Information Package
10 CFR 50.55a
Codes & Standards

Prepared For
The ASME Board on Nuclear Codes & Standards

Task Group on Regulatory Endorsement

October 25, 2004

TABLE OF CONTENTS

10 CFR 50.55a Final Rule Changes

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

10 CFR 50.55a

36FR11423

7/12/1971

Start of Codes & Standards Requirements

Statements Of Consideration

36FR11423

Publication_Date: 6/12/71

Effective_Date: 7/12/71

Codes and Standards for Nuclear Power Plants

On November 25, 1969, the Atomic Energy Commission published in the Federal Register (34 F.R. 18822) proposed amendments of its regulations in 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and 10 CFR Part 115, "Procedures for Review of Certain Nuclear Reactors Exempted from Licensing Requirements," which would establish minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of certain systems and components of boiling and pressurized water-cooled nuclear power reactor plants by requiring conformance with appropriate editions of published industry codes and standards

All interested persons were invited to submit written comments and suggestions for consideration in connection with the proposed amendments within 60 days after publication of the notice of proposed rule making in the Federal Register. Upon consideration of the comments received and other factors involved, the Commission has adopted the amendments set us below. These amendments have been changed substantially to reflect consideration of the comments received and to minimize interference with the established equipment procurement practices of the nuclear power industry. Some of the more significant changes from the proposed rule are:

a. The rule as rewritten requires that the determination of which code revisions are applicable be based on component order date rather than construction permit date. To prevent abuse of this provision and to minimize the use of outdated codes, the revised rule requires compliance with more recent codes than those in effect on the order date if the components are ordered more than a specified number of months before issuance of the construction permit.

b. The rule has been changed to make its provisions apply to future code revisions on the date they become effective unless the Commission has published a notice in the Federal Register that compliance with such requirements or any part thereof is unacceptable or unnecessary.

c. The definition of reactor coolant pressure boundary has been revised.

d. The date for compliance with the more recent industry codes has been changed from April 1, 1970, to January 1, 1971.

The Commission believes these changes adopted will eliminate most of the concerns expressed, will help to simplify and stabilize the facility licensing process, and will provide an equivalent increase in protection of the health and safety of the public to that which would be provided in the proposed rule.

Criterion 1 of the "General Design Criteria for Nuclear Power Plants" (Appendix A of Part 50) requires that structures, systems, and components of nuclear power plants which are important to safety be designed, fabricated erected, and tested to quality standards that reflect the importance of the safety functions to be performed. It has been generally recognized that, for boiling and pressurized water-cooled reactors, pressure vessels, piping, pumps, and valves which are part of the reactor coolant pressure boundary should, as a minimum, be designed, fabricated, inspected, and tested in accordance with the requirements of the applicable American Society of Mechanical Engineers (ASME) codes in effect at the time the equipment is purchased, and that protection systems (electrical and mechanical sensors and associated circuitry) should, as a minimum be designed to meet the criteria developed by the Institute of Electrical and Electronics Engineers (IEEE).

Because of the safety significance of uniform early compliance by the nuclear industry with the requirements of these ASME and IEEE codes and published code revisions, the Commission has adopted the following amendments to Parts 50 and 115, which require that certain components and systems of water-cooled reactors important to safety comply with these codes and appropriate revisions to the codes at the earliest feasible time. However, use of the ASME Code N-symbol is not required and inspection and survey systems other than those specified by ASME may be used if they provide an acceptable level of quality and safety. AEC quality assurance requirements are set forth in Appendix B to Part 50. The inspection and survey systems required by the amendments which follow may be used in partial fulfillment of these requirements to the extent that they are shown by the description of the quality assurance program required by Section 50.34(a) (7) to satisfy the applicable requirements of Appendix B.

In cases where compliance with specified code requirements, or portions thereof, would result in hardships or unusual difficulties

without a compensating increase in the level of safety, the Commission may grant exemptions under section 50.55a(b) (1). Section 50.55a(b) (2) provides a basis for the authorization of alternatives to the requirements of the specified codes and standards if it can be shown that an acceptable level of safety and quality will be provided.

The Commission considers that a significant improvement in the level of quality in design, fabrication, and testing of systems and components important to safety of water-cooled reactors will be afforded by compliance with the requirements of more recent versions of the codes than those specified in the amendments, or portions thereof, and encourages such compliance whenever practicable, regardless of the date of purchase of equipment or the provisions of these amendments.

Compliance with the provisions of the amendments and the referenced codes is intended to insure a basic, sound quality level. It may be that the special safety significance of a particular system or component will call for supplementary measures. If analysis of the system shows that such is the case, appropriate supplementary measures are expected to be adopted by applicants and licensees, or will be required by the Commission.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Parts 50 and 115 are published as a document subject to codification to be effective 30 days after publication in the Federal Register.

10 CFR 50.55a

39FR05773

3/18/1974

Added Summer 1973 Addenda for Codes and Standards

Statements Of Consideration

39FR05773

Publication_Date: 2/15/74
Effective_Date: 3/18/74

Codes and Standards for Nuclear Power Plants

On November 6, 1973, the Atomic Energy Commission published in the Federal Register (38 FR 30564) proposed amendments to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," and 10 CFR Part 115, "Procedures for Review of Certain Nuclear Reactors Exempted from Licensing Requirements," which would incorporate by reference new addenda to specified published industry codes.

The proposed amendments to Section 50.55a and Section 115.43 would provide that the editions of referenced addenda whose requirements must be met include only those addenda through the Summer 1973 Addenda as appropriate. Minor editorial changes to update references In Sections 50.55a and 115.43a were also included.

Interested persons were invited to submit written comments within 30 days. No comments were received. The Commission has adopted the proposed amendments without modification.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Parts 50 and 115 are published as a document subject to codification, to be effective on March 18, 1974.

10 CFR 50.55a

41FR06256

3/15/1976

Added ISI Requirements

Statements Of Consideration

41FR06256

Publication_Date: 2/12/76
Effective_Date: 3/15/76

Codes and Standards for Nuclear Power Plants and Technical Information

On September 30, 1974, the Atomic Energy Commission published in the Federal Register (39 FR 35180) proposed amendments to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would modify the basis for establishing which revisions of referenced codes and standards should be applied to the construction and operation of certain components of water-cooled nuclear power plants. Also the proposed amendments would incorporate by reference new addenda to the referenced codes and standards, delete obsolete references, correct typographical errors and make minor changes to Appendix G of Part 50. Interested persons were invited to submit written comments for consideration in connection with the proposed amendments by October 30, 1974.

On October 11, 1974 the Energy Reorganization Act of 1974/1/ was enacted into law. This Act provided for the abolition of the Atomic Energy Commission. Section 201 of this Act provided for the establishment of a Nuclear Regulatory Commission and a transfer to this new Commission of all licensing and related regulatory functions of the Atomic Energy Commission. In addition, section 301 of the Act provided that any proceedings pending before the AEC at the time of its abolition shall, to the extent that such proceedings relate to functions transferred by the Act, be continued.

/1/ Pub. L. 93-438 (88 Stat. 1233).

Upon consideration of the comments received and other factors involved, the Nuclear Regulatory Commission has adopted the proposed amendments with certain modifications in the form set forth below. These amendments have been changed substantially in Section 50.55a(g), "Inservice Inspection Requirement", to provide consistency in design requirements and to minimize interference with the established equipment procurement practices and inservice examination practices of the nuclear power industry. Some of the more significant changes to Section 50.55a(g) from the proposed rule are:

a. The effective rule requires that an operating license for a utilization facility be subject to the conditions specified in Section 50.55a(g), "Inservice Inspection Requirements."

b. To eliminate the misconception that the design of components needs to be continually modified and to provide a consistency between the design requirements for inspectability and the design requirements for construction, the provision on design requirements for inspectability of components has been changed to refer to the same code edition which is applied to the construction of such components.

c. The rule specifies inservice inspection requirements which apply to utilization facilities whose construction permits were issued prior to January 1, 1971.

d. Provisions in the rule for continued updating of requirements for inservice inspection to achieve compliance with more recent editions of the referenced code have been simplified and permit examination and testing programs to be updated at intervals of 40- and 20-months, respectively.

e. The rule specifies actions to be taken by a license when a revised inservice inspection program for a facility conflicts with the technical specifications or when a requirement of a subsequent edition of the referenced code is deemed impractical by the licensee and is not included in the inservice inspection program.

f. A provision has been added to the rule that the Commission may either (1) exempt the licensee from certain requirements determined to be inequitable and for which compliance may result in an undue burden without providing a significant increase in safety or (2) require the licensee to follow an augmented program when the Commission deems that additional assurance of structural reliability is necessary.

The Commission believes these changes adopted will facilitate the orderly application of new inservice inspection requirements in Section XI of the ASME Code which are incorporated by reference to operating nuclear power plants without causing significant modifications to the plant or an intolerable impact on the inservice inspection program. Also the Commission believes these changes adopted will provide an equivalent increase in the protection of the health and safety of the public to that which would be provided by the proposed rule.

The amendments to Section 50.55a set forth below which the Commission has adopted include the following:

a. References to published codes and addenda whose requirements must be met were changed to include Addenda through the Winter 1973 Addenda.

b. For a utilization facility for which a construction permit is issued on or after July 1, 1974 the rule requires that the determination of which code revision applies to a component be based on the docket date of the application for a construction permit rather than the date of issuance of the construction permit. This change should permit a more accurate assessment by the applicant of the code edition and addenda that will be in effect at the time components are ordered and thereby facilitate his procurement of long lead time components which are ordered well in advance of the construction permit date.

c. The rule modifies inservice inspection requirements applicable to components of nuclear power plants throughout the service life of the facility. Examination and testing requirements that became effective in new editions and addenda of Section XI of the ASME Code and are incorporated by reference in Section 50.55a would become applicable to all operating plants to the degree practical. The Commission will review such code changes with respect to impact on the existing operating facilities prior to incorporating by reference any new editions and addenda of Section XI.

The amendments to Appendix G conform the referenced edition and addenda of the ASME Code in that Appendix to those specified by Section 50.55a(b), including the periodic amendments and also clarify the upper-shelf energy requirements for beltline materials.

Other amendments delete references to obsolete documents and correct typographical errors.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, and Sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

41FR23931

6/14/1976

Limited 50.55a(g) to PWRs & BWRs

Statements Of Consideration

41FR23931

Publication_Date: 6/14/76
Effective_Date: 6/14/76

Codes and Standards for Nuclear Power Plants

On February 12, 1976, the Nuclear Regulatory Commission published in the Federal Register (41 FR 6256) amendments of the Commission's regulation 10 CFR Part 50, which, among other changes, modify the inservice inspection requirements applicable to components of systems of nuclear power reactors through the service life of the facility.

The prefatory language of Section 50.55a published on February 12, 1976 states that "each operating license for a utilization facility shall be subject to the conditions in paragraph (g) * * *." The code incorporated by reference in paragraph (g) applies solely to boiling and pressurized water-cooled nuclear power facilities. It appears that use of the overly broad term "utilization facility" in the prefatory language can be construed to apply the ASME Code to facilities not presently covered by it. It was not intended that Section 50.55a expand the applicability of section XI of the ASME Code to facilities other than those power reactors to which this Code applies.

Accordingly, the Commission is issuing clarifying amendments to the prefatory language of Section 50.55a and to Section 50.55a (g) to clarify this intent.

Inasmuch as the amendments set forth below are of a minor nature, good cause exists for omitting notice of proposed rule making, and public procedure thereon, as unnecessary, and for making the amendments effective June 14, 1976.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

42FR36803

8/17/1977

Added Winter 1976 Addenda III

Statements Of Consideration

42FR36803

Publication_Date: 7/18/77

Effective_Date: 8/17/77

Codes and Standards for Nuclear Power Plants

SUMMARY: The Nuclear Regulatory Commission is amending its regulation, "Codes and Standards," to incorporate by reference new addenda to specified published national codes and standards for the design, fabrication, construction, testing, and inspection of reactor components and systems. This would provide for improved methods of construction of nuclear reactor coolant-systems.

SUPPLEMENTARY INFORMATION: On March 31, 1977 the Nuclear Regulatory Commission published in the Federal Register (42 FR 17134) proposed amendments to its regulations, 10 CFR Part 50, "Licensing of Production and Utilization Facilities," which would incorporate by reference new addenda to specified published national codes and could clarify provisions in Section 50.55a and Appendix G to Part 50.

The proposed amendments would have incorporated by reference the Addenda through the Winter 1976 Addenda to Section III, Division 1, "Rules for the Construction of Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code. The Winder 975, Summer 1976, and Winter 1976 Addenda to Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code were not referenced in the proposed amendments but are expected to be referenced with modifications in a subsequent amendment to the regulations.

Interested persons were invited to submit written comments for consideration in connections with the proposed amendment by May 2, 1977. A number of adverse comments and significant questions were received in response to the notice of proposed rule making relating to the proposed changes to footnote 4 in Section 50.55 and to Appendix G to Part 50. One comment suggested that, in order not to delay the entire amendment while the adverse comments are being evaluated, Section 50.55a be amended in part to incorporate the Addenda through the Winter 1976 Addenda to Section III of the ASME Code as was proposed. After consideration of the comments the Commission has adopted the amendment to Section 50.55a set forth below which incorporates by reference the Addenda through the Winter 1976 Addenda to Section III of the ASME Code. The comments and questions on the proposed clarifying amendments will be evaluated separately and appropriate action taken accordingly.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and Section 552 and 553 of title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

43FR17337

5/24/1978

Added 1977 Edition III

Statements Of Consideration

43FR17337

Publication_Date: 4/24/78
Effective_Date: 5/24/78

Codes and Standards for Nuclear Power Plants

SUMMARY: The Nuclear Regulatory Commission is amending its regulation, Codes and Standards, to incorporate by reference a new edition and addenda of a national code that provides rules for the construction of nuclear power plane components. This amendment provides for the use of updated methods in nuclear power plant construction.

SUPPLEMENTARY INFORMATION: On January 31, 1978, the Nuclear Regulatory Commission published in the FEDERAL REGISTER (43 FR 4050) a proposed amendment to its regulations. 10 CFR Part 50, Licensing of Production and Utilization Facilities, which would incorporate by reference a new edition and new addenda to a specified national code. The Commission proposed to amend Section 50.55a to incorporate by reference the 1977 Edition and the Summer 1971 Addenda of Section III of the ASME Boiler and Pressure Vessel Code. The 1977 Edition of Section XI rules for Inservice Inspection of Nuclear Power Plant Components, for the ASME Code and Section XI addenda since the Summer 1975 Addenda are still being evaluated by the staff and are expected to be referenced with modifications in a subsequent amendment to the regulations.

Interested persons were invited to Submit written comments for consideration in connection with the proposed amendment by March 2, 1978. No adverse comments or significant questions were received in response to the notice of proposed rule making.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and Sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Part 50, are published as a document subject to codification.

10 CFR 50.55a

43FR56015

11/30/1978

Clarification For ISI

Statements Of Consideration

43FR56015

Publication_Date: 11/30/78
Effective_Date: 11/30/78

Codes and Standards for Nuclear Power Plants

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulation, "Codes and Standards," to clarify certain ambiguities to avoid misinterpretations of provisions which deal with requirements for in service inspection of nuclear power plants. These clarifications should result in more expeditious reviews and processing of in service inspection programs by the NRC.

SUPPLEMENTARY INFORMATION: In applying Section 50.55a, "Codes and Standards," of the Commission's regulations 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to current licensing actions, some confusion has resulted because of ambiguities that existed in the language. The amendments set forth below modify the language of § 50.55a to clarify the ambiguities.

Since the amendments are clarifications of regulations currently in effect and do not impose a burden to anyone applying the regulation, the Commission has found that good cause exists for omitting general notices of proposed rulemaking and public procedure thereon as unnecessary and for making the rule effective immediately without the customary 30-day waiting period.

Pursuant to the Atomic Energy Act of 1974, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendments to Title 10, Chapter I, Code of Federal Regulations, Part 50 are published as a document subject to codification.

1. In § 50.55a of 10 CFR Part 50, paragraphs (g)(4)(iii) and (g)(4)(iv) are amended by deleting the words "tests of pumps and valves for assessing operational readiness" and substituting therefor the words "tests to verify operational readiness of pumps and valves whose function is required for safety"; and paragraph (g)(6)(i) is amended by deleting the words "impractical and may grant such relief" and substituting therefor the words "impractical. The Commission may grant such relief and may impose such alternative requirements".

(Secs. 103, 104, 161i,. Pub. L. 83-703; 68 Stat. 936, 937, 948 (42 U.S.C. 2133, 2134. 2201(i)).)

10 CFR 50.55a

46FR20153

5/4/1981

Added Winter 1978 Addenda to
Summer 1979 Addenda III, XI

Statements Of Consideration

46FR20153

Publication_Date: 5/4/81
Effective_Date: 5/4/81

Codes and Standards for Nuclear Power Plants

SUMMARY: The Nuclear Regulatory Commission is amending its regulations to incorporate by reference new addenda of the ASME Boiler and Pressure Vessel Code. The sections of the ASME Code being incorporated provide rules for the construction of nuclear power plant components and specify requirements for inservice inspection of those components. Adoption of these amendments permits the use of improved methods for construction and inservice inspection of nuclear power plants.

SUPPLEMENTARY INFORMATION: On December 31, 1980, the Nuclear Regulatory Commission published in the Federal Register (45 FR 86500) proposed amendments to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed amendments revised Section 50.55a to incorporate by reference the Winter 1978 Addenda and the Summer 1979 Addenda to Section III, Division 1, "Rules for the Construction of Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code and Section XI, "Inservice Inspection of Nuclear Power Plant Components," of the ASME Code.

Interested person were invited to submit written comments for consideration in connection with the proposed amendment by February 17, 1981. No comments were received. The Commission has adopted the proposed amendment with a minor editorial revision to accommodate the incorporation by reference of the ASME Code.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

46FR63208

2/1/1982

Added winter 1979 Addenda to
Winter 1980 Addenda III, XI

Statements Of Consideration

46FR63208

Publication_Date: 12/31/81

Effective_Date: 2/1/82

Codes and Standards for Nuclear Power Plants; ASME Boiler and Pressure Vessel Code; Incorporation by Reference

SUMMARY: The Commission is amending its regulations to incorporate by reference new addenda of the ASME Boiler and Pressure Vessel Code. The sections of the ASME Code being incorporated provide rules for the construction of nuclear power plant components and specify requirements for inservice inspection of those components. Adoption of these amendments will permit the use of improved methods for construction and inservice inspection of nuclear power plants.

SUPPLEMENTARY INFORMATION: On July 27, 1981 the Nuclear Regulatory Commission published in the Federal Register (46 FR 38374) proposed amendments to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed amendments revised Section 50.55a to incorporate by reference the Winter 1979 Addenda, 1980 Edition, Summer 1980 Addenda, and the Winter 1980 Addenda of section III and the Winter 1979 Addenda, the 1980 Edition, and the Winter 1980 Addenda to section XI of the ASME Boiler and Pressure Vessel Code.

The incorporation of the new edition and addenda does not change any of the previous supplementary requirements included in the regulation. Until the ASME Code adds current requirements for inspecting the residual heat removal and emergency core cooling systems, the regulation will continue to require that these systems be inspected to the provisions cited in Section 50.55a(b)(2)(iv).

One comment was received on the proposed rule. It recommended that Section 50.55a(b)(2)(i) be revised to reflect the updated incorporation by reference of the ASME Boiler and Pressure Vessel Code, as stated in Section 50.55a(b)(2). The reason given for recommending the revision was that Section 50.55a(b)(2)(i) was being interpreted as prohibiting the use of addenda that were published and incorporated subsequent to the Summer 1978 Addenda to the 1977 Edition of the code.

As a result of the comment the title and the body of Section 50.55a(b)(2)(i) were editorally revised to clarify them.

Some of the changes effected in the addenda which are incorporated through the adoption of the amendments are:

1. Section XI requires that a system hydrostatic test be performed after all inservice repairs and replacements to Class 1 systems and components.

2. Section III requires that there be some method of remotely monitoring the position of pressure relief devices.

3. Both sections III and XI allow the practical exam, required for Nondestructive Examination (NDE) qualification, to be given by the American Society for Nondestructive Testing (ASNT) rather than the employer.

4. Section III requires that licensees meet the requirements of the national standard, ANSI/ASME N626.3-1979 "Qualification and Duties of Personnel Engaged in ASME Boiler and Pressure Vessel Code, Section III, Divisions 1 and 2, Certifying Activities."

Interested person were invited to submit written comments for consideration in connection with the proposed amendment by September 10, 1981. One editorial comment was received and the paragraphs addressing the effective edition and addenda of the ASME Code were added to the preamble in response to the comment. The Commission has adopted the proposed amendment with a minor editorial revision to accommodate the incorporation by reference of the ASME Code.

Paperwork Reduction Act Statement

The recordkeeping requirements contained in this Regulation have been approved by the Office of Management and Budget; OMB approval No. 3150-0011.

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 have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation o 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 rule does not fall within the purview of the Act.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

47FR30459

8/13/1982

Added Summer 1983 Addenda III

Statements Of Consideration

47FR30459

Publication_Date: 7/14/82
Effective_Date: 8/13/82

Codes and Standards for Nuclear Power Plants

SUMMARY: The Commission is amending its regulations to incorporate by reference the Summer 1981 Addenda of the ASME Boiler and Pressure Vessel Code. The sections of the ASME Code being incorporated provide rules for the construction of nuclear power plant components. Adoption of these amendments will permit the use of improved methods for construction.

SUPPLEMENTARY INFORMATION: On February 3, 1982 the Nuclear Regulatory Commission published in the Federal Register (47 FR 5010) proposed amendments to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed amendments revised Section 50.55a to incorporate by reference the Summer 1981 Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Some of the changes effected in the addenda which are incorporated through the adoption of the amendments are:

1. Article NCA-3000 of Section III was revised to add a requirement that N, NA, and NPT certificate holders be responsible for documentation of the review and approval of materials used by them and the preparation, accumulation, control, and protection of required records while in their custody. Also, the owner must review the materials documentation to verify that the Code Edition, Addenda, and Code Cases used satisfy NCA-1140 and are acceptable to the regulatory and enforcement authorities.

2. Article NCA-8000 of Section III was revised editorially to make it easier to read and understand. Also, two new provisions, NCA-8240(b) and NCA-8430, were added. NCA-8240(b) describes the provisions that must be met if a name plate is to be removed from an item which has been installed in a nuclear power plant system. NCA-8430 describes alternatives for compiling the Code Data Reports so that they can be traced from the Data Report Form.

3. Article NB-3500 of Section III was revised to remove the nomenclatures "normal duty valve," "severe duty valve," "standard valve," and "expected cycle," but there were no technical changes associated with dropping these nomenclatures.

4. Article NB-6000 was given an extensive editorial rewrite which mainly reorganized the paragraphs into a more comprehensive form. Also added were a subarticle on special test procedures and a subparagraph allowing the hydrostatic testing of pump and valve subassemblies.

Interested persons were invited to submit written comments for consideration in connection with the proposed amendment by May 5, 1982. One information/editorial comment on the supplementary information section of the proposed rule was received bu no significant comments were received. The necessary editorial corrections were made. The Commission has adopted the proposed amendment with a minor editorial revision to accommodate the incorporation by reference of the ASME Code.

Paperwork Reduction Act Statement

The recordkeeping requirements contained in this Regulation have been approved by the Office of Management and Budget; OMB approval No: 3150-0011.

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 have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation o 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 rule does not fall within the purview of the Act.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Nuclear Power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

48FR50878

12/7/1983

Added Summer 1982 Addenda III

Statements Of Consideration

48FR50878

Publication_Date: 11/4/83
Effective_Date: 12/7/83

Codes and Standards for Nuclear Power Plants; Summer 1982 Addenda

SUMMARY: The Commission is amending its regulations to incorporate by reference the Summer 1982 Addenda of the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code. The sections of the ASME Code being incorporated provide rules for the construction on nuclear power plant components. Adoption of these amendments will permit the use of improved methods for construction of nuclear power plants.

SUPPLEMENTARY INFORMATION: On December 22, 1982 the Nuclear Regulatory Commission published in the Federal Register proposed amendments to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed amendment would revise Section 50.55a to incorporate by reference the Summer 1982 Addenda to Section III, Division 1, "Rules for the Construction of Nuclear Power Plant Components," of the ASME Boiler and Pressure Vessel Code.

The incorporation of the new Addenda does not change any of the previous supplementary requirements included in the regulation.

Some of the changes effected in the addenda that are incorporated by this rule follow:

1. The foreword of all sections of the Code was revised regarding interpretations of the Code to restrict the authority to issue such interpretations to the American Society of Mechanical Engineers. Previously, paragraph thirteen of the foreword had permitted the National Board of Boiler and Pressure Vessel Inspectors to issue interpretations of the ASME Boiler and Pressure Vessel Code.

2. Paragraph NCA-8230(b) was revised to delete the requirement that the location of the nameplate for component supports be shown on the support drawing.

3. Code Case N-100, "Pressure Relief Valve Design," was adopted into the body of the Code as paragraph NB-3590. The Code Case will be annulled when the Summer 1982 Addenda become effective.

4. Paragraph NB-5520 was revised to update the reference to the American Society for Nondestructive Testing (ASNT) standard SNT-TC-1A, from the 1975 to the 1980 Edition. It was also revised to clearly state that even if an outside agency or ASNT provided the qualification examinations, the employer is still responsible for certifying its own personnel.

Interested persons were invited to submit written comments for consideration in connection with the proposed amendment by February 22, 1983. One comment was received on the proposed rule. The Summer 1982 Addenda invoke the June 1980 Edition of Recommended Practice No. SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," in paragraph NB-5520 in lieu of the 1975 Edition of SNT-TC-1A which was previously invoked. The commenter objected to updating the edition of SNT-TC-1A referenced by the Code because the commenter was of the opinion that the 1980 Edition of SNT-TC-1A no longer requires Level III individuals to demonstrate their ability to perform the examinations for which they are being qualified. A detailed comparison of the 1975 and 1980 Editions of SNT-TC-1A reveals that although the requirements for qualifications of Level III individuals were changed, neither edition specifically requires a demonstration of the individual's ability to perform the examination. No changes to the rule were made in response to the comment.

The NRC staff is currently working with a committee of industry representative that is developing an improved requirements document for the qualification and certification of nondestructive examination personnel. Additionally, the staff is involved in a study of the qualification and certification of various quality assurance and quality control personnel that includes consideration of nondestructive examination personnel. At this time, the staff has not concluded that hands-on practical examinations for Level III nondestructive examination personnel are warranted, and finds that the provisions of the June 1980 Edition of SNT-TC-1A are acceptable for Section III Code activities, pending the development of any new requirements.

In addition to the public comment, there was a concern raised by the NRC staff on the change in the 1980 Edition of SNT-TC-1A to the use of the word "should" in numerous places in the standard where the word "shall" had been used in the past. The concern centered around whether or not the change in language resulted in a change in the enforceability of the provisions of SNT-TC-1A. Because of the staff's concern, an inquiry, NI 83-033, was submitted to the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code Committee, asking for an interpretation of the Code's endorsement of SNT-TC-1A. The response to the inquiry was that regardless of the language used in SNT-TC-1A, the Code's endorsement of SNT-TC-1A makes the provisions of SNT-TC-1A mandatory for Code activities.

Paperwork Reduction Act Statement

This final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). Existing requirements were approved by the Office of Management and Budget approval number 3150-0011.

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 have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation o 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 of 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 rule does not fall within the purview of the Act.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Incorporation by references, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendments to Title 10, Chapter 1, Code of Federal Regulations, Part 50 are published as a document subject to codification.

10 CFR 50.55a

49FR10657

3/22/1984

Editorial Correction

Statements Of Consideration

49FR10657

Publication_Date: 3/22/84

Effective_Date:

Codes and Standards for Nuclear Power Plants

Correction

In FR Doc. 84-6953 beginning on page 9711 of the issue Thursday, March 15, 1984, make the following corrections:

1. On page 9713, third column, in the first line of section 50.55a (a)(2), "System" should be "System".
2. On page 9714, second column, in section 50.55a (c)(4) and (d)(1), replace the parenthetical expression with "May 14, 1984".
3. On the same page, third column, in the fourth line of footnote 9 to section 50.55a, "50.349b)" should be "50.34(b)".

10 CFR 50.55a

49FR09711

5/14/1984

Deleted Many Codes & Standards
and Added Class 2 & 3

Statements Of Consideration

49FR09711

Publication_Date: 3/15/84

Effective_Date: 5/14/84

Codes and Standards for Nuclear Power Plants

SUMMARY: The Commission is amending its regulation which incorporate by reference national codes and standards for the construction of nuclear power plant components. The amendments increase specific references to the ASME Boiler and Pressure Vessel Code to include subsections that provide rules for the construction of certain safety systems and that clarify existing regulations by removing obsolete provisions no longer applicable. This action establishes enforceable requirements to replace previous guidance. In addition, the amendments will ensure appropriate use of the referenced code and clarify NRC application requirements.

SUPPLEMENTARY INFORMATION: On April 13, 1982, the Nuclear Regulatory Commission published in the Federal Register (47 FR 15801) proposed amendments to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which would amend Section 50.55a, "Codes and Standards." The proposed amendments constituted a general revision of Section 50.55a designed to update NRC requirements after 10 years of experience and make them more consistent with pertinent national standards. More specifically, the proposed amendments would have (3) added specific references to parts of Section III of the ASME Boiler and Pressure Vessel Code which apply to the construction of Classes 2 and 3 components, (b) deleted obsolete references and provisions, and (c) simplified the procedure for authorizing alternatives to certain NRC requirements.

Interested persons were invited to submit written comments by June 14, 1982. Twelve letters of comment were received. Of these, seven commentors supported, in general, the main provision of the proposed amendments to add specific references to additional parts of the ASME Code. One commentor disagreed. Another commentor objected to deleting the exemption from the code requirement for applying the Code N Symbol. Most of the adverse comments dealt with the application of specific requirements. A brief summary of the more significant adverse comments and staff responses follows.

Eight of the commentors objected to the proposed provisions that applied to the classification of components used for determining which part of the ASME Code should apply. Three of the eight replies came directly from the American Nuclear Society (ANS) Standards Committee and contained detailed objections to the classification provisions and recommended that the rule apply the classification systems developed by ANS as set out in the draft standards ANS 51.1 and ANS 52.1. Five of the eight commentors endorsed the letters sent by ANS.

To resolve the question of component classification, the staff met on September 15, 1982, with representatives of ANS and other interested organizations and discussed in detail the existing NRC and industry practices for classification of safety-related components and related problems. Those persons attending the meeting agreed that classification is more appropriately a subject for a regulatory guide than for a regulation since classifications are frequently plant specific and involve so many variables that regulatory controls need to be more flexible. As a result, the NRC has revised the final rule to eliminate detailed classification of components. Instead, general guidance has been added for establishing the classifications. Further, the NRC staff has agreed to evaluate the ANS classification systems for referencing in a Regulatory Guide.

A copy of the comments received on the proposed rule and an abstract of the comments which gives the staff response to each issue raised by the commentors is available for public inspection and copying for a fee at the Commission's Public Document Room at 1717 H Street NW., Washington, DC. Single copies may be obtained by written request to the Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: A. Taboada.

Rulemaking Background

Commission policy, stated in General Design Criterion 1, "Quality Standards and Records" (10 CFR Part 50, Appendix A) and in Section 50.55a, prescribes that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. In this regard, on June 12, 1971 (36 FR 11423), the Commission incorporated by reference in Section 50.55a several codes published by the American Society of Mechanical Engineers (ASME) to be applied to the construction of components of the reactor coolant pressure boundary of water-cooled nuclear power reactors. Section III of the ASME Boiler and Pressure Vessel Code, applicable at that time only to nuclear vessels, was incorporated by references to establish standards for vessels. Other ASME codes (described below) of more general application were incorporated by reference to establish standards for piping, pumps, and valves.

Subsequently, the ASME Boiler and Pressure Vessel Code Committee expanded Section III of the Code to apply to other nuclear power plant components (including piping, pumps, and valves) in addition to nuclear vessels. The expansion included many of the appropriate provisions from the more general codes incorporated by reference in Section 50.55a on June 12, 1971.

On February 12, 1976 (41 FR 6256), the Commission amended Section 50.55a to make the expanded Section III effective code for piping, pumps, and valves of the reactor coolant pressure boundary for water-cooled nuclear plants for which construction permits were issued on or after July 1, 1974. The amendment limited use of the more general codes to plants for which construction permits were issued before July 1, 1974.

Section III of the Code is regularly updated by ASME in new editions and addenda to include new developments and to reflect experience with the use of the Code. Those parts of the new editions and addenda that pertain to the reactor coolant pressure boundary are reviewed by the Commission staff and, if acceptable, are incorporated by reference in Section 50.55a. Although not specifically included in the regulations, the remaining parts of Section III that pertain to other systems are also reviewed by the Commission staff and, if acceptable, are used in the evaluation of specific license applications. Several parts of Section III that apply to Class 2 and 3 components have been used as guidance in this manner for approximately 10 years and are referenced in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,"

Presently, other parts of the ASME Nuclear Code (Section III) cover metal containments, component supports, core support structures and concrete vessels. It is the intent of the Commission to incorporate by reference specific parts of the ASME Nuclear Code after appropriate evaluations and as adequate experience with use of each part of the Code confirms its acceptability.

Substance of the Final Rule

Now that experience has shown additional parts of Section III of the ASME Boiler and Pressure Vessel Code to be adequate for use on a general basis, the Commission is adding specific references to these additional parts in Section 50.55a. These additions include the requirements for Class 2 Components, which are found in Subsections NC and NCA of the Code, and the requirements of Class 3 Components, which are found in Subsections ND and NCA of the Code.

To clarify the requirements of Section 50.55a, the Commission is also deleting obsolete incorporations by reference of general codes which are superseded by Section III of the ASME Code. However, any previous acceptance of general codes by the Commission for nuclear plants for which construction permits were issued prior to these deletions will continue to be in effect, and deletion of these Codes from Section 50.55a should not be construed otherwise. The Codes being deleted include:

(a) For piping of the Reactor Coolant Pressure Boundary--

-- American Standard Code for Pressure Piping, ASA B31.1

-- USA Standard Code for Pressure Piping, USAS B31.1.0

-- USA Standard for Nuclear Power Piping, USAS B31.7

-- ASA B31.1 Code Cases N7, N9 and N10

(b) For pumps of the Reactor Coolant Pressure Boundary--

-- Draft ASME Code for Pumps and Valves for Nuclear Power

-- ASA B31.1 Code Cases N7, N9, and N10

(c) For values of the Reactor Coolant Pressure Boundary--

-- American Standard Code for Pressure Piping ASA B31.1

-- USA Standard Code for Pressure Piping USAS B31.1.0

-- Draft ASME Code for Pumps and Valves for Nuclear Power

-- ASA B31.1 Code Cases N2, N7, N9 and N10

The Commission is also making two procedural changes in the amendment to Section 50.55a. One change deals with the need for NRC authorization for a proposed alternative to requirements in the regulation. The amendment clarifies the existing regulation by providing that the Director of the Office of Nuclear Reactor Regulation may authorize alternatives to the requirements in Section 50.55a after demonstration by the applicant that (a) the proposed alternative would provide an acceptable level of quality and safety or (b) compliance with specific requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality or safety.

The other procedural change deletes the obsolete provision of paragraph (a) that exempts nuclear components constructed to

ASME Code rules from the Code requirement that the Code N-symbol stamp be applied to these components. This exemption was initiated when there was no provision for foreign suppliers to comply with the administrative enforcement aspect of the Code, Previously, foreign suppliers, fully qualified in other respects, could not be issued a Code N-symbol stamp to apply to components and, hence, would have been excluded from supplying components for domestic nuclear plants. This situation has been changed and foreign suppliers may now be issued Code N-symbol stamps.

Regulatory Analysis

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW., Washington, D. C. Single copies of the analysis may be obtained from A. Taboada, Office of Nuclear Regulatory Research, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, telephone (301) 443-7903.

Paperwork Reduction Act Statement

The application, reporting and recordkeeping requirements contained in this proposed regulation have been approved by the Office of Management and Budget; OMB approval No. 3151-0011.

Regulatory Flexibility Statement

This rule (Section 50.55a) requires that certain nuclear power plant components be constructed to the nuclear section of the ASME Boiler and Pressure Vessel Code, the standard used by the nuclear industry to comply with the NRC General Design Criteria and Quality Assurance Criteria. To attest that a component has been found to meet all of the specified Code requirements, the ASME Code requires that the component be stamped with a Code N-Stamp issued to manufacturers qualified through an ASME accreditation survey. The ASME Code also calls for specific inspections and verifications of records by an independent Authorized Inspection Agency (AIA) established by a legal authority (state or municipality). This amendment to Section 50.55a expands the applicability of the ASME Code and removes obsolete provisions and the exemption, previously in the rule, that the Code N Symbol Stamp need not be applied.

The direct cost incurred by a small manufacturer as a result of this amendment will vary depending on the circumstances. An estimated 550 manufacturers, approximately 10% of which may be small manufacturers, are already certified as N-Stamp holders by ASME, and it is assumed that they would not incur new costs. An ASME accreditation survey (i.e., audit and evaluation) costs a manufacturer an average of $9000 and recurs at 3 year intervals. The cost for Authorized Inspection Agency services is approximately $250 per day per inspector plus expenses. This service for a small manufacturer would typically include two (one-day) audits per year and three inspection visits per order but would vary depending on the complexity of the components being manufactured.

On this basis, the direct cost for a small manufacturer to maintain code accreditation and provide the required third party (AIA) inspection is estimated to be as high as $3500 per year assuming that the cost of inspection visits are applied to the price of the component. For multiorders the cost per order for accreditation would be less.

Approximately 30 new firms per year, some of which are expected to be small firms, apply for ASME accreditation to be N stamp holders. For these firms, particularly if new to the nuclear field, additional indirect costs may be incurred in establishing and maintaining as ASME certified shop. Such indirect costs might result from complying with, for example, code requirements for qualifying welders, establishing specific quality control programs, and maintaining appropriate procedures and records. However, since Appendices A and B of 10 CFR Part 50 already require quality standards and a quality assurance program that parallels the ASME Code requirements, and in fact, the ASME Code has become the NRC and industry standard for compliance with these appendices, the costs associated with meeting requirements of Appendices A and B are estimated to be substantially equivalent to the cost of meeting ASME Code requirements. Therefore, the NRC estimate that no major additional cost would be incurred as a result of this rule, by a small manufacturing shop providing nuclear quality products.

The NRC requested comments on the economic impact of this rule on small businesses in the proposed amendment. No comments were received from small businesses. However, two operating utilities commented that this rule would, in effect, preclude the occasional purchase of spare parts or renewal parts from some small shops who probably would not find it economically feasible to establish and maintain an ASME certified shop. NRC estimates that the number of small shops so affected would be few and would not constitute a substantial number of the small businesses involved in nuclear component construction. Further, the rule contains a provision to permit alternatives to specific ASME Code requirements, in case of hardships, that would, in effect, permit the use of such uncertified shops if an equivalent level of quality and safety were provided.

Thus, in accordance with the Regulatory Flexibility Act, 3 U.S.C. 605(b), the NRC hereby certifies that this rule will not have a significant economic impact upon a substantial number of small entities.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Incorporation by reference, Intergovernmental relations, Nuclear power plants

and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of Title 5 of the United States Code, the following amendments to 10 CFR Part 50 are published as a document subject to codification.

10 CFR 50.55a

50FR38970

10/28/1985

Added Winter 1982 Addenda to 1983
Edition III, XI

Statements Of Consideration

50FR38970

Publication_Date: 9/26/85

Effective_Date: 10/28/85

Codes and Standards for Nuclear Power Plants

SUMMARY: The Commission is amending its regulations to incorporate by reference the Winter 1982 Addenda, Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda and 1983 Edition of Section III, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and the Winter 1982 Addenda, Summer 1983 Addenda, and 1983 Edition of Section XI, Division 1, of the ASME Code. The sections of the ASME Code being incorporated provide rules for the construction of light-water-cooled nuclear power plant components and specify requirements for inservice inspection of those components. Adoption of these amendments would permit the use of improved methods for construction and inservice inspection of nuclear power plants.

SUPPLEMENTARY INFORMATION: On May 17. 1985, the Nuclear Regulatory Commission published in the Federal Register (50 CFR 20574) a proposed amendment to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to update existing references to specific editions and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), to make editorial corrections to the existing rule: to simplify the language of the rule; and to delete two obsolete provisions.

This amendment revises Section 50.55a to incorporate by reference all editions through the 1983 Edition and all addenda through the Summer 1984 Addenda that modify Division 1 rules of Section III, "Rules for the Construction of Nuclear Plant Components," and all editions through the 1983 Edition and all addenda through the Summer 1983 Addenda that modify Division 1 rules, of Section XI, "Rules for the Inservice Inspection of Nuclear Power Plant Components," of the ASME Code. The Summer 1983 Addenda for Section XI does not include technical requirements related to Division 1, but is included in the reference to prevent the confusion that might occur with a lack of continuity in the addenda references.

Editorial revisions are included in this amendment to correct certain existing footnote and paragraph references that are inconsistent with the last amendment (49 FR 9711) this rule and to simplify the language. These editorial revisions are contained entirely in Section 50.55a(g).

For facilities whose operating licenses were issued prior to March 1, 1976, this rule provided the effective date for implementing the inservice inspection requirements and for defining the effective edition and addenda of the Code for the start of the next one-third of a 120-month inspection interval after September 1, 1976. Since this one-third of an inspection interval has already been completed for all applicable facilities, Section 50.55a(g)(4)(iii) which addresses the implementation of that inspection is unnecessary and is deleted by this amendment.

Power reactors for which a notice of hearing on an application for a provisional construction permit or a construction permit had been published on or before December 31, 1970, were permitted to use the rules for construction permits prior to January 1, 1971. This amendment deletes Section 50.55a(i) which covers this provision because it is no longer necessary. Section 50.55a(c)(4) provides that for these and other facilities that received a construction permit prior to May 14, 1984, the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for the component at the time of issuance of the construction permit.

Interested persons were invited to submit written comments for consideration in connection with the proposed amendments and the draft regulatory analysis by July 16, 1985. No adverse comments or significant questions were received in response to the notice of proposed rulemaking.

Environmental Impact: Categorical Exclusion

The NRC has determined that this final rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(3). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this final rule.

Regulatory Analysis

The Commission has prepared a regulatory analysis on this regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 1717 H St. NW., Washington, DC. Single copies of the analysis may be obtained from Mr. G. C. Millman, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington. DC. 20555, Telephone (301) 443-7862.

Paperwork Reduction Act Statement

This final rule incorporates by reference information collection requirements that were reviewed by the Office of Management and Budget. The OMB approval number is 3150-0011.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule does not have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and 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.

List of Subjects in 10 CFR Part 50

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

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C 553, the NRC is adopting the following amendments to 10 CFR Part 50.

10 CFR 50.55a

53FR16051

5/5/1988

Added 1984 Addenda to 1986 Edition III, and Winter 1983 Addenda to 1986 Edition XI

Statements Of Consideration

53FR16051

Publication_Date: 5/5/88

Effective_Date: 5/5/88

Codes and Standards for Nuclear Power Plants

SUMMARY: The Commission is amending its regulations to incorporate by reference the Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition of Section III, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and the Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition of Section XI, Division 1, of the ASME Code. A limitation is placed on the use of paragraph IWB-3640 as contained in the Winter 1983 Addenda and Winter 1984 Addenda of Section XI, Division 1. This limitation requires that for certain types of welds, IWB03640 when implemented shall be used as modified by the Winter 1985 Addenda. In addition, the existing modification pertaining to the inservice inspection of pressure retaining welds in ASME Code Class 2 piping has been revised to limit its applicability up to the 1983 Edition with addenda up through the Summer 1983 Addenda. The sections of the ASME Code being incorporated by reference provide rules for the construction of light-water-cooled nuclear power plant components and specify requirements for inservice inspection of those components. Adoption of these amendments would permit the use of improved methods for construction and inservice inspection of nuclear power plants.

SUPPLEMENTARY INFORMATION:

On June 26, 1987, the Nuclear Regulatory Commission published in the Federal Register (52 FR 24015) a proposed amendment to its regulation, 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to update the reference to editions and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). This amendment revises Section 50.55a to incorporate by reference all editions through the 1986 Edition and all addenda through the Winter 1985 Addenda that modify Division 1 rules of Section III, "Rules for the Construction of Nuclear Plan Components," and, subject to certain limitations and modifications, addenda through the Winter 1985 Addenda that modify Division 1 rules of Section XI, "Rules for the Inservice Inspection of Nuclear Power Plant Components," of the ASME Code. Specifically, this amendment to Section 50.55a incorporates by reference the Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition for Division 1 rules of Section III, and the Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition for Division 1 rules of Section XI of the ASME Code. The 1986 Edition is equivalent to the 1983 Edition, as modified by the Summer 1983 Addenda, Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, and Winter 1985 Addenda. The Summer 1984 Addenda and Summer 1985 Addenda for Section XI do not include technical requirements, but are included in the reference to prevent the confusion that might occur with a lack of continuity in the addenda references.

Interested persons were invited to submit written comments for consideration in connection with the proposed amendment by August 15, 1987. Comments were received from three individuals in response to the notice of proposed rulemaking. Two of the commenters were in favor of the proposed rule, and submitted suggestions for editorial clarifications. One of these commenters was concerned that the manner proposed for specifying the endorsed editions and addenda in Sections 50.55a(b)(1) and (b)(2) was potentially confusing for this specific amendment because the latest addenda that is specified does not modify the latest edition that is specified (i.e., the Winter 1985 Addenda modifies the 1983 Edition). The staff agrees with the commenter and has modified paragraphs (b)(1) and (b)(2) to make it clear that the Winter 1985 Addenda applies to the 1983 Edition, and that the 1986 Edition is the latest ASME Code update being incorporated by reference into the regulation.

The other commenter in favor of the proposed rule believes that the proposed additional sentence in Section 50.55a(b)(2)(i) which specifies a limitation on the use of IWB-3640 for certain addenda should be provided for clarity in a separate paragraph. The staff has considered and adopted this suggestion. In this final rule, the specified limitation is contained in a new paragraph (b)(2)(v). This commenter also recommended a revision to Footnote 6 to clarify details regarding implementation of the code cases specified in the identified regulatory guides. It is the opinion of the staff that the rule should not be cluttered with such information. Therefore, that proposed revision has not been incorporated into the final rule. However, the staff is considering incorporating additional information directly into the regulatory guides to clarify their use.

Additionally, this commenter noted that the Winter 1983 Addenda to Section XI included significant improvements to the inservice inspection of Class 2 piping. This comment is correct. In particular, the rules specified in that addenda satisfy NRC staff concerns associated with the rules specified in earlier addenda for the examination of pressure retaining welds in ASME Code Class 2 piping, including residual heat removal systems, emergency core cooling systems, and containment heat removal systems. The staff previously addressed these concerns by specifying a modification in Section 50.55a(b)(2)(iv), which required that the extent of examinations for pressure retaining welds in ASME Code Class 2 piping be determined based upon specific rules in the 1974 Edition and Addenda through the Summer 1975 Addenda. Although the commenter did not make the point specifically, the proposed rule should have recognized the improvements in the Winter 1983 Addenda by incorporating a revision to limit the applicability of the required existing modification specified in paragraph (b)(2)(iv) to ASME Code editions and addenda

up to the 1983 Edition with addenda up through the Summer 1983 Addenda. This final rule incorporates this limitation to the use of the modification specified in paragraph (b)(2)(iv).

The third commenter opposed the proposed amendment. That commenter believes that the NRC should not rely on industry standards, but rather should develop its own standards based upon NRC experience and data. NRC practice is to utilize national standards, such as the ASME Code, whenever possible to define acceptable ways of implementing the NRC's basic safety regulations. This is consistent with OMB Circular No. A-119 (Revised),/1/ which provides policy and administrative guidance to federal agencies regarding participation in the development and use of voluntary standards. Consistent with this policy, the NRC staff participates actively in the development of many national standards, including the ASME Code, to ensure that NRC experience and data is part of the information base used to support development of the standard. Although the NRC staff is heavily involved in the development of the ASME Code, endorsement of the ASME Code by the NRC without exception is not an automatic action as evidenced by the existing limitations and modifications specified in Section 50.55a(b)(2) and the new limitation specified in paragraph (b)(2)(v) by this final rule.

/1/ Single copies of OMB Circular No. A-119 may be obtained from the OMB Publications Office, 726 Jackson Place NW., Washington, DC 20503, Telephone (202) 395-7332.

Paragraph IWB-3640 was incorporated into the Winter 1983 Addenda of Section XI, Division 1, to provide procedures and acceptance criteria for determining the acceptability for continued service of austenitic stainless steel piping with flaws in excess of the allowable indications specified in IWB-3514.3. Concern was expressed by the NRC staff and others that IWB-3640, as presented initially in the Winter 1983 Addenda, did not provide an acceptable level of margin against failure for materials with low toughness, such as might occur in fluxed welds (i.e., submerged arc welds (SAW) or shielded metal arc welds (SMAW)). One concern with low toughness materials was that such materials might fail at load levels below limit load. Additionally, there was concern that secondary stresses, which were not included in the stress analysis procedures required by IWB03640, might contribute to the failure of low toughness materials.

The ASME established a special task group to address the concerns associated with paragraph IWB-3640 as contained in the Winter 1983 Addenda. In the interim, the NRC staff required that licensees utilizing the procedures and acceptance criteria of IWB-3640, as contained in the Winter 1983 Addenda, apply additional safety factors in their analyses to be submitted to the staff to account for the above concerns. NRC staff acceptance criteria were provided in Generic Letter 84-11, "Inspections of BWR Stainless Steel Piping."/2/

/2/ A copy of Generic Letter 84-11 is available for inspection or copying for a fee st the NRC Public Document Room, 1717 H Street NW., Washington, DC.

In the opinion of the NRC staff, the concerns associated with material toughness have been adequately addressed by the ASME Code with the modification to paragraph IWB-3640 in the Winter 1985 Addenda. This addenda provides specific acceptance criteria for SAW and SMAW type welds, and these criteria address the concerns associated with limit load and the need to incorporate secondary stresses in the evaluation.

This amendment to Section 50.55a incorporates a limitation in paragraph (b)(2)(v) that allows for the use of paragraph IWB-3640 as contained in the Winter 1983 Addenda and Winter 1984 Addenda, for all applications permitted in that paragraph except those associated with SAW and SMAW type welds. For these welds, this amendment specifies that paragraph IWB-3640, as modified by the Winter 1985 Addenda, must be used.

Footnote 8 of Section 50.55a provides reference to the NRC Regulatory Guides that denote which ASME Code Cases have been determined to be acceptable to the NRC staff for implementation. Previously, this footnote provided reference to only Regulatory Guides 1.84 and 1.85, which denote acceptability of Section III, Division 1, Code Cases on design and fabrication, and on materials, respectively. This amendment revises Footnote 6 to incorporate a reference to Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI Division 1," which identifies the Code Cases acceptable to the NRC staff for implementation in the inservice inspection (ISI) program of light-water-cooled nuclear power plants. At present, the Implementation section of Regulatory Guide 1.147 specifies that applicants should make a specific request to the NRC to use Code Cases endorsed in the regulatory guide. The next revision of Regulatory Guide 1.147 (i.e., Revision 6 will reflect the proposed change in Footnote 6 of the regulation. It will permit the use of Code Cases endorsed in the regulatory guide without a specific request to the NRC for approval. In the interim, it is the intent of the NRC that Code Cases listed in Regulatory Guide 1.147 be used without specific application to the NRC. This amendment further revises Footnote 6 to correct the referenced titles for Regulatory Guides 1.84 and 1.85.

Section 50.55a(g) provides requirements for selecting the edition and addenda of Section XI to be complied with during the preservice inspection (Section 50.55a(g)(3), for plants whose construction permit was issued on or after July 1, 1974); the initial 10-year inspection interval (Section 50.55a(g)(4)(i)); and successive 10-year inspection intervals (Section 50.55a(9)(4)(ii). Paragraph IWA-2400 of Section XI, as revised by the Winter 1983 Addenda, incorporates rules for selecting the applicable edition and addenda of Section XI during the preservice inspection (IWA-2411); the initial 10-year inspection interval (IWA-2412); and successive 10-year inspection intervals (IWA-2413).

The criteria provided in the regulations and Section XI are effectively the same for the preservice inspection, and the successive

10-year inspection intervals, but differ for the initial 10-year inspection interval. For the initial 10-year inspection interval, the regulations specify that inservice examinations of components and inservice tests shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference on the date 12 months prior to the date of issuance of the operating license while Section XI provides that the inspection plan shall comply with the Edition and Addenda of Section XI that has been adopted by the regulatory authority 36 months after the date of issuance of the construction permit, or subsequent Editions and Addenda that have been adopted by the regulatory authority. In general, use of the Commission requirements will result in the selection of more recent edition and addenda than will use of the Section XI rules. Satisfying the requirements of Section 50.55a(g)(4)(i) for the initial 10-year inspection interval will in general, also satisfy the rules of Section XI.

It is the Commission's intent that in al cases the existing requirements in Section 50.55a(g) be the basis for selecting the edition and addenda of Section XI to be complied with during the preservice inspection, the 10-year inspection interval, and successive 10-year inspection intervals.

Subsection IWE, "Requirements for Class MC Components of Light-Water Cooled Power Plants," was added to Section XI, Division 1, in the Winter 1981 Addenda. However, 10 CFR 50.55a presently incorporates only those portions of Section XI that address the ISI requirements for Class 1, 2, and 3 components and their supports. The regulation does not currently address the ISI of containments. Since this amendment is only intended to update current regulatory requirements to include the latest ASME Code edition and addenda, the requirements of Subsection IWE would not be imposed upon Commission licensees by this amendment. The applicability of Subsection IWE is being considered separately.

Environmental Impact: Categorical Exclusion

The NRC has determined that this final rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(3). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this final rule.

Regulatory Analysis

The Commission has prepared a regulatory analysis for this amendment to the regulations. The analysis examines the costs anc benefits of the alternatives considered by the Commission. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 1717 H St. NW., Washington, DC. Single copies of the analysis may be obtained from Mr. G.C. Millman, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 492-3872.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget approval number 3150-0011.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule does not have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and operation o 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.

List of Subjects in 10 CFR Part 50

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

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is adopting the following amendments to 10 CFR Part 50.

10 CFR 50.55a

57FR34666

9/8/1992

Added 1986 Addenda to 1989 Edition
III, XI, and Augmented RPV Exams

Statements Of Consideration

57FR34666

Publication_Date: 8/6/92

Effective_Date: 9/8/92

Codes and Standards for Nuclear Power Plants

SUMMARY: The Commission is amending its regulations to incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1; of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, of the ASME Code. The final rule imposes an augmented examination of reactor vessel shell welds and separates the requirements for inservice testing from those for inservice inspection by placing the requirements for inservice testing in a separate paragraph. The ASME Code addenda and edition incorporated by reference provide updated rules for the construction of components of light-water-cooled nuclear power plants, and for the inservice inspection and inservice testing of those components. This final rule permits the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power plant components; requires expedited implementation of the expanded reactor vessel shell weld examinations specified in the 1989 Edition of Section XI; and more clearly distinguishes in the regulations the requirements for inservice testing from those for inservice inspection.

SUPPLEMENTARY INFORMATION

Background

On January 31, 1991 (56 FR 3796), the Nuclear Regulatory Commission published in the Federal Register a proposed amendment to its regulation, 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities," to update the reference to editions and addenda of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). This proposed amendment would revise Section 50.55a to incorporate by reference the 1988 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, of the ASME Code, and the 1988 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, of the ASME Code, with a specified modification. The modification would require implementation of certain requirements for containment isolation valve (CIV) testing that appear in Section XI Subsection IWV prior to the 1988 Addenda, but which do not appear in the later addenda. The amendment would impose an augmented examination of reactor vessel shell welds, and separate in the regulations the requirements for inservice testing from those for inservice inspection by placing the requirements for inservice testing in a separate paragraph.

Summary of Comments

Interested parties were invited to submit written comments for consideration in connection with the proposed amendment by April 16, 1991. Comments were received from 29 separate sources. These sources consisted of 23 utilities, one service organization representing four nuclear power plants, the Nuclear Management and Resources Council (NUMARC), one owners group (BWR Owners Group BWROG)), one state entity (Illinois Department of Nuclear Safety (IDNS)), one public citizens group (Ohio Citizens or Responsible Energy (OCRE)). and one independent consultant.

The submitted comments generally addressed one of the following subject areas: (1) The incorporation by reference of the specified later addenda and edition of Section III, Division 1, and Section XI, Division 1, of the ASME Code into Section 50.55a; (2) the endorsement of comments submitted by NUMARC; 3) the proposed modification to Section XI Subsection IWV rules for CIV testing; 4) the proposed augmented reactor vessel examination; (5) the separation of the rules for inservice inspection and inservice testing; (6) the existing scope Section 50.55a for pump and valve testing; and (7) the potential endorsement in 50.55a of ASME/ANSI OM part 4 on snubbers.

Those who commented on the updating of existing references to Section III and Section XI of the ASME Code in Section 50.55a generally noted their approval. One commentor, however, expressed significant concern with the few provision initially specified in the Section XI 1988 Addenda which expanded the existing requirement to examine one circumferential and one longitudinal reactor vessel shell weld during the 2nd and subsequent inspection intervals to essentially 100 percent of all reactor vessel shell weld during those intervals. Volumetric examination of all reactor vessel shell welds during the first inspection interval has been a requirement in Section XI since the 1975 Addenda. The commentor believes that the expanded examination is unnecessary and that examination efforts should focus on the beltline welds or welds that exceed a specified fluence level. The NRC agrees with the ASME action to expand the reactor vessel examination on the basis that the importance of the reactor vessel, and previous unexpected cracking of primary coolant pressure boundary components, requires that the expanded examinations be performed to ensure the integrity of the reactor vessel. The importance of reactor vessel integrity in protecting the public health and safety demands that periodic, comprehensive inservice examinations of the reactor vessel be made to ensure that structural degradation, if it occurs, does not go undetected. Although the beltline welds do receive the highest radiation, there is simply no assurance that service induced cracking would be limited to those welds. An examination once every ten years of essentially 100

percent of all reactor vessel shell welds is both reasonable and necessary.

The comments submitted by NUMARC relate to: (1) The proposed endorsement of a later edition and addenda of the ASME Code, which NUMARC considers to be a positive step; (2) the proposed modification to Section XI Subsection IWV (i.e., the reference to part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1087 (OM Part 10)), which NUMARC considers to be inappropriate and unnecessary on the basis that 10 CFR part 50, Appendix J testing is adequate; (3) the proposed augmented reactor vessel examination, which NUMARC recognizes to be important, but suggests that more flexibility be incorporated into the implementation provisions; and (4) the scope of Section 50.55a which NUMARC believes should not be influenced by Generic Letter 89-04. Approximately one-half of the utility commentors specifically endorsed the comments by NUMARC. In general, comments from the other utilities were consistent with one or more of the comments from NUMARC. The comments from NUMARC are discussed below, along with comments from others on the same subject.

Most of the comments addressed, in part, the proposed modification to Section XI Subsection IWV rules for containment isolatior valve testing. Utility comments supported the NUMARC comment, which expressed the belief that the current Appendix J containment leakage testing program already provides an adequate basis for assessing and controlling containment leakage and that the modification could result in a valve having to be declared inoperable immediately, in spite of the fact that the total containment leakage may be substantially less than allowable. NUMARC suggested that, in lieu of reinstating requirements for specific valves, NRC recommend to the ASME Operations & Maintenance (O&M) Committee that it perform a comprehensive review of the testing requirements for containment isolation valves and acceptance standards for those tests. IDNS agreed with the NRC position that the requirements for leakage rate analysis and provisions for corrective action should be maintained, but believed that it would be less confusing for licensees if those requirements were incorporated into the existing requirements for Type C testing in Appendix J. OCRE strongly supported the action by NRC to modify the Section XI rules for CIV testing.

The NRC concern that resulted in the proposed modification to Section XI Subsection IWV stems from the findings of two reviews and a follow-on study of Appendix J leak test results. The overall findings show that valve leakage is the primary contributor to occurrences of containment unavailability and that such occurrences generally involve small, rather than large, leaks. Risk to the public from small leakage events is very low, but the NRC is concerned that eliminating the existing Section XI requirement to analyze leakage rates and to take corrective action in the event of abnormally high leakage rates for those CIVs that do not provide a reactor coolant system pressure isolation function could reduce the ability to detect degrading valves and, thereby, could permit an unacceptable reduction in the present safety margin associated with the leak tight integrity of those CIVs and, thereby, the containment.

It was specifically noted in the proposed rule that the NRC was interested in receiving comments on the discussed basis for and content of the proposed modification, and was particularly interested in receiving comments that would provide insight and justification, based upon plant experiences, relative to the need for revising or possibly eliminating the proposed modification. Many comments were received that express concern with the proposed modification. However, these comments, which generally state the opinion that Appendix J requirements are adequate and sufficient with regard to ensuring containment integrity, are of a qualitative nature and no specific plant data or operational experiences were provided or referenced that updated the results of the earlier studies. No additional substantive information was provided for the NRC to consider relative to the need for revising or possibly eliminating the proposed modification. It has not been demonstrated, by analysis of more recent and comprehensive containment leakage test data, that containment leakage integrity can be maintained at an acceptable level without continued implementation of the existing Appendix J valve leak rate test program in conjunction with the Section XI requirement for analysis of leak rates.

Consistent with the comment by NUMARC, the NRC staff discussed the basis for OM part 10 CIV testing requirements with representatives from the ASME O&M Committee. Based on these discussions and in concert with the O&M Committee organization, the O&M Committee has initiated action to (1) perform a comprehensive review of OM part 10 CIV testing requirements and acceptance standards and (2) develop a basis document that would provide, as a minimum, a documented basis for not including the requirements for analysis of leakage rates and corrective actions in OM part 10 for those CIVs that do not provide a reactor coolant system pressure isolation function. The NRC will reevaluate the need for the modification to Section XI Subsection IWV, following review of this basis document. It is anticipated that this will occur as part of a future rulemaking proceeding that will address the incorporation by reference of the ASME O&M Code into Section 50.55a.

In the meantime, this final rule incorporates by reference the 1988 Addenda and 1989 Edition of Section XI, Division 1, with a specified modification for CIV testing that is provided in a new Section 50.55a(b)(2)(vii). The modification substantially preserves the existing requirements for analysis of leakage rates and corrective actions that exist in Subsection IWV prior to the 1988 Addenda. Specifically, the modification requires that licensees implement the requirements of Paragraph 4.2.2.3(e), "Analysis of Leakage Rates," of part 10 and Paragraph 4.2.2.3(f), "Corrective Action," of part 10, in addition to the requirements of Paragraph 4.2.2.2 of part 10, for all Category A valves that are CIVs, regardless of whether or not they provide a reactor coolant system pressure isolation function. Because paragraph 4.2.2.3(e) of part 10 is specified in the modification rather than the existing IWV-3426, the existing Section XI requirement is somewhat relaxed by permitting valve combinations rather than specific valves to be analyzed. This recognizes that, in the past, requests for relief have been granted where design constraints necessitate testing combinations of valves with permissible leak rate limits applied to valve groups. The specified modification does not require the present practice of trending NPS 6 and larger valves because that requirement has not been carried from IWV-3427(b) to OM part 10.

Section XI Subsection IWV (1988 Addenda and 1989 Edition), Subsection IWP (1988 Addenda and 1989 Edition). and Subsection IWF (1987 Addenda, 1988 Addenda, and 1989 Edition) reference ASME/ANSI OM part 10, ASME/ANSI OM part 6, and ASME/ ANSI OM part 4, respectively. During preparation of this final rule, it was recognized that Table IWA-1600-1 in the applicable Section XI addenda and edition specifies a nonexistent revision for OM part 10 and part 6, and does not specifically identify the applicable revision for OM part 4. The Section XI Subcommittee on Inservice Inspection has taken action to correct the revision reference, which, for these standards, should be the ANSI/ASME Oma-1988 Addenda to the ASME/ANSI OM-1987 Edition. To ensure that licensees are aware of the correct revision reference to the OM standards, an additional modification, Section 50.55a(b)(2)(viii), has been added to specify that the Oma-1988 Addenda is the applicable revision to the OM-1987 Edition for OM part 4, part 6, and part 10 when using the noted Section XI addenda and edition.

The NUMARC comment relative to the proposed augmented examination of the reactor vessel indicates an understanding of the NRC position on the need for this examination, but notes concern with the specifics of the proposed implementation. Specifically, NUMARC expresses concern that: (1) Better utilization of available inspection resources could be accomplished by limiting application of the augmented inspection program to the reactor vessel beltline shell welds, or by limiting implementation of the augmented examination to reactor vessel shell welds that exceed a specific neutron flux exposure (this comment differs from the one utility comment noted above relative to updating later edition and addenda of Section XI in that it only refers to the augmented examination); (2) tooling for the older Boiling Water Reactors (BWRs) may generally not be available in the time-frame needed; (3) only those reliefs which address the scope and extent of shell weld examinations should be revoked, and they should be revoked on a plant specific basis; and (4) the NRC should state its willingness to accept requests for specific new exemptions, based on the availability of suitable equipment and technology at the time of the scheduled inspection and the appropriate technical justification.

Other comments on the augmented examination include those from: BWROG, which noted concern for those plants close to the end of the current interval that could not practically incorporate the augmented examination into the current interval and would have to perform that examination during the first period of the next interval (Note: The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination normally scheduled for the interval in which the deferred examination was performed (Section 50.55a(g)(6)(ii)(A)(3), therefore, the impact of deferring the augmented examination will be reduced); IDNS, which strongly supports the NRC position regarding the augmented examination of the reactor vessel; and OCRE, which also strongly supports the augmented examination and notes that the examination will not only provide an additional assurance of safety, but will aid in understanding aging degradation phenomena which will assist licensees that wish tc pursue license renewal.

The NRC position with regard to the augmented examination of the reactor vessel, as previously stated in the Supplementary Information to the proposed rule, is that degradation of reactor vessel materials has become more of a concern recently, because: (1) Results from irradiation surveillance material tests show that certain reactor vessel materials undergo greater radiation damage than previously expected, (2) indications from operational data show that stress corrosion cracking of BWR reactor vessels is more probable than was thought several years ago, and (3) significant service induced cracking has occurred in large vessels (i.e., pressurizer, steam generators) designed and fabricated to the ASME Code. It is the judgment of the NRC that, because of new information and previous limited examinations of reactor vessels, there may exist a substantially greater potential for reactor vessel degradation, in all areas of the reactor vessel, than previously considered and that maintenance of the level of protection presumed by the regulations requires more than compliance to existing regulatory requirements. The NRC has determined that the augmented examination of reactor vessels will result in a substantial increase in the overall protection of the public health and safety, and that the costs of implementation are justified in view of the increased protection. The backfit analysis required by Section 50.109, "Backfitting," is provided as part of the regulatory analysis that supports this final rule.

However, the NRC agrees with comments that additional flexibility and specificity will improve implementation of the augmented examination of reactor vessel examination. To this end, the augmented examination of reactor vessel shell welds specified in this final rule includes the following new provisions and clarifications: (1) The revocation of previously granted reliefs is limited to those reliefs that deal with the extent of volumetric examination of reactor vessel shell welds; (2) the augmented examination will be performed in accordance with the section XI edition and addenda applicable to the inspection interval in which the examination is actually performed; (3) "essentially 100%" as used in Section 50.55a(g)(6)(ii)(A) means "more than 90 percent of the examination volume of each weld, where the reduction in examination volume is due to interference from another component, or part geometry;" (4) licensees that defer the augmented examination to the next interval are permitted to retain all existing approved reliefs for the current interval; (5) licensees with fewer than 40 months remaining in the inspection interval in effect when the rule becomes effective are permitted to extend the interval in accordance with the provisions of Section XI (1989 Edition) IWA-2430(d); (6) licensees that are unable to satisfy completely the requirements for the augmented examination may request to perform alternate examinations in accordance with Section 50.55a(g)(6)(ii)(A)(5). These items are addressed individually in the discussion below regarding provisions of the augmented reactor vessel shell weld examination.

Section 50.55a(g)(6)(ii) addresses augmented inservice inspection programs for those systems and components for which the Commission determines that added assurance of structural reliability is necessary. For that purpose, and consisent with the discussion in this final rule, Section 50.55a(g)(6)(ii)(A) has been added to require expedited implementation of the reactor vessel shell weld examinations specified in the 1989 Edition of section XI. Division 1, in item B1.10, "Shell Welds," of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table 2500-1 of subsection IWB, "Requirements for Class 1 Components of Light-Water Cooled Power Plants."

In order to ensure the applicability of the new augmented examination to all licensees, Section 50.55a(g)(6)(ii)(A)(1) revokes all previously granted reliefs relating to the extent of volumetric examination of the reactor vessel shell welds that apply to examinations for the inservice inspection interval that is in effect when the rule becomes effective subject to a specified modification. Limiting the revocation of previously granted reliefs to those that deal with the extent of the volumetric examination permits the retention of those approved reliefs that deal with issues such as specification of calibration blocks. Licensees that choose to defer the augmented examination to the next interval in accordance with Section 50.55a(g)(6)(ii)(A)(3) should note that paragraph (iv) of that section modifies the revocation of approved reliefs to permit retention of previously approved reliefs for the current interval when the augmented examination in deferred. This provision recognizes that plants that previously received relief from the section XI reactor vessel shell weld examination and satisfy the condition to defer the augmented examination may find it impractical to implement the section XI examination during the current inspection interval.

Section 50.55a(g)(6)(ii)(A)(2) requires all licensees to implement the specified augmented examination of reactor vessels during the inspection interval in effect when this rule becomes effective. subject to conditions specified in Section 50.55a(g)(6)(ii)(A)(3) and (4). Section 50.55a(g)(6)(ii)(A)(2) specifically permits the use of the augmented examination, when not deferred, as a substitute for the reactor vessel shell weld examinations scheduled for the inspection interval in effect when the rule becomes effective, and specifies that, for the purpose of this rule, "essentially 100 percent" as used in Table IWB-2500-1 means "more than 90 percent of the examination volume of each weld, where the reduced examination volume is due to interference from another component, or part geometry." This is consistent with section XI Code Case N-480, which previously has been approved for use in Regulatory Guide 1.147. It is recognized that it may be necessary to implement a combination of internal and external diameter examinations to achieve "essentially 100%" examination volume coverage for each weld. A clarification has been included in this section to note that the augmented examination may be used as a substitute for the reactor vessel shell weld examination in the interval in effect when the rule becomes effective when the augmented examination is not deferred. This is a reinforcement of Section 50.55a(g)(6)(ii)(A)(3), as it appears in both the proposed and final rule, which specifies that the deferred examination may not be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect when the rule becomes effective.

The NRC recognizes that plants with fewer than 40 months remaining in the inspection interval when this rule becomes effective may find it impractical to implement the augmented examination of the reactor vessel during that inspection interval. Therefore, Section 50.55a(g)(6)(ii)(A)(3) permits plants with fewer than 40 months remaining in the inspection interval when this rule becomes effective to defer the augmented examination until the first period of the next inspection interval. However, this same paragraph specifically prohibits the use of the deferred augmented examination as a substitute for reactor vessel shell weld examinations scheduled for the inspection interval in effect when the rule becomes effective. The intent is to ensure that the examinations are deferred only when necessary and not to have the rule encourage a 40-month delay in reactor vessel shell weld examinations. Further, Section 50.55a(g)(6)(ii)(A)(3) permits using the deferred examination, with a condition, as a substitute for reactor vessel shell weld examinations scheduled for the inspection interval in which the deferred examinations are performed. The condition is that subsequent reactor vessel shell weld examinations for successive inspection intervals be performed in the first period of the inspection interval. This condition is necessary to prevent a potential 160-month gap between reactor vessel shell weld examinations. This gap would occur if a plant used the deferred examination performed in the first period as a substitute for the scheduled examination and then deferred the examination for the next inspection interval to the end of that interval as permitted by section XI. In addition, this section specifies that licensees with fewer than 40 months remaining ir the inservice inspection interval in effect when the rule becomes effective may extend that interval in accordance with the provisions of section XI (1989 Edition) IWA-2430(d) to permit implementation of the augmented examination during the current interval. It is not the intent of the NRC to permit licensees in the second period of an inspection interval to reduce the interval length for the purpose of "being within 40 months of the end of the interval" and, thereby, deferring the augmented examination to the first period of the subsequent interval.

Section 50.55a(g)(6)(ii)(A)(4) specifies that a licensee that has either completed or has scheduled an inspection of essentially 100 percent of the length of all Examination Category B—A shell welds during the inservice inspection interval in effect when the rule becomes effective does not have to implement the required augmented examination of the reactor vessel shell welds. Primarily, this paragraph is intended to permit licensees who are in the first inspection interval to use the essentially 100 percent reactor vessel shell weld examination required for that interval by section XI to satisfy the requirement for the augmented examination of the reactor vessel. The technical objective of the augmented examination will be accomplished under these conditions. These licensees will continue to apply the current requirements of Section 50.55a(g)(4) until the next inspection interval when future examinations will be performed based on ASME section XI, 1989 Edition, or later Code edition and addenda specified in Section 50.55a(b).

The augmented examination specified in § 50.55a(g)(6)(ii)(A) is not an ASME Code requirement. It is a requirement specifically developed and additionally imposed by the Commission. Therefore, except for the specific provisions in Section 50.55a(g)(6)(ii)(A)(2) and (3) that permit using the augmented examination as a substitute for section XI required reactor vessel shell weld examinations, the closing out of an inservice inspection interval is not dependent on completion of the augmented examination. In the specific instance where the augmented examination is deferred to the first period of the next inspection interval, the current inspection interval could be closed out relative to reactor vessel shell weld examinations by implementing the regularly scheduled reactor vessel shell weld examinations as modified by previously approved applicable relief requests for that interval.

The NRC recognizes that, as noted by commentors, there may exist conditions that prevent licensees from completely satisfying

the requirements for the augmented reactor vessel shell weld examination as specified in Section 50.55a(g)(6)(ii)(A). For this reason, Section 50.55a(g)(6)(ii)(A)(5) has been added to permit licensees that make a determination that they are unable to completely satisfy the specified augmented examination to propose and use alternatives that have been authorized by the NRC's Director of the Office of Nuclear Reactor Regulation.

This final rule amends Section 50.55a to separate the requirements for inservice testing from those for inservice inspection by moving the requirements for inservice testing to a separate paragraph. Previously, Section 50.55a(g), "Inservice inspection requirements," specified the requirements for (1) preservice and inservice examinations for Class 1, Class 2, and Class 3 components and their supports, (2) system pressure tests for Class 1, Class 2, and Class 3 components, and (3) inservice testing of Class 1, Class 2, and Class 3 pumps and valves. In order to emphasize the importance of inservice testing and to distinguish more clearly its requirements from those of inservice inspection, this final rule moves the requirement for inservice testing from Section 50.55a(g), "Inservice inspection requirements," to a separate (previously reserved) Section 50.55a(f), which is titled "Inservice testing requirements." All existing requirements for inservice examination and system pressure testing are retained in Section 50.55a(g).

There is overall favorable acceptance of the separation of the requirements in the regulation for inservice testing and for inservice inspection. It is generally believed by the commentors, as it is believed by the NRC, that the separation serves to clarify and emphasize the requirements for inservice testing. Two administrative changes were made in the development of Section 50.55a(f) relative to existing Section 50.55a(g). First, Section 50.55a(f)(6)(ii) has been added to indicate the Commissions intent to impose an augmented inservice testing program if added assurance of operational readiness is deemed necessary. This paragraph only indicates intent and does not impose a specific requirement. It does parallel the existing Section 50.55a(g)(6)(ii) which specifies that the Commission may require an augmented inservice inspection program for systems and components for which it deems that added assurance of structural reliability is necessary. One utility commentor expressed concern that the addition of Section 50.55a(f)(6)(ii) would permit the Commission to impose an augmented inservice testing program without further justification. This is not the case. Any program for augmented inservice testing will be fully justified with a documented regulatory analysis that includes the appropriate backfit analysis. The intent of the NRC to perform the necessary backfit analysis is clearly demonstrated by the backfit analysis that was performed to require the augmented examination of the reactor vessel that is specified in Section 50.55a(g)(6)(ii)(A) of this final rule.

Second, this final rule includes the addition of introductory text to Section 50.55a(g) which states that the requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in Section 50.55a(f). This change is necessary because the placement of inservice testing requirements into a separate Section 50.55a(f), as included in the proposed rule, would have caused administrative inconsistencies with regard to existing references to Section 50.55a(g) for inservice testing in documents such as technical specifications, safety analysis reports, procedures, and records. With this change. existing references to Section 50.55a(g) for inservice testing will refer the user to Section 50.55a(f), where the specific requirements for inservice testing are located. The NRC recommends that as the governing documents are updated. the direct reference to Section 50.55a(f) be incorporated, as appropriate.

Two editorial revisions, relative to the previous Section 50.55a(g), are included in the new Section 50.55a(f). These editorial revisions: (1) Reserve Section 50.55a(f)(3)(i) and (ii) so that the structure of Section 50.55a(f) will parallel that of Section 50.55a(g) for the purpose of promoting easier cross-referencing between the two paragraphs; and (2) modify the reference to 120-month inspection interval in Section 50.55a(g) to 120-month interval in Section 50.55a(f), because the term "inspection interval," as used in Section XI, is used only in the context of inservice inspection. (The term "test interval" was not used because, unlike inspection interval, the 120-month time frame does not designate a period of required actions for the testing program. The 120-month interval used in Section 50.55a(f) and the 120-month inspection interval used in Section 50.55a(g) are considered by the staff to be coincident for the purpose of 120-month updating requirements.)

A number of comments were received regarding the scope of Section 50.55a as applied to pump and valve testing. These comments ranged from recommending that the scope of Section 50.55a be expanded to be consistent with the scopes of OM part 6 and part 10 which go beyond Class 1, Class 2, and Class 3 components, to recommending that the scope of Section 50.55a be limited to ASME Code classified components. One commentor expressed concern that the Supplementary Information in the proposed rulemaking addressed Generic Letter 89-04 in a way that seemed to include the letter in the rulemaking. That was not intended. To the contrary, the intent of this rulemaking is to maintain the existing scope of Section 50.55a for pump and valve testing. For plants whose construction permits were issued on or after January 1, 1971, that scope constitutes Code classified components as specified in existing Section 50.55a(g)(2) and (3) (i.e., Section 50.55a(f)(2) and (3) by this rulemaking). For those plants whose construction permits were issued prior to January 1, 1971, that scope constitutes components of the reactor coolant pressure boundary which must meet the requirements applicable to components that are classified as ASME Code Class 1, and other safety-related pumps and valves which must meet the requirements applicable to components that are classified as ASME Code Class 2 or Class 3, as specified in existing Section 50.55a(g)(1) (i.e., Section 50.55a(f)(1) by this rulemaking). The reference to the generic letter has not been included in the final rule.

A number of comments were received with regard to snubber testing which is outside the scope of this rulemaking. Commentors generally suggested that ASME OM part 4, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)," which is referenced in Subsection IWF in the 1987 Addenda, 1988 Addenda and 1989 Edition of Section XI, be incorporated by reference into Section 50.55a. Subsection IWF, "Component Supports," provides rules for the examination of component supports, and the testing of snubbers. Prior to the 1987 Addenda, Subsection IWF provided self-contained rules for

the testing of snubbers. Section 50.55a does not specify requirements for the testing of snubbers. This was clarified by the separation of requirements for inservice testing and inservice inspection. Inservice testing requirements specified in Section 50.55a(f) apply only to pumps and valves. The testing requirements specified in OM part 4 and referenced in Section XI Subsection IWF article IWF-5000 are not incorporated by reference into Section 50.55a. Requirements for the testing of snubbers are generally governed by plant technical specifications. NRC is in the process of initiating a proposed rulemaking that would, among other things, address the incorporation by reference of the ASME OM Code, which contains rules for pump. valve, and snubber testing, into Section 50.55a(f). The NRC will as a part of this future rulemaking determine the need for and acceptability of endorsing the ASME OM Code rules for snubber testing. However, in accordance with requirements for examination of component supports specified in Section 50.55a(g), licensees are required to implement the rules for examination of snubbers that are provided in OM part 4 as referenced in Subsection IWF Article IWF-5000 in the applicable Section XI addenda and edition of this final rule.

Section 50.55a(g) provides requirements for selecting the ASME Code edition and addenda of Section XI to be complied with during the preservice inspection (Section 50.55a(g)(3), for plants whose construction permit was issued on or after July 1, 1974); the initial 10-year inspection interval (Section 50.55a(g)(4)(i)); and successive 10-year inspection intervals (Section 50.55a(g)(4)(ii)). As noted in the final rule codifying the most recent amendment to Section 50.55a (May 5, 1988; 53 FR 16051), paragraph IWA-2400 of Section XI (as revised by the Winter 1983 Addenda) incorporated rules for selecting the applicable edition and addenda of Section XI during the preservice inspection (IWA-2411). the initial 10-year inspection interval (IWA-2412), and successive 10-year inspection intervals (IWA-2413). The criteria provided in the regulations and Section XI are effectively the same for the preservice inspection and the successive 10-year inspection intervals, but differ for the initial 10-year inspection interval. In general, use of the Commission requirements will result in the selection of a more recent edition and addenda than will use of the Section XI rules. Satisfying the requirements of Section 50.55a(g)(4)(i) for the initial 10-year inspection interval will, in general, also satisfy the rules of Section XI. Although the Section XI requirements for selecting editions and addenda remain unchanged in the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition, the Commission is reaffirming its intent that in all cases the existing requirements in Section 50.55a(g) be the basis for selecting the edition and addenda of Section XI to be complied with during the preservice inspection, the initial 10-year inspection interval, and the successive 10-year inspection intervals.

This final rule makes a number of editorial changes to Section 50.55a for the purpose of adopting a standard convention for imposing an obligation or expressing a prohibition. In this convention "shall" is used to impose an obligation on an individual or legal entity capable of performing the required action, "must" is used as the mandatory form when the subject of the sentence is an inanimate object, and "may not" is used to impose a prohibition. The following paragraphs are amended solely to be consistent with this convention: The introductory paragraph to the section; paragraphs (a)(1), (a)(3), (b)(2)(iii), (b)(2)(iv), (g)(1), (g)(3)(ii), (g)(3)(iii), (g)(3)(iv), introductory paragraph to (g)(4), (g)(4)(i), (g)(4)(ii), (g)(5)(i), (g)(5)(iv), (g)(6)(i), (h), and footnote 8. Other paragraphs are amended for the same editorial reason, but they also contain technical revisions relevant to other parts of this final rule. Section 50.55a(f) has been developed consistent with the noted convention.

Subsection IWE, "Requirements for Class MC Components of Light-Water-Cooled Power Plants," was added to Section XI, Division 1, in the Winter 1981 Addenda. Since Section 50.55a does not currently address the inservice inspection of containments and the scope of Section 50.55a is not affected by this final rule, the requirements of Subsection IWE are not imposed upon Commission licensees by this amendment. The incorporation by reference of Subsection IWE into Section 50.55a is presently the subject of a separate rulemaking action. Section 50.55a(b)(2)(vi) is reserved for that action.

The NRC previously alerted all holders of operating licenses or construction permits for nuclear power reactors, through NRC Information Notice No. 88-95 (IN 88-95), "Inadequate Procurement Requirements Imposed by Licensees on Vendors," to the potential that inadequate licensee procurement requirements or implementation by vendors in supplying components under the ASME Code could result in failure by these vendors to fully implement 10 CFR part 50. Appendix B (Quality Assurance Criteria). The problem, which was revealed during routine NRC inspections of vendors, resulted from the belief by some vendors that if an item was exempted by the ASME Code from Code requirements, the item was exempt from all other regulatory requirements. The apparent belief of some vendors was that since NRC endorses the ASME Code in its regulations and has accepted the various exemptions, there are, therefore, no other applicable regulatory requirements. This belief is not consistent with the NRC position. The NRC reaffirms its position which, as previously put forth in IN 88-95, states that all safety-related items, even those exempted from ASME Code requirements, are required to be manufactured under a quality assurance program that meets the requirements of 10 CFR part 50, appendix B.

Finding of No Significant Environmental Impact: Availability

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 is not a major Federal action that significantly affects the quality of the human environment and therefore an environmental impact statement is not required.

This final rule is one part of a regulatory framework directed to ensuring pressure vessel integrity, and the operational readiness of pumps and valves. Therefore, in the general sense, this rule will have a positive impact on the environment. This rule incorporates by reference into the NRC regulations improved rules contained in the ASME Code for the construction, inservice inspection, and inservice testing of components used in nuclear power plants. In addition, this rule requires an augmented examination of reactor vessel shell welds to further ensure the structural integrity of the reactor vessel. The occupational

exposures attributable to the expanded reactor vessel examinations contained in the ASME Code and the augmented examination are not expected to be significant because exposures will be limited by the use of remote examination equipment. Occupational exposures associated with the augmented reactor vessel examination will be further limited by provisions in the final rule that permit, under certain conditions, the licensee to satisfy the requirement for the augmented examination by previously scheduled or implemented reactor vessel examinations, or by deferring the examination to the next interval and using the deferred examination as a replacement for the previously scheduled examination for that interval. The actions required by applicants and licensees to implement the final rule are of an established 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 Gilbert C. Millman, Division of Engineering, Office of Nuclear Regulatory Research. U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-3848.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget approval number 3150-0011.

The public reporting burden for this collection of information is estimated to average 42 hours per response, 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, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

The Commission has prepared a regulatory analysis for this amendment to the regulations. The analysis examines the costs and benefits of the alternatives considered by the Commission. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis may be obtained from Mr. G.C. Millman, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 492-3848.

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 have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and 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 rule does not fall within the purview of the Act.

Backfit Analysis

The final rule incorporates by reference a later edition and addenda to Section III, Division 1, and, with both a technical and nontechnical modification, Section XI, Division 1, of the ASME Code; imposes an augmented examination on reactor vessels; and separates the requirements for inservice inspection from those for inservice testing.

The incorporation by reference into the regulations of later editions and addenda of Section III and Section XI of the ASME Code is not a backfit because Section III requirements apply only to new construction, except as voluntarily implemented by licensees, and because updated Section XI requirements are an integral part of the longstanding Section 50.55a(g)(4)(ii) requirement to update inservice inspection and inservice testing programs to reflect the requirements of the latest edition and addenda of Section XI incorporated by reference in Section 50.55a(b) 12 months prior to the start of the 120-month inspection interval, subject to specified limitations and modifications. The technical modification to part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987 specified in Section 50.55a(b)(2)(vii) is not a backfit because it simply retains an existing Section XI requirement for containment isolation valve testing that licensees now are required to implement in accordance with Section 50.55a(g). The nontechnical modification specified in Section 50.55a(b)(2)(viii) is not a backfit because it only serves to properly identify an incorrectly referenced standard in Section XI.

The NRC has concluded, based on the analysis required by Section 50.109(a)(3) which is provided in the regulatory analysis, that the backfit that will be imposed by the augmented reactor vessel examination specified in Section 50.55a(g)(6)(ii)(A) will result in a substantial increase in the overall protection of the public health and safety, and that the direct and indirect costs of implementation are justified in view of the increased protection.

The separation in the regulation of the inservice inspection and inservice testing requirements is an administrative reorganization of Section 50.55a that has no impact on existing technical requirements and, therefore, has no effect on backfitting.

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. 552 and 553, the NRC is adopting the following amendments to 10 CFR part 50.

10 CFR 50.55a

61FR41303

9/9/1996

Added IWE & IWL

(2) Multiplying the result by the number of tons of such citrus fruit. The applicable price for undamaged citrus fruit will be the local market price the week before damage occurred.

(f) Any production will be considered marketed or marketable as fresh fruit unless, due solely to insured causes, such production was not marketed as fresh fruit.

(g) In the absence of acceptable records of disposition of harvested citrus fruit, the disposition and amount of production to count for the unit will be the guarantee on the unit.

(h) Any citrus fruit on the ground that is not harvested will be considered totally lost if damaged by an insured cause.

13. Written Agreements

Designated terms of this policy may be altered by written agreement in accordance with the following:

(a) You must apply in writing for each written agreement no later than the sales closing date, except as provided in section (13)(e);

(b) The application for written agreement must contain all terms of the contract between you and us that will be in effect if the written agreement is not approved;

(c) If approved, the written agreement will include all variable terms of the contract, including, but not limited to, crop type or variety, the guarantee, premium rate, and price election;

(d) Each written agreement will only be valid for one year (If the written agreement is not specifically renewed the following year, insurance coverage for subsequent crop years will be in accordance with the printed policy); and

(e) An application for written agreement submitted after the sales closing date may be approved if, after a physical inspection of the acreage, it is determined that no loss has occurred and the crop is insurable in accordance with the policy and written agreement provisions.

Signed in Washington, DC, on August 2, 1996.

Kenneth D. Ackerman,

Manager, Federal Crop Insurance Corporation.

[FR Doc. 96–20195 Filed 8–7–96; 8:45 am]

BILLING CODE 3410–FA–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: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending 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 will be required to incorporate Subsection IWE and Subsection IWL into their inservice inspection (ISI) program. Licensees will also be required to expedite implementation of the containment examinations and to 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 included that will 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. The final rule specifies requirements to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's structural integrity.

EFFECTIVE DATE: September 9, 1996. The incorporation by reference of certain publications listed in the regulations is approved by the Office of the Director of the Office of the Federal Register as of September 9, 1996.

FOR FURTHER INFORMATION CONTACT: Mr. W. E. Norris, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415–6796.

SUPPLEMENTARY INFORMATION: The NRC is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's structural integrity. The rate of occurrence of degradation in containments is increasing. Appendix J to 10 CFR part 50 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 examinations. The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. There have been 32 reported occurrences of corrosion in metal containments and the liners of concrete containments. This is one-fourth of all operating nuclear power plants. Only four of the 32 occurrences were detected by current containment inspection programs. Nine of these occurrences were first identified by the NRC through its inspections or structural audits. Eleven occurrences were detected by licensees after they were alerted to a degraded condition at another site or through activity other than containment inspection. There have been 34 reported occurrences of degradation of the concrete or of the post-tensioning systems of concrete containments. This is nearly one-half of these types of containments. It is clear that current licensee containment inspection programs have not proved to be adequate to detect the types of degradation which have been reported. 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 reduction to below the minimum design thickness; (2) corrosion of the torus of the steel containment shell (wall thickness below minimum design thickness); (3) corrosion of the liner of a concrete containment to approximately half-depth; (4) grease leakage from the tendons of prestressed concrete containments; and (5) leaching as well as excessive cracking in concrete containments.

There are several General Design Criteria (GDC) 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. The GDC serve as fundamental underpinnings for many of the most safety important commitments in

licensee design and licensing bases. 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 III. 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).

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 that thickness that would ensure that the metal containment structure would continue to maintain its structural integrity under the various stressors and degradation mechanisms which could act on it.

The prestressed concrete containments of most operating nuclear reactors were designed in accordance with ACI-318 provisions taking into consideration their unique features in the design of the post-tensioning system and in 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. The liners of concrete containments provide a leak-tight barrier.

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).

None of the existing requirements, however, provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation with regard to the performance and the effectiveness of licensee containment examination programs. 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 been found to have undergone a significant level of degradation that was not detected by these programs.

The Nuclear Management and Resources Council (NUMARC) (which has since become the Nuclear Energy Institute (NEI)) developed a number of industry reports to address license renewal issues. Two of those, one for Pressurized Water Reactor (PWR) containments and the other for Boiling Water Reactor (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 NRC agrees with NEI that these ASME standards, which the industry has participated in developing, would be an effective means for managing age-related containment degradation. Thus, the NRC believes that adoption of these standards is the best approach.

Background

On January 7, 1994 (59 FR 979), the NRC published in the Federal Register a proposed amendment to its regulation, 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities," to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, and Subsection IWL, of Section XI, Division 1, of the ASME Code with specified modifications and a limitation.

Five modifications were specified in the proposed rule to address two concerns of the NRC. The first concern is that four recommendations for tendon examinations that are included in Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," Rev. 3, are not addressed in Subsection IWL (this involves four of the modifications, (§ 50.55a(b)(2)(ix)(A)-(D)). Regulatory Guide 1.35, Rev. 3, describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for ungrouted tendons in prestressed concrete containment structures. The four recommendations contained in Regulatory Guide 1.35, Rev. 3, which are not addressed by Subsection IWL, provide positions on issues such as failed wires and tendon sheathing filler grease conditions. (The ASME Code has considered the four issues involved and is in the process of adopting them into addenda of Subsection IWL). The second NRC 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. The fifth modification (§ 50.55a(b)(2)(ix)(E)) requires that inaccessible areas be evaluated when visible conditions exist that suggest the possibility of degradation of these areas.

The limitation which was included in the proposed rule specified the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL as the earliest version of the ASME Code the NRC finds acceptable. This is because this is the first edition including addenda combination acceptable to the NRC staff that incorporates the concept of base metal examinations and also provides a

comprehensive set of rules for the examination of post-tensioning systems. As originally published in 1981, 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, have promoted the incorporation of base metal examinations.

The proposed rulemaking incorporated a provision for an expedited examination schedule. This expedited examination schedule is necessary to prevent the delay in implementation of Subsection IWE and Subsection IWL (the Summary of Documented Evaluation lists each plant and the delay in implementation which would be encountered if the subsections were implemented through routine updates of the ISI programs). Provisions were incorporated in the proposed rule to ensure that the expedited examination which would be completed within 5 years from the effective date of the rule and the routine 120-month examinations did not duplicate examinations.

On March 4, 1994, the NRC received a request from the Nuclear Management and Resources Council (which has since become part of the Nuclear Energy Institute (NEI)) to extend the public comment period from March 23, 1994 until April 25, 1994, to enable NEI to "provide necessary and constructive comments on the proposed rule change." This was granted, and on March 28, 1994 (59 FR 14373), the NRC published in the Federal Register a notice of extension of the public comment period.

Summary of Comments

Comments were received from 25 separate sources. These sources consisted of 15 utilities, one service organization (Entergy Operations, Inc.) representing five nuclear plants, the Nuclear Energy Institute (NEI), the Nuclear Utility Backfitting and Reform Group (NUBARG) represented by the firm of Winston & Strawn, one owner's group (BWR Owner's Group (BWROG)), one architect and engineering firm (Stone & Webster Engineering Corporation), one public citizens group (Ohio Citizens for Responsible Energy (OCRE)), three individuals, and one consulting firm (VSL Corporation).

Comments received could be divided into three groups. The first group contains those comments which address the administrative aspects of the rule (e.g., backfit considerations, effectiveness of current containment examinations), and the modifications specified by the NRC in the proposed rule. The second and third groups contain those comments which address the technical provisions of Subsection IWE, and Subsection IWL, respectively. The summary and resolution of public comments and all of the verbatim comments which were received (grouped by subject area) are contained in the Summary of Documented Evaluation.

The majority of comments generally addressed one of the following subject areas: (1) The incorporation by reference of Subsection IWE and Subsection IWL into § 50.55a; (2) the development of guidance documents instead of regulatory requirements; (3) the rationale for the proposed backfit; (4) endorsement of the BWROG comments; and (5) the 5-year expedited implementation. These subject areas encompass the comments submitted by NEI and NUBARG, and their comments, if any, are discussed separately in each subject area.

The comments on subject area number one from those that approve of the incorporation by reference of Subsection IWE and Subsection IWL into § 50.55a, can be summarized as follows: (1) There is a need for the periodic examination of containment structures to assure the containment's pressure-retaining and leak-tight capability; (2) Section XI requirements define concise, technically sound programs to assure continuing containment integrity; and (3) input in the development of these rules was provided by all interested parties involved in containment inservice inspection—users, regulators, manufacturers, engineering organizations, and enforcement organizations.

The comments on the other four subject areas are summarized below. The resolution of public comments contains all of the comments which were received. Some of the comments resulted in modifications to the rule, and some of the comments have been transmitted to the ASME for their consideration. A discussion of the comments which led to modifications follows the summary of comments on subject area number five. The resolution of public comments package contains those comments transmitted to the ASME. Those comments asked for interpretations of the ASME Code rules.

Regarding subject area number two, eleven commenters believe that additional specific guidance in the form of a guidance document would be more appropriate than a regulation. They concur with NEI that current regulatory requirements for containment integrity and examinations are already provided by existing regulations (GDC 16 and 53 and Appendix J) and licensee commitments. If more detail on how to perform containment examinations is needed, the commenters (including NEI) state that the details could be provided in a regulatory guide, Information Notice, Generic Letter, or in an industry developed guidance document. The NRC does not believe that existing regulations and licensee commitments are adequate. Existing regulations and licensee commitments have not proved to be adequate to detect the types of problems which have been experienced in operating reactors. This is evidenced by the large number of instances of degradation that 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. Licensee containment inspection programs have generally not detected the types of degradation being reported (only four of the 32 reported instances of corrosion in Class MC containments were discovered as a result of the Appendix J general inspection). Further, the NRC does not believe that providing guidance through a regulatory guide or industry report would generally improve containment examination practices. Licensees were made aware of containment degradation through several industry notices, and yet the staff is still detecting many of occurrences of degradation. The increasing rate of occurrence of containment degradation, the number of occurrences, the extent to which some containments were degraded, the high number of instances discovered through NRC inspections or by licensees because they were alerted to a degradation condition at another site, the time-dependent mechanisms, and the results of the survey performed by the NRC Regional Offices regarding current containment inspections all point to the necessity of imposing additional requirements to ensure that containments comply with design wall thicknesses and prestressing forces. This is a compliance backfit.

With regard to subject area number three, six general comments were received from the Nuclear Utility Backfitting and Reform Group (NUBARG) and from the Nuclear Energy

Institute (NEI) (which were endorsed by other commenters) regarding the incorporation by reference of Subsection IWE and Subsection IWL which are similar in nature. The first comment is that the application of the compliance exception to this rulemaking is inappropriate, and that the proposed rule constitutes a backfit for which a cost-benefit analysis should be performed. The NRC agrees that the rulemaking is a backfit. However, as discussed under the Backfit Statement, the NRC believes that the compliance exception to the backfit rule is appropriate.

The second comment was a citation of a paragraph from the Statement of Considerations to the 1985 final backfit rule which addressed the compliance exception. That paragraph addressed "Section 50.109(a)(4) which creates exceptions for modifications necessary to bring a facility into compliance or to ensure through immediately effective regulatory action that a licensee meets a standard of no undue risk to public health and safety." Both NEI and NUBARG assert that the proposed rule is a new interpretation of how to demonstrate compliance with existing standards and therefore constitutes a backfit under 10 CFR 50.109(a)(1). The NRC does not believe that the use of the compliance exception must be confined only to the situation addressed in the Statement of Consideration to the 1985 final backfit rule—"omission or mistake of fact." In any event, the current unsatisfactory status of containment inservice inspections can be characterized fairly as, in retrospect, a mistake about and omission from the necessary elements of a satisfactory inspection program.

The third comment is that containments must experience corrosion or degradation that is so unanticipated and excessive so as to constitute a genuine compliance concern. Another commenter expressed the idea somewhat differently believing that a broad-based concern with the operability of containment structures through the industry must be demonstrated to be a compliance issue. The NRC agrees with those criteria and concludes, in fact, that there is a broad-based concern regarding the structural integrity of containment structures. The NRC's approach focuses on two questions: (1) Is the corrosion such that there is a basis for reasonably concluding that additional instances of noncompliance with the relevant GDCs, Appendix J, and/or licensee commitments at numerous plants; and (2) whether there is a basis for reasonably believing that the corrosion would have been identified and properly addressed by the licensees in the absence of additional regulatory requirements. Based on the: (1) Number of occurrences of containment degradation; (2) increasing rate of containment degradation; (3) locations of the degradation; (4) two instances where containment wall thicknesses were below minimum design wall thickness; (5) number of corrosion paths which have been reported; and (6) higher than anticipated corrosion rates in many of the occurrences, the NRC believes that containments are experiencing corrosion or degradation that is unanticipated and excessive. Further, based upon factors (1) to (6) above, the NRC concludes that additional criteria are necessary to ensure that compliance with existing requirements for minimum accepted design wall thicknesses and prestressing forces are maintained (and thereby the ability of the containment to continue to perform its intended safety function).

The fourth comment by NUBARG and NEI suggested that it is part of the anticipated process for the industry to rely upon NRC inspections and audits to identify problems and then alert the industry through NRC documents such as information notices and generic letters. During the presentation to the ACRS on February 10, 1995, NEI asserted that "[i]t really doesn't matter how the utilities identify these instances of degradation." The NRC believes that inspections conducted by licensees should be adequate to ensure that containment degradation is identified without reliance upon NRC inspections.

The fifth NEI and NUBARG comment is that to ensure compliance the NRC could take individual enforcement action rather than endorse ASME standards. The NRC believes that the best approach is to adopt the industry consensus standards (i.e., endorse ASME Section XI Subsection IWE and Subsection IWL). Containment corrosion and degradation have been reported since 1986. The patterns of degradation and the corrective actions were not immediately obvious. Given the number and the extent of the occurrences, and the variability among plants with regard to the performance and the effectiveness of containment inspections, the NRC believes that the best course of action is to endorse ISI requirements to ensure that containments comply with design wall thicknesses and prestressing forces.

The sixth comment is that GDC 16 required containments to be designed and constructed with an allowance for corrosion or degradation of the containment wall over the projected design life of the plant. NEI and NUBARG assert that "[i]t is therefore hardly surprising that, as noted in the Statement of Considerations, '[o]ver one-third of the containments have experienced corrosion or other degradation.' " Therefore, they believe there is not a broad-based concern with operability of containment structures. The NRC rejects the argument that because containments have corrosion allowances and corrosion was expected to occur that, *ipso facto*, further inspections are not necessary and the compliance exception is inappropriate. As previously pointed out, in many cases, the corrosion rate has been found to be greater than that for which the containment was designed (in some cases the rate was twice that predicted). Some of the more extreme cases of wall thinning occurred in plants with corrosion allowances. The existence of a corrosion allowance at any given plant is, of course relevant, but only in the context of determining whether a relevant requirement or commitment is likely to be violated during the OL term. A corrosion allowance simply increases the tolerance (time period) for corrosion. However, once the allowance is eroded, then concern with compliance becomes relevant. Based upon the staff's finding of the number and extent of corrosion to date, and the lack of activities to manage the degradation by many licensees, the NRC concludes that it is likely that those licensees will be in violation of applicable requirements for containment structural integrity and leak-tightness during the OL term, absent the imposition of Subsections IWE and IWL. Because licensees have been unable to ensure compliance with current regulatory requirements, the NRC believes that more specific ISI requirements, which expand upon existing requirements for the examination of containment structures in accordance with GDC 16, 53, Appendix A to 10 CFR part 50, and Appendix J to 10 CFR part 50, are needed and are justified 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 written licensee commitments (e.g., the Final Safety Analysis Report). The NRC believes that the occurrences of corrosion and other degradation would have been detected by licensees when

conducting the periodic examinations set forth in Subsection IWE and Subsection IWL.

With regard to subject area number four, six commenters believe that the Boiling Water Reactors Owner's Group (BWROG) containment inspection plan (CIP) will adequately address examinations for the primary containment when used in conjunction with other existing examination requirements such as Appendix J. The staff does not believe that the CIP is a comprehensive containment examination program. In the CIP, there is a comparison between the CIP and Subsection IWE. The CIP dismisses seven of the eighteen identified Subsection IWE examinations as not being justifiable even though some of these areas are likely to experience accelerated corrosion. The CIP enumerates the conservatisms and margins against failure in the design of Mark I and II containments and concludes that in a typical plant probabilistic risk assessment of failure, the contribution to failure of the containment steel structure is negligible. The NRC believes that the conservatisms and margins referred to are not additional tolerances which allow areas of containments to go unexamined. These conservatisms and margins were required allowances in the design because of the uncertainties in loadings, in material properties, in analysis, and in the variation of steel thicknesses. Examination of large areas of the containment cannot be dismissed as being non-critical based on conservatisms and margins when corrosion has clearly eroded the margin of safety in some cases. In addition, given that only four of the 32 occurrences of corrosion in metal containments and the liners of concrete containments were detected during the pre-integrated leakage rate test examination, the NRC does not believe that the CIP used in conjunction with other existing examination requirements such as Appendix J will adequately address examinations for the primary containment as asserted. The industry initiative that allows a decrease in the frequency of Appendix J leakage rate testing further erodes confidence in the acceptability of the BWROG approach.

Comments were received from ten sources on proposed § 50.55a(g)(6)(ii)(B) which would require a 5-year expedited examination schedule (subject area number five). Most of these comments asked for clarifications of the NRC staff intent of this provision. Some commenters interpreted this provision as a requirement to perform all of the examinations specified for a 10-year interval in 5 years, which was not the intent. § 50.55a(g)(6)(ii)(B) has been changed to clarify that for Subsection IWE, the baseline inspection will be the inservice examinations which are to be performed during the first period of the first interval. For Subsection IWL, the baseline inspection will be the required inservice examinations which correspond to the year of operation for each unit. The result of the clarification is that § 50.55a(g)(6)(ii)(B)(*1*) addresses Subsection IWE and § 50.55a(g)(6)(ii)(B)(*2*) addresses Subsection IWL. § 50.55a(g)(6)(ii)(B)(*2*) in the proposed rule has become § 50.55a(g)(6)(ii)(B)(*3*) and § 50.55a(g)(6)(ii)(B)(*3*) has become § 50.55a(g)(6)(ii)(B)(*4*) in the final rule.

There was one additional comment submitted by NEI. The proposed rule discussed NEI's (then NUMARC) position on the role of Subsection IWE and Subsection IWL in license renewal. Subsections IWE and IWL were referenced many times as one acceptable approach for managing age-related degradation. The plan for managing age-related degradation assumes that these examinations are "in current and effective use." NEI commented on the above statements in the proposed rule: "Although the BWR and PWR containment IRs [Industry Reports] do reference Subsections IWE and IWL, their identification in the IRs should not be misrepresented to imply that Subsections IWE and IWL are being implemented or that they are required for operating plants during their initial licensing term." The NRC agrees that the IRs were not to be represented as a requirement for operating licensees to implement Subsection IWE and Subsection IWL or their equivalent, and that these subsections were referenced as one acceptable approach of managing age-related degradation for the license renewal period. However, present licensee containment examination programs have not proved to be effective in detecting the types of degradation which have been reported. The number of occurrences and the extent of degradation (which includes cases of noncompliance) leads to the conclusion that additional requirements are needed for managing containment degradation during the operating term. Because Subsections IWE and IWL were developed by the ASME with industry input and found to be acceptable by NEI for managing age-related degradation for the license renewal period, the NRC believes that adoption of those programs at this time is the best approach. The NRC also believes that with implementation of Subsections IWE and IWL, the detrimental effects of containment aging will be managed during the current operating term, as well as during the license renewal term.

As a result of the comments received, there is one editorial change, two clarifications, and four modifications in the final rule. With respect to the editorial change, a commenter suggested that the wording of § 50.55a(b)(2)(ix)(D)(*2*) in the proposed rule be revised to be consistent with § 50.55a(b)(2)(ix)(D)(*1*) and § 50.55a(b)(2)(ix)(D)(*3*) of the same paragraph. § 50.55a(b)(2)(ix)(D) addresses the sampling of the grease contained in post-tensioning systems, and conditions, which if found, are reportable. The suggested wording has been adopted in the final rule.

One of the clarifications was to proposed § 50.55(g)(6)(ii)(B). This change was discussed previously in subject area number five. § 50.55a(g)(6)(ii)(B)(*1*) and § 50.55a(g)(6)(ii)(B)(*2*) require that licensees conduct the first containment examinations in accordance with Subsection IWE and Subsection IWL (1992 Edition with the 1992 Addenda), modified by § 50.55a(b)(2)(ix) and § 50.55a(b)(2)(x) 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 many of the required examinations until the end of the 10-year inspection interval. Adding the 10 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.

Paragraphs (g)(6)(ii)(B)(*3*) and (g)(6)(ii)(B)(*4*) 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)(*3*) permits 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)(*4*) allows 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.

The four modifications are: (1) § 50.55a(b)(2)(x)(A) expands the evaluation of inaccessible areas of concrete containments (Class CC) to metal containments and the liners of concrete containments (Class MC); (2) § 50.55a(b)(2)(x)(B) permits alternative lighting and resolution requirements for remote visual examination of the containment; (3) § 50.55a(b)(2)(x)(C) makes the examination of pressure retaining welds and pressure retaining dissimilar metal welds optional; and (4) § 50.55a(b)(2)(x)(D) has been added to provide an alternative sampling plan. Section 50.55a(b)(2)(x)(E), a clarification, more clearly defines the frequency of the Subsection IWE general visual examination.

The first modification, § 50.55a(b)(2)(x)(A), which expands the evaluation of inaccessible areas of concrete containments (Class CC) to metal containments and the liners of concrete containments (Class MC), was the result of a comment received on § 50.55a(b)(2)(ix)(E) of the proposed rule. The commenter believed that given the number of occurrences of corrosion in Class MC containments, the proposed provision (which only addressed concrete containments) should be expanded in the final rule to include metal containments and the liners of concrete containments.

The second modification, § 50.55a(b)(2)(x)(B), was added to the final rule to permit alternative lighting and resolution requirements for remote visual examination of the containment. Subsection IWE references the lighting and resolution requirements contained in IWA-2200. The lighting and resolution requirements contained in IWA-2200 would on a practical basis preclude remote containment examination.

The third modification, § 50.55a(b)(2)(x)(C), makes the examinations of Subsection IWE, Examination Category E-B (pressure retaining welds) and Subsection IWE, Examination Category E-F (pressure retaining dissimilar metal welds) optional. The NRC staff concludes that requiring these examinations is not appropriate. There is no evidence of problems associated with welds of this type under the given operating conditions. In addition, the occupational radiation exposure that would be incurred while performing these examinations cannot be justified. It is estimated that the total occupational exposure that would be incurred yearly in the performance of the containment weld examinations in accordance with Examination Categories E-B and E-F would be 440 person-rems.

The fourth modification, § 50.55a(b)(2)(x)(D), provides an alternative to the ASME Section XI requirements for "additional examinations" (note: additional examinations" are required during the same outage when acceptance criteria are exceeded). The alternative would allow licensees to determine the number of additional components to be examined based on an evaluation to determine the extent and nature of the degradation. Five commenters believe that the requirements for additional examinations used in other subsections of Section XI is inappropriate for containment components. Additional examinations are incorporated into Section XI to determine the extent to which degradation found in one component exists in other similar components. In some instances, a large number of additional examinations could be required. The commenters believe that a review of the operational history of containment components shows that the degradation is limited to the area in question and is not widespread. This makes the Section XI requirements for additional examinations burdensome and inappropriate for application to containments. The NRC agrees and revised the rule to permit the alternative to the Section XI requirements for additional examinations.

The NRC believes that these modifications improve the final rule and will improve the containment inspection program as set forth by Subsection IWE and Subsection IWL. Some of the public comments cited failure data which have been accumulated in recent years in support of various NRC staff activities and industry initiatives. Most of this data has been accumulated since the ASME committees developed these subsections. Without the benefit of this recently accumulated operational data, the ASME committees responsible for developing Subsection IWE and Subsection IWL modelled those subsections on other subsections of Section XI and the experience gained from application of those other subsections. With the additional insights drawn from analysis of this new data, it is apparent that many aspects of containments are unique compared to components of other systems. Some of the containment components which were expected to experience degradation, based on experience with other systems, have proved not to be susceptible to the same type of degradation. The ASME working groups are considering these issues. However, based on initial committee discussion, it is anticipated that similar changes will be made to Subsection IWE and Subsection IWL, but the length of the ASME consensus process precludes the possibility of the changes being adopted into the ASME Code in the near term. Hence, the NRC has determined to adopt the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL with the modifications which were previously discussed.

Other Provisions Contained in the Final Rule

The following paragraph was contained in the proposed rule and has not been discussed previously. This paragraph received comments which resulted in the provision being dropped in the final rule. Section 50.55a(b)(2)(x) was a provision in the proposed rule intended to provide licensees with a mechanism to merge the Subsection IWE and Subsection IWL ISI program with their routine 120-month ISI program. Those licensees who were near the end of their present 10-year ISI interval when the final rule becomes effective would have been given an additional 2 years to submit their containment ISI program. Several commenters responded that due to the time constraints of having to develop the containment ISI program and then perform the required examinations within 5 years, the additional 2 years could not be utilized. Therefore, § 50.55a(b)(2)(x) as it appeared in the proposed rule has been deleted, and § 50.55a(b)(2)(x) in the final rule contains the modifications which were added as a result of public comment on the proposed rule.

The provisions in this paragraph and the following four paragraphs were contained in the proposed rule and have not changed due to comments. Section 50.55a(b)(2)(vi) incorporates 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 incorporate the concept of base metal examinations and also provide a comprehensive set of rules for the examination of post-tensioning systems. It should be noted that the wording of this provision has been changed in the final rule in order to make it consistent with other provisions in § 50.55a(b).

Section 50.55a(b)(2)(ix) specifies 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. Section 50.55a(b)(2)(ix)(A) requires that grease caps which are accessible must be visually examined to detect grease leakage or grease cap deformation. Section 50.55a(b)(2)(ix)(B) requires the preparation of an Engineering Evaluation Report when consecutive surveillances indicate a trend of prestress loss to below the minimum prestress requirements. Section 50.55a(b)(2)(ix)(C) requires an evaluation to be performed for instances of wire failure and slip of wires in anchorages. Section 50.55a(b)(2)(ix)(D) addresses sampled sheathing filler grease and reportable conditions. A comment was received on this provision which resulted in an editorial change (this was discussed on page 12). Section 50.55a(b)(2)(ix)(E) requires that licensees evaluate the acceptability of inaccessible areas of concrete containments when conditions exist in accessible areas that suggest the possibility of degradation in inaccessible areas.

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. Subsection IWE (Class MC—metal containments) and Subsection IWL (Class CC—concrete containments) are incorporated by reference into the NRC regulations for the first time.

Section 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 rule will apply.

Section 50.55a (g)(4)(v)(A), (v)(B), and (v)(C) specify the Subsection IWE and Subsection IWL rules for inservice inspection, repair, and replacement 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 inservice inspection, repairs, and replacements of applicable components and their supports.

Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

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 is not a major Federal action that significantly affects the quality of the human environment and therefore an environmental impact statement is not required.

This final rule is one part of a regulatory framework directed to ensuring containment integrity. Therefore, in the general sense, this rule will have a positive impact on the environment. This rule incorporates by reference into the NRC regulations requirements contained in the ASME Code for the inservice inspection of the containments of nuclear power plants. The performance of containment examinations, as set forth by the provisions of this final rule, for PWRs, Ice Condensers, and BWR Mark IIs and IIIs is not expected to result in significant occupational radiation exposure (1.0 person-rems per year or 0.04 person-rems per unit averaged over 27 examinations each year). The above categories of plants, for which the occupational radiation exposure is insignificant, represent the vast majority of units (89). For BWR Mark I containments, the estimated occupational radiation exposure which would be incurred per year while performing BWR Mark I containment examination is 29.4 person-rems per year or 4.2 person-rems per unit averaged over 7 examinations per year. However, the estimated occupational radiation exposure per unit does not provide an accurate representation of the actual radiological exposure that would be incurred by any one individual. 10 CFR 20.101, "Radiation dose standards for individuals in restricted areas" only permits a whole body dose of 1.25 rem per calendar quarter. As a practical matter, licensees carefully manage the exposure incurred by any one individual by practicing and applying "as low as reasonably achievable" (ALARA) principles to protect the health and safety of personnel. In the performance of the examination of BWR Mark I containments, this is accomplished by having several individuals perform the examinations to "spread out" the exposure. In this manner, no one individual will suffer any significant health effects. It also must be kept in mind that these containment examinations are scheduled to occur at the interval of once every 3⅓ years. This provides licensees ample time for planning the examinations, and scheduling personnel in accord with ALARA considerations. Therefore, the occupational radiation exposure is insignificant given the relatively low exposure on a unit basis and the licensees' programs for controlling the impact of exposure for any one individual.

Actions required of applicants and licensees to implement containment examinations are of the same nature that applicants and licensees have been performing for many years in other Section XI ISI programs. Extension of these actions to additional components, therefore, 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 Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6796.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject

to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget, approval number 3150-0011.

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 8,000 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. The estimate of 8,000 hours for plan update and performing periodic examinations is a 2,000 hour reduction from the estimate given in the proposed rulemaking. This reduction results from changes made in response to public comment. A number of examinations have been modified or made optional greatly reducing the effort required to comply with the requirements contained in the final rule. Send comments on any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by Internet electronic mail at BJS1@NRC.GOV; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

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

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 have a significant economic impact on a substantial number of small entities. This 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 rule does not fall within the purview of the Act.

Backfit Statement

The NRC is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL to assure that the critical areas of containments are routinely inspected to detect defects that could compromise a containment's structural integrity. Based on a preponderance of reliable information, the NRC concludes that this rule is a compliance backfit, and therefore a backfit analysis is not required pursuant to 10 CFR 50.109(a)(4)(i). A summary of noncompliance is set forth below. The documented evaluation required by § 50.109(a)(4) to support this conclusion 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 Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 415-6796.

The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. There have been 32 reported occurrences of corrosion in metal containments and the liners of concrete containments. This is approximately one-fourth of all operating nuclear power plants. Only four of the 32 occurrences were detected by current licensee containment inspection programs. Nine of these occurrences were first identified by the NRC through its inspections or structural audits. Eleven occurrences were detected by licensees after they were alerted to a degraded condition at another site or through activity other than containment inspection. There have been 34 reported occurrences of degradation of the concrete or of the post-tensioning systems of concrete containments. This is nearly one-half of these types of containments. It is clear that current licensee containment inspection programs have not proved to be adequate to detect the types of degradation which have been reported. 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 reduction to below the minimum design thickness; (2) corrosion of the torus of the steel containment shell (wall thickness below minimum design thickness); (3) extensive corrosion of the liner of a concrete containment with local degradation at many locations to approximately half-depth; (4) grease leakage from the tendons of prestressed concrete containments; and (5) leaching as well as excessive cracking in concrete containments.

None of the existing requirements for containment inspection provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation with regard to the performance and the effectiveness of licensee containment examination programs. 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.

Most of those occurrences were first identified by the NRC through its inspections or audits of plant structures, or by licensees while performing an unrelated activity or, after they were alerted to a degraded condition at another site. In analyzing the reported containment degradation, it is apparent that all containments are subject to certain type(s) of degradation depending on the design. Information gathered by the staff indicates that many licensees still have not reacted to this serious safety concern and have not initiated comprehensive containment inservice inspection. As a result of the rate of occurrence of containment degradation, and the extent of containment degradation, the NRC believes that there is a basis for reasonably concluding that such degradation is widespread and affects virtually all plants. Because of the serious degradation which has occurred, the belief that additional occurrences of noncompliance with required minimum wall thicknesses and prestressing forces will be reported, and the high likelihood that some of those occurrences could result in loss of structural integrity and leak-tightness, the NRC has determined that imposition of these containment inservice inspection requirements under the compliance exception to 10 CFR 50.109(a)(4)(i) is appropriate.

The NRC believes that the final action would also result in a substantial safety increase and that the direct and indirect costs of implementation are justified in view of the significant safety benefit to be gained. The NRC believes that the inspections contained in Subsections IWE and IWL will improve significantly the ability to detect degradation and take timely action to correct degradation of containment structures. A review of early implementation of the maintenance rule (10 CFR 50.65) at nine

nuclear power plants, which is documented in NUREG–1526, indicates that most licensees assigned a low priority to the monitoring of structures. Several licensees incorrectly assumed that many of their structures are inherently reliable. This is true so long as there is no degradation. However, the degradation of structures can reduce high margins of safety to a low or negligible margin of safety. As discussed earlier, such substantial containment degradations have been detected at a large number of nuclear power plants, and their detection to date can best be characterized as happenstance. The final rule will provide for improved periodic examination of containment structures assuring that the critical areas of containment are periodically inspected to detect and take corrective action for defects that could compromise the containment's pressure-retaining and leak-tight capability. The NRC believes, therefore, that the final action can be justified as a cost-justified safety enhancement backfit, as well as a compliance backfit.

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 adopting 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. 444, 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. 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 paragraphs (b)(2) and (g)(4) to read as follows:

§ 50.55a Codes and standards.

* * * * *

(b) * * *

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Class 1, Class 2, and Class 3 components of Section XI, Division 1, and include addenda through the 1988 Addenda and editions through the 1989 Edition, and Class MC and Class CC components of Section XI, Division 1, 1992 Edition with the 1992 Addenda, subject to the following limitations and modifications:

* * * * *

(vi) *Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI.* The 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL shall be used by licensees when performing containment examinations as modified and supplemented by the requirements in § 50.55a(b)(2)(ix) and § 50.55a(b)(2)(x).

* * * * *

(ix) *Examination of concrete containments.* (A) 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) 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, an evaluation shall be performed and reported in the Engineering Evaluation Report as prescribed in IWL–3300.

(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 required by IWA–6000.

(D) The licensee shall report the following conditions, if they occur, in the ISI Summary Report required by IWA–6000:

(*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 exceeds 10 percent of the tendon net duct volume.

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

(E) For Class CC applications, 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 required by IWA–6000:

(*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) *Examination of metal containments and the liners of concrete containments.* (A) For Class MC applications, 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 required by IWA–6000:

(*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.

(B) When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA–2210–1 may be extended and the minimum illumination requirements specified in Table IWA–2210–1 may be decreased provided that the conditions or indications for which the visual

examination is performed can be detected at the chosen distance and illumination.

(C) The examinations specified in Examination Category E-B, Pressure Retaining Welds, and Examination Category E-F, Pressure Retaining Dissimilar Metal Welds, are optional.

(D) Section 50.55a(b)(2)(x)(D) may be used as an alternative to the requirements of IWE-2430.

(*1*) If the examinations reveal flaws or areas of degradation exceeding the acceptance standards of Table IWE-3410-1, an evaluation shall be performed to determine whether additional component examinations are required. For each flaw or area of degradation identified which exceeds acceptance standards, the licensee shall provide the following in the ISI Summary Report required by IWA-6000:

(*i*) A description of each flaw or area, including the extent of degradation, and the conditions that led to the degradation;

(*ii*) The acceptability of each flaw or area, and the need for additional examinations to verify that similar degradation does not exist in similar components, and;

(*iii*) A description of necessary corrective actions.

(*2*) The number and type of additional examinations to ensure detection of similar degradation in similar components.

(E) A general visual examination as required by Subsection IWE shall be performed once each period.

* * * * *

(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 that 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) of this section, 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 MC; 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.

* * * * *

(6) * * *

(ii) * * *

(B) *Expedited examination of containment.* (*1*) Licensees of all operating nuclear power plants shall implement the inservice examinations specified for the first period of the first inspection interval in Subsection IWE of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a (b)(2)(ix) by September 9, 2001. The examination performed during the first period of the first inspection interval shall serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation.

(*2*) Licensees of all operating nuclear power plants shall implement the inservice examinations which correspond to the number of years of operation which are specified in Subsection IWL of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a (b)(2)(ix) by September 9, 2001. The first examination performed shall serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation.

(*3*) The expedited examination for Class MC components may be used to satisfy the requirements of routinely scheduled examinations of Subsection IWE subject to IWA-2430(d) when the expedited examination occurs during the first containment inspection interval.

(*4*) The requirement for the expedited examination of the containment post-tensioning system may be satisfied by the post-tensioning system examinations performed after September 9, 1996 as a result of licensee post-tensioning system programs accepted by the NRC prior to September 9, 1996.

(*5*) Licensees do not have to submit to the NRC staff for approval of their containment inservice inspection program which was developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified modifications and a limitation. The program elements and the required documentation shall be maintained on site for audit.

* * * * *

Dated at Rockville, Maryland, this 12th day of June 1996.

For the Nuclear Regulatory Commission.

James M. Taylor,

Executive Director for Operations.

[FR Doc. 96-20215 Filed 8-7-96; 8:45 am]

BILLING CODE 7590-01-P

NATIONAL CREDIT UNION ADMINISTRATION

12 CFR Part 701

Supervisory Committee Audits and Verifications

AGENCY: National Credit Union Administration (NCUA).

ACTION: Final rule.

SUMMARY: The National Credit Union Administration (NCUA) is amending its regulations governing credit union supervisory committee audits and verifications. The final amendments clarify existing audit scope; expand audit scope and reporting requirements for compensated auditors only; require a comprehensive engagement letter setting forth minimum contracting terms and conditions; clarify existing working paper access requirements; expressly state available administrative sanctions for failure to comply with supervisory

10 CFR 50.55a

*62FR53932

1/1/1998

IEEE

and weight range and the number of days grazing occurred, and the amount and type of feed fed such grazing animals during any grazing period within the crop year.

(f) Animal Unit Day adjustments, as determined by CCC, may be calculated when a producer of forage predominantly grazed, provides adequate evidence, as determined by CCC, that unit forage management and maintenance practices provide different carrying capacity than practices generally provided forage acreage used to calculate the approved county expected carrying capacity.

8. Amend § 1437.9 to revise paragraph (b)(2) to read as follows:

§ 1437.9 Loss Requirements.

* * * * *

(b) * * *

(2) The failure of the producer to reseed or replant to the same crop in the county where it is practicable to reseed or replant;

* * * * *

9. Amend § 1437.11 to revise the introductory text and add paragraph (c) to read as follows:

§ 1437.11 Payments for reduced yields and prevented planting.

In the event that the area loss requirement has been satisfied for the crop and:

* * * * *

(c) The producer has sustained a loss of forage determined by CCC to be predominantly grazed in accordance with § 1437.7(l), in excess of 50 percent of the producer's expected Animal Unit Day established for the unit, the NAP payment will be determined by:

(1) Dividing the unit acreage for each species or type or variety identified on the unit by the approved carrying capacity and multiplying the result by the corresponding grazing days used as the basis for determination of the carrying capacity, totaling the result for each species or types and varieties.

(2) Multiplying the result of paragraph (c)(1) of this section by 50 percent.

(3) Multiplying the number of animals grazed by the daily allowance of corn according to type and weight range and divide the result by pounds of corn CCC determines is necessary to provide the daily energy requirement for one animal unit.

(4) Multiplying the result of paragraph (c)(3) of this section by the number of days grazing occurred to determine gross actual AUD.

(5) Adding AUD for ineligible causes of loss and incidental mechanically harvested Category 1 forage to the result of paragraph (c)(4) of this section.

(6) Subtracting AUD or equivalent value of supplemental feed fed to the grazing livestock during the crop year from the result of paragraph (c)(5) of this section.

(7) Subtracting the result of paragraph (c)(6) of this section from the result of paragraph (c)(2) of this section. If a zero or negative number results, payment cannot be calculated.

(8) Multiplying the positive result of paragraph (c)(7) of this section by:

(i) For the 1997 through 1998 crop years, 60 percent of the average market price, as determined by CCC, or any comparable coverage, as determined by CCC; or

(ii) For the 1999 and subsequent years, 55 percent of the average market price, as determined by CCC, or any comparable coverage, as determined by CCC.

Signed at Washington, DC, on October 8, 1997.

Keith Kelly,

Executive Vice President, Commodity Credit Corporation.

[FR Doc. 97-27432 Filed 10-16-97; 8:45 am]

BILLING CODE 3410-01-P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AF73

Codes and Standards; IEEE National Consensus Standard

AGENCY: Nuclear Regulatory Commission.

ACTION: Direct final rule.

SUMMARY: The Nuclear Regulatory Commission is amending its regulations to incorporate by reference IEEE Std. 603-1991, a national consensus standard for power, instrumentation, and control portions of safety systems in nuclear power plants. This action is necessary to endorse the latest version of this national consensus standard in NRC's regulations, and replace an IEEE standard currently endorsed in the NRC's regulations which has been withdrawn by the IEEE.

EFFECTIVE DATE: The final rule is effective on January 1, 1998, unless significant adverse comments are received by December 1, 1997. If the effective date is delayed, timely notice will be published in the Federal Register. The incorporation by reference of IEEE Std. 603-1991 is approved by the Director of the Federal Register as of January 1, 1998.

ADDRESSES: Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; Attention: Rulemakings and Adjudications Staff. Hand deliver comments to 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

FOR FURTHER INFORMATION CONTACT: Satish K. Aggarwal, Senior Program Manager, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 415-6005, Fax (301) 415-5074 (e-mail: SKA@NRC.GOV).

SUPPLEMENTARY INFORMATION: NRC considers this rulemaking, which endorses IEEE Std. 603-1991, to be noncontroversial because, as noted in the background discussion, there was no adverse public comment on the regulatory guide endorsing this standard. Accordingly, the Commission finds that public notice and opportunity for comment are unnecessary pursuant to 5 U.S.C. 553(b)(B). Thus, the Commission is publishing this rule in final form without seeking public comments on the amendment in a proposed rule. This action will become effective on January 1, 1998. However, if the NRC receives significant adverse comments by December 1, 1997, then the NRC will publish a document that withdraws this action, and will address the comments received in response to an identical proposed rule which is being concurrently published in the proposed rules section of this Federal Register. Any significant adverse comments will be addressed in a subsequent final rule. The NRC will not initiate a second comment period on this action in the event the direct final rule is withdrawn.

Background

In 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities," § 50.55a requires that the protection systems in nuclear power plants meet the requirements set forth in IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," in effect on the formal docket date of the application. However, IEEE Std. 279 is obsolete, has been withdrawn by IEEE and has now been superseded by IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations."

In November 1995, the NRC staff issued for public comment a draft regulatory guide, DG-1042, which was proposed Revision 1 to Regulatory Guide 1.153, "Criteria for Safety Systems." This draft regulatory guide proposed to endorse IEEE Std. 603-1991

(including the correction sheet dated January 30, 1995). Because there were no adverse public comments to Revision 1 to Regulatory Guide 1.153, the Commission believes that there is general public consensus that IEEE Std. 603-1991 provides acceptable criteria for safety systems in nuclear power plants.

Discussion

The direct final rule incorporates a national consensus standard, IEEE Std. 603-1991, for establishing minimal functional and design requirements for power, instrumentation, and control portions of safety systems for nuclear power plants into NRC regulations. This action is consistent with the provisions of the National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, which encourages Federal regulatory agencies to consider adopting industry consensus standards as an alternative to de novo agency development of standards affecting an industry. This action is also consistent with the NRC policy of evaluating the latest versions of national consensus standards in terms of their suitability for endorsement by regulations or regulatory guides.

Currently, 10 CFR 50.55 a(h) specifies that "protection systems" for plants with construction permits issued after January 1, 1971, must meet the requirements in IEEE Std. 279 in effect on the formal docket date of the application for a construction permit. IEEE Std. 279 states that a "protection system" encompasses all electric and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating those signals associated with the protective function. These signals include those that actuate reactor trip and that, in the event of a serious reactor accident, actuate engineered safeguards such as containment isolation, core spray, safety injection, pressure reduction, and air cleaning. "Protective Function" is defined by IEEE Std. 279, as "the sensing of one or more variables associated with a particular generating station condition, signal processing, and the initiation and completion of the protective action at values of the variables established in the design bases."

IEEE Std. 603-1991 uses the term "safety systems" rather than "protection systems." A "safety system" is defined by IEEE Std. 603-1991 as "a system that is relied upon to remain functional during and following design basis events to ensure: (i) The integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shut down condition, or (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the 10 CFR part 100 guidelines." A "safety function" is defined by IEEE Std. 603-1991 as "one of the processes or conditions (for example, emergency negative reactivity insertion, post-accident heat removal, emergency core cooling, post-accident radioactivity removal, and containment isolation) essential to maintain plant parameters within acceptable limits established for a design basis event."

The Commission considers that the systems covered by IEEE Std. 603-1991 and IEEE Std. 279-1971 are the same. Therefore, for purposes of paragraph (h) of 10 CFR 50.55a, "protection systems," and "safety systems" are synonymous. The Commission notes that these two terms are also synonymous with the term "safety-related systems," used elsewhere in the Commission's regulations. Therefore, licensees are expected to apply IEEE Std. 279-1971 and IEEE Std. 603-1991, as appropriate, to "safety-related systems."

This rule mandates the use of IEEE Std. 603-1991 (including the correction sheet dated January 30, 1995) for future nuclear power plants, including final design approvals, design certifications and combined licenses under 10 CFR part 52. Current licensees may continue to meet the requirements set forth in the edition or revision of IEEE Std. 279 in effect on the formal date of their application for a construction permit or may, at their option, use IEEE Std. 603-1991, provided they comply with all applicable requirements for making changes to their licensing basis. However, changes to protection systems in operating nuclear power plants initiated on or after January 1, 1998 must meet the requirements in IEEE Std. 603-1991. For purposes of this rule, "changes" to protection systems include (i) modifications, augmentation or replacement of protection systems permitted by license amendments, (ii) changes made by the licensees pursuant to procedures in 10 CFR 50.59, and (iii) plant-specific departures from a design certification rule under 10 CFR part 52. In-kind (like-for-like) replacement of protection system components are not considered changes to the protection systems.

Section 3 of IEEE Std. 603-1991 references several industry codes and standards. If the referenced standard has been endorsed in a regulatory guide, the standard constitutes a method acceptable to the Commission of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has not been endorsed in a regulatory guide, the licensees and applicants may consider and use the information in the referenced standard consistent with current regulatory practices.

Electronic Access

You may also provide comments via the NRC's interactive rulemaking website through the NRC home page (http://www.nrc.gov). This site provides the availability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking website, contact Ms. Carol Gallagher, (301) 415-5905 (e-mail: CAG@nrc.gov).

Finding of No Environmental Impact: Availability of Environmental Assessment

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 would not be a major Federal action significantly affecting the quality of the human environment and, therefore, an environment impact statement is not required. The Commission has prepared an Environmental Assessment supporting this finding of no significant environmental impact.

The NRC has sent a copy of the environmental assessment and a copy of the **Federal Register** Notice to every State liaison officer and requested their comments on the environmental assessment. The environmental assessment is available for inspection at the NRC Public Document Room, 2120 L Street NW., Washington, DC. Also, the NRC has committed itself to complying in all its actions with the Presidential Executive Order #12898—Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations, dated February 11, 1994. Therefore, the NRC also has determined that there are no disproportionate, high, and adverse impacts on minority and low-income populations. The NRC uses the following working definition of environmental justice: environmental justice means the fair treatment and meaningful involvement of all people, regardless of race, ethnicity, culture, income, or educational level with respect to the development, implementation, and enforcement of environmental laws, regulations and policies.

Paperwork Reduction Act Statement

This final rule does not contain a new or amended information collection requirement subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501, *et seq.*). Existing requirements were approved by the Office of Management and Budget, approval No. 3150-0011.

Public Protection Notification

If a document 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, an information collection.

Regulatory Analysis

The Commission has prepared a regulatory analysis which shows that the proposed amendment does not impose any new requirements or costs on current licensees who do not make changes to safety systems. However, licensees planning or proposing changes to power and instrumentation & control systems will be impacted because they will be required to meet the requirements of IEEE Std. 603-1991 for the changes even though the remainder of the plant power and I&C systems are only required to meet their current licensing basis. The draft regulatory analysis is available for inspection in the NRC Public Document Room, 2120 L Street, NW., Washington, DC.

Regulatory Flexibility Certification

As required by the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule will not have a significant economic impact on small entities. This 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 adopted by the NRC (10 CFR 2.810). Since these companies are dominant in their service areas, this rule does not fall within the purview of the Act.

Backfit Analysis

The rule requires applicants and holders of new construction permits, new operating licenses, new final design approvals, new design certifications and combined licenses to comply with IEEE Std. 603-1991 (including the correction sheet dated January 30, 1995). Changes to protection systems in existing operating plants initiated on or after January 1, 1998 must meet the requirements of IEEE Std. 603-1991. IEEE Std. 279 will continue to apply to existing nuclear power plants that do not make any changes to their protection systems, but the rule permits the licensee the option of meeting IEEE Std. 603-1991.

The backfit rule was not intended to apply to regulatory actions which change expectations of prospective applicants, and therefore the backfit rule does not apply to the portion of the rule applicable to new construction permits, new operating licenses, new final design approvals, new design certifications and combined licenses. This rule does not change the licensing basis (i.e., IEEE Std. 279) for plants that do not intend to make any changes to their power and instrumentation and control systems. However, the rule would require future changes to existing power and instrumentation and control portions of protection systems to comply with the new standard. This would not be considered a backfit, since the changes are voluntarily initiated by the licensee, or separately imposed by the NRC after a separate backfit analysis. This is consistent with past NRC practice and the discussions on backfitting in "Value-Impact Statement" prepared for Revision 1 to Regulatory Guide 1.153. A copy of the Value-Impact Statement is available for inspection or copying for a fee in the Commission's Public Document Room at 2120 L Street NW., Washington, DC, under Task DG-1042.

In summary, the NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this direct final rule because it does not impose any backfits as defined in 10 CFR 50.109(a)(1) and, therefore, a backfit analysis has not been prepared for this direct final rule.

Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

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, and 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 Reorganizations Act of 1974, as amended, and 5 U.S.C. 552 and 553, the NRC is adopting the following amendment 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. 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, and 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).

12. In § 50.55a, paragraph (h) is revised to read as follows:

§ 50.55a Codes and standards.

* * * * *

(h) *Protection and Safety Systems.* (1) IEEE Std. 603-1991 and the correction sheet dated January 30, 1995, which are referenced in paragraph (h)(3) and (h)(4), are approved for incorporation by reference by the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. Copies of IEEE Std. 603-1991 may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, Piscataway, NJ 08855. It is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, MD 20852-2738, and at the Office of the Federal Register, 800 North Capital Street, NW, Suite 700, Washington, DC. IEEE Std. 279, which is referenced in paragraph (h)(2) of this section was approved for incorporation by reference by the Director of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR part 51. Copies of this standard are also available as indicated for IEEE Std. 603-1991.

(2) Definitions.

(i) For purposes of this paragraph the terms "protection systems," "safety systems," and "safety-related systems" are synonymous.

(ii) Changes to protection systems include modification, augmentation or replacement of protection systems permitted by license amendments, changes to protection systems made by licensees pursuant to 10 CFR 50.59, and plant specific departures from a design certification rule under 10 CFR part 52.

(3) Protection systems. For nuclear power plants with construction permits issued after January 1, 1971, but prior to January 1, 1998, protection systems must meet the requirements set forth either in the Institute of Electrical and Electronics Engineers (IEEE) Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. However, changes to protection systems initiated on or after January 1, 1998 must meet the requirements set forth in IEEE Std. 603-1991, and the correction sheet dated January 30, 1995.

(4) Safety systems. For construction permits, operating licenses, final design approvals, design certifications and combined licenses issued on or after January 1, 1998, safety systems must meet the requirements set forth in IEEE Std. 603-1991, and the correction sheet, dated January 30, 1995.

Dated at Rockville, this 9th day of October, 1997.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 97-27421 Filed 10-16-97; 8:45 am]

BILLING CODE 7590-01-P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 97-ANE-38-AD; Amendment 39-10160; AD 97-21-07]

RIN 2120-AA64

Airworthiness Directives; AlliedSignal Inc. (Formerly Textron Lycoming) Model T5313B, T5317A, and T53 (Military) Turboshaft Engines

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule; request for comments.

SUMMARY: This amendment adopts a new airworthiness directive (AD) that is applicable to AlliedSignal Inc. (formerly Textron Lycoming) Model T5313B, T5317A, and T53 series military turboshaft engines approved for installation on aircraft certified in accordance with Section 21.25 of the Federal Aviation Regulations (FAR). This action requires a one-time visual inspection of accessory drive carrier assemblies for affected serial numbers (S/Ns) designating a defective assembly, and if the S/N is applicable, replacement with a serviceable assembly. This amendment is prompted by a report of an N2 overspeed condition due to a defective accessory drive carrier assembly. The actions specified in this AD are intended to prevent accessory drive carrier assembly failure, which could result in an N2 overspeed and an uncontained engine failure.

DATES: Effective November 3, 1997.

The incorporation by reference of certain publications listed in the regulations is approved by the Director of the Federal Register as of November 3, 1997.

Comments for inclusion in the Rules Docket must be received on or before December 16, 1997.

ADDRESSES: Submit comments in triplicate to the Federal Aviation Administration (FAA), New England Region, Office of the Assistant Chief Counsel, Attention: Rules Docket No. 97-ANE-38-AD, 12 New England Executive Park, Burlington, MA 01803-5299. Comments may also be sent via the Internet using the following address: "9-ad-engineprop@faa.dot.gov". Comments sent via the Internet must contain the docket number in the subject line.

The service information referenced in this AD may be obtained from AlliedSignal Aerospace, Attn: Data Distribution, M/S 64-3/2101-201, P.O. Box 29003, Phoenix, AZ 85038-9003; telephone (602) 365-2493, fax (602) 365-5577. This information may be examined at the FAA, New England Region, Office of the Assistant Chief Counsel, 12 New England Executive Park, Burlington, MA; or at the Office of the Federal Register, 800 North Capitol Street, NW., suite 700, Washington, DC.

FOR FURTHER INFORMATION CONTACT: Ray Vakili, Aerospace Engineer, Los Angeles Aircraft Certification Office, FAA, Transport Airplane Directorate, 3960 Paramount Blvd., Lakewood, CA 90712-4137; telephone (562) 627-5262, fax (562) 627-5210.

SUPPLEMENTARY INFORMATION: The Federal Aviation Administration has received a report of an N2 overspeed condition on an AlliedSignal Inc. (formerly Textron Lycoming) Model T5317A-1 turboshaft engine. The investigation revealed that the N2 overspeed condition was caused when the N2 overspeed governor bevel gear, which is part of the accessory drive carrier and cap assembly, shifted out of position. This gear shifting out of position was determined to be due to improper manufacturing of the accessory drive carrier and cap assembly, Part Number (P/N) 1-070-210-01, which is installed on the higher level assembly, accessory drive carrier assembly, P/N 1-070-220-03, 1-070-220-12, or 1-070-220-13. All accessory drive carrier assemblies, P/Ns 1-070-220-03, 1-070-220-12, and 1-070-220-13, installed after November 1, 1985, and have been identified by serial number (S/N) are subject to this inspection. This condition, if not corrected, could result in accessory drive carrier assembly failure, which could result in an N2 overspeed and an uncontained engine failure.

The FAA has reviewed and approved the technical contents of AlliedSignal Inc. Alert Service Bulletin (ASB) No. T5313B/17A-A0092, Revision 1, dated July 1, 1997; ASB No. T53-L-13B-A0092, dated June 4, 1997; and ASB No. T53-L-703-A0092, dated June 4, 1997. These ASBs describe procedures for performing a one-time visual inspection of accessory drive carrier assemblies for affected S/Ns designating a defective assembly, and if the S/N is applicable, replacement with a serviceable assembly.

Since an unsafe condition has been identified that is likely to exist or develop on other engines of the same type design, this AD is being issued to prevent accessory drive carrier assembly failure. This AD requires a one-time visual inspection of accessory drive carrier assemblies for affected S/Ns designating a potentially defective assembly, and if the S/N is applicable, replacement with a serviceable assembly. The actions are required to be accomplished in accordance with the ASBs described previously.

Since a situation exists that requires the immediate adoption of this regulation, it is found that notice and opportunity for prior public comment hereon are impracticable, and that good cause exists for making this amendment effective in less than 30 days.

Comments Invited

Although this action is in the form of a final rule that involves requirements affecting flight safety and, thus, was not preceded by notice and an opportunity

10 CFR 50.55a

**62FR66977

12/23/1997 Withdraws
*62FR53932

IEEE

Ballots invalid under this subpart shall not be counted.

§ 1209.306 Referendum report.

Except as otherwise directed, the referendum agent shall prepare and submit to the Administrator a report on results of the referendum, the manner in which it was conducted, the extent and kind of public notice given, and other information pertinent to analysis of the referendum and its results.

§ 1209.307 Confidential information.

The ballots and other information or reports that reveal, or tend to reveal, the identity or vote of any person covered under the Act shall be held confidential and shall not be disclosed.

Dated: December 11, 1997.

Sharon Bomer Lauritsen,

Associate Deputy Administrator, Fruit and Vegetable Programs.

[FR Doc. 97-32812 Filed 12-22-97; 8:45 am]

BILLING CODE 3410-02-P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AF73

Codes and Standards; IEEE National Consensus Standard, Withdrawal

AGENCY: Nuclear Regulatory Commission.

ACTION: Direct final rule; withdrawal.

SUMMARY: The Nuclear Regulatory Commission is withdrawing a direct final rule that would have amended Commission's regulations to incorporate by reference the most recent published version of IEEE Std. 603-1991, a national consensus standard for power, instrumentation, and control portions of safety systems in nuclear power plants. The NRC is taking this action because it has received significant adverse comments in response to an identical proposed rule which was concurrently published in the Federal Register.

FOR FURTHER INFORMATION CONTACT: Satish K. Aggarwal, Senior Program Manager, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 415-6005, Fax (301) 415-5074 (e-mail: SKA@NRC.GOV).

SUPPLEMENTARY INFORMATION: On October 17, 1997 (62 FR 53933), the Nuclear Regulatory Commission published in the Federal Register a direct final rule amending its regulations at 10 CFR 50.55a(h) to incorporate by reference the most recently published version of a national consensus standard. The direct final rule was to become effective on January 1, 1998. The NRC also concurrently published an identical proposed rule on October 17, 1997 (62 FR 53975). In these documents, the NRC indicated that if it received significant adverse comments in response to this action, the NRC would withdraw the direct final rule and would consider the comments received as in response to the proposed rule and address these comments in a subsequent final rule. The NRC has received significant adverse comments on the direct final rule. Therefore, the Commission is withdrawing the October 17, 1997, direct final rule. The public comments received will be addressed in a subsequent final rule issued in either a notice of final rulemaking or in a notice of withdrawal of the proposed rule.

Dated at Rockville, Maryland, this 16th day of December, 1997.

For the Nuclear Regulatory Commission.

John C. Hoyle,

Secretary of the Commission.

[FR Doc. 97-33424 Filed 12-22-97; 8:45 am]

BILLING CODE 7590-01-P

FEDERAL HOUSING FINANCE BOARD

12 CFR Part 960

[No. 97-N-10]

Questions and Answers Regarding The Affordable Housing Program

AGENCY: Federal Housing Finance Board.

ACTION: Staff interpretation of affordable housing regulations.

SUMMARY: The Federal Housing Finance Board (Finance Board) is publishing Questions and Answers regarding the Affordable Housing Program (AHP). The Questions and Answers have been prepared by staff of the Finance Board in response to questions about changes in the Finance Board's regulation governing the AHP that will go into effect on January 1, 1998. The Questions and Answers constitute informal staff guidance for Finance Board personnel, the Federal Home Loan Banks (Bank), Bank members, and program participants. The Answers are intended to be interpretive of the Finance Board's regulation governing the AHP, and are not statements of agency policy. The Questions and Answers have not been considered or approved by the Board of Directors of the Finance Board.

FOR FURTHER INFORMATION CONTACT: Richard Tucker, Deputy Director, Compliance Assistance Division, (202) 408-2848, or Janet M. Fronckowiak, Program Analyst, Compliance Assistance Division, (202) 408-2575, or Diane E. Dorius, Associate Director, Program Development Division, (202) 408-2576, Office of Policy, Federal Housing Finance Board, 1777 F Street, N.W., Washington, D.C. 20006.

SUPPLEMENTARY INFORMATION: On August 4, 1997, the Finance Board published a final rule amending its existing regulation governing the AHP. *See* 62 FR 41812 (Aug. 4, 1997). The final rule will become effective on January 1, 1998. In the months following publication of the final rule, the Finance Board has provided training to the staffs of the Banks to assist them in making a smooth transition to operation under the amended AHP regulation. A number of questions of regulatory interpretation were raised by Bank staff as a result of the Finance Board's training sessions. The staff of the Finance Board has prepared answers to the most frequently asked questions. The Questions and Answers constitute informal interpretive guidance for Finance Board personnel, the Banks, Bank members, and program participants. The Answers are intended to be interpretative of the AHP regulation, not statements of agency policy, and they have not been considered or approved by the Board of Directors of the Finance Board.

The Questions and Answers are grouped by the provision of the AHP regulation that they discuss and are presented in the same order as the regulatory provisions. The text of the Questions and Answers follows:

Text of the Questions and Answers

Questions and Answers Regarding the AHP

Definitions (§ 960.1)

Low- and Moderate-Income and Very Low-Income Household Eligibility for Current Occupants:

Q1. When a rental project involves both purchase and rehabilitation, which point in time should be used for purposes of determining household eligibility?

A1. The regulation permits a choice of determining income eligibility either at the time of completion of the purchase or at the time of completion of the rehabilitation.

Q2. In the case of projects involving the purchase or rehabilitation of rental housing with current occupants, can an occupying household that is a very low-income or a low- or moderate-income household at the time the AHP application is submitted to the Bank be deemed to be a very low-income or a low- or moderate-income household at

10 CFR 50.55a

63FR1335

1/9/1998 Corrects
**62FR66977

IEEE

10 CFR 50.55a

***64FR17944

5/13/1999

IEEE

10 CFR 50.55a

64FR23763

5/4/1999 Corrects
***64FR17944

IEEE

believed that the undersized rule change was needed to expedite that reduction. With the excess tonnage of dried prunes, the Committee also considered establishing a reserve pool and diversion program to reduce the oversupply situation. These initiatives were not supported because they would not specifically eliminate the smallest, least valuable prunes which are in oversupply. Instead, the reserve pool and diversion program would eliminate larger size prunes from human consumption outlets. Reserve pools for prunes have historically been implemented on dried prunes regardless of the size of the prunes. While the marketing order also allows handlers to remove the larger prunes from the pool by replacing them with small prunes and the value difference in cash, this exchange would be cumbersome and expensive to administer compared to this rule.

Section 8e of the Act requires that when certain domestically produced commodities, including prunes, are regulated under a Federal marketing order, imports of that commodity must meet the same or comparable grade, size, quality, or maturity requirements for the domestically produced commodity. This action does not impact the dried prune import regulation because the action to be implemented is for volume control, not quality control, purposes. The smaller diameter openings of 23/32 of an inch for French prunes and 28/32 of an inch for non-French prunes were implemented for the purpose of improving product quality. The increases to 24/32 of an inch in diameter for French prunes and 30/32 of an inch in diameter for non-French prunes are for purposes of volume control.

Therefore, the increased diameters will not be applied to imported prunes.

This action will not impose any additional reporting or recordkeeping requirements on either small or large California dried prune handlers. As with all Federal marketing order programs, reports and forms are periodically reviewed to reduce information requirements and duplication by industry and public sector agencies. In addition, as noted in the initial regulatory flexibility analysis, the Department has not identified any relevant Federal rules that duplicate, overlap, or conflict with this rule.

In addition, the Committee's meeting was widely publicized throughout the prune industry and all interested persons were invited to attend the meeting and participate in Committee deliberations on all issues. Like all Committee meetings, the December 1, 1998, meeting was a public meeting and all entities, both large and small, were able to express views on this issue. The Committee itself is composed of twenty-two members, of which seven are handlers, fourteen are producers, and one is a public member. Moreover, the Committee and its Supply Management Subcommittee have been reviewing this supply management problem for the second year, and this rule reflects their deliberations completely.

A proposed rule concerning this action was published in the **Federal Register** on January 25, 1999 (64 FR 3660). Copies of this rule were mailed or sent via facsimile to all Committee members, alternates and dried prune handlers. Finally, the rule was made available through the Internet by the U.S. Government Printing Office. The rule provided a comment period which ended April 15, 1999. No comments were received.

After consideration of all relevant material presented, including the information and recommendation submitted by the Committee and other available information, it is hereby found that this rule, as hereinafter set forth, will tend to effectuate the declared policy of the Act.

List of Subjects in 7 CFR Part 993

Marketing agreements, Plums, Prunes, Reporting and recordkeeping requirements.

For the reasons set forth in the preamble, 7 CFR part 993 is amended as follows:

PART 993—DRIED PRUNES PRODUCED IN CALIFORNIA

1. The authority citation for 7 CFR part 993 continues to read as follows:

Authority: 7 U.S.C. 601–674.

2. A new §993.406 is added to read as follows:

Note: This section will not appear in the Code of Federal Regulations.

§993.406 Undersized prune regulation for the 1999–2000 crop year.

Pursuant to §§993.49(c) and 993.52, an undersized prune regulation for the 1999–2000 crop year is hereby established. Undersized prunes are prunes which pass through openings as follows: For French prunes, 24/32 of an inch in diameter; for non-French prunes, 30/32 of an inch in diameter.

Dated: April 27, 1999.

Robert C. Keeney,

Deputy Administrator, Fruit and Vegetable Programs.

[FR Doc. 99-11078 Filed 5-3-99; 8:45 am]

BILLING CODE 3410-02-P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AF96

Codes and Standards: IEEE National Consensus Standard; Correction

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule: Correction.

SUMMARY: This document corrects a final rule appearing in the **Federal Register** on April 13, 1999 (64 FR 17944), that incorporates by reference IEEE Std. 603-1991, a national consensus standard for power, instrumentation, and control portions of safety systems in nuclear power plants. This action is necessary to correct an erroneous reference.

EFFECTIVE DATE: The final rule is effective on May 13, 1999.

FOR FURTHER INFORMATION CONTACT: Michael T. Lesar, **Federal Register** Liaison Officer, telephone (301) 415-7163.

SUPPLEMENTARY INFORMATION: On page 17946, in the third column, in the codified text at §50.55a(h)(1), on the fourteenth and twenty-first lines from the top, and at §50.55a(h)(2) on the twenty-eighth line from the top "Std. 279-1971" should be corrected to read "Std. 279."

Dated at Rockville, Maryland, this 28th day of April, 1999.

For the Nuclear Regulatory Commission.

Annette L. Vietti-Cook,

Secretary of the Commission.

[FR Doc. 99-11111 Filed 5-3-99; 8:45 am]

BILLING CODE 7590-01-P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 98-NM-202-AD; Amendment 39-11151; AD 99-09-18]

RIN 2120-AA64

Airworthiness Directives; Fokker Model F.28 Mark 0070 and Mark 0100 Series Airplanes

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule.

SUMMARY: This amendment supersedes an existing airworthiness directive (AD), applicable to certain Fokker Model F.28 Mark 0070 and Mark 0100 series

10 CFR 50.55a

****64FR51370

11/22/1999

Added 1989 Addenda to 1996 Addenda
III, XI, OM

federal register

Wednesday
September 22, 1999

Part II

Nuclear Regulatory Commission

10 CFR Part 50

Industry Codes and Standards; Amended Requirements; Final Rule

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AE26

Industry Codes and Standards; Amended Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission is amending its regulations to incorporate by reference more recent editions and addenda of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants for construction, inservice inspection, and inservice testing. These provisions provide updated rules for the construction of components of light-water-cooled nuclear power plants, and for the inservice inspection and inservice testing of those components. This final rule permits the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power plant components.

DATES: Effective November 22, 1999. The incorporation by reference of certain publications listed in the regulations is approved by the Director of the Federal Register as of November 22, 1999.

FOR FURTHER INFORMATION CONTACT: Thomas G. Scarbrough, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Telephone: 301-415-2794, or Robert A. Hermann, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Telephone: 301-415-2768.

SUPPLEMENTARY INFORMATION:

1. Background

The Nuclear Regulatory Commission (NRC) is amending its regulations to incorporate by reference the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section III, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code) with five limitations; the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section XI, Division 1, of the ASME BPV Code with three limitations; and the 1995 Edition and 1996 Addenda of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) with one limitation and one modification. The final rule imposes an expedited implementation of performance demonstration methods for ultrasonic examination systems. The final rule permits the optional implementation of the ASME Code, Section XI, provisions for surface examinations of High Pressure Safety Injection Class 1 piping welds. The final rule also permits the use of evaluation criteria for temporary acceptance of flaws in ASME Code Class 3 piping (Code Case N-523-1); mechanical clamping devices for ASME Code Class 2 and 3 piping (Code Case N-513); the 1992 Edition including the 1992 Addenda of Subsections IWE and IWL in lieu of updating to the 1995 Edition and 1996 Addenda; alternative rules for preservice and inservice testing of certain motor-operated valve assemblies (OMN-1) in lieu of stroke-time testing; a check valve monitoring program in lieu of certain requirements in Subsection ISTC of the ASME OM Code (Appendix II to the OM Code); and guidance in Subsection ISTD of the OM Code as part of meeting the ISI requirements of Section XI for snubbers. This final rule deletes a previous modification for inservice testing of containment isolation valves.

On December 3, 1997 (62 FR 63892), the NRC published a proposed rule in the Federal Register that presented an amendment to 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities," that would revise the requirements for construction, inservice inspection (ISI), and inservice testing (IST) of nuclear power plant components. For construction, the proposed amendment would have permitted the use of Section III, Division 1, of the ASME BPV Code, 1989 Addenda through the 1996 Addenda, for Class 1, Class 2, and Class 3 components with six proposed limitations and a modification.

For ISI, the proposed amendment would have required licensees to implement Section XI, Division 1, of the ASME BPV Code, 1995 Edition up to and including the 1996 Addenda for Class 1, Class 2, and Class 3 components with five proposed limitations. The proposed amendment included permission for licensees to implement Code Cases N-513, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping," and N-523, "Mechanical Clamping Devices for Class 2 and 3 Piping." The proposed

amendment also would allow licensees to use the 1992 Edition including the 1992 Addenda of Subsections IWE and IWL in lieu of updating to the 1995 Edition and the 1996 Addenda. The proposed rule included expedited implementation of Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Section XI, Division 1, with three proposed modifications. An expedited examination schedule would also have been required for a proposed modification to Section XI which addresses volumetric examination of Class 1 high pressure safety injection (HPSI) piping systems in pressurized water reactors (PWRs).

For IST, the proposed amendment would have required licensees to implement the 1995 Edition up to and including the 1996 Addenda of the ASME OM Code for Class 1, Class 2, and Class 3 pumps and valves with one limitation and one modification. The proposed amendment included permission for licensees to implement Code Case OMN-1 in lieu of stroke-time testing for motor-operated valves; Appendix II which provides a check valve condition monitoring program as an alternative to certain check valve testing requirements in Subsection ISTC of the OM Code; and Subsection ISTD of the OM Code as part of meeting the ISI requirements in Section XI for snubbers. Finally, the proposed rule would delete the modification presently in § 50.55a(b) for IST of containment isolation valves.

The NRC regulations currently require licensees to update their ISI and IST programs every 120 months to meet the version of Section XI incorporated by reference into 10 CFR 50.55a and in effect 12 months prior to the start of a new 120-month interval. The NRC published a supplement to the proposed rule on April 27, 1999 (64 FR 22580), that would eliminate the requirement for licensees to update their ISI and IST programs beyond a baseline edition and addenda of the ASME BPV Code. Under that proposed rule, licensees would continue to be allowed to update their ISI and IST programs on a voluntary basis to more recent editions and addenda of the ASME Code incorporated by reference in the regulations. Upon further review, the Commission decided to issue this final rule to incorporate by reference the 1995 Edition with the 1996 Addenda of the ASME BPV Code and the ASME OM Code with appropriate limitations and modifications. The Commission also decided to consider the proposal to eliminate the requirement to update ISI and IST programs every 120 months as a separate rulemaking effort. Following consideration of the public comments on the April 27, 1999, proposed rule, the NRC may prepare a final rule addressing the continued need for the requirement to update periodically ISI and IST programs and, if necessary, establishing an appropriate baseline edition of the ASME Code.

2. Summary of Comments

Interested parties were invited to submit written comments for consideration on the proposed rule published on December 3, 1997. Comments were received from 65 separate sources on the proposed rule. These sources consisted of 27 utilities and service organizations, the Nuclear Energy Institute (NEI), the Nuclear Utility Backfitting and Reform Group (NUBARG) represented by the firm of Winston & Strawn, the ASME Board on Nuclear Codes and Standards, the Electric Power Research Institute (EPRI), the Performance Demonstration Initiative (PDI), the Nuclear Industry Check Valve Group, the State of Illinois Department of Nuclear Safety, Oak Ridge National Laboratory, the Southwest Research Institute, three consulting firms (one firm submitted three separate letters), and 24 individuals. The commenters' concerns related principally to one or more of the proposed limitations and modifications included in the proposed rule. Many of these limitations and modifications have been renumbered in the final rule because some limitations and modifications that were contained in the proposed rule were deleted.

The proposed rule divided the proposed revisions to 10 CFR 50.55a into three groups based on the implementation schedule (i.e., 120-month update, expedited, and voluntary). These groupings have been retained in the discussion of the final rule. For each of these groups, it is indicated below in parentheses whether or not particular items are considered a backfit under 10 CFR 50.109 as discussed in Section 8, Backfit Analysis. This section provides a list of each revision and its implementation schedule, followed by a brief summary of the comments and their resolution. The summary and resolution of public comments and all of the verbatim comments which were received (grouped by subject area) are contained in Resolution of Public Comments. This document is available for inspection and copying for a fee in the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC.

2.1 List of Each Revision, Implementation Schedule, and Backfit Status.

- 120-Month Update [in accordance with §§ 50.55a(f)(4)(i) and 50.55a(g)(4)(i)]
- Section XI (Not A Backfit)

2.3.1.1 Class 1, 2, and 3 Components, Including Supports
2.3.1.2.1 Engineering Judgement (Deleted)
2.3.1.2.2 Quality Assurance
2.3.1.2.3 Class 1 Piping
2.3.1.2.4 Class 2 Piping (Deleted)
2.3.1.2.5 Reconciliation of Quality Requirements

- OM Code (Not A Backfit)

2.3.2.1 Class 1, 2, and 3 Pumps and Valves
2.3.2.3 Clarification of Scope of Safety-Related Valves Subject to IST
2.3.2.4.2 Quality Assurance
2.3.2.5.1 Motor-Operated Valve Stroke-Time Testing

- Expedited Implementation [after 6 months from the date of the final rule—Backfit]

2.4.1 Appendix VIII
2.4.1.1.1 Appendix VIII Personnel Qualification
2.4.1.1.2 Appendix VIII Specimen Set and Qualification Requirements
2.4.1.1.3 Appendix VIII Single Side Ferritic Vessel and Piping and Stainless Steel Piping Examination
2.4.3 Class 1 Piping Volumetric Examination (Deferred)

- Voluntary Implementation [may be used when final rule published—Not A Backfit]
- Section III

2.5.1.1.1 Engineering Judgement (Deleted)
2.5.1.1.2 Section III Materials
2.5.1.1.3 Weld Leg Dimensions
2.5.1.1.4 Seismic Design
2.5.1.1.5 Quality Assurance
2.5.1.1.6 Independence of Inspection
2.5.1.2.1 Applicable Code Version for New Construction

- Section XI

2.5.2.1 Subsection IWE and Subsection IWL
2.5.2.2 Flaws in Class 3 Piping; Mechanical Clamping Devices
2.5.2.3 Application of Subparagraph IWB-3740, Appendix L

- OM Code

2.5.3.1 Code Case OMN-1
2.5.3.2 Appendix II
2.5.3.3 Subsection ISTD
2.5.3.4 Containment Isolation Valves

2.2 Discussion

2.3 120-Month Update

2.3.1 Section XI

2.3.1.1 Class 1, 2, and 3 Components, Including Supports

Section 50.55a(b)(2) endorses the 1995 Edition with the 1996 Addenda of

Section XI, Division 1, for Class 1, Class 2, and Class 3 components and their supports. The proposed rule contained five limitations to address NRC positions on the use of Section XI: engineering judgment, quality assurance, Class 1 piping, Class 2 piping, and reconciliation of quality requirements. As a result of public comment, the NRC has reconsidered its positions on the use of engineering judgment and Class 2 piping. These two limitations have been eliminated from the final rule. In addition, the NRC has modified the scope of the limitation related to reconciliation of quality requirements. A discussion of each of the five proposed limitations and their comment resolution follows.

2.3.1. Limitations.

2.3.1.2.1 Engineering Judgement.

The first proposed limitation to the implementation of Section XI (§ 50.55a(b)(2)(xi) in the proposed rule) addressed an NRC position with regard to the Foreword in the 1992 Addenda through the 1996 Addenda of the BPV Code. That Foreword addresses the use of "engineering judgement" for ISI activities not specifically considered by the Code. The December 3, 1997, proposed rule contained a limitation which would have specified that licensees receive NRC approval for those activities prior to implementation.

Twenty-three commenters provided 30 separate comments on the proposed limitation to the use of engineering judgment with regard to Section XI activities. After reviewing the comments, it is apparent that the proposed rule did not accurately communicate the NRC's concerns with regard to the use of engineering judgment for Section XI activities. All of the commenters construed the limitation to prohibit the use of engineering judgment for all activities. The NRC understands that the use of engineering judgement is routinely exercised on a daily basis at each plant. It was not the NRC's intent to interject itself in this process by requiring prior approval as suggested by most commenters. The limitation was added to the proposed rule to address specific situations where engineering judgment was used and a regulatory requirement was not observed. Upon reconsideration of this issue and after reviewing all of the comments, the NRC has deleted this limitation from the final rule. The summary and the detailed discussions provided in the responses to the public comments should adequately address NRC concerns with regard to past applications of engineering judgment.

The NRC acknowledges that the use of engineering judgment is a valid and necessary part of engineering activities. However, in applying such judgment, licensees must remain cognizant of the need to assure continued compliance with regulatory requirements. Specific examples of cases where application of engineering judgment resulted in failure to satisfy regulatory requirements are discussed in detail in the Response to Public Comments, Section 2.3.1.2.1, Engineering Judgment, and Section 2.6, ASME Code Interpretations. Questions were raised by the industry regarding Interpretations, the use of engineering judgment, and related enforcement actions. At NEI's request, the NRC staff met with NEI on January 11, 1995, to discuss the use of engineering judgment and Code interpretations. On November 12, 1996, a meeting was held between representatives from the NRC and the ASME to discuss the same issues as well as the related enforcement actions. NRC Inspection Manual Part 9900, "Technical Guidance," which had been developed in response to industry questions was also discussed. The ASME representatives agreed that the NRC guidance with respect to engineering judgment was consistent with their understanding of the relationship between the ASME Code and federal regulations. The ASME stated that the NRC should not establish a formal method for reviewing ASME Code interpretations. This position was based primarily on the understanding that it would be tantamount to NRC becoming the interpreter of the Code.

It is apparent from the comments received on the proposed limitation that there is continuing confusion regarding the relationship between ASME Code requirements and NRC regulations. The NRC incorporates the ASME Code by reference into 10 CFR 50.55a. Upon adoption, the Code provisions become a part of NRC regulations as modified by other provisions in the regulations. Several commenters argued that a modification or limitation in the regulations cannot replace or overrule a Code provision or Interpretation. They also argued that, because the NRC did not accept all ASME Interpretations, the NRC was reinterpreting the Code. The NRC recognizes that the ASME is the official interpreter of the Code. However, only the NRC can determine whether the ASME Interpretation is acceptable such that it constitutes compliance with the NRC's regulations and does not adversely affect safety. The NRC cannot a priori approve Code Interpretations. While it is true that the ASME is the official interpreter of the Code, if the ASME interprets the Code in a manner which the NRC finds unacceptable (e.g., results in non-compliance with NRC regulatory requirements, a license condition, or technical specifications), the NRC can take exception to the Interpretation and is not bound by the ASME Interpretation. To put it another way, only the ASME can provide an Interpretation of the Code, but the NRC may make the determination whether that Interpretation constitutes compliance with NRC regulations. Hence, licensees need to consider the guidance on the use of Interpretations contained in the NRC Inspection Manual Part 9900, "Technical Guidance."

2.3.1.2.2 Quality Assurance.

The second proposed limitation to the implementation of Section XI [§ 50.55a(b)(2)(xii) in the proposed rule] pertained to the use of ASME Standard NQA-1, "Quality Assurance Requirements for Nuclear Facilities," with Section XI. Six comments were received and all were considered in arriving at the NRC's decision to retain the limitation as contained in the proposed rule. This limitation has been renumbered as § 50.55a(b)(2)(x) in the final rule.

As part of the licensing basis for nuclear power plants, NRC licensees have committed to certain quality assurance program provisions that are identified in both their Technical Specifications and Quality Assurance Programs. These provisions, as explained below, are taken from several sources (e.g., ASME, ANSI) and together, they constitute an acceptable Quality Assurance Program. The licensee quality assurance program commitments describe how the requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants," to 10 CFR part 50 will be satisfied by referencing applicable industry standards and the NRC Regulatory Guides (RGs) that endorsed the industry standards (e.g., the ANSI N45 series standards and applicable regulatory guides or NQA-1-1983 as endorsed by RG 1.28 (Revision 3), "Quality Assurance Program Requirements (Design and Construction)," and by prescriptive text contained in the program. Further, owners of operating nuclear power plants have committed to the additional operational phase quality assurance and administrative provisions contained in ANSI N18.7 as endorsed by RG 1.33, "Quality Assurance Program Requirements (Operations)."

Section XI references the use of either NQA-1 or the owner's Appendix B Quality Assurance Program (10 CFR part 50, Appendix B) as part of its individual provisions for a QA program. However, NQA-1 (any version) does not contain some of the quality assurance provisions and administrative controls governing operational phase activities that are contained in the ANSI standards as well as other documents which, as a group, constitute an acceptable program. When the NRC originally endorsed NQA-1, it did so with the knowledge that NQA-1 was not entirely adequate and must be supplemented by other commitments such as the ANSI standards. The later versions of NQA-1 also, by themselves, would not constitute an acceptable Quality Assurance Program. Hence, NQA-1 is not acceptable for use without the other quality assurance program provisions identified in Technical Specifications and licensee Quality Assurance Programs. The NRC staff has received questions regarding the relationship between commitments made relative to the Appendix B QA Program and Section XI as endorsed by 10 CFR 50.55a. It is apparent from public comments that there is confusion with regard to Section XI permitting the use of either NQA-1 or the owner's QA Program. The proposed limitation clarified that, when performing Section XI activities, licensees must meet other applicable NRC regulations. The limitation has been retained in the final rule to provide emphasis that licensees must comply with other applicable NRC regulations in addition to the quality assurance provisions contained in Section XI. As further clarification, the following discussion is provided.

Although not discussed in the proposed amendment to 10 CFR 50.55a, the requirements of §§ 50.34(b)(6)(ii) and 50.54(a) for establishing and revising QA Program descriptions during the operational phase are required to be followed and are not superseded or usurped by any of the requirements presently contained in 10 CFR 50.55a. Therefore, even though the present text of 10 CFR 50.55a does not take exception to applying the quality assurance provisions of NQA-1-1979 to ASME Section XI work activities, licensees of commercial nuclear power plants are required to comply not only with the QA provisions included in the Codes referenced in 10 CFR 50.55a, but also the quality assurance program developed to satisfy the requirements contained in § 50.34(b)(6)(ii). This means that, regardless of the specific quality assurance controls delineated in Section XI as referenced in 10 CFR 50.55a, licensees must meet the additional quality assurance provisions of their NRC approved quality assurance program description and other administrative controls governing operational phase activities.

2.3.1.2.3 Class 1 Piping.

The third proposed limitation to the implementation of Section XI [§ 50.55a(b)(2)(xiii) in the proposed rule] pertained to the use of Section XI, IWB-1220, "Components Exempt from Examination," that are contained in the 1989 Edition in lieu of the rules in the 1989 Addenda through the 1996 Addenda. Subparagraph IWB-1220 in these later Code addenda contain provisions from three Codes Cases: N-198-1, "Exemption from Examination for ASME Class 1 and Class 2 Piping Located at Containment Penetrations;" N-322, "Examination Requirements for Integrally Welded or Forged Attachments to Class 1 Piping at Containment Penetrations;" and N-334, "Examination Requirements for Integrally Welded or Forged Attachments to Class 2 Piping at Containment Penetrations," which the NRC found to be unacceptable. The provisions of Code Case N-198-1 were determined by the NRC to be unacceptable because industry experience has shown that welds in service-sensitive boiling water reactor (BWR) stainless steel piping, many of which are located in containment penetrations, are subjected to an aggressive environment (BWR water at reactor operating temperatures) and will experience Intergranular Stress Corrosion Cracking. Exempting these welds from examination could result in conditions which reduce the required margins to failure to unacceptable levels. The provisions of Code Cases N-322 and N-334 were determined to be unacceptable because some important piping in PWRs and BWRs was exempted from inspection. Access difficulty was the basis in the Code cases for exempting these areas from examination. However, the NRC developed the break exclusion zone design and examination criteria utilized for most containment penetration piping expecting not only that Section XI inspections would be performed but that augmented inspections would be performed. These design and examination criteria are contained in Branch Technical Position MEB 3-1, an attachment of NRC Standard Review Plan 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping."

Twenty-one comments were received on this limitation. Some commenters understood the bases for the limitation and did not believe that significant hardship would result. Many of the commenters argued that the Code cases were developed because these configurations are generally inaccessible and cannot be examined. Some argued that the piping in question is not safety significant and, thus, the examinations are unwarranted and the repairs which will be required are unnecessary.

The NRC disagrees with these comments. The provisions of § 50.55a(g)(2) require that facilities who received their construction permit on or after January 1, 1971, for Class 1 and 2 systems be designed with provisions for access for preservice inspections and inservice inspections. Several early plants with limited access have been granted plant specific relief for certain configurations. These exemptions were granted on the basis that the examinations were impractical because these plants were not designed with access to these areas. Modifications to the plant would have been required at great expense to permit examination. Therefore, narrow exceptions were granted to these early plants. For later plants, however, § 50.55a(g)(2) required that plants be constructed to provide access. The rationale for granting exemptions to early plants is not applicable to these later plants. In addition, there have been improvements in technology for the performance of examination using remote automated equipment. In designs where these welds are truly inaccessible, relief will continue to be granted when appropriate bases are provided by the licensee per § 50.55a(g)(5). With regard to the safety significance of this piping, failure of Class 1 piping within a containment penetration may lead to loss of containment integrity and an unisolable pipe break. These areas were considered break exclusion zones as part of their initial design, in part, due to the augmented examinations performed on this portion of the piping system. Further, this issue could affect the large early release frequency (LERF). For these reasons, the limitation has been retained in the final rule (§ 50.55a(b)(2)(xi)) to require licensees to use the rules for IWB-1220 that are contained in the 1989 Edition in lieu of the rules in the 1989 Addenda through the 1996 Addenda.

2.3.1.2.4 Class 2 Piping.

The fourth proposed limitation to the implementation of Section XI (§ 50.55a(b)(2)(xiv) in the proposed rule) would have confined implementation of

Section XI, IWC-1220, "Components Exempt from Examination;" IWC-1221, "Components Within RHR (Residual Heat Removal), ECC (Emergency Cool Cooling), and CHR (Containment Heat Removal) Systems or Portions of Systems;" and IWC-1222, "Components Within Systems or Portions of Systems Other Than RHR, ECC, and CHR Systems," to the 1989 Edition (i.e., it was determined that the 1989 Addenda through the 1996 Addenda were unacceptable). The provisions of Code Case N-408-3, "Alternative Rules for Examination of Class 2 Piping," were incorporated into Subsection IWC in the 1989 Addenda. These provisions contain rules for determining which Class 2 components are subject to volumetric and surface examination. The NRC limitation on the use of the Code case and its revisions has consistently been that an "applicant for an operating license should define the Class 2 piping subject to volumetric and surface examination in the Preservice Inspection for determination of acceptability by the NRC staff." Approval was required to ensure that safety significant components in the Residual Heat Removal, Emergency Core Cooling, and Containment Heat Removal systems are not exempted from appropriate examination requirements. The limitation in the proposed rule would have extended the approval required for preservice examination to inservice examination. Twenty comments were received, all disagreeing with the need for this limitation. Commenters pointed out that the information of interest is contained in the ISI program plan which is required by the Code to be submitted to the NRC. In addition, the intent of the limitation is current practice, and suitable controls are presently in place to ensure that adequate inspections of this piping are being performed. The NRC has reconsidered its bases for this limitation and agrees with the comments. Hence, the limitation has been eliminated from the final rule.

2.3.1.2.5 Reconciliation of Quality Requirements.

The fifth proposed limitation to the implementation of Section XI (§ 50.55a(b)(2)(xx) in the proposed rule) addressed reconciliation of quality requirements when implementing Section XI, IWA-4200, 1995 Addenda through the 1996 Addenda. Specifically, there were two provisions addressing the reconciliation of replacement items (§ 50.55a(b)(2)(xx)(A)) and the definition of Construction Code (§ 50.55a(b)(2)(xx)(B)). The limitation was included in the proposed rule to address the concern that, due to changes made to IWA-4200, "Items for Repair/Replacement Activities," in the 1995 Addenda, and IWA-9000, "Glossary," definition of Construction Code in the 1993 Addenda, a Section III component could be replaced with a non-Section III component, or that Construction Codes earlier than the Code of record might be used to procure components.

Twelve comments were received on the limitation. Most of the commenters stated that the limitation was too extensive; i.e., rather than taking exception to Subparagraph IWA-4200, the limitation should specifically address Subparagraph IWA-4222, "Reconciliation of Code and Owner's Requirements." Several comments suggested that the limitation be simplified to require only that "Code items shall be procured with Appendix B requirements." Additional comments were provided relating to the need to remove the limitation on the definition of Construction Code, the use of the quality provisions contained in the Construction Code, and the historical provisions contained in Section XI for reconciling of technical requirements.

The NRC has carefully reviewed the comments and agrees with the conclusions that: (1) A non-Section III item cannot be used to replace a Section III item; (2) only the same or later editions of the same Construction Code, or one that is higher in the evolutionary scale of the Code may be used; and (3) when using an earlier Construction Code, licensees must remain within the same Construction Code. The limitation has been revised in the final rule to address the reconciliation requirements contained in IWA-4222. However, changes to IWA-4222 in the 1995 Addenda specifically exempt quality assurance requirements from the reconciliation process. The various changes implemented in the 1995 Addenda, including the new definition of Construction Code, the identification of new Construction Codes, and the specific exemption to reconcile quality assurance requirements, could result in codes and standards being utilized which do not contain any quality assurance requirements, or contain quality assurance requirements which do not fully comply with Appendix B to 10 CFR part 50. Thus, the NRC has adopted the commenters' suggestion to clarify that Code items shall be procured in accordance with Appendix B requirements. Hence, when implementing the 1995 Addenda through the 1996 Addenda, the limitation (§ 50.55a(b)(2)(xvii) in the final rule) will require, in addition to the reconciliation provisions of IWA-4200, that the replacement items be purchased to the extent necessary to comply with the owner's quality assurance program description required by 10 CFR 50.34(b)(6)(ii). The rewording of the limitation addresses the NRC's concerns with regard to definitions. That portion of the proposed limitation has been eliminated from the final rule.

2.3.2 OM Code (120-Month Update).

2.3.2.1 Class 1, 2, and 3 Pumps and Valves.

This rule incorporates by reference for the first time into 10 CFR 50.55a the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code).

2.3.2.2 Background—OM Code.

Until 1990, the ASME Code requirements addressing IST of pumps and valves were contained in Section XI, Subsections IWP (pumps) and IWV (valves). The provisions of Subsections IWP and IWV were last incorporated by reference into 10 CFR 50.55a in a final rulemaking published on August 6, 1992 (57 FR 34666). In 1990, the ASME published the initial edition of the OM Code which provides rules for IST of pumps and valves. The requirements contained in the 1990 Edition are identical to the requirements contained in the 1989 Edition of Section XI, Subsections IWP (pumps) and IWV (valves). Subsequent to the publication of the 1990 OM Code, the ASME Board on Nuclear Codes and Standards (BNCS) transferred responsibility for maintenance of these rules on IST from Section XI to the OM Committee. As such, the Section XI rules for inservice testing of pumps and valves that are presently incorporated by reference into NRC regulations are no longer being updated by Section XI.

The 1990 Edition of the ASME OM Code consists of one section (Section IST) entitled "Rules for Inservice Testing of Light-Water Reactor Power Plants." This section is divided into four subsections: ISTA, "General Requirements," ISTB, "Inservice Testing of Pumps in Light-Water Reactor Power Plants," ISTC, "Inservice Testing of Valves in Light-Water Reactor Power Plants," and ISTD, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)." The testing of snubbers is governed by the ISI requirements of Section XI of the ASME BPV Code. Therefore, the rule only requires implementation of Subsections ISTA, ISTB, and ISTC. Because this final rule for the first time incorporates by reference the OM Code, the NRC has determined that the latest

endorsed Edition and Addenda of the OM Code (i.e., 1995 Edition up to and including the 1996 Addenda) should be used. Therefore, there is no need to incorporate by reference earlier Editions and Addenda of the OM Code (e.g., 1990 Edition or 1992 Edition).

2.3.2.2.1 Comments on the OM Code.

There were four commenters addressing the proposed endorsement of the OM Code. The ASME BNCS (commenter one) agreed that the action was appropriate based on the ASME moving the responsibility for developing and maintaining IST program requirements from Section XI to the OM Code. A utility (commenter two) requested clarification as to when licensees would be required to begin using the 1995 Edition with the 1996 Addenda for the OM Code. Licensees are presently required by Section XI to perform IST of pumps and valves. The regulations in 10 CFR 50.55a currently require licensees to update their IST (and ISI) programs to the latest Code incorporated by reference in § 50.55a(b) every 120 months. Hence, there is not a need to accelerate the transition to the OM Code.

A utility (commenter three) stated that changes to the OM Code that appear in the 1995 Edition with the 1996 Addenda would require their facilities to modify the test loop piping for demonstrating pump design flow rate. The NRC is aware that some licensees may have difficulty fully implementing these tests and in certain cases, due to the impracticality of implementation, a request for relief under § 50.55a(f)(5) would be appropriate. However, the OM committees developed these provisions in an effort to improve functional testing of pumps because present pump testing programs may not be capable of fully demonstrating that pumps are performing as designed. Some licensees have preoperational test loops which may be used to demonstrate full flow for this testing. Hence, the NRC has concluded that current regulatory requirements address this issue and a modification to the final rule in response to this comment is not required.

The fourth commenter (an individual) stated that the NRC was primarily responsible for the changes in the 1994 Addenda (referred to as the Comprehensive Pump Test) which will result in additional pump testing. Further, the commenter believes that the changes were more the result of pressure by the NRC than actions determined prudent by the OM committees. Hence, the conclusion is drawn that, because the changes were not instituted exclusively by the OM committees, a backfit analysis is appropriate. With respect to the addition of the Comprehensive Pump Test, the OM Code committees had decided to pursue new approaches to pump testing for a long time before its actual development. In some cases, the changes resulted in less stringent requirements or in the deletion of certain requirements. The NRC staff raised concerns with certain changes and discussed these concerns with the ASME/OM representatives in ASME/OM committee meetings. As a result, the ASME/OM decided to develop an approach to pump testing that would include a nominal "bump" test (i.e., a more frequent, but less rigorous test) complemented by a biennial "comprehensive" test (i.e., a less frequent, but more rigorous test). Subsequent changes to the 1990 OM Code were developed and adopted through a consensus process in which members of the nuclear industry are the primary participants. The NRC's position on the backfit issue is discussed in Section 8, Backfit Analysis, of the final rule, and in the response to public comments on the proposed rule. The NRC does not regard the development of the Comprehensive Pump Test to be an example of "coercion" by the NRC; rather it is an example of a properly functioning consensus process.

2.3.2.3 Clarification of Scope of Safety-Related Valves Subject to IST.

The previous language in § 50.55a(f)(1) had been interpreted by some licensees as a requirement to include all safety-related pumps and valves regardless of ASME Code Class (or equivalent) in the IST program of plants whose construction permits were issued before January 1, 1971. The NRC proposed to revise this paragraph in the draft rule amendment to clarify which safety-related pumps and valves are addressed by 10 CFR 50.55a. The intent of the revision was to ensure that the IST scope of pumps and valves for these earlier-licensed plants was similar to the scope for plants licensed after January 1, 1971. A corresponding revision was also proposed for § 50.55a(g)(1) for ISI requirements.

Fifteen separate commenters responded to the proposed clarification to § 50.55a(f)(1). During consideration of their comments, it became apparent that the proposed language in § 50.55a(f)(1) for IST did not fully accomplish its intended purpose. Instead of narrowing the IST scope of earlier-licensed plants to be consistent with the scope of later plants as intended, the proposed language inadvertently expanded the scope to include all pumps and valves in safety-related steam, water, air, and liquid-radioactive waste systems. The scope of pumps and valves to be included in IST should be dependent on the safety-related function of the component rather than the function of the system. That is, a safety-related system might include many pumps and valves. However, not all of the pumps and valves might have a safety-related function. For example, some valves in a safety-related system might be used for maintenance purposes only although they might be classified as safety-related because they are part of the safety-related system pressure boundary. Accordingly, these valves would not need to be tested under the IST program, but the welds connecting the valve to the piping might be required to be examined under the ISI program. For this reason, the NRC further concluded that, unlike the scope issue that arose in § 50.55a(f)(1) for IST, the scope issue did not apply to ISI, and a modification to the language of § 50.55a(g)(1) pertaining to ISI is not appropriate. Therefore, the existing language of § 50.55a(g)(1) will remain unchanged.

However, the need to modify the language for IST requirements exists. The final rule revises § 50.55a(f)(1) to ensure that the scope of inservice testing of pumps and valves in earlier plants is consistent with the scope applicable to later plants. This was accomplished by making the language of § 50.55a(f)(1) consistent with the scope of Paragraph 1.1 in Subsections ISTB and ISTC of the OM Code. Hence, § 50.55a(f)(1) in the final rule specifies that those pumps and valves that perform a specific function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems must meet the test requirements applicable to components which are classified as ASME Code Class 2 and Class 3 to the extent practical. The new language establishes the scope of pumps and valves that are to be included in an IST program based on the safety-related function of the pump or valve. The requirements for pumps and valves that are part of the reactor coolant pressure boundary have not been changed. This change in the regulation will clarify the scope of IST for earlier-licensed plants resulting in a more consistent scope in pump and valve IST programs for all nuclear power plants.

2.3.2.4 Limitation.

2.3.2.4.1 Quality Assurance.

The proposed rule contained one limitation (§ 50.55a(b)(3)(i)) to implementation of the OM Code addressing quality assurance (QA). This limitation pertained to the use of ASME Standard NQA-1, "Quality Assurance Requirements for Nuclear Facilities," with the OM Code. Three comments were received and all were considered in arriving at the NRC's decision to retain the limitation as contained in the proposed rule.

As part of the licensing basis for nuclear power plants, NRC licensees have committed to certain quality assurance program provisions which are identified in both their Technical Specifications and Quality Assurance Programs. These provisions are taken from several sources (e.g., ASME, ANSI) and together, they constitute an acceptable Quality Assurance Program. The licensee quality assurance program commitments describe how the requirements of appendix B to 10 CFR part 50 will be satisfied by referencing applicable industry standards and the NRC Regulatory Guides (RGs) which endorsed the industry standards (e.g., the ANSI N45 series standards and applicable regulatory guides or NQA-1-1983 as endorsed by RG 1.28, Revision 3) and by prescriptive text contained in the program. Further, owners operating nuclear power plants have committed to the additional operational phase quality assurance and administrative provisions contained in ANSI N18.7 as endorsed by RG 1.33.

The OM Code references the use of either NQA-1 or the owner's Appendix B Quality Assurance Program (10 CFR part 50, appendix B) as part of its individual provisions for a QA program. However, NQA-1 (any version) does not contain some of the quality assurance provisions and administrative controls governing operational phase activities which would be required in order to use NQA-1 in lieu of an owner's Appendix B QA Program Description. When the NRC originally endorsed NQA-1, it did so with the knowledge that NQA-1 was not entirely adequate and must be supplemented by other commitments such as the ANSI standards. The later versions of NQA-1 also, by themselves, would not constitute an acceptable Quality Assurance Program. Hence, NQA-1 is not acceptable for use without the other quality assurance program provisions identified in Technical Specifications and licensee Quality Assurance Programs. The NRC staff has received questions regarding the relationship between commitments made relative to the Appendix B QA Program and the proposed endorsement of the OM Code by 10 CFR 50.55a. It is apparent from the public comments that there is confusion with regard to the OM Code permitting the use of either NQA-1 or the owner's QA Program. The proposed limitation clarified that, when performing Section XI activities, licensees must meet other applicable NRC regulations. The limitation (§ 50.55a(b)(3)(i)) is retained in the final rule to provide emphasis that owners must comply with other applicable NRC regulations in addition to the quality provisions contained in the OM Code. The following discussion provides further clarification.

Although not discussed in the proposed amendment to 10 CFR 50.55a, the requirements of §§ 50.34(b)(6)(ii) and 50.54(a) for establishing and revising QA Program descriptions during the operational phase are required to be followed and are not superseded or usurped by any of the requirements presently contained in 10 CFR 50.55a. Therefore, even though the present text of 10 CFR 50.55a does not take exception to applying the quality provisions of NQA-1-1979 to ASME OM Code work activities, owners of commercial nuclear power plants are required to comply not only with the QA provisions included in the Codes referenced in 10 CFR 50.55a, but also the quality assurance program developed to satisfy the requirements contained in § 50.34(b)(6)(ii). This means that, regardless of the specific quality assurance controls delineated in the OM Code as referenced in 10 CFR 50.55a, owners must meet the additional quality assurance provisions of their NRC approved quality assurance program description and other administrative controls governing operational phase activities.

2.3.2.5 Modification.

2.3.2.5.1 Motor-Operated Valve Stroke-Time Testing.

The proposed rule contained a modification (§ 50.55a(b)(3)(ii)) pertaining to supplementing the stroke-time testing requirement of Subsection ISTC of the OM Code applicable for motor-operated valves (MOVs) with programs that licensees have previously committed to perform, prior to issuance of this amendment to 10 CFR 50.55a, for demonstrating the design-basis capability of MOVs. Stroke-time testing of MOVs is also specified in ASME Section XI. Seven commenters responded to the proposed change. The primary concern raised was that licensees would be required to comply with the provisions on stroke-time testing in the OM Code as well as the programs developed under their licensing commitments for demonstrating MOV design-basis capability. This might result in a duplication of activities associated with inservice testing of safety-related MOVs and the periodic verification of the design-basis capability of safety-related MOVs at nuclear power plants.

Since 1989, it has been recognized that the quarterly stroke-time testing requirements for MOVs in the Code are not sufficient to provide assurance of MOV operability under design-basis conditions. For example, in Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," the NRC stated that ASME Section XI testing alone is not sufficient to provide assurance of MOV operability under design-basis conditions. Therefore, in GL 89-10, the NRC staff requested licensees to verify the design-basis capability of their safety-related MOVs and to establish long-term MOV programs. The NRC subsequently issued GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," to provide updated guidance for establishing long-term MOV programs. Licensees have made licensing commitments pursuant to GL 96-05 that are being reviewed by the NRC staff. Most licensees have voluntarily committed to participate in an industry-wide Joint Owners Group (JOG) Program on MOV Periodic Verification. This program will help provide consistency among the individual plant long-term MOV programs.

At this time, the OM Code committees are working to update the Code with respect to its provisions for quarterly MOV stroke-time testing. For example, the ASME is considering incorporating Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants," into the OM Code. These provisions would allow users to replace quarterly MOV stroke-time testing with a combination of MOV exercising at least every refueling outage and MOV diagnostic testing on a longer interval. (The NRC has determined that, for MOVs, Code Case OMN-1 is acceptable in lieu of Subsection ISTC, with a modification. See Section 2.5.3.1 for further information.)

In light of the present weakness in the information provided by quarterly MOV stroke-time testing, this modification has been retained in the final rule. However, the NRC agrees with the

public comment that the language in the proposed rule referring to licensing commitments was cumbersome and the language has been clarified. The final rule supplements the Code requirements for MOV stroke-time testing with a provision that licensees periodically verify MOV design-basis capability. The changes to § 50.55a(b)(3)(ii) do not alter expectations regarding existing licensee commitments relating to MOV design-basis capability. Without being overly prescriptive, the final rule allows licensees to implement the regulatory requirements in a manner that best suits their particular application. The rulemaking does not require licensees to implement the JOG program on MOV periodic verification. The final rule in § 50.55a(b)(3)(iii) allows licensees the option of using ASME Code Case OMN-1 to meet the requirements of § 50.55a(b)(3)(ii).

2.4 Expedited Implementation.

2.4.1 Appendix VIII.

The proposed rule contained a requirement (§ 50.55a(g)(6)(ii)(C)) that licensees expedite implementation of mandatory Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Section XI, 1995 Edition with the 1996 Addenda. Three proposed modifications were included to address NRC positions on the use of Appendix VIII. The proposed rule would have required licensees to implement Appendix VIII for all examinations of the pressure vessel, piping, nozzles, and bolts and studs which occur after 6 months from the date of the final rule. The proposed rule would not have required any change to a licensee's ISI schedule for examination of these components, but would have required that the provisions of Appendix VIII be used for all examinations after that date.

The 1989 Addenda to Section XI added mandatory Appendix VIII to enhance the requirements for performance demonstration for ultrasonic examination (UT) procedures. In 1991, the Performance Demonstration Initiative (PDI) was organized and funded. PDI is an organization of all U. S. nuclear utilities formed for the express purpose of developing efficient, cost-effective, and technically sound implementation of the performance demonstration requirements described in the ASME Code Section XI, Appendix VIII. The EPRI NDE Center provides technical support and administration for this program on behalf of the utilities. The PDI program has been evolving. Changes to the program were being made as difficulties in implementing some Code provisions were discovered. Other changes resulted when agreements were reached on issues such as training. Finally, the program has evolved as programs were developed for each Appendix VIII supplement.

Sixty comments were received related to the proposed expedited implementation of Appendix VIII to Section XI. The issues raised by the commenters were generally uniform and narrow in scope; i.e., in agreement with the principles behind the development of Appendix VIII, but opposed to the manner in which the proposed rule would implement performance demonstration. In addition, commenters argued that implementation of Appendix VIII within 6 months from the date of the final rule was not possible because:

(1) Some Appendix VIII supplements have not yet been implemented by PDI;
(2) The number of qualified individuals is not yet sufficient;
(3) The rule would require UT personnel to requalify; and
(4) PDI's implementation of Appendix VIII differs from the Code.

The NRC staff met four times with representatives from PDI, EPRI, and NEI between the dates of May 12, 1998, and November 19, 1998, to discuss items such as the current status of the PDI program, and Appendix VIII of Section XI as modified by PDI during the development of the program. Piping, bolting, and RPV samples, for the initial phase of the program, were completed in 1994. Procedure and personnel demonstrations were initiated in April of 1994. Since that time, a large number of personnel and procedures have been qualified. However, additional time and effort will be required to complete the industry qualification process for the remaining supplements of Appendix VIII.

Subsequent to these meetings and consideration of the public comments, the NRC has reviewed the latest version of the PDI program for examination of vessels, piping, and bolting. The NRC agrees that this version will provide reasonable assurance of detecting the flaws of concern in ferritic vessels and piping. In addition, adoption in the final rule of Appendix VIII as modified by PDI during the development of the program means that the present test specimens are acceptable. The PDI program requires scanning the examination volume from both sides of the same surface of piping welds when it is accessible. Examinations performed from one side of a pipe weld may be conducted with procedures and personnel demonstrated at PDI; i.e., confirmed proficiency with single sided examinations. For the vessel weld, the volume must be examined in 4 directions from the clad-to-basemetal interface to a depth of 15 percent through-wall. Examinations performed from one side of a vessel weld may be conducted on the remaining portion of the weld volume provided the procedure shows the ability to detect flaws at angles up to 45 degrees from normal. In addition, to demonstrate equivalency to two sided examinations, the NRC staff and PDI agree that the demonstration be performed with specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel or pipe being examined. Because Appendix VIII supplements were designed for two-sided examinations, given the uniqueness in some instances of single side examinations, requalification may be necessary to demonstrate proficiency for these special cases. Single side examinations are not permitted for 15 percent of the vessel volume adjacent to the cladding, and thus cannot be used for Supplement 4 performance demonstration.

Evidence indicates that there are shortcomings in the qualifications of personnel and procedures in ensuring the reliability of nondestructive examination of the reactor vessel and other components of the reactor coolant system, the emergency core cooling systems, and portions of the steam and feedwater systems. Imposition of performance demonstration will greatly enhance the overall level of assurance of the reliability of ultrasonic examination techniques in detecting and sizing flaws. Hence, the final rule will expedite the implementation of these safety significant performance demonstration programs. The final rule will permit licensees to implement either Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Section XI, Division 1, 1995 Edition with the 1996 Addenda, or Appendix VIII as executed by PDI. Because PDI is not a consensus standards body, its program document cannot be referenced in the final rule. Thus, the PDI requirements are directly contained in the final rule in § 50.55a(b)(2)(xv).

In § 50.55a(g)(6)(ii)(C), the final rule incorporates a phased implementation of Appendix VIII over a three-year period. Licensees are required to implement the supplements to Appendix VIII according to the following schedule:

(1) Six months after the effective date of the final rule: Supplement 1,

"Evaluating Electronic Characteristics of Ultrasonic Systems," Supplement 2, "Qualification Requirements for Wrought Austenitic Piping Welds," Supplement 3, "Qualification Requirements for Ferritic Piping Welds," and Supplement 8, "Qualification Requirements for Bolts and Studs;"

(2) One year after the effective date of the final rule: Supplement 4, "Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel," and Supplement 6, "Qualification Requirements for Reactor Vessel Welds Other Than Clad/Base Metal Interface;"

(3) Two years after the effective date of the final rule: Supplement 11, "Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds;" and

(4) Three years after the effective date of the final rule: Supplement 5, "Qualification Requirements for Nozzle Inside Radius Section," Supplement 7, "Qualification Requirements for Nozzle-to-Vessel Weld," Supplement 10, "Qualification Requirements for Dissimilar Metal Piping Welds," Supplement 12, "Requirements for Coordinated Implementation of Selected Aspects of Supplements 2, 3, 10, and 11," and Supplement 13, "Requirements for Coordinated Implementation of Selected Aspects of Supplements 4, 5, 6, and 7."

Performance demonstration requirements for Supplement 9, "Qualification Requirements for Cast Austenitic Piping Welds," have not yet been initiated pending completion of the other supplements. Hence, the final rule does not address Supplement 9.

The final rule has been structured so that the equipment and procedures previously qualified under the PDI program are acceptable. Personnel previously qualified by PDI will remain qualified with the exception of a small population of individuals qualified for Supplements 4 and 6.

2.4.1.1 Modifications.

2.4.1.1.1 Appendix VIII Personnel Qualification.

The first proposed modification of Appendix VIII (§ 50.55a(b)(2)(xvii) in the proposed rule) related to its requirement that ultrasonic examination personnel meet the requirements of Appendix VII, "Qualification of Nondestructive Examination Personnel for Ultrasonic Examination," to Section XI. Appendix VII–4240 contains a requirement for personnel to receive a minimum of 10 hours of training on an annual basis. The NRC had determined that this requirement was inadequate for two reasons. The first reason was that the training does not require laboratory work and examination of flawed specimens. Signals can be difficult to interpret and, as detailed in the regulatory analysis for this rulemaking, experience and studies indicate that the examiner must practice on a frequent basis to maintain the capability for proper interpretation. The second reason is related to the length of training and its frequency. Studies have shown that an examiner's capability begins to diminish within approximately 6 months if skills are not maintained. Thus, the NRC had determined that 10 hours of annual training is not sufficient practice to maintain skills, and that an examiner must practice on a more frequent basis to maintain proper skill level. The modification in the proposed rule would have required 40 hours of annual training including laboratory work and examination of flawed specimens.

Thirty-five comments were received on this proposed modification to Appendix VIII. Many of the commenters stated that 40 hours of required training were excessive because:

(1) The EPRI NDE Center did not have the facilities which would be required to satisfy this requirement;

(2) An ample supply of training specimens would cost each site $75,000; and

(3) The requirement would result in administrative as well as cost burdens for both the utility and the vendor.

Based on the public comments and the meetings with PDI and EPRI, the NRC has reconsidered its position. The PDI program has adopted a requirement for 8 hours of training, but it is required to be hands-on practice. In addition, the training must be taken no earlier than 6 months prior to performing examinations at a licensee's facility. PDI believes that 8 hours will be acceptable relative to an examiner's abilities in this highly specialized skill area because personnel can gain knowledge of new developments, material failure modes, and other pertinent technical topics through other means. Thus, the NRC has decided to adopt in the final rule the PDI position on this matter. These changes are reflected in § 50.55a(b)(2)(xiv) of the final rule.

2.4.1.1.2 Appendix VIII Specimen Set and Qualification Requirements.

The second proposed modification of Appendix VIII (§ 50.55a(b)(2)(xviii) in the proposed rule) would have required that all flaws in the specimen sets used for performance demonstration for piping, vessels, and nozzles be cracks. For piping, Appendix VIII requires that all of the flaws in a specimen set be cracks. However, for vessels and nozzles, Appendix VIII would allow as many as 50 percent of the flaws to be notches. The NRC had previously believed that, for the purpose of demonstrating nondestructive examination (NDE) capabilities, notches are not realistic representations of service induced cracks. The flaws in the specimen sets utilized for piping by EPRI for the PDI are all cracks.

Thirty-two comments were received on this proposed modification to Appendix VIII. A majority of the commenters stated that this modification should be deleted from the rule because it would require the manufacture of new specimens and that the majority of procedure and examiner qualifications performed to date would be nullified. Many commenters argued that notches are realistic representations of cracks. Another comment was that fabrication defects should be permitted in order to test an examiner's ability to discriminate between real flaws and innocuous reflectors.

The NRC believes that flaws in test specimens used for UT should be representative of the flaws normally found or expected to be found in operating plants. Based on the public comments, the final rule in § 50.55a(b)(2)(xv) permits a population of notches and fabrication flaws on a limited basis for vessel and nozzle test specimen sets (Supplements 4, 5, 6, and 7). For these components, the NRC has concluded that a mix of cracks and notches is acceptable as long as they provide a similar detection and sizing challenge to that seen in actual service induced degradation. These types of notches will ensure that the qualification demonstration tests the ability of an examiner to discriminate between real flaws and innocuous reflectors. In addition, a mix of cracks and notches means that the present specimens can continue to be used for qualification. For wrought austenitic, ferritic, and dissimilar metal welds, however, these flaws can best be represented with cracks. Cracks span the ultrasonic spectra of flaw surface conditions from rough to smooth, jagged to straight, single to multiple tip, and tight to wide tip. Notches generally have smooth surfaces that reflect a narrow ultrasonic spectrum that represents a small population of flaws contained in components. Some variations in UT examination techniques may be more challenged with a notch located in specific locations, whereas other variations in UT examination techniques may not. With respect to

bolting, the NRC believed it would be clear that bolting was not addressed by the proposed modification. The NRC does not consider it necessary to use cracks for performance qualification for Supplement 8 as notches are appropriate reflectors in the specimen test sets.

2.4.1.1.3 Appendix VIII Single Side Ferritic Vessel and Piping and Stainless Steel Piping Examination.

The third proposed modification of Appendix VIII (§ 50.55a(b)(2)(xix) in the proposed rule) would have required that all specimens for single-side tests contain microstructures like the components to be inspected and flaws with non-optimum characteristics consistent with field experience that provide realistic challenges to the UT technique. The industry would have been required to develop specimen sets that contain microstructures similar to the types found in the components to be inspected and flaws with non-optimum characteristics (such as skew, tilt, and roughness) consistent with field experience that provide realistic challenges for single-sided performance demonstration. Appendix VIII does not distinguish specimens for two-sided examinations from those used for single-sided examination since Appendix VIII was originally developed using UT lessons learned from two-sided examinations of welds.

Thirty comments were received on this proposed modification to Appendix VIII. Many commenters stated that the NRC should delete this modification because it would invalidate the current PDI test specimens and the procedures and examiners already qualified. Another prevalent comment was that the flaws being used by PDI in vessel and piping specimens represent the microstructure and flaw orientation of postulated in-service flaws in vessel welds and, therefore, ferritic vessels should be exempted from the proposed requirement.

Based on the consideration of public comments, the final rule permits either Appendix VIII, as contained in the 1995 Edition with the 1996 Addenda, or Appendix VIII, as modified by PDI during development of the program, to be implemented. The PDI program requirements are contained in § 50.55a(b)(2)(xv). The NRC agrees that the latest version of the PDI program will provide reasonable assurance of detecting the flaws of concern in ferritic vessels and piping. In addition, adoption in the final rule of Appendix VIII as modified by PDI during the development of the PDI program means that the present test specimens are acceptable. The PDI program requires scanning the examination volume from both sides of the piping weld on the same surface when it is accessible. Examinations performed from one side of a vessel weld may be conducted with procedures and personnel demonstrated at PDI; i.e., confirmed proficiency with single sided examinations by a procedure that shows the ability to detect flaws at angles up to 45 degrees from the normal. The equipment, procedures, and personnel must demonstrate proficiency with single side examination. In addition, to demonstrate equivalency to two sided examinations, PDI requires that the demonstration be performed with specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the ferritic vessel or pipe being examined. Because Appendix VIII supplements were designed for two-sided examinations, given the uniqueness in some instances of single side examinations, requalification may be necessary to demonstrate proficiency for these special cases. Single side examinations are not permitted for 15 percent of the vessel volume adjacent to the cladding, and thus cannot be used for Supplement 4 performance demonstration.

The final rule recognizes the difficulties of performance demonstration for two sided examination of austenitic stainless steel. However, PDI does not endorse single side inspection of austenitic welds because current technology cannot consistently satisfy Appendix VIII criteria. Thus, for certain situations, the final rule in § 50.55a(b)(2)(xvi) contains criteria for demonstrating equivalency to two sided examinations.

Single side examination of wrought-to-cast stainless steel is outside the scope of the current qualification program for austenitic piping. Current technology is not reliable for detecting flaws on the opposite side of wrought-to-cast stainless steel welds. Given these shortcomings, single side examination of stainless steel piping is considered "best effort." The results of best-effort examination on the cast side of these welds is, in the NRC's view, marginal at best.

2.4.2 Generic Letter on Appendix VIII.

The proposed rule contained a summary of a draft generic letter published in the **Federal Register** for public comment on December 31, 1996 (61 FR 69120). The purpose of the generic letter was to alert the industry to the importance of using equipment, procedures, and examiners capable of reliably detecting and sizing flaws in the performance of comprehensive examinations of reactor vessels and piping. The NRC received 16 comment letters on the generic letter.

Eighteen comments were received on the summary. Many of the comments reiterated comments submitted on Appendix VIII (i.e., Section 2.4.1). Some commenters stated that the summary in the proposed rule inappropriately categorized and consolidated comments providing generalized responses to the industry's detailed comments. One commenter stated that an alternative to the proposed rule would be to mandate the use of PDI through a generic letter.

The NRC disagrees with the characterization of its consideration of the comments submitted on the generic letter. The NRC thoroughly considered each comment. Commenters generally were not in agreement with the proposed NRC action and a determination was made to withdraw the generic letter pending rulemaking. Thus, the NRC's action to withdraw the generic letter was consistent with the commenters' recommendations. The summary of the comments in the Statement of Considerations for the proposed rule was not intended to provide a detailed response to every comment received on the generic letter. The purpose of the summary was to provide some history and background related to the proposed Appendix VIII action and to alert the industry that it was the NRC's intent to withdraw the generic letter. Implementation of Appendix VIII was included in the proposed and final rules partly as a result of public comment that a generic letter should not be used to mandate new examination requirements.

2.4.3 Class 1 Piping Volumetric Examination (Deferred).

A proposed modification of Section XI (§ 50.55a(b)(2)(xv) in the proposed rule) would have required licensees of pressurized water reactor (PWR) plants to supplement the surface examination of Class 1 High Pressure Safety Injection (HPSI) system piping as required by Examination Category B–J of Table IWB–2500–1 for nominal pipe sizes (NPS) between 4 (inches) and 1+ (inches), with a volumetric (ultrasonic) examination. This requirement was proposed because:

(1) Inside diameter cracking of HPSI piping in the subject size range has been previously discovered (as detailed in NRC Generic Letter 85–20, "High Pressure Injection/Make-Up Nozzle Cracking in Babcock and Wilcox Plants," and in NRC Information Notice

97-46, "Unisolable Crack in High-Pressure Injection Piping");

(2) Failure of this line could result in a small break loss of coolant accident while directly affecting the system designed to mitigate such an event;

(3) Volumetric examinations are already required by the Code for Class 2 portions of this system (Table IWC-2500-1, Examination Category C-F-1) within the same NPS range; and

(4) Surface examinations are not highly effective in identifying cracks and flaws in piping as evidenced by events at nuclear power plants and comparisons to other examination techniques.

Implementation of this requirement was proposed to be performed during any ISI program inspection of the HPSI system performed after 6 months from the date of the final rule. Using a licensee's existing ISI schedules would result in the volumetric examinations being implemented in a reasonable period of time while not impacting lengths of outages or requiring facility shutdown solely for performance of these examinations. In light of recent industry initiatives to address Class 1 piping volumetric examination, the NRC is deferring rulemaking in this area at this time.

Fifteen comments were received on this modification to Section XI. Several concerns were raised in the comments.

(1) Volumetric examination of piping components in this size range is not very effective.

(2) Given the general ineffectiveness of volumetric examination for this piping, the occupational exposure which would be incurred outweighs the perceived need.

(3) The expedited implementation does not allow sufficient time to prepare specimen sets to comply with Appendix VIII.

(4) There was no evidence that this problem would occur in all PWRs (i.e., the concern should be limited to Babcock & Wilcox (B&W) plants which have already addressed this problem).

(5) The ASME Section XI Subcommittee on Inservice Inspection has initiated an action to address Class 1 piping.

These five concerns are addressed in order below.

As detailed in the regulatory analysis for the proposed rule, the initiation and propagation of pipe cracks at several plants have shown that surface examinations alone are not sufficient to detect the types of cracks which have occurred. It is agreed that these examinations for certain configurations may be difficult. The basic thermohydraulic phenomenon which caused the thermal fatigue cracking in the piping is well understood. However, current modeling limitations make it difficult to predict when this phenomenon will occur and at what locations. At this time, the most reliable means of detection is volumetric examination of the entire system in accordance with Section XI provisions for other Class 1 piping systems. In addition, experience has shown that, after initially discovering a section of degraded HPSI piping via leakage detection at one unit, it was possible to successfully identify similar degradation in the HPSI lines at sister units during subsequent ultrasonic examinations (in locations considered difficult to inspect). Therefore, it is the NRC's view that the usefulness of ultrasonic examinations in discovering thermal fatigue cracking in these lines has already been demonstrated in practice. Additionally, it is not clear to the NRC that the integrity of this piping can be assured in the presence of a through-wall flaw under all normal, emergency, upset, and faulted operating conditions for all PWR facilities. In short, the NRC does not believe that visual walkdowns should be the principal means of detecting leakage from pipes in these safety systems.

The NRC is aware that the imposition of any additional inspections of the reactor coolant pressure boundary may result in additional cost and/or additional worker radiation exposure depending on the plant. Some units have already implemented these examinations in response to occurrences of thermal fatigue cracking at that unit. Given the safety significance of the HPSI system (i.e., failure of this line could result in a small break loss of coolant accident while directly affecting the system designed to mitigate such an event) and the number of failures reported to date (failures have occurred in the U.S. and several foreign countries), the NRC concludes that the burden associated with such examinations is minimal.

The provisions of Appendix VIII are applicable to these examinations. The NRC staff has had several meetings with representatives from the industry's Performance Demonstration Initiative (PDI) group to discuss the status of the performance demonstration program. It is the NRC's understanding that the PDI program for piping is complete and can be implemented as soon as the administrative procedures have been developed.

The NRC does not concur that the absence of piping failures for certain portions of the HPSI system in other reactor designs precludes the need for attention to this issue in those systems at those facilities. Thermal fatigue damage attributed to diverse initiating phenomena has been reported at several facilities in the U.S. and in Europe. As discussed, it is difficult to predict when and where this phenomenon might occur. Until data consistent with the failures that occurred are determined, and the thermohydraulic phenomenon which caused the failures is reproducible by analytical means, there is limited assurance that a given analytical method will provide a reliable assessment under all potential cyclic stratification circumstances, except in special cases where the technique is obviously conservative with respect to known data. At this time, the most reliable means of detection is volumetric examination.

General Design Criterion (GDC) 14, "Reactor coolant pressure boundary," of 10 CFR part 50, appendix A, or similar provisions in the licensing basis, requires that the reactor coolant pressure boundary (of which the unisolable portions of the HPSI system are a part) be tested so as to have an extremely low probability of abnormal leakage, of propagating failure, and of gross rupture. The ASME Section XI Subcommittee on Inservice Inspection is considering the need for volumetric examination of Class 1 HPSI systems. Further, the nuclear industry has initiated a voluntary effort being coordinated by the Nuclear Energy Institute to address the issue of thermal fatigue of nuclear power plant piping. The NRC has decided to defer regulatory action on the volumetric examination of Class 1 HPSI system piping while evaluating the industry initiative and determining the need for interim action during performance of the initiative. The NRC does not believe that deferral of regulatory action in this rulemaking while evaluating the need for interim action for HPSI Class 1 weld examinations will significantly affect plant safety, because staff evaluations indicate that a minimal increase in core damage frequency would result from potentially undiscovered flaws in HPSI Class 1 piping welds over this short time period. In light of the limited benefit of surface examinations of Class 1 HPSI system piping and concerns regarding occupational radiation exposure in the performance of those examinations, this rule in § 50.55a(g)(4)(iii) endorses but does not mandate the provision in the ASME Code for surface weld examinations of Class 1 HPSI system piping.

2.5 Voluntary Implementation.

2.5.1 Section III.

The proposed rule stated that the NRC had reviewed the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section III, Division 1, for Class 1, Class 2, and Class 3 components, and had determined that they were acceptable for voluntary use with six proposed limitations. The final rule contains five limitations to the implementation of Section III. The proposed limitation on the use of engineering judgment during Section III activities has been deleted from the rule. In addition, the proposed rule stated that 10 CFR 50.55a would be modified to ensure consistency between 10 CFR 50.55a and NCA-1140. The ASME initiated an action to address this issue and requested that the NRC delete this modification from the final rule. The NRC agrees in principle with the ASME action and has deleted the modification.

The version of Section III utilized by applicants and licensees is established prior to construction as required by § 50.55a(b), (c), and (d). For operating plants, § 50.55a permits licensees to use the original construction code during the operational phase or voluntarily update to a later version which has been endorsed by 10 CFR 50.55a. Accordingly, the limitations to Section III apply to design and construction of new nuclear plants and become applicable to operating plants only if a licensee voluntarily updates to a later version.

2.5.1.1 Limitations.

2.5.1.1.1 Engineering Judgment (Deleted).

The first proposed limitation to the implementation of Section III (§ 50.55a(b)(1)(i) in the proposed rule) addressed an NRC position with regard to the Foreword in the 1992 Addenda through the 1996 Addenda of the ASME BPV Code. That Foreword addresses the use of "engineering judgement" for ISI activities not specifically considered by the Code. The proposed rule would have required licensees to receive NRC approval for those activities prior to implementation.

Twenty-three commenters provided 26 separate comments on the proposed limitation to the use of engineering judgment with regard to Section III activities. This proposed limitation has been dealt with in the same manner as the proposed limitation on the use of engineering judgment for Section XI activities. The NRC has deleted this limitation from the final rule as discussed in Section 2.3.1.2.1. The response to public comments in Section 2.3.1.2.1 addresses all of the comments which were received and provides specific examples of cases where application of engineering judgment resulted in failure to satisfy regulatory requirements.

2.5.1.1.2 Section III Materials.

The second proposed limitation to the implementation of Section III (§ 50.55a(b)(1)(ii) in the proposed rule) pertained to a reference to Part D, "Properties," of Section II, "Materials." Section II, Part D, contained many printing errors in the 1992 Edition. These errors were corrected in the 1992 Addenda. The limitation would require that Section II, 1992 Addenda, be applied when using the 1992 Edition of Section III to ensure that the design stresses intended by the ASME Code are used.

Four comments were received on the proposed limitation. One commenter agreed with the proposed action. The second commenter disagreed with the severity of the errors but had no objection to the proposed action. The third commenter stated that alerting users of the Code to such errors in a rulemaking was inappropriate. The fourth commenter argued that every version of Section II contains errors and that the NRC should recommend the use of the latest version because it contains the fewest number of errors. The limitation was not included in the proposed rule to initiate a debate over how conservative the errors were or whether the errors could cause faulty designs. There were over 160 Errata in the 1992 Edition (as identified in the 1992 Addenda) apparently because of a printing error. By comparison, there were only 16 Errata in the 1993 Addenda. The NRC was simply attempting to alert users of the Code to that fact. This limitation has been retained in the final rule to ensure that these particular design stress tables will not be used. This limitation is contained in § 50.55a(b)(1)(i) in the final rule.

2.5.1.1.3 Weld Leg Dimensions.

The third proposed limitation to the implementation of Section III [§ 50.55a(b)(1)(iii) in the proposed rule] would correct a conflict in the design and construction requirements in Subsection NB (Class 1), Subsection NC (Class 2), and Subsection ND (Class 3) of Section III, 1989 Addenda through the 1996 Addenda of the BPV Code. Two equations in NB-3683.4(c)(1), Footnote 11 to Figure NC-3673.2(b)-1, and Figure ND-3673.2(b)-1 were modified in the 1989 Addenda and are no longer in agreement with Figures NB-4427-1, NC-4427-1, and ND-4427-1. This change results in a different weld leg dimension depending on whether the dimension is derived from the text or calculated from the figures. Thus, the proposed limitation was included to ensure consistency by specifying use of the 1989 Edition for the above referenced paragraphs and figures in lieu of the 1989 Addenda through the 1996 Addenda.

Four comments were received on this proposed limitation. One commenter believed that the limitation was necessary. A second commenter believed that it was inappropriate to address Code errors in a rulemaking and this action should be accomplished through an information notice. The third commenter agreed that there appears to be a conflict, but they did not believe that the conflict would result in designs which do not satisfy the requirements and recommended deletion of the limitation. The fourth commenter stated that a conflict did not exist as a result of the changes made in the 1989 Addenda; i.e., the changes were deliberate to permit the designer an option on determining the proper weld size. However, this commenter did state that a printing error had been made in another change to the 1994 Addenda which has been corrected in the 1998 Edition.

The NRC disagrees that the limitation should be deleted from the final rule. The weld size requirements that were used in the majority of U.S. operating nuclear power plant piping systems were provided by ANSI B31.7, Nuclear Power Piping Code, ANSI B31.1, Power Piping Code, and early editions of the ASME Code, Section III. Specifically, these standards required that the minimum socket weld size equal 1.25 t but not less than ⅛ inch, where t is the nominal pipe wall thickness. The same weld size requirements as those specified in the above listed codes are also required by other nationally recognized codes and standards such as ANSI B31.3, Petroleum Refinery Piping Code. Those sizes were established as a result of many years of experience associated with the design and construction of piping systems, piping equipment, and components. In 1981, Code Case N-316, "Alternative Rules for Fillet Weld Dimensions for Socket Welded Fittings," was published permitting a reduction in socket weld sizes to 1.09 t. In essence, the Code case was developed to provide relief for certain utilities having difficulty complying with the minimum socket weld size requirement of 1.25 t. The

provisions contained in the Code case were incorporated into the 1989 Edition of the ASME Code. The NRC accepted this reduction because the new weld size was still greater than the pipe. In the 1989 Addenda of Section III of the ASME Code, the requirements for the size of socket welds were further reduced to 0.75 t which would permit welds smaller than the thickness of the pipe. The NRC is concerned with the structural integrity of a joint with a weld size which is less than the pipe wall thickness. The reduction to 0.75 t was not supported with test results or operating experience. Thus, a good technical basis has not been provided for reducing minimum socket weld sizes in nuclear power plant piping. It should be noted that the petrochemical industry has not made a corresponding change to the standards governing weld sizes in refinery piping. Hence, this limitation has been retained in § 50.55a(b)(1)(ii).

2.5.1.1.4 Seismic Design.

The fourth proposed limitation to the implementation of Section III (§ 50.55a(b)(1)(iv) in the proposed rule) pertained to new requirements for piping design evaluation contained in the 1994 Addenda through the 1996 Addenda of the ASME BPV Code. The NRC had determined that changes to articles NB-3200, "Design by Analysis," NB-3600, "Piping Design," NC-3600, "Piping Design," and ND-3600, "Piping Design," of Section III for Class 1, 2, and 3 piping design evaluation for reversing dynamic loads (e.g., earthquake and other similar type dynamic loads which cycle about a mean value) were unacceptable. The new requirements are based, in part, on industry evaluations of the test data performed under sponsorship of the EPRI and the NRC. NRC evaluations of the data do not support the changes and indicate lower margins than those estimated in earlier evaluations. The ASME has established a special working group to reevaluate the bases for the seismic design for piping.

Six comments were received on this proposed limitation to Section III. None of the commenters agreed with the proposed limitation and recommended its deletion from the final rule. The primary argument was that present seismic design of safety related piping is "overly conservative both as it relates to the seismic capacity of structures which house or support such piping as well as the potential for a reduction in overall piping safety and reliability." Several commenters stated that, while it is true that there is an ongoing review within the ASME concerning the revised criteria, the data support the revised rules.

An extensive discussion of this issue is provided in both the regulatory analysis and the response to public comments. In summary, in 1993 prior to publication of the new ASME Code rules, the NRC initiated a research program at the U.S. Department of Energy (DOE) Energy Technology Engineering Center (ETEC) to evaluate the technical basis for the Code changes, and to assess the impact of the Code changes. In December 1994, the NRC informed the ASME that there were technical concerns regarding the new criteria, and the NRC would not endorse the criteria changes in the 1994 Addenda pending the results from the research program. By letter dated May 24, 1995, the NRC restated its technical concerns, and transmitted preliminary findings from those ETEC studies which had been completed to date along with the peer review comments. After receiving comments and input from other members of the ASME BPV Code as well as representatives from other countries, the ASME established a Special Working Group—Seismic Rule (SWG-SR) in September 1995 to assess the concerns identified by the NRC and others regarding the new piping design rules, and provide a proposed resolution to address these concerns.

The ETEC efforts are now complete, and the results of the research indicate that the technical bases for the new piping design rules as published in the 1994 Addenda were incomplete. The results of the research are contained in NUREG/CR-5361, "Seismic Analysis of Piping," which was published in May 1998. The SWG-SR is considering ETEC's recommendations and is conducting some additional studies.

The NRC has concluded that additional technical bases need to be developed before the new rules could be found to be acceptable and will continue to interact via normal NRC staff participation with the Code committees. Thus, this limitation has been retained in § 50.55a(b)(1)(iii). Licensees will be permitted to use articles NB-3200, NB-3600, NC-3600, and ND-3600, in the 1989 Addenda through the 1993 Addenda, but are prohibited from using these articles as contained in the 1994 Addenda through the 1996 Addenda.

2.5.1.1.5 Quality Assurance.

The fifth proposed limitation to the implementation of Section III [§ 50.55a(b)(1)(v) in the proposed rule] pertained to the use of ASME Standard NQA-1, "Quality Assurance Requirements for Nuclear Facilities." Section III references NQA-1 as part of its individual requirements for a QA program by integrating portions of NQA-1 into the QA program defined in NCA-4000, "Quality Assurance," rather than permitting NQA-1 as a stand alone document similar to Section XI and the OM Code. Hence, even though NQA-1 by itself does not adequately describe how to satisfy the requirements of 10 CFR part 50, appendix B, the same concern does not exist regarding Section III and the use of NQA-1 as exists with Section XI. However, the limitation has been included in the final rule to provide consistency between the requirements of Section III, Section XI, and the OM Code, and to eliminate any possible confusion which could be created by not addressing the use of NQA-1 under each circumstance. The NRC had reviewed the requirements of NQA-1, 1986 Addenda through the 1992 Addenda, that are part of the incorporation by reference of Section III, and had determined that the provisions of NQA-1 are acceptable for use in the context of Section III activities. Portions of NQA-1 are integrated into Section III administrative, quality, and technical provisions which provide a complete QA program for design and construction. The additional criteria contained in Section III, such as nuclear accreditation, audits, and third party inspection, establishes a complete program and satisfies the requirements of 10 CFR part 50, appendix B (i.e., the provisions of Section III integrated with NQA-1). Licensees may voluntarily choose to apply later provisions of Section III. Hence, a limitation was included in the proposed rule which would require that the edition and addenda of NQA-1 specified by NCA-4000 of Section III be used in conjunction with the administrative, quality, and technical provisions contained in the edition of Section III being utilized.

Five comments were received on this proposed limitation. One commenter stated that the limitation was reasonable. The other commenters found the limitation confusing given that the NRC had determined that the provisions of NQA-1 were acceptable.

Section III is a design and construction code used by the manufacturers and suppliers of new Code items. However, Section III is also used for controlling the construction of replacement Code items during the operational phase at nuclear power plants. The basis for the limitation in the proposed rule was that the quality provisions contained in NQA-1 (any version) are not adequate to describe how to satisfy the applicable 10 CFR

requirements for these activities. The NRC has not taken any exceptions to the quality or administrative provisions contained in Section III. However, in the proposed limitation for Section III, the NRC emphasized that the quality provisions of NQA–1 are acceptable for use in the context of Section III activities for the construction of new and replacement Code items. Therefore, the NRC has concluded that the quality provisions contained in Section III are acceptable for the construction of new and replacement items; i.e., NQA–1 is not adequate by itself. Thus, the limitation has been retained in § 50.55a(b)(1)(iv).

2.5.1.1.6 Independence of Inspection.

The sixth proposed limitation to the implementation of Section III [§ 50.55a(b)(1)(vi) in the proposed rule] related to prohibiting licensees from using subparagraph NCA–4134.10(a), "Inspection," in the 1995 Edition through the 1996 Addenda. Before this edition and addenda, inspection personnel were prohibited from reporting directly to the immediate supervisors responsible for performing the work being inspected. However, in the 1995 Edition, NCA–4134.10(a) was modified so that independence of inspection was no longer required. This could result in noncompliance with Criterion I, "Organization," of 10 CFR part 50, appendix B. This criterion requires that persons performing QA functions report to a management level such that authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided.

Four comments were received on this limitation. One commenter stated that the proposed limitation was reasonable. The second commenter stated that this position is consistent with NRC's previous positions. The third commenter stated the change in the Code provisions had been made because the previous Code requirements exceeded the requirements of appendix B. The fourth commenter stated that there has never been a provision in appendix B that prohibited inspectors from reporting to the supervisor responsible for the work being inspected.

The NRC disagrees with both the third and fourth commenters. Criterion I, "Organization," of 10 CFR part 50, appendix B requires the establishment and execution of a quality assurance program which includes establishing and delineating in writing the authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components. In particular, Criterion I states: "These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting safety-related functions have been correctly performed." Criterion I continues by stating that "[t]he persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided." Criterion X, "Inspection," of Appendix B requires "[s]uch inspection shall be performed by individuals other than those who performed the activity being inspected."

The requirements of 10 CFR part 50, appendix B could not be met for persons performing the quality function of inspection if those persons were reporting to the individual directly responsible for meeting cost, schedule, etc. (e.g., the requirement that personnel performing quality functions, such as inspection and auditing, shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions).

As discussed in the first paragraph in this section, earlier versions of Section III contained a requirement for reporting independence. The requirement was contained in Supplement 10S–1, "Supplementary Requirements for Inspection." Supplement 10S–1, paragraph 2.1 states that, "Inspection personnel shall not report directly to the immediate supervisors who are responsible for performing the work being inspected." The Code change substitutes the more general wording in Basic Requirement 1 that applies to the overall organization. Applying this general requirement for the more specific requirements applied to independence of inspectors could promote noncompliance with established licensee QA program commitments in the absence of compensating measures. Thus, the limitation has been retained in § 50.55a(b)(1)(v). Licensees will be permitted to use the provisions contained in NCA–4134.10(a) in the 1989 Addenda through the 1994 Addenda, but will be prohibited from using these provisions as contained in the 1995 Edition through the 1996 Addenda.

2.5.1.2 Modification.

2.5.1.2.1 Applicable Code Version for New Construction.

The modification of Section III contained in the proposed rule addressed a possible conflict between NCA–1140, "Use of Code Editions, Addenda, and Cases," and 10 CFR 50.55a for new construction. NCA–1140 of Section III requires that the length of time between the date of the edition and addenda used for new construction and the docket date of the construction permit application for a nuclear power plant be no greater than three years. Section 50.55a(b)(1) requires that the edition and addenda utilized be incorporated by reference into the regulations. The possibility exists that the edition and addenda required by the ASME Code to be used for new construction would not be incorporated by reference into 10 CFR 50.55a. In order to resolve this possible discrepancy, the NRC proposed to modify existing §§ 50.55a(c)(3)(i), 50.55a(d)(2)(i), and 50.55a(e)(2)(i), to permit an applicant for a construction permit to use the latest edition and addenda which has been incorporated by reference into § 50.55a(b)(1) if the requirements of the ASME Code and the regulations cannot simultaneously be satisfied.

Three comments were received regarding this proposed modification to Section III. The ASME Board on Nuclear Codes and Standards (BNCS) agreed that there would be a conflict for new construction, but stated that the modification would preclude a Section III requirement for stamping. The BNCS recommendation was to delete this modification. The ASME is considering a Code case to resolve this by providing an alternative to NCA–1140(a)(2) which would allow an exception to this requirement when permitted by the enforcement authority. The NRC agrees with the suggested comment. The NRC, through its normal participation in the ASME committee process, will work with the appropriate ASME committees to provide an alternative when the requirements of the ASME Code and the regulations cannot simultaneously be satisfied. Hence, the proposed

modification has been deleted from the final rule.

2.5.2 Section XI (Voluntary Implementation).

The proposed rule contained provisions intended to permit licensees to voluntarily implement specific portions of the Code. One provision related to Subsection IWE and Subsection IWL of the 1995 Edition with the 1996 Addenda. Another provision related to Code Case N-513, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping," and Code Case N-523-1, "Mechanical Clamping Devices for Class 2 and 3 Piping."

2.5.2.1 Subsection IWE and Subsection IWL.

A final rule was published on August 8, 1996 (61 FR 41303), which incorporated by reference for the first time 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." The final containment rule contained a requirement for licensees to develop and implement a containment ISI program within 5 years. Some licensees have begun the development of this program. However, other licensees have expressed an interest in using later versions of the Code for this program. During review of the 1995 Edition with the 1996 Addenda, the NRC determined that the provisions contained in Subsection IWE and Subsection IWL would be acceptable when used in conjunction with the modifications contained in the final rule published on August 8, 1996 (61 FR 41303). Thus, the proposed rule contained a provision [§ 50.55a(b)(2)(vi)] to permit licensees to implement either the presently required 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda.

Twenty comments were received related to this provision. One commenter agreed with the action as proposed, and another did not object to the action but expressed a preference for the 1998 Edition. Three commenters stated that the NRC should give consideration to deferring action on this proposed amendment so that the 1998 Edition for containment ISI can be incorporated into this rulemaking. There are several provisions in Subsections IWE and IWL, 1992 Edition with the 1992 Addenda, that licensees are finding cumbersome to implement. The commenters indicated that relief requests relative to these provisions will be submitted. Because these implementation difficulties have been addressed in the 1998 Edition, incorporation of the 1998 Edition would preclude the need to seek relief. Five commenters believe that the NRC did not perform the mandatory backfit analysis for the August 8, 1996 (61 FR 41303), final rule; and, therefore, did not adequately justify its implementation. Further, the commenters believe that the NRC responses to the public comments were inadequately substantiated. Based on this, the comments argued that the proposed rule should be revised to make these subsections voluntary. Finally, one commenter believes that these subsections should be used on a trial basis before they are mandated.

The NRC has made a determination to go forward with the final rule. Given the high priority of some of the items contained in the rule, deferral of the final rule to consider the 1998 Edition for containment ISI would result in an unacceptable delay. Approval of the 1998 Edition for containment ISI would involve not only review of Subsections IWE and IWL but review of the related Code requirements such as Subsection IWA, "General Requirements," Section V, "Nondestructive Examination," and Section IX, "Welding and Brazing Qualifications." In addition, incorporation by reference of these additional Code requirements would result in the renoticing of the rule in the **Federal Register** for public comment. The NRC staff has met with NEI, EPRI, and utility representatives to discuss several industry concerns with regard to implementation of a containment ISI program. It is the NRC's understanding that these concerns can be addressed through the use of alternative examination requirements provided by an ASME Code case or the submittal of a relief request (e.g., some containment designs cannot meet Code access for examination requirements).

The NRC performed the mandatory backfit analysis for the August 8, 1996, rulemaking. Twelve commenters including NUBARG submitted comments on the documented evaluation which was performed in accordance with § 50.109(a)(4). The industry developed examination rules for containments in response to a perceived need. The reported occurrences of containment degradation and the potential for additional serious occurrences was well documented in the final rule. No technical basis has been provided for the comment that this rule should be used to revise the implementation status of Subsections IWE and IWL from mandatory to voluntary. Therefore, the provision has not been changed in the final rule. However, the proposed provision (§ 50.55a(b)(2)(ix) in the proposed rule) containing supplemental requirements for the examination of concrete containments has been renumbered as § 50.55a(b)(2)(viii) in the final rule. The proposed provision (§ 50.55a(b)(2)(x) in the proposed rule) containing supplemental requirements for the examination of metal containments and liners of concrete containments has been renumbered as § 50.55a(b)(2)(ix) in the final rule.

As licensees have begun developing their containment ISI programs, the NRC has received requests to clarify the implementation schedule for ISI of concrete containments and their post-tensioning systems. The current wording of § 50.55a(g)(6)(ii)(B)(2) requiring licensees to implement "the inservice examinations which correspond to the number of years of operation which are specified in Subsection IWL" has created confusion regarding whether the first examination of concrete is required to meet the examination schedule in Section XI, Subsection IWL, IWL-2410, which is based on the date of the Structural Integrity Test (SIT), or may be performed at any time between September 9, 1996, and September 9, 2001. In addition, the examination schedule for post-tensioning systems relative to the examination schedule for concrete was not clear. According to § 50.55a(g)(6)(ii)(B)(2) of the final rulemaking of August 8, 1996, the first examination of concrete may be performed at any time between September 9, 1996, and September 9, 2001. The intent of the rule was that, for operating plants, the date of the first examination of concrete not be linked to the date of the SIT. The first examination of concrete will set the schedule for subsequent concrete examinations. With regard to examination of the post-tensioning system, operating plants are to maintain their present 5-year schedule as they transition to Subsection IWL. For operating reactors, there is no need to repeat the 1, 3, 5-year implementation cycle.

Section 50.55a(g)(6)(ii)(B)(2) also stated that the first examination performed shall serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation. The affected plants are presently operating, but they will be performing the examination of concrete under Subsection IWL for the first time.

Because the plants are operating, a Section XI preservice examination cannot be performed. Therefore, the first concrete examination is to be an inservice examination which will serve as the baseline (the same purpose for operating plants as the preservice examination specified for plants not yet in operation). With completion of this first examination of concrete, the second 5-year ISI interval would begin. Likewise, examinations of the post-tensioning system at the nth year (e.g., the 15th year post-tensioning system examination), if performed to the requirements of Subsection IWL, are to be performed to the ISI requirements, not the preservice requirements.

The NRC has also been requested to clarify the schedule for future examinations of concrete and their post-tensioning systems at both operating and new plants. There is no requirement in Subsection IWL to perform the examination of the concrete and the examination of the post-tensioning system at the same time. The examination of the concrete under Subsection IWL and the examination of the liner plates of concrete containments under Subsection IWE may be performed at any time during the 5-year expedited implementation. This examination of the concrete and liner plate provides the baseline for comparison with future containment ISI. Coordination of these schedules in future examinations is left to each licensee. New plants would be required to follow all of the provisions contained in Subsection IWL, i.e., satisfy the preservice examination requirements and adopt the 1, 3, 5-year examination schedule linked to the Structural Integrity Test. The final rule has been clarified in § 50.55a(g)(6)(ii)(B)(*2*) with respect to the examination schedules.

The NRC has also received a request to clarify § 50.55a(g)(4)(v)(C) regarding the replacement requirements of Subsection IWL-7000 for concrete and the post-tensioning systems. Section 50.55a(g)(4)(v)(A) and (B) each state the inservice inspection, repair, and replacement requirements must be met for metal containments and metallic shell and penetration liners, respectively. However, § 50.55a(g)(4)(v)(C) states only that the inservice inspection and repair requirements applicable to concrete and the post-tensioning systems be met. This raised a question regarding whether the omission of the word "replacement" was intentional.

The intent of the rule was to require implementation of all the Articles of Subsection IWL. The failure to include "replacements" was an oversight. Section 50.55a(g)(4) requires that "* * * components which are classified as Class CC pressure retaining components and their integral attachments must meet the requirements, except for 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)." Section 50.55a(g)(4)(v)(C) has been clarified in this final rule by including "replacement" in order to eliminate any further confusion.

2.5.2.2 Flaws in Class 3 Piping.

Section 50.55a(b)(2)(xvi) in the proposed rule pertained to use of ASME Code Case N-513, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping," and Code Case N-523-1, "Mechanical Clamping Devices for Class 2 and 3 Piping." These Code cases were developed to address criteria for temporary acceptance of flaws (including through-wall leaking) of moderate energy Class 3 piping where a Section XI Code repair may be impractical for a flaw detected during plant operation (i.e., a plant shutdown would be required to perform the Code repair). In the past, licensees had to request NRC staff approval to defer Section XI Code repair for these Class 3 moderate energy (200 °F, 275 psig) piping systems. The NRC had determined that Code Case N-513 is acceptable except for the scope and Section 4.0. Code Case N-523-1 is acceptable without limitation. When using Code Case N-523-1, it should be noted that the Code case erroneously references Table NC-3321-2, rather than Table NC-3321-1 for pressure-retaining clamping devices designed by stress analysis. The use of Code Case N-513, with the limitations, and Code Case N-523-1 will obviate the need for licensees to request approval for deferring repairs; thus saving NRC and licensee resources.

Section 1.0(a) of the Scope to Code Case N-513 limits the use of the requirements to Class 3 piping. However, Section 1.0(c) would allow the flaw evaluation criteria to be applied to all sizes of ferritic steel and austenitic stainless steel pipe and tube. Without some limitation on the scope of the Code case, the flaw evaluation criteria could be applied to components such as pumps and valves, and pressure boundary leakage; applications for which the criteria should not be utilized. Thus, paragraph (B) of the proposed provision limited the use of Code Case N-513 to those applications for which it was developed.

The first paragraph of Section 4.0 of Code Case N-513 contains the flaw acceptance criteria. The criteria provide a safety margin based on service loading conditions. The second paragraph of Section 4.0, however, would permit a reduction of the safety factors based on a detailed engineering evaluation. Criteria and guidance are not provided for justifying a reduction, or limiting the amount of reduction. The NRC had determined that this provision was unacceptable because the second paragraph could permit available margins to become unacceptably low. Hence, § 50.55a(b)(2)(xvi)(A) of the proposed provision required that, when implementing Code Case N-513, the specific safety factors in the first paragraph of Section 4.0 must be satisfied.

There were seven commenters on the proposed use of these Code cases. One commenter agreed with the proposed action. Five commenters believed that the endorsement of these Code cases in a rulemaking is not appropriate. Five commenters disagreed with the limitations to Code Case N-513.

The reason for incorporating the Code cases in the proposed rule was that § 50.55a(g)(4) requires the application of Section XI during all phases of plant operation. Under Section XI structural and operability requirements, piping containing indications greater than 75 percent of the pipe thickness are deemed unsatisfactory for continued service. A limitation must be included in the rulemaking to modify the above mentioned Section XI regulatory requirements. Because regulatory guides are not mandatory, inclusion of the Code cases in Regulatory Guide 1.147 would not modify the Section XI repair requirements. In addition, the preparation of these relief requests consumes considerable industry resources, and the review and issuance consume considerable NRC staff resources. Therefore, the NRC is implementing this limited use of these Code cases through the final rule.

With regard to the limitations on the use of Code Case N-513, some commenters questioned the restrictions and believe that the Code case should be permitted in other applications such as socket welded connections. The Code case has been approved for use on moderate energy Class 3 piping and tubing (which is the ASME scope of the Code case). The NRC does not believe that the criteria are applicable to socket welds because NDE methods are not available for adequate flaw characterization. In addition, the NRC

does not agree that the level of reduction of safety margins which would be permitted by the Code case is appropriate. The margins available in an unflawed component are expected to be higher than for a degraded component. Margins less than the minimums specified for Level A, B, C, and D loading conditions are not acceptable. Hence, these restrictions have been maintained in the final rule except for the limitation related to original construction. The NRC agrees with commenters that any defects remaining from construction that have been determined by evaluation to be permissible are acceptable and has removed this limitation from the final rule. Code Cases N-513 and N-523-1 are addressed in § 50.55a(b)(2)(xiii) of the final rule.

2.5.2.3 Application of Subparagraph IWB-3740, Appendix L.

Appendix L of Subparagraph IWB-3740 permits a licensee to demonstrate that a component is acceptable with regard to cumulative fatigue effects by performing a flaw tolerance evaluation of the component as an alternative to meeting the fatigue requirements of Section III. The NRC has reviewed Appendix L and determined that its use is generally acceptable. However, licensees should be aware of the following two items, which have been under consideration by certain ASME committees and may affect future revisions of Appendix L. The first item is that the assumption of a postulated flaw with a fixed aspect ratio of 6 may not be conservative depending on the extent of cumulative usage factor (CUF) criteria exceedance along the surface of the component. The assumption of a fixed aspect ratio can have an impact on crack growth rates and projected remaining fatigue life in a component. The second item pertains to the influence of environmental effects on both fatigue usage and crack growth evaluations in Appendix L. Environmental crack growth data from laboratory studies indicate the potential for a growth rate which is different from that currently reflected in a draft Section XI Code case which has been under ASME consideration. In addition, some environmental effects data on fatigue usage are available that may be considered for a revision to Section III.

2.5.3 OM Code (Voluntary Implementation).

The proposed rule contained three provisions [§§ 50.55a(b)(3)(iii), 50.55a(b)(3)(iv), and 50.55a(b)(3)(v)] pertaining to voluntary implementation of alternatives to specific OM Code requirements. The first provision involved implementation of ASME Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants," in lieu of stroke time testing as required in Subsection ISTC, with a modification. The second provision involved implementation of a check valve condition monitoring program under Appendix II as an alternative to the testing or examination provisions contained in Subsection ISTC, with three modifications. The third provision involved use of Subsection ISTD to satisfy certain ISI requirements for snubbers provided in ASME BPV Code, Section XI. Each of these provisions is discussed separately below.

2.5.3.1 Code Case OMN-1.

Section 50.55a(b)(3)(iii) of the proposed rule addressed the voluntary implementation of Code Case OMN-1 in lieu of stroke time testing as required for motor-operated valves (MOVs) in Subsection ISTC. In particular, Code Case OMN-1 permits licensees to replace quarterly stroke-time testing of MOVs with a program of exercising on intervals of one year or one refueling outage (whichever is longer) and diagnostic testing on longer intervals. As indicated in Attachment 1 to GL 96-05, the Code case meets the intent of the generic letter, but with certain limitations which were discussed in the generic letter. For MOVs, Code Case OMN-1 is acceptable in lieu of Subsection ISTC, except for leakage rate testing (ISTC 4.3) which must continue to be performed. In addition, OMN-1 contains a maximum MOV test interval of 10 years, which the NRC supports. However, the NRC believed it prudent to include the modification requiring licensees to evaluate the information obtained for each MOV, during the first 5 years or three refueling outages (whichever is longer) of use of the Code case, to validate assumptions made in justifying a longer test interval. These conditions on the use of OMN-1 were included in the rule as a modification [§ 50.55a(b)(3)(iii)(A) in the final rule].

Paragraph 3.7 of OMN-1 discusses the use of risk insights in implementing the provisions of the Code case such as those involving MOV grouping, acceptance criteria, exercising requirements, and testing frequency. For example, Paragraph 3.6.2 of OMN-1 states that exercising more frequently than once per refueling cycle shall be considered for MOVs with high risk significance. Since the proposed rule was issued, the NRC has reviewed plant-specific requests to use OMN-1 and has determined that a clarification of the rule is appropriate regarding the provision in the Code case for the consideration of risk insights if extending the exercising frequencies for MOVs with high risk significance beyond the quarterly frequency specified in the ASME Code. In particular, licensees should ensure that increases in core damage frequency and/or risk associated with the increased exercise interval for high-risk MOVs are small and consistent with the intent of the Commission's Safety Goal Policy Statement (51 FR 30028; August 21, 1986). The NRC also considers it important for licensees to have sufficient information from the specific MOV, or similar MOVs, to demonstrate that exercising on a refueling outage frequency does not significantly affect component performance. The information may be obtained by grouping similar MOVs and staggering the exercising of MOVs in the group equally over the refueling interval. This clarification is provided in § 50.55a(b)(3)(iii)(B) of the final rule.

Thus, Code Case OMN-1 is acceptable as an optional alternative to MOV stroke-time test requirements with

(1) The modification that, at 5 years or three refueling outages (whichever is longer) from initial implementation of Code Case OMN-1, the adequacy of the test interval for each MOV must be evaluated and adjusted as necessary; and

(2) The clarification of the provision in OMN-1 for the establishment of exercise intervals for high risk MOVs in that the licensee will be expected to ensure that the potential increase in core damage frequency and risk associated with extending exercise intervals beyond a quarterly frequency is small and consistent with the intent of the Commission's Safety Goal Policy Statement.

In addition, as noted in GL 96-05, licensees are cautioned that, when implementing Code Case OMN-1, the benefits of performing a particular test should be balanced against the potential adverse effects placed on the valves or systems caused by this testing. Code Case OMN-1 specifies that an IST program should consist of a mixture of static and dynamic testing. While there may be benefits to performing dynamic testing, there are also potential detriments to its use (i.e., valve damage). Licensees should be cognizant of this for each MOV when selecting the appropriate method or combination of methods for the IST program.

Seven commenters responded to the proposed voluntary use of Code Case

OMN-1. All of the commenters agreed with the action to permit use of the Code case. However, four of the commenters did not believe that it was appropriate to do so in a rulemaking. Two commenters believe that the rule codifies individual licensee responses to Generic Letters 89-10 and 96-05 which is unnecessary. Two commenters did not believe that the NRC had adequately justified limits on the test intervals.

The proposed rule referenced Code Case OMN-1 as one method for developing a long-term MOV program that satisfies the recommendations of GL 96-05. This issue is closely related to Section 2.3.2.5.1. The amendment does not require the use of Code Case OMN-1. Licensees will be allowed the option of using the Code case as an alternative to the Code-required provisions for MOV stroke-time testing with the specified limitation and clarification. The voluntary use of Code Case OMN-1 by a licensee (in accordance with the rule and GL 96-05) would resolve weaknesses in the Code requirements for quarterly MOV stroke-time testing, and would also address the need to establish a long-term MOV program in response to GL 96-05.

With regard to the concerns that the rule would require licensees to comply with the provisions on stroke-time testing in the OM Code and also with the programs developed under their licensing commitments for demonstrating MOV design-basis capability, it has been recognized since 1989 that the quarterly stroke-time testing requirements for MOVs in the ASME Code are not sufficient to provide assurance of MOV operability under design-basis conditions. For example, in GL 89-10, the NRC stated that ASME BPV Code, Section XI, testing alone is not sufficient to provide assurance of MOV operability under design-basis conditions. Therefore, in GL 89-10, the NRC requested licensees to verify the design-basis capability of their safety-related MOVs and to establish long-term MOV programs. The NRC subsequently issued GL 96-05 to provide updated guidance for establishing long-term MOV programs. However, the NRC agrees with the public comment that the language in the proposed rulemaking referring to licensing commitments is cumbersome. The paragraph has been revised in the final rule to be performance-based to focus on maintaining MOV design-basis capability.

With regard to the question of limits on test intervals, the amendment does not limit the diagnostic test interval in Code Case OMN-1 for MOVs to 5 years or three refueling outages. In endorsing the allowable use of Code Case OMN-1, the amendment states that the adequacy of the test interval for each MOV shall be evaluated and adjusted as necessary but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of Code Case OMN-1. In other words, the amendment requires when applying Code Case OMN-1, prior to extending diagnostic test intervals for a specific MOV beyond 5 years (or three refueling outages), that the licensee evaluate test information on similar MOVs to ensure that the aging mechanisms are sufficiently understood such that the MOV will remain capable of performing its safety function over the entire diagnostic test interval. After evaluating the test information on similar MOVs, a licensee can extend the diagnostic test interval on other MOVs beyond 5 years or three refueling outages up to 10-year limit specified in Code Case OMN-1.

2.5.3.2 Appendix II.

Paragraph ISTC 4.5.5 of Subsection ISTC permits the owner to use Appendix II, "Check Valve Condition Monitoring Program," of the OM Code as an alternative to the testing or examination provisions of ISTC 4.5.1 through ISTC 4.5.4. If an owner elects to use Appendix II, the provisions of Appendix II become mandatory per OM Code requirements. However, upon reviewing the appendix, the NRC determined that the requirements in Appendix II must be supplemented in three areas. The first area is testing or examination of the check valve obturator movement to both the open and closed positions to assess its condition and confirm acceptable valve performance. Bi-directional testing of check valves was approved by the ASME OM Main Committee for inclusion in the 1996 Addenda to the Code. The NRC agrees with the need for a required demonstration of bi-directional exercising movement of the check valve disc. Single direction flow testing of check valves, as an interpreted requirement, will not always detect degradation of the valve. The classic example of this faulty testing strategy is that the departure of the disc would not be detected during forward flow tests. The departed disc could be lying in the valve bottom or another part of the system, and could move to block flow or disable another valve. Although the ASME's Working Group on Check Valves (OM Part 22) is considering Code rules for bi-directional testing of check valves, Appendix II does not presently require it. Hence, the modification in § 50.55a(b)(3)(iv)(A) was included so that an Appendix II condition monitoring program includes bi-directional testing of check valves to assess their condition and confirm acceptable valve performance (as is presently required by the OM Code).

The second area needing supplementation is the length of test interval. Appendix II would permit a licensee to extend check valve test intervals without limit. Under the current check valve IST program, most valves are tested quarterly during plant operation. The interval for certain valves has been extended to refueling outages. The NRC has concluded that operating experience exists at this time to support longer test intervals for the condition monitoring concept. A policy of prudent and safe interval extension dictates that any additional interval extension must be limited to one fuel cycle, and this extension must be based on sufficient experience to justify the additional time. Condition monitoring and current experience may qualify some valves for an initial extension to every other fuel cycle, while trending and evaluation of the data may dictate that the testing interval for some valves be reduced. Extensions of IST intervals must consider plant safety and be supported by trending and evaluating both generic and plant-specific performance data to ensure the component is capable of performing its intended function over the entire IST interval. Thus, the modification (§ 50.55a(b)(3)(iv)(B)) limits the time between the initial test or examination and second test or examination to two fuel cycles or three years (whichever is longer), with additional extensions limited to one fuel cycle. The total interval is limited to a maximum of 10 years. An extension or reduction in the interval between tests or examinations would have to be supported by trending and evaluation of performance data.

The third area in Appendix II which the NRC determined should be supplemented is the requirement applicable to a licensee who discontinues a condition monitoring program. A licensee who discontinues use of Appendix II, under Subsection ISTC 4.5.5, is required to return to the requirements of Subsection ISTC 4.5.4. However, the NRC has concluded that the requirements of ISTC 4.5.1 through ISTC 4.5.4 must be also met. Hence, if the monitoring program is discontinued, the modification [§ 50.55a(b)(3)(iv)(C)] specifies that licensees implement the provisions of ISTC 4.5.1 through ISTC 4.5.4.

Thirty-four comments were received relative to the proposed voluntary implementation of Appendix II. There were seven comments supporting the

option to utilize the requirements of Appendix II. Most of the commenters did not agree with the limitations on the use of Appendix II. However, during its June 1997 meeting, the ASME's Working Group on Check Valves (OM Part 22) identified the following issues related to Condition Monitoring (as reported in the December 1, 1997, meeting minutes) that still needed to be resolved: consideration of safety significance; trending; interval limits; step-wise interval limits; and bi-directional testing. The proposed modifications addressed these issues. Based on its interaction with OM-22, the NRC believes the ASME will address these issues in future updates of the Code.

Condition Monitoring, as described in Appendix II, is a program consisting of a general process without specified requirements, interval extension limits, and criteria. Condition Monitoring is a new Code approach with a promise of better detection of check valve degradation, improved valve performance, and maintaining reliable component capability over extended intervals, while adjusting test and examination intervals. The Condition Monitoring approach has not yet been implemented. Therefore, the nuclear industry lacks sufficient experience upon which to provide confidence of a uniform industry application of the process, or that equivalent requirements and interval extension limits will be applied, or assurance that components are capable of maintaining safe and reliable performance over extended intervals. Failure to ensure proper implementation of the process without specified requirements, interval extension limits, and criteria could result in inadvertent degradation in safety. Ensuring proper implementation could present an unwieldy compliance and inspection process for the NRC and licensees. The modifications to Appendix II contained in the rule provide for a safe and prudent progression of extending test and examination intervals consistent with historical experience and performance expectations. In addition, the modifications allow the licensee to conduct self-compliance inspections and minimize the expenditure of owner and NRC resources. Hence, the NRC has concluded that the modifications are justified and they have been retained in the final rule.

The NRC considers the Condition Monitoring approach of Appendix II for check valves to be a significant improvement over present Code requirements, and encourages licensees to implement Appendix II. Where a licensee's Code of record is an earlier edition or addenda of the ASME Code, the regulations in § 50.55a(f)(4)(iv) allow the licensee to implement portions of subsequent Code editions and addenda that are incorporated by reference in the regulations subject to the limitations and modifications listed in the rule, and subject to Commission approval. The NRC staff will favorably consider a request by a licensee under § 50.55a(f)(4)(iv) to apply Appendix II, in advance of incorporating the 1995 Edition with the 1996 Addenda of the ASME OM Code as its Code of record, if the licensee justifies the following in its submitted request:

(1) The modifications to Appendix II contained in the rule have been satisfied; and

(2) All portions of the 1995 Edition with the 1996 Addenda of the OM Code that apply to check valves are implemented for the remaining check valves not included in the Appendix II program.

2.5.3.3 Subsection ISTD.

Article IWF-5000, "Inservice Inspection Requirements for Snubbers," of the ASME BPV Code, Section XI, 1996 Addenda, requires examinations and tests of snubbers at nuclear power plants as part of the licensee's ISI program in accordance with ASME/ANSI OM, Part 4. Some licensees control testing of snubbers through plant technical specifications. Although the ASME BPV Code, Section XI, establishes ISI requirements for examination and tests of snubbers, the ASME OM Code also provides guidance on snubber examination and testing in Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants." The proposed rule (§ 50.55a(b)(3)(v)) stated that licensees may use the guidance in Subsection ISTD, OM Code, 1995 Edition with the 1996 Addenda, for testing snubbers. The final rule (§ 50.55a(b)(3)(v)) clarifies that Subsection ISTD, OM Code, 1995 Edition, up to and including the 1996 Addenda may be used to meet certain ISI requirements for snubbers provided in IWF-5000 of the ASME BPV Code, Section XI. The licensee must still meet those requirements of IWF-5000, Section XI, not included in or addressed by Subsection ISTD. Consistent with IWF-5000, the rule specifies that preservice and inservice examinations must be performed using the VT-3 visual examination method in IWA-2213.

Eleven comments were received on the endorsement of Subsection ISTD of the ASME OM Code. Seven commenters indicated that some owners have modified their Technical Specifications Snubber Surveillance Requirements to follow the provisions of GL 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions," to move the specific visual inspection and functional testing requirements to a Technical Requirements Manual. The NRC has addressed these comments in the final rule by referencing technical specifications or licensee-controlled documents for snubber test or examination requirements.

One commenter noted that Article IWF-5000, Section XI, requires examination of snubbers be performed in accordance with ASME OM-1987, Part 4. Licensees of plants with a large number of snubbers have found the required visual inspection schedule in Part 4 to be excessively restrictive. As a result, some licensees have expended a significant amount of resources and have subjected plant personnel to unnecessary radiological exposure to comply with the visual examination requirements. Many licensees have been granted relief based on application of the snubber visual inspection intervals contained in GL 90-09. The final rule allows licensees to use the snubber visual inspection interval contained in Table ISTD 6.5.2-1, "Refueling Outage-Based Visual Examination Table," Subsection ISTD, OM Code, as an alternative to the Table in OM-1987, Part 4. Table ISTD 6.5.2-1 is substantially similar to the guidance provided in GL 90-09 for snubber visual inspection intervals. The final rule should help resolve the concerns regarding the visual inspection schedule in OM-1987, Part 4.

Some commenters proposed Subsection ISTD as an acceptable alternative to the preservice and inservice examination requirements in IWF-5000, Section XI. The NRC has not accepted this suggestion because some preservice and inservice examinations for snubbers are not included in the OM Code. For example, Subsection ISTD does not address inspection of integral and non-integral attachments, such as lugs, bolting, pins, and clamps. Further, Subsection ISTD does not address snubbers in systems required to maintain the integrity of reactor coolant pressure boundary.

Section 2.5.3.3, "Subsection ISTD," of the Statement of Considerations for the proposed rule (62 FR 63903; December 3, 1997) stated that inservice testing of dynamic restraints or snubbers is governed by plant technical specifications and, thus, has never been included in 10 CFR 50.55a. It was apparent from comments received on

this section that this statement was confusing and needed to be clarified. First, it is true that 10 CFR 50.55a never directly required inservice testing of snubbers although the language in the current rule would appear to indicate otherwise. The language in the current rule states in § 50.55a(f)(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 * * * set forth in section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda * * *" (emphasis added). Although the language clearly states that "components (including supports)" are within the scope of inservice testing, and it appears that inservice testing of snubbers is included under this statement, this statement was an editorial error. In the 1992 final rule amending 10 CFR 50.55a to more clearly distinguish the requirements for inservice testing from those for inservice inspection (57 FR 34666; August 6, 1992), paragraph (g) was split into two separate paragraphs—paragraph (f) for inservice testing and paragraph (g) was retained for inservice inspection. In the 1992 final rule, similar requirements that applied to both inservice inspection and inservice testing were carried over from paragraph (f) to paragraph (g). The terminology, "components (including supports)," which existed in paragraph (g) was changed in paragraph (f) to read, "pumps and valves," except in this one instance. Therefore, the Commission views this error as an editorial oversight. In the final rule, the language in paragraph (f)(4) has been corrected to read, "pumps and valves," instead of "components (including supports)."

Based on this discussion, § 50.55a never directly required inservice testing of snubbers. However, confusion resulted because some licensees interpreted this to mean that the NRC was implying that inservice testing of snubbers was never a regulatory requirement. Inservice testing of snubbers is a regulatory requirement and has been for many years. Section 50.55a(g)(4) requires that ASME Code Class 1, 2, and 3 components (including supports) must meet the inservice inspection requirements of ASME Code, Section XI. Article IWF–5000 of Section XI, "Inservice Inspection Requirements for Snubbers," provides requirements for the examination and testing of snubbers in nuclear power plants. Therefore, inservice testing of snubbers is required by 10 CFR 50.55a because it incorporates by reference Section XI requirements including Article IWF–5000. Inservice testing of snubbers has been a requirement in IWF–5000 since Subsection IWF was first issued in the Winter 1978 Addenda of the ASME Code, Section XI.

2.5.3.4 Containment Isolation Valves.

The proposed rule contained a provision to delete the existing modification in § 50.55a(b)(2)(vii) for IST of containment isolation valves (CIVs), which was added to the regulations in a rulemaking published on August 6, 1992 (57 FR 34666). That rulemaking incorporated by reference, among other things, the 1989 Edition of ASME Section XI, Subsection IWV that endorsed part 10 of ASME/ANSI OMa–1988 for valve inservice testing. A modification to the testing requirements of part 10 related to CIVs was included in the rulemaking indicating that paragraphs 4.2.2.3(e) and 4.2.2.3(f) of part 10 were to be applied to CIVs. Since that time, the ASME OM Committee has performed a comprehensive review of OM Part 10 CIV testing requirements and acceptance standards, and has developed a basis document supporting removal of the requirements for analysis of leakage rates and corrective actions in Part 10 for those CIVs that do not provide a reactor coolant system pressure isolation function. The NRC reviewed this OM Committee basis document and determined that the modification addressing CIVs could be removed from the regulation. The requirements of 10 CFR part 50, Appendix J, ensure adequate identification analysis, and corrective actions for leakage monitoring of CIVs. There were four separate commenters on the proposed deletion of this modification and all were in agreement with the action. The final rule deletes this requirement.

2.6 ASME Code Interpretations.

The ASME issues "Interpretations" to clarify provisions of the ASME BPV and OM Codes. Requests for interpretation are submitted by users and, after appropriate committee deliberations and balloting, responses are issued by the ASME. Generally, the NRC agrees with these interpretations. However, in a few cases interpretations have been issued which conflicted with or were inconsistent with NRC requirements. Following the guidance in these interpretations resulted in noncompliance with the regulations. Some cases were discussed earlier on engineering judgment. Additional discussion is provided on the use of interpretations in the Response to Public Comments. The proposed rule contained a discussion of NRC concerns related to ASME Code Interpretations, and referenced part 9900, Technical Guidance, of the NRC Inspection Manual. Part 9900 provides that licensees should exercise caution when applying Interpretations as they are not specifically part of the incorporation by reference into 10 CFR 50.55a and have not received NRC approval.

Twenty-two comments were submitted by 21 separate commenters. Interpretations were also discussed in Sections 2.3.1.2.1 and 2.5.1.1.1 as the use of engineering judgment and interpretations is intrinsically linked. Many of the commenters believe that the NRC position on ASME Code Interpretations is inconsistent. The NRC recognizes that the ASME is the official interpreter of the Code, but the NRC will not accept ASME interpretations that, in NRC's opinion, are contrary to NRC requirements or may adversely impact facility operations. It should be noted that, considering the large number of Code interpretations that are issued, there have been very few cases where the NRC has taken exception to an ASME interpretation. Interpretations have been of great benefit in clarifying the Code. The NRC is not restricting the use of ASME Code interpretations. A proposed limitation on their use was not placed in 10 CFR 50.55a; the discussion being limited to the Statement of Considerations. The purpose of the discussion was to merely alert Code users to be prudent when applying interpretations.

As discussed in Section 2.3.1.2.1, a meeting was held on November 12, 1996, between representatives from the ASME and the NRC (in part because of the continuing questions from the industry regarding ASME interpretations). The guidance given in NRC Inspection Manual, Part 9900, regarding ASME Code interpretations was discussed. ASME representatives stated that the guidance is consistent with the ASME's understanding of the relationship between the ASME Code and NRC regulations. There were discussions regarding the mechanism for the NRC to inform the ASME of Code interpretations to which the NRC takes exception. It was agreed that the NRC should not establish a formal method for reviewing ASME Code interpretations for acceptance. This conclusion was based primarily on the understanding that it would be tantamount to the NRC becoming the interpreter of the Code. It was agreed that any concerns the NRC has regarding specific ASME Code interpretations would be brought to the ASME's attention through the NRC

staff's normal interaction with the Code. This has been routine practice for many years.

Many commenters suggested that the NRC should adopt all interpretations because the ASME is the official interpreter of the Code. The NRC cannot a priori approve interpretations as suggested. This would delegate the NRC's statutory oversight responsibility to the ASME. In addition, the NRC cannot accept an interpretation when it conflicts with regulatory requirements. Finally, an interpretation may not be accepted that changes the requirements of the Code subsequent to the NRC endorsement of a particular edition or addenda in 10 CFR 50.55a. Several commenters stated that the NRC should accept interpretations because, interpretations do not change the Code, they clarify it. As discussed in the responses to the public comments, there is evidence in a few cases to the contrary.

2.7 Direction Setting Issue 13.

The proposed rule contained a discussion of issues under consideration relative to the Commission's endorsement of ASME Codes. The first item discussed was an October 21, 1993, Cost Beneficial Licensing Action (CBLA) submittal from Entergy Operations, Inc., requesting relief from the requirement to update ISI and IST programs to the latest ASME Code edition and addenda incorporated by reference into 10 CFR 50.55a. The underlying premise of the request was that a licensee should not be required to upgrade its ISI and IST programs without considering whether the costs of the upgrade are warranted in light of the increased safety afforded by the updated Code edition and addenda. The second item discussed was the National Technology Transfer and Advancement Act of 1995, Public Law 104-113. The Act directs Federal agencies to achieve greater reliance on technical standards developed by voluntary consensus standards development organizations. The third item was Direction Setting Issue (DSI) 13, which is part of an NRC Commission Strategic Assessment and Rebaselining Initiative. The Commission has directed the NRC staff to address how industry initiatives should be evaluated, and to evaluate several issues related to NRC endorsement of industry codes and standards. As part of this evaluation, the NRC staff is addressing issues relevant to the NRC's endorsement of the ASME Code, including periodic updating, the impact of 10 CFR 50.109 (the Backfit Rule), and streamlining the process for NRC review and endorsement of the ASME Code.

Thirty-five comments were received from 21 commenters. Eight of the commenters supported NRC endorsement of the ASME Code, but submitted comments encouraging more timely endorsement. The Nuclear Energy Institute (NEI), the ASME Board on Nuclear Codes and Standards, and one utility requested that the NRC hold public meetings regarding the proposed rule. The reasons cited were: (1) Difficulties in implementing Appendix VIII as modified by the NRC; (2) concerns with the number of modifications and limitations and their content; and (3) licensee use of ASME Code editions later than 1989 should be voluntary and NRC staff endorsement need not be reflected in revisions to 10 CFR 50.55a.

With regard to the comments related to difficulties in implementing Appendix VIII as modified by the NRC, as discussed under Section 2.4.1, the NRC staff met with representatives from PDI, EPRI, and NEI on May 12, 1998, and again on June 18, 1998, to discuss items such as the current status of the PDI program, and Appendix VIII as modified during the development of the PDI program. The final rule endorses the latest version of Appendix VIII as modified by PDI during the development of the PDI program which, the NRC believes, satisfies the industry's concerns relative to this issue.

Nine commenters stated that the modifications and limitations in the proposed rule violate or are contrary to the spirit of the National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, which codified OMB Circular A-119. However, the NRC disagrees that Pub. L. 104-113 requires, without exception, the use of industry consensus standards. Section 12(d)(3) clearly allows agencies to decline to adopt voluntary consensus standards if they are inconsistent with applicable law or otherwise impractical. Furthermore, the Commission believes that it is in keeping with the intent of the Act if industry consensus standards are endorsed with limitations, rather than failing to endorse them in their entirety because of a few objectionable provisions. Ten commenters suggested that the modifications and limitations, in effect, reject the ASME consensus process. Some further suggested that many of the issues had not previously been brought to the ASME's attention. The NRC disagrees that the limitations and modifications exemplify NRC's failure to accept the consensus process of standards development. There are several examples, such as the new Section III piping seismic design criteria, which illustrate that the consensus process failed to consider the NRC representatives' comments that the bases for some of the criteria were flawed. This has been conclusively confirmed through additional testing performed by ETEC. Nearly all of the issues had previously been brought to the attention of committee members directly or as a result of public issuances such as NUREGs and generic communications.

On April 27, 1999 (64 FR 22580), the NRC published a supplement to the proposed rule dated December 3, 1997 (63 FR 63892), that would eliminate the requirement for licensees to update their ISI and IST programs beyond a baseline edition and addenda of the ASME BPV Code. Under the proposed rule, licensees would continue to be allowed to update their ISI and IST programs to more recent editions and addenda of the ASME Code incorporated by reference in the regulations. In a Staff Requirements Memorandum dated June 24, 1999, the Commission directed the NRC staff to complete expeditiously the issuance of the final rule to incorporate by reference the 1995 Edition with the 1996 Addenda of the ASME BPV Code and the ASME OM Code with appropriate limitations and modifications, and to consider the elimination of the requirement to update ISI and IST programs every 120 months as a separate rulemaking effort. The NRC is currently reviewing the public comments received on the proposed rule dated April 27, 1999. The NRC will indicate the decision regarding the need for periodic updating of ISI and IST programs and, if necessary, an appropriate baseline edition of the ASME Code following the review of public comments.

2.8 Steam Generators.

ASME Code requirements for repair of heat exchanger tubes by sleeving were added to Section XI in the 1989 Addenda. This portion of the Code contains requirements for sleeving of heat exchanger tubes by several methods (e.g., explosion welding, fusion welding, expansion, etc.). The NRC has reviewed the Code requirements for sleeving and determined that they are acceptable. However, it should be recognized that, typically, there are other relevant requirements that need to be addressed for the application of sleeving to steam generator tubing. Some of the other requirements are as follows: periodic inservice inspections, repair of sleeves containing flaws exceeding the plugging limit (i.e., tube repair criteria), structural design and operational leakage limits. All of these sleeving requirements (ASME Code and

otherwise) would need to be addressed in the technical specifications sleeving license amendment request. Thus, the NRC determination that the ASME Code sleeving requirements are acceptable should be kept in perspective.

2.9 Future Revisions of Regulatory Guides Endorsing Code Cases.

Section 50.55a indicates the ASME Code edition and addenda which have been approved for use by the NRC. In addition, Footnote 6 to 10 CFR 50.55a references NRC Regulatory Guide 1.84, "Design and Code Case Acceptability—ASME Section III Division 1," NRC Regulatory Guide 1.85, "Materials Code Case Acceptability—ASME Section III Division 1," and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI Division 1," which list the ASME Code cases that have been determined suitable by the NRC for use and may be applied to: (1) The design and construction of a particular component; or (2) the performance of inservice examination of systems and components. A determination has been made that the regulatory guide process must change in order to assure that the Code cases endorsed in the Regulatory Guides are incorporated by reference into the regulations and constitute legally-binding alternatives to the existing requirements in § 50.55a. Draft Revision 31 to Regulatory Guide 1.84, draft Revision 31 to Regulatory Guide 1.85, and draft Revision 12 to Regulatory Guide 1.147 were published for public comment in May 1997. The final regulatory guides were published in May 1999, in accordance with the present process. Future revisions to these regulatory guides, however, will be accompanied by rulemaking which will change the footnote reference to indicate the acceptable regulatory guide revisions, and to reflect approval for incorporation by reference of the endorsed Code cases by the Office of the Federal Register.

3. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104-113, requires that agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or otherwise impractical. In this final rule, the NRC is amending its regulations to incorporate by reference more recent editions and addenda of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants for construction, inservice inspection, and inservice testing as identified in the SUPPLEMENTARY INFORMATION of this document.

4. Finding of No Significant Environmental Impact

Based upon an environmental assessment, 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 will not have a significant effect on the quality of the human environment and therefore an environmental impact statement is not required.

The final rule is one part of a regulatory framework directed to ensuring pressure boundary integrity and the operational readiness of pumps and valves. The final rule incorporates provisions contained in the ASME BPV Code and the OM Code for the construction, inservice inspection, and inservice testing of components used in nuclear power plants. These provisions have been updated to incorporate improved technology and methodology. Therefore, in the general sense, the final rule would have a positive impact on the environment.

The final rule endorses ASME BPV Code, Section XI, 1995 Edition with the 1996 Addenda. As most of the technical changes to this edition/addenda merely incorporate improved technology and methodology, imposition of these requirements is not expected to either increase or decrease occupational exposure. However, imposition of paragraphs IWF-2510, Table IWF-2500-1, Examination Category F-A, and IWF-2430, will result in fewer supports being examined which will decrease the occupational exposure compared to present support inspection plans. It is estimated that an examiner receives approximately 100 millirems for every 25 supports examined. Adoption of the new provisions is expected to decrease the total number of supports to be examined by approximately 115 per unit per interval. Thus, the reduction in occupational exposure is estimated to be 460 millirems per unit each inspection interval or 50.14 rems for 109 units.

The final rule endorses the 1995 Edition with the 1996 Addenda of the ASME OM Code. The provisions of the OM Code are not expected to either increase or decrease occupational exposure. The types of testing associated with the 1995 Edition with the 1996 Addenda of the OM Code are essentially the same as the OM standards contained in the 1989 Edition of Section XI referenced in a final rule published on August 6, 1992 (57 FR 34666).

Actions by applicants and licensees in response to the final rule are of the same nature as those applicants and licensees have been performing for many years. Therefore, this action should not increase the potential for a negative environmental impact.

The Commission has determined, in accordance with the National Environmental Policy Act of 1969, as amended and the Commission's regulations in subpart A of 10 CFR part 51, that this rulemaking is not a major action significantly affecting the quality of the human environment, and, therefore, an environmental impact statement is not required. This final rule amends the NRC regulations pertaining to ISI and IST requirements for nuclear power plant components. The current regulations in 10 CFR 50.55a incorporates by reference the 1989 Edition of the ASME BPV Code, Section III, Division 1; the 1989 Edition of the ASME BPV Code, Section XI, Division 1, for Class 1, Class 2, and Class 3 components; the 1992 Edition with the 1992 Addenda of the ASME BPV Code, Section XI, Division 1, for Class MC and Class CC components; and the 1989 Edition of the ASME BPV Code, Section XI, Division 1, for Class 1, Class 2, and Class 3 pumps and valves. The Commission is amending its regulations to incorporate by reference the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section III, Division 1, of the ASME BPV Code with five limitations; the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section XI, Division 1, of the ASME BPV Code with three limitations; and the 1995 Edition and 1996 Addenda of the ASME OM Code with one limitation and one modification. The final rule imposes an expedited implementation of performance demonstration methods for ultrasonic examination systems. The final rule permits the optional implementation of the ASME Code, Section XI, provisions for surface examinations of High Pressure Safety Injection Class 1 piping welds. The final rule also permits the use of evaluation criteria for temporary acceptance of flaws in ASME Code Class 3 piping (Code Case N-523-1); mechanical clamping devices for ASME Code Class 2 and 3 piping (Code Case N-513); the 1992 Edition including the 1992 Addenda of Subsections IWE and IWL

in lieu of updating to the 1995 Edition and 1996 Addenda; alternative rules for preservice and inservice testing of certain motor-operated valve assemblies (OMN-1) in lieu of stroke-time testing; a check valve monitoring program in lieu of certain requirements in Subsection ISTC of the ASME OM Code (Appendix II to the OM Code); and guidance in Subsection ISTD of the OM Code as part of meeting the ISI requirements of Section XI for snubbers. This final rule deletes a previous modification for inservice testing of containment isolation valves. The editions and addenda of the ASME BPV Code and OM Code incorporated by reference provide updated rules for the construction of components of light-water-cooled nuclear power plants, and for the inservice inspection and inservice testing of those components. This final rule permits the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power plant components. For these reasons, the Commission concludes that this rule should have no significant adverse impact on the operation of any licensed facility or the environment surrounding these facilities.

The conclusion of this environmental assessment is that there will be no significant offsite impact to the general public from this action. However, the general public should note that the NRC has also committed to comply with Executive Order (EO) 12898, "Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations," dated February 11, 1994, in all its actions. Therefore, the NRC has also determined that there is no disproportionately high adverse impacts on minority and low-income populations. In the letter and spirit of EO 12898, the NRC is requesting public comment on any environmental justice considerations or questions that the public thinks may be related to this final rule. The NRC uses the following working definition of "environmental justice": the fair treatment and meaningful involvement of all people, regardless of race, ethnicity, culture, income, or education level with respect to the development, implementation, and enforcement of environmental laws, regulations, and policies. Comments on any aspect of the environmental assessment, including environmental justice may be submitted to the NRC.

The NRC will send a copy of this final rule including the foregoing Environmental Assessment to every State Liaison Officer.

The environmental assessment is available for inspection at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC. Single copies of the environmental assessment are available from Thomas G. Scarbrough, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Telephone: 301-415-2794, or Robert A. Hermann, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Telephone: 301-415-2768.

5. Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*). These requirements were approved by the Office of Management and Budget approval number 3150-0011.

The public reporting burden for this information collection is estimated to average 85 person-hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information.

Public Protection Notification

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

6. Regulatory Analysis

The Commission has prepared a regulatory analysis on this final regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. The analysis 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 Thomas G. Scarbrough, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Telephone: 301-415-2794, or Robert A. Hermann, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Telephone: 301-415-2768.

7. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this rule will not have a significant economic impact on a substantial number of small entities. This final rule involves the licensing and 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. Public comment received on this section suggested that the implementation of Appendix VIII of ASME BPV Code, Section XI, on performance qualification for ultrasonic testing might negatively impact small entities that contract their examination personnel to nuclear power plants. However, the final rule permits licensees to implement either Appendix VIII as contained in the 1995 Edition with the 1996 Addenda of the ASME Code, or Appendix VIII as implemented by the industry's PDI program. As a result, the NRC is unaware of any small entities in this area of expertise that are adversely affected such that they cannot satisfy either Appendix VIII as written or as implemented by PDI and endorsed in the rule.

8. Backfit Analysis

The NRC regulations in 10 CFR 50.55a require that nuclear power plant owners—

(1) Construct Class 1, Class 2, and Class 3 components in accordance with the rules provided in Section III, Division 1, "Requirements for Construction of Nuclear Power Plant Components," of the ASME BPV Code;

(2) Inspect Class 1, Class 2, Class 3, Class MC (metal containment) and Class CC (concrete containment) components in accordance with the rules provided in Section XI, Division 1, "Requirements for Inservice Inspection of Nuclear Power Plant Components," of the BPV Code; and

(3) Test Class 1, Class 2, and Class 3 pumps and valves in accordance with the rules provided in Section XI, Division 1.

The amendment to 10 CFR 50.55a endorses the 1995 Edition with the 1996 Addenda of Section XI, Division 1, of the ASME BPV Code for ISI of Class 1, Class 2, Class 3, Class MC, and Class CC components; and the 1995 Edition with the 1996 Addenda of the ASME OM Code for IST of Class 1, Class 2, and Class 3 pumps and valves. The final rule requires licensees to implement Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Section XI, Division 1, as contained in the 1995 Edition with the 1996 Addenda of the ASME BPV Code, or Appendix VIII as

modified during the development of the PDI program.

Under § 50.55a(a)(3), licensees may voluntarily update to the 1989 Addenda through the 1996 Addenda of Section III of the BPV Code, with limitations. In addition, the modification for containment isolation valve inservice testing that applied to the 1989 Edition of the BPV Code has been deleted.

The NRC regulations currently require licensees to update their ISI and IST programs every 120 months to the version of Section XI incorporated by reference into 10 CFR 50.55a 12 months prior to the start of a new 10-year interval. In the past, the NRC position has consistently been that 10 CFR 50.109 does not ordinarily require a backfit analysis of the routine 120-month update to 10 CFR 50.55a. The basis for the NRC position is that

(1) Section III, Division 1, update applies only to new construction (i.e., the edition and addenda to be used in the construction of a plant are selected based upon the date of the construction permit and are not changed thereafter, except voluntarily by the licensee);

(2) Licensees understand that 10 CFR 50.55a requires that they update their ISI and IST programs every 10 years to the latest edition and addenda of the ASME Code that were incorporated by reference in 10 CFR 50.55a and in effect 12 months before the start of the next inspection interval; and

(3) The ASME Code is a national consensus standard developed by participants with broad and varied interests where all interested parties (including the NRC and utilities) participate; the consensus process includes an examination of the cost and benefits of proposed Code revisions.

This consideration is consistent with both the intent and spirit of the backfit rule (i.e., NRC provides for the protection of the public health and safety, and does not unilaterally impose undue burden on applicants or licensees). Finally, to ensure that any interested member of the public that may not have had an opportunity to participate in the national consensus standard process is able to communicate with the NRC, proposed rules are published in the Federal Register. However, it should be noted that the Commission's initial endorsement of new subsections or appendices which would expand the scope of 10 CFR 50.55a to, *e.g.*, include components that are not presently considered by the regulation (e.g., containment structures under Subsection IWE and Subsection IWL) would be subject to the Backfit Rule, unless one or more of the exceptions to 10 CFR 50.109(a)(4) apply.

The Nuclear Utility Backfitting and Reform Group (NUBARG) and the Nuclear Energy Institute (NEI) each raised a concern with regard to the NRC's position on routine updates to 10 CFR 50.55a. Both NUBARG and NEI believe that, contrary to the NRC's determination, the routine updating of 10 CFR 50.55a to incorporate by reference new ASME Code provisions for ISI and IST constitutes a backfit for which a backfit analysis is required. The NRC has reviewed all of NUBARG's and NEI's comments in detail and has concluded that neither NUBARG nor NEI raise legal concerns which would alter the previous legal conclusion that the Backfit Rule does not require a backfit analysis of routine updates to 10 CFR 50.55a to incorporate new ASME Code ISI and IST requirements. Based on the historical evolution of the ISI requirements in 10 CFR 50.55a, the NRC believes it manifest that the "automatic update" of ISI programs under § 50.55a(g) exists in tandem with the periodic updating and endorsement of new Code editions and addenda for ISI under § 50.55a(b), and that the Commission intended that they be treated as an integrated regulatory structure for ISI which should not be subject to the Backfit Rule except in limited circumstances as discussed above. However, even though the NRC has determined that updating and endorsement of new Code editions and addenda are not subject to the Backfit Rule, the NRC is still considering these issues in the context of DSI 13. In particular, on April 27, 1999 (64 FR 22580), the NRC published a supplement to the proposed rule dated December 3, 1997 (62 FR 63892), to eliminate the requirement for licensees to update their ISI and IST programs beyond a baseline edition and addenda of the ASME BPV Code. Under that proposed rule, licensees would continue to be allowed to update their ISI and IST programs to more recent editions and addenda of the ASME Code incorporated by reference in the regulations. Upon further review, the Commission decided to complete the issuance of this final rule endorsing the 1995 Edition with the 1996 Addenda of the ASME BPV Code and the ASME OM Code with appropriate limitations and modifications and to consider the elimination of the requirement to update ISI and IST programs every 120 months as a separate rulemaking effort. Following consideration of the public comments on the April 27, 1999, proposed rule, the NRC may prepare a final rule addressing the continued need for the requirement to update periodically ISI and IST programs and, if necessary, establishing an appropriate baseline edition of the ASME Code.

The provisions for IST of pumps and valves were originally contained in Section XI Subsections IWP and IWV of the ASME BPV Code, but have now been moved by ASME to a new OM Code. Section XI, 1989 Edition was incorporated by reference in the August 6, 1992, rulemaking (57 FR 34666). The 1990 OM Code standards, Parts 1, 6, and 10 of ASME/ANSI-OM-1987, are identical to Section XI, 1989 Edition. This amendment is an administrative change simply referencing the 1995 Edition with the 1996 Addenda of the OM Code. Therefore, imposition of the 1995 Edition with the 1996 Addenda of the OM Code is not a backfit.

Appendix VIII to ASME BPV Code, Section XI, or Appendix VIII as modified during the development of the PDI program will be used to demonstrate the qualification of personnel and procedures for performing nondestructive examination of welds in components of systems that include the reactor coolant system and the emergency core cooling systems in nuclear power facilities. These performance demonstration programs will greatly increase the reliability of detection and sizing of cracks and flaws. Current requirements have been demonstrated not to be able to consistently and accurately identify and size cracks and flaws and thus are not effective. The Appendix delineates a method for qualification of the personnel and procedures. Appendix VIII changes the Code rules from a prescriptive set of requirements to a performance based approach that allows for implementation of improved technology without changes to the regulations. Performance demonstration would normally be imposed by the 120-month update requirement but, because of its importance, implementation of Appendix VIII is being expedited by the rulemaking. Because of the fundamental change in the nature of the qualification requirements, Appendix VIII is being considered a backfit. The proposed rule would have required licensees to implement Appendix VIII, including the modifications, for all examinations of the pressure vessel, piping, nozzles, and bolts and studs which occur after 6 months from the date of the final rule. However, based on public comment, the final rule adopts a phased implementation approach for Appendix VIII, ranging from 6 months to 3 years, depending on the supplement. The final rule will not require any change to a licensee's ISI schedule for examination of these components, but will require

that the provisions of Appendix VIII as contained in the 1995 Edition with the 1996 Addenda (as supplemented by the final rule) or Appendix VIII as modified during the development of the PDI program (as supplemented by the final rule) be used for all examinations after that date rather than the UT procedures and personnel requirements presently being utilized by licensees.

On the basis of the documented evaluation required by § 50.109(a)(4), the NRC has concluded that imposition of Appendix VIII is necessary to bring the facilities described into compliance with GDC 14, 10 CFR Part 50, Appendix A, or similar provisions in the licensing basis for these facilities, and Criterion II, "Quality Assurance Program," and Criterion XVI, "Corrective Actions," of appendix B to 10 CFR part 50. Criterion II requires, in part, that a QA program shall take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality and the need for verification of quality by inspection and test. Evidence indicates that there are shortcomings in the qualifications of personnel and procedures in ensuring the reliability of the examinations. These safety significant revisions to the Code include specific requirements for UT performance demonstration, with statistically based acceptance criteria for blind testing of UT systems (procedures, equipment, and personnel) used to detect and size flaws. Criterion XVI requires that measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected. Because of the serious degradation which has occurred, and the belief that additional occurrences of noncompliance with GDC 14, and Criteria II and XVI will occur, the NRC has determined that imposition of Appendix VIII beginning 6 months after the final rule has been published under the compliance exception to § 50.109(a)(4)(i) is appropriate. Therefore, a backfit analysis is not required and the cost-benefit standards of § 50.109(a)(3) do not apply. A complete discussion is contained in the documented evaluation.

The rationale for application of the backfit rule and the backfit justification for the various items contained in this final rule are contained in the regulatory analysis and documented evaluation. The regulatory analysis and documented evaluation are available for inspection at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC. Single copies of the regulatory analysis and documented evaluation are available from Thomas G. Scarbrough, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–2794, or Robert A. Hermann, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–2768.

9. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

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. 552 and 553, the NRC is adopting 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: Sections 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. 444, 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. 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 as follows:

a. By removing paragraph (b)(2)(vii);

b. By redesignating and revising paragraphs (b)(2)(viii), (b)(2)(ix), and (b)(2)(x) as (b)(2)(vii), (b)(2)(viii), and (b)(2)(ix), respectively;

c. By adding paragraphs (b)(1)(i) through (b)(1)(v), (b)(2)(x) through (b)(2)(xvii), (b)(3), (g)(4)(iii), and (g)(6)(ii)(C); and

d. By revising the introductory paragraph, the introductory text of paragraph (b), paragraph (b)(1), the introductory text of paragraph (b)(2), paragraph (b)(2)(vi), the introductory text of paragraph (f), paragraphs (f)(1), the introductory text of paragraph (f)(3), paragraphs (f)(3)(iii), (f)(3)(iv), the introductory text of paragraph (f)(4), paragraph (g)(1), the introductory text of paragraph (g)(3), paragraph (g)(3)(i), the introductory paragraph of (g)(4), and paragraphs (g)(4)(v)(C), (g)(6)(ii)(B)(*1*), and (g)(6)(ii)(B)(*2*), to read as follows:

§ 50.55a Codes and standards.

Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section and each construction permit for a utilization facility is subject to the following conditions in addition to those specified in § 50.55.

* * * * *

(b) The ASME Boiler and Pressure Vessel Code, and the ASME Code for Operation and Maintenance of Nuclear Power Plants, which are referenced in the following paragraphs, were approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the **Federal Register**. Copies of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016. They are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

Copies are also available at the Office of the Federal Register, 800 N. Capitol Street, Suite 700, Washington, DC.

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions through the 1995 Edition and addenda through the 1996 Addenda, subject to the following limitations and modifications:

(i) *Section III Materials.* When applying the 1992 Edition of Section III, licensees must apply the 1992 Edition with the 1992 Addenda of Section II of the ASME Boiler and Pressure Vessel Code.

(ii) *Weld leg dimensions.* When applying the 1989 Addenda through the 1996 Addenda of Section III, licensees may not apply paragraph NB-3683.4(c)(1), Footnote 11 to Figure NC-3673.2(b)-1, and Figure ND-3673.2(b)-1.

(iii) *Seismic design.* Licensees may use Articles NB-3200, NB-3600, NC-3600, and ND-3600 up to and including the 1993 Addenda, subject to the limitation specified in paragraph (b)(1)(ii) of this section. Licensees shall not use these Articles in the 1994 Addenda through the 1996 Addenda.

(iv) *Quality assurance.* When applying editions and addenda later than the 1989 Edition of Section III, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1986 Edition through the 1992 Edition, are acceptable for use provided that the edition and addenda of NQA-1 specified in NCA-4000 is used in conjunction with the administrative, quality, and technical provisions contained in the edition and addenda of Section III being used.

(v) *Independence of inspection.* Licensees may not apply NCA-4134.10(a) of Section III, 1995 Edition with the 1996 Addenda.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1, and include editions through the 1995 Edition and addenda through the 1996 Addenda, subject to the following limitations and modifications:

* * * * *

(vi) *Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI.* Licensees may use either the 1992 Edition with the 1992 Addenda or the 1995 Edition with the 1996 Addenda of Subsection IWE and Subsection IWL as modified and supplemented by the requirements in § 50.55a(b)(2)(viii) and § 50.55a(b)(2)(ix) when implementing the containment inservice inspection requirements of this section.

(vii) *Section XI References to OM Part 4, OM Part 6 and OM Part 10 (Table IWA-1600-1).* When using Table IWA-1600-1, "Referenced Standards and Specifications," in the Section XI, Division 1, 1987 Addenda, 1988 Addenda, or 1989 Edition, the specified "Revision Date or Indicator" for ASME/ANSI OM Part 4, ASME/ANSI Part 6, and ASME/ANSI Part 10 must be the OMa-1988 Addenda to the OM-1987 Edition. These requirements have been incorporated into the OM Code which is incorporated by reference in paragraph (b)(3) of this section.

(viii) *Examination of concrete containments.* Licensees applying Subsection IWL, 1992 Edition with the 1992 Addenda, shall apply all of the modifications in this paragraph. Licensees choosing to apply the 1995 Edition with the 1996 Addenda shall apply paragraphs (b)(2)(viii)(A), (viii)(D)(*3*), and (viii)(E) of this section.

(A) 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) 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, an evaluation must be performed and reported in the Engineering Evaluation Report as prescribed in IWL-3300.

(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 anchorage. A difference of more than 10 percent must be identified in the ISI Summary Report required by IWA-6000.

(D) The licensee shall report the following conditions, if they occur, in the ISI Summary Report required by IWA-6000:

(*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 exceeds 10 percent of the tendon net duct volume;

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

(E) For Class CC applications, 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 required by IWA-6000:

(*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.

(ix) *Examination of metal containments and the liners of concrete containments.*

(A) For Class MC applications, 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 as required by IWA-6000:

(*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.

(B) When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

(C) The examinations specified in Examination Category E-B, Pressure Retaining Welds, and Examination Category E-F, Pressure Retaining Dissimilar Metal Welds, are optional.

(D) Section 50.55a(b)(2)(ix)(D) may be used as an alternative to the requirements of IWE-2430.

(*1*) If the examinations reveal flaws or areas of degradation exceeding the acceptance standards of Table IWE-3410-1, an evaluation must be performed to determine whether additional component examinations are required. For each flaw or area of degradation identified which exceeds acceptance standards, the licensee shall

provide the following in the ISI Summary Report required by IWA-6000:

(*1*) A description of each flaw or area, including the extent of degradation, and the conditions that led to the degradation;

(*ii*) The acceptability of each flaw or area, and the need for additional examinations to verify that similar degradation does not exist in similar components, and;

(*iii*) A description of necessary corrective actions.

(*2*) The number and type of additional examinations to ensure detection of similar degradation in similar components.

(E) A general visual examination as required by Subsection IWE must be performed once each period.

(x) *Quality Assurance.* When applying Section XI editions and addenda later than the 1989 Edition, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1979 Addenda through the 1989 Edition, are acceptable as permitted by IWA-1400 of Section XI, if the licensee uses its 10 CFR Part 50, Appendix B, quality assurance program, in conjunction with Section XI requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA-1 must govern Section XI activities. Further, where NQA-1 and Section XI do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to Section XI activities.

(xi) *Class 1 piping.* Licensees may not apply IWB-1220, "Components Exempt from Examination," of Section XI, 1989 Addenda through the 1996 Addenda, and shall apply IWB-1220, 1989 Edition.

(xii) Reserved.

(xiii) *Flaws in Class 3 Piping.* Licensees may use the provisions of Code Case N-513, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping," Revision 0, and Code Case N-523-1, "Mechanical Clamping Devices for Class 2 and 3 Piping." Licensees choosing to apply Code Case N-523-1 shall apply all of its provisions. Licensees choosing to apply Code Case N-513 shall apply all of its provisions subject to the following:

(A) When implementing Code Case N-513, the specific safety factors in paragraph 4.0 must be satisfied.

(B) Code Case N-513 may not be applied to:

(*1*) Components other than pipe and tube, such as pumps, valves, expansion joints, and heat exchangers;

(*2*) Leakage through a flange gasket;

(*3*) Threaded connections employing nonstructural seal welds for leakage prevention (through seal weld leakage is not a structural flaw, thread integrity must be maintained); and

(*4*) Degraded socket welds.

(xiv) *Appendix VIII personnel qualification.* All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Appendix VIII specimen set and qualification requirements.* The following provisions may be used to modify implementation of Appendix VIII of Section XI, 1995 Edition with the 1996 Addenda. Licensees choosing to apply these provisions shall apply all of the provisions except for those in § 50.55a(b)(2)(xv)(F) which are optional.

(A) When applying Supplements 2 and 3 to Appendix VIII, the following examination coverage criteria requirements must be used:

(*1*) Piping must be examined in two axial directions and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available.

(*2*) Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds, full coverage credit from a single side may be claimed only after completing a successful single sided Appendix VIII demonstration using flaws on the opposite side of the weld.

(B) The following provisions must be used in addition to the requirements of Supplement 4 to Appendix VIII:

(*1*) Paragraph 3.1, Detection acceptance criteria—Personnel are qualified for detection if the results of the performance demonstration satisfy the detection requirements of ASME Section XI, Appendix VIII, Table VIII-S4-1 and no flaw greater than 0.25 inch through wall dimension is missed.

(*2*) Paragraph 1.1(c), Detection test matrix—Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB-3500, need not be included as detection flaws. For procedures applied from the inside surface, use the minimum thickness specified in the scope of the procedure to calculate a/t. For procedures applied from the outside surface, the actual thickness of the test specimen is to be used to calculate a/t.

(C) When applying Supplement 4 to Appendix VIII, the following provisions must be used:

(*1*) A depth sizing requirement of 0.15 inch RMS shall be used in lieu of the requirements in Subparagraphs 3.2(a) and 3.2(b).

(*2*) In lieu of the location acceptance criteria requirements of Subparagraph 2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the flaw type requirements of Subparagraph 1.1(e)(1), a minimum of 70 percent of the flaws in the detection and sizing tests shall be cracks. Notches, if used, must be limited by the following:

(*i*) Notches must be limited to the case where examinations are performed from the clad surface.

(*ii*) Notches must be semielliptical with a tip width of less than or equal to 0.010 inches.

(*iii*) Notches must be perpendicular to the surface within ± 2 degrees.

(*4*) In lieu of the detection test matrix requirements in paragraphs 1.1(e)(2) and 1.1(e)(3), personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(D) The following provisions must be used in addition to the requirements of Supplement 6 to Appendix VIII:

(*1*) Paragraph 3.1, Detection Acceptance Criteria—Personnel are qualified for detection if:

(*i*) No surface connected flaw greater than 0.25 inch through wall has been missed.

(*ii*) No embedded flaw greater than 0.50 inch through wall has been missed.

(*2*) Paragraph 3.1, Detection Acceptance Criteria—For procedure qualification, all flaws within the scope of the procedure are detected.

(*3*) Paragraph 1.1(b) for detection and sizing test flaws and locations—Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB-3500, need not be included as detection flaws. Flaws which are less than the allowable flaw size, as defined in IWB-3500, may be used as detection and sizing flaws.

(*4*) Notches are not permitted.

(E) When applying Supplement 6 to Appendix VIII, the following provisions must be used:

(*1*) A depth sizing requirement of 0.25 inch RMS must be used in lieu of the requirements of subparagraphs 3.2(a), 3.2(c)(2), and 3.2(c)(3).

(*2*) In lieu of the location acceptance criteria requirements in Subparagraph

2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the length sizing criteria requirements of Subparagraph 3.2(b), a length sizing acceptance criteria of 0.75 inch RMS must be used.

(*4*) In lieu of the detection specimen requirements in Subparagraph 1.1(e)(1), a minimum of 55 percent of the flaws must be cracks. The remaining flaws may be cracks or fabrication type flaws, such as slag and lack of fusion. The use of notches is not allowed.

(*5*) In lieu of paragraphs 1.1(e)(2) and 1.1(e)(3) detection test matrix, personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(F) The following provisions may be used for personnel qualification for combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII qualification. Licensees choosing to apply this combined qualification shall apply all of the provisions of Supplements 4 and 6 including the following provisions:

(*1*) For detection and sizing, the total number of flaws must be at least 10. A minimum of 5 flaws shall be from Supplement 4, and a minimum of 50 percent of the flaws must be from Supplement 6. At least 50 percent of the flaws in any sizing must be cracks. Notches are not acceptable for Supplement 6.

(*2*) Examination personnel are qualified for detection and length sizing when the results of any combined performance demonstration satisfy the acceptance criteria of Supplement 4 to Appendix VIII.

(*3*) Examination personnel are qualified for depth sizing when Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII flaws are sized within the respective acceptance criteria of those supplements.

(G) When applying Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification, the following additional provisions must be used, and examination coverage must include:

(*1*) The clad to base metal interface, including a minimum of 15 percent T (measured from the clad to base metal interface), shall be examined from four orthogonal directions using procedures and personnel qualified in accordance with Supplement 4 to Appendix VIII.

(*2*) If the clad-to-base-metal-interface procedure demonstrates detectability of flaws with a tilt angle relative to the weld centerline of at least 45 degrees, the remainder of the examination volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction. This must be accomplished using a procedure and personnel qualified for single-side examination in accordance with Supplement 6. Subsequent examinations of this volume may be performed using examination techniques qualified for a tilt angle of at least 10 degrees.

(*3*) The examination volume not addressed by § 50.55a(b)(2)(xv)(G)(*1*) is considered fully examined if coverage is obtained in one parallel and one perpendicular direction, using a procedure and personnel qualified for single sided examination when the provisions of § 50.55a(b)(2)(xv)(G)(*2*) are met.

(*4*) Where applications are limited by design to single side access, credit may be taken for the full volume provided the examination volume is covered from a single direction perpendicular to the weld and the weld volume is examined from at least one direction parallel to the weld.

(H) When applying Supplement 5 to Appendix VIII, at least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation shall be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches.

(I) When applying Supplement 5, Paragraph (a), to Appendix VIII, the following provision must be used in calculating the number of permissible false calls:

(*1*) The number of false calls allowed must be D/10, with a maximum of 3, where D is the diameter of the nozzle.

(J) When applying the requirements of Supplement 5 to Appendix VIII, qualifications for the nozzle inside radius performed from the outside surface may be performed in accordance with Code Case N–552, "Qualification for Nozzle Inside Radius Section from the Outside Surface," provided that 10 CFR 50.55a(b)(2)(xv)(I)(*1*) is also satisfied.

(K) When performing nozzle-to-vessel weld examinations, the following provisions must be used when the requirements contained in Supplement 7 to Appendix VIII are applied for nozzle-to-vessel welds in conjunction with Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification.

(*1*) For examination of nozzle-to-vessel welds conducted from the bore, the following provisions are required to qualify the procedures, equipment, and personnel:

(*i*) For detection, a minimum of four flaws in one or more full-scale nozzle mock-ups must be added to the test set. The specimens must comply with Supplement 6, Paragraph 1.1, to Appendix VIII, except for flaw locations specified in Table VIII S6–1. Flaws may be either notches, fabrication flaws or cracks. Seventy five percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in § 50.55a(b)(2)(xv)(K)(*4*), Table VIII–S7–1—Modified, except flaws perpendicular to the weld are not required. There may be no more than two flaws from each category, and at least one subsurface flaw must be included.

(*ii*) For length sizing, a minimum of four flaws as in § 50.55a(b)(2)(xv)(K)(*1*)(*i*) must be included in the test set. The length sizing results must be added to the results of combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII. The combined results must meet the acceptance standards contained in § 50.55a(b)(2)(xv)(E)(*3*

(*iii*) For depth sizing, a minimum of four flaws as in § 50.55a(b)(2)(xv)(K)(*1*)(*i*) must be included in the test set. Their depths must be distributed over the ranges of Supplement 4, Paragraph 1.1, to Appendix VIII, for the inner 15 percent of the wall thickness and Supplement 6, Paragraph 1.1, to Appendix VIII, for the remainder of the wall thickness. The depth sizing results must be combined with the sizing results from Supplement 4 to Appendix VIII for the inner 15 percent and to Supplement 6 to Appendix VIII for the remainder of the wall thickness. The combined results must meet the depth sizing acceptance criteria contained in §§ 50.55a(b)(2)(xv)(C)(*1*), 50.55a(b)(2)(xv)(E)(*1*), and 50.55a(b)(2)(xv)(F)(*3*).

(*2*) For examination of reactor pressure vessel nozzle-to-vessel welds conducted from the inside of the vessel,

(*i*) The clad to base metal interface and the adjacent examination volume to a minimum depth of 15 percent T (measured from the clad to base metal interface) must be examined from four orthogonal directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII as modified by §§ 50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C).

(*ii*) When the examination volume defined in § 50.55a(b)(2)(xv)(K)(*2*)(*i*) cannot be effectively examined in all four directions, the examination must be

augmented by examination from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*).

(*iii*) The remainder of the examination volume not covered by § 50.55a(b)(*2*)(xv)(K)(*2*)(*ii*) or a combination of § 50.55a(b)(2)(xv)(K)(*2*)(i) and § 50.55a(b)(2)(xv)(K)(*2*)(*ii*), must be examined from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*), or from the vessel shell using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G).

(*3*) For examination of reactor pressure vessel nozzle-to-shell welds conducted from the outside of the vessel,

(*i*) The clad to base metal interface and the adjacent metal to a depth of 15 percent T, (measured from the clad to base metal interface) must be examined from one radial and two opposing circumferential directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C), for examinations performed in the radial direction, and Supplement 5 to Appendix VIII, as modified by § 50.55a(b)(2)(xv)(J), for examinations performed in the circumferential direction.

(*ii*) The examination volume not addressed by § 50.55a(b)(2)(xv)(K)(*3*)(*i*) must be examined in a minimum of one radial direction using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G).

(*4*) Table VIII-S7-1, "Flaw Locations and Orientations," Supplement 7 to Appendix VIII, is modified as follows:

TABLE VIII–S7–1—MODIFIED

Flaw Locations and Orientations

	Parallel to weld	Perpendicular to weld
Inner 15 percent	X	X
OD Surface	X	
Subsurface	X	

(L) As a modification to the requirements of Supplement 8, Subparagraph 1.1(c), to Appendix VIII, notches may be located within one diameter of each end of the bolt or stud.

(xvi) *Appendix VIII single side ferritic vessel and piping and stainless steel piping examination.*

(A) Examinations performed from one side of a ferritic vessel weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and §§ 50.55a(b)(2)(xv) (B) through (G), on specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel being examined.

(B) Examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and § 50.55a(b)(2)(xv)(A).

(xvii) *Reconciliation of Quality Requirements.* When purchasing replacement items, in addition to the reconciliation provisions of IWA-4200, 1995 Edition with the 1996 Addenda, the replacement items must be purchased, to the extent necessary, in accordance with the owner's quality assurance program description required by 10 CFR 50.34(b)(6)(ii).

(3) As used in this section, references to the OM Code refer to the ASME Code for Operation and Maintenance of Nuclear Power Plants, and include the 1995 Edition and the 1996 Addenda subject to the following limitations and modifications:

(i) *Quality Assurance.* When applying editions and addenda of the OM Code, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1979 Addenda, are acceptable as permitted by ISTA 1.4 of the OM Code, provided the licensee uses its 10 CFR part 50, Appendix B, quality assurance program in conjunction with the OM Code requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA-1 govern OM Code activities. If NQA-1 and the OM Code do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to OM Code activities.

(ii) *Motor-Operated Valve stroke-time testing.* Licensees shall comply with the provisions on stroke time testing in OM Code ISTC 4.2, 1995 Edition with the 1996 Addenda, and shall establish a program to ensure that motor-operated valves continue to be capable of performing their design basis safety functions.

(iii) *Code Case OMN-1.* As an alternative to § 50.55a(b)(3)(ii), licensees may use Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants," Revision 0, 1995 Edition with the 1996 Addenda, in conjunction with ISTC 4.3, 1995 Edition with the 1996 Addenda. Licensees choosing to apply the Code case shall apply all of its provisions.

(A) The adequacy of the diagnostic test interval for each valve must be evaluated and adjusted as necessary but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of ASME Code Case OMN-1.

(B) When extending exercise test intervals for high risk motor-operated valves beyond a quarterly frequency, licensees shall ensure that the potential increase in core damage frequency and risk associated with the extension is small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(iv) *Appendix II.* The following modifications apply when implementing Appendix II, "Check Valve Condition Monitoring Program," of the OM Code, 1995 Edition with the 1996 Addenda:

(A) Valve opening and closing functions must be demonstrated when flow testing or examination methods (nonintrusive, or disassembly and inspection) are used;

(B) The initial interval for tests and associated examinations may not exceed two fuel cycles or 3 years, whichever is longer; any extension of this interval may not exceed one fuel cycle per extension with the maximum interval not to exceed 10 years; trending and evaluation of existing data must be used to reduce or extend the time interval between tests.

(C) If the Appendix II condition monitoring program is discontinued, then the requirements of ISTC 4.5.1 through 4.5.4 must be implemented.

(v) *Subsection ISTD.* Article IWF-5000, "Inservice Inspection Requirements for Snubbers," of the ASME BPV Code, Section XI, provides inservice inspection requirements for examinations and tests of snubbers at nuclear power plants. Licensees may

use Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants," ASME OM Code, 1995 Edition up to and including the 1996 Addenda, in lieu of the requirements for snubbers in Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to their technical specifications or licensee controlled documents. Preservice and inservice examinations shall be performed using the VT-3 visual examination method described in IWA-2213.

* * * * *

(f) *Inservice testing requirements.* Requirements for inservice inspection of Class 1, Class 2, Class 3, Class MC, and Class CC components (including their supports) are located in § 50.55a(g).

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, pumps and valves must meet the test requirements of paragraphs (f)(4) and (f)(5) of this section to the extent practical. Pumps and valves which are part of the reactor coolant pressure boundary must meet the requirements applicable to components which are classified as ASME Code Class 1. Other pumps and valves that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems (in meeting the requirements of the 1986 Edition, or later, of the Boiler and Pressure Vessel or OM Code) must meet the test requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

* * * * *

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

* * * * *

(iii)(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

(iv)(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 2 and 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

* * * * *

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the inservice test requirements, except design and access provisions, set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (f)(3) of this section and that 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.

* * * * *

(g) * * *

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued before January 1, 1971, components (including supports) must meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves, and their supports must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

* * * * *

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i) Components (including supports) which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular component.

* * * * *

(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 that 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) of this section, subject to the limitation listed in paragraph (b)(2)(vi) of this section and the modifications listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section, to the extent practical within the limitation of design, geometry and materials of construction of the components.

* * * * *

(iii) Licensees may, but are not required to, perform the surface examinations of High Pressure Safety

Injection Systems specified in Table IWB-2500-1, Examination Category B-J, Item Numbers B9.20, B9.21, and B9.22.

* * * * *

(v) * * *

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

* * * * *

(6) * * *

(ii) * * *

(B) *Expedited examination of containment.*

(*1*) Licensees of all operating nuclear power plants shall implement the inservice examinations specified for the first period of the first inspection interval in Subsection IWE of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a(b)(2)(ix) by September 9, 2001. The examination performed during the first period of the first inspection interval must serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation.

(*2*) Licensees of all operating nuclear power plants shall implement the inservice examinations which correspond to the number of years of operation which are specified in Subsection IWL of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a(b)(2)(viii) by September 9, 2001. The first examination performed must serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation. The first examination of concrete must be performed prior to September 10, 2001, and the date of the examination need not comply with the requirements of IWL-2410(a) or IWL-2410(b). The date of the first examination of concrete must be used to determine the 5-year schedule for subsequent examinations subject to the provisions of IWL-2410(c).

* * * * *

(C) *Implementation of Appendix VIII to Section XI.* (*1*) The Supplements to Appendix VIII of Section XI, Division 1, 1995 Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code must be implemented in accordance with the following schedule: Supplements 1, 2, 3, and 8—May 22, 2000; Supplements 4 and 6—November 22, 2000; Supplement 11—November 22, 2001; and Supplements 5, 7, 10, 12, and 13—November 22, 2002.

* * * * *

Dated at Rockville, MD this 26th day of August, 1999.

For the Nuclear Regulatory Commission.

William D. Travers,

Executive Director for Operations.

[FR Doc. 99-24256 Filed 9-21-99; 8:45 am]

BILLING CODE 7590-01-P

10 CFR 50.55a

66FR16390

3/26/2001 Corrects
****64FR51370

Corrected Appendix VIII Sizing

212(a)(9)(B) and (c) of the Act. If the alien is not in a period of stay authorized by the Attorney General, the fact that he or she is a grandfathered alien does not prevent the alien from accruing unlawful presence under section 212(a)(9)(B) and (C) of the Act.

(n) *Evidentiary requirement to demonstrate physical presence on December 21, 2000.* (1) Unless the qualifying immigrant visa petition or application for labor certification was filed on or before January 14, 1998, a principal grandfathered alien must establish that he or she was physically present in the United States on December 21, 2000, to be eligible to apply to adjust status under section 245(i) of the Act. If no one document establishes the alien's physical presence on December 21, 2000, he or she may submit several documents establishing his or her physical presence in the United States prior to, and after December 21, 2000.

(2) To demonstrate physical presence on December 21, 2000, the alien may submit Service documentation. Examples of acceptable Service documentation include, but are not limited to:

(i) A photocopy of the Form I–94, Arrival-Departure Record, issued upon the alien's arrival in the United States;

(ii) A photocopy of the Form I–862, Notice to Appear;

(iii) A photocopy of the Form I–122, Notice to Applicant for Admission Detained for Hearing before Immigration Judge, issued by the Service on or prior to December 21, 2000, placing the applicant in exclusion proceedings under section 236 of the Act (as in effect prior to April 1, 1997);

(iv) A photocopy of the Form I–221, Order to Show Cause, issued by the Service on or prior to December 21, 2000, placing the applicant in deportation proceedings under section 242 or 242A of the Act (as in effect prior to April 1, 1997);

(v) A photocopy of any application or petition for a benefit under the Act filed by or on behalf of the applicant on or prior to December 21, 2000, which establishes his or her presence in the United States, or a fee receipt issued by the Service for such application or petition.

(3) To demonstrate physical presence on December 21, 2000, the alien may submit other government documentation. Other government documentation issued by a Federal, state, or local authority must bear the signature, seal, or other authenticating instrument of such authority (if the document normally bears such instrument), be dated at the time of issuance, and bear a date of issuance not later than December 21, 2000. For this purpose, the term Federal, state, or local authority includes any governmental, educational, or administrative function operated by Federal, state, county, or municipal officials. Examples of such other documentation include, but are not limited to:

(i) A state driver's license;

(ii) A state identification card;

(iii) A county or municipal hospital record;

(iv) A public college or public school transcript;

(v) Income tax records;

(vi) A certified copy of a Federal, state, or local governmental record which was created on or prior to December 21, 2000, shows that the applicant was present in the United States at the time, and establishes that the applicant sought on his or her own behalf, or some other party sought on the applicant's behalf, a benefit from the Federal, state, or local governmental agency keeping such record;

(vii) A certified copy of a Federal, state, or local governmental record which was created on or prior to December 21, 2000, that shows that the applicant was present in the United States at the time, and establishes that the applicant submitted an income tax return, property tax payment, or similar submission or payment to the Federal, state, or local governmental agency keeping such record;

(viii) A transcript from a private or religious school that is registered with, or approved or licensed by, appropriate State or local authorities, accredited by the State or regional accrediting body, or by the appropriate private school association, or maintains enrollment records in accordance with State or local requirements or standards.

(4) To demonstrate physical presence on December 21, 2000, the alien may submit non-government documentation. Examples of documentation establishing physical presence on December 21, 2000, may include, but are not limited to:

(i) School records;

(ii) Rental receipts;

(iii) Utility bill receipts;

(iv) Any other dated receipts;

(v) Personal checks written by the applicant bearing a bank cancellation stamp;

(vi) Employment records, including pay stubs;

(vii) Credit card statements showing the dates of purchase, payment, or other transaction;

(viii) Certified copies of records maintained by organizations chartered by the Federal or State government, such as public utilities, accredited private and religious schools, and banks;

(ix) If the applicant established that a family unit was in existence and cohabiting in the United States, documents evidencing the presence of another member of the same family unit; and

(x) For applicants who have ongoing correspondence or other interaction with the Service, a list of the types and dates of such correspondence or other contact that the applicant knows to be contained or reflected in Service records.

(5)(i) The adjudicator will evaluate all evidence on a case-by-case basis and will not accept a personal affidavit attesting to physical presence on December 21, 2000, without requiring an interview or additional evidence to validate the affidavit.

(ii) In all cases, any doubts as to the existence, authenticity, veracity, or accuracy of the documentation shall be resolved by the official government record, with records of the Service and the Executive Office for Immigration Review (EOIR) having precedence over the records of other agencies. Furthermore, determinations as to the weight to be given any particular document or item of evidence shall be solely within the discretion of the adjudicating authority (i.e., the Service or EOIR). It shall be the responsibility of the applicant to obtain and submit copies of the records of any other government agency that the applicant desires to be considered in support of his or her application.

Dated: March 20, 2001.

John Ashcroft,

Attorney General.

[FR Doc. 01–7373 Filed 3–21–01; 3:32 pm]

BILLING CODE 4410–10–P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AE26

Industry Codes and Standards; Amended Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule; correcting amendment.

SUMMARY: This document corrects a final rule appearing in the Federal Register on September 22, 1999 (64 FR 51370), and reflected in the 2000 revision of the Code of Federal

Regulations. This action corrects the final rule by specifying the use of a flaw length sizing criterion for reactor vessel qualification. This correction is necessary for clarity and consistency in the regulations.

DATES: Effective March 26, 2001.

FOR FURTHER INFORMATION CONTACT: Donald G. Naujock [telephone (301) 415-2767, e-mail DGN@nrc.gov] of the Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

SUPPLEMENTARY INFORMATION:

Background

On September 22, 1999 (64 FR 51370), a final rule " Industry Codes and Standards; Amended Requirements" was published in the Federal Register. The purpose of the rule was to permit the use of improved methods in § 50.55a for construction, inservice inspection and inservice testing of nuclear power plant components. The rule, in part, permits licensees to modify implementation of Appendix VIII to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code) provided that certain provisions specified in the regulations were followed. Paragraph (b)(2)(xv)(C) addressed the provisions regarding application of Supplement 4 to Appendix VIII. After the final rule was published, an error was discovered in paragraph (b)(2)(xv)(C)(*1*). Paragraph (b)(2)(xv)(C)(*1*) properly stipulated the use of a flaw depth sizing criterion, but failed to specify the use of an appropriate flaw length sizing criterion for reactor vessel qualification. It has always been the intent of the NRC to require the use of both depth and length criteria for flaw sizing qualification. This intent is evident in paragraph (b)(2)(xv)(F)(*2*) of § 50.55a which stipulates that length sizing qualifications must satisfy the acceptance criterion of Appendix VIII, Supplement 4.

With respect to a length sizing criterion, it was the intent of the NRC to specify in the final rule, the use of 0.75 inch root mean square (RMS) length sizing criterion in lieu of Appendix VIII, Supplement 4, Subparagraph 3.2(b). Since 1995, the NRC has supported the 0.75 inch RMS numeric value as an appropriate length sizing criterion for reactor vessels. This numeric value is the same as the length sizing criterion referenced in (b)(2)(xv)(E)(*3*).

Need for Correction

As published, the Federal Register and the Code of Federal Regulations contain an error which is misleading and needs to be corrected.

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. 552 and 553, the NRC is adopting the following amendment 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: Sections 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. 444, 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. 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. In § 50.55a, paragraph (b)(2)(xv)(C)(*1*) is revised to read as follows:

§ 50.55a Codes and standards.

* * * * *

(b) * * *

(2) * * *

(xv) * * *

(C) * * *

(*1*) A depth sizing requirement of 0.15 inch RMS shall be used in lieu of the requirement in Subparagraph 3.2(a), and a length sizing requirement of 0.75 inch RMS shall be used in lieu of the requirement in Subparagraph 3.2(b).

* * * * *

Dated at Rockville, Maryland, this 20th day of March, 2001.

For the Nuclear Regulatory Commission.

Michael T. Lesar,

Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration.

[FR Doc. 01-7352 Filed 3-23-01; 8:45 am]

BILLING CODE 7590-01-P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 73

[Docket No. FAA-2001-9059; Airspace Docket No. 01-AWA-1]

RIN 2120-AA66

Establishment of Prohibited Area P-49 Crawford; TX

AGENCY: Federal Aviation Administration (FAA), DOT.

ACTION: Final rule.

SUMMARY: This action establishes Prohibited Area 49 (P-49) over the Crawford, TX, residence of the President of the United States. The FAA is taking the action to enhance security in the immediate vicinity of the presidential residence and assist the United States Secret Service in accomplishing its mission of providing security for the President of the United States.

EFFECTIVE DATES: 0901 UTC, May 17, 2001.

FOR FURTHER INFORMATION CONTACT: Steve Rohring, Airspace and Rules Division, ATA-400, Office of Air Traffic Airspace Management, Federal Aviation Administration, 800 Independence Avenue, SW., Washington, DC, 20591; telephone: (202) 267-8783.

SUPPLEMENTARY INFORMATION:

Background

On March 7, the Department of the Treasury, United States Secret Service requested that the FAA establish a prohibited area at Crawford, TX, to enhance the level of security provided the President. In order to provide adequate safeguards for the protection of the President, it is necessary to designate certain airspace above the presidential residence at Crawford, TX, as a prohibited area. Under the provision of Section 73.83, no person may operate an aircraft within that area

10 CFR 50.55a

*****67FR60520

10/28/2002

Added 1997 Addenda to 2000 Addenda
III, XI, OM

PART 121—POSSESSION OF BIOLOGICAL AGENTS AND TOXINS

3. The authority citation for part 121 continues to read as follows:

Authority: Secs. 211–213, Title II, Pub. L. 107–188, 116 Stat. 647 (7 U.S.C. 8401).

4. In § 121.1, the definitions for *biological agent* and *toxin* are revised to read as follows:

§ 121.1 Definitions.

Biological agent. Any microorganism (including, but not limited to, bacteria, viruses, fungi, rickettsiae, or protozoa), or infectious substance, or any naturally occurring, bioengineered, or synthesized component of any such microorganism or infectious substance, capable of causing:

(1) Death, disease, or other biological malfunction in a human, an animal, a plant, or another living organism;

(2) Deterioration of food, water, equipment, supplies, or material of any kind; or

(3) Deleterious alteration of the environment.

* * * * *

Toxin. The toxic material or product of plants, animals, microorganisms (including, but not limited to, bacteria, viruses, fungi, rickettsiae, or protozoa), or infectious substances, or a recombinant or synthesized molecule, whatever their origin and method of production, and includes:

(1) Any poisonous substance or biological product that may be engineered as a result of biotechnology produced by a living organism; or

(2) Any poisonous isomer or biological product, homolog, or derivative of such a substance.

Done in Washington, DC, this 20th day of September, 2002.

Peter Fernandez,

Acting Administrator, Animal and Plant Health Inspection Service.

[FR Doc. 02–24423 Filed 9–25–02; 8:45 am]

BILLING CODE 3410–34–P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AG61

Industry Codes and Standards; Amended Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to incorporate by reference a later edition and addenda of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code) and the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) to provide updated rules for construction, inservice inspection (ISI), and inservice testing (IST) of components in light-water cooled nuclear power plants. This final rule incorporates by reference the latest edition and addenda of the ASME BPV and OM Codes that have been approved for use by the NRC subject to certain limitations and modifications.

EFFECTIVE DATE: October 28, 2002. The incorporation by reference of certain publications in this rule is approved by the Director of the Office of the Federal Register as of October 28, 2002

ADDRESSES: The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC at 1–800–397–4209, (301) 415–4737, or by email to pdr@nrc.gov. The availability of the Regulatory Analysis, Environmental Assessment, and Resolution of Public Comments associated with this rulemaking is further discussed in Section 5 below, under **SUPPLEMENTARY INFORMATION.**

FOR FURTHER INFORMATION CONTACT: Stephen Tingen, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Alternatively, you may contact Mr. Tingen at (301) 415–1280, or via e-mail at: *sgt@nrc.gov.*

SUPPLEMENTARY INFORMATION:

1. Background

On August 3, 2001 (66 FR 40626), the NRC published a **Federal Register** notice that presented a proposed rule to amend 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed rule would revise the requirements for construction, ISI, and IST of nuclear power plant components. For construction, the proposed rule would permit the use of Section III, Division 1, of the ASME BPV Code, 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda for Class 1, Class 2, and Class 3 components with no new modifications or limitations.

For ISI, the proposed rule would permit the use of Section XI, Division 1, of the ASME BPV Code, 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda for Class 1, Class 2, Class 3, Class MC, and Class CC components with new modifications and limitations.

For IST, the proposed rule would permit the use of the ASME OM Code, 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda for Class 1, Class 2, and Class 3 pumps and valves with one new modification.

In the same **Federal Register** notice, the Commission withdrew a proposed rule (64 FR 22580; April 27, 1999) that would have eliminated the requirement for licensees to update their ISI and IST programs every 120 months beyond a

baseline edition and addenda of the ASME Code. That withdrawal was a final action—not part of the proposed rule.

2.0 Public Comments on Proposed Rule; and Final Rule

Interested parties submitted written comments on the proposed rule published on August 3, 2001 (66 FR 40626). Comments were received from 17 separate sources. These sources consisted of 10 utilities, 4 service organizations, and 3 individuals. In response to the public comments, the NRC has either removed or revised some modifications and limitations that were proposed. A summary of the public comments applicable to the proposed rule and their resolution are provided in the following sections. Public comments on the proposed rule that are not addressed in the final rule are addressed in the Resolution of Public Comments. The availability of the Resolution of Public Comments is further discussed in Section 5 below.

The NRC has considered and resolved the public comments and has revised the final rule accordingly. The NRC is publishing these final regulations in § 50.55a to incorporate by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Division 1 rules of Section III of the ASME BPV Code; the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Division 1 rules of Section XI of the ASME BPV Code; and the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code for construction, ISI, and IST of components in nuclear power plants. Section III of the ASME BPV Code is acceptable for use with no new limitations or modifications. Section XI of the ASME BPV Code is acceptable for use subject to limitations and modifications. The ASME OM Code is acceptable for use subject to one modification.

This final rule also revises the regulations in § 50.55a that licensees use to modify the implementation of Appendix VIII, "Performance Demonstration for Ultrasonic Examinations Systems," to Section XI of the ASME BPV Code. The amendment clarifies existing ultrasonic (UT) examination qualification requirements in § 50.55a. The amendment also adds new requirements to clarify the coordination of Appendix VIII with other parts of Section XI.

2.1 Section III

There were no public comments on the proposed rule concerning Section III. This final rule revises § 50.55a(b)(1) to incorporate by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III of the ASME BPV Code; clarify that the 1963 Edition was the initial edition of Section III incorporated by reference in the regulations; and extend the applicability of the existing regulations in §§ 50.55a(b)(1)(ii), 50.55a(b)(1)(iii), and 50.55a(b)(1)(v) to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III of the ASME BPV Code.

2.2 Section XI

Public comments on the proposed rule concerning Section XI are addressed in the following sections. This final rule revises § 50.55a(b)(2) to incorporate by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV Code; clarify that the 1970 Edition was the initial edition of Section XI incorporated by reference in the regulations; and extend the applicability of the existing regulations in §§ 50.55a(b)(2)(viii), 50.55a(b)(2)(ix), 50.55a(b)(2)(xi), 50.55a(b)(2)(xv), and 50.55a(b)(2)(xvii) to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV Code. This final rule also deletes the existing regulations in § 50.55a(g)(6)(ii)(B)(*1*) through (*4*) because the implementation dates have expired and all licensees have completed the requirements or have been approved by an exemption for a delay. The existing requirement that was formerly § 50.55a(g)(6)(ii)(B)(*5*) is redesignated as § 50.55a(g)(6)(ii)(B).

Although § 50.55a(b)(2)(vi) is not addressed in the proposed rule, one commenter stated that § 50.55a(b)(2)(vi) should be revised to include references to the 1998 Edition through the 2000 Addenda of the ASME Code for the ISI of Class MC and Class CC components. The commenter noted that §§ 50.55a(b)(2)(viii) and (ix) in the proposed rule reference the 1998 Edition through the 2000 Addenda, therefore, § 50.55a(b)(2)(vi) should also reference the 1998 Edition through the 2000 Addenda.

The NRC agrees that § 50.55a(b)(2)(vi) should be revised to clarify the applicability of the 1998 Edition through the 2000 Addenda to containment ISI programs but does not agree with the revision suggested by the commenter. The statement of considerations for the final rule published on September 22, 1999 (64 FR 51370), states that either the 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda of IWE and IWL must be used to develop and implement a containment ISI program within 5 years. The NRC finds that the existing requirements in § 50.55a(b)(2)(vi) only address the applicable edition and addenda of IWE and IWL to be used during initial 120-month interval for the ISI of Class CC and Class MC components. Therefore, § 50.55a(b)(2)(vi) is revised to clarify that the 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda of IWE and IWL must be used when implementing the initial 120-month interval for the ISI of Class MC and Class CC components, and that successive 120-month interval updates must be implemented in accordance with § 50.55a(g)(4)(ii).

The proposed rule would add a new § 50.55a(g)(6)(ii)(B)(*1*) to clarify the start date of the first 120-month interval for the ISI of Class MC and Class CC components. Some commenters indicated that § 50.55a(g)(6)(ii)(B)(*1*) in the proposed rule did not clarify the start date of the first 120-month interval for the ISI of Class MC and Class CC components. Other commenters suggested a revised regulation that they thought would be more appropriate.

The NRC finds that the proposed regulation regarding the start date of the first 120-month interval for the ISI of Class MC and Class CC components has created confusion rather than clarifying existing requirements as intended. The clarification in the proposed rule would also create a hardship for many licensees in reestablishing the start date of their first 120-month containment ISI interval. It was not the intent of the NRC to alter the 10-year examination interval in IWE or the 5-year examination interval in IWL already established by licensees. Licensees are permitted to schedule examinations of Class MC and Class CC components in accordance with the requirements in IWE and IWL. Therefore, the clarification of the first 120-month interval start date in § 50.55a(g)(6)(ii)(B)(*1*) in the proposed rule is not adopted.

In responding to this clarification, several commenters indicated that the 10-year IWE and 5-year IWL examination intervals must coincide with the 120-month interval update in § 50.55a(b)(4)(ii). The NRC does not agree that the 10-year IWE and 5-year IWL examination intervals must coincide with the 120-month interval update in § 50.55a(b)(4)(ii). The 10-year IWE and 5-year IWL examination intervals are independent of the 120-month interval update in § 50.55a(g)(4)(ii). Section 50.55a(g)(4)(ii) does not prohibit licensees from updating to a later edition and addenda of the ASME Code midway through a

10-year IWE or 5-year IWL examination interval.

In responding to this clarification, several commenters implied that the final rule dated August 8, 1996 (61 FR 41303), requiring licensees to develop and implement a containment ISI program for Class MC components in accordance with IWE of Section XI, authorized the extension of the first period inspection from 40 months to 60 months in duration. The NRC does not agree. The schedule in the containment final rule did not extend the duration of the 40-month inspection period required by IWE. This issue was addressed in the response to Question 13 in a letter to the Nuclear Energy Institute from NRC dated May 30, 1997.

In responding to this clarification, several commenters indicated that the final rules dated August 8, 1996 (61 FR 41303), and September 22, 1999 (64 FR 51370), create a hardship when implementing 120-month interval updates required by § 50.55a(g)(4)(ii). The NRC agrees with this comment. The final rule dated August 8, 1996, required licensees to implement an ISI program for Class MC and Class CC components using the 1992 Edition with the 1992 Addenda of IWE and IWL. The final rule dated September 22, 1999, required licensees to implement Appendix VIII UT qualification requirements using the 1995 Edition with the 1996 Addenda of Section XI. Consequently, the schedule for 120-month interval updates for the ISI of Class MC and Class CC components, Appendix VIII UT qualification requirements, and the ISI of Class 1, 2, and 3 components might not coincide. This creates a hardship for licensees because ISI programs are required to maintain up to 3 separate editions and addenda of Section XI—one edition and addenda applicable to the ISI of Class MC and Class CC components, another edition and addenda applicable to the ISI of Class 1, 2, and 3 components, and a third edition and addenda applicable to Appendix VIII UT qualification requirements. Therefore, licensees may wish to synchronize 120-month interval updates such that the same edition and addenda of Section XI apply to the ISI of Class MC and Class CC components, Appendix VIII UT qualification requirements, and the ISI of Class 1, 2, and 3 components. Licensees wishing to synchronize their 120-month intervals may submit a request in accordance with § 50.55a(a)(3) to obtain authorization to extend or reduce 120-month intervals.

2.2.1 Owner-Defined Requirements for Class CC and Class MC Components

The proposed rule addresses NRC concerns with "owner-defined" requirements in IWE and IWL. Revisions to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE and IWL permit each licensee to define requirements that were previously defined in the ASME Code.

A number of commenters indicated that "owner-defined" requirements are acceptable because the regulations in Appendix B of 10 CFR 50, "Quality Assurance Criteria For Nuclear Power Plants and Fuel Reprocessing Plants," and the Responsible Engineer/Individual oversight provisions (as delineated in IWE and IWL) ensure that requirements defined by the owner are properly implemented.

The NRC does not agree that the quality assurance requirements in Appendix B of 10 CFR 50 and the oversight duties of the Responsible Engineer/Individual alone are adequate to ensure that owner-defined requirements are properly implemented. The final rule published on August 8, 1996 (61 FR 41303), required licensees to develop and implement a containment ISI program for Class MC and Class CC components in accordance with IWE and IWL. The final containment rule stated that the rule was needed because none of the existing requirements provide specific guidance on how to inspect containment surfaces. This lack of guidance resulted in a large variation with regard to the performance and the effectiveness of licensee containment inspection programs. Based on the results of inspections and audits, as well as plant operational experiences, it was clear to the NRC that without specific guidance, several licensee containment inspection programs were unable to detect degradation that could ultimately result in a compromise to the containment pressure-retaining capability. Some containment structures had been found to have undergone a significant level of degradation that was not detected by existing programs. Given the number and the extent of the occurrences, and the variability among plants with regard to the performance and the effectiveness of containment inspections, the NRC believed that the prudent course of action was to impose the more specific ISI inspection requirements in the 1992 Edition with the 1992 Addenda of IWE and IWL. The containment final rule imposed requirements that define comprehensive and technically sound methods that ensure uniform containment inspection results among all licensees.

The NRC believes that it is inappropriate to approve Code provisions that do not contain specific containment inspection guidance when prior experience demonstrates that specific containment inspection guidance is necessary. The quality assurance provisions in Appendix B of 10 CFR 50 and the oversight duties of the Responsible Engineer/Individual do not ensure uniform containment inspection results among all licensees. Furthermore, the quality assurance provisions in Appendix B of 10 CFR 50 did not prevent the previous problems associated with a lack of guidance. Reliance on Appendix B of 10 CFR 50 resulted in a large variation in the performance and effectiveness of licensees' containment inspection programs that contributed to the NRC issuing the containment final rule.

2.2.1.1 Visual Examination Qualification Requirements (Class CC Components)

Section 50.55a(b)(2)(viii)(F) in the proposed rule would require that personnel who conduct visual examinations of containment surfaces be qualified in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2300 in place of the "owner-defined" qualification provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWL–2310(d). Prior to the 1998 Edition, the NRC-approved provisions in IWA–2300 were used to define the qualification requirements for personnel who conduct visual examinations of containment surfaces. The qualification requirements were revised in IWL–2310(d), 1997 Addenda, to allow the owner to define the qualification requirements for personnel who perform visual examinations of concrete and tendon anchorage hardware, wires, or strands. However, the new Code provision does not provide any criteria that the licensee must use when developing qualification requirements. Therefore, the NRC proposed that licensees continue to use the provisions in IWA–2300 to qualify personnel who perform visual inspections of containment concrete surfaces and tendon anchorage hardware, wires, or strands.

Several commenters recommended that the NRC specify the use of a more generic standard for qualification of containment examiners such as ANSI N45.2.6, "Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants," to define personnel qualification provisions in

place of the requirements in IWA–2300. One commenter stated that licensees typically commit to meet the requirements of Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (Revision 1, September 1980)," or another NRC-approved standard that endorses ANSI N45.2.6. Another commenter noted that use of the qualification standards of IWA–2300, as proposed by the NRC, is not appropriate because they were designed for examinations associated with piping systems and their supports and not containment examinations.

The NRC disagrees with the comments because the use of "owner-defined" qualification requirements or a generic quality assurance standard to qualify containment examiners does not provide adequate guidance to ensure that examiners are qualified to inspect containment surfaces. The NRC prefers instead that the ASME Code identify the specific elements deemed necessary to ensure containment inspection qualification programs are adequate, or describe specific criteria that licensees must use to qualify personnel performing containment examinations. Although the existing qualification provisions in IWA–2300 were not developed specifically for qualifying examiners of concrete containment surfaces, they provide the most practical criteria that are presently available for qualification of personnel that conduct visual examinations of containment surfaces. The NRC notes that many of the changes in the later editions and addenda of IWE and IWL are more suited to containment examinations than earlier editions and addenda. The NRC withdrew Regulatory Guide 1.58 on July 31, 1991 (56 FR 36175). Therefore, the NRC no longer endorses the use of ANSI 45.2.6 for the ISI of containment surfaces in operating nuclear power plants. Section 50.55a(b)(2)(viii)(F) in the proposed rule is adopted without change.

2.2.1.2 Visual Examination Qualification Requirements (Class MC and Liners of Class CC)

Section 50.55a(b)(2)(ix)(F) of the proposed rule would require that personnel who conduct visual examinations of containment surfaces be qualified in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2300 of in place of the "owner-defined" qualification provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda IWE–2330(a). Prior to the 1998 Edition, the NRC approved provisions in IWA–2300 were used to define the qualification requirements for personnel who conduct visual examinations of containment surfaces.

There was one public comment on § 50.55a(b)(2)(ix)(F), which is discussed in the following Section 11, Backfit Analysis. In consideration of the public comment, the qualification requirements for personnel that conduct visual inspections of containment surfaces have been revised to require that VT–1 and VT–3 examinations must be conducted in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2200. Personnel conducting examinations in accordance with the VT–1 or VT–3 examination method shall be qualified in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2300.

2.2.1.3 General and Detailed Visual Examinations

Section 50.55a(b)(2)(ix)(G) in the proposed rule would require that the general and detailed visual examinations required by the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–2310(b) and IWE–2310(c) meet the VT–1 and VT–3 examination method provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2210 in place of the "owner-defined" general and detailed visual examination provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–2310(a), and allow licensees to continue to extend Table IWA–2210–1 maximum direct examination distance and decrease Table IWA–2210–1 minimum illumination requirements as currently stated in § 50.55(b)(2)(ix)(B).

The distance and illumination requirements in § 50.55a(b)(2)(ix)(G) in the proposed rule have been removed because these requirements are addressed in the existing § 50.55a(b)(2)(ix)(B). There was one public comment on § 50.55a(b)(2)(ix)(G), which is discussed in the following Section 11, Backfit Analysis. In consideration of the public comment, § 50.55a(b)(2)(ix)(G) is revised to require that the VT–1 and VT–3 examination methods in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2200 be used to conduct specific visual examinations in Table IWE–2500–1 in place of the "owner-defined" visual examination methods in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–2310(b) and IWE–2310(c). The VT–3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE–2500–1, and the VT–1 examination method must be used to conduct the examination in Item E4.11 of Table IWE–2500–1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval.

2.2.2 Examination of Containment Bolted Connections

Section 50.55a(b)(2)(ix)(H) of the proposed rule would require that the acceptance standard in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWC–3513 be used to evaluate flaws in pressure-retaining bolting greater than or equal to 51 millimeters [2.0 inches] in diameter which are identified during the examination of containment surfaces. The acceptance standard would be used in place of the "owner-defined" acceptance standard in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–3510.1.

Several commenters stated that § 50.55a(b)(2)(ix)(H) of the proposed rule is unnecessary because there are no substantial differences between the revised standard for bolting materials in the 1998 Edition and the standard for bolting materials in editions and addenda earlier than the 1998 Edition. The NRC disagrees. The bolting standard for bolting materials in the editions and addenda of IWE–3515.1 earlier than the 1998 Edition was significantly revised in the 1998 Edition. Prior to the 1998 Edition, IWE–3515.1 stated that bolting material must be examined in accordance with the material specification for defects which may cause the bolted connection to violate either the containment leak-tight or structural integrity. IWE–3515.1 was revised and renumbered as IWE–3510.3 in the 1998 Edition to require that the owner define the standard for examining bolting materials. Since containment bolting is not unique from other bolting applications in Section XI, the NRC finds that the examination of containment bolting should be consistent with other Section XI bolting examination requirements.

A number of commenters stated that IWC–3513 is not the appropriate standard to use to evaluate flaws in pressure-retaining bolting. One commenter recommended that IWB–3517.1 be used in place of IWC–3513. The NRC agrees and finds that the visual examination criteria for bolting in IWE–3517.1 is an acceptable standard because it enures that the integrity of reused bolting is maintained. Section 50.55a(b)(2)(ix)(H) is revised to require that bolting material be examined in accordance with the material specification or the 1997 Addenda, 1998

Edition, 1999 Addenda, and 2000 Addenda of IWB–3517.1.

Section 50.55a(b)(2)(ix)(I) in the proposed rule would require licensees to supplement the containment bolted connection examination requirements in Items E1.10 and E1.11 of the 1998 Edition, 1999 Addenda, and 2000 Addenda of Table IWE–2500–1 with additional requirements for examining inaccessible areas of containment bolting.

One commenter stated that since the ASME Code requires that accessible areas of containment bolted connections be more frequently examined in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE than in the earlier editions and addenda, bolting examination requirements have been enhanced. The NRC disagrees. Although the revised provisions increase the frequency of accessible examinations of containment bolting, the revised provisions reduce the frequency of examinations of inaccessible areas of containment bolting. The 1992 Edition with the 1992 Addenda and the 1995 Edition with the 1996 Addenda of IWE provide acceptable provisions for conducting examinations of the accessible and inaccessible areas of containment bolted connections. Item No. E8.10 of Table IWE–2500–1 requires that a visual examination of the individual parts of the bolted connection using the VT–1 visual examination method be performed whenever a connection is disassembled during a scheduled ISI inspection. Item E8.20 of Table IWE–2500–1 requires that a bolt torque or tension test be performed on bolted connections that have not been disassembled during the inspection interval. A bolt torque or tension test provides an indication of the integrity of the inaccessible areas of a bolted connection. The requirements in Items E8.10 and E8.20 requiring that containment bolting either be disassembled and examined (VT–1), or torque tested every interval were deleted in the 1998 Edition of IWE.

Several commenters suggest that § 50.55a(b)(ix)(I) of the proposed rule be revised to allow the option of conducting visual examinations of the inaccessible areas of containment bolted connections during maintenance that requires a bolted connection be disassembled or during visual examinations that are conducted during scheduled ISI inspections. In consideration of the public comments, the modification that was formerly § 50.55a(b)(ix)(I) in the proposed rule is revised in the final rule to allow licensees the option of performing visual examinations of inaccessible areas of containment bolted connections during maintenance evolutions or scheduled inspections. Any bolted connections that are disassembled during the scheduled performance of Item E1.11 examinations must be examined using the VT–3 examination method. Flaws or degradation identified during the performance of this VT–3 examination must be examined in accordance with the VT–1 examination method. The criteria in the material specification, or the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWB–3517.1 must be used to evaluate bolting flaws or degradation. As an alternative to performing the VT–3 examination during the scheduled performance of Item E1.11, VT–3 examination of bolting may be conducted whenever containment bolting in Item E1.11 is disassembled for any reason. Sections 50.55a(b)(ix)(I) and 50.55a(b)(ix)(H) in the proposed rule have been combined as § 50.55a(b)(ix)(H) in this final rule.

2.2.3 Acceptance Standard for Surfaces Requiring Augmented Ultrasonic Examinations

Section 50.55a(b)(2)(ix)(J) in the proposed rule would require that the ultrasonic (UT) examination acceptance standard specified in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–3511.3 for Class MC pressure-retaining components also apply to metallic liners of Class CC pressure-retaining components. A UT acceptance standard is needed for metallic liners of Class CC pressure-retaining components to evaluate conditions that are identified during an examination that may be unacceptable. Therefore, the NRC proposed to continue to use the UT acceptance standard in IWE–3511.3 for metallic liners of Class CC pressure-retaining components.

Several commenters stated that § 50.55a(b)(2)(ix)(J) of the proposed rule is not needed because the provisions in IWE–3122.3 provide an appropriate standard for evaluating degradation and aging of metallic liners of Class CC pressure-retaining components. The NRC disagrees. Item E4.12 of the 1998 Edition, 1999 Addenda, and 2000 Addenda of Table IWE–2500–1, states that IWE–3511 is the acceptance standard for UT examinations. IWE–3122.3 is not referenced in Table IWE–2500–1 as an acceptance standard. The acceptance standard in IWE–3511 addresses Class MC pressure-retaining components and does not address metallic liners of Class CC pressure-retaining components. Prior to the 1995 Addenda to Section XI, the standard in IWE–3511 addressed Class MC pressure-retaining components and metallic liners of Class CC pressure-retaining components. IWE–3511 was revised in the 1995 Addenda to address only Class MC pressure-retaining components. The NRC believes that the acceptance standard in the 1995 Addenda, 1996 Addenda, 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–3511 is incomplete because it does not address metallic liners of Class CC pressure-retaining components.

Several commenters stated that § 50.55a(b)(2)(ix)(J) of the proposed rule is inappropriate because a concrete metallic liner can be allowed to significantly degrade and still accomplish its design function. Therefore, imposing an acceptance limit of 10 percent of the nominal wall thickness is extremely conservative and unwarranted.

The NRC disagrees and believes that the UT acceptance limit of 10 percent of the nominal wall thickness is warranted. Concrete containments are constructed with metallic liners as the final leak-tight barrier against radioactive releases to the atmosphere. By the virtue of being anchored to the concrete, the liner carries stresses and strains imparted by the concrete in addition to the loads of the liner itself. General or pitting corrosion occurring in a large area of the liner creates discontinuities in the liner behavior under accident pressure and earthquake loads which would result in a high stress concentration area in the liner. The model tests on concrete containments (*e.g.*, NUREG/CR–5431, "Round-Robin Analysis of the Behavior of a 1:6 Scale Reinforced Concrete Containment Model Pressurized to Failure: Posttest Evaluation") have shown that once a liner tear occurs due to high stress concentration, the containment losses its ability to retain radioactive releases. Thus, the liner integrity must be monitored and maintained during the operating life of the containment. The modification in the proposed rule is identical to what was approved for use by the ASME Code in the 1995 Edition and earlier editions and addenda of the ASME Code. Section 50.55a(b)(2)(ix)(J) of the proposed rule is presented here in the final rule as § 50.55a(b)(2)(ix)(I). Section § 50.55a(b)(2)(ix)(I) is otherwise adopted without change.

2.2.4 Containment Penetration Piping

Section 50.55a(b)(2)(xii)(A) in the proposed rule would prohibit welds in high-energy fluid system piping that are located inside a containment penetration assembly or encapsulated by a guard pipe from being exempted from the examination provisions of

Subsection IWC as permitted by the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWC–1223. The revised Code provisions appeared to be inconsistent with NRC's guidelines on "break exclusion zone" design and examination criteria for containment penetration piping. Specifically, Branch Technical Position EMEB 3–1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," an attachment to NRC Standard Review Plan (SRP) Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with Postulated Rupture of Piping" (NUREG–0800), allows that breaks and cracks in high-energy fluid piping in containment penetration areas need not be postulated provided that certain criteria are met. These criteria include a commitment that where guard pipes are used, the enclosed portion of fluid system piping should be seamless construction and without circumferential welds unless specific access provisions are made to permit inservice volumetric examination of the longitudinal and circumferential welds; and a 100 percent volumetric inservice examination of all pipe welds is conducted during each inspection interval as defined in IWA–2400 of Section XI of the ASME BPV Code. Licensees may have made commitments to follow the provisions in SRP 3.6.2 as a part of their licensing design basis.

The commenters stated that § 50.55a(b)(2)(xii)(A) of the proposed rule is unnecessary because the regulatory requirements associated with high energy line breaks are independent from the scope of Section XI. Commenters also noted that it is inappropriate for the NRC to impose limitations to maintain commitments used to license plants.

The NRC agrees that the regulatory guidelines associated with high energy line breaks are separate from the regulatory requirements associated with the ISI of nuclear power plant components. The intent of § 50.55a(b)(2)(xii)(A) in the proposed rule was to ensure that licensee commitments regarding high energy line breaks in Branch Technical Positions under SRP 3.6.2 would not be eliminated from a misapplication of the exemption allowed in IWC–1223. The NRC concludes that it is the responsibility of each licensee to ensure that changes to later editions and addenda of the ASME Code are not misapplied to licensing design bases commitments, and that it is inappropriate for the NRC to impose modifications or limitations in § 50.55a to ensure that commitments, not directly related to Section XI requirements but part of the licensing design basis, are maintained. Therefore, § 50.55a(b)(2)(xii)(A) in the proposed rule is not adopted.

Section 50.55a(b)(2)(xii)(B) in the proposed rule would require that piping that penetrates the containment that is connected to a system not in the scope of Section XI (*i.e.*, not safety-related) be pressure tested in accordance with the 1996 Addenda and earlier editions and addenda of IWA–5110(c).

A number of commenters stated that § 50.55a(b)(2)(xii)(B) is unnecessary because the Type C local leak rate test (LLRT) in Appendix J of 10 CFR 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," provides an acceptable method for ensuring the leak-tight integrity of the containment penetration piping, and that the test requirements in the editions and addenda of IWA–5110(c) earlier than the 1997 Addenda are redundant. The commenters stated that test equipment used for LLRT is capable of detecting extremely small leakage, and that the regulations in Appendix J of 10 CFR 50 contain acceptance criteria for leakage identified during testing. Commenters also noted that Appendix J does not differentiate between measured leakage emanating out of the piping and out of the containment isolation valves. However, the commenters noted that this determination is unnecessary because the Appendix J maximum allowable leakage limit accounts for all leakage regardless of where it emanates.

The NRC agrees that Appendix J provides an acceptable method for testing the leak-tightness of the containment penetration piping. Appendix J of 10 CFR 50 requires that piping between the containment isolation valves be pressurized with air during seat leak testing of the containment isolation valves. Any leakage emanating from the piping and containment isolation vales is measured and evaluated in accordance the criteria in Appendix J. The NRC finds that the Appendix J Type C LLRT provides an acceptable basis for ensuring the containment penetration piping integrity when the only safety function of the containment penetration piping is to provide containment integrity. Therefore, § 50.55a(b)(2)(xii)(B) in the proposed rule is not adopted.

2.2.5 Certification of Nondestructive Examination (NDE) Personnel

Section 50.55a(b)(2)(xviii)(A) in the proposed rule would require that all Level I and Level II NDE personnel be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA–2314, and the 1999 Addenda and 2000 Addenda of IWA–2314(a) and IWA–2314(b). Prior to 1997, Level I and II NDE personnel were recertified on a 3-year interval.

A number of commenters objected to § 50.55a(b)(2)(xviii)(A) in the proposed rule. The commenters explained that the 1996 Addenda and earlier editions and addenda of IWA–2314 require that Level I and Level II personnel be recertified by qualification examination every 3 years, and that Level III personnel be recertified by qualification examination every 5 years. The commenters stated that the 5-year recertification interval should also be acceptable for Level I and Level II personnel because the 5-year recertification interval for Level III personnel has been approved by the NRC since 1989. The commenters also disagreed with the NRC position that available data do not support recertification examinations at a frequency of every 5 years. On the contrary, the commenters stated that since the recertification interval was increased from 3 to 5 years in 1989 for Level III personnel, there is no data to support that a decrease in proficiency of Level III personnel has occurred. The commenters claimed that the improved annual practice requirements for UT examiners ensure the proficiency of UT examiners is maintained throughout the 5-year period. One commenter stated that Section XI is one of the few standards that require recertification by examination every 3 years, and that other countries recertify personnel every 5 to 10 years.

The NRC did not approve the extension of the recertification frequency from 3 years to 5 years in the proposed rule because the proficiency of examination personnel decreases over time, and available data do not support recertification examinations at a frequency of every 5 years. Although one commenter (a licensee) stated that it has a 100 percent recertification pass rate, the public comments did not provide or reference any data that NRC could review that supports extending the recertification frequency of Level 1 and Level 2 NDE personnel from 3 years to 5 years. Therefore, the NRC is not approving the extension of the recertification interval for Level I and Level II NDE personnel from 3 to 5 years at this time. Section 50.55a(b)(2)(xviii)(A) in the proposed rule is adopted without change.

Section 50.55a(b)(2)(xviii)(B) in the proposed rule would supplement the alternative qualification provisions for

VT–2 visual examination personnel in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2316 with the requirements that VT–2 examination personnel pass an initial test and then be retested every 3 years.

Commenters indicated that the intent of IWA–2316 is to only qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with IWA–5211(a) and (b), and objected to § 50.55a(b)(2)(xviii)(B) in the proposed rule on the basis that experienced plant personnel such as system engineers, licensed and non-licensed operators, and maintenance staff perform the VT–2 examinations. The commenters argue that the basic knowledge level of these types of personnel is adequate to inspect plant systems during leakage tests. The commenters also note that the NRC has granted relief allowing licensees to implement the VT–2 visual examination qualification conditions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2316 without requiring initial tests and periodic retests. The commenters also noted that the existing NRC-approved requirements in the 1995 Edition with the 1996 Addenda of IWA–2300 require that personnel who conduct NDE be qualified in accordance with CP–189. The commenters stated that VT–2 qualification requirements are not in the scope of CP–189 nor are they addressed in CP–189 because there are no unique technical requirements associated with performing VT–2 examinations. VT–2 examinations are conducted to detect evidence of leakage from pressure-retaining components during system pressure tests. The use of special equipment, examination techniques, and evaluation of test results associated with other NDE methods such as volumetric and surface examinations are not applicable to VT–2 examinations. VT–2 examinations do not include the evaluation of the material conditions of components, such as degraded conditions like loose bolting or corrosion. The commenters also stated that the proposed § 50.55a(b)(2)(xviii)(B) is unnecessary because plant administrative procedures require that personnel involved in testing be briefed prior to the test, and special requirements for conducting the VT–2 examinations are covered during the pretest brief.

The NRC agrees that there are no special or unique technical requirements associated with performing VT–2 examinations that require personnel to observe for leakage of liquids or condensation during system leakage or hydrostatic testing. However, VT–2 visual examiners also conduct other evolutions that are more complex than observing for leakage during a leakage or hydrostatic tests. Visual examiners that are VT–2 qualified also perform bubble, halogen diode leak, and mass spectrometer testing requiring the use of special equipment and examination techniques. The NRC believes that VT–2 qualification requirements in IWA–2316 should be limited to personnel that only observe for leakage of liquids or condensation during system leakage or hydrostatic testing. Therefore, § 50.55a(b)(2)(xviii)(B) is revised to clarify that IWA–2316 may only be used to qualify personnel that observe for leakage during the performance of system leakage and that hydrostatic tests are to be conducted in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–5211(a) and (b).

Section 50.55a(b)(2)(xviii)(C) in the proposed rule would supplement the alternative qualification provisions for VT–3 visual examination personnel in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2317 with the requirements that VT–3 examination personnel pass an initial test and then retested every 3 years.

Several commenters objected to § 50.55a(b)(2)(xviii)(C) in the proposed rule because experienced personnel are familiar with the performance of VT–3 examinations, and the VT–3 examination is a straightforward technique. The NRC does not agree because the material condition of many different types of components are required to be evaluated during the performance of VT–3 examinations, and there are different technical acceptance criteria specified for the many different components. For example, the acceptance criteria for examining bolting is different from the acceptance criteria for examining containment metal surfaces. Furthermore, there are critical technical requirements associated with the minimum illumination, distance, and character height that must be adhered to when performing VT–3 examinations. There are a number of options available to the VT–3 examiner that complicate qualification requirements. For example, remote visual examination can be substituted for direct visual examination resulting in the use of special test equipment. The NRC concludes that testing is required to demonstrate that VT–3 examiners are knowledgeable regarding the different requirements associated with the examination method, and that these testing requirements are consistent with other NDE methods in CP–189 that require testing to demonstrate the required knowledge. Therefore, § 50.55a(b)(2)(xviii)(C) is revised to clarify that the alternative qualification provisions for VT–3 visual examination personnel in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2317 may be used provided that VT–3 examination personnel pass an initial test and a retest every 3 years.

2.2.6 Substitution of Alternative Methods

Section 50.55a(b)(2)(xix) in the proposed rule would prohibit the use of the provision in IWA–2240 (1998 Edition, 1999 Addenda, and 2000 Addenda) and IWA–4520(c) (1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda), which allows alternative examination methods, a combination of methods, or newly developed techniques to be substituted for the methods specified in the Construction Code, provided the Authorized Nuclear Inservice Inspector (ANII) is satisfied that the results are demonstrated to be equivalent or superior to those in the Construction Code. The revision to IWA–2240 changed the applicability of the paragraph from Section XI only (ISI) to both Sections III and XI (design/construction and ISI).

A number of commenters stated that editions and addenda of Section XI approved by the NRC since the 1974 Edition of Section XI allow ANIIs to approve the substitution of alternative methods, a combination of methods, or newly developed techniques for the examinations specified in Section XI, Division 1. For example, the ANII can approve the substitution of an eddy current examination for a surface examination requirement in IWB and IWC of Section XI provided the ANII is satisfied that the results of the eddy current examination are equivalent or superior to those of the surface examination. Most of the commenters stated that the NRC should accept the revised provisions in the 1998 Addenda, 1999 Addenda, and 2000 Addenda of IWA–2240 and the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–4520(c) that extend the substitution of the alternative examination provisions to the examinations specified in the Construction Code when performing repair/replacement activities. The commenters stated that ANII qualifications require detailed knowledge of the different examination methods addressed in Section XI and the Construction Code. One commenter stated that ASME QAI–1–1995, "Qualifications for Authorized

Inspection," is the applicable qualification standard that must be used to qualify ANIIs, and that ASME QAI-1-1995 requires that ANIIs be certified in Section XI and Construction Code requirements. An example of a use of the revised provisions provided by the commenters indicated that in some instances it may be a hardship or impractical to perform a radiographic (RT) examination during a Section XI repair/replacement activity as specified in the Construction Code. The revised provisions in Section XI would allow the substitution of an alternative method such as an UT examination for the RT examination provided that the ANII is satisfied that the results of the UT examination are equivalent or superior to the RT examination specified in the Construction Code.

The NRC agrees that the provisions in IWA-2240 that allow the ANII to approve the substitution of alternative examination methods, a combination of methods, or newly developed techniques for the methods specified in Section XI, Division 1, have been approved by the NRC since 1974. The NRC has reviewed the qualification standard in ASME QAI-1-1995, and agrees that ANIIs are required to be knowledgeable regarding the NDE methods, qualification requirements, and other requirements in Section XI and the Construction Code. However, the NRC believes that the substitution of alternative methods for those specified in the Construction Code is significantly more complex than what was previously approved by the NRC in editions and addenda of IWA-2240 earlier than the 1998 Edition. For example, there are many factors that have to be evaluated when substituting a UT examination for an RT examination required by the Construction Code. Consideration needs to be given to the thickness of the weld, volume of the UT examination, appropriate UT technique, UT examination coverage criteria, UT examination procedure (Section V or Section XI), and performance demonstration methodology; calibration block material, thickness, and size; flaw evaluation acceptance criteria, and demonstration and qualification criteria for single-sided UT examinations. Weld material would also be a critical factor when considering the substitution of a UT examination for an RT examination. It may not be appropriate to allow the substitution of a UT examination for an RT examination for certain materials such as ferritic or austenitic cast products and corrosion resistant cladding with butt welds. Substitution of a UT examination for an RT examination may be acceptable for dissimilar metal welds but would require additional factors to be evaluated. The NRC finds that there is a lack of guidance in the Code to ensure proper consideration of factors when substituting alternative examinations for the examinations specified in the Construction Code. The NRC believes that a standardized repeatable methodology that can be consistently used among all licensees is needed not only to demonstrate that the alternative method is equivalent or superior to that specified in the Construction Code, but also to ensure consistent application and implementation of IWA-2240 and IWA-4520(c). Furthermore, the NRC notes that the ASME is currently developing a Code Case that will provide the necessary guidance to allow the substitution of a UT examination with an RT examination when an RT examination is required by the Construction Code. Therefore, § 50.55a(b)(2)(xix) in the proposed rule is adopted without change.

2.2.7 System Leakage Tests

Section 50.55a(b)(2)(xx) in the proposed rule would have required that the pressure and temperature hold time requirements in the 1995 Edition of IWA-5213(a) be applied in place of the revised provisions of the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA-5213(a) when performing system leakage tests.

Many commenters objected to this modification because pressure and temperature hold time requirements in the 1995 Edition of IWA-5213(a) imposed by the modification place what the commenters believe to be an undue burden on utilities. One commenter noted that the ASME is currently developing a new revision to IWA-5213 to clarify the pressure and temperature hold time requirements in IWA-5213(a). A number of commenters stated that the NRC is arbitrarily choosing the pressure and temperature hold times in the 1995 Edition, and that the NRC should justify the use of the pressure and temperature hold times in the 1995 Edition.

The NRC normally requires that the Code revision most recently approved by the NRC be used when it does not approve the use of a later Code provision. Since the NRC has not approved the elimination of the pressure and temperature hold times in 1995 Addenda of IWA-5213, the NRC proposed to require the use of the pressure and temperature provisions in the 1995 Edition. The NRC agrees with the commenters that the changes in the 1989 Addenda through the 1995 Edition in conjunction with the proposed modification would create unintended test conditions. For example, some systems are not designed to operate at test conditions for the period of time necessary to meet the hold time conditions. Also, hold times are not necessary for leakage tests of Class 1 components because these leakage tests are normally performed following each refueling outage as the reactor is heating up. The heatup process of the reactor is performed within the pressure-temperature constraints of the heatup curve in the plant technical specifications. These constraints limit the rate of temperature and pressure increase resulting in a heatup period of several hours. In light of the substantial length of time required for the reactor heatup process, sufficient time is available for leakage from the Class 1 system to collect in sufficient quantity to be detectable by visual examination. Holding the Class 1 components for additional time at this temperature and pressure is unnecessary to accomplish the purpose of the pressure test.

In consideration of the public comments, the NRC has revised the pressure and temperature hold time requirements in § 50.55a(b)(2)(xx) to be consistent with the revisions recommended in several of the public comments (to use the provisions contained in the 1989 Edition of the ASME Code). This is also consistent with the current ASME proposed revision to the pressure and temperature hold times in IWA-5213. Section 50.55a(b)(2)(xx) requires a 10-minute holding time after attaining test pressure for Class 2 and Class 3 components that do not normally operate during operation, and no holding time is required for the remaining Class 2 and Class 3 components provided that system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

2.2.8 Table IWB-2500-1 Examination Requirements

Section 50.55a(b)(2)(xxi)(A) in the proposed rule would require licensees to use the provisions in the 1998 Edition of Table IWB-2500-1, Examination Category B-D, for Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) when using the 1999 Addenda and the 2000 Addenda. The 1999 Addenda eliminated the pressurizer and steam generator (SG) nozzle inside-radius inspections in Table IWB-2500-1, Examination Category B-D, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B).

Several commenters summarized the results of a white paper developed by the ASME that provides the technical basis for eliminating the pressurizer and SG nozzle inside-radius UT examinations from Table IWB–2500–1. The commenters explained the difficulties associated with performing UT examinations of pressurizer and SG nozzle inside radii. Radiation exposure to personnel who conduct the UT examinations is a significant concern because the pressurizer and SG nozzles are located in very high radiation areas. The geometry and material of the nozzles significantly complicate the UT examination procedure making it difficult to obtain meaningful UT data. The commenters stated that the basis for eliminating the pressurizer and SG nozzle inner radius examinations is that a review of UT and visual examination data from pressurizer and SG nozzle inner radius examinations reveal that no service-induced flaws were detected in any of the examinations performed. Commenters claimed that pressurizer and SG nozzle cracking incidents have not occurred at any nuclear facilities, and that structural integrity evaluations of the nozzles indicate that leakage would occur from a through-wall flaw before any integrity problems would occur (*ie.*, the nozzle would leak before it failed). In addition, a risk assessment indicated that the failure probability of the nozzles is extremely low under plant operating conditions, and shows that there is no change in risk if pressurizer and SG nozzle inner radius examinations are eliminated. Finally, the commenters stated that the NRC has granted relief from the pressurizer and SG inside-radius UT examination requirements in Table IWB–2500–1 to many licensees because of these concerns associated with occupational exposure and difficulty in obtaining meaningful UT data.

The NRC disagrees. Operating history alone does not provide adequate justification to eliminate examinations of the pressurizer and SG nozzle inside radii because operational experience also has demonstrated that components degrade as they age. Although pressurizer and SG nozzle cracking incidents have not occurred, cracks have been identified in other nozzles such as the feedwater nozzles. Furthermore, a leak-before-break evaluation is not adequate justification to eliminate the examination of the pressurizer and SG nozzle inside radii because the primary purpose of the ISI requirements in Section XI is to identify and correct component degradation before it becomes significant. Leakage from any pressurizer or SG nozzle would be significant because such leakage would represent an unisolable breach of the reactor coolant pressure boundary.

The NRC agrees that a number of licensees have requested relief from the UT examination requirements for SG nozzle inner radii and pressurizer nozzle inner radii. In these cases, the NRC has authorized, as an alternative to UT examination, the performance of a visual examination which utilizes equipment with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack. The flaw length acceptance criteria specified for the UT examination in Table IWB–3512–1 is applicable to the visual examination. The primary degradation mode for these nozzles is fatigue which produces hairline surface indications that network along the circumference of the nozzle at the inner radius section. Ultrasonic examination of the inner radii from the outside surface should detect these indications. However, even with the use of improved technology from the outside surface, the complex geometry of these nozzle inner radius sections prevents complete coverage. Visual examination for some of these nozzles from the inside surface is easier and less costly to accomplish, and coverage is more complete. The examinations can be performed when the pressurizer and SG are opened for other maintenance or inspection activities. Use of video equipment with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack is similar to UT examination regarding the capability of detecting fatigue-type cracks on nozzle inside radii before they become detrimental to structural integrity.

Therefore, § 50.55a(b)(2)(xxi)(A) is revised to allow the option of performing a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria of Table IWB–3512–1 in place of a UT examination. Section 50.55a(b)(2)(xxi)(A) requires that licensees use the provisions of Table IWB–2500–1, Examination Category B–D, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition when using the 1999 Addenda and the 2000 Addenda. A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Table IWB–3512–1 may be performed in place of a UT examination.

Section 50.55a(b)(2)(xxi)(B) in the proposed rule would require that licensees apply the provisions in the 1995 Edition of Table IWB–2500–1, Examination Category B–G–2, Item B7.80 when using the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda. The 1995 Edition and earlier editions and addenda of Section XI require a visual examination of control rod drive (CRD) housing bolting using the VT–1 visual examination method whenever the CRD housing is disassembled. The requirement to examine CRD bolting whenever the CRD housing is disassembled was deleted in the 1995 Addenda.

Several commenters stated that § 50.55a(b)(2)(xxi)(B) should be deleted because the skill of the craft and maintenance practices are sufficient to ensure that bolting is not damaged during maintenance activities. The NRC agrees that the scope of Section XI does not normally include examinations that are conducted during routine maintenance activities, but notes there may be maintenance-related activities associated with ISI. The ISI of components to verify that service-related degradation is not occurring is within the scope of Section XI.

The majority of the commenters stated that no degradation of CRD bolting has occurred in 30 years of experience, and hence the requirement to examine the CRD bolting should be eliminated. The NRC disagrees. Operating history alone does not provide adequate justification to eliminate examinations of CRD bolting because operational experience also has demonstrated that components degrade as they age. Furthermore, the NRC is aware of an example where CRD bolting was replaced in two units because examination of CRD bolting identified cracks.

Several commenters stated that the NRC is misinterpreting the ASME Code because Item B7.80 of Table IWB–B7.80 does not require that the CRD housing be disassembled to perform the examination of CRD bolting. The NRC notes that although the Code does not require disassembly of the CRD housing to examine the bolting, Item B7.80 of Table IWB–2500–1 in the 1995 Edition and earlier editions and addenda of Section XI states that the extent and frequency of the examination is to include bolts, studs, and nuts in CRD housings when disassembled. The NRC finds that the 1995 Edition and earlier editions and addenda of Section XI only require that CRD bolting be examined when the CRD housing is disassembled

such as during a repair or maintenance activity.

Several other commenters stated that since CRD mechanisms are usually contaminated and in high radiation areas, elimination of the bolting examinations reduces radiation exposure to personnel. The NRC notes that CRD bolting is normally relocated to a storage area after disassembly of the CRD housing. Therefore, VT–1 examination personnel typically examine the bolting when it is removed and remotely located from the CRD mechanism, reducing the exposure to individuals.

One commenter requested that the NRC revise § 50.55a(b)(2)(xxi)(B) to include a statement that only CRD bolting that is reused is required to be examined. It was the NRC's intent to require examination of the CRD bolting material only if it was to be reused. Therefore, § 50.55a(b)(2)(xxi)(B) is revised to clarify that only CRD bolting that is reused must be re-examined.

Section 50.55a(b)(2)(xxi)(C) in the proposed rule would require that licensees use the provisions in the 1995 Addenda of Table IWB–2500–1, Examination Category B–K, Item B10.10, when using the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda for the examination of welded attachments to pressure vessels. The 1997 Addenda permits performance of a single-side surface examination in place of a surface examination from both sides of the weld, whereas the 1995 Addenda requires the performance of a single-side volumetric examination of the weld in place of surface examination of the inaccessible surface if surface examination from both sides of the weld is not performed.

Several commenters noted that volumetric examination of reactor pressure vessel (RPV) skirt welds is not practical because UT calibration blocks were typically not supplied for RPV skirt welds and the UT performance demonstration requirements of Appendix VIII do not address RPV support attachment welds. If a licensee wanted to perform a volumetric examination in place of surface examination of both surfaces, it would have to fabricate its own calibration blocks and sample specimens, develop its own procedures, and set up its own demonstration program.

The NRC recognizes that UT examination of RPV skirt welds is not addressed in Appendix VIII at this time. However, the applicable examination requirements are addressed in Article I–2000 of Section XI which in turn references Section V of the ASME BPV Code. Furthermore, Section V of the ASME BPV Code addresses the qualification and use of suitable alternative calibration blocks.

Commenters stated that access under the RPV bottom head for performing a visual examination is a confined space that is also a high radiation area. The inside surface geometry is such that preparation for a surface examination is difficult, thus extending the time spent in the high radiation area. The commenters conclude that the radiation exposure to personnel who examine the inside surface of the RPV skirt weld is not justified. The NRC agrees that access to such confined spaces is very difficult. However, the NRC also believes that the 1995 Addenda of the ASME Code, which already provides for an alternative UT examination in place of a surface examination of the inaccessible surface, appropriately accommodates the commenters concerns. These UT examinations are performed on the accessible surface of the RPV skirt welds. Therefore, personnel are not required to enter the confined space area under the RPV bottom head.

Commenters also stated that RPV skirt weld materials are very flaw-tolerant, with slow flaw-propagation rates. Flaws originating on the inside surface would grow through-wall long before their length would threaten the structural integrity/function of the weld. The NRC notes that the assumption that flaws will be detected before affecting structural integrity is an assumption based on limited surface examination experience and is not supported by rigorous study. The commenters have not presented any analyses or studies which support such an assumption.

Commenters stated that RPV skirt welds are similar to non-pressure boundary core shroud circumferential welds in boiling water reactors. The commenters also stated that safety analyses performed by the Boiling Water Reactor Vessel & Internals Program found that core shroud circumferential welds could be cracked through-wall for 360° and still perform their function. The NRC considers the inference that the structural performance, response, and safety implications of operating with a significantly cracked RPV skirt weld is no different than operating with significantly cracked core shroud circumferential welds to be inappropriate. Operation with cracked core shroud welds has been extensively evaluated for all operating and accident loading conditions. The core shroud is contained within the confines of the reactor pressure vessel with positive restraints holding it in place to assure integrity and adequate coolant flow through the core. However, operation with a significantly cracked RPV skirt weld has not been evaluated. Therefore, the NRC has no basis to conclude that operation under such conditions is acceptable. Commenters also claim that the excellent service history of RPV skirt welds demonstrates that inside surface examinations of welds is not warranted. The NRC considers that operating history alone does not provide adequate justification to eliminate examinations of components because operational experience has also demonstrated that components degrade as they age. Therefore, § 50.55a(b)(2)(xxi)(C) in the proposed rule is adopted without change.

2.2.9 Supplemental Annual Training Requirements for Ultrasonic Examiners

Section 50.55a(b)(2)(xxii) in the proposed rule would require licensees to apply the UT examiner supplemental annual training provisions in the 1998 Edition of Paragraph VII–4240 of Appendix VII, in place of the revised provisions in the 1999 Addenda and 2000 Addenda of VII–4240.

Several commenters stated that the NRC position on training requirements for UT examiners in § 50.55a(b)(2)(xxii) of the proposed rule is inconsistent with the NRC position on training requirements for UT examiners in final rule 64 FR 51370 (September 22, 1999). The commenters noted that the final rule imposed § 50.55a(b)(2)(xiv) because the 10-hour classroom training requirement in VII–4240 was inadequate. The commenters stated that Code Case N–583, "Annual Training Alternative," was developed by the ASME to specifically address the NRC concern with the 10-hour classroom training requirement in the 1995 Edition and 1996 Addenda of VII–4240. Code Case N–583 was incorporated into the 1999 Addenda of VII–4240, replacing the 10-hour classroom training requirement with an 8-hour training requirement to analyze data from material or welds containing flaws similar to those that may be encountered during UT examinations. The commenters stated that the revised training requirements in the 1999 Addenda of VII–4240 are an improvement over the training requirements in the 1998 Edition and earlier editions and addenda of VII–4240. The revised training requirements provide specific criteria that result in uniform training programs among all licensees.

The commenters have clarified to the NRC that the training requirements in the 1999 Addenda and 2000 Addenda of VII–4240 specify hands on training in

place of classroom training. Therefore, § 50.55a(b)(2)(xxii) in the proposed rule is not adopted because after further clarification, the NRC finds that the training requirements in 1999 Addenda and 2000 Addenda of VII–4240 are consistent with the NRC position on training requirements for UT examiners in final rule 64 FR 51370 (September 22, 1999).

Commenters requested that licensees be allowed to substitute the supplemental practice in the 1999 Addenda and 2000 Addenda of VII–4240 for the existing hands-on training requirement in § 50.55a(b)(2)(xiv). The NRC finds that the supplemental practice as described in VII–4240 of Supplement VII of Section XI, 1999 Addenda and 2000 Addenda, is an acceptable alternative to the existing hands-on training requirement in § 50.55a(b)(2)(xiv) provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. Therefore, § 50.55a(b)(2)(xiv) is revised to allow the option of performing the supplemental practice as described in VII–4240 of Supplement VII of Section XI, 1999 Addenda and 2000 Addenda, or the existing hands-on training requirement.

2.2.10 Underwater Welding

Section 50.55a(b)(2)(xxiii) in the proposed rule would require licensees to demonstrate the acceptability of the underwater welding method through the use of a mockup using material with similar neutron fluence levels, when welding irradiated material underwater in accordance with the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–4660.

Several commenters stated that the use of a mockup to demonstrate the acceptability of an underwater welding method is impractical due to unavailability of materials with similar neutron fluence levels, personnel exposure, high-cost of mockups, and handling and disposal requirements. The commenters also stated that the industry is currently developing an acceptable underwater welding technique for irradiated materials in conjunction with the Boiling Water Reactor Vessel & Internals Project that will be submitted to the NRC for approval.

The NRC proposed the use of a mockup because underwater weld repairs using conventional welding techniques on in-vessel components exposed to high neutron fluences may be unsuccessful due to helium-induced cracking and radiation damage, unless special welding techniques are used. The NRC has revised the proposed underwater welding mockup requirement because of the impracticality of developing and using a mockup with similar neutron fluence levels. Section 50.55a(b)(2)(xxiii) is revised to prohibit the use of the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–4660 to weld irradiated material underwater. Licensees must obtain NRC approval in accordance with § 50.55a(a)(3) of the technique used to weld irradiated material underwater. Section 50.55a(b)(2)(xxiii) of the proposed rule is presented here in the final rule as § 50.55a(b)(2)(xii).

2.3 Appendix VIII to Section XI

This final rule extends the applicability of the existing regulations in § 50.55a(b)(2)(xv) to the 1997 Addenda, the 1998 Edition, 1999 Addenda, and 2000 Addenda of Appendix VIII of Section XI of the ASME BPV Code.

2.3.1 Examination Coverage for Dissimilar Metal Pipe Welds

The existing requirements in § 50.55a(g)(6)(ii)(C)(1) state that Supplement 10, "Qualification Requirements for Dissimilar Metal Piping Welds," of Appendix VIII to Section XI must be implemented by November 22, 2002. Therefore, the proposed rule would have updated § 50.55a(b)(2)(xv)(A) to reference Supplement 10. Specifically, the proposed rule would revise §§ 50.55a(b)(2)(xv)(A)(*1*) and (A)(*2*) to provide UT examination coverage criteria for dissimilar metal piping welds. Examination coverage criteria for dissimilar metal piping welds are specified in the 1989 Edition and earlier editions and addenda of Appendix III of Section XI. Appendix VIII was added in the 1989 Addenda of Section XI, and Section XI would require that the UT examination criteria for piping welds in Appendix VIII supercede the examination criteria in Appendix III. Although Appendix VIII addresses qualification of personnel, procedures, and equipment used to conduct UT examinations of dissimilar metal piping welds, Appendix VIII (unlike Appendix III) does not address UT examination coverage criteria for dissimilar metal piping welds.

The commenters agreed that §§ 50.55a(b)(2)(xv)(A), (A)(*1*) and (A)(*2*) should be revised to provide UT examination coverage criteria for dissimilar metal piping welds. However, the commenters did not agree with the examination coverage criteria in § 50.55a(b)(2)(xv)(A)(*2*) of the proposed rule requiring that dissimilar metal welds be examined from the austenitic side of the weld when examination from both sides is not possible. The commenters stated that § 50.55a(b)(2)(xv)(A)(*2*) should be revised to allow coverage from either the austenitic or ferritic side of the weld when UT examination from both sides is not possible because the composition of the base material is of minor consequence when compared to the effects of the austenitic weld material. Furthermore, the commenters argued that the examination should be conducted from the side of the weld that is most accessible.

The NRC does not agree that the composition of the base material is of minor consequence when compared to the effects of austenitic weld material. There is a higher probability and reliability of identifying flaws in dissimilar metal welds when using a UT procedure qualified to perform examinations from the austenitic side than when using a UT procedure qualified to perform examinations from the ferritic side. Therefore, coverage from the austenitic side of the weld is preferred when UT examination from both sides is not possible.

Sections 50.55a(b)(2)(xv)(A) and (A)(*1*) in the proposed rule are adopted without change. Section 50.55a(b)(2)(xv)(A)(*2*) is revised to clarify that dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld, and that the examination from the austenitic side of the weld may be used to perform examinations from either side of the weld.

2.3.2 Reactor Vessel Single Side Examinations

The proposed rule would remove the existing § 50.55a(b)(2)(xv)(G)(*4*) because the examination criteria are redundant with the examination criteria contained in § 50.55a(b)(2)(xv)(G)(*3*) and, therefore unnecessary. Both §§ 50.55a(b)(2)(xv)(G)(*3*) and (*4*) allow credit for the full volume when the examination volume is covered from a perpendicular and parallel direction. There were no public comments on the proposed revision; therefore, § 50.55a(b)(2)(xv)(G)(*4*) is removed.

2.3.3 Qualification Test Samples

The revision to § 50.55a(b)(2)(xv)(K)(*1*)(*i*) in the proposed rule would resolve a discrepancy between the existing §§ 50.55a(b)(2)(xv)(K)(*1*)(*i*) and 50.55a(b)(2)(xv)(K)(*4*). Currently, § 50.55a(b)(2)(xv)(K)(*1*)(*i*) states that

flaws which are perpendicular to the weld *are not required to be included in* the qualification test sample. This requirement conflicts with a provision in § 50.55a(b)(2)(xv)(K)(*4*), which states that test samples must contain flaws that are perpendicular to the weld in the inner 15 percent of the weld, but that these same flaws are not required to be located in the outer 85 percent of the weld. There were no public comments on the proposed revision; therefore, the revision to § 50.55a(b)(2)(xv)(K)(*1*)(*i*) is adopted without change.

2.3.4 Implementation of Appendix VIII to Section XI

Section 50.55a(b)(2)(xv)(M) in the proposed rule would clarify that only those provisions in Supplement 12 to Appendix VIII that relate to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations must be implemented. Supplement 12 provides coordinated implementation provisions for the performance demonstrations in Supplements 2, 3, 10, and 11 of Appendix VIII; however, with the exception of the coordinated implementation of Supplement 3 to Supplement 2 performance demonstration, the other coordinated implementation provisions in Supplement 12 are incomplete. Supplement 12 does not provide provisions for implementing single-side examinations as part of the coordinating process, or provide provisions for the coordinated implementation of Supplement 2 or Supplement 11 performance demonstrations to Supplements 3 and 10. There were no public comments on the proposed § 50.55a(b)(2)(xv)(M); therefore, § 50.55a(b)(2)(xv)(M) is adopted without change.

Section 50.55a(g)(6)(ii)(C)(*1*) in the proposed rule would clarify that Appendix VIII to Section XI, 1995 Edition with the 1996 Addenda, as well as its supplements, are mandatory and must be implemented. Although the final rule that implemented Appendix VIII (64 FR 51370; September 22, 1999) requires a phased implementation of Appendix VIII over a 3-year period, the final rule addressed the implementation of the Appendix VIII supplements only and failed to mention the implementation of Appendix VIII itself. The failure to address the implementation of Appendix VIII was an oversight. Section 50.55a(g)(6)(ii)(C)(*1*) in the proposed rule would also eliminate Supplements 12 and 13 of Appendix VIII from the implementation schedule that is currently in § 50.55a(g)(6)(ii)(C)(*1*). Supplements 12 and 13 coordinate the *implementation of selected aspects of* Supplements 2, 3, 4, 5, 6, 7, 10, and 11 of Appendix VIII. Since the implementation schedule for Supplements 2, 3, 4, 5, 6, 7, 10, and 11 of Appendix VIII is addressed in § 50.55a(g)(6)(ii)(C)(*1*), the imposition of a mandatory implementation date for Supplements 12 and 13 is redundant. There were no public comments on either of the proposed revisions; therefore, the revisions to § 50.55a(g)(6)(ii)(C)(*1*) are adopted without change.

Section 50.55a(g)(6)(ii)(C)(*2*) in the proposed rule would clarify that the requirements of Appendix VIII and the supplements to Appendix VIII to Section XI, of the 1995 Edition with the 1996 Addenda are mandatory when implementing the 1989 Edition and earlier editions and addenda of IWA–2232 of Section XI. Paragraph IWA–2232 provides rules for conducting UT examinations. Appendix VIII was introduced into Section XI in the 1989 Addenda. Before that time, Appendix VIII did not exist in Section XI. Therefore, the 1989 Edition and earlier editions and addenda of IWA–2232 do not reference Appendix VIII. It is not clear to some licensees that they are required to perform UT examinations using personnel, procedures, and equipment qualified in accordance with Appendix VIII. The NRC believes that the final rule dated September 22, 1999 (64 FR 51370), by imposing an expedited implementation of the supplements to Appendix VIII to Section XI, 1995 Edition with the 1996 Addenda, makes it clear that all licensees are required to implement the provisions of Appendix VIII, including those licensees implementing the 1989 Edition or earlier editions and addenda of IWA–2232.

A commenter pointed out that § 50.55a(g)(6)(ii)(C)(*2*) in the proposed rule is inconsistent with the statement of considerations for the proposed rule. The NRC agrees. The purpose of § 50.55a(g)(6)(ii)(C)(*2*) in the proposed rule was to clarify the relationship between the 1989 Edition and earlier editions and addenda of IWA–2232 of Section XI, and Appendix VIII of Section XI. However, in making this clarification, the NRC inadvertently worded § 50.55a(g)(6)(ii)(C)(*2*) such that licensees would be required to update their Appendix VIII program to the latest edition and addenda of Section XI incorporated by reference in § 50.55a(b)(2) following every update. It was not the intent of the NRC to revise the existing 120-month inspection interval update requirement. Therefore, § 50.55a(g)(6)(ii)(C)(*2*) is revised to clarify that licensees implementing the 1989 Edition and earlier editions and addenda of IWA–2232 of Section XI must implement the 1995 Edition with the 1996 Addenda of Appendix VIII of Section XI.

2.4 ASME OM Code

The final rule revises § 50.55a(b)(3) to incorporate by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code, and extends the applicability of the existing regulations in §§ 50.55a(b)(3)(ii), 50.55a(b)(3)(iii), 50.55a(b)(3)(iv), and 50.55a(b)(3)(v) to the 1997 Addenda, 1998 Edition, 1999 Addenda, and the 2000 Addenda of the ASME OM Code. Subsections of the ASME OM Code were renumbered in the 1998 Edition; therefore, §§ 50.55a(b)(3)(ii), 50.55a(b)(3)(iii), and 50.55a(b)(3)(iv) are revised and § 50.55a(b)(3)(iv)(D) is added to account for the renumbering.

Although the technical requirements in § 50.55a(b)(3)(ii) were not revised in the proposed rule, several commenters stated that the reference to motor-operated valve (MOV) stroke-time testing in the existing § 50.55a(b)(3)(ii) is confusing because there are other MOV test requirements in the ASME OM Code (such as position indication and seat leakage testing) that are applicable in addition to stroke-time testing. The commenters suggested that a licensee might incorrectly interpret § 50.55a(b)(3)(ii) as requiring that only MOV stroke-time testing be performed in accordance with the OM Code. The NRC believes the current regulation clearly states that licensees must meet all of the ASME Code provisions for testing MOVs. The NRC is not aware of any misunderstanding among licensees regarding the intent of the regulatory requirement for MOVs. However, to avoid any potential confusion in the future, § 50.55a(b)(3)(ii) is revised to clarify that licensees must comply with the provisions of the ASME OM ISTC Code for testing MOVs.

Section 50.55a(b)(3)(vi) in the proposed rule would require an exercise interval of 2 years for manual valves within the scope of the ASME OM Code rather than the exercise interval of 5 years specified in the 1999 Addenda and the 2000 Addenda of the ASME OM Code. The 1998 Edition of the ASME OM Code specified an exercise interval of 3 months for manual valves within the scope of the Code. The 1999 Addenda to the ASME OM Code revised ISTC–3540 to extend the exercise frequency for manual valves to 5 years.

A number of commenters stated that § 50.55a(b)(3)(vi) in the proposed rule should be withdrawn because sufficient justification exists to allow the extension of the exercise interval for manual valves to 5 years. The justification for the 5-year frequency is the simplicity of manual valves (limited number of failure causes) and that the ASME OM Code allows other valves (safety and relief valves) to be tested on a 5-year or longer frequencies.

The NRC does not agree that there is sufficient justification to extend the exercise interval for manual valves to 5 years. The NRC review of licensee IST programs indicate that manual valves are exercised every 3 months except in instances where it is impractical to operate valves during unit operation. Valves are then exercised when the unit is in a cold shutdown condition, and the exercise frequency cannot exceed 2 years. Therefore, a 2-year interval for exercising manual valves is justified because the available manual valve exercise data supports the 2-year interval. The NRC has approved longer test intervals for other types of valves in the ASME OM Code but the longer test intervals include additional means to determine component degradation. For example, although the ASME OM Code test strategy for Class 2 and 3 relief valves has a testing interval of 10 years, Class 2 and 3 relief valves are subject to grouping and sample expansion if there is a test failure. Manual valves that are required to be exercised are not subject to grouping and sample expansion. Furthermore, obstruction from silting or blockage, or corrosion of valve internals are possible failure modes for safety-related manual valves that are not applicable to other types of valves with longer test intervals. Exercising manual valves minimizes both of these failure modes and also allows for more immediate detection if an obstruction or corrosion induced failure occurs. Section 50.55a(b)(3)(vi) is revised to clarify that the interval for exercising manual valves may not exceed 2 years when using the 1999 Addenda and 2000 Addenda of ISTC–3540. Licensees are not prohibited from exercising manual valves more frequently than every 2 years.

3. Section-by-Section Analysis of Substantive Changes

Paragraph (b)(1). This paragraph incorporates by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III, Division 1, of the ASME BPV Code. New applicants for a nuclear power plant submitting an application for a construction permit under 10 CFR 50 or design certification under 10 CFR 52 are required to use the 1998 Edition up to and including the 2000 Addenda for the design and construction of the reactor coolant pressure boundary and Quality Group B and C components.

Paragraph (b)(1)(ii). This paragraph extends the applicability of the existing regulation on weld leg dimension requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III of the ASME BPV Code. Applicants and licensees using these Edition and Addenda are not allowed to apply paragraph NB–3683.4(c)(1), Footnote 11 to Figure NC–3673.2(b)–1, and Figure ND–3673.2(b)–1.

Paragraph (b)(1)(iii). This paragraph extends the applicability of the existing regulation on seismic design requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III of the ASME BPV Code. Applicants and licensees using these edition and addenda are not allowed to use Articles NB–3200, NB–3600, NC–3600, and ND–3600.

Paragraph (b)(1)(v). This paragraph extends the applicability of the existing regulation on independence of inspection requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III of the ASME BPV Code. Applicants and licensees using these edition and addenda are not allowed to apply Sub-subparagraph NCA–4134.10(a).

Paragraph (b)(2). This paragraph incorporates by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI, Division 1, of the ASME BPV Code. Licensees of nuclear power plants are required to use the 1998 Edition up to and including the 2000 Addenda when updating their ISI programs in their subsequent 120-month interval under § 50.55a(g)(4)(ii).

Paragraph (b)(2)(vi). This paragraph clarifies that either the 1992 Edition with the 1992 Addenda or the 1995 Edition with the 1996 Addenda of Subsection IWE and Subsection IWL as modified and supplemented by the requirements in § 50.55a(b)(2)(viii) and § 50.55a(b)(2)(ix) must be used when implementing the initial 120-month inspection interval for the containment inservice inspection requirements. Successive 120-month interval updates must be implemented in accordance with § 50.55a(g)(4)(ii).

Paragraph (b)(2)(viii). This paragraph extends the applicability of the existing regulation in paragraph (b)(2)(viii)(E) on concrete containment examination requirements to the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWL, and clarifies that the new modification in paragraph (b)(2)(viii)(F) applies only to the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWL.

Paragraph (b)(2)(viii)(F). This paragraph requires that personnel who perform visual inspections of containment surfaces and tendon anchorage hardware, wires, or strands be qualified in accordance with IWA–2300 in place of the "owner-defined" personnel qualification provision in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWL–2310(d).

Paragraph (b)(2)(ix). This paragraph clarifies that the existing modifications in paragraphs (b)(2)(ix)(A) through (E) of this section on examination of metal containments and liners of Class CC components apply to the 1992 Edition with the 1992 Addenda or the 1995 Edition with the 1996 Addenda of IWE. It also extends the applicability of the regulations in paragraphs (b)(2)(ix)(A) and (b)(2)(ix)(B) to the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE, and clarifies that the new modifications in paragraphs (b)(2)(ix)(F) through (I) apply only to the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE.

Paragraph (b)(2)(ix)(F). This paragraph requires that VT–1 and VT–3 examinations of containment surfaces be conducted in accordance with IWA–2200, and that personnel who perform visual inspections of containment surfaces be qualified in accordance with IWA–2300 in place of the "owner-defined" examination and personnel qualification provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE.

Paragraph (b)(2)(ix)(G). This paragraph requires that the VT–3 examination method be used to conduct the examinations in Items E1.12 and E1.20 in the 1998 Edition, 1999 Addenda, and 2000 Addenda of Table IWE–2500–1 in place of the "owner-defined" general visual examination provisions; the VT–1 examination method be used to conduct the examination in Item E4.11 of Table IWE–2500–1 in place of "owner-defined" detailed visual examinations; and an examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval.

Paragraph (b)(2)(ix)(H). This paragraph supplements the examination requirements for containment bolted connections that are in Item E1.11 of the 1998 Edition, 1999 Addenda, and 2000 Addenda of Table IWE–2500–1. Containment bolted connections that are disassembled during the scheduled

performance of the examinations in Item E1.11 of Table IWE–2500–1 must be examined using the VT–3 examination method. Flaws or degradation identified during the performance of a VT–3 examination must be examined in accordance with the VT–1 examination method. The criteria in the material specification or IWB–3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT–3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT–3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

Paragraph (b)(2)(ix)(I). This paragraph requires that the UT examination acceptance standard specified in the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–3511.3 for Class MC pressure-retaining components also apply to metallic liners of Class CC pressure-retaining components.

Paragraph (b)(2)(xi). This paragraph extends the applicability of the existing regulation on the use of IWB–1220 to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV Code. Licensees using editions and addenda later than the 1989 Addenda of Section XI are prohibited from exempting components from volumetric and surface examination as allowed by IWB–1220.

Paragraph (b)(2)(xii). This paragraph prohibits the use of the irradiated material underwater weld provisions in the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–4660. Licensees must obtain NRC authorization in accordance with § 50.55a(a)(3) of the method used to weld irradiated material underwater.

Paragraph (b)(2)(xiv). This paragraph allows 8 hours of annual practice as described in VII–4240 of Supplement VII of Section XI, 1999 Addenda and 2000 Addenda, to be performed in place of the existing hands-on training requirement in paragraph (b)(2)(xiv), provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

Paragraph (b)(2)(xv). This paragraph extends the applicability of the existing regulations on Appendix VIII specimen set and qualification requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV Code. Licensees choosing to use these modifications are required to apply all the modifications under paragraph (b)(2)(xv) except for those in (b)(2)(xv)(F) which are optional.

Paragraphs (b)(2)(xv)(A), (A)(1), and (A)(2). These paragraphs update the UT examination coverage criteria to include examination coverage criteria for dissimilar metal piping welds when using personnel, procedures and equipment that are qualified in accordance with Supplement 10 of Appendix VII to Section XI. Dissimilar metal welds must be examined axially and circumferentially. Where examination from both sides is not possible on dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld and may be used to perform examinations from either side of the weld.

Paragraph (b)(2)(xv)(G)(4). Paragraph (b)(2)(xv)(G)(*4*) is removed. This requirement is redundant given the requirement in paragraph (b)(2)(xv)(G)(*3*) and is unnecessary. As a result, this revision involves no substantive change.

Paragraph (b)(2)(xv)(K)(1)(i). This paragraph clarifies that flaws perpendicular to the weld located in the outer eighty-five (85) percent of the weld are not required to be included in the qualification test sample. The revision neither increases nor decreases current requirements, but clarifies conflicting requirements that currently exist.

Paragraph (b)(2)(xv)(M). This paragraph clarifies that only the provisions in Supplement 12 to Appendix VIII that are related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are required to be implemented.

Paragraph (b)(2)(xvii). This paragraph extends the applicability of the existing regulation on reconciliation of quality requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV Code. Licensees using IWA–4200 of this edition and these addenda are required to procure replacement and repair items under its approved quality assurance program required by Appendix B of 10 CFR 50. The limitation does not permit licensees to use IWA–4200 to procure repair and replacement items to be used in ASME Code safety-related applications that are manufactured under a non-nuclear code or non-nuclear standard without an approved quality assurance program.

Paragraph (b)(2)(xviii)(A). This paragraph requires that Level I and II NDE personnel be recertified on a 3-year interval in lieu of the 5-year interval specified in IWA–2314.

Paragraph (b)(2)(xviii)(B). This paragraph requires that IWA–2316 may only be used to qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with IWA–5211(a) and (b).

Paragraph (b)(2)(xviii)(C). This paragraph requires that when qualifying VT–3 examination personnel in accordance with IWA–2317, the proficiency of the training must be demonstrated by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

Paragraph (b)(2)(xix). This paragraph prohibits the use of the provisions in IWA–2240 and IWA–4520(c) which would allow alternative examination methods, a combination of methods, or newly developed techniques to be substituted for the methods specified in the Construction Code during repair and replacement activities.

Paragraph (b)(2)(xx). This paragraph supplements the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–5213(a) to require a 10-minute hold time after attaining test pressure for Class 2 and Class 3 components that are not in use during normal operating conditions, and no hold time for the remaining Class 2 and Class 3 components provided that system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

Paragraph (b)(2)(xxi)(A). This paragraph requires that licensees perform pressurizer and steam generator nozzle inside-radius inspections of Table IWB–2500–1, Examination Category B-D, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition. The 1999 Addenda and the 2000 Addenda of Section XI are not permitted to be used. A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, using the allowable flaw length criteria in Table IWB–3512–1, may be performed in place of a UT examination.

Paragraph (b)(2)(xxi)(B). This paragraph requires that the CRD bolting examinations of Table IWB–2500–1, Examination Category B-G–2, Item

B7.80, of the 1995 Addenda of Section XI be retained only for used CRD bolting in ISI programs when using the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI.

Paragraph (b)(2)(xxi)(C). This paragraph requires that the attachment weld single-side volumetric examination of Table IWB–2500–1, Examination Category B–K, Item B10.10, of the 1995 Addenda of Section XI be retained in ISI programs when using the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI.

Paragraph (b)(3). This paragraph incorporates by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code. Licensees of nuclear power plants are required to use the 1998 Edition up to and including the 2000 Addenda when updating their inservice testing programs in their subsequent 120-month inspection interval under § 50.55a(f)(4)(ii).

Paragraph (b)(3)(ii). This paragraph extends the applicability of the existing regulations on MOV test requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code. Licensees using this edition and these addenda are required to establish a program to ensure that MOVs continue to be capable of performing their design basis safety functions. This paragraph clarifies that licensees must comply with the provisions of the ASME OM ISTC Code for testing MOVs, and reconciles the different subsection and paragraph numbers of the ASME OM Code that were renumbered in the 1998 Edition and subsequent editions and addenda.

Paragraph (b)(3)(iii). This paragraph extends the applicability of the existing regulation that permits the use of Code Case OMN–1 in place of stroke time test requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code, and reconciles those subsections of the ASME OM Code that were renumbered in the 1998 Edition. The modification continues to allow, as a voluntary alternative, the use of Code Case OMN–1 in place of the stroke-time testing requirements of paragraph (b)(3)(ii) when using this edition and these addenda.

Paragraph (b)(3)(iv). This paragraph extends the applicability of the existing regulations in paragraphs (b)(3)(iv)(A), (B), and (C) on check valve condition monitoring requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code. There are no substantive changes in the requirements. This paragraph also reconciles the different subsection and paragraph numbers of the ASME OM Code that were renumbered in the 1998 Edition and subsequent editions and addenda.

Paragraph (b)(3)(iv)(D). There are no substantive changes to the check valve condition monitoring requirements in ASME OM Code in this paragraph. This paragraph reconciles the different subsection and paragraph numbers of that were renumbered in the 1998 Edition and subsequent editions and addenda.

Paragraph (b)(3)(v). This paragraph extends the applicability of the existing snubber ISI requirements to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME OM Code.

Paragraph (b)(3)(vi). This paragraph requires that manual valves within the scope of the ASME OM Code be exercised on a 2-year interval rather than the 5-year interval specified in the 1999 Addenda and 2000 Addenda of the ASME OM Code, provided that adverse conditions do not require more frequent testing. Paragraph ISTC–3540 of the ASME OM Code describes adverse conditions as harsh service environment, lubricant hardening, corrosive or sediment-laden process fluid, or degraded valve components.

Paragraph (g)(6)(ii)(B). The paragraph removes the containment examination requirements in §§ 50.55a(g)(6)(ii)(B)(*1*) through (*4*) because the implementation dates have expired and all licensees have completed the requirements (or a delay has been approved by an exemption); and redesignates the existing § 50.55a(g)(6)(ii)(B)(*5*) as § 50.55a(g)(6)(ii)(B). Licensees do not have to submit to the NRC staff for approval of their containment inservice inspection programs which were developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified modifications and limitations. The program elements and the required documentation must be maintained on site for audit.

Paragraph (g)(6)(ii)(C)(1). This paragraph clarifies that Appendix VIII to Section XI, 1995 Edition with the 1996 Addenda, as well as its supplements, must be implemented. Supplements 12 and 13 of Appendix VIII are eliminated from the implementation schedule.

Paragraph (g)(6)(ii)(C)(2). This paragraph clarifies the requirements of Appendix VIII and the supplements to Appendix VIII to Section XI. Licensees implementing the 1989 Edition and earlier editions and addenda of IWA–22323 of Section XI must implement the 1995 Edition with the 1996 Addenda of Appendix VIII of Section XI.

4. Generic Aging Lessons Learned Report

In July 2001, the NRC issued, "Generic Aging Lessons Learned (GALL) Report," NUREG–1801, Volumes 1 and 2, for use by applicants in preparing their license renewal applications. The GALL report evaluates existing generic programs, documents the basis for determining when generic existing programs are adequate without change, and documents when generic existing programs should be augmented for license renewal. Section XI, Division 1, of the ASME BPV Code is one of the generic existing programs in the GALL report that is evaluated as an aging management program (AMP) for license renewal. Subsections IWB, IWC, IWD, IWF, IWE, and IWL of the 1995 Edition up to and including the 1996 Addenda of Section XI of the ASME BPV for ISI were evaluated in the GALL report and the conclusions in the GALL report are valid for these edition and addenda.

In the GALL report Sections XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," XI.S1, "ASME Section XI, Subsection IWE," XI.S2, "ASME Section XI, Subsection IWL," and XI.S3, "ASME Section XI, Subsection IWF," describe the evaluation and technical basis for determining the adequacy of Subsections IWB, IWC, IWD, IWE, IWL, and IWF, respectively. In addition, many other AMPs in the GALL report rely in part, but to a lesser degree, on the requirements in the ASME Code, Section XI (*i.e.*, XI.M3, XI.M4, XI.M5, XI.M6, XI.M7, XI.M8, XI.M9, XI.M11, XI.M12, XI.M13, XI.M14, XI.M15, XI.M16, XI.M18, XI.M24, XI.M25, and XI.M32). These AMPs were evaluated for 10 specific elements with such attributes as scope of program, preventive actions, parameters monitored/inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience. If an applicant takes credit for a program in GALL, it is incumbent on the applicant to ensure that the plant program contains all the elements of the referenced GALL program. The GALL report contains one acceptable way to manage aging effects for license renewal. An applicant may propose alternatives for NRC review in its plant-specific license renewal application.

The NRC has completed an evaluation of Subsections IWB, IWC, IWD, IWE, IWF, and IWL of Section XI of the ASME BPV Code (1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda) as part of the § 50.55a

amendment process to ensure that the conclusions of the GALL report remain valid. Although some of the revisions in Section XI of the ASME BPV Code relax the provisions of the 1995 Edition with the 1996 Addenda, the revisions are acceptable (except as discussed below) and the conclusions of the GALL report remain valid. Accordingly, an applicant may use Subsections IWB, IWC, IWD, IWE, IWF, and IWL of Section XI of the ASME BPV Code (1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda) as acceptable alternatives to the requirements of the 1995 Edition up to and including the 1996 Addenda of the ASME Code, Section XI, referenced in the GALL AMPs without the need to submit these alternatives for NRC review in its plant-specific license renewal application. Similarly, a licensee approved for license renewal that relied on the GALL AMPs may use Subsections IWB, IWC, IWD, IWE, IWF, and IWL of Section XI of the ASME BPV Code (1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda) as acceptable alternatives to the AMPs described in the GALL report.

Several of the revisions to Subsections IWB, IWE, and IWL that are discussed in the preceding Section 2, Public Comments on Proposed Rule; and Final Rule, might affect the validity of the conclusions in the GALL report because provisions in the 1995 Edition up to and including the 1996 Addenda that address examination requirements and acceptance standards have been relaxed or eliminated in the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda. The new limitations and modifications in § 50.55a(b) require that the revised provisions be supplemented with additional inspection requirements as a condition for their use. The conclusions of the GALL report remain valid for the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME BPV Code with the use of these new limitations and modifications as discussed in this final rulemaking. However, it should be noted that the NRC is imposing these limitations and modifications to ensure consistency and an acceptable level of safety in the examination requirements and acceptance standards, and not solely to validate the conclusions in the GALL report.

The GALL report identified areas of the 1995 Edition with the 1996 Addenda of Section XI of the ASME Code that require augmentation for license renewal. A license renewal applicant may either augment their AMPS in these areas as described in the GALL report, or propose alternatives for NRC review in its plant-specific license renewal application. The GALL report's conclusions with respect to augmentation in connection with a license renewal application also apply when implementing the 1998 Edition, 1999 Addenda, and 2000 Addenda of Section XI of the ASME Code.

5. Availability of Documents

The NRC is making the documents identified below available to interested persons through one or more of the following methods as indicated.

Public Document Room (PDR). The NRC Public Document Room is located at 11555 Rockville Pike, Rockville, Maryland.

Rulemaking Website (Web). The NRC's interactive rulemaking Website is located at *http://ruleforum.llnl.gov.* These documents may be viewed and downloaded electronically via this Website.

NRC's Public Electronic Reading Room (PERR). The NRC's public electronic reading room is located at *http://www.nrc.gov/reading-rm/adams.html.*

NRC Staff Contact. Single copies of the Federal Register Notice, Regulatory Analysis, Environmental Assessment, and Resolution of Public Comments be obtained from Stephen Tingen, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Alternatively, you may contact Mr. Tingen at (301) 415–1280, or via e-mail at: *sgt@nrc.gov.*

Document	PDR	Web	PERR	NRC staff
Federal Register Notice	X	X	....	X
Regulatory Analysis	X	X	ML 022130308	X
Environmental Assessment	X	X	ML 022130316	X
Resolution of Public Comments	X	X	ML 022130320	X
Public Comments	X	X	ML 021480072	X

6. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104–113, requires that Federal agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or is otherwise impractical. In this final rule, the NRC is amending its regulations to incorporate by reference a later edition and addenda of Sections III and XI of the ASME BPV Code and the ASME OM Code, for construction, ISI, and IST of nuclear power plant components, as identified in the preceding Section 2, Public Comments on Proposed Rule; and Final Rule.

A number of commenters stated that the NRC approval of the ASME Code with exceptions (*i.e.*, modifications and limitations) does not meet the spirit of Pub. L. 104–113. The NRC disagrees because although Pub. L. 104–113 requires Federal agencies to use industry consensus standards to the extent practical, it does not require Federal agencies to endorse a standard in its entirety, nor does it forbid Federal agencies from endorsing industry consensus standards with limitations or modifications. The law does not prohibit an agency from generally adopting a voluntary consensus standard while taking exception to specific portions of the standard if those provisions are deemed to be "inconsistent with applicable law or otherwise impractical." Furthermore, taking specific exceptions furthers the Congressional intent of Federal reliance on voluntary consensus standards because it allows the adoption of substantial portions of consensus standards without the need to reject the standards in their entirety because of limited provisions which are not acceptable to the agency. Moreover, there is no legislative history suggesting that Congress intended agencies to take an "all or nothing" approach to endorsement of voluntary consensus standards under the Act, and the OMB guidance implementing Pub. L. 104–113 does not address the matter. The discussion in the statement of considerations of the limitations and modifications is sufficient to satisfy the requirements of Section 12(d)(3) of Pub. L. 104–113, and the relevant requirements of OMB Circular A–119 (1998). In light of these factors, the NRC concludes that the explanations for the modifications and limitations to the ASME BPV and OM Codes, as set forth

in the statement of considerations for this final rule, satisfy the requirements of Section 12(d)(3) of Pub. L. 104–113, and OMB Circular A–119.

7. Finding of No Significant Environmental Impact: Availability

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 is not a major Federal action significantly affecting the quality of the human environment, and, therefore, an environmental impact statement is not required.

This rulemaking will not significantly increase the probability or consequences of accidents; no changes are being made in the types of any effluents that may be released off-site; the environmental assessment for this rule demonstrates that there is a small decrease in occupational exposure; and there is no significant increase in public radiation exposure. Therefore, there are no significant radiological impacts associated with the action. The rulemaking does not involve non-radiological plant effluents and has no other environmental impact. Therefore, no significant non-radiological impacts are associated with the action.

The determination for this rule is that there will be no significant off-site impact to the public from this action. The NRC has prepared an environmental assessment on this final rule. The environmental assessment is available as indicated in Section 5, Availability of Documents, under the Supplementary Information heading.

The NRC requested the views of the States on the environmental assessment for the rule and did not receive any comments from the States.

8. Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*). These requirements were approved by the Office of Management and Budget, approval number 3150–0011.

Because the rule will reduce existing information collection requirements, the public burden for these information collections is expected to be decreased by 14 hours per licensee. This reduction includes the time required for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the information collection. Send comments on any aspect of these information collections, including suggestions for further reducing the burden, to the Records Management Branch (T–6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by Internet electronic mail to *infocollects@nrc.gov*; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB–10202 (3150–0011), Office of Management and Budget, Washington, DC 20503.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number

9. Regulatory Analysis

The NRC has prepared a regulatory analysis on this final rule. The analysis examines the costs and benefits of the action considered by the Commission. The regulatory analysis is available as indicated in Section 5, Availability of Documents, under the Supplementary Information heading.

One commenter stated that the regulatory analysis for the proposed amendment failed to address the values and impacts associated with a number of the modifications and limitations in the proposed rule. The NRC notes that the purpose of the regulatory analysis is to identify any significant values and impact associated with updating from the 1995 Edition with the 1996 Addenda to the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of the ASME Code. Therefore, modifications and limitations that require licensees to use the existing Code provisions in the 1995 Edition with the 1996 Addenda of the ASME Code are not addressed in the regulatory analysis.

10. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this rule will not have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and 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 size standards established by the NRC (10 CFR 2.810).

11. Backfit Analysis

The NRC's Backfit Rule in 10 CFR 50.109 states that the Commission shall require the backfitting of a facility only when it finds the action to be justified under specific standards stated in the rule. Section 50.109(a)(1) defines backfitting as the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position after issuance of the construction permit or the operating license or the design approval.

Section 50.55a requires nuclear power plant licensees to construct ASME Boiler and Pressure Vessel Code (BPV Code) Class 1, 2, and 3 components in accordance with the rules provided in Section III, Division 1, of the ASME BPV Code; inspect Class 1, 2, 3, Class MC, and Class CC components in accordance with the rules provided in Section XI, Division 1, of the ASME BPV Code; and test Class 1, 2, and 3 pumps and valves in accordance with the rules provided in the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code). This final rule incorporates by reference the 1997 Addenda, 1998 Edition, 1999 Addenda, and 2000 Addenda of Section III, Division 1, of the ASME BPV Code; Section XI, Division 1, of the ASME BPV Code; and the ASME OM Code.

Incorporation by reference of later editions and addenda of Section III, Division 1, of the ASME BPV Code is prospective in nature. The later editions and addenda do not affect a plant that has received a construction permit or an operating license or a design that has been approved, because the edition and addenda to be used in constructing a plant are, by rule, determined on the basis of the date of the construction permit, and are not changed thereafter, except voluntarily by the licensee. Thus, incorporation by reference of a later edition and addenda of Section III, Division 1, does not constitute a "backfitting" as defined in § 50.109(a)(1).

Incorporation by reference of later editions and addenda of Section XI, Division 1, of the ASME BPV Code and the ASME OM Code affect the ISI and IST programs of operating reactors. However, the Backfit Rule generally does not apply to incorporation by reference of later editions and addenda of the ASME BPV (Section XI) and OM Codes for the following reasons—

(1) The NRC's longstanding policy has been to incorporate later versions of the

ASME Codes into its regulations. This is codified in § 50.55a which requires licensees to revise their ISI and IST programs every 120 months to the latest edition and addenda of Section XI of the ASME BPV Code and the ASME OM Code incorporated by reference into § 50.55a that is in effect 12 months prior to the start of a new 120-month ISI and IST interval. Thus, when the NRC endorses a later version of the Code, it is implementing this longstanding policy and requirement.

(2) ASME BPV and OM Codes are national consensus standards developed by participants with broad and varied interests, in which all interested parties (including the NRC and utilities) participate. This consideration is consistent with both the intent and spirit of the Backfit Rule (i.e., the NRC provides for the protection of the public health and safety, and does not unilaterally imposed undue burden on applicants or licensees).

Other circumstances where the NRC does not apply the Backfit Rule to the endorsement of a later Code are as follows—

(1) When the NRC takes exception to a later ASME BPV or OM Code provision, but merely retains the current existing requirement, prohibits the use of the use of the later Code provision, or limits the use of the later Code provision, the Backfit Rule does not apply because the NRC is not imposing new requirements. However, the NRC explains any such exceptions to the Code in the Statement of Considerations for the rule. Sections 50.55a(b)(2)(viii)(F), (b)(2)(ix)(F), (b)(2)(ix)(G), (b)(2)(ix)(H), (b)(2)(xviii)(A), (B) and (C), (b)(2)(xix), (b)(2)(xxi)(A) and (C) in this final rule either retain current existing requirements, prohibit the use of the later Code provision, or limit the use of the later Code provision.

(2) When an NRC exception relaxes an existing ASME BPV or OM Code provision but does not prohibit a licensee from using the existing Code provision. Section 50.55a(b)(3)(vi) in this final rule relaxes the use of an existing Code provision but does not prohibit a licensee from using the existing Code provision.

There are some circumstances where the NRC considers it appropriate to treat as a backfit the endorsement of a later ASME BPV or OM Code—

(1) When the NRC endorses a later provision of the ASME BPV or OM Code that takes a substantially different direction from the currently existing requirements, the action is treated as a backfit. An example was the NRC's initial endorsement of Subsections IWE and IWL of Section XI, which imposed containment inspection requirements on operating reactors for the first time. The final rule dated August 8, 1996 (61 FR 41303), incorporated by reference in § 50.55a the 1992 Edition with the 1992 Addenda of IWE and IWL of Section XI to require that containments be routinely inspected to detect defects that could compromise a containment's structural integrity. This action expanded the scope of § 50.55a to include components that were not considered by the existing regulations to be within the scope of ISI. Since those requirements involved a substantially different direction, they were treated as backfits, and justified in accordance with the standards of 10 CFR 50.109. There are no provisions similar to this in the final rule.

(2) When the NRC requires implementation of later ASME BPV or OM Code provision on an expedited basis, the action is treated as a backfit. This applies when implementation is required sooner than it would be required if the NRC simply endorsed the Code without any expedited language. An example was the final rule dated September 22, 1999 (64 FR 51370), which incorporated by reference the 1989 Addenda through the 1996 Addenda of Section III and Section XI of the ASME BPV Code, and the 1995 Edition with the 1996 Addenda of the ASME OM Code. The final rule expedited the implementation of the 1995 Edition with the 1996 Addenda of Appendix VIII of Section XI of the ASME BPV Code for qualification of personnel and procedures for performing UT examinations. The expedited implementation of Appendix VIII was considered a backfit because licensees were required to implement the new requirements in Appendix VIII prior to the next 120-month ISI program inspection interval update. Another example was the final rule dated August 6, 1992 (57 FR 34666), which incorporated by reference in § 50.55a the 1986 Addenda through the 1989 Edition of Section III and Section XI of the ASME BPV Code. The final rule added a requirement to expedite the implementation of the revised reactor vessel shell weld examinations in the 1989 Edition of Section XI. Imposing these examinations was considered a backfit because licensees were required to implement the examinations prior to the next 120-month ISI program inspection interval update. There are no provisions similar to this in the final rule.

(3) When the NRC takes an exception to a ASME BPV or OM Code provision and imposes a requirement that is substantially different from the current existing requirement as well as substantially different than the later Code.

In §§ 50.55a(b)(2)(xv)(A), (A)(*1*) and (A)(*2*) that are discussed in the preceding Section 2, Final Rule and Comments on Proposed Rule, the NRC is adopting dissimilar metal piping weld ultrasonic (UT) examination coverage requirements. The NRC concludes that the addition of dissimilar metal piping weld UT examination coverage requirements to the regulation is necessary to correct the omission by the ASME BPV Code to ensure adequate protection of public health and safety. This backfit falls into the "adequate protection" exception under 10 CFR 50.109(a)(4)(ii), and the documented evaluation required by 10 CFR 50.109(a)(6) is below. Therefore, a backfit analysis under 10 CFR 50.109(a)(3) is not required.

Documented Evaluation

Dissimilar metal piping weld examination coverage requirements, although contained in the 1989 Edition, and earlier editions and addenda of Appendix III of Section XI of the ASME BPV Code, are not addressed in later editions and addenda of Section XI. Appendix VIII was added in the 1989 Addenda of Section XI, and the UT examination criteria for piping welds in Appendix VIII supercede the examination criteria for piping welds in Appendix III. Although Appendix VIII addresses qualification of personnel, procedures, and equipment used to conduct UT examinations of dissimilar metal piping welds, Appendix VIII (unlike Appendix III) does not define UT examination coverage criteria for dissimilar metal piping welds. Therefore, the addition of dissimilar metal piping weld examination coverage requirements to the regulation is necessary to correct the omission by the ASME BPV Code.

The purpose of ISI is to monitor for degrading conditions and ensure that any flaws which develop during service can be detected, sized, and evaluated, and that components with unacceptable flaws are repaired or replaced to adequately maintain the integrity of the pressure boundary. Another purpose of ISI is to identify any possible generic-type defects that were unforeseen during the design stage so that corrective actions can be taken prior to a breach of the pressure boundary. Although plants have generally been designed with sufficient margin so that important components will not crack or undergo excessive degradation, uncertainties in the definition of

service-induced loads and operating environments may have led to a less than optimum choice of materials, and may have permitted degradation mechanisms to progress more rapidly, or allowed different mechanisms to be active during plant operation, than were foreseen in the design.

Section XI defines inspection criteria for ISI and indicates allowable flaw sizes (with margin) based on fracture mechanics for various locations within reactor coolant pressure boundary (RCPB) components. If a flaw is found that exceeds the allowable size, (1) the component must be repaired, or (2) a safety analysis must be conducted, using fracture mechanics, to show that the flaw will not grow to an extent that could impair the integrity of the component. To conduct reliable and credible safety evaluations using fracture mechanics, information from the UT examination is required regarding the flaw size, shape, orientation, and location within the component. Consequently, examination information is key to detecting flaws and assessing the continued reliability and safety of flawed RCPB components.

Dissimilar metal welds are used to connect RCPB components. Operating history shows serious degradation of RCPB dissimilar metal welds have occurred at several nuclear power plants in the United States and at one foreign nuclear power plant. The NRC believes that additional occurrences are possible. Therefore, comprehensive and technically sound UT examination coverage criteria for dissimilar metal piping welds are needed to ensure that each facility provides adequate protection to the health and safety of the public. Sections 50.55a(b)(2)(xv)(A), (A)(*1*) and (A)(*2*) impose requirements that define comprehensive and technically sound UT examination coverage criteria for dissimilar metal piping welds that ensure uniform examination results among all licensees. These UT examination coverage requirements are necessary to detect flaws in dissimilar metal welds in RCPB components, thereby maintaining an extremely low probability of abnormal leakage or rapidly propagating failure, and gross rupture.

The remaining portion of this section addresses public comments related to backfitting or backfit issues on the proposed rule.

A number of commenters raised a generic concern with regard to the NRC's position on imposing exceptions (*i.e.*, modification or limitation) to consensus standards that are incorporated by reference in the Code of Federal Regulations. The commenters believe that, contrary to the NRC's determination, imposing any modification or limitation to the ASME Code constitutes a backfit for which a backfit analysis is required. Commenters stated that NRC is required to demonstrate that modifications and limitations result in an increase in quality or safety.

The NRC has reviewed the comments and has concluded that the commenters do not raise concerns which would alter the previous conclusion that the Backfit Rule does not require a backfit analysis of the modifications and limitations imposed by the NRC in the final rule. Furthermore, many of the modifications and limitations imposed during previous routine updates of § 50.55a have established a precedence for determining which modifications or limitations are backfits or require a backfit analysis (final rules dated August 6, 1992 (57 FR 34666), August 8, 1996 (61 FR 41303), and September 22, 1999 (64 FR 51370)). The NRC finds that the application of the backfit requirements to modifications and limitations in the current rule are consistent with the application of backfit requirements to modifications and limitations in previous rules. Since the modification and limitations in the current rule are not considered backfits or do not require backfit analyses, the NRC is not required to demonstrate that the new modifications and limitations result in an increase in quality or safety.

Section 50.55a(b)(2)(ix)(F) of the proposed rule would require that personnel who conduct visual examinations of containment surfaces be qualified in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2300 in place of the "owner-defined" qualification provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda IWE–2330(a). One commenter stated that the NRC is imposing additional qualification requirements for personnel that conduct general visual examinations in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE that were not imposed on general visual examinations conducted in accordance with earlier editions and addenda of IWE.

The NRC agrees with the commenter. The NRC proposed additional qualification requirements for personnel that conduct general visual examinations. Editions and addenda of IWE earlier than the 1998 Edition required the use of the VT–1 visual inspection method, the VT–3 visual inspection method, and a general visual inspection. The provisions in IWA–2300 were used to define the qualification requirements for personnel that conduct VT–1 and VT–3 visual examinations; however, detailed qualification requirements were not provided in the ASME Code for personnel that conduct general visual examinations. There are significant changes in the visual examination requirements in the 1998 Edition of IWE. Paragraph IWE–2330(a) requires that the licensee define the qualification requirements for personnel that conduct all visual examinations of containment surfaces, and a number of visual examinations are recategorized as general visual examinations that were formerly categorized as VT–1 or VT–3 in earlier editions and addenda of IWE. The intent of § 50.55a(b)(2)(ix)(F) in the proposed rule was not to allow licensees to use "owner-defined" qualification requirements to qualify personnel that conduct examinations that were formerly categorized as VT–1 or VT–3. However, the NRC inadvertently worded the modification such that additional qualification requirements would also be imposed on personnel that conduct general visual examinations. Therefore, the qualification requirements for personnel that conduct visual inspections of containment surfaces are revised in the final rule to require that personnel who conduct VT–1 and VT–3 visual examinations of containment surfaces be qualified in accordance with the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2300.

Section 50.55a(b)(2)(ix)(G) in the proposed rule would require that the general visual examinations required by the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–2310(b) and IWE–2310(c) meet VT–3 examination method provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWA–2210 in place of the "owner-defined" general and detailed visual examination provisions in the 1998 Edition, 1999 Addenda, and 2000 Addenda of IWE–2310(a).

One commenter stated that it is inappropriate for the NRC to impose § 50.55a(b)(2)(ix)(G) without performing a backfit analysis because the modification increases the frequency of VT–3 examinations of containment surfaces beyond that which was previously required in the editions and addenda of IWE earlier than the 1998 Edition. The commenter is correct. It was not the intent of the NRC to increase the frequency of VT–3 visual examinations of containment surfaces. The NRC inadvertently worded the modification such that the frequency of VT–3 examinations of containment areas was increased. Therefore, § 50.55a(b)(2)(ix)(G) is revised in the

final rule to require that VT-3 visual examinations for certain containment areas be performed once during each 10-year inspection interval which is consistent with the provisions in the editions and addenda of IWE earlier than the 1998 Edition.

Sections 50.55a(b)(2)(ix)(J), (b)(2)(xx), and (b)(2)(xxi)(B) in the proposed rule involve provisions in Section XI that were deleted in the 1995 Addenda that the NRC is reinstating in the final rule (§ 50.55a(b)(2)(ix)(J) of the proposed rule is renumbered as § 50.55a(b)(2)(ix)(I) in the final rule). Section 50.55a(b)(2)(xxiii) of the proposed rule involves underwater welding provisions in Section XI that were added in the 1996 Addenda that the NRC is prohibiting the use of in the final rule (§ 50.55a(b)(2)(xxiii) of the proposed rule is renumbered as § 50.55a(b)(2)(xii) in the final rule).

Several commenters stated that it is inappropriate for the NRC to reinstate or prohibit the use of these Code provisions because the elimination or addition of these Code provisions was previously accepted by the NRC the final rule dated September 22, 1999 (64 FR 51370). The NRC disagrees. These modifications were not included in the final rule that incorporated by reference the 1995 Addenda and 1996 Addenda of Section XI in 10 CFR 50.55a (64 FR 51370) due to an oversight by the NRC. The NRC did not identify that these Code provisions were eliminated or added when it reviewed the 1995 Addenda and 1996 Addenda of Section XI. The NRC has determined that these modifications should only apply to those licensees who implement the 1997 Addenda and later editions and addenda of Section XI, and should not be backfit to those licensees who update their ISI programs to the 1995 Edition with the 1996 Addenda in accordance with 10 CFR 50.55a(g)(4)(ii). The NRC has determined it is acceptable not to backfit the licensees who update their ISI programs to the 1995 Edition with the 1996 Addenda, because those licensees will be required at the next 10-year interval to update their ISI programs to include or prohibit the relevant Code provisions. Thus, any problems would be caught during the next 10-year interval. The reinstatement or prohibition of the relevant Code provisions are not considered backfits, because they are imposed only as part of the routine updating required as part of the 120-month updating, and do not constitute a significant change to, or fundamental modification of the existing ISI program.

Section 50.55a(b)(3)(vi) in the proposed rule would prohibit the extension of the exercise interval for manual valves from 3 months to 5 years when using the 1999 Addenda and 2000 Addenda of ISTC-3540. One commenter stated that the NRC should delete § 50.55a(b)(3)(vi) or conduct a backfitting analysis justifying the imposition of the proposed modification.

The NRC disagrees that a backfit analysis is required for § 50.55a(b)(3)(vi). The intent of the ASME consensus process was to extend the exercise interval for manual valves, and in this case, the NRC is accommodating the ASME consensus process to the extent that the NRC believes the extended exercise interval can be justified (*i.e.*, 2 years). In this case the NRC is allowing a relaxation from the current requirements, but not as much of a relaxation as the later Code would allow. Licensees are free to continue to implement the existing requirement (e.g., testing every three months).

The proposed rule would add a new § 50.55a(g)(6)(ii)(B)(*1*) to clarify the start date of the first 120-month interval for the ISI of Class MC and Class CC components. One commenter noted that since licensees have already established the start date of the first 120-month interval for the ISI of Class MC and Class CC components, it is a backfit for the NRC to now impose a different start date than that already established by licensees. The NRC agrees with this comment. Therefore, § 50.55a(g)(6)(ii)(B)(*1*) in the proposed rule is not adopted.

12. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

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. 552 and 553, the NRC is adopting 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, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 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 as amended by Pub L. 102-486, sec. 2902, 106 Stat. 3123 (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 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:

(a) removing paragraphs (b)(2)(xv)(G)(*4*), (g)(6)(ii)(B)(*1*) through (g)(6)(ii)(B)(*4*);

(b) redesignating and revising paragraph (g)(6)(ii)(B)(*5*) as (g)(6)(ii)(B);

(c) revising the introductory text of paragraph (b)(1), paragraphs (b)(1)(ii), (b)(1)(iii) and (b)(1)(v), the introductory text of paragraph (b)(2), paragraph (b)(2)(vi), the introductory text of paragraphs (b)(2)(viii) and (b)(2)(ix), paragraphs (b)(2)(xi), and (b)(2)(xiv), the introductory text of paragraph (b)(2)(xv), paragraphs (b)(2)(xv)(A), (b)(2)(xv)(K)(*1*)(*i*) and (b)(2)(xvii), the introductory text of paragraph (b)(3), paragraph (b)(3)(ii), the introductory text of paragraphs (b)(3)(iii) and (b)(3)(iv), and paragraphs (b)(3)(v) and (g)(6)(ii)(C)(*1*); and

(d) adding paragraphs (b)(2)(viii)(F), (b)(2)(ix)(F) through (b)(2)(ix)(I), (b)(2)(xii), (b)(2)(xv)(M), (b)(2)(xviii) through (b)(2)(xxi), (b)(3)(iv)(D), (b)(3)(vi), and (g)(6)(ii)(C)(2).

The amended text is set forth to read as follows:

§ 50.55a Codes and standards.

* * * * *

(b) * * *

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, and include the 1963 Edition through

1973 Winter Addenda, and the 1974 Edition (Division 1) through the 2000 Addenda (Division 1), subject to the following limitations and modifications:

* * * * *

(ii) *Weld leg dimensions.* When applying the 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section, licensees may not apply paragraph NB–3683.4(c)(1), Footnote 11 to Figure NC–3673.2(b)–1, and Figure ND–3673.2(b)–1.

(iii) *Seismic design.* Licensees may use Articles NB–3200, NB–3600, NC–3600, and ND–3600 up to and including the 1993 Addenda, subject to the limitation specified in paragraph (b)(1)(ii) of this section. Licensees may not use these Articles in the 1994 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

* * * * *

(v) *Independence of inspection.* Licensees may not apply NCA–4134.10(a) of Section III, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, and include the 1970 Edition through the 1976 Winter Addenda, and the 1977 Edition (Division 1) through the 2000 Addenda (Division 1), subject to the following limitations and modifications:

* * * * *

(vi) *Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI.* Licensees may use either the 1992 Edition with the 1992 Addenda or the 1995 Edition with the 1996 Addenda of Subsection IWE and Subsection IWL as modified and supplemented by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section when implementing the initial 120-month inspection interval for the containment inservice inspection requirements of this section. Successive 120-month interval updates must be implemented in accordance with paragraph (g)(4)(ii) of this section.

* * * * *

(viii) *Examination of concrete containments.* Licensees applying Subsection IWL, 1992 Edition with the 1992 Addenda, shall apply paragraphs (b)(2)(viii)(A) through (b)(2)(viii)(E) of this section. Licensees applying the 1995 Edition with the 1996 Addenda shall apply paragraphs (b)(2)(viii)(A), (b)(2)(viii)(D)(*3*), and (b)(2)(viii)(E) of this section. Licensees applying the 1998 Edition with the 1999 and 2000 Addenda shall apply paragraphs (b)(2)(viii)(E) and (b)(2)(viii)(F) of this section.

* * * * *

(F) Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA–2300. The "owner-defined" personnel qualification provisions in IWL–2310(d) are not approved for use.

(ix) *Examination of metal containments and the liners of concrete containments.* Licensees applying Subsection IWE, 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda, shall satisfy the requirements of paragraphs (b)(2)(ix)(A) through (b)(2)(ix)(E) of this section. Licensees applying the 1998 Edition with the 1999 Addenda and 2000 Addenda shall satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I) of this section.

* * * * *

(F) VT–1 and VT–3 examinations must be conducted in accordance with IWA–2200. Personnel conducting examinations in accordance with the VT–1 or VT–3 examination method shall be qualified in accordance with IWA–2300. The "owner-defined" personnel qualification provisions in IWE–2330(a) for personnel that conduct VT–1 and VT–3 examinations are not approved for use.

(G) The VT–3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE–2500–1, and the VT–1 examination method must be used to conduct the examination in Item E4.11 of Table IWE–2500–1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval. The "owner-defined" visual examination provisions in IWE–2310(a) are not approved for use for VT–1 and VT–3 examinations.

(H) Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE–2500–1 must be examined using the VT–3 examination method. Flaws or degradation identified during the performance of a VT–3 examination must be examined in accordance with the VT–1 examination method. The criteria in the material specification or IWB–3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT–3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT–3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

(I) The ultrasonic examination acceptance standard specified in IWE–3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

* * * * *

(xi) *Class 1 piping.* Licensees may not apply IWB–1220, "Components Exempt from Examination," of Section XI, 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, and shall apply IWB–1220, 1989 Edition.

(xii) *Underwater Welding.* The provisions in IWA–4660, "Underwater Welding," of Section XI, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use on irradiated material.

* * * * *

(xiv) *Appendix VIII personnel qualification.* All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. Licensees applying the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section may use the annual practice requirements in VII–4240 of Supplement VII of Section XI in place of the 8 hours of annual hands-on training provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Appendix VIII specimen set and qualification requirements.* The following provisions may be used to modify implementation of Appendix VIII of Section XI, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. Licensees choosing to apply these provisions shall apply all of the following provisions under paragraph (b)(2)(xv) except for those in paragraph (b)(2)(xv)(F) which are optional.

(A) When applying Supplements 2, 3, and 10 to Appendix VIII, the following examination coverage criteria requirements must be used:

(*1*) Piping must be examined in two axial directions, and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available. Dissimilar metal welds must be examined axially and circumferentially.

(*2*) Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds or dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld and may be used to perform examinations from either side of the weld.

* * * * *

(K) * * *

(*1*) * * *

(*i*) For detection, a minimum of four flaws in one or more full-scale nozzle mock-ups must be added to the test set. The specimens must comply with Supplement 6, paragraph 1.1, to Appendix VIII, except for flaw locations specified in Table VIII S6–1. Flaws may be either notches, fabrication flaws or cracks. Seventy-five (75) percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in paragraph (b)(2)(xv)(K)(*4*) of this section, Table VIII–S7–1—Modified, with the exception that flaws in the outer eighty-five (85) percent of the weld need not be perpendicular to the weld. There may be no more than two flaws from each category, and at least one subsurface flaw must be included.

* * * * *

(M) When implementing Supplement 12 to Appendix VIII, only the provisions related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are to be applied.

* * * * *

(xvii) *Reconciliation of Quality Requirements.* When purchasing replacement items, in addition to the reconciliation provisions of IWA–4200, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, the replacement items must be purchased, to the extent necessary, in accordance with the licensee's quality assurance program description required by 10 CFR 50.34(b)(6)(ii).

(xviii) *Certification of NDE personnel.* (A) Level I and II nondestructive examination personnel shall be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA–2314, and IWA–2314(a) and IWA–2314(b) of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(B) Paragraph IWA–2316 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may only be used to qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with IWA–5211(a) and (b), 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(C) When qualifying visual examination personnel for VT–3 visual examinations under paragraph IWA–2317 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, the proficiency of the training must be demonstrated by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

(xix) *Substitution of alternative methods.* The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA–2240 must be applied. The provisions in IWA–2240, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use. The provisions in IWA–4520(c), 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, allowing the substitution of alternative examination methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code are not approved for use.

(xx) *System leakage tests.* When performing system leakage tests in accordance IWA–5213(a), 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, a 10-minute hold time after attaining test pressure is required for Class 2 and Class 3 components that are not in use during normal operating conditions, and no hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

(xxi) *Table IWB–2500–1 examination requirements.* (A) The provisions of Table IWB–2500–1, Examination Category B–D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) in the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB–3512–1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may be performed in place of an ultrasonic examination.

(B) The provisions of Table IWB–2500–1, Examination Category B–G–2, Item B7.80, that are in the 1995 Edition are applicable only to reused bolting when using the 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(C) The provisions of Table IWB–2500–1, Examination Category B–K, Item B10.10, of the 1995 Addenda must be applied when using the 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(3) As used in this section, references to the OM Code refer to the ASME Code for Operation and Maintenance of Nuclear Power Plants, and include the 1995 Edition through the 2000 Addenda subject to the following limitations and modifications:

* * * * *

(ii) *Motor-Operated Valve testing.* Licensees shall comply with the provisions for testing motor-operated valves in OM Code ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC–3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, and shall establish a program to ensure that motor-operated valves continue to be capable of performing their design basis safety functions.

(iii) *Code Case OMN–1.* As an alternative to paragraph (b)(3)(ii) of this section, licensees may use Code Case OMN–1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants," Revision 0, in

conjunction with ISTC 4.3, 1995 Edition with the 1996 and 1997 Addenda, or ISTC-3600, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section. Licensees choosing to apply the Code Case shall apply all of its provisions.

* * * * *

(iv) *Appendix II.* Licensees applying Appendix II, "Check Valve Condition Monitoring Program," of the OM Code, 1995 Edition with the 1996 and 1997 Addenda, shall satisfy the requirements of paragraphs (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(C) of this section. Licensees applying Appendix II, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, shall satisfy the requirements of paragraphs (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(D) of this section.

* * * * *

(D) The provisions of ISTC-3510, ISTC-3520, and ISTC-3540 in addition to ISTC-5221 must be implemented if the Appendix II condition monitoring program is discontinued.

(v) *Subsection ISTD.* Article IWF-5000, "Inservice Inspection Requirements for Snubbers," of the ASME BPV Code, Section XI, provides inservice inspection requirements for examinations and tests of snubbers at nuclear power plants. Licensees may use Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants," ASME OM Code, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, in place of the requirements for snubbers in Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to their technical specifications or licensee-controlled documents. Preservice and inservice examinations must be performed using the VT-3 visual examination method described in IWA-2213.

(vi) *Exercise interval for manual valves.* Manual valves must be exercised on a 2-year interval rather that the 5-year interval specified in paragraph ISTC-3540 of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, provided that adverse conditions do not require more frequent testing.

* * * * *

(g) * * *

(6) * * *

(ii) * * *

(B) Licensees do not have to submit to the NRC staff for approval of their containment inservice inspection programs which were developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified modifications and limitations. The program elements and the required documentation must be maintained on site for audit.

(C) * * *

(*1*) Appendix VIII and the supplements to Appendix VIII to Section XI, Division 1, 1995 Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code must be implemented in accordance with the following schedule: Appendix VIII and Supplements 1, 2, 3, and 8—May 22, 2000; Supplements 4 and 6—November 22, 2000; Supplement 11—November 22, 2001; and Supplements 5, 7, and 10—November 22, 2002.

(*2*) Licensees implementing the 1989 Edition and earlier editions and addenda of IWA-2232 of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code must implement the 1995 Edition with the 1996 Addenda of Appendix VIII and the supplements to Appendix VIII of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code.

* * * * *

Dated at Rockville, Maryland this 9th day of September 2002.

For the U.S. Nuclear Regulatory Commission.

William D. Travers,

Executive Director For Operations.

[FR Doc. 02-23811 Filed 9-25-02; 8:45 am]

BILLING CODE 7590-01-P

DEPARTMENT OF THE TREASURY

Office of Thrift Supervision

12 CFR Parts 560, 590, and 591

[No. 2002-43]

RIN 1550-AB51

Alternative Mortgage Transaction Parity Act; Preemption

AGENCY: Office of Thrift Supervision, Treasury.

ACTION: Final rule.

SUMMARY: The Alternative Mortgage Transaction Parity Act (AMTPA) authorizes state chartered housing creditors to make, purchase, and enforce alternative mortgage transactions without regard to any state constitution, law, or regulation. To rely on AMTPA, certain state chartered housing creditors must comply with regulations on alternative mortgage transactions issued by the Office of Thrift Supervision (OTS). In today's rulemaking, OTS revises its rules identifying the OTS regulations that apply under AMTPA. OTS will no longer identify its regulations on prepayments and late charges for state chartered housing creditors.

OTS is also revising its limits on the amount of late charges that may be assessed on loans secured by first liens on residential manufactured homes under part 590, which addresses the preemption of state usury laws. In addition, OTS is making a minor technical change to the definition of reverse mortgage in part 591, which addresses the preemption of state due-on-sale laws.

EFFECTIVE DATE: January 1, 2003.

FOR FURTHER INFORMATION CONTACT: Theresa Stark, Senior Project Manager, Compliance Policy, (202) 906-7054; Karen Osterloh, Special Counsel, Regulations and Legislation Division, (202) 906-6639, Office of Thrift Supervision, 1700 G Street, NW., Washington, DC 20552.

SUPPLEMENTARY INFORMATION:

I. Alternative Mortgage Transaction Parity Act Regulations (§ 560.220)

The Alternative Mortgage Transaction Parity Act (AMTPA) [1] permits state chartered housing creditors [2] to make, purchase, and enforce alternative mortgage transactions if the creditors comply with regulations governing such transactions issued by federal regulators. AMTPA applies to loans with any alternative payment features that vary from conventional fixed-rate, fixed-term mortgage loans, such as variable rates, balloon payments, or call features. It allows state chartered housing creditors to engage in alternative mortgage transactions notwithstanding "any State constitution, law, or regulation," provided the transactions are made in conformity with regulations issued by one of three federal regulators.[3] Housing creditors, other than state chartered commercial banks and state chartered

[1] 12 U.S.C. 3801 *et seq.*

[2] A "housing creditor" is a depository institution, a lender approved by the Secretary of Housing and Urban Development for participation in certain mortgage insurance programs, "any person who regularly makes loans, credit sales or advances secured by interests in properties referred to in [AMTPA]; or * * * any transferee of any of them." To qualify as a state housing creditor and take advantage of preemption, AMTPA specifically provides that the creditor must be "licensed under applicable State law and [remain or become] subject to the applicable regulatory requirements and enforcement mechanisms provided by State law." 12 U.S.C. 3802(2).

[3] 12 U.S.C. 3803(c).

10 CFR 50.55a

67FR64033

10/17/2002 Corrects
*****67FR60520

Corrected IWE & IWL
Intervals

Rules and Regulations

Federal Register

Vol. 67, No. 201

Thursday, October 17, 2002

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AG61

Industry Codes and Standards; Amended Requirements: Correction

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule; correction.

SUMMARY: On September 26, 2002 (67 FR 60520), the U.S. Nuclear Regulatory Commission (NRC) published a final rule amending its regulations to incorporate by reference a later edition and addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) to provide updated rules for construction, inservice inspection (ISI), and inservice testing (IST) of components in light-water cooled nuclear power plants. This action corrects two erroneous references to the NRC's regulations made in the supplementary information accompanying the final rule.

EFFECTIVE DATE: October 28, 2002.

FOR FURTHER INFORMATION CONTACT: Stephen Tingen, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Alternatively, you may contact Mr. Tingen at (301) 415–1280, or via e-mail at: *sgt@nrc.gov.*

SUPPLEMENTARY INFORMATION: In the final rule, published on September 26, 2002 (67 FR 60520), on page 60521, in the third column, in the third full paragraph, the first and second sentences are corrected to read as follows:

In responding to this clarification, several commenters indicated that the 10-year IWE and 5-year IWL examination intervals must coincide with the 120-month interval update in § 50.55a(g)(4)(ii). The NRC does not agree that the 10-year IWE and 5-year IWL examination intervals must coincide with the 120-month interval update in § 50.55a(g)(4)(ii).

Dated at Rockville, Maryland, this 9th day of October, 2002.

For the Nuclear Regulatory Commission.

Michael T. Lesar,

Federal Register Liaison Officer.

[FR Doc. 02–26342 Filed 10–16–02; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 170

RIN 3150–AH03

Cost Recovery for Contested Hearings Involving U.S. Government National Security Initiatives

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to allow the agency to recover its costs associated with contested hearings on licensing actions involving U.S. Government national security initiatives through licensing fees assessed to the affected applicant or licensee. This final rule is a special exception to the Commission's longstanding policy of not charging this type of fee for contested hearings. In this case, the Commission will charge its contested hearing costs directly to the involved licensee or applicant rather than recovering its costs through the annual fees assessed to all licensees within the affected class.

EFFECTIVE DATE: November 18, 2002.

ADDRESSES: The comments received are available electronically at the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html.* From this site, the public can gain entry into the NRC's Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. For more information, contact the NRC Public Document Room (PDR) Reference staff at 1–800–397–4209, or 301–415–4737, or by email to *pdr@nrc.gov.* If you do not have access to ADAMS, or if there are problems in accessing the documents located in ADAMS, please contact the PDR.

Comments received may also be viewed via the NRC's interactive rulemaking website (*http://ruleforum.llnl.gov*). This site provides the ability to upload comments as files (any format), if your web browser supports that function. For information about the interactive rulemaking site, contact Ms. Carol Gallagher, 301–415–5905; e-mail *CAG@nrc.gov.*

FOR FURTHER INFORMATION CONTACT: Robert Carlson, telephone 301–415–8165, Office of the Chief Financial Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

SUPPLEMENTARY INFORMATION:

I. Background

The NRC has a longstanding policy of charging the affected applicant part 170 licensing fees to recover the agency's costs for any uncontested hearings that the NRC holds on applications to construct a power reactor or enrichment facility. These hearings are mandated by statute. However, the NRC's costs for all contested hearings[1] have been recovered through part 171 annual fees assessed to the members of the particular class of licensee to which the applicant belongs.

The NRC published the final rule establishing the part 170 and part 171 fees for FY 2002 on June 24, 2002, (67 FR 42612) after considering a comment

[1] A contested proceeding is defined in 10 CFR 2.4 as (1) a proceeding in which there is a controversy between the staff of the Commission and the applicant for a license concerning the issuance of the license or any of the terms or conditions thereof or (2) a proceeding in which a petition for leave to intervene in opposition to an application for a license has been granted or is pending before the Commission.

10 CFR 50.55a

68FR40469

8/7/2003

Added Reg. Guides For Code Cases

Rules and Regulations

Federal Register
Vol. 68, No. 130
Tuesday, July 8, 2003

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AG86

Incorporation by Reference of ASME BPV and OM Code Cases

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to incorporate by reference NRC Regulatory Guides listing Code cases published by the American Society of Mechanical Engineers (ASME) which the NRC has reviewed and found to be acceptable for use. These Code cases provide alternatives to requirements in the ASME Boiler and Pressure Vessel Code (BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) pertaining to construction, inservice inspection and inservice testing of nuclear power plant components. This action incorporates by reference three regulatory guides that address NRC review and approval of ASME-published Code cases. Therefore, the Code cases listed in these regulatory guides are incorporated by reference into the NRC's regulations.

EFFECTIVE DATE: August 7, 2003. The incorporation by reference of certain publications listed in the regulation is approved by the Director of the Office of the Federal Register as of August 7, 2003.

FOR FURTHER INFORMATION CONTACT: Harry S. Tovmassian, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–3092, e-mail *hst@nrc.gov*.

SUPPLEMENTARY INFORMATION:

Background

New editions of the ASME BPV and OM Codes are issued every three years and addenda to the editions are issued annually. It has been the Commission's policy to update 10 CFR 50.55a to incorporate the ASME Code editions and addenda by reference. Section 50.55a was last amended on September 26, 2002 (67 FR 60520), to incorporate by reference the 1998 Edition of these Codes, up to and including the 2000 Addenda. The ASME also publishes Code cases for Section III and Section XI quarterly and Code cases for the OM Code yearly. Code cases are alternatives to the requirements of the ASME BPV Code and the OM Code. In the past the NRC staff's practice was to review these Code cases and find them either acceptable, conditionally acceptable, or unacceptable for use by NRC licensees. These Code cases were then listed in periodically revised regulatory guides (RGs), together with information on their acceptability. Footnote 6 to § 50.55a referred to those RGs listing Code cases determined by the staff to be "suitable for use." However, the publication dates and version numbers of the RGs were not specified in Footnote 6 and these RGs had not been approved by the Director of the *Office of the Federal Register* (OFR) for incorporation by reference into the Code of Federal Regulations.

Discussion

The NRC identified a concern with the practice of generally referencing the RGs addressing ASME Code cases in Footnote 6 to § 50.55a. The notice and comment provisions of the Administrative Procedure Act (APA) (5 U.S.C. 551 *et seq.*), as amended, were arguably not satisfied by this practice. To address this matter, on March 19, 2002 (67 FR 12488), the NRC published a proposed amendment to revise § 50.55a to incorporate by reference the regulatory guides which list the ASME Code cases approved or conditionally approved by the NRC.

This final rule amends 10 CFR 50.55a(b) to incorporate by reference the RGs listing ASME BPV and OM Code cases which are approved for use by NRC licensees (including their revision numbers) into Title 10 of the Code of Federal Regulations. The text of existing Footnote 6 to § 50.55a is deleted and all references to Footnote 6 in § 50.55a have been removed and replaced by language that specifies, where appropriate, that the optional ASME Code cases that are incorporated by reference in § 50.55a(b) may be applied in lieu of the corresponding requirements of the ASME Codes. Sections 50.55a(b)(4), (b)(5), and (b)(6), which specify limitations upon the implementation of approved ASME Code cases, have been added.

Over the past several years, NRC licensees have expressed their dissatisfaction about the length of time it takes the NRC to review and approve Code cases. To improve the efficiency of the process for endorsement of ASME Code cases, the NRC plans to proceed as follows for future updates. First, the NRC will review Code cases and revise the RGs periodically to indicate Code cases approved for use by NRC licensees. The NRC will issue the draft RGs for comment before issuance of the final RGs. At approximately the time each set of final guides is issued, the NRC will also issue the next set of proposed guides. Second, the NRC will conduct rulemakings to incorporate by reference the revised RGs into § 50.55a. The NRC will complete each rulemaking within a short time of the issuance of the applicable final RGs. Where these rulemakings do not involve any significant questions of policy, they will be issued in accordance with the rulemaking authority delegated to the NRC's Executive Director for Operations under NRC Management Directives 6.3 and 9.17. To expedite the issuance of subsequent rules, the NRC will conduct these rulemakings without preparing a rulemaking plan. If the rulemakings are not controversial and significant adverse comment is not expected, the NRC will incorporate by reference future revisions of the RGs through the issuance of direct final rules. These actions should expedite the NRC process for reviewing and approving ASME Code cases.

Resolution of Public Comments

In response to the publication of the proposed rule, the NRC received eight letters commenting on various aspects of the rulemaking. The letters came from utilities, law firms representing utilities, and the Nuclear Energy Institute (NEI). NEI sent a second letter to supplement its first letter. The following sections address the various issues raised by the public commenters.

1. Comment

On December 28, 2001 (66 FR 67335), the NRC published a notice of the availability of proposed revisions to RGs 1.84 and 1.147 and a new proposed RG (temporarily designated DG–1089 but subsequently given a permanent designation of RG 1.192) and solicited public comments. One rule commenter that had responded to the December 28, 2001, notice requests that the NRC consider the comments he submitted on the proposed regulatory guides as part of this rulemaking.

Response

The NRC has considered the public comments received in response to its December 28, 2001, notice and has resolved those comments by modifying the guides, as appropriate, or providing its rationale for not doing so. The public comments received and the NRC resolution of the comments on the guides is available to the public as indicated in the "Availability of Documents" section of these statements of consideration. The NRC finds no reason to further consider those comments as part of this rulemaking.

2. Comment

One commenter believes that as an alternative to this rulemaking, the NRC should consider developing a Web site (1) where the individual Code cases could be posted for public comment and for subsequent NRC acceptance (identifying any limitations on or exceptions to the use of the Code cases), or (2) where revisions to the RGs could be posted for comment each time the NRC proposes to endorse a Code case. Either method would allow individual Code cases to be reviewed by the NRC, posted for public comment, and accepted for use by licensees within 3-to-6 months of the ASME publication of the Code case, as compared to the 3-to-5 years between past revisions of the RGs.

Response

The commenter's suggestion does not appear to be in compliance with the notice and comment provisions of the APA. The APA requires notice of proposed rulemaking to be published in the Federal Register. As discussed earlier, the Code cases listed in the RGs to be incorporated by reference provide alternatives to compliance with the ASME Codes, which are rules by virtue of their incorporation by reference into § 50.55a. Accordingly, it is the NRC's view that any generally applicable alternatives to the endorsed ASME Codes must be considered requirements, and are therefore subject to the notice and comment requirements of the APA.

3. Comment

Some commenters state that the NRC's regulations already allow "generic" approval of Code cases as alternatives to the requirements in § 50.55a. In accordance with § 50.55a(a)(3), alternatives "may be used when authorized by the Director of the Office of Nuclear Reactor Regulation" if the alternatives provide an acceptable level of quality and safety. Several commenters believe that the NRC's acceptance on a generic basis could be authorized in a generic communication, such as a regulatory issue summary. These commenters recommend that if the NRC determines that the current provisions do not allow a generic approval in this manner, the NRC should provide a generic approval process similar to § 50.55a(a)(3) that would not require continued rulemaking for endorsement of Code cases.

Response

The NRC does not agree that the provisions in § 50.55a(a)(3) provide for generic approval of Code cases. This paragraph allows licensees, on a license-specific basis, to request NRC's review and approval of alternatives to the requirements in the ASME Codes. The purpose of § 50.55a(a)(3) is to provide a mechanism for individual licensees to request approval to implement measures (including Code cases) not generically approved by the NRC in order to meet specific licensee needs. The NRC does not believe that it may through rulemaking adopt a procedure for "generically" approving alternatives to ASME Code provisions which are incorporated by reference in § 50.55a if the procedure does not meet the requirements for rulemaking or an "order" under the APA.

4. Comment

While acknowledging that the recommendation is beyond the scope of this rulemaking, one commenter suggests that the NRC explore whether § 50.55a should be revised to no longer reference the editions and addenda of the ASME Code. The editions and addenda of the Code and the Code cases could be put into RGs which provide a means by which the revised regulation could be met. The commenter believes that both the editions and addenda of the Code sections and Code cases could be approved more efficiently in this manner.

Response

The NRC previously considered this approach but rejected it because of the difficulty, the length of time, and substantial resources that would be necessary to develop a rule that sets forth general requirements for inservice inspection (ISI), inservice testing (IST), and the construction of nuclear power plant components. The NRC agrees that if the rule were revised to no longer incorporate the ASME Codes by reference, then this rulemaking to approve Code cases would not be necessary. However, as a practical matter, to avoid imposing a backfit, the rule would likely have to include a grandfather provision that would allow licensees to use current ASME Code requirements already incorporated by reference into § 50.55a. Thus, the NRC would still be faced with the task of conducting rulemakings to approve Code cases for the grandfathered licensees.

5. Comment

Several commenters urge the NRC to expedite the process for reviewing and approving ASME Code cases. One commenter believes that the proposed rule is inconsistent with the NRC strategic goal of improving the efficiency of the regulatory process. Alternative approaches for streamlining the process should be explored.

Response

The NRC agrees that the process of approving ASME BPV and OM Code cases through incorporation by reference into the regulations is cumbersome and that a more efficient process would better satisfy the NRC's goal of streamlining the regulatory process. As mentioned in the Discussion section of these statements of consideration, the NRC is planning several actions that it believes will improve the timeliness of the incorporation by reference process. Other actions to improve the efficiency of the Code case approval process are discussed in the Resolution of Public Comments on Guides document published in conjunction with the publication of the RGs in question (*see* "Availability of Documents" section). However, any streamlining of the process must comply with applicable law.

6. Comment

One commenter recommends that, if the NRC believes it must use the rulemaking process to incorporate by reference its Code case approvals, maximum use should be made of the direct final rule process to enable

licensees to implement Code cases sooner.

Response

The NRC agrees with this comment and is considering the feasibility of taking this approach with future rulemakings of this type. Direct final rules are published together with companion proposed rules containing the identical regulations. If there is no significant adverse public comment on the direct final rule during the comment period, the rule becomes a final rule within a specified number of days after publication. If one or more significant adverse comment is received, the direct final rule is withdrawn and the proposed rule is treated as though no direct final rule had been published. There is no further opportunity for public comment. However, the NRC cautions that the RGs in question may control the timeliness in this matter. Unless a method to streamline the RG publication process is developed, efficiencies arrived at by using direct final rules may be minor.

7. Comment

Several commenters object to the wording in proposed § 50.55a(i)(2)(ii), as well as the parallel wording of §§ 50.55a (i)(3)(ii) and (i)(4)(ii). The proposed language would require that users of a Code case implement newly approved versions of the Code case along with any modifications or limitations. The commenters argue that this is inconsistent with the existing requirements in §§ 50.55a (f)(4)(ii) and (g)(4)(ii), which permit licensees to defer implementation of new ASME Code criteria.

Response

The NRC agrees that the proposed rule language would require licensees who have implemented a Code case to implement additional modifications and limitations if the Code case is revised in the future. In general, this is contrary to NRC's intention. The NRC intends that once an approved Code case is implemented by an applicant or licensee, it may continue to apply the Code case until it updates its Code of Record for the component being constructed or until the end of the licensee's current 120-month ISI or IST update interval, as applicable. Accordingly, the proposed rule language has been modified in the final rule §§ 50.55a (b)(4)(ii), (b)(5)(ii), and (b)(6)(ii) (corresponding to §§ 50.55a (i)(2)(ii), (i)(3)(ii), and (i)(4)(ii) in the proposed rule) to clarify the NRC's intention in this regard. An exception to this would be when the NRC's initial approval of the Code case by a specific licensee is conditioned by including language that requires the licensee to apply any limitations or conditions specified in a revised RG that approves that Code case. Accordingly, the final rule states that the licensee may apply the previous version of the Code case "as authorized," which refers to the NRC's condition in the initial approval of the Code case for use by a specific licensee.

8. Comment

One commenter states that proposed § 50.55a(i)(2)(iv) is not "conducive to" use with repair/replacement activities under Section XI of the ASME Code and the Section XI Code cases. Replacement items are procured over time and many different editions and addenda of Section III may be referenced for different items. Therefore, the phrase in the proposed rule "* * * until the licensee updates its Section III Code of Record" could be interpreted as referring to a singular event rather than an action that occurs many times. Adding the phrase "for the item being constructed" would clarify that a licensee can use an annulled Code case until it procures the specific item to an updated Section III.

The commenter is also concerned about situations in which the licensee implements a Code case to a certain edition of the Code, but later updates his Code of Record to a later edition of the Code. In some instances the updated Code of Record will not have the Code case approved because it has been incorporated into the Code. The commenter recommends the following wording to resolve both concerns: "A licensee that has initiated implementation of a Code case that is subsequently annulled by the ASME may continue to apply that Code case until the licensee updates its Section III Code of Record for the item being constructed to an edition or addenda of Section III that has incorporated the case."

Response

The NRC agrees with these comments and has amended § 50.55a(b)(4)(iii) in the final rule to read as the commenter suggests with some further clarifications, as follows: "Application of an annulled Code case is prohibited unless an applicant or licensee applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.84. If an applicant or licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.84, the applicant or licensee may continue to apply the Code case until it updates its Code of Record for the component being constructed."

9. Comment

A commenter requests that the NRC retain Footnote 6 of § 50.55a and amend it to reference a new RG which is temporarily designated as DG–1112. Although this RG, which has been designated NRC Regulatory Guide 1.193, lists Code cases that the NRC has reviewed and not approved, the commenter believes that it would be useful to licensees because they could still implement the Code cases through the provisions of § 50.55a(a)(3), if the NRC's concerns are adequately resolved.

Response

The NRC does not believe that it is appropriate to reference RGs that list disapproved ASME Code cases. The fact the NRC has not incorporated a Code case by reference simply means that the Code case has not received generic NRC approval, and therefore may not be applied without prior NRC review and approval. Referencing RGs which list disapproved Code cases may give the appearance that the NRC has generically disapproved the Code cases in question, which is incorrect. As the commenter points out, disapproved Code cases may be proposed through the relief request process permitted by § 50.55a(a)(3). Also, the NRC does not believe that the lack of a reference to Regulatory Guide 1.193 presents a hardship to licensees. Licensees are generally aware of its existence and availability and may make use of it as they see fit. Thus, the final rule does not reference this RG.

10. Comment

Several commenters recommend that the incorporation by reference of the RGs listing the NRC-approved ASME BPV and OM Code cases be placed in § 50.55a(b), instead of in a new § 50.55a(i) as in the proposed rule, because of the similarity of the requirements.

Response

During the preparation of the proposed rule, the staff considered several options for integrating the incorporation by reference of the RGs with the remaining requirements in § 50.55a and sought public comment on this question. The staff agrees with the commenters that incorporation by reference of the RGs listing NRC-approved Code cases should be co-located with the incorporation by reference of the various ASME BPV and OM Code editions and addenda. Thus, this final rule expands § 50.55a(b) to include the incorporation by reference

of the RGs and adds paragraphs (b)(4), (b)(5), and (b)(6) to specify the implementation requirements.

11. Comment

Sections 50.55a(i)(2)(iv), 50.55a(i)(3)(iv), and 50.55a(i)(4)(iv) of the proposed rule state that licensees could no longer apply an NRC-approved annulled Code case if the NRC later determines the Code case is unacceptable for use and revises § 50.55a or the applicable regulatory guide (1.84, 1.147, or 1.192) to prohibit continued application of the annulled Code case. Several commenters state that revising § 50.55a or the applicable regulatory guide (1.84, 1.147, or 1.192) to prohibit continued application of the NRC-approved annulled Code case for the remainder of the interval is a backfit.

Response

The NRC agrees that any revision to § 50.55a prohibiting the continued application of an annulled Code case for the remainder of an interval would be a backfit that must be justified in accordance with § 50.109. In order to avoid confusion, the requirement in the proposed rule prohibiting the continued application of an annulled Code case previously approved by the NRC is deleted in the final rule. However, if in the future, an NRC-approved Code case is annulled, allowed to expire, or revised because the Code case is no longer adequate, the NRC will consider amending § 50.55a and the applicable regulatory guide to prohibit continued application of the Code case. The NRC will justify the revision to § 50.55a in accordance with the requirements in § 50.109.

12. Comment

Several commenters recommend that the phrase, "or the optional ASME Code cases listed in the RGs incorporated by reference in paragraph (i) of this section" be added in six other paragraphs in § 50.55a where reference is made to the use of ASME BPV or OM Code provisions.

Response

The phrase in question occurred in various locations of § 50.55a in the proposed rule where the regulations in the current rule had referred the reader to Footnote 6 (which references the RGs listing approved Code cases). The NRC agrees with the commenter that the reference to the use of the optional ASME BPV and OM Code cases should also be included in the specified paragraphs. The NRC has modified §§ 50.55a(f)(3)(iii)(B), (f)(3)(iv)(B), (f)(4)(i), (f)(4)(ii), (g)(4)(i), and (g)(4)(ii), accordingly.

13. Comment

One commenter states that the incorporation by reference of the ASME Code cases in § 50.55a is unnecessary because the ASME issues Code cases as alternative rules applicable for a 3-year period, after which the Code cases are incorporated into the ASME Code, annulled, or renewed, and because § 50.55a has provisions for endorsement of future editions and addenda of the ASME Code. The commenter also believes the process is inefficient and unlawful because it introduces new regulatory positions without satisfying the requirements of the Backfit Rule, 10 CFR 50.109.

Response

The Commission agrees that once the provisions of a Code case are incorporated into an edition or addenda of the ASME BPV or OM Code, and those editions and addenda of the Codes are incorporated by reference, there is no need for incorporation by reference of those alternative requirements. However, from the time that the Code case is published by the ASME to the time it is listed in an incorporated edition or addenda of the Codes, there is no legal mechanism for the NRC to approve its use other than through the provisions of § 50.55a(a)(3) for requesting approval of alternatives. This requires a case-by-case review and approval, which is time consuming and wasteful of agency resources. Therefore, the Commission has determined that rulemaking approving the use of alternatives to the required ASME Code provisions specified in § 50.55a is the most efficient course of action that complies with applicable law.

This rulemaking contains no requirements that satisfy the definition of a backfit as specified in 10 CFR 50.109(a)(1). The initial application of a Code case is voluntary on the part of the licensee. Absent approval of the NRC, either on a license-specific basis or through a generic rulemaking, a licensee is not legally authorized to use an ASME Code case. Hence, any limitations on the use of Code cases are not backfits as defined in § 50.109(a)(1).

14. Comment

Several commenters believe that the NRC is acting contrary to the intent of Congress in passing the National Technology Transfer and Advancement Act of 1995, (Pub. L. 104–113), which was implemented through Office Management and Budget (OMB) Circular A–119 and NRC Management Directive 6.5, "NRC Participation on Development and Use of Consensus Standards." These commenters believe the NRC has not identified regulations that are in direct conflict with the published Code case or documented a regulatory basis for imposing limitations or modifications.

Response

The NRC does not agree with the commenters' opinion that the NRC has not fully complied with the letter and intent of Public Law 104–113 and the associated guidance. Public Law 104–113, requires that Federal agencies use technical standards that are developed by voluntary consensus standards bodies unless the use of these standards is inconsistent with applicable law or is otherwise impractical. The statute does not require Federal agencies to endorse a standard in its entirety, nor does it forbid Federal agencies to endorse industry consensus standards with limitations or modifications, if the agencies deem the provisions of the standards to be inconsistent with applicable law or otherwise impractical. Endorsing a voluntary consensus standard with limitations, modifications, or exceptions furthers the congressional intent of Federal reliance on voluntary consensus standards by allowing the adoption of substantial portions of consensus standards. Agencies need not reject the standards in their entirety because a few provisions are not acceptable. Moreover, there is no legislative history suggesting that Congress intended agencies to take an "all or nothing" approach to the endorsement of voluntary consensus standards under the Act, and the OMB guidance implementing Public Law 104–113 does not address the matter. The discussions of the limitations and modifications in the RGs and the document on the Resolution of Public Comments on the RGs are sufficient to satisfy the requirements of section 12(d)(3) of Public Law 104–113, and the relevant requirements of OMB Circular A–119 (1998).

15. Comment

According to one commenter the proposed rulemaking is unlawful because it is not in compliance with the Backfit Rule, 10 CFR 50.109. (The commenter provided no explanation of why the proposed rule is in conflict with the Backfit Rule.)

Response

Section 50.109(a)(2) requires that the NRC perform a backfit analysis for any backfits, as defined in § 50.109(a)(1), that it seeks to impose, unless the

backfits fall into one or more of the delineated exceptions. A backfit is a modification of or addition to systems, components, or design of a facility, or the design approval or manufacturing license, or procedures or organization required to design, construct or operate a facility, any of which may result from a new or amended provision in the Commission's rules or the imposition of a staff regulatory position interpreting the Commission rules that is either new or different from previously applicable staff positions. As discussed in the responses to Comments 11 and 13, the Commission finds that this final rule does not contain any requirements which satisfy the definition of a backfit, and consequently, a backfit analysis is not required.

16. Comment

One commenter states that when Code cases are interpretive of the regulations (or provide an alternative means for achieving compliance with a requirement), they need not be incorporated by reference and that licensees should be permitted to use them with no further NRC action.

Response

The NRC agrees that Code cases that are purely interpretations of the regulations incorporated by reference in § 50.55a(b) need not be incorporated by reference. However, the Code cases incorporated by reference in § 50.55a, with or without modifications or limitations, constitute alternatives to the requirements in § 50.55a and not interpretations. Therefore, the NRC believes that incorporating the RGs by reference is the proper treatment of these alternative requirements.

Paragraph-by-Paragraph Discussion

On December 28, 2001 (66 FR 67335), the NRC published proposed revisions to RGs 1.84 and 1.147 and a new proposed RG [temporarily designated DG–1089]. The NRC has considered the public comments on these RGs and has resolved those comments by modifying the guides, as appropriate, or providing its rationale for not doing so. Previously, RG 1.84, Revision 31, listed only Section III Code cases related to design and fabrication, and RG 1.85, Revision 31, listed Section III Code cases related to materials and testing. Revision 32 to RG 1.84 lists for the first time in one guide all Section III Code cases that have been approved for use by the NRC. The staff intents to withdraw RG 1.85 when the ensuing revisions to the RGs are published. This rulemaking incorporates by reference Regulatory Guide 1.84, Revision 32, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," Regulatory Guide 1.147, through Revision 13, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code."

1. Paragraph 50.55a(b)

In the proposed rule (March 19, 2002: 67 FR 12488), the language of incorporation by reference of the RGs and the implementation requirements were contained in a new postulated paragraph (i). The NRC requested public comment on the proposed placement of these requirements. As discussed in Comment 10 and the corresponding NRC response, the Commission has decided to place the incorporation by reference of the RGs listing NRC-approved ASME BPV and OM Code case in § 50.55a(b) and the corresponding implementation requirements in §§ 50.55a(b)(4), (b)(5), and (b)(6). In this manner, the incorporation by reference of the RGs listing NRC-approved Code cases would be located with the incorporation by reference of the editions and addenda of the ASME BPV and OM Codes and be more organizationally consistent. Thus § 50.55a(b) has been expanded and now contains the language of incorporation by reference of the RGs listing NRC-approved ASME BPV and OM Code cases and identifies each RG by title and revision number.

Section 50.55a(b) now specifies the applicable RGs for incorporation by reference in addition to the editions and addenda of the ASME Boiler and Pressure Vessel Code and Code for Operation and Maintenance of Nuclear Power Plants. This paragraph incorporates by reference NRC Regulatory Guide 1.84, Revision 32, NRC Regulatory Guide 1.147, through Revision 13, and NRC Regulatory Guide 1.192. This final rule incorporates all of the revisions of Regulatory Guide 1.147 because some licensees continue to apply Code cases listed as approved in earlier revisions to this RG and if these revisions were not incorporated by reference the further use of these Code cases would be prohibited. Similarly, Revision 14 of Regulatory Guide 1.147 will be incorporated by reference in the same fashion because it has already been prepared in draft form and major reformatting of that document would result in a substantial delay in issuing the final version. However, the NRC will format Revision 15 of Regulatory Guide 1.147 so that it provides the current status of all Section XI Code cases and at that time the incorporation by reference of previous revisions of that RG will be superceded.

The RGs incorporated by reference in this final rule list Code cases applicable to Section III of the ASME BPV Code, Section XI of the ASME BPV Code, and the ASME OM Code, respectively, that have been approved unconditionally, or with conditions and limitations specified by the NRC, as alternatives to specific Code provisions. NRC approval of the use of Code cases listed in these RGs is granted only if the limitations and conditions, if any, are applied.

Sections 50.55a(b)(4), (b)(5), and (b)(6) require that licensees or applicants initially applying a Code case which is listed in one of the RGs as acceptable apply the most recent version of the Code case listed in the RG. If a licensee or applicant is applying a particular version of an approved Section III Code case, and a later version is incorporated into the applicable RG as acceptable, the licensee or applicant may continue to apply the earlier version of the Code case until it updates its Code of Record for the component being constructed. A licensee may continue to apply the earlier version of a Section XI or OM Code case until the end of the licensee's current 120-month ISI or IST update interval, including any adjustments to the interval permitted under Paragraphs IWA–2430(c)(1) and (e) of Section XI of the ASME BPV Code or Paragraphs ISTA 2.2.3(d) and (e) of the OM Code.

Sections 50.55a (b)(4), (b)(5), and (b)(6) also specify that a licensee is permitted to apply an annulled or expired Code case provided that it has been applied before it has been listed as expired or annulled in RG 1.84, 1.147, or 1.192. A licensee implementing an approved Section III Code case that is subsequently listed as annulled or expired in RG 1.84 may continue to apply that Code case until it updates its Code of Record for the component being constructed. A licensee implementing an approved Section XI or OM Code case that is subsequently listed as annulled or expired in RG 1.147 or RG 1.192 may continue to apply that Code case until it updates its ISI or IST program to an edition or addenda of the Code that has incorporated the Code case. In most circumstances, a Code case is annulled or allowed to expire by the ASME because the Code case is included in a later edition or addenda of the ASME BPV or OM Codes. When a licensee updates its construction, ISI or IST Code of Record, the provisions of the Code can then be applied instead of the annulled or expired Code case. In any event, a licensee may continue to use the annulled or expired Code case

until the end of its 120-month ISI/IST interval or until it updates its construction Code of Record, unless the NRC specifically prohibits its continued use by modifying the RG or 10 CFR 50.55a and performing a backfit analysis in accordance with the provisions in 10 CFR 50.109.

In the proposed rule, §§ 50.55a(i)(2)(iv), (i)(3)(iv), and (i)(4)(iv) contained language implying that § 50.55a or the RGs could specifically prohibit the use of annulled Code cases. As noted in the Response to Comment 11, this language is unnecessary and has been removed in the final rule.

2. Paragraphs 50.55a(c)(3), (d)(2), and (e)(2)

Current references to Footnote 6 in §§ 50.55a(c)(3), (d)(2), and (e)(2) have been removed, and text has been added to indicate that the optional ASME Code cases referred to are those listed in the RGs that are incorporated by reference in § 50.55a(b).

3. Paragraphs 50.55a(f)(2), (f)(3)(iii)(A), (f)(3)(iv)(A), (g)(2), (g)(3)(i), and (g)(3)(ii)

Currently, §§ 50.55a(f)(2), (f)(3)(iii)(A), (f)(3)(iv)(A), (g)(2), (g)(3)(i), and (g)(3)(ii) do not specifically mention ASME Code cases but have a reference to Footnote 6. These references to Footnote 6 have been removed and text has been added to indicate that the optional ASME Code cases referred to are those listed in the RGs that are incorporated by reference in § 50.55a(b).

4. Paragraphs 50.55a(f)(3)(iii)(B), (f)(3)(iv)(B), (f)(4)(i), (f)(4)(ii), (g)(4)(i), and (g)(4)(ii)

Sections 50.55a(f)(3)(iii)(B), (f)(3)(iv)(B), (f)(4)(i), (f)(4)(ii), (g)(4)(i), and (g)(4)(ii) have been amended to indicate that the ASME Code cases listed in the RGs that are incorporated by reference in § 50.55a may be applied in lieu of corresponding ASME BPV or OM Code requirements.

5. Footnote 6, 10 CFR 50.55a

Footnote 6 has been removed from § 50.55a and the footnote number has been reserved. Footnote 6 to § 50.55a formerly stated that ASME Code cases suitable for use are listed in RGs 1.84, 1.85, and 1.147. These Code cases are now approved for use by specific language in §§ 50.55a(c)(3), (d)(2), (e)(2), (f)(2), (f)(3)(iii)(A), (f)(3)(iii)(B), (f)(3)(iv)(A), (f)(3)(iv)(B), (f)(4)(i), (f)(4)(ii), (g)(2), (g)(3)(i), (g)(3)(ii), (g)(4)(i), and (g)(4)(ii). Footnote 6 also stated that the use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to § 50.55a(a)(3). This text is being removed because it is unnecessary; licensees continue to have the option of requesting approval to use Code cases not incorporated by reference into § 50.55a under § 50.55a(a)(3).

Availability of Documents

The NRC is making the documents identified below available to interested persons through one or more of the following:

Public Document Room (PDR). The NRC Public Document Room is located at 11555 Rockville Pike, Public File Area O–1 F21, Rockville, Maryland.

Rulemaking Web site. The NRC's interactive rulemaking Web site is located at *http://ruleforum.llnl.gov*. The documents may be viewed and downloaded electronically via this Web site.

The NRC's Public Electronic Reading Room (PERR). The NRC's public electronic Reading Room is located at *http://www.nrc.gov/reading-rm.html*.

The NRC staff contact (NRC Staff). Single copies of the final rule, the regulatory analysis, the environmental assessment, and the regulatory guides may be obtained from Harry S. Tovmassian, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Alternatively, you may contact Mr. Tovmassian at (301) 415–3092 or via e-mail to: *hst@nrc.gov*.

Document	PDR	Web	PERR	NRC staff
Environmental Assessment	x	x	ML030690244	x
Regulatory Analysis	x	x	ML031490533	x
Regulatory Guide 1.192	x		ML030730430	x
Regulatory Guide 1.84, Revision 32	x		ML030730417	x
Regulatory Guide 1.147, Revisions 0 to 12	x		ML031560264	x
Regulatory Guide 1.147, Revision 13	x		ML030730423	x
Regulatory Guide 1.193	x		ML030730440	x
Resolution of Public Comments on Guides	x		ML030730448	x

Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Public Law 104–113, requires agencies to use technical standards developed or adopted by voluntary consensus standards bodies unless the use of such standards is inconsistent with applicable law or is otherwise impractical. The NRC is amending its regulations to incorporate by reference regulatory guides that list ASME BPV and OM Code cases which have been approved by the NRC. ASME Code cases, which are ASME-approved alternatives to the provisions of ASME Code editions and addenda, constitute national consensus standards, as defined in Public Law 104–113 and OMB Circular A–119. They are developed by bodies whose members (including the NRC and utilities) have broad and varied interests.

These statements of consideration provide the reasons for modifying or limiting the applicability of ASME Code cases otherwise approved for use by the NRC as alternatives to current ASME Code provisions incorporated by reference into § 50.55a. The treatment of ASME BPV and OM Code cases, and modifications and conditions placed on them, in this final rule does not conflict with any policy on agency use of consensus standards specified in OMB Circular A–119.

Finding of No Significant Environmental Impact: Availability

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 is not a major Federal action significantly affecting the quality of the human environment, and, therefore, an environmental impact statement is not required.

This rulemaking will not significantly increase the probability or consequences of accidents; no changes are being made in the types of effluents that may be released off site; and there is no significant increase in public radiation exposure. Therefore, there are no significant environmental impacts

associated with the action. Therefore, the NRC determines that there will be no significant off site impact to the public from this action.

The basis for NRC's finding is set forth in an environmental assessment on this final rule. The environmental assessment is available as indicated in the Availability of Documents section under the Supplementary Information heading. The NRC requested the views of the States on the environmental assessment for the rule and did not receive any comments from the States.

Paperwork Reduction Act Statement

This final rule decreases the burden on licensees for recordkeeping and reporting requirements related to examinations, tests, and repair and replacement activities during refueling outages and the recordkeeping requirements associated with welding procedures. The annual public burden reduction for this information collection is estimated to average 59 hours for each of 172 responses. Because the burden for this information collection is insignificant, OMB clearance is not required. The existing requirements were approved by OMB, approval number 3150-0011.

Public Protection Notification

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.

Regulatory Analysis

The ASME Code cases listed in the RGs provide voluntary alternatives to the provisions in the ASME BPV Code and OM Code for construction, ISI, and IST of specific structures, systems, and components used in nuclear power plants. Implementation of these Code cases is not required. Licensees use NRC-approved ASME Code cases to reduce regulatory burden or gain additional operational flexibility. It would be difficult for the NRC to provide these advantages independent of the ASME Code case publication process without a considerable additional resource expenditure by the agency. The NRC has prepared a regulatory analysis addressing the qualitative benefits of the alternatives considered in this rulemaking and comparing the costs associated with each alternative. The regulatory analysis is available for inspection in the NRC Public Document Room, located at One White Flint North, 11555 Rockville Pike, Rockville, Maryland, Room O-1 F21. Single copies of the analysis may be obtained from Harry S. Tovmassian, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, telephone (301) 415-3092, e-mail *hst@nrc.gov.*

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, (5 U.S.C. 605(b)), the Commission certifies that this final rule will not have a significant economic impact on a substantial number of small entities. This final rule affects only the licensing and 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 size standards established by the NRC (10 CFR 2.810).

Backfit Analysis

The provisions in this rulemaking permit, but do not require, licensees to apply Code cases that have been reviewed and approved by the NRC, sometimes with modifications or conditions. Therefore, the implementation of an approved Code case is voluntary and does not constitute a backfit. Thus the Commission finds that these amendments do not involve any provisions that constitute a backfit as defined in 10 CFR 50.109(a)(1), that the backfit rule does not apply to this final rule, and that a backfit analysis is not required.

Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

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 forth 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. 552 and 553, the NRC is adopting the following amendments to 10 CFR Part 50.

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

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 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, as amended by Pub. L. 102-486, sec. 2902, 106 Stat. 3123 (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 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—

■ a. Revising the introductory text of paragraph (b), the introductory text of paragraph (c)(3), paragraph (c)(3)(iv), the introductory text of paragraph (d)(2), paragraph (d)(2)(iii), the introductory text of paragraph (e)(2), paragraphs (e)(2)(iii), (f)(2), (f)(3)(iii)(A), (f)(3)(iii)(B), (f)(3)(iv)(A), (f)(3)(iv)(B), (f)(4)(i), (f)(4)(ii), (g)(2), (g)(3)(i), (g)(3)(ii), (g)(4)(i) and (g)(4)(ii);

■ b. Adding paragraphs (b)(4), (b)(5), and (b)(6); and

■ c. Removing the text of Footnote 6 and reserving the footnote number.

§ 50.55a Codes and standards.

* * * * *

(b) The ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants, which are referenced in paragraphs (b)(1), (b)(2), and (b)(3) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. NRC Regulatory Guide 1.84, Revision 32, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (June 2003); NRC Regulatory Guide 1.147 (Revision 0—

February 1981), including Revision 1 through Revision 13 (June 2003), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"; and Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code" (June 2003), have been approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. These regulatory guides list ASME Code cases which the NRC has approved in accordance with the requirements in paragraphs (b)(4), (b)(5), and (b)(6). Copies of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016. Also, copies of these Codes and NRC Regulatory Guides 1.84, Revision 32; 1.147, through Revision 13; and 1.192 are available for inspection and copying for a fee at the Office of the Federal Register, 800 N. Capitol Street, Suite 700, Washington, DC, as well as the NRC Technical Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852–2738. Single copies of Regulatory Guides may be obtained free of charge by writing the Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by fax to (301) 415–2289; or by email to *DISTRIBUTION@NRC.GOV.*

* * * * *

(4) *Design, Fabrication, and Materials Code Cases.* Licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in NRC Regulatory Guide 1.84, Revision 32, without prior NRC approval subject to the following:

(i) When an applicant or licensee initially applies a listed Code case, the applicant or licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If an applicant or licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the applicant or licensee may continue to apply the previous version of the Code case as authorized, or may apply the later version of the Code case, including any NRC-specified conditions placed on its use, until it updates its Code of Record for the component being constructed.

(iii) Application of an annulled Code case is prohibited unless an applicant or licensee applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.84. If an applicant or licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.84, the applicant or licensee may continue to apply the Code case until it updates its Code of Record for the component being constructed.

(5) *Inservice Inspection Code Cases.* Licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in Regulatory Guide 1.147 through Revision 13, without prior NRC approval subject to the following:

(i) When a licensee initially applies a listed Code case, the licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If a licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code case as authorized or may apply the later version of the Code case, including any NRC-specified conditions placed on its use.

(iii) Application of an annulled Code case is prohibited unless a licensee previously applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.147. Any Code case listed as annulled in any Revision of Regulatory Guide 1.147 which a licensee has applied prior to it being listed as annulled, may continue to be applied by that licensee to the end of the 120-month interval in which the Code case was implemented.

(6) *Operation and Maintenance of Nuclear Power Plants Code Cases.* Licensees may apply the ASME Operation and Maintenance Nuclear Power Plants Code cases listed in Regulatory Guide 1.192 without prior NRC approval subject to the following:

(i) When a licensee initially applies a listed Code case, the licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If a licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code case as authorized or may apply the later version of the Code case, including any NRC-specified conditions placed on its use.

(iii) Application of an annulled Code case is prohibited unless a licensee previously applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.192. If a licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.192, the licensee may continue to apply the Code case to the end of the current 120-month interval.

(c) * * *

(3) The Code edition, addenda, and optional ASME Code cases to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but—

* * * * *

(iv) The optional Code cases applied to a component must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

* * * * *

(d) * * *

(2) The Code edition, addenda, and optional ASME Code cases to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but—

* * * * *

(iii) The optional Code cases must be those listed in the NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(e) * * *

(2) The Code edition, addenda, and optional ASME Code cases to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA–1140, subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but—

* * * * *

(iii) The optional Code cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(f) * * *

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, pumps and valves which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice tests for operational readiness set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, or 1.192 that are

incorporated by reference in paragraph (b) of this section) in effect 6 months before the date of issuance of the construction permit. The pumps and valves may meet the inservice test requirements set forth in subsequent editions of this Code and addenda which are incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, or 1.192 that are incorporated by reference in paragraph (b) of this section), subject to the applicable limitations and modifications listed therein.

(3) * * *

(iii) * * *

(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases that are listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section) referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

(iv) * * *

(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 2 and 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code cases listed in the NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section) referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

* * * * *

(4) * * *

(i) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, or 1.192 that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

* * * * *

(g) * * *

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) in effect six months before the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of this Code which are incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section), subject to the applicable limitations and modifications.

(3) * * *

(i) Components (including supports) which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular component.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular component.

* * * * *

(4) * * *

(i) Inservice examinations of components and system pressure tests

conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

* * * * *

Dated at Rockville, Maryland, this 10th day of June, 2003. For the Nuclear Regulatory Commission.

William D. Travers,

Executive Director for Operations.

[FR Doc. 03–17027 Filed 7–7–03; 8:45 am]

BILLING CODE 7590–01–P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 25

[Docket No. NM246; Special Conditions No. 25–231–SC]

Special Conditions: Embraer Model 170–100 and 170–200 Airplanes; Sudden Engine Stoppage; Operation Without Normal Electrical Power; Interaction of Systems and Structures

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final special conditions; correction.

SUMMARY: This document corrects a typographical error that appeared in Final Special Conditions 25–231–SC, which were published in the Federal Register on April 23, 2003 (68 FR 19933). The typographical error resulted in inadvertent repetition of the following language:

In lieu of compliance with 14 CFR 25.1351(d), the following special conditions apply:

This language correctly appears in the section of the special conditions entitled Operation Without Normal Electrical Power. This same language incorrectly appears in the section entitled Interaction of Systems and Structure and should be stricken.

EFFECTIVE DATE: April 10, 2003.

FOR FURTHER INFORMATION CONTACT: Tom Groves, FAA, International Branch, ANM–116, Transport Airplane Directorate, Aircraft Certification Service, 1601 Lind Avenue SW., Renton, Washington 98055–4056; telephone (425) 227–1503; facsimile (425) 227–1149.

SUPPLEMENTARY INFORMATION: Final special conditions for Embraer Model 170–100 and 170–200 airplanes were published in the Federal Register on April 23, 2003 [68 FR 19933]. These special conditions pertained to sudden engine stoppage, operation without normal electrical power, and interaction of systems and structures.

As published, the final special conditions contained an inadvertent repetition of certain language on page 19935. After the section entitled Operation Without Normal Electrical Power, the language "In lieu of compliance with 14 CFR 25.13519(d), the following special conditions apply:" should remain. In the section entitled Interaction of Systems and Structure, that language should be stricken.

Since no other part of the final special conditions has been changed, the final special conditions are not being republished.

The effective date of the final special conditions remains April 10, 2003.

Issued in Renton, Washington on June 23, 2003.

Ali Bahrami,

Acting Manager, Transport Airplane Directorate, Aircraft Certification Service.

[FR Doc. 03–17112 Filed 7–7–03; 8:45 am]

BILLING CODE 4910–13–P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 2002–SW–25–AD; Amendment 39–13217; AD 2003–13–15]

RIN 2120–AA64

Airworthiness Directives; Schweizer Aircraft Corporation Model 269A, 269A–1, 269B, 269C, and TH–55A Helicopters

AGENCY: Federal Aviation Administration, DOT.

ACTION: Final rule.

SUMMARY: This amendment supersedes an existing airworthiness directive (AD), applicable to Schweizer Aircraft Corporation (Schweizer) Model 269A, 269A–1, 269B, 269C, and TH–55A helicopters, that currently requires inspecting the lugs on certain aft cluster fittings and each aluminum end fitting on certain tailboom struts. Modifying or replacing each strut assembly within a specified time period and serializing certain strut assemblies are also required by the existing AD. This amendment requires the same actions as the existing AD, and also requires a one-time inspection and repair, if necessary, of certain additional cluster fittings, and replacement and modification of certain cluster fittings within 150 hours time-in-service (TIS) or 6 months, whichever occurs first. This amendment is prompted by the need to expand the applicability to include certain Hughes-manufactured cluster fittings and to provide a terminating action for the repetitive dye-penetrant inspections of the cluster fittings. The actions specified by this AD are intended to prevent failure of a tailboom support strut or a cluster fitting, which could cause rotation of a tailboom into the main rotor blades, and subsequent loss of control of the helicopter.

DATES: Effective August 12, 2003.

The incorporation by reference of certain publications listed in the regulations is approved by the Director of the Federal Register as of August 12, 2003.

ADDRESSES: The service information referenced in this AD may be obtained from Schweizer Aircraft Corporation, P.O. Box 147, Elmira, New York 14902. This information may be examined at the FAA, Office of the Regional Counsel, Southwest Region, 2601 Meacham Blvd., Room 663, Fort Worth, Texas; or at the Office of the Federal Register, 800 North Capitol Street, NW., suite 700, Washington, DC.

10 CFR 50.55a

69FR58804

11/1/2004

Added 2003 Addenda
III, XI, OM

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AH24

Industry Codes and Standards; Amended Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to incorporate by reference the 2001 Edition and the 2002 and 2003 Addenda of Division 1 of Section III of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPV Code); the 2001 Edition and the 2002 and 2003 Addenda of Division 1 rules of Section XI of the ASME BPV Code; and the 2001 Edition and the 2002 and 2003 Addenda of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) to provide updated rules for constructing and inspecting components and testing pumps and valves in light-water cooled nuclear power plants. This final rule incorporates by reference the latest edition and addenda of the ASME BPV and OM Codes that have been approved for use by the NRC subject to certain limitations and modifications. The NRC is also withdrawing its approval of Subsection NH of the 1995 through 2000 Addenda of Section III of the ASME BPV Code.

DATES: Effective November 1, 2004. The incorporation by reference of certain publications in this rule is approved by the Director of the Office of the Federal Register as of November 1, 2004.

ADDRESSES: The NRC maintains an Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. The documents may be accessed through the NRC's Public Electronic Reading Room on the Internet at *http://www.nrc.gov/reading-rm/adams.html*. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC at 1–800–397–4209, (301) 415–4737, or by e-mail to *pdr@nrc.gov*. The availability of the Regulatory Analysis and the Environmental Assessment is further discussed in Section 5 of this rule.

FOR FURTHER INFORMATION CONTACT: Stephen Tingen, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Alternatively, you may contact Mr. Tingen at (301) 415–1280, or via e-mail at: *sgt@nrc.gov*.

SUPPLEMENTARY INFORMATION:

1. Background

On January 7, 2004 (69 FR 879), the NRC published a proposed rule to amend 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." The proposed rule presented revised requirements for construction, inservice inspection (ISI), and inservice testing (IST) of nuclear power plant components for public comment. For construction, the proposed rule would have permitted the use of Section III, Division 1, of the ASME BPV Code, 2001 Edition and the 2002 and 2003 Addenda for Class 1, Class 2, and Class 3 components with one new modification.

For ISI, the proposed rule would have permitted the use of Section XI, Division 1, of the ASME BPV Code, 2001 Edition and the 2002 and 2003 Addenda for Class 1, Class 2, Class 3, Class MC, and Class CC components with new modifications and limitations.

For IST, the proposed rule would have permitted the use of the ASME OM Code, 2001 Edition and the 2002 and 2003 Addenda for Class 1, Class 2, and Class 3 pumps and valves with no new modifications or limitations.

2.0 Public Comments Received on Proposed Rule; and Final Rule

Fifty-five comments on the proposed rule were received from utilities, service organizations, and individuals. In response to the public comments, the NRC has either removed or revised some modifications and limitations that were proposed. A summary of the public comments applicable to the proposed rule and their resolution are provided in the following sections.

The NRC has considered and resolved the public comments and incorporated changes into the final rule. The NRC is publishing the final rule in § 50.55a to incorporate by reference the 2001 Edition and the 2002 and 2003 Addenda of Division 1 rules of Section III of the ASME BPV Code; the 2001 Edition and the 2002 and 2003 Addenda of Division 1 rules of Section XI of the ASME BPV Code; and the 2001 Edition and the 2002 and 2003 Addenda of the ASME OM Code for construction, ISI, and IST of components in nuclear power plants. The 2001 Edition and the 2002 and 2003 Addenda of Sections III and XI of the ASME BPV Code are acceptable for use subject to limitations and modifications. The 2001 Edition and the 2002 and 2003 Addenda of the ASME OM Code is acceptable for use with no new limitations or modifications.

2.1 Section III

The proposed rule would have revised § 50.55a(b)(1) to incorporate by reference the 2001 Edition and the 2002 and 2003 Addenda of Division 1 of Section III of the ASME BPV Code subject to modifications and limitations. Accordingly, the existing modification and limitation for weld leg dimensions and independence of inspection in §§ 50.55a(b)(1)(ii) and 50.55a(b)(1)(v), respectively, would continue to apply when using the 2001 Edition through 2003 Addenda of Section III, Division 1, of the ASME BPV Code. The existing modification and limitation in §§ 50.55a(b)(1)(ii) and 50.55a(b)(1)(v) would continue to apply to the 2001 Edition through 2003 Addenda of Section III because the earlier Code provisions on which these regulations are based were not revised in the 2001 Edition through 2003 Addenda of Section III to address the underlying issues which led to the NRC to impose the modification and limitation. There were no public comments received on §§ 50.55a(b)(1) and 50.55a(b)(1)(v). Therefore, §§ 50.55a(b)(1) and 50.55a(b)(1)(v) are adopted without change in this final rule.

10 CFR 50.55a(b)(1)(ii)—Weld Leg Dimensions

One commenter stated that the footnote to circumferential fillet welded and socket welded joints in Figures NC–3673.2(b)–1 and ND–3673.2(b)–1 of Section III was renumbered in the Code. The NRC agrees. Footnote 11 to Figures NC–3673.2(b)–1 and ND–3673.2(b)–1 is referenced in the existing regulation in § 50.55a(b)(1)(ii). Footnote 11 to Figures NC–3673.2(b)–1 and ND–3673.2(b)–1 was renumbered as Footnote 7 in the 1997 Addenda. Footnote 7 was renumbered as Footnote 11 in the 2000 Addenda. Footnote 11 was renumbered as Footnote 13 in the 2002 Addenda. Although the footnote was renumbered

in the Code, the contents of the footnote have not been revised. In consideration of this public comment, the existing regulation in § 50.55a(b)(1)(ii) is revised in this final rule to reference the contents of the footnote instead of referencing the footnote number. The revised § 50.55a(b)(1)(ii) states that the footnote to circumferential fillet welded and socket welded joints in Figures NC–3673.2(b)–1 and ND–3673.2(b)–1 that permits a socket weld leg dimension to be less than 1.09 of the nominal wall thickness of the pipe is not approved for use when using the 1989 Addenda through 2003 Addenda of Section III. This revision does not change the requirements in a substantive manner.

10 CFR 50.55a(b)(1)(iii) and 10 CFR 50.55a(b)(1)(vi)—Seismic Design

The proposed rule would have revised the existing limitation for seismic design in § 50.55a(b)(1)(iii) to prohibit the use of Articles NB–3200, NB–3600, NC–3600, and ND–3600 when using the 1994 Addenda through 2000 Addenda of Section III. The proposed rule stated that the limitation in § 50.55a(b)(1)(iii) does not apply to the 2001 Edition through 2003 Addenda of Section III because the earlier Code provisions on which this regulation was based were revised in the 2001 Edition through 2003 Addenda of Section III to address a number of the underlying issues which led the NRC to impose the limitation on the ASME Code provisions. Section 50.55a(b)(1)(vi) in the proposed rule would have allowed use of these articles when using the 2001 Edition and 2002 and 2003 Addenda of Section III with certain limitations and modifications. However, in consideration of public comment, the revisions to § 50.55a(b)(1)(iii) and § 50.55a(b)(1)(vi) in the proposed rule are not adopted in this final rule.

Section 50.55a(b)(1)(vi) of the proposed rule would have permitted the use of the alternative method for evaluating reversing dynamic building filtered loads and seismic loads in the 2001 Edition and the 2002 and 2003 Addenda of Section III Division 1 of the ASME BPV Code subject to modifications and limitations. However, § 50.55a(b)(1)(vi)(A) of the proposed rule would have prohibited the use of the alternative method for evaluating reversing dynamic loads for piping subject to loads generated by reflected waves caused by flow transients in NB–3200, NB–3600, NC–3600, and ND–3600. In addition, § 50.55a(b)(1)(vi)(B) of the proposed rule would have prohibited the use of inelastic analyses for evaluating reversing dynamic loads in NB–3228.6. Also, § 50.55a(b)(1)(vi)(C) of the proposed rule would have provided an alternate Level B stress limit for reversing dynamic loads. Section 50.55a(b)(1)(vi)(D) of the proposed rule would have supplemented the requirements for the calculation of inertial moment. Section 50.55a(b)(1)(vi)(E) of the proposed rule would have prohibited the use of the B_2 'stress indices specified in ND–3655(b)(3) and would have required that the allowable B_2 'stress indices specified in NB–3656(b)(3) and NC–3655(b)(3) be used instead of the allowable B_2 'stress indices specified in ND–3655(b)(3). Section 50.55a(b)(1)(vi)(F) of the proposed rule would have allowed the use of an allowable stress limit of $6S_M$ in the evaluation of the range of resultant moment only when it could be demonstrated that the global piping system response to the anchor movement does not create significant inelastic strain concentrations when using the provisions in NB–3656(b)(4), NC–3655(b)(4), and ND–3655(b)(4). S_M is the design stress intensity limit for a material and is tabulated in Section II of the ASME Code. A demonstration that the anchor movement does not create significant inelastic strain concentrations would not have been required if an allowable stress limit of $3S_M$ were used instead of $6S_M$ in the evaluation of the range of resultant moment.

The NRC received a large number of public comments on the modifications and limitations in § 50.55a(b)(1)(vi). The public comments provided technical reasoning why the modifications and limitations in § 50.55a(b)(1)(vi) were unnecessary and recommended their deletion. For example, ASME submitted an 83 page position paper in response to the modifications and limitations in (b)(1)(vi) of the proposed rule. It should be noted that the NRC's concerns regarding the alternative method for evaluating reversing dynamic building filtered loads and seismic loads began with changes in the 1994 Addenda through 1996 Addenda and were discussed in an amendment to § 50.55a issued in September 1999 (64 FR 51370). The ASME formed a special working group to evaluate the NRC's concerns. Although the special working group resolved some the NRC's concerns, a few significant issues remain.

The ASME submittal also recommended that the NRC prohibit the use of the revised seismic design provisions in the 2001 Edition and the 2002 and 2003 Addenda of Section III at this time. The ASME stated that the NRC and ASME should resolve their technical differences over the modifications and limitations in § 50.55a(b)(1)(vi) before permitting the use of revised seismic design provisions in the 2001 Edition and 2002 and 2003 Addenda of Section III. The NRC agrees. This would allow the NRC to discuss the technical details including recent piping dynamic testing in a more comprehensive manner. In consideration of public comments, the revision to § 50.55a(b)(1)(iii) in the proposed rule and the modifications and limitations in § 50.55a(b)(1)(vi) in the proposed rule are not adopted in this final rule. The existing limitation for seismic design in § 50.55a(b)(1)(iii) is revised in this final rule to prohibit the use of Articles NB–3200, NB–3600, NC–3600, and ND–3600 when using the 1994 Addenda through 2003 Addenda of Section III.

10 CFR 50.55a(b)(1)(vii)—Subsection NH

Section 50.55a(b)(1)(vii) in the proposed rule would have prohibited the use of Subsection NH of the 2001 Edition through 2003 Addenda of Section III of the ASME BPV Code and would have withdrawn current approval of Subsection NH of the 1995 Addenda through 2000 Addenda of Section III of the ASME BPV Code. The scope of Subsection NH includes Class 1 components that function in water, steam, sodium, helium, or any other process fluid. The special design provisions in Subsection NH apply to Class 1 components that are required to function at elevated metal temperatures where creep and relaxation effects may be significant and for which the stress limits and design provisions in Subsection NB of Section III are not applicable. These stress limits and design provisions of Subsection NB are applicable only to service conditions where creep and relaxation effects do not exist. The proposed rule stated that the elevated temperature provisions in Subsection NH, applicable to certain Class 1 components in future advanced reactor designs such as liquid metal and high-temperature gas-cooled reactor designs, have not been reviewed by the NRC for technical adequacy because the design provisions in Subsection NH were thought not to be applicable to any currently operating nuclear power plant nor to any currently approved standard advanced light-water reactor plant design.

A commenter stated that prohibiting the use of Subsection NH because the NRC has not performed a technical review is not adequate justification. The commenter stated that the NRC should provide technical reasons why Subsection NH is not approved for use.

The NRC disagrees and, with the exception of the application of Subsection NH to pressurizer heater sleeves constructed from Type 316 stainless steel, is unable to provide technical comments on Subsection NH at this time because it has not performed a comprehensive review of Subsection NH. A public comment on the proposed rule indicated that Subsection NH is used for the design and construction of pressurizer heater sleeves (a pressure boundary component). Accordingly, the NRC is approving the use of Subsection NH for this application. The maximum service condition for Type 316 stainless steel components that are designed and constructed in accordance with the currently approved provisions in Subsection NB is 800 °F because the reduction in material strength due to creep and relaxation effects are negligible at temperatures below 800 °F. Subsection NH provides specialized design and construction provisions when temperatures exceed 800 °F. The temperature of Type 316 stainless steel pressurizer heater sleeves reaches approximately 900 °F; therefore, Subsection NH is applicable. At 900 °F, creep and relaxation effects reduce the allowable stress at 800 °F by approximately 10 percent for Type 316 stainless steel. Therefore, a 100 °F increase in temperature above 800 °F does not significantly reduce the material strength of Type 316 stainless steel. The use of pressurizer heater sleeves constructed of Type 316 stainless steel is limited to only one type of reactor plant design in the United States. Pressurizer heater sleeves in other reactor plant designs are constructed of different materials and the temperature of the pressurizer heater sleeves in the other designs does not exceed 800 °F. Furthermore, many years operating experience indicate that pressurizer heater sleeves have not experienced creep and relaxation effects. Accordingly, the NRC concludes that the use of Subsection NH for Type 316 stainless steel pressurizer heater sleeves is technically acceptable and will provide reasonable assurance of adequate protection to public health and safety.

The NRC has not performed a full technical review of Subsection NH for other Class 1 components in future advanced reactor designs such as liquid metal and high-temperature gas-cooled reactor designs where service conditions could reach 1500 °F. At these service conditions, creep and relaxation are more pronounced. Therefore, the NRC is unable to approve the use of Subsection NH for components other than Type 316 stainless steel pressurizer heater sleeves. In consideration of public comment, § 50.55a(b)(1)(vii) is revised to allow the application of Subsection NH to Type 316 stainless steel pressurizer heater sleeves only where service conditions do not cause the component to reach temperatures exceeding 900 °F. Section 50.55a(b)(1)(vii) in the proposed rule is renumbered as § 50.55a(b)(1)(vi) in this final rule. Section 11, "Backfit Analysis," below, has been revised to address this last comment.

2.2 Section XI

The proposed rule would have revised § 50.55a(b)(2) to incorporate by reference the 2001 Edition and the 2002 and 2003 Addenda of Division 1 of Section XI of the ASME BPV Code subject to proposed modifications and limitations. Accordingly, the existing modifications and limitations for quality assurance, Class 1 piping, underwater welding, reconciliation of quality requirements, certification of nondestructive examination personnel, substitution of alternative method, and Table IWB–2500–1 examination requirements in § 50.55a(b)(2)(x), § 50.55a(b)(2)(xi), § 50.55a(b)(2)(xii), § 50.55a(b)(2)(xvii), § 50.55a(b)(2)(xviii), § 50.55a(b)(2)(xix), and § 50.55a(b)(2)(xxi), respectively, would continue to apply when using the 2001 Edition through 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. The existing modifications and limitations in § 50.55a(b)(2)(x), § 50.55a(b)(2)(xi), § 50.55a(b)(2)(xii), § 50.55a(b)(2)(xvii), § 50.55a(b)(2)(xviii), § 50.55a(b)(2)(xix), and § 50.55a(b)(2)(xxi) would continue to apply to the 2001 Edition through 2003 Addenda of Section XI because the earlier Code provisions on which these regulations are based were not revised in the 2001 through 2003 Addenda of Section XI to address the underlying issues which led the NRC to impose the modifications and limitations. There were no public comments on § 50.55a(b)(2), § 50.55a(b)(2)(x), § 50.55a(b)(2)(xi), § 50.55a(b)(2)(xii), § 50.55a(b)(2)(xviii), § 50.55a(b)(2)(xix), and § 50.55a(b)(2)(xxi). Therefore, § 50.55a(b)(2), § 50.55a(b)(2)(x), § 50.55a(b)(2)(xi), § 50.55a(b)(2)(xii), § 50.55a(b)(2)(xviii), § 50.55a(b)(2)(xix), and § 50.55a(b)(2)(xxi) are adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xvii)—Reconciliation of Quality Requirements

One commenter stated that the existing modification in § 50.55a(b)(2)(xvii) for the reconciliation of quality requirements is no longer applicable because a footnote was added to IWA–4222 that resolves the issue. The footnote was added in the 1999 Addenda to Section XI and clarifies that the provision in IWA–4222(a)(2) does not negate the requirement to implement the Owner's quality assurance program nor does it affect Owner commitments to regulatory and enforcement authorities. The NRC agrees that § 50.55a(b)(2)(xvii) is no longer applicable because the footnote addresses NRC reasons for initially implementing § 50.55a(b)(2)(xvii) in final rule dated September 22, 1999 (64 FR 51374). In consideration of this public comment, § 50.55a(b)(2)(xvii) is revised in this final rule to be applicable only when using the 1995 Addenda through 1998 Edition of Section XI.

10 CFR 50.55a(b)(2)—Footnote 10

The proposed rule would have added Footnote 10 to § 50.55a(b)(2) to indicate that the NRC has issued Order EA–03–009 which imposed enhanced reactor pressure vessel (RPV) head inspections at pressurized water reactors (PWRs). In February 2003, the NRC issued the Order to licensees of PWRs to establish interim inspection requirements that would ensure adequate protection of public health and safety. The Order was revised on February 20, 2004. The Order imposes enhanced requirements for PWR licensees that supplement areas of Section XI of the ASME BPV Code to ensure the structural and leakage integrity of the reactor coolant pressure boundary. The requirements imposed by the Order do not conflict with the requirements in Section XI of the ASME BPV Code but are needed to enhance Code requirements. Licensees are required to meet the requirements in the Order as a supplement to the requirements in the 2001 Edition with the 2002 and 2003 Addenda of Section XI of the ASME BPV Code. Licensees of PWRs using editions and addenda of Section XI of the ASME Code earlier than the 2001 Edition are currently required to apply the requirements in the Order to supplement the use of their applicable Code of record.

One commenter incorrectly interpreted Footnote 10 in the proposed rule. The commenter stated that Footnote 10 would incorporate the requirements of the Order into 10 CFR 50.55a. The NRC notes that it never intended to incorporate the requirements of the Order into 10 CFR 50.55a in this rulemaking. This final rule does not incorporate the requirements of the Order into 10 CFR 50.55a; it simply alerts the reader to the Order. Footnote 10 is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(viii)—Examination of Concrete Containments

This proposed rule would have revised the existing modification for examination of concrete containments in § 50.55a(b)(2)(viii) to apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. The modification in § 50.55a(b)(2)(viii) continues to apply to the 2001 Edition through 2003 Addenda of Section XI because the earlier ASME BPV Code provisions on which this regulation was based were not revised in the 2001 Edition through 2003 Addenda of Section XI to address the underlying issues which led the NRC to impose the modification of the ASME Code provisions. The proposed rule would have also revised the existing modification for examination of concrete containments in § 50.55a(b)(2)(viii) to require a new modification, which is discussed below, when using the 2001 Edition through 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. There were no public comments received on § 50.55a(b)(2)(viii) in the proposed rule. Therefore, § 50.55a(b)(2)(viii) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(viii)(G)—Corrosion Protection Medium (CPM)

Section 50.55a(b)(2)(viii)(G) of the proposed rule would have required that CPM be restored in accordance with the quality assurance program requirements specified in IWA–1400 when using the 2001 Edition through 2003 Addenda of Section XI. IWL–4110 of Section XI defines the scope of the repair and replacement activities associated with concrete containments. IWL–4110(b) specifies those items that are exempt from repair and replacement activity requirements. A new provision, IWL–4110(b)(3), was added in the 2002 Addenda exempting the removal, replacement, or addition of the concrete containment post-tensioning system CPM from repair and replacement requirements. Prior to the 2002 Addenda, IWL–4000 specifies that the CPM must be restored following a concrete containment post-tensioning system repair and replacement activity.

CPM is applied to containment post-tension system components to prevent corrosion. The function of the containment post-tension system is to ensure the structural integrity of the concrete containment structure under design basis loadings, and CPM is relied upon to maintain the integrity of the containment post-tension system. Therefore, the restoration of the concrete containment post-tensioning system CPM is important to ensure that the containment integrity and load capacity satisfy design basis requirements under accident conditions. For example, the acceptable concentration of water soluble chlorides, nitrates and sulfides of the replacement CPM must be verified. The amount of CPM to be installed and the method used to apply the CPM must be specified.

One commenter stated that the provisions in IWL–2500 must be applied to the restoration of CPM, and that these provisions were not revised in the 2002 Addenda. The commenter stated that quality assurance requirements must be applied when implementing IWL–2500. The NRC disagrees. The NRC believes that the provisions in IWL–2500 are not applicable to items that are exempt from Code repair and replacement activity requirements. Therefore, § 50.55a(b)(2)(viii)(G) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(ix)—Examination of Metal Containments and the Liners of Concrete Containments

The proposed rule would have revised the existing modification for examination of metal containments and the liners of concrete containments in § 50.55a(b)(2)(ix) to apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. The proposed rule stated that with the exception of the visual examination requirements specified in § 50.55a(b)(2)(ix)(B), the modification in § 50.55a(b)(2)(ix) would continue to apply to the 2001 Edition through 2003 Addenda of Section XI because the earlier Code provisions on which this regulation was based were not revised in the 2001 Edition through 2003 Addenda of Section XI to address the underlying issues which led to the NRC to impose the modification on the ASME Code provisions. The minimum illumination and distance visual examination provisions in Table IWA–2210–1 in Section XI were revised in the 2003 Addenda and are equivalent to the minimum illumination and distance visual examination requirements in § 50.55a(b)(2)(ix)(B). Therefore, the proposed rule revised the existing modification for examination of metal containments and the liners of concrete containments in § 50.55a(b)(2)(ix) to specify that § 50.55a(b)(2)(ix)(B) does not apply when using the 2001 Edition with the 2002 and 2003 Addenda of Section XI, Division 1, of the ASME BPV Code.

Several commenters stated that the revision to Table IWA–2210–1 in the 2003 Addenda of Section XI was rescinded by a special Erratum in December 2003. Therefore, the existing modification in § 50.55a(b)(2)(ix)(B) should continue to apply when using the 2001 Edition with the 2002 and 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. The NRC agrees. In consideration of the public comment, § 50.55a(b)(2)(ix) is revised in this final rule to require that § 50.55a(b)(2)(ix)(B) continue to apply when using the 2001 Edition and the 2002 and 2003 Addenda of Section XI.

10 CFR 50.55a(b)(2)(xiii)—Flaws in Class 3 Piping

The proposed rule would have revised § 50.55a(b)(2)(xiii) to eliminate the authorization to use Code Case N–513. The existing regulation in § 50.55a(b)(2)(xiii) authorizes the use of Code Cases N–513 and N–523–1. Code Case N–513 is now approved in Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." Regulatory Guide 1.147 (Revision 13) was incorporated by reference into § 50.55a in a final rule dated July 8, 2003 (68 FR 40469). Thus, it is no longer necessary to authorize the use of Code Case N–513 in § 50.55a(b)(2)(xiii) because this code case is included in Regulatory Guide 1.147. Section 50.55a(b)(2)(xiii) would continue to approve the use of Code Case N–523–1 because Code Case N–523–1 is currently not included in Regulatory Guide 1.147. There were no public comments received on § 50.55a(b)(2)(xiii) and therefore is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xiv)—Appendix VIII Personnel Qualification

The proposed rule would have revised the existing modification for Appendix VIII personnel qualification in § 50.55a(b)(2)(xiv) to apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. The modification in § 50.55a(b)(2)(xiv) continues to apply to the 2001 Edition through 2003 Addenda of Section XI because the earlier Code provisions on which this regulation was based were not revised in the 2001 Edition through 2003 Addenda of Section XI to address the underlying issues which led to the NRC to impose the modification. The proposed rule also revised § 50.55a(b)(2)(xiv) to correct an oversight. The existing regulation incorrectly states that the annual practice requirements in VII–4240 of Supplement VII of Section XI may be used. The reference to Supplement VII is incorrect; it should be Appendix VII.

Therefore, the proposed rule stated that § 50.55a(b)(2)(xiv) should be revised to state that the annual practice requirements in VII–4240 of Appendix VII of Section XI may be used.

One commenter requested that the existing annual training requirements in § 50.55a(b)(2)(xiv) be revised to change the required number of hours of training that must be completed before performing ultrasonic examinations. The NRC declines to make this change because the proposed rule did not suggest an amendment to the required number of hours of training that must be completed before performing ultrasonic examinations, and the NRC currently does not have a basis for supporting such a change. There were no other public comments received on § 50.55a(b)(2)(xiv). Therefore, § 50.55a(b)(2)(xiv) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xv)—Appendix VIII Qualification and Coverage Requirements

The proposed rule would have revised the existing modification for Appendix VIII specimen set and qualification requirements in § 50.55a(b)(2)(xv) to apply to the 2001 Edition of Section XI, Division 1, of the ASME BPV Code. The modification in § 50.55a(b)(2)(xv) would continue to apply to the 2001 Edition of Section XI because the earlier Code provisions on which this regulation was based were not revised in the 2001 Edition of Section XI to address the underlying issues which led the NRC to impose the modification. There were no public comments received on § 50.55a(b)(2)(xv). Therefore, § 50.55a(b)(2)(xv) is adopted without change in this final rule.

The proposed rule would have revised the existing regulation in § 50.55a(b)(2)(xv)(C)(*1*) to specify that the flaw depth sizing provisions in Subparagraph 3.2(c) of Supplement 4 to Appendix VIII are not applicable when Appendix VIII is implemented in accordance with § 50.55a(b)(2)(xv). Section 50.55a(b)(2)(xv) currently provides an alternative method that licensees may use for implementing Appendix VIII and the supplements to Appendix VIII. The existing regulation specifies that the flaw depth sizing provisions in Subparagraph 3.2(a) of Supplement 4 to Appendix VIII are not applicable when using the flaw depth sizing provisions specified in § 50.55a(b)(2)(xv)(C)(*1*). This revision is needed to correct an oversight that the flaw depth sizing provisions in Subparagraph 3.2(c) of Supplement 4 to Appendix VIII also do not apply when using the flaw depth sizing provisions specified in § 50.55a(b)(2)(xv)(C)(*1*). Thus, the flaw depth sizing provisions in § 50.55a(b)(2)(xv)(C)(*1*) were revised in the proposed rule to also reference Subparagraph 3.2(c) of Supplement 4 to Appendix VIII. There were no public comments received on § 50.55a(b)(2)(xv)(C)(*1*). Therefore, § 50.55a(b)(2)(xv)(C)(*1*) is adopted without change in this final rule.

The proposed rule would have revised the existing regulation in § 50.55a(b)(2)(xv)(J) to eliminate the approval to use Code Case N–552. Code Case N–552 is now approved in Regulatory Guide 1.147, Revision 13, which was incorporated by reference into § 50.55a in a final rule dated July 8, 2003 (68 FR 40469). Thus, it is no longer necessary to approve the use of Code Case N–552 in § 50.55a(b)(2)(xv)(J) because this code case is included in Regulatory Guide 1.147. There were no public comments received on § 50.55a(b)(2)(xv)(J). Therefore, § 50.55a(b)(2)(xv)(J) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xx)—System Leakage Test

The proposed rule would have revised the existing modification for system leakage tests in § 50.55a(b)(2)(xx) to continue prohibiting the use of certain system leakage test provisions in the 1997 Addenda through 2001 Edition of Section XI, Division 1 of the ASME BPV Code. The proposed rule stated that the modification in § 50.55a(b)(2)(xx) does not apply to the 2002 and 2003 Addenda of Section XI because the earlier Code provisions on which this regulation was based were revised in the 2002 Addenda of Section XI to address the underlying issues which led to the NRC to impose the modification of the ASME Code provisions. The revised system leakage test provisions in IWA–5213(a) are equivalent to the existing requirements in § 50.55a(b)(2)(xx).

One commenter stated that the system leakage test provisions in IWA–5213(a) were revised in the 2003 Addenda of Section XI not the 2002 Addenda as stated in the proposed rule. The NRC agrees. In consideration of the public comment, § 50.55a(b)(2)(xx) is revised in this final rule so that the modification applies when using IWA–5213(a), 1997 through 2002 Addenda.

10 CFR 50.55a(b)(2)(xxii)—Surface Examination

Section 50.55a(b)(2)(xxii) in the proposed rule would have prohibited the use of a new provision in IWA–2220 allowing ultrasonic (UT) examination. The provisions of Code Case N–615, "Ultrasonic Examination as a Surface Examination Method for Category B–F and B–J Piping Welds," were incorporated into IWA–2220 in the 2001 Edition of Section XI of the ASME BPV Code. Code Case N–615 and IWA–2220 allow a surface examination to be conducted using a UT examination method. The UT examination is conducted from the inside surface of certain piping welds. Other allowable surface examination methods (magnetic particle or liquid penetrant) are conducted from the outside surface of certain piping welds. The purpose of the these surface examinations is to identify flaws in the outer surface of the weld. Revision 13 to Regulatory Guide 1.147 did not approve the use of Code Case N–615 and the proposed rule would have prohibited the use of the same UT examination specified in IWA–2220. There are no provisions in Section XI that address qualification requirements and performance demonstration criteria and requirements to ensure proper consideration of flaws in the outer surface of a piping weld when conducting a UT examination from the inside surface of the piping weld.

One commenter stated that the proposed § 50.55a(b)(2)(xxii) should be deleted because IWA–2220 provides an acceptable UT performance demonstration requirement. The NRC disagrees. For example, IWA–2220 does not provide test specimen requirements, piping weld material requirements, acceptable flaw types, performance demonstration detection acceptance criteria, nor acceptable pipe specimen thickness.

A number of commenters requested that § 50.55a(b)(2)(xxii) be revised to allow IWA–2220 surface examinations be conducted by UT examination provided that the UT examination method has been demonstrated by a successful performance demonstration. The commenters stated that their revision addresses the NRC concern that there are no qualification requirements or performance demonstration criteria in Section XI for conducting a UT examination from the inside surface of the piping weld. The NRC disagrees. The revision, as proposed by the commenters, does not address the concern in the proposed rule. Appendix I of Section XI requires that all piping examinations be performed in accordance with Appendix VIII qualified procedures and personnel. The final rule dated September 22, 1999 (64 FR 51370), requires that licensees implement Appendix VIII and the supplements to Appendix VIII on an expedited basis. The NRC imposed this requirement on an expedited basis

because there were shortcomings in the qualifications of personnel and procedures in ensuring the reliability of nondestructive examination of the reactor vessel and other components of the reactor coolant system pressure boundary. The NRC believes that the imposition of performance demonstration in Appendix VIII and its supplements has enhanced the overall level of assurance of the reliability of UT examination techniques in detecting and sizing flaws. The NRC is not approving the use of new UT provision in IWA–2220 because qualification requirements and performance demonstration criteria for the new UT provision are not addressed in Appendix VIII. Therefore, § 50.55a(b)(2)(xxii) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xxiii)—IWA–4461.4.2 Evaluation of Thermally Cut Surfaces

Section 50.55a(b)(2)(xxiii) of the proposed rule would have required that all the adverse effects associated with the elimination of mechanical processing following a thermal removal process listed in IWA–4461.4.2(a)(1) through (5) be considered by tests, inspections and analyses. Tests, inspections and analyses are further discussed below. IWA–4461.4 requires that the surface left in service after the metal is removed by a thermal removal process be mechanically processed. A thermal removal process is used to remove metal from a weld or base metal. Thermal removal processes include oxyacetylene cutting, carbon arc gouging, plasma cutting, metal disintegration machining and electrodischarge machining. Thermal removal processes can leave cracks, stress risers, very rough surfaces or heavy oxidations on the surface of the metal. Mechanical processing involves the removal of any defects from a surface of the metal by grinding, machining or filing, for example. Subparagraph IWA–4461.4.2 was added in the 2001 Edition to allow the elimination of mechanical processing of a thermally cut surface when, due to field conditions, mechanical processing is deemed impractical. IWA–4461.4.2 allows the elimination of mechanical processing of thermally cut surfaces provided that the adverse effects associated with the elimination of mechanical processing listed in IWA–4461.4.2(a)(1) through (5) are considered by an evaluation. The adverse effects listed in IWA–4461.4.2(a)(1) through (5) include soundness of cut, material toughness, corrosion resistance, stresses, and oxidation or other contamination. The proposed rule stated that it is unclear if all the adverse effects listed in IWA–4461.4.2(a)(1) through (5) are required to be considered by evaluation or are licensees supposed to determine which of the adverse effects listed in IWA–4461.4.2(a)(1) through (5) would be applicable. The proposed rule stated that tests, inspections, and analyses would be required to evaluate the adverse effects listed in IWA–4461.4.2(a)(1) through (5). The proposed rule did not describe any specific test, inspection or analysis. Licensees would be responsible for determining the appropriate test, inspection, and analysis for each of the items listed in IWA–4461.4.2(a)(1) through (5).

Several commenters explained that the provision IWA–4461.4.2(a) requires that the evaluation shall include all those adverse effects listed in IWA–4461.4.2(a)(1) through (5) in the evaluation. Other commenters stated that not all of the adverse effects listed in IWA–4461.4.2(a)(1) through (5) are applicable to all thermal processes and that IWA–4461.4.2(c) requires that the evaluation document any adverse effects listed in IWA–4461.4.2(a)(1) through (a)(5) that are not applicable in the Repair/Replacement Plan. Commenters also stated that it is unreasonable for NRC to require tests, inspections, and analyses to address each of the adverse effects listed in IWA–4461.4.2(a)(1) through (5) to eliminate mechanical processing of a thermally cut surface. The tests, inspections, and analyses as proposed in § 50.55a(b)(2)(xxiii) would make it impracticable for a licensee to use the provisions in IWA–4461.4.2.

The NRC believes that it is impracticable to justify the elimination of mechanical processing of a thermally cut surface in an evaluation as specified in IWA–4461.4.2. It is not possible to evaluate the adverse effects that can occur as a result of thermal cutting without performing appropriate tests, inspections, and analyses. For example, the provisions in IWA–4461.4.2 could be used to eliminate mechanical processing for a carbon arc-gouging cut that removed a hanger in a high radiation area. If the cut is made too close to the load-bearing component, the metal on the load-bearing component could be affected by an errant arc touching the load-bearing surface or allowing some of the cutting spatter to become attached to the load-bearing surface leaving an arc strike, a heat-affected zone or a stress riser on the surface. The area around the cut must be inspected to make certain that the cutting has not damaged the surface of the component. Elimination of the inspection in a documented evaluation would not be adequate even for this simple thermal cutting example. Furthermore, the cut must be a safe distance from the surface of the component to eliminate any possibility of leaving a mechanical (a rough, oxidized or carburized surface) or metallurgical (a heat affected zone) stress riser near or in the surface of the component. If the cut is made too close to the final surface, a heat-affected zone from the cut could be left in the final load-bearing surface or a very rough, highly oxidized or carburized surface could be left very near the final load-bearing surface. The exact distance from the cut surface must be determined by an analysis or qualification testing of the configuration, not by a documented evaluation.

The NRC agrees with the comment that the test, inspection, and analysis provisions in § 50.55a(b)(2)(xxiii) of the proposed rule would make it impracticable for a licensee to use IWA–4461.4.2. Therefore, § 50.55a(b)(2)(xxiii) is revised in this final rule to prohibit the use of the new provisions in IWA–4461.4.2.

10 CFR 50.55a(b)(2)(xxiv)—UT Performance Demonstration and Coverage Requirements

Section 50.55a(b)(2)(xxiv) in the proposed rule would have prohibited the use of Appendix VIII and the supplements to Appendix VIII, and Article I–3000 in the 2002 and 2003 Addenda of Section XI of the ASME BPV Code. The elements of the Performance Demonstration Initiative (PDI) program were added to Appendix VIII and its supplements and Article I–3000 in the 2002 Addenda. PDI is an organization formed for the purpose of developing efficient, cost-effective, and technically sound UT performance demonstration methods to meet Appendix VIII requirements. The PDI program has evolved as programs were developed for each Appendix VIII supplement. Article I–3000, Examination Coverage, was also added in the 2002 Addenda to provide UT examination coverage criteria for certain welds.

The final rule dated September 22, 1999 (64 FR 51370), requires licensees to implement Appendix VIII and its supplements. The essential elements of the PDI program were added to the final rule as § 50.55a(b)(2)(xv). Section 50.55a(b)(2)(xv) also provides UT examination coverage criteria. Licensees are currently implementing Appendix VIII and its supplements in accordance with § 50.55a(b)(2)(xv). Although the NRC, ASME, and PDI have made considerable progress in the

development of UT qualification and inspection requirements, the addition of the PDI program into Section XI are not complete at this time. As a result, differences exist between the modifications in § 50.55a(b)(2)(xv), and the provisions in Appendix VIII and its supplements and Article I–3000 in the 2002 and 2003 Addenda of Section XI of the ASME BPV Code. Therefore, Appendix VIII and its supplements and the UT coverage criteria in Article I–3000 can not be implemented in accordance with § 50.55a(b)(2)(xv) when using the 2002 and 2003 Addenda. Consequently, the proposed rule would have prohibited the use of Appendix VIII and its supplements and Article I–3000 beyond the 2001 Edition.

The proposed rule stated that conflicts exist between the modifications in § 50.55a(b)(2)(xv), and the UT coverage provisions in Article I–3000 in the 2002 and 2003 Addenda. Several commenters stated that the use of the term "conflicts" in the proposed rule was inappropriate. The NRC agrees and should have used term "differences" instead of "conflicts." Commenters acknowledged that there are differences between the UT coverage requirements in Article I–3000 and the UT coverage requirements in § 50.55a(b)(2)(xv).

A number of commenters requested that the proposed limitation in § 50.55a(b)(2)(xxiv) be revised to allow the use of the UT coverage requirements in Article I–3000. Commenters stated that the NRC should accept the UT coverage requirements in Article I–3000 as an alternative to the UT coverage requirements in § 50.55a(b)(2)(xv). The NRC disagrees. Article I–3000 requires that the UT coverage provisions be applied when using UT examination procedures, equipment, and personnel qualified by performance demonstration in accordance with Appendix VIII. The NRC believes that allowing the use of the UT coverage requirements in Article I–3000 would require revising the existing UT coverage requirements in § 50.55a(b)(2)(xv) to provide licensees the choice of continuing to use the existing UT coverage requirements in § 50.55a(b)(2)(xv) or using the UT coverage requirements in Article I–3000. It is not the NRC's intention to periodically revise § 50.55a(b)(2)(xv) to add new elements of the PDI program as the program evolves. The purpose of the modification in § 50.55a(b)(2)(xv) is to provide a short-term solution that allows licensees to implement an Appendix VIII program. The long-term solution is to add the elements of the PDI program to Section XI or develop a code case that can be used to implement Appendix VIII and remove § 50.55a(b)(2)(xv) from 10 CFR 50.55a. Therefore, § 50.55a(b)(2)(xxiv) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xxv)—Mitigation of Defects by "Modification"

Section 50.55a(b)(2)(xxv) in the proposed rule would have prohibited the use of the provisions in IWA–4340 when using the 2001 Edition and the 2002 and 2003 Addenda of Section XI of the ASME BPV Code. IWA–4340 was added in the 2000 Addenda and provides requirements for the mitigation of defects by "modification." Paragraph IWA–4340 allows a defect to remain in a component provided that the defect can be eliminated from the pressure boundary by "modification."

Commenters stated that although additional provisions were added in the 2000 Addenda, Section XI has always allowed mitigation of defects by "modification." Commenters objected to the NRC prohibiting the use of this longstanding Code requirement. Commenters also stated that prohibiting the use of IWA–4340 would significantly impact licensees in terms of cost, resources, and plant shutdowns. IWA–4340 "modifications" can be designed and installed by most plants within the 72-hour technical specification allowed outage time. These "modifications" are typically used when replacement or excavation and repair welding of the defect cannot be performed within the technical specification allowed outage time. Commenters stated that it is not unusual for a plant to install several "modifications" in an operating cycle. Commenters stated that licensees would have to request authorization of an alternative pursuant to § 50.55a(a)(3) to install modifications if use of IWA–4340 is prohibited. This would result in a significant increase in regulatory burden, costs, and plant outage time and would also adversely impact NRC resources. The NRC disagrees that the mitigation of a defect by "modification" in Section XI is a longstanding Code provision. Section XI does not specifically address mitigation of defects by "modification" in the editions and addenda prior to the 2000 Addenda. The NRC is also unaware of any ASME Section XI interpretation that specifically addresses mitigation of defects by "modification." Furthermore, the NRC has authorized many alternatives pursuant to § 50.55a(a)(3) that are similar to those in IWA–4340. These alternatives were authorized on a case-by-case basis and addressed pressure testing, flaw growth evaluation, and reexamination requirements. Licensees believed these modifications were not permitted by the ASME Code, and therefore, concluded that authorizations of alternatives were necessary. Although some Section XI code cases address repair of defects on a limited basis, such as the use of weld overlays, new provisions for repairing defects were added in the 2000 Addenda.

One commenter stated that the NRC had previously approved the use of provisions that are similar to those in IWA–4340. The commenter stated that the NRC should approve the same provisions in IWA–4340. The NRC agrees that, in some instances, it had previously approved the use of mitigative methods or alternatives that could fall under the provisions of IWA–4340, but the methods approved by the NRC were significantly more comprehensive than those in IWA–4340. For example, the NRC approved the use of Code Case N–504–2, "Alternative Rule for Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping," in Regulatory Guide 1.147. The NRC notes that the provisions in Code Case N–504–2 are significantly more comprehensive than the provisions required by IWA–4340. The NRC has also authorized use of weld overlays as corrective action for intergranular stress corrosion cracking in plant-specific submittals. Authorization was based on adequate flaw evaluation, examination frequency, and pressure testing provided by licensees in their proposed alternative. However, the NRC has also disapproved the use of mitigative methods that would be allowed under IWA–4340. For example, the NRC disapproved the use of Code Case N–562–1, "Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping," in Regulatory Guide 1.193, "ASME Code Cases Not Approved For Use." The NRC disapproved the use of Code Case N–562–1 because the ASME Code and the code case do not provide criteria for determining the rate of the extent of degradation of the repair or surrounding base metal and do not specify examination requirements.

The proposed rule stated that IWA–4520(b)(2) exempts piping, pump and valve welding or brazing that does not penetrate through the pressure boundary from any pressure test. Since the modification to mitigate the defect will become the new pressure boundary and the modification may be attached to the pressure boundary by welds that do not penetrate through the pressure boundary, pressure testing would not be required. The NRC proposed to not

accept the elimination of pressure testing requirements for a modification that will function as a pressure boundary.

Commenters stated that the reference to IWA–4520(b)(2) in the proposed rule is incorrect. The NRC agrees. The NRC intended to reference IWA–4540(b)(3) in the proposed rule. IWA–4540(b)(3) exempts piping, pump and valve welding or brazing that does not penetrate through the pressure boundary from pressure testing, not IWA–4520(b)(2).

Commenters did not discuss if the pressure test exemption in IWA–4540(b)(3) would be applicable to IWA–4340 "modifications." They simply stated that Section XI requires a pressure test for new welds that are a part of the pressure boundary. The NRC agrees that pressure testing for new pressure boundary weld is a requirement. However, the NRC is concerned that licensees could interpret the provisions in IWA–4540(b)(3) that pressure tests are not required for certain IWA–4340 modifications such as an encapsulation of a defect that does not yet, but eventually could, breach the pressure boundary for example. The NRC believes that pressure testing the "modification" is necessary to validate the structural integrity of the "modification."

The proposed rule stated that IWA–4340(c) requires that each licensee define the successive examinations to be performed after the completion of the "modification." The purpose of the successive examinations is to monitor the defect to detect propagation beyond the limits of the "modification" and, when practicable, to validate the projected growth of the defect. The Code is unclear as to whether it permits a defect to propagate outside the physical boundary of the "modification" or requires that a licensee's examination program predict propagation of the defect such that the licensee would be able to identify, in advance, a defect that is expected to propagate outside the area physically modified such that corrective action could be taken.

Commenters explained that a flaw outside of the modification might be acceptable until it reached the condition of a defect. The condition would be unacceptable if the flaw propagated into a defect. Commenters also indicated that because each "modification" is unique, it is not possible to specify examination frequency criteria that could be applied to all defects that are mitigated by "modification." Commenters stated that IWA–4340(c) requires that, if practicable, the growth of the defect be predicted and licensees establish an examination method that would demonstrate that the defect has not propagated beyond the limits of the "modification." The examinations would also validate the predicted growth assumptions. In other cases, it may not be practical to predict the growth of the defect. Commenters stated that the examination frequency would have to account for this condition. The NRC believes that IWA–4340(c) is unacceptable because it does not specify minimum periodic examinations that are capable of validating the predicted defect growth assumptions. The NRC believes that it is appropriate for the Code to establish minimum periodic examination requirements. Licensees may always do more than Code minimum requirements.

One commenter states that it is inappropriate for the NRC to modify the use of Code provisions that were previously accepted by the NRC. The NRC disagrees. The modification in § 50.55a(b)(2)(xxv) was not included in the final rule that incorporated by reference the 2000 Addenda of Section XI in § 50.55a (67 FR 60520: September 26, 2002) due to an oversight by the NRC. The NRC did not identify that these Code provisions were added when it reviewed the 2000 Addenda of Section XI. The NRC has determined that this modification should only apply to those licensees who implement the 2001 Edition and later editions and addenda of Section XI, and should not be backfit to those licensees who update their ISI programs to the 1998 Edition with the 1999 and 2000 Addenda in accordance with § 50.55a(g)(4)(ii). The NRC has determined it is acceptable not to backfit the licensees who update their ISI programs to the 1998 Edition with the 1999 and 2000 Addenda because those licensees will be required at the next 10-year interval to update their ISI programs to prohibit the relevant Code provisions. Thus, any problems would be caught during the next 10-year interval. The prohibition of the relevant Code provisions is not considered a backfit because they are imposed only as part of the routine updating required as part of the 120-month updating and do not constitute a significant change to, or fundamental modification of, the existing ISI program.

Although not discussed in the proposed rule, the NRC has additional concerns about the use of IWA–4340. For example, Section XI, Appendix I, Ultrasonic Examination, directs users to the specific examination methods to be followed, including the performance demonstration requirements of Appendix VIII for certain components. IWA–4340(a) states that defects shall be characterized using nondestructive examination but has no specific requirements regarding nondestructive examination methods to be used. The NRC believes that IWA–4340(a) should specify the qualification requirements and examination methods by reference to existing rules in the Code where applicable, or where not applicable, the process to be followed to demonstrate the capability of the techniques to be used.

IWA–4340 could be used to mitigate non-planar defects, such as caused by flow accelerated corrosion or microbiological induced corrosion. The ASME has issued certain code cases, such as Code Cases N–561–1, "Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping," and N–562–1, dealing with wall thickness restoration for non-planar defects. The NRC has found these code cases to be unacceptable because of the absence of criteria concerning the extent and rate of degradation of the repair and reinspection frequencies and because the root cause of the degradation may not be mitigated. For similar reasons, the NRC finds IWA–4340 unacceptable for use to mitigate non-planar defects.

Licensees have proposed to mitigate circumferential defects above the partial penetration weld in control rod drive nozzles by partially removing the defect and replacing the removed material with weldment, thereby "embedding" the defect. The NRC has found such proposals to be unacceptable because of the possibility of additional cracking in the embedding weld and because of safety concerns posed by severance of the nozzle. The NRC finds IWA–4340 unacceptable because it could be used to mitigate such defects.

Under IWA–4340, if a defect were to propagate beyond the limits of a modification, a licensee could perform repeated repairs to the same location. The NRC believes this is unacceptable because it would represent a failure of the original evaluation to correctly predict the projected growth of the defect.

For these reasons, § 50.55a(b)(2)(xxv) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xxvi)—Pressure Testing Mechanical Joints

Section 50.55a(b)(2)(xxvi) of the proposed rule would have supplemented the test provisions in IWA–4540 of the 2001 Edition and the 2002 and 2003 Addenda of Section XI of the ASME BPV Code to require that Class 1, 2, and 3 mechanical joints be

pressure tested in accordance with IWA–4540(c) of the 1998 Edition of Section XI. The requirements to pressure test Class 1, 2, and 3 mechanical joints undergoing repair and replacement activities were deleted in the 1999 Addenda of Section XI. Therefore, pressure testing of mechanical joints is no longer required by Section XI when performing IWA–4000 repair and replacement activities. The proposed rule would have retained the pressure and testing requirements in IWA–4540(c) of the 1998 Edition when using the 2001 Edition through 2003 Addenda because there was no justification for eliminating the requirements for pressure testing Class 1, 2, and 3 mechanical joints. Pressure testing of mechanical joints affected by repair and replacement activities is necessary to ensure and verify the integrity of the pressure boundary. In the proposed rule, the NRC requested that commenters provide additional information that can be used to justify the elimination of the pressure tests requirements in IWA–4540(c) of the 1998 Edition of Section XI.

Several commenters stated that the Code requirement to conduct a system leakage test during operation at nominal operating pressure to verify leakage after reassembly of a mechanical joint was deleted in the 1999 Addenda of Section XI. The commenters indicated that this Code requirement was deleted because mechanical joint leakage is not prohibited by Section XI. The commenters contend that Section XI does not provide leakage acceptance criteria, and it has always been the responsibility of each licensee to determine if the leakage is acceptable and if corrective action is required. Furthermore, they contend that the purpose of the system leakage test in the 1998 Edition and earlier editions and addenda of Section XI is to monitor for leakage not verify the structural integrity of the pressure boundary. One commenter pointed out that the revised system leakage test requirements in the 1999 Addenda and later editions and addenda are consistent with the construction requirements for mechanical joint leakage in Section III of the ASME Code. Section III does not prohibit leakage at mechanical connections and only requires that mechanical connection leakage not mask leakage at other joints. Commenters stated that operators and system engineers periodically monitor systems for leakage and evaluate if corrective action is warranted when leakage is identified. Commenters also stated that post maintenance test programs specify requirements for leak testing mechanical connections following reassembly. Section XI does not provide any acceptance criteria for mechanical joint leakage following reassembly, and it has always been the responsibility of licensees to determine if corrective action is warranted.

The NRC and commenters generally agree that repaired or replaced mechanical joints should be pressure tested following Code repair and replacement activities. However, the NRC and commenters disagree on the role of the Code for providing this guidance. The NRC believes that it is inappropriate to rely on regulations or programs other than the Code, such as testing requirements in Appendix B of 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to provide detailed test requirements for mechanical joint repair and replacement activities. With the exception of Section XI, there are no other NRC regulations that provide detailed guidance on pressure testing mechanical joints that are repaired or replaced in accordance with Section XI. The test requirements in Section XI are technically correct and are also consistent with the test requirements in Appendix B of 10 CFR Part 50. After consideration of public comments, the NRC finds that Code pressure testing of mechanical joints after repair and replacement activities is still warranted, and that reliance on programs which are not under Code jurisdiction is not an appropriate substitute for specifying Code repair and replacement requirements.

One commenter states that it is inappropriate for the NRC to modify the use of Code provisions that were previously accepted by the NRC. The NRC disagrees. The modification in § 50.55a(b)(2)(xxvi) was not included in the final rule that incorporated by reference the 1999 Addenda of Section XI in § 50.55a (67 FR 60520: September 26, 2002) due to an oversight by the NRC. The NRC did not identify that these Code provisions were added when it reviewed the 1999 Addenda of Section XI. The NRC has determined that this modification should only apply to those licensees who implement the 2001 Edition and later editions and addenda of Section XI, and should not be backfit to those licensees who update their ISI programs to the 1998 Edition with the 1999 and 2000 Addenda in accordance with § 50.55a(g)(4)(ii). The NRC has determined it is acceptable not to backfit the licensees who update their ISI programs to the 1998 Edition with the 1999 and 2000 Addenda, because those licensees will be required at the next 10-year interval to update their ISI programs to prohibit the relevant Code provisions. Thus, any problems would be caught during the next 10-year interval. The prohibition of the relevant Code provisions is not considered a backfit because they are imposed only as part of the routine updating required as part of the 120-month updating and do not constitute a significant change to, or fundamental modification of, the existing ISI program.

For these reasons, § 50.55a(b)(2)(xxvi) is adopted without change in this final rule.

10 CFR 50.55a(b)(2)(xxvii)—Removal of Insulation

The proposed modification in § 50.55a(b)(2)(xxvii) consisted of two parts. The first part would have supplemented a new provision in IWA–5242(a) to require that insulation be removed before conducting visual examinations on bolting susceptible to stress corrosion cracking (SCC). The purpose of IWA–5242 is to periodically examine bolted connections for evidence of boric acid leakage. The 17–4 precipitation-hardened (PH) stainless steels and the 410 stainless steels installed in borated systems are susceptible to SCC when aged at a temperature below 1100 °F or have a Rockwell Method C hardness value above 30. A–286 stainless steel studs or bolts are also susceptible to SCC when preloaded to 100,000 pounds per square inch or higher. Thus, the insulation must be removed to visually examine these bolting materials. Code Case N–616, "Alternative Requirements for VT–2 Visual Examination of Classes 1, 2, and 3 Insulated Pressure Retaining Bolted Connections Section XI, Division 1," included, among other things, a provision allowing bolted connections with certain bolting materials to be examined without removing the insulation. However, this could prevent identification of signs of degraded bolting if the bolting is susceptible to SCC. The provisions of Code Case N–616 were added to IWA–5242(a) in the 2003 Addenda of Section XI of the ASME BPV Code. The NRC also conditionally accepted the use of Code Case N–616 in Regulatory Guide 1.147, by requiring that insulation be removed to examine 17–4 PH stainless steel or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or with a Rockwell Method C hardness value above 30; and A–286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.

One commenter stated that the ASME determined that a VT–2 visual examination may not be able to detect

SCC in 17–4 PH and 410 stainless steel installed in borated systems and recommended that NRC not adopt the modification in § 50.55a(b)(2)(xxvii) requiring removal of insulation prior to examining 17–4 PH and 410 stainless steel studs or bolts. The NRC agrees that it is not the intent of a VT–2 visual examination to detect SCC. However, VT–2 visual examination is an effective method for determining when conditions necessary to support SCC, such as boric acid leakage on or near a bolted connection, are present. The NRC believes that it is not prudent to attempt to detect boric acid leakage with insulation in place on connections bolted with materials susceptible to SCC. For these reasons, § 50.55a(b)(2)(xxvii) requiring that insulation be removed when conducting visual examinations on bolting susceptible to SCC is adopted without change in this final rule.

The second part of § 50.55a(b)(2)(xxvii) in the proposed rule would have supplemented IWA–5242(a) to require that a VT–2 examination of bolted connections be performed during system leakage tests. One commenter noted that the reason for this part of the proposed modification was not specifically addressed in the statement of considerations for the proposed rule. The NRC agrees. The proposed rule identified two areas in IWA–5242(a) that need to be supplemented, and the statement of considerations only described one of the areas. The reason for the second part of § 50.55a(b)(2)(xxvii) is as follows. Requirement (a) of Code Case N–533–1, "Alternative Requirements for VT–2 Visual Examination of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections," states that a "system pressure test and VT–2 visual examination shall be performed each refueling outage for Class 1 connections and each period for Class 2 and 3 connections without removal of insulation." With the exception of Requirement (a), the other provisions of Code Case N–533–1 were added to IWA–5242(a) in the 2003 Addenda of Section XI of the ASME BPV Code. The NRC proposed this modification because it appeared that all of the provisions of Code Case N–533–1 were not added in the 2003 Addenda. After further review, the NRC concludes that VT–2 examination of insulated bolted connections during system leakage tests is required by Tables IWB/C/D–2500–1 and by IWA–5241 of Section XI. Tables IWB/C/D–2500–1 require VT–2 visual examination during system leakage testing for all pressure retaining components. Paragraph IWA–5241 requires VT–2 visual examination of the accessible external exposed surfaces of pressure-retaining components for evidence of leakage and applies to insulated and non-insulated components. Therefore, the proposed requirement that a VT–2 examination of bolted connections be performed during system leakage tests is not adopted in this final rule.

10 CFR 50.55a(b)(2)(xxviii)—Reconciliation of Quality Assurance Requirements

Section 50.55a(b)(2)(xxviii) of the proposed rule would have supplemented a new provision in IWA–4226.1 to require that repair/replacement components be manufactured, procured, and controlled as safety-related under a quality assurance program meeting the requirements of Appendix B to 10 CFR Part 50. The proposed rule stated that the purpose of IWA–4226.1 (2003 Addenda) and Code Case N–554–2, "Alternative Requirements for Reconciliation of Replacement Items and Addition of New Systems," Section XI, Division 1 is to provide requirements for reconciling design requirements when using later editions of a construction code or Section III. The proposed rule stated that IWA–4226.1 and Code Case N–554–2 do not require reconciliation of the quality assurance requirements for certification, Code symbol stamping, data reports, and authorized inspection. For example, a component manufactured in a commercial shop that does not have a quality assurance program could be used in a safety-related application without having to reconcile quality assurance requirements. In Regulatory Guide 1.147, the NRC conditionally accepted the use of Code Case N–554–2 by requiring that repair/replacement components be manufactured, procured, and controlled as safety-related under a quality assurance program meeting the requirements of Appendix B to 10 CFR Part 50. The modification in § 50.55a(b)(2)(xxviii) in the proposed rule would have imposed the same quality assurance requirements on IWA–4226.1.

One commenter stated that the proposed modification in § 50.55a(b)(2)(xxviii) would prevent licensees from using a commercial grade dedication program to fabricate or procure components that are no longer available through an Appendix B supplier. The commenter proposed a revision to § 50.55a(b)(2)(xxviii) that would allow licensees to use a commercial grade dedication program to fabricate or procure components, if necessary. The NRC notes that it was not the intent of the modification in § 50.55a(b)(2)(xxviii) in the proposed rule to prevent licensees from using a commercial grade dedication program to fabricate or procure components that are no longer available through an Appendix B supplier. Another commenter stated the proposed modification in § 50.55a(b)(2)(xxviii) is unnecessary because the revision to IWA–4226.1 in the 2003 Addenda is not associated with the fabrication or procurement of components. This same commenter stated that a component manufactured in a commercial shop that does not have a quality assurance program would not be permitted in an application within the jurisdiction of Section XI unless that practice was permitted by the original Construction Code. In this case, a licensee may purchase replacement material, parts, or components from a commercial vendor and dedicate them for use in a nuclear power plant in accordance with its quality assurance program. The NRC agrees with the second commenter. The proposed modification in § 50.55a(b)(2)(xxviii) is unnecessary because the revision to IWA–4226.1 (2003 Addenda) does not change component procurement or fabrication requirements. Furthermore, the existing modification in § 50.55a(b)(2)(xvii), Reconciliation of Quality Requirements, requires that replacement parts be purchased, to the extent necessary, in accordance with the licensee's quality assurance program. In consideration of public comments, § 50.55a(b)(2)(xxviii) is not adopted in this final rule.

2.3 ASME OM Code

The proposed rule would have revised § 50.55a(b)(3) to incorporate by reference the 2001 Edition and the 2002 and 2003 Addenda of the ASME OM Code. Accordingly, the existing modifications for motor-operated valves, snubbers, and manual valves in § 50.55a(b)(3)(ii), § 50.55a(b)(3)(v), and § 50.55a(b)(3)(vi), respectively, would continue to apply when using the 2001 Edition through 2003 Addenda of the ASME OM Code. The modifications in § 50.55a(b)(3)(ii), § 50.55a(b)(3)(v), and § 50.55a(b)(3)(vi) continue to apply to the 2001 Edition through 2003 Addenda of ASME OM Code because the earlier Code provisions on which these regulations are based were not revised in the 2001 Edition through 2003 Addenda of the ASME OM Code to address the underlying issues which led to the NRC to impose the modifications. There were no public comments

received on § 50.55a(b)(3), § 50.55a(b)(3)(ii), § 50.55a(b)(3)(v), and § 50.55a(b)(3)(vi) and, therefore, these provisions are adopted without change in this final rule.

10 CFR 50.55a(b)(3)(i)—Quality Assurance

The proposed rule would have revised the existing quality assurance requirements in § 50.55a(b)(3)(i) to state that ISTA–1500 is applicable when using the 1998 Edition and later editions and addenda of the ASME OM Code. Subsections of the ASME OM Code were renumbered in the 1998 Edition; therefore, § 50.55a(b)(3)(i) is revised to account for the renumbering. This revision does not change requirements in a substantive manner. There were no public comments received on § 50.55a(b)(3)(i) and, therefore, this provision is adopted without change in this final rule.

10 CFR 50.55a(b)(3)(iii)—Code Case OMN–1

The proposed rule would have revised § 50.55a(b)(3)(iii) to eliminate the authorization to use Code Case OMN–1. The existing regulation in § 50.55a(b)(3)(iii) authorizes the use of Code Case OMN–1. Code Case OMN–1 is now approved in Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code." Regulatory Guide 1.192 (Revision 0) was incorporated by reference into § 50.55a in a final rule dated July 8, 2003 (68 FR 40469). Thus, it is no longer necessary to authorize the use of Code Case OMN–1 in § 50.55a(b)(3)(iii) because this code case is now included in Regulatory Guide 1.192. There were no public comments received on § 50.55a(b)(3)(iii) and, therefore, this provision is adopted without change in this final rule.

10 CFR 50.55a(b)(3)(iv)—Check Valve Monitoring Program

The proposed rule would have revised the existing modification for the check valve monitoring program in § 50.55a(b)(3)(iv) to continue prohibiting use of the 1995 Edition through 2002 Addenda of the ASME OM Code. The modification in (b)(3)(iv) does not apply to the 2003 Addenda of the ASME OM Code because the earlier Code provisions on which this regulation was based were revised in the 2003 Addenda of the ASME OM Code to address the underlying issues which led to the NRC to impose the modification. The check valve monitoring program requirements in Appendix II of the 2003 Addenda of the ASME OM Code are equivalent to the check valve monitoring program requirements in § 50.55a(b)(3)(iv). There were no public comments received on (b)(3)(iv) and, therefore, this provision is adopted without change in this final rule.

3. Section-by-Section Analysis for 50.55a

Paragraph (b)(1). This paragraph requires new applicants for a nuclear power plant who submit an application for a construction permit under 10 CFR Part 50 after the effective date of this rule use the 2001 Edition and the 2002 and 2003 Addenda of Section III, Division 1 of the ASME BPV Code for the design and construction of the reactor coolant pressure boundary and Quality Group B and C components. The statement of considerations for the proposed rule (69 FR 886) indicated that the proposed rule would require, inter alia, applicants for design certifications under 10 CFR Part 52 to use the 2001 Edition and the 2002 and 2003 Addenda of Section III, Division 1 of the ASME BPV Code. However, the language of the proposed rule did not provide for such applicability, and upon further consideration, the NRC believes that additional issues relating to the application of ASME Code to design certifications and other regulatory processes in Part 52 need to be considered. Accordingly, the NRC has decided not to extend by rulemaking these ASME BPV Code provisions to design certifications, and no rule change is necessary to accomplish this. This paragraph also requires that existing modifications and limitations for weld leg dimensions, seismic design, and independence of inspection in §§ 50.55a(b)(1)(ii), 50.55a(b)(1)(iii), and 50.55a(b)(1)(v), respectively, apply to the 2001 Edition through 2003 Addenda of Section III, Division 1 of the ASME BPV Code.

Paragraph (b)(1)(ii). This paragraph reconciles the change in footnote numbers in Figures NC–3673.2(b)–1 and ND–3673.2(b)–1 in Section III, Division 1 of the ASME BPV Code that were renumbered. There are no substantive changes in this paragraph.

Paragraph (b)(1)(vi). This paragraph approves the use of Subsection NH, "Class 1 Components in Elevated Temperature Service," 1995 Addenda through 2003 Addenda, for only the design and construction of Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the component to reach temperatures exceeding 900 °F. Licensees may not employ the special design methodologies for high temperatures described in Subsection NH for the design and construction of other Class 1 reactor coolant pressure boundary component applications absent specific approval by the NRC.

Paragraph (b)(2). This paragraph requires licensees of nuclear power plants to use the 2001 Edition and the 2002 and 2003 Addenda of Section XI, Division 1 of the ASME BPV Code when updating their inservice inspection programs in their subsequent 120-month interval under § 50.55a(g)(4)(ii). Existing modifications and limitations for quality assurance, Class 1 piping, underwater welding, certification of nondestructive examination personnel, substitution of alternative method, and Table IWB–2500–1 examination requirements in §§ 50.55a(b)(2)(x), 50.55a(b)(2)(xi), 50.55a(b)(2)(xii), 50.55a(b)(2)(xviii), 50.55a(b)(2)(xix), and 50.55a(b)(2)(xxi), respectively, apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1 of the ASME BPV Code. This paragraph also adds Footnote 10 which states that enhanced reactor pressure vessel head inspections have been imposed by order at pressurized water reactors, and that the NRC will determine the need for supplemental inspection requirements to be imposed through rulemaking.

Paragraph (b)(2)(viii). This paragraph requires that the existing modification for examination of concrete containments in § 50.55a(b)(2)(viii) apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1 of the ASME BPV Code, and that a new modification, § 50.55a(b)(2)(viii)(G), apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(viii)(G). This new paragraph requires that corrosion protection medium be restored in accordance with the quality assurance program requirements specified in IWA–1400 following IWL–4000 repair and replacement activities conducted on concrete containment post-tensioning systems when using the 2001 Edition through 2003 Addenda Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(ix). This paragraph requires that the existing modification for examination of metal containments and the liners of concrete containments in § 50.55a(b)(2)(ix) apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(xiii). This paragraph no longer includes the authorization to use Code Case N–513. Authorization to use Code Case N–513 is now provided in Regulatory Guide 1.147, which has been incorporated by reference into § 50.55a.

Paragraph (b)(2)(xiv). The paragraph requires that the existing modification for Appendix VIII personnel qualification in § 50.55a(b)(2)(xiv) apply to the 2001 Edition through 2003 Addenda of Section XI, Division 1, of the ASME BPV Code. The paragraph also corrects an oversight by clarifying that the annual practice requirements in VII–4240 of Appendix VII of Section XI, Division 1 of the ASME BPV Code may be used.

Paragraph (b)(2)(xv). This paragraph requires the existing modification for Appendix VIII specimen set and qualification requirements in § 50.55a(b)(2)(xv) apply to the 2001 Edition of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(xv)(C)(1). This paragraph specifies that the flaw depth sizing provisions in Subparagraph 3.2(c) of Supplement 4 to Appendix VIII of Section XI, Division 1 of the ASME BPV Code are not applicable when Appendix VIII is implemented in accordance with the provisions in § 50.55a(b)(2)(xv).

Paragraph (b)(2)(xv)(J). This paragraph no longer includes the authorization to use Code Case N–552. Authorization to use Code Case N–552 is now provided in Regulatory Guide 1.147, which has been incorporated by reference into § 50.55a. Paragraph (b)(2)(xv)(J) is reserved for future use.

Paragraph (b)(2)(xvii). This paragraph limits the existing modification for reconciliation of quality requirements in § 50.55a(b)(2)(xvii) to apply only to the 1995 Addenda through 1998 Edition of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(xx). This paragraph limits the existing modification for system leakage tests in § 50.55a(b)(2)(xx) to apply only to the 1997 Addenda through 2002 Addenda of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(xxii). This new paragraph prohibits the use of the provision in IWA–2220, 2001 Edition and the 2002 and 2003 Addenda of Section XI, Division 1 of the ASME BPV Code, that allows the use of an ultrasonic examination method to conduct a surface examination. Licensees must conduct an IWA–2220 surface examination using magnetic particle, liquid penetrant, or eddy current method.

Paragraph (b)(2)(xxiii). This new paragraph prohibits the use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA–4461.4.2 of the 2001 Edition through 2003 Addenda of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(2)(xxiv). This new paragraph prohibits the use of Appendix VIII and the supplements to Appendix VIII and Article I–3000 of the 2002 and 2003 Addenda of Section XI, Division 1 of the ASME BPV Code. Licensees are required to implement Appendix VIII and its supplements in accordance with the alternative provided in paragraph (b)(2)(xv). Licensees are also required to use the coverage requirements in paragraph (b)(2)(xv).

Paragraph (b)(2)(xxv). This new paragraph prohibits the use of IWA–4340, 2001 Edition and the 2002 and 2003 Addenda of Section XI that allows the mitigation of defects by modification.

Paragraph (b)(2)(xxvi). This new paragraph requires that the Class 1, 2, and 3 mechanical joint pressure and test provisions in IWA–4540(c) of the 1998 Edition of Section XI of the ASME Code be used when repair and replacement activities are conducted in accordance with the 2001 Edition and the 2002 and 2003 Addenda of Section XI of the ASME BPV Code.

Paragraph (b)(2)(xxvii). This new paragraph requires that the insulation be removed from 17–4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A–286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher when performing visual examinations in accordance with IWA–5242 of the 2003 Addenda of Section XI, Division 1 of the ASME BPV Code.

Paragraph (b)(3). This paragraph requires licensees of nuclear power plants to use the 2001 Edition and the 2002 and 2003 Addenda of the ASME OM Code when updating their inservice test programs in their subsequent 120-month inspection intervals under § 50.55a(f)(4)(ii). This paragraph also requires the existing modifications and limitations for quality assurance, motor-operated valve testing, snubbers, and manual valves in §§ 50.55a(b)(3)(i), 50.55a(b)(3)(ii), 50.55a(b)(3)(v), and 50.55a(b)(3)(vi), respectively, apply to the 2001 Edition through 2003 Addenda of the ASME OM Code.

Paragraph (b)(3)(i). This paragraph reconciles the different subsection and paragraph numbers of the ASME OM Code that were renumbered in the 1998 Edition and subsequent editions and addenda. There are no substantive changes in this paragraph.

Paragraph (b)(3)(iii). This paragraph no longer includes the authorization to use Code Case OMN–1. Authorization to use Code Case OMN–1 is now provided in Regulatory Guide 1.192 which has been incorporated by reference into § 50.55a. Paragraph (b)(3)(iii) is reserved for future use.

Paragraph (b)(3)(iv). This paragraph limits the existing modification for the check valve monitoring program in § 50.55a(b)(3)(iv) to the 1995 Edition through 2002 Addenda of the ASME OM Code.

4. Generic Aging Lessons Learned Report

In July 2001, the NRC issued, "Generic Aging Lessons Learned (GALL) Report," NUREG–1801, Volumes 1 and 2, for use by applicants in preparing their license renewal applications. The GALL report evaluates existing generic programs, documents the bases for determining when generic existing programs are adequate without change, and documents when generic existing programs should be augmented for license renewal. Section XI, Division 1 of the ASME BPV Code is one of the generic existing programs in the GALL report that is evaluated as an aging management program (AMP) for license renewal. Subsections IWB, IWC, IWD, IWF, IWE, and IWL of the 1995 Edition up to and including the 1996 Addenda of Section XI of the ASME BPV Code for inservice inspection were evaluated in the GALL report, and the conclusions in the GALL report are valid for these edition and addenda.

In the GALL report Sections XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," XI.S1, "ASME Section XI, Subsection IWE," XI.S2, "ASME Section XI, Subsection IWL," and XI.S3, "ASME Section XI, Subsection IWF," describe the evaluation and technical bases for determining the adequacy of Subsections IWB, IWC, IWD, IWE, IWL, and IWF, respectively. In addition, many other AMPs in the GALL report rely in part, but to a lesser degree, on the requirements in the ASME Code, Section XI (*i.e.*, XI.M3, XI.M4, XI.M5, XI.M6, XI.M7, XI.M8, XI.M9, XI.M11, XI.M12, XI.M13, XI.M14, XI.M15, XI.M16, XI.M18, XI.M24, XI.M25, and XI.M32).

The NRC has completed an evaluation of Subsections IWB, IWC, IWD, IWE, IWF, and IWL of Section XI of the ASME BPV Code (2001 Edition and the 2002 and 2003 Addenda) as part of the § 50.55a amendment process to determine if the conclusions of the GALL report are also applicable for AMPs that rely upon the ASME Code editions and addenda which are incorporated by reference into § 50.55a by the final rule. The NRC finds that the 2001 Edition and 2002 and 2003 Addenda of Sections III and XI of the ASME BPV Code are acceptable and the conclusions of the GALL report remain valid. Accordingly, an applicant may

use Subsections IWB, IWC, IWD, IWE, IWF, and IWL of Section XI of the ASME BPV Code (2001 Edition and the 2002 and 2003 Addenda) as acceptable alternatives to the requirements of the 1995 Edition up to and including the 1996 Addenda of the ASME Code, Section XI referenced in the GALL AMPs without the need to submit these alternatives for NRC review in its plant-specific license renewal application. Similarly, a licensee approved for license renewal that relied on the GALL AMPs may use Subsections IWB, IWC, IWD, IWE, IWF, and IWL of Section XI of the ASME BPV Code (2001 Edition and the 2002 and 2003 Addenda) as acceptable alternatives to the AMPs described in the GALL report. However, a licensee must assess and follow applicable NRC requirements with regard to changes to its licensing basis.

The GALL report identified areas of the 1995 Edition with the 1996 Addenda of Section XI of the ASME Code that require augmentation for license renewal. A license renewal applicant may either augment their AMPs in these areas as described in the GALL report or propose alternatives for NRC review in its plant-specific license renewal application. The GALL report's conclusions with respect to augmentation in connection with a license renewal application also apply when implementing the 2001 Edition and the 2002 and 2003 Addenda of Section XI of the ASME Code.

5. Availability of Documents

The NRC is making the documents identified below available to interested persons through one or more of the following methods as indicated.

Public Document Room (PDR). The NRC Public Document Room is located at 11555 Rockville Pike, Rockville, Maryland.

Rulemaking Web site (Web). The NRC's interactive rulemaking Web site is located at *http://ruleforum.llnl.gov.* These documents may be viewed and downloaded electronically via this Web site.

NRC's Public Electronic Reading Room (PERR). The NRC's public electronic reading room is located at *http://www.nrc.gov/reading-rm/adams.html.*

NRC Staff Contact. Single copies of the Federal Register Notice, Regulatory Analysis, Environmental Assessment, and Resolution of Public Comments can be obtained from Stephen Tingen, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Alternatively, you may contact Mr. Tingen at (301) 415–1280, or via e-mail at: *sgt@nrc.gov.*

Document	PDR	Web	PERR	NRC staff
Order EA–03–009	X	X	ML 030380470	X
Revised Order EA–03–009	X	X	ML 040220181	X
SECY–03–0078	X	X	ML 030700408	X
Federal Register Notice	X	X	ML 041200758	X
Regulatory Analysis	X	X	ML 041200761	X
Environmental Assessment	X	X	ML 041200768	X
Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 13.	X	X	ML 040230509.	
Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," Revision 0.	X	X	ML 030730430.	
NUREG–1801, "Generic Aging Lessons Learned (GALL) Report".	X	X	Volume 1—ML 012060392, Volume 2—ML 012060514.	

6. Voluntary Consensus Standards

The National Technology Transfer and Advancement Act of 1995, Pub. L. 104–113, requires that if agencies establish technical standards, the agencies use technical standards that are developed or adopted by voluntary consensus standards bodies unless the use of such a standard is inconsistent with applicable law or is otherwise impractical. Pub. L. 104–113 requires Federal agencies to use industry consensus standards to the extent practical, however, it does not require Federal agencies to endorse a standard in its entirety. The law does not prohibit an agency from generally adopting a voluntary consensus standard while taking exception to specific portions of the standard if those provisions are deemed to be "inconsistent with applicable law or otherwise impractical." Furthermore, taking specific exceptions furthers the Congressional intent of Federal reliance on voluntary consensus standards because it allows the adoption of substantial portions of consensus standards without the need to reject the standards in their entirety because of limited provisions which are not acceptable to the agency.

The NRC is amending its regulations to incorporate by reference a more recent edition and addenda of Sections III and XI of the ASME BPV Code and ASME OM Code for construction, inservice inspection, and inservice testing of nuclear power plant components. ASME BPV and OM Codes are national consensus standards developed by participants with broad and varied interests in which all interested parties (including the NRC and licensees of nuclear power plants) participate. In a staff requirements memorandum dated September 10, 1999, the Commission indicated its intent that a rulemaking identify all portions of an adopted voluntary consensus standard which are not adopted and to provide a justification for not adopting such portions. The portions of the ASME BPV Code and OM Code which the NRC does not adopt, or partially adopts, are identified in Section 2 of this final rule and the regulatory analysis. The justification for not adopting portions of the ASME BPV Code, as set forth in these statements of consideration and regulatory analysis for this rule satisfy the requirements of Section 12(d)(3) of Pub. L. 104–113, Office of Management and Budget (OMB) Circular A–119 and the Commission's direction in the staff requirements memorandum dated September 10, 1999.

7. Finding of No Significant Environmental Impact: Availability

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 is not be a major Federal action significantly affecting the quality of the human environment, and therefore, an environmental impact statement is not required.

This rulemaking will not significantly increase the probability or consequences of accidents; no changes are being made in the types of effluents that may be

released off-site; there is no increase in occupational exposure; and, there is no significant increase in public radiation exposure. Therefore, there are no significant radiological impacts associated with the proposed action. The rulemaking does not involve non-radiological plant effluents and has no other environmental impact. Therefore, no significant non-radiological impacts are associated with the action.

The determination of this environmental assessment is that there will be no significant off-site impact to the public from this action. The NRC has prepared an environmental assessment on this final rule. The environmental assessment is available as indicated in Section 5, Availability of Documents, under the SUPPLEMENTARY INFORMATION heading.

The NRC requested the views of the States on the environmental assessment for the rule and did not receive any comments from the States.

8. Paperwork Reduction Act Statement

This final rule decreases the burden on licensees for recordkeeping requirements related to examinations, tests, and repair and replacement activities. The industry annual public burden reduction for this information collection is estimated at 713 hours. Because the burden reduction for this information collection is insignificant, Office of Management and Budget (OMB) clearance is not required. Existing requirements were approved by the OMB, approval number 3150–0011.

Public Protection Notification

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

9. Regulatory Analysis

The NRC has prepared a regulatory analysis on this final rule. The analysis is available for review in the NRC's Public Document Room, located in One White Flint North, 11555 Rockville Pike, Rockville, Maryland. The regulatory analysis is available as indicated in Section 5, Availability of Documents, under the SUPPLEMENTARY INFORMATION heading.

10. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This rule affects only the licensing and 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 forth in regulations issued by the Small Business Administration at 13 CFR Part 121.

11. Backfit Analysis

The NRC's Backfit Rule, 10 CFR 50.109, states that the Commission shall require the backfitting of a facility only when it finds the action to be justified under specific standards stated in the rule. Section 50.109(a)(1) defines backfitting as the modification of or addition to systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to design, construct or operate a facility; any of which may result from a new or amended provision in the Commission rules or the imposition of a regulatory staff position interpreting the Commission rules that is either new or different from a previously applicable staff position after issuance of the construction permit or the operating license or the design approval.

Section 50.55a requires nuclear power plant licensees to construct ASME BPV Code Class 1, 2, and 3 components in accordance with the rules provided in Section III, Division 1 of the ASME BPV Code; inspect Class 1, 2, 3, Class MC, and Class CC components in accordance with the rules provided in Section XI, Division 1 of the ASME BPV Code; and test Class 1, 2, and 3 pumps and valves in accordance with the rules provided in the ASME OM Code. This rule incorporates by reference the 2001 Edition and the 2002 and 2003 Addenda of Section III, Division 1 of the ASME BPV Code; Section XI, Division 1 of the ASME BPV Code; and the ASME OM Code.

Incorporation by reference of more recent editions and addenda of Section III, Division 1 of the ASME BPV Code does not affect a plant that has received a construction permit or an operating license or a design that has been approved because the edition and addenda to be used in constructing a plant are, by rule, determined on the basis of the date of the construction permit and are not changed thereafter except voluntarily by the licensee. Thus, incorporation by reference of a more recent edition and addenda of Section III, Division 1 does not constitute a "backfitting" as defined in § 50.109(a)(1).

Incorporation by reference of more recent editions and addenda of Section XI, Division 1, of the ASME BPV Code and the ASME OM Code affect the inservice inspection (ISI) and inservice testing (IST) programs of operating reactors. However, the Backfit Rule generally does not apply to incorporation by reference of later editions and addenda of the ASME BPV Code (Section XI) and OM Code. The NRC's longstanding policy has been to incorporate later versions of the ASME Codes into its regulations. This is codified in § 50.55a which requires licensees to revise their ISI and IST programs every 120 months to the latest edition and addenda of Section XI of the ASME BPV Code and the ASME OM Code incorporated by reference into § 50.55a that is in effect 12 months prior to the start of a new 120-month ISI and IST interval. Thus, when the NRC endorses a later version of the Code, it is implementing this longstanding policy and requirement.

Other circumstances where the NRC does not apply the Backfit Rule to the endorsement of a later Code are as follows:

(1) When the NRC takes exception to a later ASME BPV Code or OM Code provision but merely retains the current existing requirement, prohibits the use of the later Code provision, limits the use of the later Code provision, or supplements the provisions in a later Code, the Backfit Rule does not apply because the NRC is not imposing new requirements. However, the NRC explains any such exceptions to the Code in the Statement of Considerations and regulatory analysis for the rule.

(2) When an NRC exception relaxes an existing ASME BPV Code or OM Code provision but does not prohibit a licensee from using the existing Code provision, the Backfit Rule does not apply because the NRC is not imposing new requirements.

(3) Modifications and limitations imposed during previous routine updates of § 50.55a have established a precedent for determining which modifications or limitations are backfits or require a backfit analysis (final rules dated August 6, 1992 (57 FR 34666), August 8, 1996 (61 FR 41303), September 22, 1999 (64 FR 51370), and September 26, 2002 (67 FR 60520)). The application of the backfit requirements to modifications and limitations in the current rule are consistent with the application of backfit requirements to modifications and limitations in previous rules.

There are some circumstances in which the endorsement of a later ASME BPV Code or OM Code introduces a

backfit. In these cases, the NRC would perform a backfit analysis or documented evaluation in accordance with § 50.109. These include the following:

(1) When the NRC endorses a later provision of the ASME BPV Code or OM Code that takes a substantially different direction from the existing requirements, the action is treated as a backfit. An example was the NRC's initial endorsement of Subsections IWE and IWL of Section XI which imposed containment inspection requirements on operating reactors for the first time. The final rule dated August 8, 1996 (61 FR 41303), incorporated by reference in § 50.55a the 1992 Edition with the 1992 Addenda of IWE and IWL of Section XI to require that containments be routinely inspected to detect defects that could compromise a containment's structural integrity. This action expanded the scope of § 50.55a to include components that were not considered by the existing regulations to be within the scope of ISI. Since those requirements involved a substantially different direction, they were treated as backfits, and justified in accordance with the standards of 10 CFR 50.109.

(2) When the NRC requires implementation of later ASME BPV Code or OM Code provision on an expedited basis, the action is treated as a backfit. This applies when implementation is required sooner than it would be required if the NRC simply endorsed the Code without any expedited language. An example was the rule dated September 22, 1999 (64 FR 51370), which incorporated by reference the 1989 Addenda through the 1996 Addenda of Section III and Section XI of the ASME BPV Code and the 1995 Edition with the 1996 Addenda of the ASME OM Code. The final rule expedited the implementation of the 1995 Edition with the 1996 Addenda of Appendix VIII of Section XI of the ASME BPV Code for qualification of personnel and procedures for performing ultrasonic examinations. The expedited implementation of Appendix VIII was considered a backfit because licensees were required to implement the new requirements in Appendix VIII prior to the next 120-month ISI program inspection interval update. Another example was the final rule dated August 6, 1992 (57 FR 34666), which incorporated by reference in § 50.55a the 1986 Addenda through the 1989 Edition of Section III and Section XI of the ASME BPV Code. The final rule added a requirement to expedite the implementation of the revised reactor vessel shell weld examinations in the 1989 Edition of Section XI. Imposing these examinations was considered a backfit because licensees were required to implement the examinations prior to the next 120-month ISI program inspection interval update.

(3) When the NRC takes an exception to a ASME BPV Code or OM Code provision and imposes a requirement that is substantially different from the existing requirement as well as substantially different than the later Code. An example was the adoption of dissimilar metal piping weld UT examination coverage requirements in the final rule dated September 26, 2002 (67 FR 60529), that incorporated by reference in § 50.55a the 1997 though 2000 Addenda of Section XI. Dissimilar metal piping weld examination coverage requirements, although contained in the 1989 Edition and earlier editions and addenda of Section XI, are not addressed in the 1989 Addenda and later editions and addenda of Section XI. Therefore, the addition of dissimilar metal piping weld examination coverage requirements to the regulation was necessary.

10 CFR 50.55a(b)(1)(vi)—Subsection NH

The modification, § 50.55a(b)(1)(b)(vi), adds a new limitation on the use of Subsection NH of the 1995 through 2003 Addenda of Section III of the ASME BPV Code for the design and construction of Class 1 reactor coolant pressure boundary components. Subsection NH was added to Section III of the ASME BPV Code in the 1995 Addenda. The NRC has determined that this subsection was adopted in a final rule dated September 22, 1999 (64 FR 51370), without performing an adequate technical review.

As discussed earlier, the NRC has determined that Subsection NH has been used to design and construct Type 316 stainless steel pressurizer heater sleeves that reach temperatures of up to 900 °F, and that the use of Subsection NH for this application is acceptable. However, the NRC has not performed a full technical review of Subsection NH for other Class 1 components in future advanced reactor designs such as liquid metal and high-temperature gas-cooled reactor designs where service conditions could reach 1500 °F. Section 50.55a(b)(1)(vi) in this final rule limits the application of Subsection NH to only pressurizer heater sleeves constructed from Type 316 stainless steel material where service conditions do not cause the component to reach temperatures exceeding 900 °F. The Backfit Rule does not apply to this limitation because, with the exception of Type 316 stainless steel pressurizer heater sleeves, licensees have not applied the provisions in Subsection NH to other Class 1 reactor coolant pressure boundary components. The Backfit Rule does not apply to rules that revise requirements that existing licensees have not applied or for future combined license applicants and design certification applicants even though such a rule may impact an applicant or licensee who was considering applying the provisions of Subsection NH to Class 1 reactor coolant pressure boundary components. For these reasons, the NRC concludes that limiting the application of Subsection NH to only Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the component to reach temperatures exceeding 900 °F does not constitute a backfit as defined in 10 CFR 50.109(a)(1).

12. Small Business Regulatory Enforcement Fairness Act

In accordance with the Small Business Regulatory Enforcement Fairness Act of 1996, the NRC has determined that this action is not a major rule and has verified this determination with the Office of Information and Regulatory Affairs of OMB.

13. Miscellaneous Public Comments on Proposed Rule

Class MC Supports

Several commenters stated that the ISI requirements for Class MC supports are not specifically addressed in § 50.55a(g). The commenters requested that NRC revise § 50.55a(g)(4) to clarify that Class MC supports must be included in ISI programs. The NRC disagrees with the commenters. The existing regulation in § 50.55a(g) states that Class MC components and their "integral attachments" must meet the ISI requirements set forth in Section XI. The use of "integral attachment" in the regulation is consistent with the terminology used in Subsection IWF of Section XI (see Figure IWF–1300–1). The provisions for the ISI of Class 1, 2, 3, and MC Component supports are included in the scope of Subsection IWF. The use of the term "integral attachment" is used in Table IWF–1300–1 and includes welded supports to MC components.

NRC Participation on ASME Code Committees

Several commenters stated that the number of modifications and limitations imposed by the NRC on later editions and addenda of the ASME Codes have

significantly increased and that the ASME and NRC committee members should strive to minimize the number of modifications and limitations. The NRC agrees that the number of modifications and limitations should be kept to a minimum. OMB Circular A–119, "Federal Participation in the Development and Use of voluntary Consensus Standards and in Conformity Assessment Activities," requires agency representatives on committees to ascertain the views of the agency to the extent possible and express views consistent with established agency views. It should be noted, however, that unanticipated events occasionally change the NRC position on an issue during final consideration.

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. 552 and 553, the NRC is adopting 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, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846); sec. 1704, 112 Stat. 2750 (44 U.S.C. 3504 note).

Section 50.7 also issued under Public Law 95–601, sec. 10, 92 Stat. 2951(42 U.S.C. 5841). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955 as amended (42 U.S.C. 2131, 2235), sec. 102, Public Law 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, Public Law 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:

■ (a) Removing and Reserving paragraphs (b)(2)(xv)(J) and (b)(3)(iii).

■ (b) Revising the introductory text of paragraph (b)(1), paragraph (b)(1)(ii), the introductory text of paragraph (b)(2), the introductory text of paragraphs (b)(2)(viii) and (b)(2)(ix), paragraph (b)(2)(xiii), paragraph (b)(2)(xiv), and the introductory text of paragraph(b)(2)(xv), paragraph (b)(2)(xv)(C)(*1*), paragraph (b)(2)(xvii), paragraph (b)(2)(xx), the introductory text of paragraph (b)(3), paragraph (b)(3)(i), and the introductory text of paragraph (b)(3)(iv).

■ (c) Adding paragraphs (b)(1)(vi), (b)(2)(viii)(G), and (b)(2)(xxii) through (b)(2)(xxvii), and Footnote 10.

§ 50.55a Codes and standards.

* * * * *

(b) * * *

(1) As used in this section, references to Section III of the ASME *Boiler and Pressure Vessel Code* refer to Section III, and include the 1963 Edition through 1973 Winter Addenda, and the 1974 Edition (Division 1) through the 2003 Addenda (Division 1), subject to the following limitations and modifications:

* * * * *

(ii) *Weld leg dimensions.* When applying the 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section, licensees may not apply paragraph NB–3683.4(c)(1), the footnote to circumferential fillet welded and socket welded joints in Figure NC–3673.2(b)–1 that permit a socket weld leg dimension to be less than 1.09 of the nominal wall thickness of the pipe or the footnote to circumferential fillet welded and socket welded joints in figure ND–3673.2(b)–1 that permit a socket weld leg dimension to be less than 1.09 of the nominal wall thickness of the pipe.

* * * * *

(vi) *Subsection NH.* The provisions in Subsection NH, "Class 1 Components in Elevated Temperature Service," 1995 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section, may only be used for the design and construction of Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the component to reach temperatures exceeding 900 °F.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, and include the 1970 Edition through the 1976 Winter Addenda, and the 1977 Edition (Division 1) through the 2003 Addenda (Division 1), subject to the following limitations and modifications:[10]

* * * * *

(viii) *Examination of concrete containments.* Licensees applying Subsection IWL, 1992 Edition with the 1992 Addenda, shall apply paragraphs (b)(2)(viii)(A) through (b)(2)(viii)(E) of this section. Licensees applying Subsection IWL, 1995 Edition with the 1996 Addenda, shall apply paragraphs (b)(2)(viii)(A), (b)(2)(viii)(D)(*3*), and (b)(2)(viii)(E) of this section. Licensees applying Subsection IWL, 1998 Edition through the 2000 Addenda shall apply paragraphs (b)(2)(viii)(E) and (b)(2)(viii)(F) of this section. Licensees applying Subsection IWL, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, shall apply paragraphs (b)(2)(viii)(E) through (b)(2)(viii)(G) of this section.

* * * * *

(G) Corrosion protection material must be restored following concrete containment post-tensioning system repair and replacement activities in accordance with the quality assurance program requirements specified in IWA–1400.

(ix) *Examination of metal containments and the liners of concrete containments.* Licensees applying Subsection IWE, 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda, shall satisfy the requirements of paragraphs (b)(2)(ix)(A) through (b)(2)(ix)(E) of this section. Licensees applying Subsection IWE, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, shall satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I) of this section.

* * * * *

(xiii) *Mechanical clamping devices.* Licensees may use the provisions of Code Case N–523–1, "Mechanical Clamping Devices for Class 2 and 3 Piping." Licensee choosing to apply Code Case N–523–1 shall apply all of its provisions.

(xiv) *Appendix VIII personnel qualification.* All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. Licensees applying the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section may use the annual practice requirements in VII–

4240 of Appendix VII of Section XI in place of the 8 hours of annual hands-on training provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Appendix VIII specimen set and qualification requirements.* The following provisions may be used to modify implementation of Appendix VIII of Section XI, 1995 Edition through the 2001 Edition. Licensees choosing to apply these provisions shall apply all of the following provisions under this paragraph except for those in § 50.55a(b)(2)(xv)(F) which are optional.

* * * * *

(C) * * *

(*1*) A depth sizing requirement of 0.15 inch RMS must be used in lieu of the requirements in Subparagraphs 3.2(a) and 3.2(c), and a length sizing requirement of 0.75 inch RMS must be used in lieu of the requirement in Subparagraph 3.2(b).

* * * * *

(J) [Reserved]

* * * * *

(xvii) *Reconciliation of Quality Requirements.* When purchasing replacement items, in addition to the reconciliation provisions of IWA-4200, 1995 Addenda through 1998 Edition, the replacement items must be purchased, to the extent necessary, in accordance with the licensee's quality assurance program description required by 10 CFR 50.34(b)(6)(ii).

* * * * *

(xx) *System leakage tests.* When performing system leakage tests in accordance IWA-5213(a), 1997 through 2002 Addenda, a 10-minute hold time after attaining test pressure is required for Class 2 and Class 3 components that are not in use during normal operating conditions, and no hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

(xxii) *Surface Examination.* The use of the provision in IWA-2220, "Surface Examination," of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, that allow use of an ultrasonic examination method is prohibited.

(xxiii) *Evaluation of Thermally Cut Surfaces.* The use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA-4461.4.2 of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are prohibited.

(xxiv) *Incorporation of the Performance Demonstration Initiative and Addition of Ultrasonic Examination Criteria.* The use of Appendix VIII and the supplements to Appendix VIII and Article I-3000 of Section XI of the ASME BPV Code, 2002 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, is prohibited.

(xxv) *Mitigation of Defects by Modification.* The use of the provisions in IWA-4340, "Mitigation of Defects by Modification," Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section are prohibited.

(xxvi) *Pressure Testing Class 1, 2, and 3 Mechanical Joints.* The repair and replacement activity provisions in IWA-4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(xxvii) *Removal of Insulation.* When performing visual examinations in accordance with IWA-5242 of Section XI, 2003 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of the section, insulation must be removed from 17-4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A-286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.

(3) As used in this section, references to the OM Code refer to the ASME *Code for Operation and Maintenance of Nuclear Power Plants,* and include the 1995 Edition through the 2003 Addenda subject to the following limitations and modifications:

(i) *Quality Assurance.* When applying editions and addenda of the OM Code, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1979 Addenda, are acceptable as permitted by ISTA 1.4 of the 1995 Edition through 1997 Addenda or ISTA-1500 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, provided the licensee uses its 10 CFR Part 50, Appendix B, quality assurance program in conjunction with the OM Code requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA-1 govern OM Code activities. If NQA-1 and the OM Code do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to OM Code activities.

* * * * *

(iii) [Reserved]

(iv) *Appendix II.* Licensees applying Appendix II, "Check Valve Condition Monitoring Program," of the OM Code, 1995 Edition with the 1996 and 1997 Addenda, shall satisfy the requirements of (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(C) of this section. Licensees applying Appendix II, 1998 Edition through the 2002 Addenda, shall satisfy the requirements of (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(D) of this section.

* * * * *

Footnotes to § 50.55a:

* * * * *

[10] Supplemental inservice inspection requirements for reactor vessel pressure heads have been imposed by Order EA-03-09 issued to licensees of pressurized water reactors. The NRC expects to develop revised supplemental inspection requirements, based in part upon a review of the initial implementation of the order, and will determine the need for incorporating the revised inspection requirements into 10 CFR 50.55a by rulemaking.

Dated at Rockville, Maryland this 14th day of September, 2004.

For the U.S. Nuclear Regulatory Commission.

Luis A. Reyes,

Executive Director for Operations.

[FR Doc. 04-21561 Filed 9-30-04; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

10 CFR Part 73

RIN 3150-AH53

Criminal History Check: Assessment of Application Fee

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to reflect an administrative change in the method of calculating the agency's application fee for criminal history checks requested by licensees. The amendment establishes the application fee amount as the sum of the user fee charged by the Federal Bureau of Investigation (FBI) for performing

10 CFR 50.55a

Current Revision As Of
January 1, 2004

§ 50.55a Codes and standards.

Each operating license for a boiling or pressurized water-cooled nuclear power facility is subject to the conditions in paragraphs (f) and (g) of this section and each construction permit for a utilization facility is subject to the following conditions in addition to those specified in § 50.55.

(a)(1) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

(2) Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), (f), and (g) of this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section.

(3) Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

(b) The ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants, which are referenced in paragraphs (b)(1), (b)(2), and (b)(3) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. NRC Regulatory Guide 1.84, Revision 32, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III" (June 2003); NRC Regulatory Guide 1.147 (Revision 0—February 1981), including Revision 1 through Revision 13 (June 2003), "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1"; and Regulatory Guide 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code" (June 2003), have been approved for incorporation by reference by the Director of the Office of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR part 51. These regulatory guides list ASME Code cases which the NRC has approved in accordance with the requirements in paragraphs (b)(4), (b)(5), and (b)(6). Copies of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016. Also, copies of these Codes and NRC Regulatory Guides 1.84, Revision 32; 1.147, through Revision 13; and 1.192 are available for inspection and copying for a fee at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html*, as well as the NRC Technical Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852-2738. Single copies of Regulatory Guides may be obtained free of charge by writing the Distribution Services Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by fax to (301) 415-2289; or by email to *DISTRIBUTION@NRC.GOV*.

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, and include the 1963 Edition through 1973 Winter Addenda, and the 1974 Edition (Division 1) through the 2000 Addenda (Division 1), subject to the following limitations and modifications:

Section III Materials. When applying the 1992 Edition of Section III, licensees must apply the 1992 Edition with the 1992 enda of Section II of the ASME Boiler and Pressure Vessel Code.

(ii) *Weld leg dimensions.* When applying the 1989 Addenda through the latest edition and addenda incorporated by

reference in paragraph (b)(1) of this section, licensees may not apply paragraph NB-3683.4(c)(1), Footnote 11 to Figure NC-3673.2(b)-1, and Figure ND-3673.2(b)-1.

(iii) *Seismic design.* Licensees may use Articles NB-3200, NB-3600, NC-3600, and ND-3600 up to and including the 1993 Addenda, subject to the limitation specified in paragraph (b)(1)(ii) of this section. Licensees may not use these Articles in the 1994 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

(iv) *Quality assurance.* When applying editions and addenda later than the 1989 Edition of Section III, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1986 Edition through the 1992 Edition, are acceptable for use provided that the edition and addenda of NQA-1 specified in NCA-4000 is used in conjunction with the administrative, quality, and technical provisions contained in the edition and addenda of Section III being used.

(v) *Independence of inspection.* Licensees may not apply NCA-4134.10(a) of Section III, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, and include the 1970 Edition through the 1976 Winter Addenda, and the 1977 Edition (Division 1) through the 2000 Addenda (Division 1), subject to the following limitations and modifications:

(i) *Limitations on specific editions and addenda.* When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (b)(2) of this section are not affected by these limitations.

(ii) *Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB-2500 and IWB-2500- 1 and Category B-J).* If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB- 2500 and Table IWB- 2600 Category B-J of Section XI of the ASME Code in the 1974 Edition and addenda through the Summer 1975 Addenda or other requirements the Commission may adopt.

(iii) *Steam generator tubing (modifies Article IWB- 2000).* If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB- 2000, the inservice inspection program for steam generator tubing is governed by the requirements in the technical specifications.

(iv) *Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC-2520 or IWC-2520-1, Category C-F).* (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, must be examined. When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for these systems must be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda.

(B) For a nuclear power plant whose application for a construction permit was docketed prior to July 1, 1978, when applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

(v) Evaluation procedures and acceptance criteria for austenitic piping (applies to IWB- 3640). When applying the Winter 1983 Addenda and Winter 1984 Addenda, the rules of paragraph IWB- 3640 may be used for all applications permitted in that paragraph, except those associated with submerged arc welds (SAW) or shielded metal arc welds (SMAW). For SAW or SMAW, use paragraph IWB- 3640, as modified by the Winter 1985 Addenda.

(vi) *Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI.* Licensees may use either the 1992 Edition with the 1992 Addenda or the 1995 Edition with the 1996 Addenda of Subsection IWE and Subsection IWL as modified and supplemented by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section when implementing the initial 120-month inspection interval for the containment inservice inspection requirements of this section. Successive 120-month interval updates must be implemented in accordance with paragraph (g)(4)(ii) of this section.

(vii) *Section XI References to OM Part 4, OM Part 6 and OM Part 10 (Table IWA-1600-1).* When using Table IWA-1600-1, "Referenced Standards and Specifications," in the Section XI, Division 1, 1987 Addenda, 1988 Addenda, or 1989 Edition, specified "Revision Date or Indicator" for ASME/ANSI OM Part 4, ASME/ANSI Part 6, and ASME/ANSI Part 10 must be OMa-1988 Addenda to the OM-1987 Edition. These requirements have been incorporated into the OM Code which is incorporated by reference in paragraph (b)(3) of this section.

(viii) *Examination of concrete containments.* Licensees applying Subsection IWL, 1992 Edition with the 1992 Addenda, shall apply paragraphs (b)(2)(viii)(A) through (b)(2)(viii)(E) of this section. Licensees applying the 1995 Edition with the 1996 Addenda shall apply paragraphs (b)(2)(viii)(A), (b)(2)(viii)(D)(*3*), and (b)(2)(viii)(E) of this section. Licensees applying the 1998 Edition with the 1999 and 2000 Addenda shall apply paragraphs (b)(2)(viii)(E) and (b)(2)(viii)(F) of this section.

(A) 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) 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, an evaluation must be performed and reported in the Engineering Evaluation Report as prescribed in IWL-3300.

(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 anchorage. A difference of more than 10 percent must be identified in the ISI Summary Report required by IWA-6000.

(D) The licensee shall report the following conditions, if they occur, in the ISI Summary Report required by IWA-6000:

(*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 exceeds 10 percent of the tendon net duct volume;

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

(E) For Class CC applications, 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 required by IWA-6000:

(*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.

(F) Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA-2300. The "owner-defined" personnel qualification provisions in IWL-2310(d) are not approved for use.

(ix) *Examination of metal containments and the liners of concrete containments.* Licensees applying Subsection IWE, 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda, shall satisfy the requirements of paragraphs (b)(2)(ix)(A) through (b)(2)(ix)(E) of this section. Licensees applying the 1998 Edition with the 1999 Addenda and 2000 Addenda shall satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I) of section.

(A) For Class MC applications, 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 as required by IWA-6000:

(*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.

(B) When performing remotely the visual examinations required by Subsection IWE, the maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.

(C) The examinations specified in Examination Category E-B, Pressure Retaining Welds, and Examination Category E-F, Pressure Retaining Dissimilar Metal Welds, are optional.

(D) Section 50.55a(b)(2)(ix)(D) may be used as an alternative to the requirements of IWE-2430.

(*1*) If the examinations reveal flaws or areas of degradation exceeding the acceptance standards of Table IWE-3410-1, an evaluation must be performed to determine whether additional component examinations are required. For each flaw or area of degradation identified which exceeds acceptance standards, the licensee shall provide the following in the ISI Summary Report required by IWA-6000:

(*i*) A description of each flaw or area, including the extent of degradation, and the conditions that led to the degradation;

(*ii*) The acceptability of each flaw or area, and the need for additional examinations to verify that similar degradation does not exist in similar components, and;

(*iii*) A description of necessary corrective actions.

(*2*) The number and type of additional examinations to ensure detection of similar degradation in similar components.

(E) A general visual examination as required by Subsection IWE must be performed once each period.

(F) VT-1 and VT-3 examinations must be conducted in accordance with IWA-2200. Personnel conducting examinations in accordance with the VT-1 or VT-3 examination method shall be qualified in accordance with IWA-2300. The "owner-defined" personnel qualification provisions in IWE-2330(a) for personnel that conduct VT-1 and VT-3 examinations are not approved for use.

(G) The VT-3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE-2500-1, and the VT-1 examination method must be used to conduct the examination in Item E4.11 of Table IWE-2500-1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE-2500-1 using the VT-3 examination method must be conducted once each interval. The "owner-defined" visual examination provisions in IWE-2310(a) are not approved for use for VT-1 and VT-3 examinations.

(H) Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE-2500-1 must be examined using the VT-3 examination method. Flaws or degradation identified during the performance of a VT-3 examination must be examined in accordance with the VT-1 examination method. The criteria in the material specification or IWB-3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT-3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT-3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

(I) The ultrasonic examination acceptance standard specified in IWE-3511.3 for Class MC pressure-retaining component must also be applied to metallic liners of Class CC pressure-retaining components.

(x) *Quality Assurance.* When applying Section XI editions and addenda later than the 1989 Edition, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1979 Addenda through the 1989 Edition, are acceptable as mitted by IWA-1400 of Section XI, if the licensee uses its 10 CFR Part 50, Appendix B, quality assurance program, in junction with Section XI requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA-1 must govern Section XI activities. Further, where NQA-1 and Section XI do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to Section XI activities.

(xi) *Class 1 piping.* Licensees may not apply IWB-1220, "Components Exempt from Examination," of Section XI, 1989 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, and shall apply IWB-1220, 1989 Edition.

(xii) *Underwater Welding.* The provisions in IWA-4660, "Underwater Welding," of Section XI, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use on irradiated material.

(xiii) *Flaws in Class 3 Piping.* Licensees may use the provisions of Code Case N-513, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping," Revision 0, and Code Case N-523-1, "Mechanical Clamping Devices for Class 2 and 3 Piping." Licensees choosing to apply Code Case N-523-1 shall apply all of its provisions. Licensees choosing to apply Code Case N-513 shall apply all of its provisions subject to the following:

(A) When implementing Code Case N-513, the specific safety factors in paragraph 4.0 must be satisfied.

(B) Code Case N-513 may not be applied to:

(*1*) Components other than pipe and tube, such as pumps, valves, expansion joints, and heat exchangers;

(*2*) Leakage through a flange gasket;

(*3*) Threaded connections employing nonstructural seal welds for leakage prevention (through seal weld leakage is not a structural flaw, thread integrity must be maintained); and

(*4*) Degraded socket welds.

(xiv) *Appendix VIII personnel qualification.* All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 8 hours of annual hands-on training on specimens that contain cracks. Licensees applying the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section may use the annual practice requirements in VII-4240 of Supplement VII of Section XI in place of the 8 hours of annual hands-on training provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Appendix VIII specimen set and qualification requirements.* The following provisions may be used to modify implementation of Appendix VIII of Section XI, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. Licensees choosing to apply these provisions shall apply all of the following provisions under paragraph (b)(2)(xv) except for those in paragraph (b)(2)(xv)(F) which are optional.

(A) When applying Supplements 2, 3, and 10 to Appendix VIII, the following examination coverage criteria requirements must be used:

(1) Piping must be examined in two axial directions, and when examination in the circumferential direction is required, the circumferential examination must be performed in two directions, provided access is available. Dissimilar metal welds must be examined axially and circumferentially.

Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds or dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws

on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld and may be used to perform examinations from either side of the weld.

(B) The following provisions must be used in addition to the requirements of Supplement 4 to Appendix VIII:

(*1*) Paragraph 3.1, Detection acceptance criteria--Personnel are qualified for detection if the results of the performance demonstration satisfy the detection requirements of ASME Section XI, Appendix VIII, Table VIII-S4-1 and no flaw greater than 0.25 inch through wall dimension is missed.

(*2*) Paragraph 1.1(c), Detection test matrix--Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB-3500, need not be included as detection flaws. For procedures applied from the inside surface, use the minimum thickness specified in the scope of the procedure to calculate a/t. For procedures applied from the outside surface, the actual thickness of the test specimen is to be used to calculate a/t.

(C) When applying Supplement 4 to Appendix VIII, the following provisions must be used:

(*1*) A depth sizing requirement of 0.15 inch RMS shall be used in lieu of the requirement in Subparagraph 3.2(a) and a length sizing requirement of 0.75 inch RMS shall be used in lieu of the requirement in Subparagraph 3.2(b).

(*2*) In lieu of the location acceptance criteria requirements of Subparagraph 2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y directions.

(*3*) In lieu of the flaw type requirements of Subparagraph 1.1(e)(1), a minimum of 70 percent of the flaws in the detection and sizing tests shall be cracks. Notches, if used, must be limited by the following:

(*i*) Notches must be limited to the case where examinations are performed from the clad surface.

(*ii*) Notches must be semielliptical with a tip width of less than or equal to 0.010 inches.

(*iii*) Notches must be perpendicular to the surface within ± 2 degrees.

(*4*) In lieu of the detection test matrix requirements in paragraphs 1.1(e)(2) and 1.1(e)(3), personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(D) The following provisions must be used in addition to the requirements of Supplement 6 to Appendix VIII:

(*1*) Paragraph 3.1, Detection Acceptance Criteria--Personnel are qualified for detection if:

(*i*) No surface connected flaw greater than 0.25 inch through wall has been missed.

(*ii*) No embedded flaw greater than 0.50 inch through wall has been missed.

(2) Paragraph 3.1, Detection Acceptance Criteria--For procedure qualification, all flaws within the scope of the procedure are detected.

(*3*) Paragraph 1.1(b) for detection and sizing test flaws and locations--Flaws smaller than the 50 percent of allowable flaw size, as defined in IWB-3500, need not be included as detection flaws. Flaws which are less than the allowable flaw size, as defined in IWB-3500, may be used as detection and sizing flaws.

(*4*) Notches are not permitted.

(E) When applying Supplement 6 to Appendix VIII, the following provisions must be used:

(*1*) A depth sizing requirement of 0.25 inch RMS must be used in lieu of the requirements of subparagraphs 3.2(a), 3.2(c)(2), and 3.2(c)(3).

(*2*) In lieu of the location acceptance criteria requirements in Subparagraph 2.1(b), a flaw will be considered detected when reported within 1.0 inch or 10 percent of the metal path to the flaw, whichever is greater, of its true location in the X and Y ·ctions.

(*3*) In lieu of the length sizing criteria requirements of Subparagraph 3.2(b), a length sizing acceptance criteria of 0.75 inch RMS must be used.

(*4*) In lieu of the detection specimen requirements in Subparagraph 1.1(e)(1), a minimum of 55 percent of the flaws must be cracks. The remaining flaws may be cracks or fabrication type flaws, such as slag and lack of fusion. The use of notches is not allowed.

(*5*) In lieu of paragraphs 1.1(e)(2) and 1.1(e)(3) detection test matrix, personnel demonstration test sets must contain a representative distribution of flaw orientations, sizes, and locations.

(F) The following provisions may be used for personnel qualification for combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII qualification. Licensees choosing to apply this combined qualification shall apply all of the provisions of Supplements 4 and 6 including the following provisions:

(*1*) For detection and sizing, the total number of flaws must be at least 10. A minimum of 5 flaws shall be from Supplement 4, and a minimum of 50 percent of the flaws must be from Supplement 6. At least 50 percent of the flaws in any sizing must be cracks. Notches are not acceptable for Supplement 6.

(*2*) Examination personnel are qualified for detection and length sizing when the results of any combined performance demonstration satisfy the acceptance criteria of Supplement 4 to Appendix VIII.

(*3*) Examination personnel are qualified for depth sizing when Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII flaws are sized within the respective acceptance criteria of those supplements.

When applying Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and ·plement 6 qualification, the following additional provisions must be used, and examination coverage must include:

(*1*) The clad to base metal interface, including a minimum of 15 percent T (measured from the clad to base metal interface), shall be examined from four orthogonal directions using procedures and personnel qualified in accordance with Supplement 4 to Appendix VIII.

(*2*) If the clad-to-base-metal-interface procedure demonstrates detectability of flaws with a tilt angle relative to the weld centerline of at least 45 degrees, the remainder of the examination volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction. This must be accomplished using a procedure and personnel qualified for single-side examination in accordance with Supplement 6. Subsequent examinations of this volume may be performed using examination techniques qualified for a tilt angle of at least 10 degrees.

(*3*) The examination volume not addressed by § 50.55a(b)(2)(xv)(G)(*1*) is considered fully examined if coverage is obtained in one parallel and one perpendicular direction, using a procedure and personnel qualified for single sided examination when the provisions of § 50.55a(b)(2)(xv)(G)(2) are met.

(H) When applying Supplement 5 to Appendix VIII, at least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation shall be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches.

(I) When applying Supplement 5, Paragraph (a), to Appendix VIII, the following provision must be used in calculating the number of permissible false calls:

(*1*) The number of false calls allowed must be D/10, with a maximum of 3, where D is the diameter of the nozzle.

'hen applying the requirements of Supplement 5 to Appendix VIII, qualifications for the nozzle inside radius performed t. . the outside surface may be performed in accordance with Code Case N-552, "Qualification for Nozzle Inside Radius Section from the Outside Surface," provided that 10 CFR 50.55a(b)(2)(xv)(I)(*1*) is also satisfied.

(K) When performing nozzle-to-vessel weld examinations, the following provisions must be used when the requirements contained in Supplement 7 to Appendix VIII are applied for nozzle-to-vessel welds in conjunction with Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification.

(*1*) For examination of nozzle-to-vessel welds conducted from the bore, the following provisions are required to qualify the procedures, equipment, and personnel:

(*i*) For detection, a minimum of four flaws in one or more full-scale nozzle mock-ups must be added to the test set. The specimens must comply with Supplement 6, paragraph 1.1, to Appendix VIII, except for flaw locations specified in Table VIII S6-1. Flaws may be either notches, fabrication flaws or cracks. Seventy-five (75) percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in paragraph (b)(2)(xv)(K)(*4*) of this section, Table VIII-S7-1--Modified, with the exception that flaws in the outer eighty-five (85) percent of the weld need not be perpendicular to the weld. There may be no more than two flaws from each category, and at least one subsurface flaw must be included.

(*ii*) For length sizing, a minimum of four flaws as in § 50.55a(b)(2)(xv)(K)(*1*)(*i*) must be included in the test set. The length sizing results must be added to the results of combined Supplement 4 to Appendix VIII and Supplement 6 to Appendix VIII. The combined results must meet the acceptance standards contained in § 50.55a(b)(2)(xv)(E)(*3*).

(*iii*) For depth sizing, a minimum of four flaws as in § 50.55a(b)(2)(xv)(K)(*1*)(*i*) must be included in the test set. Their depths must be distributed over the ranges of Supplement 4, Paragraph 1.1, to Appendix VIII, for the inner 15 percent of the wall thickness and Supplement 6, Paragraph 1.1, to Appendix VIII, for the remainder of the wall thickness. The depth sizing results must be combined with the sizing results from Supplement 4 to Appendix VIII for the inner 15 percent and to Supplement 6 to Appendix VIII for the remainder of the wall thickness. The combined results must meet the depth sizing acceptance criteria contained in §§ 50.55a(b)(2)(xv)(C)(*1*), 50.55a(b)(2)(xv)(E)(*1*), and 50.55a(b)(2)(xv)(F)(*3*).

(*2*) For examination of reactor pressure vessel nozzle-to-vessel welds conducted from the inside of the vessel,

(*i*) The clad to base metal interface and the adjacent examination volume to a minimum depth of 15 percent T (measur from the clad to base metal interface) must be examined from four orthogonal directions using a procedure and personn qualified in accordance with Supplement 4 to Appendix VIII as modified by §§ 50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C).

(*ii*) When the examination volume defined in § 50.55a(b)(2)(xv)(K)(*2*)(*i*) cannot be effectively examined in all four directions, the examination must be augmented by examination from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*).

(*iii*) The remainder of the examination volume not covered by § 50.55a(b)(*2*)(xv)(K)(*2*)(*ii*) or a combination of § 50.55a(b)(2)(xv)(K)(2)(i) and § 50.55a(b)(2)(xv)(K)(*2*)(*ii*), must be examined from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*), or from the vessel shell using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G).

(*3*) For examination of reactor pressure vessel nozzle-to-shell welds conducted from the outside of the vessel,

(*i*) The clad to base metal interface and the adjacent metal to a depth of 15 percent T, (measured from the clad to base metal interface) must be examined from one radial and two opposing circumferential directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C), for examinations performed in the radial direction, and Supplement 5 to Appendix VIII, as modified by § 50.55a(b)(2)(xv)(J), for examinations performed in the circumferential direction.

(*ii*) The examination volume not addressed by § 50.55a(b)(2)(xv)(K)(*3*)(*i*) must be examined in a minimum of one radial direction using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 to Appendix VIII, as modified by §§ 50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G)

(*4*) Table VIII-S7-1, "Flaw Locations and Orientations," Supplement 7 to Appendix VIII, is modified as follows:

Table VIII-S7-1--Modified

Flaw Locations and Orientations		
	Parallel to weld	Perpendicular to weld
...er 15 percent	X	X
OD Surface	X	.....
Subsurface	X	.....

(L) As a modification to the requirements of Supplement 8, Subparagraph 1.1(c), to Appendix VIII, notches may be located within one diameter of each end of the bolt or stud.

(M) When implementing Supplement 12 to Appendix VIII, only the provisions related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are to be applied.

(xvi) *Appendix VIII single side ferritic vessel and piping and stainless steel piping examination.*

(A) Examinations performed from one side of a ferritic vessel weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and §§ 50.55a(b)(2)(xv) (B) through (G), on specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel being examined.

(B) Examinations performed from one side of a ferritic or stainless steel pipe weld must be conducted with equipment, procedures, and personnel that have demonstrated proficiency with single side examinations. To demonstrate equivalency to two sided examinations, the demonstration must be performed to the requirements of Appendix VIII as modified by this paragraph and § 50.55a(b)(2)(xv)(A).

...i) *Reconciliation of Quality Requirements.* When purchasing replacement items, in addition to the reconciliation ...isions of IWA-4200, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, the replacement items must be purchased, to the extent necessary, in accordance with the licensee's quality assurance program description required by 10 CFR 50.34(b)(6)(ii).

(xviii) *Certification of NDE personnel.* (A) Level I and II nondestructive examination personnel shall be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA-2314, and IWA-2314(a) and IWA-2314(b) of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(B) Paragraph IWA-2316 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may only be used to qualify personnel that observe for leakage during system leakage and hydrostatic tests conducted in accordance with IWA-5211(a) and (b), 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(C) When qualifying visual examination personnel for VT-3 visual examinations under paragraph IWA-2317 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, the proficiency of the training must be demonstrated by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

(xix) *Substitution of alternative methods.* The provisions for the substitution of alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA-2240 must be applied. The provisions in IWA-2240, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, are not approved for use. The provisions in IWA-4520(c), 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, allowing the substitution of alternative examination methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code are not approved for use.

(...) *System leakage tests.* When performing system leakage tests in accordance IWA-5213(a), 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, a 10-minute hold time after

attaining test pressure is required for Class 2 and Class 3 components that are not in use during normal operating conditions, and no hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

(xxi) *Table IWB-2500-1 examination requirements.* (A) The provisions of Table IWB-2500-1, Examination Category B-D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) in the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section. A visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB-3512-1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section, may be performed in place of an ultrasonic examination.

(B) The provisions of Table IWB-2500-1, Examination Category B-G-2, Item B7.80, that are in the 1995 Edition are applicable only to reused bolting when using the 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(C) The provisions of Table IWB-2500-1, Examination Category B-K, Item B10.10, of the 1995 Addenda must be applied when using the 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(2) of this section.

(3) As used in this section, references to the OM Code refer to the ASME Code for Operation and Maintenance of Nuclear Power Plants, and include the 1995 Edition through the 2000 Addenda subject to the following limitations and modifications:

(i) *Quality Assurance.* When applying editions and addenda of the OM Code, the requirements of NQA-1, "Quality Assurance Requirements for Nuclear Facilities," 1979 Addenda, are acceptable as permitted by ISTA 1.4 of the OM Code, provided the licensee uses its 10 CFR part 50, Appendix B, quality assurance program in conjunction with the OM Code requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA-1 govern OM Code activities. If NQA-1 and the OM Code do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to OM Code activitie

(ii) *Motor-Operated Valve testing.* Licensees shall comply with the provisions for testing motor-operated valves in OM Code ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC-3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, and shall establish a program to ensure that motor-operated valves continue to be capable of performing their design basis safety functions.

(iii) *Code Case OMN-1.* As an alternative to paragraph (b)(3)(ii) of this section, licensees may use Code Case OMN-1, "Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor-Operated Valve Assemblies in Light Water Reactor Power Plants," Revision 0, in conjunction with ISTC 4.3, 1995 Edition with the 1996 and 1997 Addenda, or ISTC-3600, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section. Licensees choosing to apply the Code Case shall apply all of its provisions.

(A) The adequacy of the diagnostic test interval for each valve must be evaluated and adjusted as necessary but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of ASME Code Case OMN-1.

(B) When extending exercise test intervals for high risk motor-operated valves beyond a quarterly frequency, licensees shall ensure that the potential increase in core damage frequency and risk associated with the extension is small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(iv) *Appendix II.* Licensees applying Appendix II, "Check Valve Condition Monitoring Program," of the OM Code, 1995 Edition with the 1996 and 1997 Addenda, shall satisfy the requirements of paragraphs (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(C) of this section. Licensees applying Appendix II, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, shall satisfy the requirements of paragraphs (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(D) of this section.

(A) Valve opening and closing functions must be demonstrated when flow testing or examination methods (nonintrusive, disassembly and inspection) are used;

(B) The initial interval for tests and associated examinations may not exceed two fuel cycles or 3 years, whichever is

longer; any extension of this interval may not exceed one fuel cycle per extension with the maximum interval not to exceed 10 years; trending and evaluation of existing data must be used to reduce or extend the time interval between tests.

(C) If the Appendix II condition monitoring program is discontinued, then the requirements of ISTC 4.5.1 through 4.5.4 must be implemented.

(D) The provisions of ISTC-3510, ISTC-3520, and ISTC-3540 in addition to ISTC-5221 must be implemented if the Appendix II condition monitoring program is discontinued.

(v) *Subsection ISTD.* Article IWF-5000, "Inservice Inspection Requirements for Snubbers," of the ASME BPV Code, Section XI, provides inservice inspection requirements for examinations and tests of snubbers at nuclear power plants. Licensees may use Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants," ASME OM Code, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, in place of the requirements for snubbers in Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to their technical specifications or licensee-controlled documents. Preservice and inservice examinations must be performed using the VT-3 visual examination method described in IWA-2213.

(vi) *Exercise interval for manual valves.* Manual valves must be exercised on a 2-year interval rather that the 5-year interval specified in paragraph ISTC-3540 of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (b)(3) of this section, provided that adverse conditions do not require more frequent testing.

(4) *Design, Fabrication, and Materials Code Cases.* Licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in NRC Regulatory Guide 1.84, Revision 32, without prior NRC approval subject to the following:

(i) When an applicant or licensee initially applies a listed Code case, the applicant or licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If an applicant or licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the applicant or licensee may continue to apply the previous version of the Code case as authorized, or may apply the later version of the Code case, including any NRC-specified conditions placed on its use, until it updates its Code of Record for the component being constructed.

(iii) Application of an annulled Code case is prohibited unless an applicant or licensee applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.84. If an applicant or licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.84, the applicant or licensee may continue to apply the Code case until it updates its Code of Record for the component being constructed.

(5) *Inservice Inspection Code Cases.* Licensees may apply the ASME Boiler and Pressure Vessel Code cases listed in Regulatory Guide 1.147 through Revision 13, without prior NRC approval subject to the following:

(i) When a licensee initially applies a listed Code case, the licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If a licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code case as authorized or may apply the later version of the Code case, including any NRC-specified conditions placed on its use.

(iii) Application of an annulled Code case is prohibited unless a licensee previously applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.147. Any Code case listed as annulled in any Revision of Regulatory Guide 1.147 which a licensee has applied prior to it being listed as annulled, may continue to be applied by that licensee to the end of the 120-month interval in which the Code case was implemented.

(6) *Operation and Maintenance of Nuclear Power Plants Code Cases.* Licensees may apply the ASME Operation and Maintenance Nuclear Power Plants Code cases listed in Regulatory Guide 1.192 without prior NRC approval subject to the following:

(i) When a licensee initially applies a listed Code case, the licensee shall apply the most recent version of that Code case incorporated by reference in this paragraph.

(ii) If a licensee has previously applied a Code case and a later version of the Code case is incorporated by reference in this paragraph, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code case as authorized or may apply the later version of the Code case, including any NRC-specified conditions placed on its use.

(iii) Application of an annulled Code case is prohibited unless a licensee previously applied the listed Code case prior to it being listed as annulled in Regulatory Guide 1.192. If a licensee has applied a listed Code case that is later listed as annulled in Regulatory Guide 1.192, the licensee may continue to apply the Code case to the end of the current 120-month interval.

(c) *Reactor coolant pressure boundary*. (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III[4, 5] of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

(2) Components which are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, *Provided*:

(i) In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or

(ii) The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

(3) The Code edition, addenda, and optional ASME Code cases to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA-1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but--

(i) the edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section,

(ii) the ASME Code provisions applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition,

(iii) the ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 edition, and

(iv) The optional Code cases applied to a component must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(4) For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit.

(d) *Quality Group B components*. (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B[9] must meet the requirements for Class 2 Components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code edition, addenda, and optional ASME Code cases to be applied to the systems and components identified i paragraph (d)(1) of this section must be determined by the rules of paragraph NCA-1140, Subsection NCA of Section III the ASME Boiler and Pressure Vessel Code, but--

(i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section,

he ASME Code provisions applied to the systems and

(iii) components may be dated no earlier than the 1980 Edition, and The optional Code cases must be those listed in the NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(e) *Quality Group C components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group C[9] must meet the requirements for Class 3 components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code edition, addenda, and optional ASME Code cases to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA-1140, subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but--

(i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section,

(ii) the ASME Code provisions applied to the systems and

(iii) components may be dated no earlier than the 1980 Edition, and The optional Code cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (b) of this section.

(f) *Inservice testing requirements.* Requirements for inservice inspection of Class 1, Class 2, Class 3, Class MC, and Class CC components (including their supports) are located in § 50.55a(g).

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, pumps and valves must meet the test requirements of paragraphs (f)(4) and (f)(5) of this section to the extent ...tical. Pumps and valves which are part of the reactor coolant pressure boundary must meet the requirements applicable ...omponents which are classified as ASME Code Class 1. Other pumps and valves that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for safety-related systems (in meeting the requirements of the 1986 Edition, or later, of the Boiler and Pressure Vessel or OM Code) must meet the test requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, pumps and valves which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice tests for operational readiness set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, or 1.192 that are incorporated by reference in paragraph (b) of this section) in effect 6 months before the date of issuance of the construction permit. The pumps and valves may meet the inservice test requirements set forth in subsequent editions of this Code and addenda which are incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, or 1.192 that are incorporated by reference in paragraph (b) of this section), subject to the applicable limitations and modifications listed therein.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i)--(ii) [Reserved]

(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME Boiler ' Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases that listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of the section) referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

(iv)(A) Pumps and valves, in facilities whose construction permit was issued before November 22, 1999, which are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 2 and 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code cases listed in the NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section) referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

(v) All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the inservice test requirements, except design and access provisions, set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (f)(3) of this section and that 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.

(i) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.192 that is incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120- month interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, or 1.192 that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) [Reserved]

(iv) Inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

(5)(i) The inservice test program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (f)(4) of this section.

(ii) If a revised inservice test program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. The licensee shall submit this application, as specified in § 50.4, at least 6 months before the start of the period during which the provisions become applicable, as determined by paragraph (f)(4) of this section.

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determination.

(5) Where a pump or valve test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice test program as permitted by paragraph (f)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the test is determined to be impractical.

(6)(i) The Commission will evaluate determinations under paragraph (f)(5) of this section that code requirements are impractical. The Commission may grant relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice test program for pumps and valves for which the Commission deems that added assurance of operational readiness is necessary.

(g) *Inservice inspection requirements.* Requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in § 50.55a(f).

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued before January 1, 1971, components (including supports) must meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves, and their supports must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) in effect six months before the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of this Code which are incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section), subject to the applicable limitations and modifications.

(3) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after July 1, 1974:

(i) Components (including supports) which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular component.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME Boiler and Pressure Vessel Code incorporated by reference in paragraph (b) of this section (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section) applied to the construction of the particular component.

(iii)-(iv) [Reserved]

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

(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 that 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) of this section, subject to the limitation listed in paragraph (b)(2)(vi) of this section and the modifications listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section, to the extent practical within the limitation of design, geometry and materials of construction of the components.

(i) Inservice examinations of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months before the date of issuance of the operating license (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in NRC Regulatory Guide 1.147, through Revision 13, that are incorporated by reference in paragraph (b) of this section), subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) Licensees may, but are not required to, perform the surface examinations of High Pressure Safety Injection System specified in Table IWB-2500-1, Examination Category B-J, Item Numbers B9.20, B9.21, and B9.22.

(iv) Inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

(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 MC; and

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

(5)(i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

(ii) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specificat to the revised program. The licensee shall submit this application, as specified in § 50.4, at least six months before the start of the period during which the provisions become applicable, as determined by paragraph (g)(4) of this section.

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determinations.

(5)(iv) Where an examination requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination is determined to be impractical.

(6)(i) The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

(A) Augmented examination of reactor vessel.

(*1*) All previously granted reliefs under § 50.55a to licensees for the extent of volumetric examination of reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of subsection IWB in applicable edition and addenda of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, during the inservice inspection interval in effect on September 8, 1992 are hereby revoked, subject to the specific modification in § 50.55a(g)(6)(ii)(A)(*3*)(*iv*) for licensees that defer the augmented examination in accordance with § 50.55a(g)(6)(ii)(A)(*3*).

(*2*) All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB- 2500-1 of subsection IWB of the 1989 Edition of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in § 50.55a(g)(6)(ii)(A)(*3*) and (*4*). The augmented examination, when not deferred in accordance with the provisions of § 50.55a(g)(6)(ii)(A)(*3*), shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inservice inspection interval in effect on September 8, 1992, and may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992. For the purpose of this augmented examination, essentially 100% as used in Table IWB- 2500-1 means more than 90 percent of the examination volume of each weld, where the reduction in coverage is due to interference by another component, or part geometry.

(*3*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may defer the augmented reactor vessel examination specified in § 50.55a(g)(6)(ii)(A)(*2*) to the first period of the next inspection interval under the following conditions:

(*i*) The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992.

(*ii*) The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed.

(*iii*) If the deferred augmented examination is used as a substitute for the normally scheduled reactor vessel shell weld examination, subsequent reactor vessel shell weld examinations must be performed during the first period of successive inspection intervals.

(*iv*) Licensees that defer the augmented examination, as permitted herein, may retain all previously granted reliefs that otherwise would be revoked by § 50.55a(g)(6)(ii)(A)(*1*) for the inservice inspection interval in effect on September 8, 1992.

(*v*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may extend that interval in accordance with the provisions of section XI (1989 Edition) IWA-2430(d) for the purpose of

implementing the augmented examination during that interval.

(*vi*) The deferred augmented examination shall be performed in accordance with the related procedures specified in th section XI edition and addenda applicable to the inspection interval in which the augmented examination is performed.

(*4*) The requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100 percent of the reactor vessel shell welds specified in § 50.55a(g)(6)(ii)(A)(*2*) that has been completed, or is scheduled for implementation with a written commitment, or is required by § 50.55a(g)(4)(i), during the inservice inspection interval in effect on September 8, 1992.

(*5*) Licensees that make a determination that they are unable to completely satisfy the requirements for the augmented reactor vessel shell weld examination specified in § 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination requirements that would provide an acceptable level of quality and safety. The licensee may use the proposed alternative when authorized by the Director of the Office of Nuclear Reactor Regulation.

(B) Licensees do not have to submit to the NRC staff for approval of their containment inservice inspection programs which were developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified modifications and limitations. The program elements and the required documentation must be maintained on site for audit.

(C) *Implementation of Appendix VIII to Section XI.* (*1*) Appendix VIII and the supplements to Appendix VIII to Section XI, Division 1, 1995 Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code must be implemented in accordance with the following schedule: Appendix VIII and Supplements 1, 2, 3, and 8--May 22, 2000; Supplements 4 and 6--November 22, 2000; Supplement 11--November 22, 2001; and Supplements 5, 7, and 10--November 22, 2002.

(*2*) Licensees implementing the 1989 Edition and earlier editions and addenda of IWA-2232 of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code must implement the 1995 Edition with the 1996 Addenda of Appendix VIII and the supplements to Appendix VIII of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code.

(h) *Protection and safety systems.* (1) IEEE Std. 603-1991, including the correction sheet dated January 30, 1995, which is referenced in paragraphs (h)(2) and (h)(3) of this section, is approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of IEEE Std. 603-1991 may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, Piscataway, NJ 08855. The standard is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Md; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202-741-6030, or go to: *http://www.archives.gov/federal_register/code_of_federal_regulations/ibr_locations.html.* IEEE Std. 279, which is referenced in paragraph (h)(2) of this section, was approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of IEEE Std. 279 are also available as indicated for IEEE Std. 603-1991.

(2) Protection systems. For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603-1991 and the correction sheet dated January 30, 1995.

(3) Safety systems. Applications filed on or after May 13, 1999 for preliminary and final design approvals (10 CFR Part 52, Appendix O), design certifications, and construction permits, operating licenses and combined licenses that do not reference a final design approval or design certification, must meet the requirements for safety systems in IEEE Std. 603-1991 and the correction sheet dated January 30, 1995.

Footnotes to § 50.55a:

1--3 [Reserved]

4 USAS and ASME Code addenda issued prior to the Winter 1977 Addenda are considered to be "in effect" or "effective" 6 months after their date of issuance *and* after they are incorporated by reference in paragraph (b) of this section. Addenda

to the ASME Code issued after the Summer 1977 Addenda are considered to be "in effect" or "effective" after the date of publication of the addenda *and* after they are incorporated by reference in paragraph (b) of this section.

5 For ASME Code Editions and Addenda issued prior to the Winter 1977 Addenda, the Code Edition and Addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the Winter 1977 Addenda and subsequent editions and addenda the method for determining the applicable Code editions and addenda is contained in Paragraph NCA 1140 of Section III of the ASME Code.

6-8 [Reserved]

9 Guidance for quality group classifications of components which are to be included in the safety analysis reports pursuant to § 50.34(a) and § 50.34(b) may be found in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants," and in Section 3.2.2 of NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants."

[36 FR 11424, June 12, 1971]

EDITORIAL NOTE: For Federal Register citations affecting § 50.55a, see the List of CFR Sections Affected.

10 CFR 50.55a

NRC REGULATORY ISSUE SUMMARY 2004-12

7/28/2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, DC 20555-0001

July 28, 2004

NRC REGULATORY ISSUE SUMMARY 2004-12 CLARIFICATION ON USE OF LATER EDITIONS AND ADDENDA TO THE ASME OM CODE AND SECTION XI

ADDRESSEES

All holders of operating licenses for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to clarify the requirements for inservice testing (IST) and inservice inspection (ISI) when using later editions and addenda of the American Society of Mechanical Engineers (ASME) *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), and Section XI of the ASME *Boiler and Pressure Vessel Code* (B&PV Code), as required by Title 10, Section 50.55a, paragraphs (f)(4)(iv) and (g)(4)(iv) of the *Code of Federal Regulations.*

This RIS requires no action or written response on the part of an addressee.

BACKGROUND INFORMATION

The regulations in 10 CFR 50.55a(f)(4) and (g)(4) establish the effective ASME Code edition and addenda to be used by licensees for performing inservice inspections of components (including supports) and inservice testing of pumps and valves. Paragraph 50.55a(f)(4)(iv) states that inservice test of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of

ML042090436

Section 50.55a, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. In addition, it allows the use of portions of editions or addenda provided that all related requirements of the respective editions or addenda are met. Similarly, for inservice examination and system pressure tests, Paragraph 50.55a(g)(4)(iv) allows the use of subsequent editions and addenda, and portions thereof, incorporated by reference in paragraph (b) of Section 50.55a, subject to the limitations and modifications listed in paragraph (b), and subject to Commission approval.

Paragraphs 50.55a(f)(4)(ii) and (g)(4)(ii) require the use of the latest edition and addenda that has been incorporated by reference 1 year prior to the beginning of each 120-month interval. This is considered the Code of Record. As stated in paragraphs (f)(4)(iv) and (g)(4)(iv), the use of later editions and addenda of the Code is subject to Commission approval. Licensees that plan to use for their IST and ISI programs later editions and addenda of the ASME OM Code or Section XI that have been incorporated by reference into 10 CFR 50.55a must obtain prior approval pursuant to 10 CFR 50.55a(f)(4)(iv) or (g)(4)(iv). The licensees may request this approval by submitting a letter to the NRC Document Control Desk.

SUMMARY OF ISSUE

The staff has identified several instances in which nuclear power plants licensees have used portions of editions and addenda of the OM Code and/or the B&PV Code issued after their Code of Record without requesting approval from the Commission. In addition, several licensees have raised questions recently about whether staff approval is necessary prior to using subsequent editions and addenda of the Code (or portions thereof). The staff is issuing this RIS in order to clarify this matter.

Paragraphs 10 CFR 50.55a(f)(4)(iv) and (g)(4)(iv) require that inservice tests of pumps and valves, inservice examinations of components, and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval. The fact that these Code editions and addenda have been incorporated by reference into the regulations does not imply that Commission approval has already been given. Licensees must request approval to use later Code editions and addenda via a letter to the NRC; they may not just update their IST or ISI program. When requesting to use editions and addenda of the ASME Code that have not yet been incorporated by reference, licensees must request authorization to use these later editions and addenda as an alternative to the regulations pursuant to 10 CFR 50.55a(a)(3).

Some licensees mistakenly believe that requesting approval to use a later Code edition or addenda is the same as requesting relief from certain Code requirements. The request to use a later edition and addenda is not a relief request; it is simply a request to use a later Code. Relief requests are addressed under 10 CFR 50.55a(f)(5)(iv) and (g)(5)(iv). Similarly, the use of alternatives to Code requirements is addressed under 10 CFR 50.55a(a)(3). The amount of written documentation needed for a request to use a later Code edition and addenda is significantly less than for a relief request or a request to use an alternative requirement. For

example, licensees are not required to justify requests to use the later Code editions and addenda. In contrast, when submitting a relief request, licensees must provide the basis for why the Code requirement is impractical. Likewise, when requesting authorization for use of an alternative pursuant to 10 CFR 50.55a(a)(3), licensees must demonstrate an acceptable level of quality and safety, or demonstrate why compliance with the Code requirement would result in hardship or unusual difficulty without a compensating level of quality and safety.

If portions of a later Code edition and addenda are used, licensees must assure that all related requirements of the respective editions and addenda are met. A discussion of the related requirements should be included in the letter to the NRC. The regulations do not specify when the letter must be submitted, only that it be submitted before using the proposed later Code edition and addenda.

BACKFIT DISCUSSION

This RIS clarifies current regulatory requirements. The RIS imposes no new requirements and requires no action or written response. Therefore, it does not constitute a backfit under 10 CFR 50.109, and the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment was not published in the *Federal Register* because this RIS is informational and pertains to a staff position that does not represent a departure from current regulatory requirements and practice.

SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996

The NRC has determined that this action is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not request any information collections, and, therefore, is not subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

CONTACT

Please direct any questions about this matter to the technical contact listed below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Terrence Reis, Acting Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contact: Jorge Hernandez, NRR
301-415-4093
E-mail: jeh3@nrc.gov

Attachment: List of Recently Issued NRC Regulatory Issue Summaries

LIST OF RECENTLY ISSUED
NRC REGULATORY ISSUE SUMMARIES

Regulatory Issue Summary No.	Subject	Date of Issuance	Issued to
2003-18, Supplement 1	Use of Nuclear Energy INSTITUTE (NEI) 99-01, "Methodology for Development of Emergency Action Levels," Revision 4, Dated January 2003	07/13/2004	All holders of operating licenses for nuclear power reactors and licensees that have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
2004-11	Supporting Information Associated with Requests For Withholding Proprietary Information	06/29/2004	All submitters of proprietary information to the Nuclear Regulatory Commission.
2004-10	Preparation And Scheduling of Operator Licensing Examinations	06/14/2004	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
2004-09	Status on Deferral of Active Regulation of Ground-water Protection At In Situ Leach Uranium Extraction Facilities	06/07/2004	All holders of materials licenses for uranium and thorium recovery facilities.
2004-08	Results of the License Termination Rule Analysis	05/28/2004	All holders of operating licenses for nuclear power reactors, research and test reactors, as well as decommissioning sites.
2004-07	Release of Final Review Standard (RS)-002, "Processing Applications for Early Site Permits"	05/19/2004	All holders of operating licenses for nuclear power reactors, all applicants for early site permits (ESPs), and all prospective vendors of nuclear power plants in the United States.

Note: NRC generic communications may be received in electronic format shortly after they are issued by subscribing to the NRC listserver as follows:

To subscribe send an e-mail to <listproc@nrc.gov >, no subject, and the following command in the message portion:

subscribe gc-nrr firstname lastname

OL = Operating License
CP = Construction Permit

10 CFR 50.55a

NRC REGULATORY ISSUE SUMMARY 2004-16

10/19/2004

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

October 19, 2004

NRC REGULATORY ISSUE SUMMARY 2004-16: USE OF LATER EDITIONS AND ADDENDA TO ASME CODE SECTION XI FOR REPAIR/REPLACEMENT ACTIVITIES

ADDRESSEES

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

INTENT

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to clarify the requirements in Title 10 of the Code of Federal Regulations, Part 50, paragraph 55a(g)(4)(iv) (10 CFR 50.55a(g)(4)(iv)), and the use of American Society of Mechanical Engineers (ASME) Code Case N-389-1, "Alternative Rules for Repairs, Replacements, or Modifications-Section XI, Division 1," when applying the rules of later editions and addenda of ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Plant Components."

BACKGROUND

Section 50.55a governs the use of codes and standards, including the ASME Boiler and Pressure Vessel Code (ASME Code). Paragraph 50.55a(g)(4)(iv) provides requirements on the use of later editions and addenda of the ASME Code for the inservice examination of components and system pressure tests.

ASME Code Case (Code Case) N-389-1 provides guidance on which editions and addenda of ASME Code Section XI licensees may use when making a repair, replacement, or modification to nuclear plant components. Specifically, the Code Case states that licensees may use later editions and addenda of ASME Code Section XI than those specified in the owner's inservice inspection program, provided that all related requirements are met. The Code Case further states that the later editions and addenda need to be accepted by the enforcement and regulatory authorities having jurisdiction at the plant site.

Code Cases provide alternatives to existing ASME Code requirements. Those Code Cases that the NRC finds acceptable are included in NRC's Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." Regulatory Guide 1.147,

Revision 13, was incorporated by reference into 10 CFR 50.55a, "Codes and Standards," by a final rule (68 FR 40469, July 8, 2003). The NRC approved Code Case N-389-1 in Regulatory Guide 1.147, Revision 13, with no conditions. The provisions of Code Case N-389-1 have also been incorporated into Section XI of the ASME Code in the 1995 edition through 1995 addenda. It should be noted that the phrase used in the latest ASME Code Section XI for repair, replacement, and modification is "repair/replacement activities."

SUMMARY OF ISSUE

Questions have arisen regarding the need for licensees to seek prior NRC review and approval to use the later editions and addenda of the ASME Code Section XI for repair/replacement activities, in accordance with the requirements of 10 CFR 50.55a(g)(4)(iv). The NRC is aware that some licensees may have used later editions and addenda of the ASME Code without prior NRC review and approval. Several licensees have expressed the view that because Code Case N-389-1 is applicable only to repair/replacement activities and because 10 CFR 50.55a(g)(4)(iv) only addresses inservice examination requirements, the prior NRC approval requirement of 10 CFR 50.55a(g)(4)(iv) does not apply to the use of Code Case N-389-1. Other licensees have expressed that there is an apparent conflict between 10 CFR 50.55a(g)(4)(iv) and 10 CFR 50.55a(b)(5) regarding the need for NRC approval when implementing Code Case N-389-1.

Paragraph 50.55a(b)(2) incorporates by reference those editions and addenda of the ASME Code Section XI, that the NRC finds acceptable, subject to certain limitations and modifications. Currently, paragraph 50.55a(b)(2) incorporates by reference the ASME Code Section XI from the 1970 edition through the 2000 addenda (Division 1).

Paragraph 50.55a(b)(5) incorporates by reference Regulatory Guide 1.147, Revision 13, including those Code Cases which the NRC finds acceptable. Paragraph 50.55a(b)(5) states that "...licensees may apply the ASME Boiler and Pressure Vessel Code Cases listed in Regulatory Guide 1.147 through Revision 13, without prior NRC approval..."

Paragraph 50.55a(g)(4) states, in part, that throughout the service life of a nuclear power facility, components must meet the requirements set forth in Section XI of editions of the ASME Code and addenda. The repair, replacement, and modification of plant components are not explicitly mentioned in 10 CFR 50.55a(g)(4) and associated subparagraphs. However, these activities are specifically mentioned in ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." The NRC staff considers that these activities are not separate and distinct from, but are included under, inservice examinations. Therefore, the requirements of 10 CFR 50.55a(g)(4)(iv) are applicable to repair/replacement activities.

Paragraph 50.55a(g)(4)(iv) states specifically that inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda of the ASME Code provided that they are incorporated by reference in 10 CFR 50.55a(b), subject to the limitations and modifications listed in 10 CFR 50.55a(b) and subject to Commission approval. Portions of editions or addendum may be used provided that all related requirements of the respective editions or addenda are met.

The NRC finds no conflict between the provisions of Code Case N-389-1 and 10 CFR 50.55a(g)(4)(iv) because the Code Case is silent on the issue of prior NRC review and approval regarding the use of later editions and addenda of the ASME Code for repair/replacement activities. In addition, the primary purpose of Code Case N-389-1 is the ASME Code's response to licensees' inquiries regarding which editions and addenda of Section XI may be used as alternatives to those specified in the Owner's Inservice Inspection Program.

Moreover, with regard to the differences in requirements between 10 CFR 50.55a(g)(iv) and 10 CFR 50.55a(b)(5), the requirements in 10 CFR 50.55a(b)(5) apply to adopting and codifying Code Cases, whereas 10 CFR 50.55a(g)(4)(iv) specifies the overall inservice inspection requirements including requirement for repair/replacement activities. Therefore, the requirements in 10 CFR 50.55a(b)(5) are subordinated to the requirements in 10 CFR 50.55a(g)(4)(iv). The purpose of the NRC review and approval for the use of later ASME editions and addenda is to provide and monitor consistency in the use of the appropriate ASME editions and addenda in repair/replacement activities of nuclear plant components.

On the basis of the above discussion, the NRC concludes that licensees who wish to use provisions of subsequent editions and addenda of the ASME Code Section XI for activities, including repair/replacement activities (e.g., Code Case N-389-1), must receive prior NRC review and approval as required by 10 CFR 50.55a(g)(4)(iv).

BACKFIT DISCUSSION

This RIS requires no action or written response and is, therefore, not a backfit under 10 CFR 50.109. Consequently, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because it is informational and pertains to a staff position that does not depart from current regulatory requirements and practice.

SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996

The NRC has determined that this action is not a new rule and therefore is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain an information collection request and therefore is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

CONTACT

Please direct any questions about this matter to the technical contact listed below or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

/RA/
Francis M. Costello, Acting Chief
Reactor Operations Branch
Division of Inspection Program Management
Office of Nuclear Reactor Regulation

Technical Contact: John Tsao, NRR
301-415-2702
jct@nrc.gov

Attachment: List of Recently Issues NRC Regulatory Issue Summaries

LIST OF RECENTLY ISSUED NRC REGULATORY ISSUE SUMMARIES

Regulatory Issue Summary No.	Subject	Date of Issuance	Issued to
2004-15	Emergency Preparedness Issues: Post 9/11	10/18/2004	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
2004-14	Focusing Resources in the Office of Nuclear Reactor Regulation as a Result of Review of Security Plan Changes	09/20/2004	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
2004-13	Consideration of Sheltering in Licensee's Range of Protective Action Recommendations	08/02/2004	All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.
2004-12	Clarification on Use of Later Editions and Addenda to the ASME OM Code and Section XI	07/28/2004	All holders of operating licenses for nuclear power reactors except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Note: NRC generic communications may be received in electronic format shortly after they are issued by subscribing to the NRC listserver as follows:

To subscribe send an e-mail to <listproc@nrc.gov >, no subject, and the following command in the message portion:

subscribe gc-nrr firstname lastname

OL = Operating License
CP = Construction Permit