[7590-01-P]

NUCLEAR REGULATORY COMMISSION

10 CFR PART 50

RIN 3150-AC93

# Codes and Standards for Nuclear Power Plants; Subsection IWE and Subsection IWL

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) proposes to amend its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants," of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) with specified modifications and a limitation. Subsection IWE of the ASME Code provides rules for inservice inspection, repair, and replacement of Class MC pressure retaining components and their integral attachments and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments in light-water cooled power plants. Subsection IWL of the ASME Code provides rules for inservice inspection and repair of the reinforced concrete and the post-tensioning systems of Class CC components. Licensees would be required to incorporate Subsection IWE and Subsection IWL into their routine inservice inspection (ISI) program. Licensees would also

be required to expedite implementation of the containment examinations and complete the expedited examination in accordance with Subsection IWE and Subsection IWL within 5 years of the effective date of this rule. Provisions have been proposed that would prevent unnecessary duplication of examinations between the expedited examination and the routine 120-month ISI examinations. Subsection IWE and Subsection IWL have not been previously incorporated by reference into the NRC regulations. This proposed amendment would specify requirements to assure that the critical areas of containments are routinely inspected to detect defects that could compromise a containment's pressure-retaining integrity.

DATES: Comment period expires (75 days after publication in the Federal Register). Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Written comments or suggestions may be submitted to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch. Deliver comments to: 11555 Rockville Pike, Rockville, MD between 7:45 am and 4:15 pm Federal workdays. Copies of the regulatory analysis, the environmental assessment and finding of no significant impact, the supporting statement submitted to the Office of Management and Budget, and comments received may be examined in the Commission's Public Document Room at 2120 L Street NW. (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Mr. W. E. Norris, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3805, or Mr. H. L. Graves, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3813.

SUPPLEMENTARY INFORMATION:

## Background

The NRC is taking the proposed action for the purpose of ensuring that containments continue to maintain or exceed minimum accepted design wall thicknesses and prestressing forces as provided for in industry standards used to design containments (e.g., Section III and Section VIII of the ASME Code, and the American Concrete Institute Standard ACI-318), as reflected in license conditions, technical specifications, and licensee commitments (e.g., the Final Safety Analysis Report). The NRC also believes enhanced ISI examinations are needed and are justified to supplement existing requirements specified in General Design Criterion (GDC) 16, and GDC 53, Appendix A to 10 CFR Part 50, and Appendix J to 10 CFR Part 50. Appendix J requires a general visual inspection of the containment but does not provide specific guidance on how to perform the necessary containment examinations. This has resulted in a large variation with regard to the performance and the effectiveness of containment inspections. In view of the increasing rate of occurrences of degradation in containments and variability of present containment examinations, the NRC has determined that it is necessary to include more detailed requirements for the periodic examination of containment structures in the regulations to assure that the critical areas of containments are periodically inspected to detect defects that could compromise the containment's pressure-retaining and leak-tight capability. Recent changes and additions to the ASME Code include provisions to address the concerns outlined above. The NRC proposes to make these provisions mandatory by amending 10 CFR 50.55a to incorporate by reference these additional portions of the ASME Code (Subsection IWE and Subsection IWL). Subsection IWE and Subsection IWL have not been previously incorporated by reference into the NRC's regulations.

The rate of occurrence of corrosion and degradation of containments has been increasing at operating nuclear power plants. Since 1986, twenty-one (21) instances of corrosion in steel containments have been reported. In two cases, thickness measurements of the walls revealed areas where the wall thickness was at or below the minimum design thickness. Since the early 1970s, thirty-one (31) incidents of containment degradation related to post-tensioning systems of concrete containments have been reported. Four recent additional incidents which involved grease leakage from tendons have been investigated. In addition to grease leakage, these incidents showed signs of leaching of the concrete.

Over one-third of the operating containments have experienced corrosion or other degradation. Almost one-half of these occurrences were found by the NRC through its inspections or audits of plant structures, or by licensees because they were alerted to a degraded condition at another site. Examples of degradation not found by licensees, but initially detected at plants through NRC inspections include: steel containment shell corrosion in the drywell sand cushion region (wall thickness reduced to below minimum design thickness); steel containment shell torus corrosion (wall thickness at or near minimum design thickness); grease leakage from the tendons of prestressed concrete containments, and water seepage, as well as concrete cracking in concrete containments.

There are several GDC criteria and ASME Code sections which establish minimum requirements for the design, fabrication, construction, testing, and performance of structures, systems, and components important to safety in water-cooled nuclear power plants. Criterion 16, "Containment design," requires the provision of reactor containment and associated systems to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as required for postulated accident conditions. Section III and Section VIII of the ASME Code, and the American Concrete Institute provide design specifications for minimum wall thicknesses and prestressing forces of containments, and these are reflected in license conditions, technical specifications, and licensee commitments for the operating plants.

Criterion 53, "Provisions for containment testing and inspection," requires that the reactor containment design permit: (1) appropriate periodic inspection of all important areas, such as penetrations; (2) an appropriate surveillance program; and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows. Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," of 10 CFR Part 50 contains specific rules for leakage testing of containments. Paragraph V. A. of Appendix J requires that a general inspection of the accessible interior and exterior surfaces of the containment structures and components be performed prior to any Type A test to uncover any evidence of structural deterioration that may affect either the containment structural integrity or leak-tightness. (Type A test means tests intended to measure the primary reactor containment overall integrated leakage rate: (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter). None of these existing requirements, however, provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation in licensee containment examination programs, such that there have been cases of noncompliance with GDC 16. Based on the results of inspections and audits, as well as plant operational experiences, it is clear that many licensee containment examination programs have not detected degradation that could ultimately result in a compromise to the pressure-retaining capability. Some containment structures have also been found to have undergone a significant level of degradation that was not detected by these programs.

The NRC believes that more specific ISI requirements, which expand upon existing requirements for the examination of containment structures in accordance with GDC 53 and Appendix J , are needed and are justified for the purpose of ensuring that containments continue to maintain minimum design wall thicknesses and prestressing forces as provided for in industry standards used to design containments (e.g., Section III and Section VIII of the ASME Code, and the American Concrete Institute Standard ACI-318), as reflected in license conditions, technical specifications, and written licensee commitments (e.g., the Final Safety Analysis Report). There exists a serious concern, based on actual operating experience, regarding continued compliance by the operating plants with existing requirements for ensuring containment minimum design wall thicknesses and prestressing forces if the proposed action is not taken. The NRC also believes that the occurrences of corrosion and other degradation discussed above would have been detected by licensees implementing the comprehensive periodic examinations set forth in Subsection IWE and Subsection IWL of the ASME Code proposed for incorporation by reference into 10 CFR 50.55a.

The Nuclear Management and Resources Council (NUMARC) has developed a number of industry reports to address license renewal issues. Two of them, one for PWR containments and the other for BWR containments, were developed for the purpose of managing age-related degradation of containments on a generic basis. The NUMARC plan for containments relies on the examinations contained in Subsection IWE and Subsection IWL to manage age-related degradation, and this plan assumes that these examinations are "in current and effective use." In the BWR Containment Industry Report, NUMARC concluded that "On account of these available and established methods and techniques to adequately manage potential degradation due to general corrosion of freestanding metal containments, no additional measures need to be developed and, as such, general corrosion is not a license renewal concern if the containment minimum wall thickness is maintained and verified." Similarly, in the PWR Containment Industry Report, NUMARC concluded that potentially significant degradation of concrete surfaces, the post-tensioning system, and the liners of concrete containments could be managed effectively if periodically examined in accordance with the requirements contained in Subsection IWE and Subsection IWL.

The five modifications, which are contained in one paragraph of the proposed rule, address two concerns of the NRC. The first concern is that certain recommendations for tendon examinations that are included in Regulatory Guide 1.35, Rev. 3, are not addressed in Subsection IWL (this involves four of the modifications, (ix)(A)-(D)). The ASME Code has considered these four issues and has adopted them in Subsection IWL. These issues will be published in future addenda. The second concern is that if there is visible evidence of degradation of the concrete (e.g., leaching, surface cracking) there may also be degradation of inaccessible areas. This fifth modification ((ix)(E)) contains a provision which would require an evaluation of inaccessible areas when visible conditions exist that could result in degradation of these areas.

The limitation specifies the 1992 Edition with 1992 Addenda of Subsection IWE and Subsection IWL as the earliest version of the ASME Code the NRC finds acceptable. This edition and addenda combination incorporates the concept of base metal examinations and would provide a comprehensive set of rules for the examination of post-tensioning systems. As originally published, Subsection IWE preservice examination and inservice examination rules focused on the examination of welds. This weld-based examination philosophy was established in the 1970s as plants were being constructed. It was based on the premise that the welds in pressure vessels and piping were the areas of greatest concern. As containments have aged, degradation of base metal, rather than welds, has been found to be the issue of concern. The 1991 Addenda to the 1989 Edition, the 1992 Edition and the 1992 Addenda to Section XI, Subsection IWE, all have furthered the incorporation of base metal examinations.

The proposed rulemaking incorporates a provision for an expedited examination schedule. This expedited examination schedule is necessary to prevent a delay in the implementation of Subsection IWE and Subsection IWL

(Table 4 of Enclosure 2 lists each plant and the delay in implementation which would be encountered without an expedited implementation schedule). Provisions have been incorporated in the proposed rule so that the expedited examination which would be required 5 years after the effective date of the rule and the routine 120-month examinations are not duplicated.

The NRC has reviewed the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL of Section XI of the ASME Code and has found that with the specified modifications these subsections of Section XI address current experience and provide a sound basis for ensuring the structural integrity of containments. NRC endorsement of Subsection IWE and Subsection IWL in its regulations would provide a method of improving containment examination practices by incorporating rules into the regulatory process that have received industry participation in their development and acceptance by the NRC.

Existing § 50.55a(g), "Inservice inspection requirements," specifies the requirements for preservice and inservice examinations for Class 1 (Class 1 refers to components of the reactor coolant pressure boundary), Class 2 (Class 2 quality standards are applied to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are part of the reactor coolant pressure boundary (e.g., systems designed for residual heat removal and emergency core cooling)), and Class 3 (Class 3 quality standards are applied to radioactive-waste-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves (not part of the reactor coolant pressure boundary)) components and their supports. Neither Subsection IWE (Class MC -- metal containments) nor Subsection IWL (Class CC -

- concrete containments) is presently incorporated by reference into the NRC regulations.

Proposed § 50.55a(g)(4) specifies the containment components to which the ASME Code Class MC and Class CC inservice inspection classifications incorporated by reference in this proposed rule would apply.

Proposed § 50.55a (g)(4)(v)(A), (v)(B), and (v)(C) specify Subsection IWE and Subsection IWL rules for repairs and replacements of metal and concrete containments. This is consistent with the long-standing intent and ongoing application by NRC and licensees to utilize the rules of Section XI when performing repairs and replacements of applicable components and their supports.

Proposed § 50.55a(b)(2)(vi) would incorporate a limitation specifying the 1992 Edition with 1992 Addenda of Subsection IWE and Subsection IWL as the earliest ASME Code version the NRC finds acceptable. This edition and addenda combination incorporates the concept of base metal examinations and provides a comprehensive set of rules for the examination of post-tensioning systems.

Proposed § 50.55a(b)(2)(ix) would specify five modifications that must be implemented when using Subsection IWL. Four of these issues are identified in Regulatory Guide 1.35, Revision 3, but are not currently addressed in Subsection IWL.

Proposed § 50.55a(g)(4)(v) requires that licensees incorporate containment examinations as part of their routine 120-month inspection program. It is recognized that when this rule becomes effective, plants within 2 years of the end of the 120-month interval may have difficulty developing and completing the containment examination program in a timely manner. Therefore, proposed § 50.55a (b)(2)(x) specifies that licensees with less than 2 years remaining in their present ISI interval may complete the Subsection IWE and the Subsection IWL portions of their ISI update within 2 years from the end of the present ISI interval. This is intended to provide licensees with sufficient time to develop the initial ISI plan and to facilitate maintenance of one ISI plan instead of two separate plans (i.e, the current Section XI ISI plan, and the Subsection IWE and Subsection IWL plan). In order to further reduce the burden on licensees and NRC staff, the Subsection IWE and Subsection IWL portions of the ISI plan will not have to be submitted to the NRC for approval. Licensees may simply retain their initial Subsection IWE and Subsection IWL plans at the site for audit.

Proposed § 50.55a(g)(6)(ii)(B)(1) would require that licensees conduct the first containment examinations in accordance with Subsection IWE and Subsection IWL (1992 Edition with the 1992 Addenda), modified by proposed § 50.55a(b)(2)(ix) within 5 years of the effective date of the final rule. This expedited examination schedule is necessary to prevent possible delays in the implementation of Subsection IWE by as much as 20 years and Subsection IWL by as much as 15 years. Subsection IWE, Table IWE-2500-1, permits the deferral of most of the required examinations until the end of the 10-year inspection interval. Adding the ten years that could pass before some utilities are required to update their ISI plans, a period of 20 years could pass before the first examinations would take place. Subsection IWL is based on a 5-year inspection interval. Adding the possible 10 years before update of existing ISI plans, a period of 15 years could pass before the examinations were performed by plants that have not voluntarily adopted the provisions of

Regulatory Guide 1.35, Rev. 3. Expediting implementation of the containment examinations is considered necessary because of the problems that have been identified at various plants, the need to establish expeditiously a baseline for each facility, and the need to identify any existing degradation.

Proposed paragraphs (g)(6)(ii)(B)(2) and (g)(6)(ii)(B)(3) would each provide a mechanism for licensees to satisfy the requirements of the routine containment examinations and the expedited examination without duplication. Paragraph (g)(6)(ii)(B)(2) would permit licensees to avoid duplicating examinations required by both the periodic routine and expedited examination programs. This provision is intended to be useful to those licensees that would be required to implement the expedited examination during the first periodic interval that routine containment examinations are required. Paragraph (g)(6)(ii)(B)(3) would allow licensees to use a recently performed examination of the post-tensioning system to satisfy the requirements for the expedited examination of the containment post-tensioning system. This situation would occur for licensees who perform an examination of the post-tensioning system using Regulatory Guide 1.35 between the effective date of this rule and the beginning of the expedited examination.

## Submission of Comments in Electronic Format

The comment evaluation process will be improved if each comment is identified with document title, section heading, and paragraph number addressed. In addition to the original paper copy, submitters are encouraged to provide a copy of their letter in an electronic format on IBM PC compatible

3.5- or 5.25-inch diskettes. Data files should be provided as WordPerfect documents. ASCII text is also acceptable or, if formatted text is required, data files should be provided in IBM Revisable-Form Text/Document Content Architecture (RFT/DCA) format. The format and version should be identified on the diskette's external label.

## Finding of No Significant Environmental Impact

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, if adopted, would not be a major Federal action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required.

This proposed rule is one part of a regulatory framework directed to ensuring containment integrity. Therefore, in the general sense, the proposed rule would have a positive impact on the environment. The proposed rule would incorporate by reference in the NRC regulations requirements contained in the ASME Code for the inservice inspection of the containments of nuclear power plants. Actions required of applicants and licensees to implement the proposed rule are of a routine nature that should not increase the potential for a negative environmental impact.

The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from Mr. W. E. Norris, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301)492-3805, or Mr. H. L. Graves, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301)492-3813.

## Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The public reporting burden for this collection of information is estimated to average 4,000 hours per response for development of an initial inservice inspection plan and 10,000 hours per response for the update of the plan and periodic examinations, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, DC 20503.

## Documented Evaluation

The Commission has prepared a draft summary of documented evaluation on this proposed regulation. The draft evaluation is available for inspection in the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis may be obtained from Mr. W. E. Norris, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301)492-3805, or from Mr. H. L. Graves, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301)492-3813.

The Commission requests public comment on the draft summary of documented evaluation. Comments on the draft evaluation may be submitted to the NRC as indicated under the ADDRESSES heading.

## Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this proposed rule does not fall within the purview of the Act.

## Backfit Statement

The NRC is taking the proposed action for the purpose of ensuring that containment structures continue to maintain or exceed minimum accepted design wall thicknesses and prestressing forces as provided for in industry standards used to design containment structures, as reflected in license conditions technical specifications, and licensee commitments. Therefore, under 10 CFR 50.109(a)(4)(i) a backfit analysis need not be prepared for this rule. A summary of the documented evaluation required by § 50.109(a)(4) to support this conclusion is set forth below.

GDC 16, "Containment design," requires the provision of reactor containment and associated systems to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as required for postulated accident conditions.

Criterion 53, "Provisions for containment testing and inspection," requires that the reactor containment design permit: (1) appropriate periodic inspection of all important areas, such as penetrations; (2) an appropriate surveillance program; and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows. Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," of 10 CFR Part 50 contains specific rules for leakage testing of containments. Paragraph V. A. of Appendix J requires that a general inspection of the accessible interior and exterior surfaces of the containment structures and components be performed prior to any Type A test to uncover any evidence of structural deterioration that may affect either the containment structural integrity or leak-tightness (Type A test means tests intended to measure the primary reactor containment overall integrated leakage rate: (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter). None of these existing requirements, however, provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation in licensee containment examination programs, such that there have been cases of noncompliance with GDC 16. Based on the results of inspections and audits, and plant operational experiences, it is clear that many licensee containment examination programs have not detected degradation that could result in a compromise of pressure-retaining capability. The location and extent of corrosion or degradation in a containment can be critical to the containment's behavior during an accident.

The metal containment structure of operating nuclear power plants were designed in accordance with either Section III, Subsection NE, "Class MC Components," or Section VIII, of the ASME Code. These subsections contain provisions for the design and construction of metal containment structures, including methods for determining the minimum required wall thicknesses. The minimum wall thickness is determined so that the metal containment structure will continue to maintain its structural integrity under the various stressors and degradation mechanisms which act on it.

The American Concrete Institute Standard ACI-318 contains provisions for designing and constructing the post-tensioning systems of concrete containment structures, including methods for determining the prestressing forces. The post-tensioning system is designed so that the concrete containment structure will continue to maintain its structural integrity under the various stressors and degradation mechanisms which act on it.

These requirements for minimum design wall thicknesses and prestressing forces as provided in these industry standards used to design containment structures are reflected in license conditions, technical specifications, and licensee commitments (e.g., the Final Safety Analysis Report).

The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. Over one-third of operating containment structures have experienced corrosion or other degradation. Almost one-half of the occurrences were first identified by the NRC through its inspections or structural audits, or by licensees because they were alerted to a degraded condition at another site. Examples of degradation not found by licensees, but initially detected at plants through NRC inspections include 1) corrosion of steel containment shells in the drywell sand cushion region, resulting in wall thickness reduced to below the minimum design thickness; 2) corrosion of the torus of the steel containment shell (wall thickness at or near minimum design thickness); 3) grease leakage from the tendons of prestressed concrete containments; and 4) water seepage, as well as concrete cracking in concrete containments.

The NRC believes that more specific ISI requirements, that expand upon existing requirements for the examination of containment structures in accordance with GDC 53, and Appendix J are needed and are justified to ensure that containment structures continue to maintain or exceed minimum accepted design wall thicknesses and prestressing forces as reflected in license conditions, technical specifications, or licensee commitments. Based on actual operating experience, a serious concern exists regarding continued compliance by the operating plants with existing requirements for ensuring containment minimum design wall thicknesses and prestressing forces if the proposed action is not taken. The NRC also believes that the occurrences of corrosion and other degradation discussed above would have been detected by licensees when conducting the comprehensive periodic examinations set forth in Subsection IWE and Subsection IWL of the ASME Code, as proposed for incorporation by reference into 10 CFR 50.55a.

Recent changes and additions to the ASME Code include provisions to address the concerns outlined above; and the staff proposes to make these provisions mandatory by amending 10 CFR 50.55a to incorporate by reference these additional portions of the ASME Code (Subsection IWE and Subsection IWL). The Commission concludes that this proposed backfit is necessary to ensure compliance with GDCs 16 and 53, Appendix J, minimum design wall thicknesses in metal containments, and the prestressing forces of concrete containments, which are applicable to all licensees through license conditions, technical specifications, and licensee commitments.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Criminal Penalties, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 533, the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd) and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.55a is amended by adding paragraphs (b)(2)(vi), (b)(2)(ix), (b)(2)(x), (g)(4)(v), and (g)(6)(ii)(B), and revising the introductory text of paragraph (g)(4) to read as follows:

§ 50.55a Codes and standards.

* * * * *

(b) * * *

(2) * * *

(vi) Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI. When using Subsection IWE and Subsection IWL, the 1992 Edition with the 1992 Addenda is the only acceptable Edition and Addenda.

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(ix) Examination of concrete containments.

(A) All grease caps that are accessible must be visually examined to detect grease leakage or grease cap deformations. Grease caps must be removed for this examination when there is evidence of grease cap deformation that indicates deterioration of anchorage hardware.

(B) An Engineering Evaluation Report must be prepared as prescribed in IWL-3300(a), (b), (c), and (d) when evaluation of consecutive surveillances of prestressing forces for the same tendon or tendons in a group indicates a trend of prestress loss such that the tendon force(s) would be less than the minimum design prestress requirements before the next inspection interval.

(C) When the elongation corresponding to a specific load (adjusted for effective wires or strands) during retensioning of tendons differs by more than 10 percent from that recorded during the last measurement, an evaluation must be performed to determine whether the difference is related to wire failures or slip of wires in anchorages. A difference of more than 10 percent must be identified in the ISI Summary Report.

(D) The licensee shall identify the following conditions, if they occur, in the ISI Summary Report:

(1) The sampled sheathing filler grease contains chemically combined water exceeding 10 percent by weight or the presence of free water;

(2) The absolute difference between the amount removed and the amount replaced may not exceed 10 percent of the tendon net duct volume.

(3) Grease leakage is detected during general visual examination of the containment surface.

(E) The licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report:

(1) A description of the type and estimated extent of degradation, and the conditions that led to the degradation;

(2) An evaluation of each area, and the result of the evaluation, and;

(3) A description of necessary corrective actions.

(x) Subsection IWE and Subsection IWL inservice inspection plans. Licensees that have less than 2 years remaining in their present 120-month inservice inspection interval on (insert effective date of the final rule) may defer completion of the Subsection IWE and Subsection IWL portions of the inspection plan for the next 120-month inspection interval for up to 2 years from the end of the present interval.

*     *     *     *     *

(g) * * *

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components. Components which are classified as Class MC pressure retaining components and their integral attachments, and components which are classified as Class CC pressure retaining components and their integral attachments must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of the ASME Boiler and Pressure Vessel Code and Addenda that are incorporated by reference in paragraph (b), subject to the limitation listed in paragraph (b)(2)(vi) and the modifications listed in paragraphs (b)(2)(ix) and (b)(2)(x) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

* * * * *

(v) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued after January 1, 1956:

(A) Metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class MC;

(B) Metallic shell and penetration liners which are pressure retaining components and their integral attachments in concrete containments must meet the inservice inspection, repair, and replacement requirements applicable to components which are classified as ASME Code Class CC; and

(C) Concrete containment pressure retaining components and their integral attachments, and the post-tensioning systems of concrete containments must meet the inservice inspection and repair requirements applicable to components which are classified as ASME Code Class CC.

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(6) * * *

(ii) * * *

(B) <u>Expedited examination of containment.</u>

(1) Licensees of all operating nuclear power plants shall implement the examinations specified for the first inspection interval in Subsection IWE and Subsection IWL of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a (b)(2)(ix) by (a date will be inserted that is 5 years later than the effective date of the final rule).

(2) The expedited examination may be used to satisfy the requirements of routinely scheduled examinations of Subsection IWE subject to IWA-2430(c) when the expedited examination occurs during the first containment inspection interval.

(3) The requirement for the expedited examination of the containment post-tensioning system may be satisfied by written commitments that are in place before (insert the effective date of the final rule) for examinations of the post-tensioning system.

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Dated at Rockville, Maryland this 3d day of January 1994.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.