U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-1284

Proposed Revision 1 to Regulatory Guide RG 1.199

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ANCHORING COMPONENTS AND STRUCTURAL SUPPORTS IN CONCRETE

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes a method acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for compliance with NRC regulations for the design, installation, testing, evaluation, and quality assurance (QA) of anchors (steel embedments) used for component and structural supports in concrete.

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 "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:

    - 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 to determine 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 without loss of capability to perform their safety functions, reflecting the appropriate

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This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, http://www.regulations.gov, by searching for draft regulatory guide DG-1284. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of this guide, 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 http://www.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. ML17258A579. The regulatory analysis may be found in ADAMS under Accession No. ML17258A580.

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 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 overall QA requirements for safety-related 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 methods the NRC staff considers acceptable for complying with the provisions of 10 CFR Part 50 and 10 CFR Part 52, which refer to 10 CFR Part 50, Appendix B, for establishing and implementing a QA program for the design and construction of nuclear power plants and fuel reprocessing plants.

- RG 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)” (Ref. 4), provides guidance for the analysis, design, construction, testing, and evaluation of safety related nuclear concrete structures, excluding concrete reactor vessels and concrete containments.

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 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), under control 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, 3150-0151, 3150-0009 and 3150-0132), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503; e- mail: oira_submission@omb.eop.gov.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

This revision of RG 1.199 (Revision 1) addresses changes in the codes and standards endorsed by Revision 0 of the guide. The revised codes and standards endorsed by Revision 1 include (1) American Concrete Institute (ACI) 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," Appendix D, "Anchoring to Concrete" (Ref. 5), (2) ACI 355.2-07, "Qualification of Post-Installed Mechanical Anchors in Concrete and Commentary" (Ref. 6), and (3) American Society for Testing and Materials (ASTM) E488/E488M-15, "Standard Test Methods for Strength of Anchors in Concrete Elements" (Ref. 7). Table 1 summarizes the changes to these codes and standards.

Table 1 Summary of Changes in Codes and Standards

ACI 349-13, Appendix D	Replaces	Appendix B to ACI 349-01 (Ref. 8)
ACI 355.2-07	Replaces	ACI 355.2-01 (Ref. 9)
ASTM E488/E488M-15	Replaces	ASTM E488-96 (Ref. 10)

Harmonization with International Standards

The NRC staff considered guidance from the International Atomic Energy Agency and International Organization for Standardization and did not identify any additional guidance applicable to this RG.

Documents Discussed in Staff Regulatory Guidance

This RG endorses 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 guide endorses the procedures and standards of ACI 349-13, Appendix D, ACI 355.2-07, and ASTM E488/E488M-15 for the design, installation, testing, evaluation, and QA of anchors (steel embedments) used for component and structural supports in concrete, subject to the exceptions and clarifications discussed in the regulatory guidance positions below.

The guidance is applicable to the types of anchors discussed in Sections D.1, "Definitions," and D.2, "Scope," of Appendix D to ACI 349-13. Adhesive anchors, deformed wire (bar) anchors and hooked bolts (J- or L-bolt) are not within the scope of this RG.

1. Application of Appendix D to ACI 349-13

1.1 Testing and Materials

1.1.1 In addition to the provisions in Section D.3.3 of Appendix D, the NRC staff considers the testing recommendations defined in ASTM E488/E488M-15 to be acceptable as a guide for establishing a testing program. Test methods not covered by ASTM E488/E488M-15 should be established using good engineering judgment. ACI 355.2-07 provides guidance acceptable to the NRC staff for determining whether post-installed mechanical anchors are acceptable. A representative number of anchors should be tested under the applicable environmental conditions for qualification purposes.

1.1.2 The concrete constituents, anchors, and embedded materials should be compatible with the anticipated environmental conditions to which they will be subjected during the life of the plant.

1.1.3 The effects of stress corrosion-cracking of high-strength anchor bolts (actual measured yield strength greater than or equal to 150 ksi) should be considered when selecting bolting material, lubricants, and installation specifications in order to maintain structural bolting integrity (Ref. 11).

1.2 Loads and Load Combinations

The loads and load combinations provided in Regulatory Positions 3 and 4 of RG 1.142 should be used. Loads and forces on embedments should be evaluated to account for baseplate flexibility and eccentricity of connections and the dynamic (strain rate and low-cycle fatigue) effects of loads and forces.

1.3 Design requirements

Design requirements should follow the provisions in Sections D.3–D.11 of ACI 349-13, with load factors and strength reduction factors consistent with the loads and load combinations in Regulatory Position C.1.2.

1.4 Installation

In addition to the provision in Section D.9 of ACI 349-13, "Installation of Anchors," the following should be considered in the installation of anchor bolts:

a. Installers should have the training and qualifications specified by the licensed design professional and anchor manufacturer.

b. Installation standards are consistent with accepted industry-specified tolerances.

c. Proper anchor type, grade, and material are used.

d. Embedment depth is proper.

e. Edge distance and spacing of anchors are to specified values.

f. Anchor is threaded properly.

g. Plate thickness meets specified size and thickness values.

h. Plate bolt-hole size is within established limits.

i. Correct bolt diameter and length are used.

j. Concrete has the specified full design strength.

k. Concrete is sound (free of voids).

l. Grout has been mixed and installed to specifications.

The following additional checks specifically for the installation of post-installed anchors should be considered:

a. Location of anchors avoids conflicts with existing embedments such as reinforcing steel.

b. Hole diameter in concrete is correct.

c. Drill hole angularity in concrete is within established limits.

d. Bolt hole has been cleared of drill dust.

e. Anchor has been correctly preloaded.

2. Inspection

Licensees and applicants should use their construction inspectors with the training and qualifications specified by the licensed design professional and anchor manufacturer. All anchors should be inspected to verify that they are of the specified size and type. Anchor systems that are external (that part or portion of the anchor that is not embedded in the concrete-visible part) to the concrete surface should be inspected to assure adequate performance during the life of the structure. In addition to the provisions in Section D.9.2 of Appendix D to ACI 349-13, the following 6-step inspection program should be used to verify the proper installation of post-installed anchors:

2.1 Are the nut and anchor bolt tight? This step will detect certain types of installation defects: oversized holes, total lack of preload, loose nuts, damaged subsurface concrete, and missing plug (for shell type). To implement this step, it is necessary to place a calibrated device on the bolt head or nut and to apply a torque. A well-installed bolt should not rotate under the torque applied equal to about 20 percent of the normal installation torque.

2.2 Are there washers between the equipment base and the anchor bolt nut or bolt head? All bolts should have washers. Oversize washers should be used for thin equipment bases. Lock washers should be used where even low-level vibration exists.

2.3 Is the bolt spacing in accordance with the anchorage design?

2.4 Is the distance between the bolt and any free concrete surface in accordance with the anchorage design (edge condition)?

2.5 Is the concrete sound and uncracked? This inspection element will detect gross defects in the concrete that could affect the holding power of expansion anchor bolts. Hairline shrinkage cracks in the vicinity of an expansion bolt are not a matter of concern as long as the design strength is based on cracked concrete. If cracks in the vicinity exceed about 0.01 inch (0.3 millimeter), the design strength should be appropriately reduced.

2.6 Is there a significant gap between the equipment base and the concrete surface? This inspection element will identify situations in which the equipment base is raised. This detail causes concern because shear forces result in flexural stresses in the anchor bolt. A gap of less than about 0.25 inch is not significant and should be ignored, except, for equipment that contains essential relays (a relay whose function is essential to plant safety in an earthquake), there should be no gap between the base of the equipment and the surface of the concrete at the bolt or anchor location. Anchorages with gaps larger than about 0.25 inch should be evaluated in more detail.

For maximum assurance, all six of these steps should be performed for all bolts. However, adequate assurance can be achieved by a less extensive inspection program. Inspection steps 2.2 through 2.6, which are simple and mainly visual, should be applied to each bolt together with a sampling approach for the tightness check in step 2.1. Appendix A to this guide provides a sampling program for expansion anchor bolts.

3. Quality Assurance

The methods for design and construction QA found in RG 1.28, “Quality Assurance Program Criteria,” should be used.

4. Anchorage to Masonry

Anchorage to masonry structures and components is outside the scope of ACI 349-13. Anchors discussed in this guide should not be used to attach seismic Category I components or systems to concrete block (masonry) walls that are seismically qualified, except for extremely low-load applications. When it is impossible to avoid their application, users should verify through appropriate means (e.g., pull test) that the supports are structurally acceptable.

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 which 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

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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.

acceptability of the licensee's request, then the staff may request that the licensee either follow the 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 the NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 12), and in NUREG-1409, "Backfitting Guidelines," (Ref. 13).

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

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

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

5. American Concrete Institute (ACI) 349-13, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary," Appendix D, "Anchoring to Concrete," Farmington Hills, MI, 2013.[4]

6. ACI 355.2-07, "Qualification of Post-Installed Mechanical Anchors in Concrete and Commentary," Farmington Hills, MI, 2007.

7. American Society for Testing and Materials, (ASTM) E488/E488M-15, "Standard Test Methods for Strength of Anchors in Concrete Elements," West Conshohocken, PA, 2015.[5]

8. ACI 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," Appendix B, "Anchoring to Concrete," Farmington Hills, MI, 2001.

9. ACI 355.2-01/ACI 355.2R-01, "Evaluating the Performance of Post-Installed Mechanical Anchors in Concrete" (ACI 355.2-01) and "Commentary" (ACI 355.2R-01), Farmington Hills, MI, 2001.

10. ASTM, E488-96, "Standard Test Methods for Strength of Anchors in Concrete and Masonry Elements," West Conshohocken, PA, 1996.

11. NUREG-2191, Vol. 2, "Generic Aging Lessons Learned for Subsequent License Renewal (GALL-SLR) Report," Washington, DC.

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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 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.

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/.

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

13. NRC, NUREG-1409, "Backfitting Guidelines," Washington, DC.

APPENDIX A

Sampling Program for Anchor Bolts

Perform inspection step 2.1 in Regulatory Position 2 on at least 25 percent of the bolts in every equipment anchorage. If the selected bolts do not pass the inspection, perform step 2.1 on all bolts in the anchorage.

OR

Perform inspection step 2.1 on a randomly selected statistical sample of bolts. The size of the sample and the number of nonconformances should be such that there is a 95-percent confidence of no more than 5 percent nonconforming bolts. This can be determined as follows:

$$R' = R + Z\left(\frac{R(1-R)}{n}\right)^{1/2}\left(\frac{N-n}{N-1}\right)^{1/2}$$

Where:

R' = Upper limit of the true defect rate at a specified confidence level (R' = 0.05 in this application)

R = Defect rate observed in sample

Z = Confidence coefficient for a normally distributed statistical model of test data (For a 95% confidence level, Z= 1.65)

n = Test sample size

N = Total population from which test sample was selected.

Table A-1 gives the allowable number of nonconforming bolts as a function of the population size *N* and the test sample size *n*[1].

When the failure rate for this check exceeds the limitations corresponding to 95-percent confidence of no more than 5 percent nonconforming bolts, the installation procedure should be considered to be unacceptable.

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1 This sampling program adopts the essential features of the sampling program in NRC IE Bulletin 79-02, “Pipe Support Base Plate Designs Using Concrete Expansion Bolts,” Revision 2, dated November 8, 1979 (Ref. A.1), which was adapted from Electric Power Research Institute (EPRI)-NP-5228, “Seismic Verification of Nuclear Power Plant Equipment Anchorage,” issued May 1987 (Ref. A.2).

Table A-1. Allowable Number of Nonconforming Anchors (R × n)

Total Population Size (N)	Test Sample Size (n)											
	40	**60**	**80**	**100**	**150**	**200**	**250**	**300**	**350**	**400**	**450**	**500**
100	1	2	3	5								
200	1	1	2	3	6	10						
300	1	1	2	3	5	7	10	15				
400	1	1	2	3	5	7	9	12	15	20		
500	1	1	2	3	5	7	9	12	14	17	20	25
600	1	1	2	3	5	7	9	11	14	16	19	22
700	1	1	2	3	4	7	9	11	13	16	18	21
800	1	1	2	3	4	6	9	11	13	16	18	21
900	1	1	2	3	4	6	8	11	13	15	18	20
1000	1	1	2	3	4	6	8	11	13	15	17	20

REFERENCES

A.1 NRC IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Bolts," Revision 2, Washington, DC, November 8, 1979. (ADAMS Accession No. ML080310536)

A.2 Electric Power Research Institute NP-5228, "Seismic Verification of Nuclear Plant Equipment Anchorage," Volumes 1 and 2, Palo Alto, California, May 1987.[6]

6 Copies of Electric Power Research Institute (EPRI) documents may be obtained by contacting the Electric Power Research Institute, 3420 Hillview Avenue, Palo Alto, CA 94304, telephone: (650) 855-2000 or through the EPRI Web site at http://my.epri.com/portal/server.pt.