear structures is typically earthquake for which cracking is not a significant issue. In any case, the concerns expressed regarding deflection control, ductility and overstrength are not particularly relevant to the design basis for most safety-related nuclear structures. We urge NCR to rethink its exception to the use of grade 75/80 reinforcement for flexure as allowed by ACI 349.
2.	ACI 349 Committee Officers	5	2.6	N	The reasoning for the one-third rule is provided in ACI 318-08 R10.5.3, where it is explained that the use of the minimum area of steel in a thick slab, wall or shell may result in severe congestion of reinforcing steel. ACI 349-13 uses the commentary in ACI 318-08 unless modified. We would request that NRC review the background provided in the 318 commentary as the minimum

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					reinforcing requirement affects the constructability of many nuclear structures. Please note that this provision has been in the code for many years and is not a recent addition.
3.	ACI 349 Committee Officers	5	3.1	N	When To is combined with Dead and Live in normal loading conditions, i.e. load combinations 9-1 thru 9-4, the concrete sections are most likely taken as uncracked and gross member properties are used. The cracked properties for the seismic load cases are used (ASCE 4 and 43) for the accident and abnormal loading conditions, including thermal loading. Combining operational temperature loading and uncracked member properties is conservative (since cracking relieves these stresses), and applying the 1.6 factor as requested by the NRC position increases the conservatism to a degree that we believe is unwarranted. The resulting effects of applying 1.6 to To is also undesirable because this results in more reinforcement required in the member which in turn increases the stiffness of the member and hence making it attract greater thermal stresses. This is counter-productive to the reinforced concrete member.
4.	ACI 349 Committee Officers	6	3.1	N	In as much as Ro is computed mainly from thermally-induced elongation of piping, it is not clear why this should be associated with enhanced uncertainty as stated in the NRC position. Note also that there is already significant conservatism associated with the use of an envelope of temperatures for these cases. Please note that the nuclear industry has long struggled with the difficulty of dealing with temperature loads on nuclear structures. The self-relieving nature of the temperature load makes it less critical than other loads. Adding larger load factors sends a wrong message to the designers that the way to deal with temperature is to make the structure stronger. This again is counter-productive to a rational design. Furthermore, the codes recognize the cumulative approach contained in ASCE 7, which holds that as an increasing number of loading types are combined, the less likely it is that the peaks of these loads will occur concurrently. Ro is consistently addressed in this regard in ASCE 43, ACI 349 and AISC N690.

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5.	ACI 349 Committee Officers	6	3.1	N	We disagree with the NRC position to require a load factor of 1.0 for live load. The load factors in Chapter 9 are associated with lower strength (phi) factors; the Appendix C load factors are used with higher strength (phi) factors. These factors cannot be mixed. Thus, increasing load factors in Chapter 9 to match those of Appendix C erroneously alters the global safety factor. We note also that ACI 318 allows live load reductions that result in an equivalent load factor of 0.5L. These reductions are not permitted in nuclear safety-related construction. We strongly recommend that the NRC review their position in this regard. Regarding the load factor in ACI 349-13 for live load, as explained in the commentary of ASCE 7 Section C2.3, the loads used in design account for the maximum lifetime value as well as arbitrary point-in-time values, with the maximum lifetime value always controlling. When many different types of loads are superimposed in a load combination, as is the case for abnormal or extreme load combinations, the arbitrary point-in-time value or the mean value of the load (accounting for industry variation) should be used. The mean value varies between 0.5 to 0.8 of the maximum lifetime value. The value of 0.8L is used for load combination 9-5 to 9-9 on this basis.
6.	ACI 349 Committee Officers	6	3.7	N	We strongly recommend that the NRC reconsiders their recommendation to use a phi factor of 0.6 for shear critical walls for both load combinations from Chapter 9 of ACI 349 as well as for load combinations from Appendix C of ACI 349. It is an error to use a phi of 0.6 for load combinations in Appendix C. ACI 318 requires 0.75 for shear critical walls using Appendix C load combinations and ACI 349 recommended 0.85. With regard to the deviation from ACI 318, the 349 Committee discussed this issue extensively. Most of the walls in nuclear safety-related structures are squat (low-aspect ratio) flanged walls. Some of these walls could be deemed shear critical. Shear critical walls are walls where the measured ultimate shear strength is less than the value associated with the wall's expected flexural strength. Establishing the flexural strength for squat walls of the type encountered in nuclear construction (considering non-aligned

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					openings) is best addressed experimentally. Numerical simulation is problematic since it involves predicting a three-dimensional yield line and there was general disagreement on how that is done. There was general agreement that the trigger for shear-critical walls as defined in ACI 318 is appropriate for ACI 318-type walls. These rectangular walls have a well-defined B stress region, are relatively short in wall length, are tall in wall height, and generally have aligned openings from top to bottom. In such cases, the wall can be approximated as a 2-D cantilever column that develops a hinge near the foundation and thus the moment Mp can be approximated as well as the associated value of Vp, whereby if Vu>Vp then the wall is deemed shear critical. Walls in nuclear safety-related construction do not generally lend themselves to this type of assessment. Walls often extend over the entire length of the structure because they are also used for shielding. These walls are thick, stocky walls with intersecting walls (flanged walls or barbells) with non-aligned utility openings, comprising exclusively D stress regions better solved using the strut and tie method (because the openings do not align) than the conventional bending model. Computation of a cantilever bending value for such walls would be wholly inaccurate, and thus this method of defining shear-critical walls is not available to the nuclear design community. There were two additional reasons for the deviation from ACI 318. The first is that the committee concluded that the ACI 318 design equation for shear, especially the ones in Chapter 11 of ACI 318-08 under-predicted the capacity of the ACI 349-type walls. Work by Gulec et al. (ACI Structural Journal, Vol 105, Issue 4, pgs. 488-497) showed that for ACI 318-compliant walls that are shear critical both the Chapter 21 and Chapter 11 equations result in median nominal capacities to measured capacities ratios of 1.08 and 0.86, respectively. Furthermore, it was noted that the standard deviation for shear capacity for the regular ACI 318-compliant squat rectangular walls and those deemed shear critical was approximately the same at 0.53 and 0.54 for the equations in Chapter 21 and 0.38 and 0.42 for the expressions provided in Chapter 11. Note that the shear critical wall data had approximately the

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					same scatter in this research for ACI 318-compliant rectangular walls. The design median strength under-predicts the measured strength by 35% for Chapter 11 equations and 20% for the expressions in Chapter 21. The use of a 0.75 phi factor shifts the 50^{th} percentile nominal strength to an approximately 90^{th} percentile design strength. ASCE 4 predicts seismic demands for DBE shaking at the 80^{th} percentile exceedance probability. ACI 349 design strength equations deliver capacities between 90^{th} and 98^{th} percentile exceedance probability. Taken together they ensure a performance target of 1% or less frequency of unacceptable performance under DBE shaking (Ref. ASCE 43). Secondly, ACI 318 introduced a phi factor of 0.6 not because of increased scatter associated with the measured shear strength of regular squat walls vs shear-critical walls. The penalty was introduced to offset reductions in seismic load associated with energy dissipation, wherein ductility and degraded shear strength are critical for the expected behavior. This is not the case with walls in nuclear safety-related construction. Nuclear safety-related structures are designed to be subjected to multiple cycles of loading to peak strength in safe shut down DBE shaking. The ductility demands for these walls are intentionally kept low. The structures that we build for these power plants are designed to behave essentially elastic under the design basis earthquake. The accompanying drift associated with the design basis earthquake is very small. The reduced ductility identified by the NRC is deemed acceptable since the ductility demand is indeed very low for nuclear safety-related structures. This is not the case for building structures designed in accordance with ACI 318. We thank you for your attention in this matter.
7.	ACI 349 Committee Officers	4	B	C	Why is NRC endorsing a portion of ACI 359-15 in DG -1283 when DG-1283 does not cover Reactors and Containment Structures?
8.					
9.					

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U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1283

Proposed Revision 3 to Regulatory Guide 1.142

Issue Date: April 2019
Technical Lead: M. Sircar

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.

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not receiv approval and does not represent a final NRC staff position. Public comments are being solicited on this DG and its associated regu should be accompanied by appropriate supporting data. Comments may be submitted through the Federal-rulemaking Web site, ht searching for draft regulatory guide DG-1283. Alternatively, comments may be submitted to the Rules, Announcements, and Dire Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date in *Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at https://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/. The DG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML16172A240. The regulatory analysis may be found in ADAMS under Accession No. ML16172A239.

  - 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, 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.

Related Guidance

- RG 1.28, “Quality Assurance Program Criteria (Design and Construction)” (Ref. 3), describes acceptable methods for establishing and implementing quality assurance (QA) programs to comply with the requirements of Appendix B to 10 CFR Part 50 during the design and construction phases of nuclear power plants and fuel reprocessing plants.

- 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), provides guidance for the design, construction, and testing of concrete radiation shields in nuclear power plants.

- RG 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments” (Ref. 6), provides guidance on containment design, construction and testing.

- RG 1.199, “Anchoring Components and Structural Supports in Concrete” (Ref. 7), 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. 8), describes considerations for aircrafts impacts for new nuclear power reactors.

- NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (Ref. 9), 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. 10), 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 agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. Regulatory guides 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.

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 of RG 1.142 (Revision 3) was updated to endorse, with certain exceptions, American Concrete Institute (ACI) 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary" (Ref. 11), 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." It also endorses a portion of ACI 359-15, "Code for Concrete Reactor Vessels and Containments (Ref. 12), related to mechanical splices.

Background

ACI 349-13 is based on ACI 318-08, "Building Code Requirements for Structural Concrete and Commentary" (Ref. 13), 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. Only those regulatory positions with supporting information are listed below.

Regulatory Position 1

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 1 provides conditions that complement the ultimate strength design (USD) approach in ACI 349-13.

Regulatory Position 2

2.1 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.

2.3 Use of high-strength (HS) reinforcement (Grade 75 and 80) as used in ACI 349-13 is not endorsed. 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.

2.4 Regulatory Position 2.4 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. 14) 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. 15) deformed reinforcement.

2.5 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 2.5 increases the area of shear reinforcement parallel to the flexural tension reinforcement to $0.0025b_w s_2$.

2.6 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.

2.7 Section 7.12.4 of ACI 349-13 does not specifically address the minimum reinforcement for a foundation base mat. Regulatory Position 2.7 specifies the minimum reinforcement requirement for a foundation base mat.

2.8 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 3

3.1 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 3.1.

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 3.1.

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 3.1, 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.

3.2 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. In addition, in cases where the criteria in ASCE 37 conflict with this RG, licensees and applicants should follow the positions in this RG.

3.6 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.

3.7 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, $\varphi = 0.60$ is used for shear critical members. Commentary section R9.3.1 of ACI 349-13 recommends using $\varphi = 0.75$ for such members but the justification provided for using higher φ value is not defensible. The NRC Staff position is to use $\varphi = 0.60$, which provides margin to account for uncertainties and variability in the shear capacity and the consequences of brittle failure. This discussion also applies to Appendix C, Section C.9.3 and RC.9.3.4 of ACI 349-13

3.8 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 4

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.

4.2.3 The technical background information for this regulatory position is identical to that for Regulatory Position 3.7.

Regulatory Position 5

This position alerts the applicant to the jurisdictional boundaries between ACI 349-13 and ACI 359-15 to capture structure-to-structure interaction effects. 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 359-15 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. 16)

- Safety Guide NS-G-1.6, "Seismic Design and Qualification for Nuclear Power Plants" (Ref. 17)

- Safety Guide NS-G-2.6, "Maintenance, Surveillance and In-Service Inspections in Nuclear Power Plants" (Ref. 18)

- Safety Guide NS-G-3.6, "Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants" (Ref. 19)

- Safety Report 70, Management System Standards "Comparison between IAEA GS-R-3 and ASME NQA-1-2008 and NQA-1a-2009 Addenda" (Ref. 20)

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 of RG 1.142 (Revision 3) endorses, with certain exceptions, ACI 349-13 and a portion of ACI 359-15 related to mechanical splices. They provide an adequate basis for complying with the NRC's regulations with regard to the analysis, design, construction, testing, and evaluation of safety-related concrete structures (other than reactor vessels and containment structures), subject to the positions listed below.

1. Regulatory Position 1

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 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.

2. Regulatory Position 2

2.1 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.

2.2 For performance criteria of mechanical splices, follow the additional splice qualification requirements in CC-4333 from ASME B&PV Code, Section III, Division 2 (ACI 359-15), with the exception that the strength requirement of an individual mechanical splice should be in accordance with Section 12.14.3.2 of ACI 349-13.

Test and evaluate the mechanical splices in accordance with ASTM A1034, "Standard Test Methods for Testing Mechanical Splices for Steel Reinforcing Bars" (Ref. 21).

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.

Conduct the slip test for two mechanical splice samples in accordance with Section 10.7 of ASTM A1034 to a predetermined load equal to one-half the specified yield strength ($0.5F_y$) of the steel reinforcing bar. Ensure that the measured slip does not exceed the values in Table 1 of this RG. The table was derived from California Department of Transportation “Authorization and Acceptance Criteria for Mechanical Couplers on ASTM A706 and ASTM A615 Reinforcing Steel” (Ref. 22), and “California Test 670 – Methods of Tests for Mechanical and Welded Reinforcing Steel Splices” (Ref. 23). Should either of the two mechanical splice samples not meet the slip acceptance criteria in Table 1, conduct a retest in which all remaining static tensile test specimens are evaluated for slip before static tensile testing and meet the slip requirements; if the specimens do not meet the slip requirements, the splices should be rejected.

Table 1: Total Slip Acceptance Criteria

Reinforcing Bar No.	Total Slip (inches)
#4 - #6	0.02
#7 - #9	0.028
#10 - #11	0.036
#14	0.048
#18	0.060

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

2.4 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.

2.5 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 $0.0025b_ws_2$..

2.6 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.

2.7 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.

2.8 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.

3. Regulatory Position 3

The staff endorses the load combinations in Section 9.2 of ACI 349-13, as modified in Regulatory Position 3.1 and outlined in Table 2 (modifications of load factors indicated in ***bold italic*** font).

3.1 **Table 2. 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)
$U = 1.2(D + F) + 1.6(L + H \mathbf{+} \boldsymbol{R_o} + \boldsymbol{T_o}) + 1.4C_{cr} + 0.5(L_r \text{ or } S \text{ or } R)$	(9-2) (NL)
$U = 1.2(D + F) + 0.8L + H \mathbf{+} \boldsymbol{R_o} + \boldsymbol{T_o} + 1.4C_{cr} + 1.6(L_r \text{ or } S \text{ or } R)$	(9-3) (NL)
$U = 1.2(D + F) + 1.6(L + H \mathbf{+} \boldsymbol{R_o} + E_o) \mathbf{+} \boldsymbol{1.4C_{cr}}$	(9-4) (SEL)
$U = 1.2(D + F) + 1.6(L + H \mathbf{+} \boldsymbol{R_o} + W)$	(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)
Legend	
NL = Normal Loading condition SEL = Severe Environmental Loading condition	
EEL = Extreme Environmental Loading condition AL = Abnormal Loading condition	

3.2 Construction loads need to 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.

3.3 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).

3.4 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.

3.5 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.

3.6 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.

3.7 The NRC staff does not endorse use of strength reduction factor $\varphi = 0.75$ in section 9.3.2.3 and R9.3.1of ACI 349-13 for shear critical members such as low-rise shear walls, portions of walls between openings or diaphragms. To address the lower ductility of shear critical members versus flexure-critical members, use $\varphi = 0.60$ for shear critical members.

3.8 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.

4. **Regulatory Position 4**

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

4.1 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.

4.2 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 3 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 3 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

4.2.1 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).

4.2.2 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.

4.2.3 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 3.7.

4.3 Appendix D, "Anchoring to Concrete"

RG 1.199 endorses Appendix D with exceptions.

4.4 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:

4.4.1 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, $(\mu) = 1.0$

b. For a localized area in the structure, $(\mu) = 3.0$

4.4.2 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

4.4.3 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).

4.4.4 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.

4.4.5 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 2.3 gives the NRC staff's position on this subject.

4.4.6 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.

4.4.7 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 2.8). The zone of influence induced by the concentrated loads may be determined, for example, by an analysis.

5. Regulatory Position 5

The applicant or licensee needs to carefully examine and establish the applicability of the standards in ACI 349-13 and ACI 359-15 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 359-15, 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 purpose of this section is to provide information on how applicants and licensees [1] may use this guide and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting" and any applicable finality provisions in 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

Use by Applicants and Licensees

Applicants and licensees may voluntarily [2] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations. Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged.

Licensees may use the information in this RG for actions that do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Licensees may use the information in this RG or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this RG. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the RG, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this RG, generic communication or promulgation of a rule requiring the use of this RG without further backfit consideration.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this RG, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this RG are part of the licensing basis of the facility. However, unless this RG is part of the license for a facility, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this RG constitutes a violation.

If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised RG and (2) the specific subject matter of this RG is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the

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1 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "applicants" refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

2 In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

guidance in this RG or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52.

Additionally, an existing applicant may be required to comply with new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this RG or requesting or requiring the licensee to implement the methods or processes in this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 26), and in NUREG-1409, "Backfitting Guidelines." (Ref. 27).

REFERENCES [3]

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), Regulatory Guided (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 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," Washington, DC.

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

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

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

10. American Society of Civil Engineers/Structural Engineering Institute, ASCE/SEI 37-14, "Design Loads on Structures during Construction," Reston, VA,[4] 2014.

11. American Concrete Institute (ACI) 349-13, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary," Farmington Hills, MI,[5] 2013.

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3 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.

4 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.

5 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

12. American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 2 (ACI 359), "Code for Concrete Containments Rules for Construction of Nuclear Facility Components," ASME, New York, NY, 2015.[6]

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

14. American Society for Testing and Materials (ASTM) A615, "Standard Specification for Deformed and Plain Carbon-Steel Bars for Concrete," West Conshohocken, PA, 2019.[7]

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

16. International Atomic Energy Agency (IAEA), Safety Guide NS-G-1.5, "External Events Excluding Earthquakes in the Design of Nuclear Power Plants," Vienna, Austria. [8]

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

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

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

20. 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.

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

22. California Department of Transportation (CALTRANS), "Authorization and Acceptance Criteria For Mechanical Couplers on ASTM A706 and ASTM A615 Reinforcing Steel,"[9] Sacramento, CA, 2017.

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

[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/.

[8] 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.

[9] Copies of California Department of Transportation (CALTRANS) reports can be obtained by writing to the Department at 1120 N Street, Sacramento, CA 95814, by telephone at 916-654-2852, or at http://www.caltrans.ca.gov/

23. CALTRANS, “California Test 670 - Methods of Tests for Mechanical and Welded Reinforcing Steel Splices,” Sacramento, CA, 2013.

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

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[10].

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

27. NRC, NUREG-1409, “Backfitting Guidelines,” Washington, DC.

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10 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/