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

A. INTRODUCTION

This regulatory guide has been revised to provide guidance to licensees and applicants on methods acceptable to the NRC staff for complying with the NRC's regulations in the design, evaluation, and quality assurance of safety-related nuclear concrete structures, excluding concrete reactor vessels and concrete containments.

General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. In addition, GDC 2, "Design Bases for Protection Against Natural Phenomena," and GDC 4, "Environmental and Dynamic Effects Design Bases," require, in part, that such SSCs be designed to withstand the effects of natural phenomena and to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation and postulated accidents. Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 establishes overall quality assurance for SSCs important to safety. Appendix

S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50, states, in part, requirements for the implementation of General Design Criterion 2 with respect to earthquakes.[1]

For concrete containments, Regulatory Guide 1.136, “Materials, Construction, and Testing of Concrete Containments,” and Section 3.8.1, “Concrete Containments,” of NUREG-0800, “Standard Review Plan,” provide guidance for complying with the NRC’s regulations. Licensees and applicants may propose means other than those specified by the provisions of the Regulatory Position of this guide for meeting applicable regulations. No new requirements are being imposed by this regulatory guide. Implementation of this guidance by licensees will be on a strictly voluntary basis.

The information collections contained in this regulatory guide are covered by the requirements of 10 CFR Part 50, which were approved by the Office of Management and Budget, approval number 3150-0011. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

B. DISCUSSION

ANSI/ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures," was prepared by Committee 349 of the American Concrete Institute (ACI) and was published by the ACI in February 1998. The code requirements are principally based on the 1995 edition of ACI 318, "Building Code Requirements for Structural Concrete," with modifications to accommodate the loadings and performance requirements specific to nuclear power plants.

Some sections of ACI 318-1995 contain criteria on certain aspects of concrete structural design that are more current than those of ACI 349-97. When this is the case, ACI 318-1999 is recommended for use. ACI 318 has long been the basis for the design of concrete buildings in the United States, and it has been used by the NRC staff initially in evaluating the adequacy of concrete structures in nuclear power plants.

ACI 349-97 contains requirements for the design and construction of safety-related concrete structures. This regulatory guide delineates the extent to which ACI 349-97, except for Appendix B, "Steel Embedments," is acceptable to the NRC staff. In a separate action, the staff intends to endorse Appendix B of ACI 349-97 in a regulatory guide that is being developed to address anchoring components and structural supports in concrete.

Because the ACI 349 provisions may be revised through supplements and Code revisions, the staff may update this guide periodically.

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[1] Appendix S to 10 CFR Part 50 applies to applicants for a design certification or combined license pursuant to 10 CFR Part 52 or a construction permit or operating license issued pursuant to 10 CFR Part 50 after January 10, 1997. However, for either an operating license applicant or holder whose construction permit was issued before January 10, 1997, the earthquake engineering criteria in Section VI of Appendix A to 10 CFR Part 100 continues to apply.

USE OF ACI 349-97 AND OTHER RELATED STANDARDS

On January 10, 1997, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” to 10 CFR Part 50 became effective. The specification of the safe shutdown earthquake (SSE) and operating basis earthquake (OBE) in Appendix S, with the load combinations in ACI 349-97, are considered in this revision of the regulatory guide. As noted in the Introduction to this guide, Appendix S to 10 CFR Part 50 is not applicable to existing plants.

The ACI 349-97 commentary recommends the use of various ANSI standards for developing quality assurance programs related to the design and construction of safety-related concrete structures. Quality assurance programs developed in accordance with the applicable portions of these ANSI standards and the regulatory guides that endorse them provide an acceptable means of complying with the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

ASME NQA-2-1983 provides standards, such as those for in-process testing of concrete, concrete materials, and reinforcing bar splices and for repairs and inspection of concrete construction, that are not adequately included in ACI 349-97. Hence, ASME NQA-2-1983 provides supplementary provisions for the construction of safety-related concrete structures.

Field testing requirements for reinforcing bar splices have changed and new systems of mechanical connections might require different types of in-process testing than that stated in ASME NQA-2-1983 for Cadweld splices. For such cases, it is advisable to keep informed of the actions of the ACI-ASME Joint Committee on ASME Section III, Division 2 (ACI 359), i.e., its code cases and code changes and those of the ACI 349 Committee and the extent to which they are acceptable to the NRC staff.

Such phrases as "unless approved by the Engineer" are frequently used in ACI 349-97. For example, Section 6.4 of ACI 349-97 states that “all construction joints shall be made according to the design drawings or shall be approved by the Engineer.” It is recommended that applicants carefully evaluate the possible interpretations of such discretion in each case. To the extent possible, the appropriate construction specifications should describe the criteria under which such discretion may be used by the Engineer.

DISCUSSION OF REGULATORY POSITIONS

This regulatory guide sets forth 15 Regulatory Positions on the use of ACI 349-97. The reasons for each of these regulatory positions are as follows.

Regulatory Position 1. Concrete structures within a containment could be designed in accordance with the standards of ACI 349-97. However, the pressure-resisting portion of the drywell of Mark III containments (e.g., General Electric boiling-water reactors (BWRs)), the dividing barrier of ice-condenser containments, and the dividing slab between the drywell and the wetwell of Mark II containments (e.g., General Electric BWRs) are required to maintain a certain degree of leaktightness during a loss-of-coolant accident in order to achieve the goals of Criterion 16, “Containment Design,” of Appendix A to 10 CFR Part 50. Therefore, the NRC staff reviews these structures on a case-by-case basis as indicated in the Standard Review Plan. To include these structures under ACI 349-97, the following additional provisions are to be added to ACI 349-97.

a. Provision for crack control under service loads, including test pressure load;

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 2. This position emphasizes the need to evaluate concrete structures for their effectiveness as radiation shields, when they are so intended. Some specific guidance for this purpose may be obtained from ANSI/ANS 6.4-1997. This is the current ANSI standard for radiation shielding.

Regulatory Position 3. In complex structural systems, the definitions of structural components such as walls, slabs, and foundations in ACI 349-97 may not be adequate for nuclear safety-related structures. This position alerts the designer to consider whether or not structural components are acting as parts of flexural frames, etc.

Since structural components generally have some flexure, they should be designed as a frame when the flexural moment from seismic loads is a large percentage of the flexural capacity. A ratio of two-thirds has been recommended such that flexure from seismic loads alone would be within the design capacity even under a seismic margin earthquake equal to 150% of the SSE.

Regulatory Position 4. Section 1.3.1 of ACI 349-97 enumerates work stages during which the concrete quality assurance (QA) inspector is required to ensure compliance with ACI 349-97. However, it lacks specific requirements for qualifying inspectors. This position provides an acceptable method of qualifying inspectors.

Regulatory Position 5. Regulatory Position 5 describes an acceptable alternative to the compressive strength test frequencies of ACI 349-97 and of ASME/NQA-2. The frequency for in-process compressive strength tests of concrete as recommended by ASME/NQA-2 is every 100 cu yd. ACI 349-97, on the other hand, requires the frequency of this testing to be every 150 cu yd or at least once a day. ACI 349-97 also provides a relaxation if some consistency in quality control is ensured in the moving averages of these tests based on the reduction in the standard deviation of the test data from the target strength. Public comments on the April 1978 issue of this Regulatory Guide 1.142 indicate that the cost and inconvenience in placement of concrete as a result of the more frequent testing (i.e., every 100 cu yd) considerably outweigh the benefit derived (i.e., more sampling data) from this requirement, particularly when the size of the members tends toward mass concrete placement. After a careful review, the NRC staff has recommended a position of gradual relaxation similar to that of ACI 349-97, maintaining, however, the starting frequency at every 100 cu yd. This position also recommends that the test frequency be increased as soon as the test data indicate a higher standard deviation than had been used in arriving at the decreased test frequency.

Regulatory Position 6. The load factors recommended by the NRC staff are to a large extent based on the design philosophy of ACI 318, with some amount of added conservatism. 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, etc., do not lend themselves to the same degree of confidence in assessing its effect as that for a dead load computation. The NRC staff thus has recommended a load factor of 1.2 for T_o in Regulatory Position 6.1. The load factor on T_o is reduced from 1.3 to 1.2 based on the statistical quantitative data gathered in NUREG/CR-3315, "A Consensus Estimation Study of Nuclear Power Plant Structural Loads" (May 1983).

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 has recommended a comparatively conservative load factor for Regulatory Position 6.2.

Regulatory Position 7. Loads that are due to malevolent vehicle assault, aircraft impact, and accidental explosion should be considered with the same load factors as tornado wind W_t. For a discussion of these loads see NUREG/CR-5733, "Re-evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.143" (August 1999).

Regulatory Position 8. Hydrodynamic loads associated with seismic loads are to be considered with the same load factors as the seismic load. Other hydrodynamic loads associated with accidents should be taken as Y_j, as stated in Regulatory Position 8.

Regulatory Position 9. At present, the loads that are due to pool dynamics and associated load combinations are reviewed by the NRC staff on a case-by-case basis. This position clarifies this practice.

Regulatory Positions 10 and 11. These positions endorse Appendix C of ACI 349-97 with certain clarifying exceptions that reflect the existing review practices of the NRC staff.

For specific cases such as a pressurized tunnel, limiting the ductility to the elastic range could be conservative.

In Regulatory Position 10.6, increase in the material strength (i.e., dynamic increase factor, DIF) could be realized only when the material is subjected to very high strain rates of loading, normally associated with impactive loadings. If a structure is found to be responding in a static or semi-static manner to a dynamic loading (i.e., dynamic load factor (DLF) <1.2), the materials of the structure would not undergo very high strain rate that would increase the material strength. Though there is no direct relationship between DLF and DIF, Regulatory Position 10 restricts the use of DIF when the DLF is lower than 1.2.

Regulatory Position 12. This position endorses Appendix A of ACI 349-97.

Regulatory Position 13. This position is intended as a guide until ACI 349-97 addresses composite or modular construction.

Regulatory Position 14. This position supplements the requirements for slabs and walls that frame into concrete containments and participate in resisting accident and seismic loads. The licensee should establish jurisdictional boundaries between the containment and other structures that frame into the containment.

Regulatory Position 15. This position invokes the state-of-the-art requirements in ACI 318-99 for members that are subject to torsion and combined shear and torsion.

C. REGULATORY POSITION

The procedures and requirements described in ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures" (except Appendix B), are generally acceptable to the NRC staff. They are considered to provide an adequate basis for complying with the NRC's regulations with regard to the design and construction of safety-related concrete structures other than reactor vessels and containments subject to the following:

1. Structures required to withstand pressures and to maintain a certain degree of leaktightness during operating and accident conditions will be reviewed in accordance with the provisions of Section 3.8.3 of the Standard Review Plan, NUREG-0800. To include these structures under ACI 349-97, the following additional provisions should be added to ACI 349-97.

a. Provision for crack control under service loads, including test pressure load;

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. When concrete structures are used to provide radiation shielding, provisions of ANSl/ANS 6.4-1997 (Appendix A) are applicable to the extent that they enhance the radiation shielding function of these structures. Reduction in shielding effectiveness from embedments, penetrations, and openings should be fully evaluated.

3. Where structural components, normally defined as walls, slabs, and foundations, actually 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 349-97, in addition to Chapters 13, 14, and 15 as appropriate.

The response of structural components should be considered as consistent with the response of structural frames when the flexural moment from seismic loads exceeds two-thirds of the design flexural capacity of the section in the absence of axial forces.

4. In addition to meeting the standards of Section 1.3.1 of ACI 349-97, the concrete QA inspectors should have sufficient experience in reinforced and prestressed concrete practice as applied to the construction of nuclear power plants. The inspectors should be thoroughly familiar with the applicable ACI and ASTM standards (e.g., ACI 311, "Guide for Concrete Inspection"). The examiners or inspectors who are qualified according to Appendix Vll of Section III, Division 2, of the ASME Boiler and Pressure Vessel Code (ACI 359) are acceptable as inspectors.

5. In lieu of the frequency of compressive strength testing specified by Section 5.6.1.1 of ACI 349-97 or that specified by ASME/NQA-2, the following is acceptable:

> Samples for strength tests of concrete should be taken at least once per day for each class of concrete placed or at least once for each 100 cu yd of concrete placed. When the standard deviation for 30 consecutive tests of a given class is less than 600 psi, the amount of concrete placed between tests may be increased by 50 cu yd for each 100 psi the standard deviation is below 600 psi, except that the minimum testing rate should not be less than one test for each shift when concrete is placed on more than one shift per day or not less than one test for each 200 cu yd of concrete placed. The test frequency should revert to once for each 100 cu yd placed if the data for any 30 consecutive tests indicate a higher standard deviation than the value controlling the decreased test frequency.

6. The load factors used in Section 9.2.1 of ACI 349-97 are acceptable to the staff except for the following:

6.1 In load combinations 9, 10, and 11, 1.2 T_o should be used in place of 1.05 T_o.

6.2 In load combination 6, 1.4Pa should be used in place of 1.25Pa.

7. Loads that are due to malevolent vehicle assault, aircraft impact, and accidental explosion should be taken as W_t in load combination 5.

8. 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 cases 4 and 8, and E_O in load cases 2, 7, and 10. All other hydrodynamic loads should be taken as Y_j, in load combinations 7 and 8.

9. The consideration of loads that are due to pool dynamics for the concrete structures in pressure-suppression containments will be evaluated on a case-by-case basis.

10. The local exceedance of section strengths in accordance with Appendix C of ACI 349-97 is acceptable in analyses for impactive or impulsive effects of Yr, Yj, and Ym in load combinations 7 and 8, load combinations of tornado-generated missiles, and loads described in Regulatory Position 7 in load combination 5 except for the following:

10.1 The deformation and degradation of the structure resulting from such an analysis must not cause loss of function of any safety-related structures, systems, or components.

10.2 The section strengths should be adequate to satisfy these load combinations without the impactive or impulsive effects.

10.3 In Section C.3.5 of ACI 349-97, the maximum permissible ductility ratios (μ) when a concrete structure is subjected to a pressure pulse caused by compartment pressurization or external explosion (blast) loading should be as follows.

10.3.1 For the structure as a whole μ =1.0 except as noted in 10.5.

10.3.2 For a localized area in the structure μ = 3.0.

10.4 In Section C.3.7 of ACI 349-97, where shear controls the design, the maximum permissible ductility ratios should be as follows.

10.4.1 When shear is carried by concrete alone, μ = 1.0.

10.4.2 When shear is carried by a combination of concrete and stirrups or bent bars, μ = 1.3.

10.5 In Section C.3.8 of ACI 349-97, the maximum permissible ductility ratio in flexure should be as follows:

10.5.1 When the compressive load is greater than $0.1f'_c A_g$ or one-third of that which would produce balanced conditions, whichever is smaller, the maximum permissible ductility ratio should be 1.0.

10.5.2 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 as given in C.3.3 or C.3.4 of ACI 349-97.

10.5.3 The permissible ductility ratio should vary linearly from 1.0 to that given in C.3.3 or C.3.4 of ACI 349-97 for conditions between those specified in 10.5.1 and 10.5.2.

10.6 In Section C.2.1 of ACI 349-97, the dynamic increase factor is to be considered as 1.0 for all materials when the dynamic load factor associated with the impactive or impulsive loading is less than 1.2.

11. The local exceedance of section strengths in accordance with Appendix C of ACI 349-97 is acceptable under the impactive and impulsive loadings associated with malevolent vehicle assault, aircraft impact, turbine missiles, and a localized pressure transient during an explosion, subject to the applicable exceptions of Regulatory Position 10.

12. The generic criteria of Appendix A, "Thermal Consideration," of ACI 349-97 are acceptable for the analysis of structures under loads T_o and T_a.

13. The design of composite members used in modular construction should conform to the intent of Code provisions of Chapter 10.14 and Chapter 17 of ACI 349 (i.e., the same rules used in computing the strength of regular reinforced concrete should apply). Until ACI 349-97 contains more specific requirements for modular construction, future designs will be evaluated on a case-by-case basis.

14. Slabs and walls that frame into concrete containments and will participate in resisting accident and seismic loads should meet the standards of ACI 349-97 or ACI 359 as appropriate.

15. Members that are subject to torsion and combined shear and torsion should be evaluated to the standards of Section 11.6 of ACI 318-99 instead of the requirements of Section 11.6 of ACI 349-97.

D. IMPLEMENTATION

The purpose of this section is to provide information to licensees and applicants regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the methods described in this guide reflecting public comments will be used in the evaluation of safety-related nuclear concrete structures, excluding concrete reactor vessels and concrete containments. Current licensees may, at their option, comply with the guidance in this regulatory guide.

REFERENCES

ACI 311.4 R-95, "Guide for Concrete Inspection," American Concrete Institute, 1995.[1]

ACI 318-[1989 (revised 1992, 1995, and 1999], "Building Code Requirements for Structural Concrete," American Concrete Institute.[1]

ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures" (and its supplements), American Concrete Institute, 1997.[1]

ACI 359, "Code for Concrete Reactor Vessels and Containments," Section III, Division 2, American Concrete Institute, 1999.[1] (Endorsed in Regulatory Guide 1.136, "Materials, Construction, and Testing of Concrete Containments.")

ANS 6.4-1997, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," American Nuclear Society, 1997.[2]

ASME NQA-1-1983, "Quality Assurance Requirements for Nuclear Facility Applications," American Society of Mechanical Engineers, 1983. (ASME NQA-1-1997 has not been endorsed by the NRC.)[3]

ASME NQA-2-1983, "Quality Assurance Requirements for Nuclear Power Plants," with ASME NQA-2a-1985, Addenda to ASME NQA-2-1983, American Society of Mechanical Engineers.[3]

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 3.8.1, "Concrete Containments," USNRC, ???NEED DATE OF SECTION[4]

NUREG/CR-3315, "A Consensus Estimation Study of Nuclrear Power Plant Structural Loads," USNRC, May 1983.[4]

NUREG/CR-5733, "Re-Evaluation of Regulatory Guidance Provided in Regulatory Guides 1.142 and 1.143," USNRC, August 1999.[4]

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[1] Copies may be obtained from the American Concrete Institute, P.O. Box 9094, Farmington Hills, Michigan 48333.

[2] Copies may be obtained from the American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60525.

[3] Copies may be obtained from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990.

[4] Copies are available at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202)512-1800); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161; telephone (703)487-4650; <http://www.ntis.gov/ordernow>. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 11555 Rockville Pike, Rockville, MD; the PDR's mailing address is USNRC PDR, Washington, DC 20555; telephone (301)415-4737 or (800)397-4209; fax (301)415-3548; email is PDR@NRC.GOV.

Regulatory Guide 1.136, "Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containments," USNRC, June 1981.[5]

[5] Single copies of regulatory guides may be obtained free of charge by writing the Reproduction and Distribution Services Section, OCIO, USNRC, Washington, DC 20555-0001, or by email to <DISTRIBUTION@NRC.GOV>, or by fax to (301)415-2289. Active guides may also be purchased from the National Technical Information Service on a standing order basis. Details on this service may be obtained by writing NTIS, 5285 Port Royal Road, Springfield, VA 22161; telephone (703)487-4650; online <http://www.ntis.gov/ordernow>. Copies of certain guides and many other NRC documents are available electronically on the internet at NRC's home page at <WWW.NRC.GOV> in the Reference Library. Documents are also available through the Electronic Reading Room (NRC's ADAMS document system, or PARS) at the same web site.

REGULATORY ANALYSIS

1. STATEMENT OF THE PROBLEM

In October 1981, Revision 1 of Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," was issued to describe acceptable methods for complying with the NRC's regulations with regard to design, evaluation, and quality assurance of safety-related concrete structures. This was accomplished by the conditional endorsement of the ACI 349-1976 edition of the American Concrete Institute's "Code Requirements for Nuclear Safety Related Concrete Structures" (for design and evaluation) and the American Nuclear Society (ANSI) Standard N45.2.5 (for quality assurance requirements). When Regulatory Guide 1.142 was issued, the NRC staff had planned to update the guide periodically because it anticipated continuing changes in the status of ACI 349-1976 code provisions through supplements and code revisions. The NRC has not updated Regulatory Guide 1.142 as planned.

Since the issuance of Revision 1 of Regulatory Guide 1.142, ACI 349-1976 has been revised and published as a new edition in 1980, 1985, 1990 (addenda and errata published), and 1997. Further, ANSI N45.2.5 is now essentially obsolete with quality assurance requirements for nuclear facility construction that are now found in ASME NQA-1. Therefore, the guidance in Regulatory Guide 1.142 has become outdated. Many nuclear power plant licensees have been using various editions of the ACI 349 code when making structural modifications to their plants. Also, the Electric Power Research Institute in its Utility Requirements Documents for Passive and Evolutionary (Advanced) Reactors, and the applicants who applied for design certification of standardized advanced reactors such as Westinghouse (AP600), General Electric (ABWR), and ABB/Combustion Engineering (System 80+), used the 1990 edition of the ACI 349 code. In addition, licensees who have applied for license renewal have referenced various editions of ACI 349 in their applications.

More importantly, the staff maintains that endorsement of the latest edition of ACI 349 and reference to current quality assurance standards will result in better evaluation of safety-related concrete structures (other than reactor vessels and containments) because the latest ACI 349 code is based on current industry practices and reflects the state of the art.

2. OBJECTIVE

The objective of the regulatory action is to update NRC guidance on the design, evaluation, and quality assurance of safety-related concrete structures (other than reactor vessels and containments).

3. ALTERNATIVES AND CONSEQUENCES OF PROPOSED ACTION

3.1 Alternative 1 - Do Not Revise Regulatory Guide 1.142

Under this alternative, Regulatory Guide 1.142 would not be revised and licensees would continue to rely on the current version of Regulatory Guide 1.142 with references from the mid-1970s. The staff acknowledges that many licensees currently involved in the construction and modification of safety-related concrete structures may, as a matter of practice, already rely on more recent editions of ACI 349 and ASME NQA-1. This alternative is considered the baseline, or no action alternative.

3.2 Alternative 2 (Update Regulatory Guide 1.142)

The staff has identified the following consequences associated with adopting Alternative 2.

3.2.1. Licensees would use the latest consensus standards available, thereby the design, evaluation, and quality assurance of safety-related concrete structures would be improved. The staff views the latest standards as improved because they incorporate the latest technology and knowledge on the subject.

3.2.2. Regulatory efficiency would be improved by reducing uncertainty as to what is acceptable and by encouraging consistency in the design, evaluation, and quality assurance of safety-related concrete structures. The benefits to the industry and the NRC will be to the extent this occurs. NRC reviews would be facilitated because licensee submittals should be more predictable and consistent analytically. Similarly, licensees' adherence to the latest consensus standards should benefit licensees by reducing the likelihood for follow-up questions and possible revisions to licensees' plans.

3.2.3. An updated Regulatory Guide 1.142 would result in cost savings to both the NRC and industry. From the NRC's perspective, relative to the baseline, NRC will incur one-time incremental costs to develop the regulatory guide for comment and to finalize the regulatory guide. However, the NRC should also realize cost savings associated with the review of licensee submittals. In the staff's view, the continuous and on-going cost savings associated with these reviews should more than off-set this one-time cost.

On balance, it is expected that industry would realize a net savings, as their one-time incremental cost to review and comment on a revised regulatory guide would be more than compensated for by the efficiencies (e.g., reduced follow-up questions and revisions) associated with each licensee submission.

3.2.4. The use of industry consensus standards that are already being used by licensees would enhance the continued use of the guidance contained in the ACI 349 code, thereby avoiding costs related to a "new" agency-prepared standard. This approach would also comply with the Commission's directive that standards developed by consensus bodies be utilized per Public Law

104-113, “National Technology and Transfer Act of 1995” (see NRC Management Directive 6.5, “NRC Participation in the Development and Use of Consensus Standards”).

4. CONCLUSION

Based on this regulatory analysis, it is recommended that the NRC revise Regulatory Guide 1.142. The staff concludes that the proposed action will reduce unnecessary burden on the part of both the NRC and its licensees, and it will result in an improved process for the design, evaluation, and quality assurance of safety-related concrete structures. Furthermore, the staff sees no adverse effects associated with revising Regulatory Guide 1.142.

BACKFIT ANALYSIS

The regulatory guide does not require a backfit analysis as described in 10 CFR 50.109(c) because it does not impose a new or amended provision in the NRC rules or a regulatory staff position interpreting the NRC rules that is either new or different from a previous applicable staff position. In addition, this regulatory guide does not require the modification or addition to systems, structures, components, or design of a facility or the procedures or organization required to design, construct, or operate a facility. Rather, a licensee or applicant can select a preferred method for achieving compliance with a license or the rules or the orders of the Commission as described in 10 CFR 50.109(a)(7). This regulatory guide provides an opportunity to use industry-developed standards, if that is a licensee’s or applicant’s preferred method.