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OFFICE OF NUCLEAR REGULATORY RESEARCH

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

A. INTRODUCTION

General Design Criteria 1, "Quality Standards and Records," 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Basis," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," require, in part, that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed and that they be designed to withstand the effects of postulated accidents and environmental conditions as well as those effects associated with normal operating conditions. This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to safety-related concrete structures (other than reactor vessels and containments) for nuclear power plants. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

ANSI/ACI 349-76,[1] "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 October 1976. The standard is based on the 1971 edition of ACI 318 (ANSI A89.1-1972), "Building Code Requirements for Reinforced Concrete," with modifications to accommodate the loadings and performance requirements peculiar to nuclear power plants. ACI 318 has long been the basis for the design of concrete buildings in the United States and has been used by the NRC staff as a starting point in evaluating the adequacy of concrete structures in nuclear power plants.

ACI 349-76 delineates requirements for the design and construction of safety-related concrete structures. This regulatory guide delineates the extent to which the Code and its 1979 supplement, except for Appendix B, "Steel Embedments," are acceptable to the NRC staff. The staff intends to endorse Appendix B of the Code in a regulatory guide being developed to address component support anchors.

Appendix A to this guide lists pertinent national standards, including those mentioned above.

Because of the continuing change in the status of the Code provisions through supplements and Code revisions, the staff plans to update this guide at periodic intervals.

Use of the Code and Other Related Standards

The Code 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.

ANSI N45.2.5 provides requirements 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 the Code. Hence, ANSI N45.2.5, as endorsed by Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants," provides supplementary provisions for the construction of safety-related concrete structures.

In-process testing requirements for reinforcing bar splices are undergoing changes and new systems of mechanical

*The substantial number of changes in this revision has made it impractical to indicate the changes with lines in the margin.

[1] In this guide, ACI 349-76 is referred to as "the Code."

connections might require different types of in-process testing than the one stated in ANSI N45.2.5 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 the extent to which they are acceptable to the NRC staff.

The Code contains frequent use of such phrases as "unless 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 specification should describe the criteria under which such discretion may be used by the Engineer.

Discussion of Regulatory Positions

Position 1. Concrete structures within a containment could be designed by the requirements of the Code. However, the pressure-resisting portion of the drywell of Mark III containments, the dividing barrier of ice-condenser containments, and the dividing slab between the drywell and the wetwell of Mark II containments are required to maintain a certain degree of leaktightness during a loss-of-coolant accident. Therefore, the NRC staff reviews these structures on a case-by-case basis as indicated in the Standard Review Plan. In order to include these structures under the Code, the following additional provisions are needed in the Code:

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.

c. Provisions for preoperational testing and inservice inspections.

Position 2. This portion emphasizes the need to evaluate concrete structures for their effectiveness as radiation shields, when so intended. Some specific guidance for the purpose may be obtained from ANSI N101.6-1972 (now withdrawn). It is expected that the Code in conjunction with ANSI/ANS 6.4 will pick up the applicable provisions of the withdrawn standard in an appropriate manner.

Position 3. All safety-related concrete structures are required to resist a specified magnitude of earthquake and its aftershocks. A concrete moment frame can be designed as an ordinary moment frame or a special moment frame depending upon its deformation characteristics under lateral loads. If it is designed to withstand the anticipated reversal of the earthquake loading by formation of hinges in the areas of maximum moments, it is required to be designed as a special moment frame, and the provisions of Appendix A of ACI 318 are applicable to such frames. Even when designed as an ordinary moment frame, where its theoretical deformations are restricted and the energy dissipation is assumed to occur through elastic deformations, it is a prudent approach to design and detail the frame to ensure that, under unanticipated and unaccounted for superposition of loading due to aftershocks, it is capable of functioning as a special moment frame.

Position 4. Section 1.3.1 of the Code enumerates work stages during which the inspector is required to ensure compliance with the Code. However, it lacks specific requirements for qualifying inspectors. This position provides an acceptable method of qualifying inspectors.

Position 5. Position 5 describes an acceptable alternative to the compressive strength test frequencies of the Code or of ANSI N45.2.5. The frequency for in-process compressive strength tests of concrete as recommended by ANSI N45.2.5 is every 100 cu yd. The Code, on the other hand, requires the frequency of this testing to be every 150 cu yd or at least once a day. The Code 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 guide 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 the Code, 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.

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 the 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 dead loads. The NRC staff thus has recommended in regulatory position 6.a. the same load factor (1.7) for T_o as that for a live load.

To a certain degree, the structural systems required to withstand pressures are related to the release of radioactivity to the atmosphere. 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 the comparatively conservative load factors of regulatory positions 6.b and 6.c.

The load factor on Operating Basis Earthquake (OBE) is conservatively derived from the stipulation of ACI 318 for earthquake. It is recognized that the OBE is not the same as the earthquake addressed in ACI 318. Further research is needed to justify the lower (i.e., required by the Code) load factor on OBE.

Position 7. A clarifying regulatory position is taken on the use of load F (as defined in Section 9.1.1.1 of the Code) for hydrostatic loadings due to ground water. The position states that, when the hydrostatic effects due to the ground water variations in the surrounding soil are considered, they should be considered as a part of H (lateral earth pressure) loading. This position is consistent with the intent of the ACI 318 Committee with regard to the use of F as expressed in a letter dated March 30, 1979, from the chairman of the committee to NRC.[2]

Position 8. The magnitude of differential settlement (as opposed to uniform settlement) is difficult to predict with any certainty. Moreover, the differential settlement of certain structures sets up permanent stresses in these structures. Hence, this position recommends consideration of differential settlement in all load combinations.

Position 9. At present, the loads 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.

Positions 10 & 11. These positions endorse Appendix C of the Code with certain clarifying exceptions. Exceptions are the existing review practices of the NRC staff.

Position 12. This position endorses Appendix A of the Code.

C. REGULATORY POSITION

The procedures and requirements described in ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures," and its 1979 Supplement (except Appendix B) are generally acceptable to the NRC staff. They are considered to provide an adequate basis for complying with the Commission'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 specific provisions of Standard Review Plan 3.8.3.[3]

2. When concrete structures are used to provide radiation shielding, provisions of ANSI/ANS 6.4-1977 (see Appendix A) are applicable to the extent that they enhance the radiation shielding function of these structures. Reduction in shielding effectiveness due to embedments, penetrations, and openings should be fully evaluated.

3. The Code lacks specific requirements to ensure the ductility of concrete moment frames. Adherence to the requirements of Appendix A to ANSI/ACI 318-77 is acceptable.

4. In addition to the requirements of Section 1.3.1 of the Code, the 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. The examiners/inspectors qualified according to Appendix VII 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 required by Section 4.3.1 of the Code or that required by ANSI N45.2.5 as endorsed by Regulatory Guide 1.94, the following is acceptable:

> Samples for strength tests of concrete should be taken at least once every shift 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 less than one test for each 200 cu yd of concrete placed. The test frequency should revert back to each 100 cu yd placed as soon as the test data of any 30 consecutive tests indicate a higher standard deviation than the value controlling the decreased test frequency.

6. The load factors used in Section 9.3.1 of the Code are acceptable to the staff except for the following:

a. In load combinations (9), (10), and (11), $1.3T_o$ should be used in place of $1.05T_o$.

b. In load combination (6), $1.5P_a$ should be used in place of $1.25P_a$.

c. In load combination (7), $1.25P_a$ and $1.25E_o$ should be used in place of $1.15P_a$ and $1.15E_o$, respectively.

d. In load combinations (2) and (10), $1.9E_o$ and $1.4E_o$ should be used in place of $1.7E_o$ and $1.3E_o$, respectively.

7. When the lateral and vertical pressures of liquids are due to the normal ground water variation in the soil surrounding the structure, the load factors of H loading of

[2] Attachment 2 to NRC memorandum dated April 14, 1979, from R.E. Lipinski to F. P. Schauer on the subject: "Proposed Change on SEB (Structural Engineering Branch) Position with Regard to Load Factor for Hydrostatic Pressure in Structural Design." Available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street NW., Washington, D.C.

[3] NUREG 75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition." Copies may be obtained from the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Sales Manager.

Section 9.3.1 should be applied to these forces or their related internal moments and forces.

8. In Section 9.3.2 the effects of differential settlement should be included in load combinations (1) through (11).

9. The consideration of loads 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 the Code is acceptable in analyses for impactive or impulsive effects of Y_r, Y_j, and Y_m in load combinations (7) and (8), and those of tornado-generated missiles in load combination (5) except for the following:

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

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

c. In Section C.3.4, the permissible ductility ratios (μ) when a concrete structure is subjected to a pressure pulse due to compartment pressurization or external explosion (blast) loading should be as follows:

(1) For the structure as a whole $\mu < 1.0$.

(2) For a localized area in the structure $\mu < 3.0$.

d. In Section C.3.7, where shear controls the design, the permissible ductility ratios should be as follows:

(1) When shear is carried by concrete alone, $\mu < 1.0$.

(2) When shear is carried by combination of concrete and stirrups or bent bars, $\mu < 1.3$.

11. The local exceedance of section strengths in accordance with Appendix C of the Code is also acceptable under the impactive and impulsive loadings associated with aircraft impact, turbine missiles, and a localized pressure transient during an explosion, subject to the applicable exceptions of regulatory position C.10.

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

D. IMPLEMENTATION

Except in those cases in which an applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described in this guide will be used in the evaluation of (1) all construction permit applications, (2) standard reference system preliminary design applications (PDA) or Type-2 final design applications (FDA-2), and (3) licenses to manufacture after November 1, 1981, except those portions of a construction permit application that:

1. Reference an approved standard reference system preliminary or final design (PDA or FDA), or applications for such approval.

2. Reference an approved standard duplicate plant preliminary or final design (PDDA or FDDA).

3. Reference parts of a base plant design qualified and approved for replication.

4. Reference a plant design approved or under review for approval to manufacture under a Manufacturing License, or applications for such approval.

The NRC staff does not intend to reevaluate existing plants or plants under construction on the basis of this guide. However, a licensee may at any time elect to use this or a later revision of this guide, following appropriate revision to applicable licensing commitments.

APPENDIX A

PERTINENT NATIONAL STANDARDS

The following is a list of national standards that are pertinent to safety-related concrete structures for nuclear power plants (other than reactor vessels and containments). If a standard is endorsed by a regulatory guide, this is indicated.

ANSI/ACI 349-76 and its supplements, "Code Requirements for Nuclear Safety Related Concrete Structures."[1]

ACI 318-71 (ANSI A89.1-1972), "Building Code Requirements for Reinforced Concrete."[1]

ANSI/ACI 318-77, "Building Code Requirements for Reinforced Concrete."[1]

ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.28)

ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage, and Handling of Items of Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.38)

ANSI N45.2.3-1973, "Housekeeping During the Construction Phase of Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.39)

ANSI N45.2.5-1974, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete, Structural Steel, Soils, and Foundations During the Construction Phase of Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.94)

[1] Copies may be obtained from the American Concrete Institute, P.O. Box 4754, Redford Station, Detroit, Michigan 48219.

[2] Copies may be obtained from the American Society of Mechanical Engineers, 345 East 47th Street, New York, N.Y. 10017.

ANSI N45.2.6-1978, "Qualification for Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.58)

ANSI N45.2.9-1974, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.88)

ANSI N45.2.11-1974, "Quality Assurance Requirements for Design of Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.64)

ANSI/ASME N45.2.12-1977, "Quality Assurance Program Auditing Requirements for Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.144)

ANSI N45.2.13-1976, "Supplementary Quality Assurance Requirements for Procurement Documents for Nuclear Power Plants."[2] (Endorsed by Regulatory Guide 1.123)

ANSI/ASME N45.2.23-1978, "Qualification of Quality Assurance Program Audit Personnel for Nuclear Facilities."[2] (Endorsed by Regulatory Guide 1.146)

ASME Section III, Division 2 (ACI 359), "Code for Concrete Reactor Vessels and Containments,"[1, 2] 1980.

ANSI/ANS 6.4-1977, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants."[3] The NRC staff intends to endorse this standard in a regulatory guide. A draft regulatory guide, EM 805-5, was issued for comment in February 1979.

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

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Docket No. 50-400-OL Official Exh. No. 24
In the matter of Carolina Power
Staff ____ IDENTIFIED ✓
Applicant ____ RECEIVED ✓
Intervenor ____ REJECTED 4
Cont'g Off'r ____
Contractor ____ DATE 10-30-84
Other ____ Witness C. Stokes
Reporter G. Walsh