Proposed Rules

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This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AE26

Industry Codes and Standards; Amended Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) regulations require that nuclear power plant owners 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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code), 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 ASME BPV Code, and test Class 1, Class 2, and Class 3 pumps and valves in accordance with the rules provided in Section XI, Division 1, of the ASME BPV Code.

The NRC proposes to amend 10 CFR 50.55a to revise the requirements for construction, inservice inspection (ISI), and inservice testing (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, 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 require licensees to implement Section XI, Division 1, of the ASME BPV Code, 1995 Edition with the 1996 Addenda, for Class 1, Class 2, and Class 3 components with five proposed limitations. Licensees would be permitted to implement: Code Case N–513 which addresses flaws in low and moderate energy Class 3 piping; Code Case N–523 which addresses the temporary use of mechanical clamps in Class 2 and 3 piping; and Subsection IWE and Subsection IWL, 1995 Edition with the 1996 Addenda.

The proposed rule would expedite implementation of Appendix VIII, ''Performance Demonstration for Ultrasonic Examination Systems,'' to Section XI, Division 1, with three proposed modifications. An expedited implementation schedule would also be required for a proposed modification to Section XI which addresses volumetric examination of the Class 1 high pressure safety injection (HPSI) system in pressurized water reactors (PWRs).

For IST, the proposed amendment would require licensees to implement the 1995 Edition with the 1996 Addenda of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) for Class 1, Class 2, and Class 3 pumps and valves with one limitation and one modification. 10 CFR 50.55a has been clarified with respect to which pumps and valves are to be included in a licensee's IST program. Licensees would be permitted to implement: Code Case OMN–1 with one modification in lieu of stroke time testing; Appendix II (which is an alternative to the check valve condition monitoring program provisions contained in Subsection ISTC of the OM Code) with three proposed modifications; and Subsection ISTD for the IST of snubbers. Finally, based upon supporting information received since the last rulemaking, the modification presently in § 50.55a for containment isolation valve inservice testing has been deleted.

The Statement of Considerations concludes by clarifying the NRC position regarding ASME Code Interpretations, and discussing NRC Direction Setting Issue Number 13 (DSI–13) with regard to NRC endorsement of industry codes and standards.

DATES: Submit comments by March 3, 1998. Comments received after this date will be considered if it is practical to do so, but the Commission is able to ensure consideration only for comments received on or before this date.

ADDRESSES: Comments may be sent to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. ATTN: Rulemaking and Adjudications Staff. Hand deliver comments to 11545 Rockville Pike, Rockville, Maryland, 20852, between 7:30 am and 4:15 pm on Federal workdays.

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 website, contact Ms. Carol Gallagher, (301) 415–5905; e-mail CAG@nrc.gov.

Single copies of this proposed rulemaking may be obtained by written request or telefax to 301–415–2260 or from Frank C. Cherny, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6786, or Wallace E. Norris, Division of Engineering Technology, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6796. Certain documents related to this rulemaking, including comments received, may be examined at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. These same documents may also be viewed and downloaded via the interactive rulemaking website as established by NRC for this rulemaking.

FOR FURTHER INFORMATION CONTACT: Frank C. Cherny, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6786, or Wallace E. Norris, Division of Engineering Technology, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6796.

SUPPLEMENTARY INFORMATION:

1. Background

The NRC is proposing to amend 10 CFR 50.55a, which defines the requirements for applying industry codes and standards to nuclear power plants. Section 50.55a presently requires that nuclear power plant owners (1) construct Class 1, Class 2, and Class 3 components in accordance with the rules provided in the 1989 Edition of Section III, Division 1, ''Requirements for Construction of Nuclear Power Plant Components'' of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code), (2) inspect Class 1, Class 2, and Class 3 components in accordance with the rules provided in the 1989 Edition of Section XI, Division 1, ''Requirements for Inservice Inspection of Nuclear Power Plant Components,'' of the ASME BPV Code with certain limitations and modifications, (3) inspect Class MC (metal containment) and Class CC (concrete containment) components in accordance with the rules provided in the 1992 Edition with the 1992 Addenda of Section XI, Division 1, with certain modifications, and (4) test Class 1, Class 2, and Class 3 pumps and valves in accordance with the rules provided in the 1989 Edition of Section XI, Division 1, of the ASME BPV Code with certain limitations and modifications. Every 120 months licensees are required to update their ISI and IST programs to meet the version of Section XI incorporated by reference into § 50.55a and in effect 12 months prior to the start of a new 120-month interval.

The NRC proposes to amend 10 CFR 50.55a to 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, 1989 Addenda through the 1996 Addenda, for Class 1, Class 2, and Class 3 components. Six proposed limitations to the implementation of Section III are included which address the issues of engineering judgement, Section III materials, weld leg dimensions, seismic design, quality assurance, and independence of inspection. A modification has been included addressing the applicable Code version for new construction.

For ISI, the proposed amendment would require licensees to implement Section XI, Division 1, of the ASME BPV Code, 1995 Edition with the 1996 Addenda, for Class 1, Class 2, and Class 3. Five proposed limitations to the implementation of Section XI are included which address the issues of engineering judgement, quality assurance, Class 1 piping, Class 2 piping, and reconciliation of replacement items. Licensees would be permitted to implement Code Case N–513 which addresses flaws in low and moderate energy Class 3 piping, and Code Case N–523 which addresses the temporary use of mechanical clamps in Class 2 and 3 piping. Licensees would also be permitted to implement Subsection IWE and Subsection IWL, 1995 Edition with the 1996 Addenda.

The proposed rule would expedite implementation of Appendix VIII, ''Performance Demonstration for Ultrasonic Examination Systems,'' to Section XI, Division 1. Three proposed modifications to the implementation of Appendix VIII are included to address the issues of personnel qualification, specimen set cracks, and specimen set microstructure. An expedited implementation schedule would also be required for a proposed modification to Section XI which addresses volumetric examination of the Class 1 high pressure safety injection (HPSI) system in pressurized water reactors (PWRs).

For IST, the proposed amendment would require licensees to implement the 1995 Edition with the 1996 Addenda of the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) for Class 1, Class 2, and Class 3 pumps and valves. 10 CFR 50.55a has been clarified with respect to which pumps and valves are to be included in a licensee's IST program. A proposed limitation is included which addresses the issue of quality assurance (QA). A proposed modification to the implementation of the OM Code is included which addresses stroke time testing. Licensees would be permitted to implement Code Case OMN–1 with one modification in lieu of stroke time testing. In addition, Appendix II to the OM Code is an alternative to the check valve condition monitoring program provisions contained in Subsection ISTC of the OM Code. Three proposed modifications to the implementation of Appendix II are included which supplement the appendix check valve condition monitoring program. Licensees would be permitted to use Subsection ISTD for the IST of snubbers. Finally, based upon supporting information received since the last rulemaking, the modification presently in § 50.55a for containment isolation valve inservice testing has been deleted.

The mechanism for endorsement of the ASME standards, which has been used since the first endorsement in 1971, has been to incorporate by reference the ASME BPV Code rules into § 50.55a. The regulation identifies which editions and addenda of the BPV Code have been approved for use by the NRC. On August 6, 1992 (57 FR 34666), the NRC published a final rule in the **Federal Register** to amend 10 CFR Part 50, ''Domestic Licensing of Production and Utilization Facilities.'' This final rule amended § 50.55a to incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, and the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, of the BPV Code, with specified modifications. The amendment imposed an augmented examination of reactor vessel shell welds. The amendment also separated the requirements for IST of pumps and valves from those for ISI of other components by placing the requirements for inservice testing in a

separate paragraph. For IST of pumps and valves, the regulation, through its incorporation by reference of the 1989 Edition of Section XI, endorsed Part 1, ''Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices,'' Part 6, ''Inservice Testing of Pumps in Light-Water Reactor Power Plants,'' and Part 10, ''Inservice Testing of Valves in Light-Water Reactor Power Plants,'' of ASME/ANSI OMa–1988 to ASME/ANSI OM–1987.

On August 8, 1996 (61 FR 41303), the NRC published a final rule in the **Federal Register** to amend 10 CFR 50.55a to incorporate by reference for the first time ASME Section XI, Division 1, 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.'' Subsection IWE provides criteria for visual inspection of the surface of metal containments, the steel liners of concrete containments, pressure-retaining bolts, and seals and gaskets. Subsection IWL provides criteria for visual inspection of concrete pressure-retaining shells and shell components and for the examination of unbonded post-tensioning systems.

2. Summary of Proposed Revisions to § 50.55a

The revisions to § 50.55a which would result from adoption of the 1989 Addenda through the 1996 Addenda have been divided into three groups based on the proposed implementation schedule (i.e., 120-month update, expedited, and voluntary). For each of these groups, it is indicated 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 discussion of the proposed revisions.

2.1 List of Each Revision and Implementation Schedule

120-Month Update [in accordance with § 50.55a(g)(4)(i) and § 50.55a(f)(4)(i)]
Section XI (Not A Backfit)
Class 1, 2, and 3 Components, Including Supports
Limitations
Engineering Judgement
Quality Assurance
Class 1 Piping
Class 2 Piping
Reconciliation of Quality Requirements
OM Code (Not A Backfit)
Class 1, 2, and 3 Pumps and Valves
Clarification of Safety-Related Valves
Limitation
Quality Assurance
Modification
Stroke Time Testing
Expedited Implementation [after 6 months from the date of the final rule—Backfit]
Section XI
Appendix VIII (including three modifications)
Personnel Qualification
Specimen Set Cracks
Specimen Set Microstructure
Class 1 Piping Volumetric Examination
Voluntary Implementation [may be used when final rule published]
Section III (Not A Backfit)
Class 1, 2, and 3 Components
Limitations
Engineering Judgement
Section III Materials
Weld Leg Dimensions
Seismic Design
Quality Assurance
Independence of Inspection
Modification
Applicable Code Version for New Construction
Section XI (Not A Backfit)
Subsections IWE and IWL, 1995 Edition with the 1996 Addenda
Flaws in Class 3 Piping; Mechanical Clamping Devices
Limitation on Scope
OM Code (Not A Backfit)
Code Case OMN–1
Limitation on Length of Test Interval
Appendix II (including three modifications)
Valve Opening and Closing Functions
Limitation of Length of Initial Test Interval
Condition Monitoring Program
Subsection ISTD
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) together with § 50.55a(g)(4) of the proposed rule would require that licensees implement the 1995 Edition with the 1996 Addenda of Section XI, Division 1, for Class 1, Class 2, and Class 3 components and their supports. Five proposed limitations would be included to address NRC positions on the use of Section XI.

2.3.1.2 Limitations

2.3.1.2.1 Engineering Judgement

The first proposed limitation to the implementation of Section XI would address 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. Proposed paragraph 50.55a(b)(2)(xi) would require that when a licensee relies on engineering judgement for activities or evaluations of components or systems within the scope of § 50.55a that are not directly addressed by the BPV Code, the licensee must receive NRC approval for those activities or evaluations pursuant to 10 CFR 50.55a(a)(3).

2.3.1.2.2 Quality Assurance

The second proposed limitation to the implementation of Section XI pertains to the use of NQA–1 with Section XI. Section XI references the use of either NQA–1 or the Owner's Appendix B Quality Assurance Program (10 CFR Part 50, Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants'') as part of its individual requirements for a QA program. At present, § 50.55a endorses the 1989 Edition of the ASME Code which references NQA–1–1979 for Section XI. The 1996 Addenda of the ASME Code references NQA–1–1992 for Section XI.

The NRC has reviewed the requirements of NQA–1, 1986 Addenda through the 1992 Addenda, that are part of the incorporation by reference of Section XI, and has determined that by itself, NQA–1 would not adequately describe how to satisfy the requirements of 10 CFR Part 50, Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' since there are various aspects of operational phase QA and administrative controls which are not addressed by NQA–1.

10 CFR 50.34(b)(6)(ii) requires that ''The information on the controls to be used for a nuclear power plant or a fuel reprocessing plant shall include a discussion of how the applicable requirements of Appendix B will be satisfied.'' This information is required to be submitted to the NRC as part of the Final Safety Analysis Report (FSAR). Standard Review Plan (SRP) 17.2, ''Quality Assurance During the Operations Phase,'' states that ''The QA program description presented in the FSAR must discuss how each criterion of Appendix B will be met.'' Further, the SRP states ''The acceptance criteria include a commitment to comply with the regulatory positions presented in the appropriate issue of the Regulatory Guides including the requirements of ANSI Standard N45.2.12 and the Branch

Technical Position listed in subsection V of SRP Section 17.1. Thus, the commitment constitutes an integral part of the QA program description and requirements.'' The NRC has determined that the provisions of NQA–1, 1986 Addenda through the 1992 Addenda, would not satisfy the criteria specified in SRP 17.2 for describing how the requirements of Appendix B will be satisfied for operational activities. There are numerous areas where American National Standards Institute (ANSI) standards or NRC regulatory positions, which have been long-standing cornerstones of an Owner's QA Program, are either nonmandatory or missing altogether from the NQA–1 provisions. However, the Owner's Section XI QA Program, which has been approved by the NRC, is adequate. Thus, the Commission has determined that the requirements of NQA–1, 1986 Addenda through the 1992 Addenda, are acceptable for use in the context of Section XI, as permitted by IWA–1400, provided the licensee utilizes its 10 CFR Part 50, Appendix B, QA program in conjunction with Section XI. Changes to a licensee's QA program shall be made in accordance with 10 CFR 50.54(a). Further, where NQA–1 and Section XI do not address the commitments contained in the licensee's Appendix B QA program description, such commitments shall be applied to Section XI activities. Proposed § 50.55a(b)(2)(xii) contains the requirement addressing licensee's commitments related to Section XI.

2.3.1.2.3 Class 1 Piping

The third proposed limitation to the implementation of Section XI would require licensees to use the rules for 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. These later Code addenda contain provisions of Code 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–324, ''Examination Requirements for Integrally Welded or Forged Attachments to Class 2 Piping at Containment Penetrations;'' which were found to be unacceptable. Because the NRC had previously determined the Code cases to be unacceptable, they were not endorsed in any revision of Regulatory Guide 1.147, ''Inservice Inspection Code Case Acceptability—ASME Section XI, Division 1.'' 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 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–324 were determined to be unacceptable because some important piping was exempted from inspection. Access difficulties was the basis in the Code cases for exempting these areas from examination, but 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.'' Thus, proposed § 50.55a(b)(2)(xiii) would 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, contained in § 50.55a(b)(2)(xiv), would confine 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,'' 1989 Addenda through the 1996 Addenda. 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 had previously determined that the provisions of the Code Case were acceptable if the licensee defined the Class 2 piping subject to volumetric and surface examination and received approval prior to implementation. 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. Thus, the requirements contained in IWC–1220, IWC–1221, and IWC–1222, 1989 Addenda through the 1996 Addenda, for determining the components subject to examination and establishing examination requirements for Class 2 piping may be used if the licensee defines the Class 2 piping subject to volumetric and surface examination, and submits this information to the NRC for approval pursuant to § 50.55a(a)(3).

2.3.1.2.5 Reconciliation of Quality Requirements

The fifth proposed limitation to the implementation of Section XI addresses reconciliation of replacement items [§ 50.55a(b)(2)(xx)(A)] and the definition of Construction Code [§ 50.55a(b)(2)(xx)(B)]. Changes to IWA–4222, ''Reconciliation of Owner's Requirements,'' in the 1995 Addenda would permit a replacement item produced at a facility not having a 10 CFR Part 50, Appendix B qualified program to be used in safety-related applications. With regard to the definition of Construction Code, a new definition of Construction Code appeared in IWA–9000, ''Glossary,'' in the 1993 Addenda. Due to the changes made in IWA–4200 in the 1995 Addenda, the change in definition could result in standards being utilized which do not contain any QA requirements, or contain QA requirements that do not fully comply with Appendix B. Thus, when implementing the 1995 Addenda through the 1996 Addenda, § 50.55a(b)(2)(xx)(A) would require reconciliation of replacement items to the original QA requirements. Section 50.55a(b)(2)(xx)(B) would require a licensee to reconcile replacement items to the Construction Code and to the QA requirements as described in the Owner's QA program.

Section XI Article IWA–4000 provides rules and requirements for the repair and replacement of pressure retaining components and their supports. Versions of IWA–4000 previous to the 1995 Addenda permitted a licensee to purchase a replacement item to the standards of the original Construction Code or a later version, provided that the technical requirements of an item such as design and fabrication, as well as the nontechnical requirements (identified as administrative requirements in IWA–4222) such as QA

and Authorized Inspection of the later version were reconciled with those of the original Construction Code and Owner's Requirements. Reconciliation ensures that the replacement item meets certain standards of quality so that it is satisfactory for the specified design and operating conditions. In the 1995 Addenda, the provisions of Code Case N–554, ''Alternative Requirements for Reconciliation of Replacement Items,'' were incorporated into an extensive rewrite of IWA–4200. As a result of these changes to IWA–4200, specifically IWA–4222(a)(2), the nontechnical requirements for Class 1, 2, and 3 safety-related replacement items would no longer need to be reconciled which may result in noncompliance with 10 CFR Part 50, Appendix B. NRC regulations require that any item which performs a safety-related function must meet Appendix B. Appendix B invokes, among other things, controls on suppliers of safety-related items. By not requiring reconciliation of the administrative requirements, the provisions in IWA–4222(a)(2) of the 1995 Addenda through the 1996 Addenda, would allow vendors having a QA program which does not meet Appendix B to be utilized, and may result in noncompliance with Appendix B. These deficiencies could be resolved if the Code provided for commercial grade item dedication in accordance with 10 CFR Part 21, ''Reporting of Defects and Noncompliance.'' However, IWA–4222 does not address commercial grade dedication. In addition, it should be pointed out that a separate Code Case which provides an alternative for a specific provision in IWA–4200, Code Case N–567, ''Alternative Requirements for Class 1, 2, and 3 Replacement Components,'' was modified to require the reconciliation of nontechnical requirements before the Code Case was approved. Therefore, an inconsistency exists between the Code and a Code Case. Thus, when implementing the 1995 Addenda through the 1996 Addenda, § 50.55a(b)(2)(xx)(A) would require reconciliation of replacement items to the original QA requirements.

The provisions of the Code in IWA–4222(a)(2) discussed above address newly manufactured replacement parts. A further limitation on the use of Article IWA–4200 in the 1995 Addenda through the 1996 Addenda is contained in § 50.55a(b)(2)(xx)(B). IWA–4222(b) addresses the use of items from a facility which was shutdown or for which construction was halted. IWA–4222(b) permits the use of either the administrative requirements of the Construction Code of the item being replaced or the administrative requirements of the Construction Code of the item being used for replacement. However, the definition of ''Construction Code'' was changed in the 1993 Addenda. In versions of Section XI previous to the 1993 Addenda, Construction Code was defined in IWA–9000, ''Glossary,'' as ''the body of technical requirements that governed the construction of the item.'' Included in the body of technical requirements that governed the construction of the item was a requirement to reconcile the Owner's specification requirements, which included NRC regulatory requirements, and applicable Owner design and procurement specifications that invoke technical and nontechnical requirements (e.g., 10 CFR Part 50, Appendix B). In the 1993 Addenda, the definition became nationally recognized Codes such as ASME, Specifications such as the American Society of Testing and Materials (ASTM), and designated Code Cases. Either definition of Construction Code would include the original Construction Codes for the design and construction of piping, such as B31.1, ''Power Piping,'' and B31.7, ''Nuclear Piping,'' and those for the design and construction of storage tanks, such as the American Petroleum Institute (API) 620, ''Design and Construction of Large, Welded, Low-Pressure Storage Tanks,'' and API 650, ''Welded Steel Tanks for Oil Storage.'' However, many of these standards utilized for construction do not contain any QA requirements, or they contain QA requirements that do not fully comply with Appendix B. Therefore, in order to satisfy Appendix B, QA requirements similar to or meeting Appendix B were invoked in the Owner's original procurement documents. Thus, when implementing IWA–4200 (including subparagraphs IWA–4221, IWA–4222, IWA–4223, IWA–4224, and IWA–5224), § 50.55a(b)(2)(xx)(B) would require a licensee to reconcile replacement items to the Construction Code and to the QA requirements as described in the Owner's QA program.

2.3.2 OM Code (120-Month Update)

2.3.2.1 Class 1, 2, and 3 Pumps and Valves

The proposed amendment to § 50.55a(f)(4) would require that IST of pumps and valves be performed in accordance with the ASME ''Code for Operation and Maintenance of Nuclear Power Plants'' (OM Code). A proposed new section, § 50.55a(b)(3), would specify the editions and addenda of the OM Code that have been incorporated by reference into § 50.55a. Paragraph 50.55a(b)(3) together with § 50.55a(f)(4) of the proposed rule would require that licensees implement the 1995 Edition with the 1996 Addenda of the OM Code. Existing § 50.55a(f)(1) has been modified to clarify which pumps and valves are to be included in the IST program. One proposed limitation to implementation of the OM Code addressing QA, and one proposed modification of the OM Code addressing stroke time testing have been included.

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 IWP and IWV were last incorporated by reference into § 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). The ASME Board on Nuclear Codes and Standards has transferred responsibility for 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 ASME 1990 Edition of the 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 IST of snubbers is governed by plant technical specifications and, thus, has never been included in § 50.55a. Therefore, this proposed rule only requires implementation of Subsections ISTA, ISTB, and ISTC. However, § 50.55a(b)(3)(v) would permit licensees to implement Subsection ISTD of the 1996 Addenda by making a change to their technical specifications in accordance with applicable NRC requirements.

2.3.2.3 Clarification of Safety-Related Valves

The existing § 50.55a(f)(1) has been interpreted by some licensees to mean that all safety-related pumps and valves regardless of ASME Code Class (or equivalent) were to be included in the IST program. The NRC proposes to modify this paragraph to clarify that the provisions of § 50.55a(f)(1) apply only to pumps and valves in steam, water, air, and liquid radioactive waste systems that perform a function to shut down the reactor, maintain the reactor in a safe shutdown condition, mitigate the consequences of an accident, or provide overpressure protection for such systems.

2.3.2.4 Limitation

2.3.2.4.1 Quality Assurance

The limitation to the implementation of the OM Code pertains to the use of NQA–1, ''Quality Assurance Requirements for Nuclear Facilities,'' with the OM Code. The OM Code references the use of either NQA–1 or the Owner's Appendix B Quality Assurance Program as part of its individual requirements for a QA program. At present, § 50.55a endorses NQA–1–1979 for the OM Code. The 1996 Addenda also endorses NQA–1–1979. Thus, the 1996 OM Code has not endorsed a later version of NQA–1. Because this rulemaking would incorporate the OM Code by reference into § 50.55a for the first time, a limitation is included to address the same issues discussed previously in the Section XI section on QA.

The NRC has determined that the provisions of NQA–1, 1979 Addenda, would not adequately describe how to satisfy the requirements of Appendix B as satisfied by § 50.34(b)(6)(ii). Further, there are various aspects of operational phase QA and administrative controls which are not addressed by NQA–1. There are numerous areas where American National Standards Institute (ANSI) standards or NRC regulatory positions, which are specified in SRP 17.2, are either nonmandatory or missing altogether from the NQA–1 provisions. However, the Owner's QA Program, which has been approved by the NRC, is adequate. Thus, the NRC has determined that the requirements of NQA–1–1979, that are part of the incorporation by reference of the OM Code, is acceptable for use in the context of the OM Code, as permitted by ISTA 1.4, provided the licensee utilizes its 10 CFR Part 50, Appendix B, QA program in conjunction with the OM Code. Changes to licensee's QA program shall be made in accordance with 10 CFR 50.54. Further, where NQA–1 and the OM Code do not address the commitments contained in the licensee's Appendix B QA program description, such commitments shall be applied to OM Code activities. Proposed § 50.55a(b)(3)(i) addresses licensee's commitments related to the OM Code.

2.3.2.5 Modification

2.3.2.5.1 Stroke Time Testing

Proposed § 50.55a(b)(3)(ii) would require that the stroke time testing requirement of Subsection ISTC of the OM Code applicable for motor-operated valves (MOVs) be supplemented with programs that licensees have previously committed to perform, prior to issuance of this amendment to § 50.55a, for demonstrating the design basis capability of MOVs. Stroke time testing of MOVs has been specified in ASME Section XI and is currently required by § 50.55a(f). This same testing is required by the OM Code. This testing is a useful tool and complements other tests used to verify MOV function. Variation in measured stroke times can indicate valve degradation. Additionally, periodic stroking provides valve exercise and some measure of on-demand reliability. However, as discussed in NRC Generic Letter (GL) 89–10 ''Safety-Related Motor-Operated Valve Testing and Surveillance'' dated June 28, 1989, it is now recognized that the stroke time testing alone is not sufficient to provide assurance of MOV capability under design-basis conditions.

Subsequent to licensees implementing programs pursuant to GL 89–10, the NRC issued Generic Letter 96–05, ''Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,'' on September 18, 1996. This generic letter requested licensees to establish a program, or to ensure the effectiveness of their current program, to verify on a periodic basis that safety-related motor-operated valves continue to be capable of performing their safety functions within the current licensing bases of the facility. Prior to issuance of this rule, licensees have made licensing commitments pursuant to GL 96–05 that have been reviewed by the NRC staff. Most licensees have committed to participate in the Joint Owners Group (JOG) Program on MOV Periodic Verification. The JOG program includes three phases: (1) licensees will establish an interim static diagnostic testing program developed by JOG with a test frequency based on margin and safety significance; (2) JOG will coordinate a dynamic testing program over the next 5 years that includes approximately 150 MOVs with participating licensees each testing a few MOVs three times over this interval; and (3) based on the results of the dynamic testing program, JOG will establish a long-term periodic test program. Proposed § 50.55a(b)(3)(ii) would require that licensees supplement the stroke time testing requirements of the OM Code with these commitments.

2.4 Expedited Implementation

2.4.1 Appendix VIII

The proposed rule would require 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 would be included to address NRC positions on the use of Appendix VIII. Licensees would be required 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. The proposed rule would not require any change to a licensee's ISI schedule for examination of these components, but would require that the provisions of Appendix VIII be used for all examinations after that date rather than the ultrasonic testing (UT) procedures and personnel requirements presently being utilized by licensees.

Appendix VIII provides the requirements for performance demonstration for ultrasonic testing (UT) procedures, equipment, and personnel used to detect flaws and size flaws. Its requirements are applicable to all UT performed for Class 1, Class 2, and Class 3 items (i.e., reactor vessel, nozzles, piping, and bolting and studs). These requirements are also to be utilized when implementing the augmented inservice inspection program for reactor vessel shell welds presently required by § 50.55a(g)(6)(ii)(A). The NRC has reviewed the 1995 Edition with the 1996 Addenda of Appendix VIII and has determined that the provisions contained in this appendix should be used with three modifications (addressed below). This mandatory appendix would normally be adopted as part of the routine 120-month update specified in § 50.55a(g)(4), but because of the importance of the Appendix VIII program, the NRC has determined that its requirements should be implemented after 6 months from the date of the final rule. The performance demonstration requirements in Appendix VIII would

substantially improve the ability of an examiner to detect and characterize flaws in examined components. UT procedures and personnel requirements are presently contained in Section XI but, as detailed in the documented evaluation required by § 50.109(a)(4), personnel qualified to Appendix VIII are significantly better at detecting flaws. The industry's Performance Demonstration Initiative (PDI) established a process in accordance with Appendix VIII for reactor vessel, nozzle, piping, and bolting examinations. PDI has received considerable support from the industry, and every licensee has contributed financially. The majority of the cost of PDI was in setting up the samples, which has been completed. Proposed § 50.55a(g)(6)(ii)(C)(*1*) would require licensees to utilize the improved requirements in Appendix VIII for all examinations of reactor vessels (including nozzles), piping, and bolting performed after 6 months from the date of the final rule. To date, the PDI program has qualified over 300 individuals for piping and five teams for vessel examinations. Thus, the NRC does not believe that a 6-month implementation period would result in hardship.

2.4.1.1 Modifications

2.4.1.1.1 Appendix VIII Personnel Qualification

The first proposed modification of Appendix VIII relates 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 first appeared in Section XI in the 1988 Addenda and was incorporated by reference into § 50.55a in a final rule published on August 6, 1992 (57 FR 34666). The NRC believes that the requirement in Appendix VII–4240 for personnel to receive a minimum of 10 hours of training on an annual basis is inadequate. Proposed § 50.55a(b)(2)(xvii) would require that all personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII receive 40 hours of annual training which includes 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. In addition, these studies have shown that this capability begins to diminish within approximately 6 months if skills are not maintained. Thus, 10 hours of annual training is not sufficient practice to maintain skills. The NRC believes that a minimum of 40 hours of annual training, not 10 hours, is required to maintain an examiner's abilities in this highly specialized skill area. The NRC expects that licensees would distribute the training over the course of the year to ensure that interpretation skills do not diminish.

2.4.1.1.2 Appendix VIII Specimen Set Cracks

The second proposed modification of Appendix VIII would require 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% of the flaws to be notches. For the purpose of demonstrating nondestructive examination (NDE) capabilities, notches are not realistic representations of service induced cracks. An inspector cannot properly interpret service induced cracks by qualifying with specimens containing notches. Notches are easier to detect than flaws because notches have a higher amplitude and simpler signal characteristics. Notches are easier to interpret and, in fact, the probability of detecting notches can be much higher than the probability of detecting cracks under similar conditions. In addition, Appendix VIII provides a screening test that uses a relatively small sample size containing few flaws. If some of the flaws are replaced by notches that are unrealistic, the screening test becomes ineffective. Because of these considerations, the flaws in the specimen sets utilized for piping by EPRI for the PDI are all cracks. The regulatory analysis for this rulemaking contains a detailed discussion of the importance of using cracks in the specimens. Thus, proposed § 50.55a(b)(2)(xiii) would require that all flaws in the specimen sets used for performance demonstration be cracks.

2.4.1.1.3 Appendix VIII Specimen Set Microstructure

The third proposed modification of Appendix VIII would require 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. Appendix VIII does not distinguish specimens for two-sided examinations from those used for single-sided examination.

Appendix VIII was originally developed using UT lessons learned from two-sided examinations of welds. This UT experience provided the input for designing specimens and selecting, locating, and characterizing flaws. Studies have shown that defect characteristics such as shape, size, depth, tilt angle, skew angle, roughness, and crack tip affect the probability of detecting a particular flaw. For example, it was demonstrated in one particular study (Reference 22 in the documented evaluation) that a particular flaw was over three times more reflective in one direction, thus easier to detect, than in the opposite direction. Specimens designed for two-sided examination may not have defects which are appropriate for single-sided performance demonstration; i.e., the specimens may not adequately test an examiners proficiency in detecting flaws. Therefore, in order to proceed with the effort of qualifying UT systems (equipment, procedures, and personnel) for single-sided examinations, proposed § 50.55a(b)(2)(xx) would require the industry to develop sets of specimens 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.

2.4.2 Generic Letter on Appendix VIII

A draft generic letter was published in the **Federal Register** (61 FR 69120) for public comment on December 31, 1996, 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 generic letter stated that even though the need for improvement clearly existed, the staff had reached the conclusion that immediate backfitting of Appendix VIII in advance of this proposed rulemaking was not warranted. This conclusion was based on consideration of defense-in-depth measures, Code margins in component design, leakage monitoring systems, and also that Appendix VIII was already being applied to selected piping subject to intergranular stress corrosion cracking. The NRC received 16 comment letters on the generic letter.

The comments generally were very similar and can be summarized in the following five items: (1) it is inappropriate to request licensees to voluntarily commit to a program in a

generic letter; (2) the urgency for licensee's to voluntarily commit to implementing Appendix VIII is inconsistent with the statement in the generic letter that a safety concern does not exist that would warrant immediate backfitting in advance of the rulemaking; (3) the performance-based qualification program of Appendix VIII should be approved an alternative to the current ASME Code, and Appendix VIII as implemented by PDI should be recognized as an acceptable alternative for Appendix VIII; (4) the NRC should provide guidance on incorporating Appendix VIII and/or PDI into plant-specific ISI programs; and (5) the generic letter would request that licensees update their UT ISI and augmented inspection commitments to a Code edition not yet referenced in the regulations.

With regard to the first comment, the NRC disagrees that it is inappropriate to request licensees to voluntarily commit to a program in a generic letter. This is one mechanism available to the NRC for alerting licensees, for example, to degraded conditions which may unacceptably affect the function of safety-related components. The second comment takes the generic letter statement out of context. What the generic letter actually stated was that a safety concern did not exist to warrant immediate backfitting in advance of the rulemaking because of defense-in-depth measures, Code margins in design, and that Appendix VIII was already being applied to selected piping subject to intergranular stress corrosion cracking. The NRC strongly disagrees that Appendix VIII and Appendix VIII as implemented by PDI should be alternatives to the present Code rules. As detailed in the documented evaluation for backfitting Appendix VIII, it has been demonstrated that examiners previously considered qualified under Section XI generally have marginal UT skills. This was evident from the discouragingly low percentage of examiners initially satisfying the screening criteria for detecting flaws under the PDI program. Comment four regarding guidance on incorporating Appendix VIII into present ISI programs, and comment five regarding Code edition are automatically resolved in a rulemaking format.

At the time the generic letter was issued, this proposed rulemaking was still under development. The purpose of the generic letter was to alert the industry to the (1) generally poor performance in detecting flaws and (2) the Commission's intent to endorse Appendix VIII via rulemaking. Publication of a final rule would obviate the need for the generic letter.

2.4.3 Class 1 Piping Volumetric Examination

A proposed modification of Section XI would require licensees of pressurized water reactor plants to supplement the surface examination of Class 1 High Pressure Safety Injection Systems (HPSI) 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 is 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, and (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. Thus, not only are the requirements between Class 1 and Class 2 inconsistent (with the Class 1 portions being subject to less stringent testing requirements as compared with Class 2 portions of the same type of piping), but operating experience has shown that these reactor coolant pressure boundary (RCPB) pipe examinations need to be more comprehensive. Proposed § p50.55a(b)(2)(xv) would require licensees to supplement the Section XI required surface examination for the Class 1 portion of the HPSI system with volumetric examination in order to ensure the integrity of the reactor coolant pressure boundary as required by General Design Criteria (GDC) 14, 10 CFR Part 50, Appendix A, or similar provisions in the licensing basis for these facilities, and Criteria II and XVI of 10 CFR Part 50, Appendix B. Licensees would be required to perform the volumetric examination during any ISI program inspection of the HPSI system performed after 6 months from the date of the final rule. Utilization of licensee's existing ISI schedules will 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.

2.5 Voluntary Implementation

2.5.1 Section III

The NRC has reviewed the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, and 1996 Addenda of Section III, Division 1, for Class 1, Class 2, and Class 3 components, and has determined that they are acceptable for voluntary use with six proposed limitations. In addition, § 50.55a would be modified to ensure consistency between § 50.55a and NCA–1140.

The version of Section III utilized by licensees is chosen prior to construction. Section 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 § 50.55a. Accordingly, the proposed limitations to Section III become effective only when a licensee voluntarily updates to a later version. The modification would only apply to a applicant for a new construction permit.

2.5.1.1 Limitations

2.5.1.1.1 Engineering Judgement

The first proposed limitation to the implementation of Section III would establish an NRC restriction 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 construction activities not specifically considered by the Code. Proposed paragraph 50.55a(b)(1)(i) would require that when a licensee relies on engineering judgement for activities or evaluations of components or systems within the scope of § 50.55a that are not directly addressed by the BPV Code, the licensee must receive NRC approval for those activities or evaluations pursuant to § 50.55a(a)(3).

2.5.1.1.2 Section III Materials

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

2.5.1.1.3 Weld Leg Dimensions

The third proposed limitation to the implementation of Section III would

correct a conflict in the design and construction requirements in Subsection NB (Class 1 Components), 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, to ensure consistency, proposed § 50.55a(b)(1)(iii) would require that licensees use the 1989 Edition for the above referenced paragraphs and figures in lieu of the 1989 Addenda through the 1996 Addenda.

2.5.1.1.4 Seismic Design

The fourth proposed limitation to the implementation of Section III pertains to new requirements for piping design evaluation contained in the 1994 Addenda through the 1996 Addenda of the BPV Code. The NRC has determined that changes to subarticles 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) are unacceptable. The new requirements are based on the premise that loads such as earthquake loads are not capable of producing collapse or gross distortion of a component. The requirements, in part, are based on General Electric evaluations of the test data performed under sponsorship of the Electric Power Research Institute (EPRI) and the NRC. However, 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. Thus, in proposed § 50.55a(b)(1)(iv), licensees would be permitted to use articles NB–3200, NB–3600, NC–3600, and ND–3600, in the 1989 Addenda through the 1993 Addenda, but would be prohibited from using these requirements 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 pertains to the use of NQA–1, ''Quality Assurance Requirements for Nuclear Facilities,'' with Section III. 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.'' At present, § 50.55a endorses the 1989 Edition of the ASME Code which references NQA–1–1986 for Section III. The 1996 Addenda of the ASME Code references NQA–1–1992 for Section III.

The NRC has reviewed the requirements of NQA–1, 1986 Addenda through the 1992 Addenda, that are part of the incorporation by reference of Section III, and has 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. NQA–1 by itself would not adequately describe how to satisfy the requirements of 10 CFR Part 50, Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.'' 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 Appendix B (i.e., the provisions of Section III integrated with NQA–1). Because licensees may voluntarily choose to apply later provisions of Section III, proposed § 50.55a(b)(1)(v) contains a limitation 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.

2.5.1.1.6 Independence of Inspection

The sixth proposed limitation to the implementation of Section III would prohibit licensees from using subparagraph NCA–4134.10(a), ''Inspection,'' in the 1995 Edition through the 1996 Addenda. Prior to this edition and addenda, NCA–4134.10(a) required that the provisions of NQA–1, ''Quality Assurance Program Requirements for Nuclear Facilities,'' Basic Requirement 10, ''Inspection,'' and Supplement 10S–1, ''Supplementary Requirements for Inspection,'' be utilized without exception. In the 1995 Edition, NCA–4134.10(a) was modified so that paragraph 2 of Supplement 10S–1 and the requirements for independence of inspection were no longer required. Supplement 10S–1, 2.1, states that ''Inspection Personnel shall not report directly to the immediate supervisors who are responsible for performing the work being inspected.'' Subparagraph 2.2 states ''Each person who verifies conformance of work activities for purposes of acceptance shall be qualified to perform the assigned task.'' By exempting Supplement 10S–1 paragraph 2 from the requirements of NCA–4134.10, Section III could promote noncompliance with 10 CFR 50, Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' Criterion 1, ''Organization.'' 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. Thus, in proposed § 50.55a(b)(1)(vi), licensees would be permitted to use the provisions contained in NCA–4134.10(a), in the 1989 Addenda through the 1994 Addenda, but would be prohibited from using these provisions 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 proposed modification of Section III addresses a possible conflict between NCA–1140 and § 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 nuclear power plant be no greater than three years. Paragraph 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 § 50.55a. In order to resolve this possible discrepancy, the NRC proposes 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.

2.5.2 Section XI (Voluntary Implementation)

Licensees would be permitted to update from the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL to the 1995 Edition with the 1996 Addenda. In addition, licensees could implement 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

Many of the provisions in Section XI Subsection IWL, ''Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,'' pertaining to the inspection of the tendons of concrete containments were based on guidance contained in Regulatory Guide 1.35, ''Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments.'' A final rule published on August 8, 1996 (61 FR 41303) incorporated 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. At that time, there were several key positions in the regulatory guide addressing the trending of prestress losses, unanticipated tendon elongation, grease leakage, and excessive water in the sampled sheathing filler grease not addressed in Subsection IWL because the ASME Code committees had not yet completed consideration of these positions. Due to the importance of these positions, the final rule addressed them in paragraphs 50.55a(b)(2)(ix)(A) through 50.55a(b)(2)(ix)(D)(3). In addition, the final rule contained § 50.55a(b)(2)(ix)(E) which addressed the occurrence of degradation in inaccessible areas of containments.

Since publication of the 1992 Addenda, the ASME Code committees have completed their consideration of those regulatory guide positions. Most have been incorporated into subsequent edition and addenda, and the 1995 Edition with the 1996 Addenda addresses all of the modifications listed above except grease leakage and degradation in inaccessible areas. Thus, licensees would be required to utilize the modifications presently in § 50.55a addressing grease leakage and degradation in inaccessible areas. The NRC has determined that the provisions contained in Subsection IWE and Subsection IWL, 1995 Edition with the 1996 Addenda Code, in conjunction with the modifications, would be acceptable.

The final rule published on August 8, 1996 (61 FR 41303) incorporated Subsection IWE and Subsection IWL into § 50.55a for the first time. The final rule contained a requirement for licensees to develop and implement a containment ISI program within five years. Each plant had a pre-existing ISI program to address Class 1, Class 2, and Class 3 components. The rule left it to the licensee's discretion whether to have two separate ISI programs, or merge the containment ISI program with the pre-existing program.

It has been over a year since the final rule was issued, and some licensees have begun the development of a containment ISI program to comply with the required 5-year implementation period. This containment ISI program will be based on the 1992 Edition with the 1992 Addenda as required by the final rule. However, other licensees have indicated that they will request NRC approval pursuant to § 50.55a(a)(3) to use later editions and addenda of Subsection IWE and Subsection IWL before this proposed rule becomes final. Thus, to provide flexibility, § 50.55a(b)(2)(vi) has been modified. Licensees would be permitted to implement either the presently required 1992 Edition with the 1992 Addenda, or the latest containment examination provisions; i.e., 1995 Edition with the 1996 Addenda.

For those licensees implementing the 1992 Edition with the 1992 Addenda, all of the modifications contained in paragraphs 50.55a(b)(2)(ix)(A) through 50.55a(b)(2)(ix)(D)(3) must be applied as presently required by § 50.55a. Licensees wishing to implement the 1995 Edition with the 1996 Addenda would be required to apply paragraphs 50.55a(b)(2)(ix)(A), 50.55a(b)(2)(ix)(D)(3), and 50.55a(b)(2)(ix)(E). Paragraph § 50.55a(b)(2)(ix) would thus be modified. According to § 50.55a(g)(6)(ii)(B)(1), the containment examinations performed during the 5-year implementation period are those examinations which are required by Subsection IWE during the first period of what will be the first containment inspection interval. (Since Subsection IWL is based on a 5-year schedule, standard Section XI periods do not apply for the examination of concrete containments and their post-tensioning systems). With completion of the first period examinations, the second period of the first containment ISI interval would begin. The end of the third period completes the first containment ISI interval, a containment ISI 120-month update has been completed, and the second containment ISI interval would begin.

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. According to § 50.55a(g)(6)(ii)(B)(2) of the final rulemaking, the first examination of concrete may be performed at any time between September 9, 1996, and September 9, 2001. The date of the first examination of concrete is not conditional upon compliance with Subsection IWL–2410 or the SIT. The purpose of the italicized words is to maintain the present 5-year schedule for examination of the post-tensioning system as operating plants 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 five-year Subsection IWL ISI period 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 ISI schedule.

2.5.2.2 Flaws in Class 3 Piping

Proposed § 50.55a(b)(2)(xvi) would permit licensees to use 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.'' Section XI contains repair methods for pipes with a flaw exceeding acceptable limits. These repairs restore the integrity of the flawed piping. There are certain cases, however, 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 effect the Code repair). For many safety-related piping systems, immediate repair is required regardless of plant status. However, it has been determined that under certain conditions, temporary acceptance of flaws, including through-wall leaking, of low and moderate energy Class 3 piping is acceptable provided that the conditions are met, and the repair is effected during the next outage. At present, licensees must request NRC staff approval to defer Section XI Code repair for these Class 3 moderate energy (200 xF, 275 psig) piping systems. The NRC has reviewed Code Case N–513 and Code Case N–523–1 and has determined that Code Case N–523–1 is acceptable. Code Case N–513 is acceptable except for the scope and Section 4.0.

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, original construction deficiencies, and pressure boundary leakage; applications for which the criteria should not be utilized. Thus, the NRC has determined that the Code Case shall not be applied to: (1) components other than pipe and tube, such as pumps, valves, expansion joints, and heat exchangers; (2) the discovery and repair of flaws or deficiencies remaining from original construction; (3) leakage through a flange gasket; (4) threaded connections employing nonstructural seal welds for leakage prevention (through seal weld leakage is not a structural flaw, thread integrity must be maintained); and (5) degraded socket welds. A proposed limitation would be added in § 50.55a(b)(2)(xvi)(B) which would preclude the use of Code Case N–513 for these applications.

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. No criteria or guidance is given for justifying a reduction, or limiting the amount of reduction. The acceptance criteria of the first paragraph are based on sound principles. The second paragraph would allow ever finer calculation until the available margins became unacceptably low. A limitation would be added in proposed § 50.55a(b)(2)(xvi)(A) requiring that when implementing Code Case N–513, the specific safety factors in the first paragraph of Section 4.0 be satisfied. The use of Code Case N–513, with the limitations, and Code Case N–523–1 would obviate the need for licensees to request approval for deferring repairs, thus saving NRC and licensee resources.

2.5.3 OM Code (Voluntary Implementation)

Licensees would be permitted to implement Code Case OMN–1 in lieu of stroke time testing as required in Subsection ISTC. Licensees would also be permitted to implement Appendix II as an alternative to the condition monitoring program provisions contained in Subsection ISTC. However, licensees choosing to implement Appendix II would be required to apply the three proposed modifications to Appendix II to supplement check valve condition monitoring. In addition, licensees would be permitted to use Subsection ISTD for the IST of snubbers.

2.5.3.1 Code Case OMN–1

An alternative to the provisions contained in § 50.55a(b)(3)(ii) is included in proposed § 50.55a(b)(3)(iii) which would permit licensees to voluntarily implement ASME Code Case OMN–1, ''Alternative Rules for Preservice and Inservice Testing of Certain Electric Motor Operated Valve Assemblies in LWR Power Plants.'' The NRC has determined that for motor-operated valves, 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. 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. The NRC supports the OMN–1 maximum motor-operated valve test interval of 10 years based on current knowledge and experience, but believes it prudent to require that licensees evaluate the information obtained for each motor-operated valve during the first five years of use of the Code case, or three refueling outages (whichever is longer) to validate assumptions made in justifying a longer test interval. These limitations on the use of OMN–1 would be added to the rule as a modification in § 50.55a(b)(3)(iii)(A). Thus, Code Case OMN–1 is acceptable in lieu of Subsection ISTC, other than leakage rate testing requirements, with the modification that five years or three refueling outages (whichever is longer) from initial implementation of Code Case OMN–1, the adequacy of the test interval for each motor-operated valve must be evaluated and adjusted as necessary.

In addition, as noted in GL 96–05, licensees are cautioned when implementing Code Case OMN–1 that 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.

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. However, upon reviewing the appendix, the NRC has determined that the requirements in Appendix II must be supplemented. The first area that the NRC believes requires supplementation is the demonstration of acceptable valve performance. Appendix II requires no testing or examination of the check

valve obturator movement to both the open and closed positions. Testing or examination of the check valve obturator in one direction only cannot assure the unambiguous detection of a functionally degraded check valve. The valve obturator must be tested or examined in both the opening and closing directions to assess its condition and confirm acceptable performance. Proposed § 50.55a(b)(3)(iv)(A) would require bi-directional testing of check valves.

Length of test interval is the second area of Appendix II where the NRC believes the rules must be supplemented. Appendix II was first incorporated into the OM Code in the 1996 Addenda. Thus, the operating experience database does not yet exist to support long term test intervals for the condition monitoring concept. 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. Under the appendix, a licensee would be able to extend the interval without limit. 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. Interval changes or extensions must be justified and limited within the existing performance and experience database. Condition monitoring and the current experience data base 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. Proposed § 50.55a(b)(3)(iv)(B) would limit 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, and the total interval would be 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 final area in Appendix II which the Commission believes 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 IST 4.5.5 is required to return to the requirements of IST 4.5.4. However, the NRC believes the requirements of IST 4.5.1 through IST 4.5.4 must be also met. Hence, if the monitoring program is discontinued, proposed § 50.55a(b)(3)(iii)(C) would require a licensee to implement the provisions of IST 4.5.1 through IST 4.5.4.

2.5.3.3 Subsection ISTD

The IST of dynamic restraints or snubbers is governed by plant technical specification and, thus, has never been included in § 50.55a. However, the NRC has reviewed Subsection ISTD, 1995 Edition with the 1996 Addenda, and has determined that the provisions for IST of snubbers are an acceptable alternative to the requirements contained in the plant technical specifications. Subsection ISTD, 1996 Addenda, includes new provisions for service life monitoring of snubbers. The new provisions require that the service lives of snubbers be predicted and evaluated to ensure that the service life will not be exceeded before the next scheduled refueling outage. These new provisions simply formalize preventative maintenance practices presently found in most plants. Because the IST of snubbers is governed by plant technical specifications, Subsection ISTD is not included in the proposed mandatory requirements of the rulemaking, but licensees may choose to voluntarily implement Subsection ISTD, 1995 Edition with the 1996 Addenda, by processing a change to their technical specifications. This proposed modification is contained in § 50.55a(b)(3)(v).

2.5.3.4 Containment Isolation Valves

The proposed amendment would 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 effective 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 OMa1988 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. As noted in the ''Supplementary Information'' for the August 6, 1992 rulemaking, the ASME Operations and Maintenance (OM) Committee had 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 Part 10 for those CIVs that do not provide a reactor coolant system pressure isolation function. The NRC made a commitment via the Supplementary Information to reevaluate the need for the modification to Section XI, Subsection IWV, following review of this OM Committee basis document. This basis document was transmitted to the NRC in a letter from Steve Weinman, Secretary, OM Committee, to Eric S. Beckjord, Director, Office of Nuclear Regulatory Research, dated February 16, 1994. The NRC has determined that the requirements of 10 CFR 50, Appendix J, ensure adequate identification analysis, and corrective actions for leakage monitoring of CIVs, and that the existing modification in § 50.55a(b)(2)(vii) should be deleted. The regulatory analysis for this proposed rule contains a detailed discussion of the basis document findings and the NRC staff evaluation.

2.6 ASME Code Interpretations

The ASME issues Interpretations to clarify provisions of the BPV and OM Codes. Requests for Interpretations are submitted by users, and after appropriate committee deliberations and balloting, responses are issued by the ASME. Generally, the NRC agrees with these interpretations. When the NRC incorporates by reference specific editions and addenda into its regulations, the NRC has a certain understanding of those editions and addenda. Because an Interpretation is issued subsequent to issuance of the provision to which it refers, the Interpretation may affect that understanding. While the NRC acknowledges that the ASME is the official interpreter of the Code, the NRC will not accept ASME interpretations that, in NRC's opinion, are contrary to NRC requirements or may adversely impact facility operations. Interpretations have been issued which in some cases, conflicted with or were inconsistent with NRC requirements. These resulted in enforcement actions. Of particular concern are Code Interpretations that may be implemented following initiation of enforcement action by the NRC. ASME Code Interpretations were discussed in 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 § 50.55a and have not received NRC approval.

2.7 DSI–13

Since 1992, when the Commission last revised § 50.55a to endorse new ASME Code Editions and addenda (57 FR 34666), several developments have occurred which have raised some fundamental issues with respect to the Commission's endorsement of ASME Codes. First, on October 21, 1993, Entergy Operations, Inc. submitted a request that would relieve it from updating its ISI and IST programs to the last ASME Code edition and addenda incorporated by reference into § 50.55a. The underlying premise of the request was that a licensee should not be required to upgrade its ISI and IST program without considering whether the costs of the upgrade are warranted in light of the increased safety afforded by the updated Code edition and addenda. Though the request was later withdrawn, the underlying premise resulted in NRC reconsideration of the 120-month update. Requiring Code updates every 120-months is still under active consideration. However, the proposed rule has been prepared under the traditional approach; i.e., licensees would be required to update their ISI and IST programs every 120-months to the latest edition and addenda incorporated by reference into § 50.55a. If a decision is reached subsequent to publication of the proposed rule that is adverse to this approach, this position will be corrected prior to publication of the final rule.

Second, the National Technology Transfer and Advancement Act of 1995, PL 104–113, was signed into law on March 7, 1996. The Act directs federal agencies to achieve greater reliance on technical standards developed by voluntary consensus standards development organizations. Finally, the Commission commenced a Strategic Assessment and Rebaselining Initiative. One of the issues addressed in this effort was Direction Setting Issue (DSI) 13, which raised the question, ''In performing its regulatory responsibilities, what consideration should the NRC give to industry activities.'' A draft paper addressing DSI–13 was published for public comment on September 16, 1996, after which the Commission held public meetings to facilitate understanding of the issues and receive comments on the DSI–13 draft paper. Based on the public comments, 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 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.

2.8 Steam Generators

ASME Code requirements for repair of heat exchanger tubes by sleeving were added to Section XI in the 1989 Addenda. Minimum Code requirements for tube sleeving was added to the Code so that licensees would not have to develop sleeving programs and have them approved by the NRC on a case-by-case basis. The NRC has reviewed the Code requirements for sleeving and determined that they are acceptable. However, it should be recognized that there are other relevant requirements, and that a considerable amount of effort is presently being expended due to the number of occurrences of degraded steam generator tubing. For example, licensees are required by either 10 CFR 50.55a(f) or by the plant technical specifications to perform periodic inservice inspections and to repair (e.g., sleeving) or remove from service (by installing plugs in the tube ends) all tubes found to contain flaws exceeding the plugging limit (i.e., tube repair criteria). In addition, current technical specifications contain operational leakage limits. Licensee's have frequently found it necessary to implement measures beyond minimum Code and technical specification requirements to ensure adequate tube integrity when significant degradation problems are encountered. Thus, the NRC determination that the sleeving requirements are acceptable should be kept in perspective.

3. 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, if adopted, would not have a significant effect on the quality of the human environment and therefore an environmental impact statement is not required.

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

The proposed rule would impose the 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, would result in fewer supports being examined which would 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 proposed rule would impose Appendix VIII to Section XI, 1995 Edition with the 1996 Addenda, BPV Code, for the first time and would expedite its implementation. Appendix VIII provides rules for the performance demonstration of ultrasonic examination systems, procedures, and personnel. Implementation of this appendix should result in a decrease in occupational exposure. Appendix VIII qualified procedures and personnel should reduce repeat ultrasonic testing (UT), which could reduce occupational exposure. In addition, flaws should be detected at an earlier stage of growth resulting in less extensive repair operations, which could further reduce occupational exposure.

The proposed rule would incorporate by reference into the regulations the 1995 Edition with the 1996 Addenda of the OM Code. Imposition of the OM Code is 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 required of applicants and licensees to implement the proposed 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 NRC has sent a copy of the Environmental Assessment and the proposed rule 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 (Lower Level), Washington, DC. Single copies of the environmental assessment are available from Frank C. Cherny, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6786, or Wallace E. Norris, Division of Engineering Technology, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6796.

4. Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 *et seq.*). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The public reporting burden for this information collection is estimated to average 67 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. The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the information collections contained in the proposed rule and on the following issues:

1. Is the proposed information collection necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?
2. Is the estimate of burden accurate?
3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?
4. How can the burden of the information collection be minimized, including the use of automated collection techniques?

Send comments on any aspect of this proposed collection of information, including suggestions for further 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.

Comments to OMB on the information collections or on the above issues should be submitted by January 2, 1998. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date.

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.

5. Regulatory Analysis

The Commission has prepared a draft regulatory analysis on this proposed regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. The draft analysis is available for inspection in the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington DC. The Commission requests public comment on the draft analysis. Single copies of the analysis may be obtained from Frank C. Cherny, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6786, Wallace E. Norris, Division of Engineering Technology, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6796.

6. 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 proposed 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.

7. Backfit Analysis

The Nuclear Regulatory Commission (NRC) regulations, 10 CFR 50.55a, requires 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 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (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. Licensees are required to update every 120 months to the version of Section XI incorporated by reference into § 50.55a 12 months prior to the start of a new ten year interval.

The proposed amendment to § 50.55a would require licensees to update ISI in accordance with Section XI of the ASME BPV Code and IST in accordance with the ASME OM Code. Licensees would be required to implement the 1995 Edition with the 1996 Addenda of (1) Section XI, Division 1 for Class 1, Class 2, Class 3, Class MC, and Class CC components; (2) the ''Code for Operation and Maintenance of Nuclear Power Plants'' (OM Code) for Class 1, Class 2, and Class 3 pumps and valves; and (3) Appendix VIII, ''Performance Demonstration for Ultrasonic Examination Systems,'' to Section XI, Division 1. As permitted by § 50.55a(a)(3), licensees may voluntarily update to the 1989 Addenda through the 1996 Addenda of Section III of the BPV Code, with limitation. In addition, the modification for containment isolation valve inservice testing that applied to the 1989 Edition of the BPV Code has been deleted. Licensees will continue to be required to update their ISI and IST programs every 120 months to the version of Section XI and the OM Code incorporated by reference and in effect at least 12 months prior to the start of a new 120-month interval.

The NRC position on the routine 120-month update to § 50.55a has consistently been that 10 CFR 50.109 does not require a backfit analysis of the routine 120-month update to § 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 § 50.55a requires that they update their inservice inspection program every 10 years to the latest edition and addenda of Section XI that were incorporated by reference in § 50.55a and in effect 12 months before

the start of the next inspection interval, and (3) endorsing and updating references to the ASME Code, a national consensus standard developed by the participants (including the NRC) with broad and varied interests, 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**.

The provisions for IST of pumps and valves were originally contained in Section XI Subsections IWP and IWV. 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 proposed 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, ''Performance Demonstration for Ultrasonic Examination Systems,'' to Section XI would 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. Appendix VIII would greatly enhance the reliability of detection and sizing of cracks and flaws, and it delineates a method for qualification of the personnel and procedures. The appendix 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 expedited implementation schedule, the imposition of Appendix VIII is being considered a backfit. Licensees would be required 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. The proposed rule would not require any change to a licensee's ISI schedule for examination of these components, but would require that the provisions of Appendix VIII be used for all examinations after that date rather than the UT procedures and personnel requirements presently being utilized by licensees.

The NRC has concluded, on the basis of the documented evaluation required by § 50.109(a)(4), that imposition of Appendix VIII, which would greatly enhance the overall level of assurance of the safety and reliability of ultrasonic examination techniques in detecting and sizing flaws, 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 Criteria II and XVI, of 10 CFR Part 50, Appendix B.

The modification to Section XI to require licensees to supplement the surface examination of the Class 1 portion (RCPB) of the HPSI system with volumetric examination would ensure the integrity of the reactor coolant system pressure boundary and maintenance of emergency core cooling system operability. The operability of this system is necessary to ensure the protection of the public health and safety, and the NRC has concluded, on the basis of the documented evaluation required by § 50.109(a)(4), that licensees must supplement the Section XI required surface examination for the Class 1 portion of the HPSI system with volumetric examination in order to ensure the integrity of the reactor coolant pressure boundary as required by GDC 14, 10 CFR Part 50, Appendix A, or similar provisions in the licensing basis for these facilities, and Criteria II and XVI, of 10 CFR Part 50, Appendix B. Volumetric examination would be required during any ISI program inspection of the HPSI system performed after 6 months from the date of the final rule.

GDC 14, ''Reactor coolant pressure boundary,'' (RCPB) or similar provisions in the licensing basis for these facilities, specify that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture. There has recently been an occurrence of gross rupture in the Class 1 portion of a HPSI system, and a number of occurrences of abnormal leakage in the RCPB in other plants.

Imposition of Appendix VIII and the HPSI volumetric examination is also necessary to bring the facilities described into compliance with Criteria II, ''Quality Assurance Program,'' and Criteria XVI, ''Corrective Actions,'' of Appendix B to 10 CFR Part 50. Criteria 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. Criteria 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. In analyzing the occurrences of pipe break and leakage, it is apparent that the RCPB is subject to certain types of degradation. Information gathered by the NRC staff indicates that many licensees have not reacted to this serious safety concern by performing more comprehensive examinations. The NRC believes that there is a basis for reasonably concluding that such degradation could occur in virtually all PWRs. 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 be reported, the NRC has determined that imposition of Appendix VIII and volumetric examination of the HPSI system 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 proposed 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 Frank C. Cherny, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–6786, or Wallace E. Norris, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission,

Washington, DC 20555–0001, Telephone: 301–415–6796.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalties, 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. 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

Authority: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1444, 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 removing and reserving paragraphs (b)(2)(vii) and (g)(4)(iv), adding paragraphs (b)(2)(xi) through (b)(2)(xx), (b)(3), (g)(6)(ii)(A)(*6*), and (g)(6)(ii)(C), and revising the introductory text of paragraph (b), paragraph (b)(1), the introductory text of paragraph (b)(2), paragraphs (b)(2)(iv), (b)(2)(vi), (b)(2)(viii), the introductory text of paragraph (b)(2)(ix), paragraphs (c)(3), (d)(2), (e)(2), the introductory text of paragraph (f), paragraphs (f)(1), (f)(2), (f)(3)(iii), (f)(3)(iv), the introductory text of paragraph (f)(4), paragraphs (g)(1), (g)(3)(i), the introductory text of paragraph (g)(4), paragraphs (g)(6)(ii)(A)(*1*), (g)(6)(ii)(A)(*2*), and Footnotes 5 and 7 to read as follows:

§ 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 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, United Engineering Center, 345 East 47th Street, New York, NY 10017. They are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852–2738.

(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) Engineering judgement. When a licensee relies on engineering judgment for activities or evaluations of components or systems within the scope of 10 CFR 50.55a that are not directly addressed by the ASME Boiler and Pressure Vessel Code, the NRC must approve the activities or evaluations pursuant to 10 CFR 50.55a(a)(3).

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

(iii) Weld leg dimensions. When applying the 1989 Addenda through the 1996 Addenda of Section III, licensees shall not apply paragraph NB–3683.4(c)(1), Footnote 11 to Figure NC–3673.2(b)–1, and Figure ND–3673.2(b)–1, and shall continue to use the requirements in the 1989 Edition for this paragraph and figures.

(iv) Seismic design. Licensees may use Articles NB–3200, NB–3600, NC–3600, and ND–3600 through the 1993 Addenda, subject to the limitation specified in (b)(1)(iii) of this section. Licensees shall not use the provisions in the 1994 Addenda through the 1996 Addenda for these Articles.

(v) 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 Addenda are acceptable for use provided that both NQA–1 and the quality assurance provisions specified in NCA–4000 are used in conjunction with the administrative, quality, and technical provisions contained in the edition and addenda of Section III being utilized.

(vi) Independence of inspection. Licensees shall not apply NCA–4134.10(a) of Section III, 1995 Edition with the 1996 Addenda, and shall use NCA–4134.10(a), 1994 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:

* * * * *

(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.

* * * * *

(vi) Effective edition and addenda of Subsection IWE and Subsection IWL, Section XI. Licensees shall 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)(ix) and § 50.55a(b)(2)(x).

(vii) [Reserved]

(viii) 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 shall be the OMa–1988 Addenda to the OM–1987 Edition. These requirements have been incorporated into the 1990 Edition of the OM Code which is incorporated by reference in paragraph (b)(3) of this section.

(ix) 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)(ix)(A), (D)(3), and (E) of this section.

* * * * *

(xi) Engineering judgment. When a licensee relies on engineering judgment for activities or evaluations of components or systems within the scope of 10 CFR 50.55a that are not directly addressed by the ASME Boiler and Pressure Vessel Code, the NRC must approve the activities or evaluations pursuant to 10 CFR 50.55a(a)(3).

(xii) 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, provided the licensee utilizes its 10 CFR Part 50, Appendix B, quality assurance program, in conjunction with Section XI requirements. Changes to licensee's quality assurance program shall be made in accordance with 10 CFR 50.54(a). In addition, where NQA–1 and Section XI do not address the commitments contained in the licensee's Appendix B quality assurance program description, such commitments shall be applied to Section XI activities.

(xiii) Class 1 piping. Licensees shall not apply IWB–1220, ''Components Exempt from Examination,'' of Section XI, 1989 Addenda through the 1996 Addenda, and shall apply IWB–1220, 1989 Edition.

(xiv) Class 2 piping. Prior to applying the provisions of IWC–1220, ''Components Exempt from Examination,'' IWC–1221, ''Components Within RHR, ECC, and CHR Systems or Portions of Systems,'' and IWC–1222, ''Components Within Systems or Portions of Systems Other Than RHR, ECC, and CHR Systems,'' 1989 Addenda through the 1996 Addenda, licensees shall define the Class 2 piping subject to volumetric and surface examination, and submit this information for approval by the NRC staff pursuant to § 50.55a(a)(3) prior to implementation.

(xv) Class 1 piping volumetric examination. When performing weld examinations of High Pressure Safety Injection Systems, as required by Table IWB–2500–1, Examination Category B–J, Item Numbers B9.20, B9.21, and B9.22, all licensees of pressurized water reactor facilities shall perform volumetric examination of the Class 1 portion of the system after [insert 6 months from the date of the final rule].

(xvi) Flaws in Class 3 piping moderate energy (200 xF, 275 psig) piping. Licensees may use the provisions of Code Case N–513, ''Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping,'' Rev 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 shall not be applied to:

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

(*2*) The discovery and repair of flaws or deficiencies remaining from original construction;

(*3*) Leakage through a flange gasket;

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

(*5*) Degraded socket welds.

(xvii) Appendix VIII personnel qualification. All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII shall receive 40 hours of annual training that includes laboratory work and examination of flawed specimens.

(xviii) Appendix VIII specimen set cracks. All flaws in the specimen sets used for performance demonstration for piping, vessels, and nozzles shall be cracks.

(xix) Appendix VIII specimen set microstructure. All specimens for single-side tests shall contain microstructures of the type found in components to be inspected, and flaws with non-optimum characteristics consistent with field experience that provide realistic challenges to the UT techniques.

(xx) Reconciliation of Quality Requirements. The following limitations apply when implementing Section XI, IWA–4200, 1995 Addenda through the 1996 Addenda:

(A) Licensees shall not apply IWA–4200, of Section XI, 1995 Addenda through the 1996 Addenda, for reconciliation of the administrative requirements for replacement items, and shall reconcile the administrative requirements with the original Construction Code and the Owner's requirements as required by the 1995 Edition.

(B) Licensees shall not apply the definition of Construction Code in IWA–9000, ''Glossary,'' 1993 Addenda through the 1996 Addenda, and shall apply the definition of Construction Code in IWA–9000, 1992 Edition.

(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 addenda through the 1996 Addenda and editions through the 1995 Edition subject to the following limitations and modifications:

(i) *Quality Assurance.* When applying editions and addenda of the OM Code, 1990 and later, 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 utilizes its 10 CFR Part 50, Appendix B, quality assurance program, in conjunction with the OM Code requirements. Changes to licensee's quality assurance program shall be made in accordance with 10 CFR 50.54(a). In addition, where NQA–1 and the OM Code do not address the commitments contained in the licensee's Appendix B quality assurance program description, such commitments shall be applied to OM Code activities.

(ii) 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 the programs developed under their licensing commitments for demonstrating design basis capability of motor-operated valves.

(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 Operated Valve Assemblies in LWR Power Plants,'' Rev. 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 test interval for each valve shall be evaluated and adjusted as necessary but not later than five years or three refueling outages (whichever is longer) from initial implementation of ASME Code Case OMN–1.

(B) [Reserved]

(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 shall not exceed two fuel cycles or 3 years, whichever is longer; any extension of this interval shall not exceed one fuel cycle per extension with the maximum interval not to exceed 10 years; trending and evaluation of existing data shall be used to reduce or extend time the 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 shall be implemented.

(v) Subsection ISTD. Licensees may use Subsection ISTD, OM Code, 1995 Edition with the 1996 Addenda, by making a change to their technical specifications in accordance with applicable NRC requirements. Licensees choosing to apply the subsection shall apply all of its provisions.

(c) * * *

(3) The Code Edition, Addenda, and optional 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, and, in case of conflict between paragraph (b)(1) of this section and paragraph NCA–1140, the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section shall be applied,

(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

* * * * *

(d) * * *

(2) The Code Edition, Addenda, and optional Code Cases6 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 Vessel and Pressure Code, but:

(i) The edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, and, in case of conflict between paragraph (b)(1) of this section and paragraph NCA–1140, the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section shall be applied,

(ii) The ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and

(iii) The ASME Code Cases6 must have been determined suitable for use by the NRC.

(e) * * *

(2) The Code Edition, Addenda, and optional Code Cases6 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, and, in case of conflict between paragraph (b)(1) of this section and paragraph NCA–1140, the latest edition and addenda incorporated by reference in paragraph (b)(1) of this section shall be applied,

(ii) The ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and

(iii) The ASME Code Cases must have been determined suitable for use by the NRC.

(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 in steam, water, air, and liquid-radioactive-waste systems 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 such 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, 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 of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda6 in effect 6 months prior to the date of issuance of the construction permit. The pumps and valves may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to limitations and modifications listed therein.

(3) * * *

(iii)(A) Pumps and valves, in facilities whose construction permit was issued before [insert effective date of the final rule], 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 Addenda6 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 [insert effective date of the final rule], 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 [insert effective date of rule], 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 Addenda6 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 [insert effective date of the final rule], 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 pressure vessels, piping, pumps and valves, and their supports in steam, water, air, and liquid-radioactive-waste systems that provide pressure boundary integrity for systems that perform a function to shut down the reactor or maintain the reactor in a safe shutdown condition, or mitigate the consequences of an accident, must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

* * * * *

(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 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 Addenda6 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 Xl 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 paragraph (b)(2)(ix) and (b)(2)(x) of this section, to the extent practical within the limitation of design, geometry and materials of construction of the components.

* * * * *

(iv) [Reserved]

(6) * * *

(ii) * * *

(A)(*1*) All previously granted reliefs under § 50.55a to licensees for the extent of volumetric examination of reactor vessel shell welds specified in Item BI.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 81.10 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.

* * * * *

(*6*) Augmented examinations of reactor vessel shell welds that are performed in accordance with § 50. 55a(g)(6)(ii)(A) after [insert 6 months from the date of the final rule] must be performed in accordance with § 50.55a(g)(6)(ii)(C).

* * * * *

(C) *Application of Appendix VIII to Section Xl Examinations.*

(*1*) All reactor vessel (including nozzles) ultrasonic examinations, all piping ultrasonic examinations, and all bolting ultrasonic examinations performed after *insert 6 months from the date of the final rule* must be

performed in accordance with Appendix VIII of Section Xl, Division 1, 1995, Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code.

(*2*) [Reserved]

* * * * *

[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.

* * * * *

[7] For purposes of this regulation the proposed IEEE–279 became ''in effect'' on August 30, 1968, and the revised issue IEEE–279–1971 became ''in effect'' on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the NRC Library, Two White Flint North, 11545, Rockville Pike, Rockville, Maryland 20852–2738.

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Dated at Rockville, MD this 27th day of October 1997.

For the Nuclear Regulatory Commission.

L. Joseph Callan,

Executive Director for Operations.

[FR Doc. 97–31588 Filed 12–2–97; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

10 CFR Parts 50 and 70

RIN 3150–AF87

Criticality Accident Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to provide light-water nuclear power reactor licensees with greater flexibility in meeting the requirement that licensees authorized to possess more than a small amount of special nuclear material (SNM) maintain a criticality monitoring system in each area where the material is handled, used, or stored. This action is taken as a result of the experience gained in processing and evaluating a number of exemption requests from power reactor licensees and NRC's safety assessments in response to these requests that concluded that the likelihood of criticality was negligible.

DATES: Comments on the proposed rule must be received on or before January 2, 1998.

ADDRESSES: Mail comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Attention: Rulemaking and Adjudication Staff. Hand deliver comments to 11555 Rockville Pike, Maryland, between 7:45 am and 4:15 pm on Federal workdays.

Copies of any comments received may be examined at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC.

For information on submitting comments electronically, see the discussion under Electronic Access in the Supplementary Information section.

FOR FURTHER INFORMATION CONTACT: Stan Turel, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone (301) 415–6234, e-mail spt@nrc.gov.

SUPPLEMENTARY INFORMATION: For additional information see the Direct Final Rule published in the rules section of this **Federal Register**.

Procedural Background

Because NRC considers this action noncontroversial and routine, we are publishing this proposed rule concurrently as a direct final rule. The direct final rule will become effective on February 17, 1998. However, if the NRC receives significant adverse comments on the direct final rule by January 2, 1998, then the NRC will publish a document that withdraws the direct final rule. If the direct final rule is withdrawn, the NRC will address in a Final Rule the comments received in response to the proposed revisions in a subsequent final rule. Absent significant modifications to the proposed revisions requiring republication, the NRC will not initiate a second comment period for this action in the event the direct final rule is withdrawn.

Electronic Access

You may also provide comments via the NRC's interactive rulemaking web site 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 site, contact Ms. Carol Gallagher, (301) 415–6215; e-mail CAG@nrc.gov.

List of Subjects

10 CFR Part 50

Antitrust, Classified information, Criminal penalties, Fire prevention, Intergovernmental relations, Nuclear power plants and reactors, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

10 CFR Part 70

Criminal penalties, Hazardous materials transportation, Material control and accounting, Nuclear materials, Packaging and containers, Radiation protection, Reporting and recordkeeping requirements, Scientific equipment, Security measures, Special nuclear material.

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, the National Environmental Policy Act of 1969, as amended, and 5 U.S.C. 553, the NRC is considering adopting the following amendments to 10 CFR Parts 50 and 70.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for 10 CFR 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, as amended by Pub. L. 102–486, sec. 2902, 106 Stat. 3123, (42 U.S.C. 5851). Sections 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L. 91–190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97–415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. Section 50.68 is added under the center heading ''Issuance, Limitations, and Conditions of Licenses and Construction Permits'' to read as follows: