NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150–AE26

Industry Codes and Standards; Amended Requirements

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

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

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

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

SUPPLEMENTARY INFORMATION:

1. Background

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

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

For ISI, the proposed amendment would have required licensees to implement Section XI, Division 1, of the ASME BPV Code, 1995 Edition up to and including the 1996 Addenda for Class 1, Class 2, and Class 3 components with five proposed limitations. The proposed amendment included permission for licensees to implement Code Cases N–513, ''Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping,'' and N–523, ''Mechanical Clamping Devices for Class 2 and 3 Piping.'' The proposed amendment also would allow licensees to use the 1992 Edition including the 1992 Addenda of Subsections IWE and IWL in lieu of updating to the 1995 Edition and the 1996 Addenda. The proposed rule included expedited implementation of Appendix VIII, ''Performance Demonstration for Ultrasonic Examination Systems,'' to Section XI, Division 1, with three proposed modifications. An expedited examination schedule would also have been required for a proposed modification to Section XI which addresses volumetric examination of Class 1 high pressure safety injection (HPSI) piping systems in pressurized water reactors (PWRs).

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

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

2. Summary of Comments

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

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

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

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

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

- OM Code (Not A Backfit)

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

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

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

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

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

- Section XI

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

- OM Code

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

2.2 Discussion

2.3 120-Month Update

2.3.1 Section XI

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

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

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

2.3.1. Limitations.

2.3.1.2.1 Engineering Judgement.

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

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

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

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

2.3.1.2.2 Quality Assurance.

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

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

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

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

2.3.1.2.3 Class 1 Piping.

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

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

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

2.3.1.2.4 Class 2 Piping.

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

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

2.3.1.2.5 Reconciliation of Quality Requirements.

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

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

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

2.3.2 OM Code (120-Month Update).

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

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

2.3.2.2 Background—OM Code.

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

The 1990 Edition of the ASME OM Code consists of one section (Section IST) entitled ''Rules for Inservice Testing of Light-Water Reactor Power Plants.'' This section is divided into four subsections: ISTA, ''General Requirements,'' ISTB, ''Inservice Testing of Pumps in Light-Water Reactor Power Plants,'' ISTC, ''Inservice Testing of Valves in Light-Water Reactor Power Plants,'' and ISTD, ''Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers).'' The testing of snubbers is governed by the ISI requirements of Section XI of the ASME BPV Code. Therefore, the rule only requires implementation of Subsections ISTA, ISTB, and ISTC. Because this final rule for the first time incorporates by reference the OM Code, the NRC has determined that the latest endorsed Edition and Addenda of the OM Code (i.e., 1995 Edition up to and including the 1996 Addenda) should be used. Therefore, there is no need to incorporate by reference earlier Editions and Addenda of the OM Code (e.g., 1990 Edition or 1992 Edition).

2.3.2.2.1 Comments on the OM Code.

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

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

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

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

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

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

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

2.3.2.4 Limitation.

2.3.2.4.1 Quality Assurance.

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

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

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

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

2.3.2.5 Modification.

2.3.2.5.1 Motor-Operated Valve Stroke-Time Testing.

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

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

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

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

2.4 Expedited Implementation.

2.4.1 Appendix VIII.

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

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

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

(1) Some Appendix VIII supplements have not yet been implemented by PDI;

(2) The number of qualified individuals is not yet sufficient;

(3) The rule would require UT personnel to requalify; and

(4) PDI's implementation of Appendix VIII differs from the Code.

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

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

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

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

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

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

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

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

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

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

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

2.4.1.1 Modifications.

2.4.1.1.1 Appendix VIII Personnel Qualification.

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

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

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

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

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

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

2.4.1.1.2 Appendix VIII Specimen Set and Qualification Requirements.

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

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

The NRC believes that flaws in test specimens used for UT should be representative of the flaws normally found or expected to be found in operating plants. Based on the public comments, the final rule in § 50.55a(b)(2)(xv) permits a population of notches and fabrication flaws on a limited basis for vessel and nozzle test specimen sets (Supplements 4, 5, 6, and 7). For these components, the NRC has concluded that a mix of cracks and notches is acceptable as long as they provide a similar detection and sizing challenge to that seen in actual service induced degradation. These types of notches will ensure that the qualification demonstration tests the ability of an examiner to discriminate between real flaws and innocuous reflectors. In addition, a mix of cracks and notches means that the present specimens can continue to be used for qualification. For wrought austenitic, ferritic, and dissimilar metal welds, however, these flaws can best be represented with cracks. Cracks span the ultrasonic spectra of flaw surface conditions from rough to smooth, jagged to straight, single to multiple tip, and tight to wide tip. Notches generally have smooth surfaces that reflect a narrow ultrasonic spectrum that represents a small population of flaws contained in components. Some variations in UT examination techniques may be more challenged with a notch located in specific locations, whereas other variations in UT examination techniques may not. With respect to bolting, the NRC believed it would be clear that bolting was not addressed by the proposed modification. The NRC does not consider it necessary to use cracks for performance qualification for Supplement 8 as notches are appropriate reflectors in the specimen test sets.

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

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

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

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

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

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

2.4.2 Generic Letter on Appendix VIII.

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

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

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

2.4.3 Class 1 Piping Volumetric Examination (Deferred).

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

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

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

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

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

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

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

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

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

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

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

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

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

These five concerns are addressed in order below.

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

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

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

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

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

2.5 Voluntary Implementation.

2.5.1 Section III.

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

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

2.5.1.1 Limitations.

2.5.1.1.1 Engineering Judgment (Deleted).

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

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

2.5.1.1.2 Section III Materials.

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

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

2.5.1.1.3 Weld Leg Dimensions.

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

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

The NRC disagrees that the limitation should be deleted from the final rule. The weld size requirements that were used in the majority of U.S. operating nuclear power plant piping systems were provided by ANSI B31.7, Nuclear Power Piping Code, ANSI B31.1, Power Piping Code, and early editions of the ASME Code, Section III. Specifically, these standards required that the minimum socket weld size equal 1.25 t but not less than ⅛ inch, where t is the nominal pipe wall thickness. The same weld size requirements as those specified in the above listed codes are also required by other nationally recognized codes and standards such as ANSI B31.3, Petroleum Refinery Piping Code. Those sizes were established as a result of many years of experience associated with the design and construction of piping systems, piping equipment, and components. In 1981, Code Case N–316, ''Alternative Rules for Fillet Weld Dimensions for Socket Welded Fittings,'' was published permitting a reduction in socket weld sizes to 1.09 t. In essence, the Code case was developed to provide relief for certain utilities having difficulty complying with the minimum socket weld size requirement of 1.25 t. The provisions contained in the Code case were incorporated into the 1989 Edition of the ASME Code. The NRC accepted this reduction because the new weld size was still greater than the pipe. In the 1989 Addenda of Section III of the ASME Code, the requirements for the size of socket welds were further reduced to 0.75 t which would permit welds smaller than the thickness of the pipe. The NRC is concerned with the structural integrity of a joint with a weld size which is less than the pipe wall thickness. The reduction to 0.75 t was not supported with test results or operating experience. Thus, a good technical basis has not been provided for reducing minimum socket weld sizes in nuclear power plant piping. It should be noted that the petrochemical industry has not made a corresponding change to the standards governing weld sizes in refinery piping. Hence, this limitation has been retained in § 50.55a(b)(1)(ii).

2.5.1.1.4 Seismic Design.

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

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

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

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

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

2.5.1.1.5 Quality Assurance.

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

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

Section III is a design and construction code used by the manufacturers and suppliers of new Code items. However, Section III is also used for controlling the construction of replacement Code items during the operational phase at nuclear power plants. The basis for the limitation in the proposed rule was that the quality provisions contained in NQA–1 (any version) are not adequate to describe how to satisfy the applicable 10 CFR requirements for these activities. The NRC has not taken any exceptions to the quality or administrative provisions contained in Section III. However, in the proposed limitation for Section III, the NRC emphasized that the quality provisions of NQA–1 are acceptable for use in the context of Section III activities for the construction of new and replacement Code items. Therefore, the NRC has concluded that the quality provisions contained in Section III are acceptable for the construction of new and replacement items; i.e., NQA–1 is not adequate by itself. Thus, the limitation has been retained in § 50.55a(b)(1)(iv).

2.5.1.1.6 Independence of Inspection.

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

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

The NRC disagrees with both the third and fourth commenters. Criterion I, ''Organization,'' of 10 CFR part 50, appendix B requires the establishment and execution of a quality assurance program which includes establishing and delineating in writing the authority and duties of persons and organizations performing activities affecting the safety-related functions of structures, systems, and components. In particular, Criterion I states: ''These activities include both the performing functions of attaining quality objectives and the quality assurance functions. The quality assurance functions are those of (a) assuring that an appropriate quality assurance program is established and effectively executed and (b) verifying, such as by checking, auditing, and inspection, that activities affecting safety-related functions have been correctly performed.'' Criterion I continues by stating that ''[t]he persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided.'' Criterion X, ''Inspection,'' of Appendix B requires ''[s]uch inspection shall be performed by individuals other than those who performed the activity being inspected.''

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

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

2.5.1.2 Modification.

2.5.1.2.1 Applicable Code Version for New Construction.

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

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

2.5.2 Section XI (Voluntary Implementation).

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

2.5.2.1 Subsection IWE and Subsection IWL.

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

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

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

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

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

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

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

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

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

The intent of the rule was to require implementation of all the Articles of Subsection IWL. The failure to include ''replacements'' was an oversight. Section 50.55a(g)(4) requires that ''* * * components which are classified as Class CC pressure retaining components and their integral attachments must meet the requirements, except for design and access provisions and preservice examination requirements, set forth in Section XI of the ASME Boiler and Pressure Vessel Code and Addenda that are incorporated by reference in paragraph (b).'' Section 50.55a(g)(4)(v)(C) has been clarified in this final rule by including ''replacement'' in order to eliminate any further confusion.

2.5.2.2 Flaws in Class 3 Piping.

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

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

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

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

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

With regard to the limitations on the use of Code Case N–513, some commenters questioned the restrictions and believe that the Code case should be permitted in other applications such as socket welded connections. The Code case has been approved for use on moderate energy Class 3 piping and tubing (which is the ASME scope of the Code case). The NRC does not believe that the criteria are applicable to socket welds because NDE methods are not available for adequate flaw characterization. In addition, the NRC does not agree that the level of reduction of safety margins which would be permitted by the Code case is appropriate. The margins available in an unflawed component are expected to be higher than for a degraded component. Margins less than the minimums specified for Level A, B, C, and D loading conditions are not acceptable. Hence, these restrictions have been maintained in the final rule except for the limitation related to original construction. The NRC agrees with commenters that any defects remaining from construction that have been determined by evaluation to be permissible are acceptable and has removed this limitation from the final rule. Code Cases N–513 and N–523–1 are addressed in § 50.55a(b)(2)(xiii) of the final rule.

2.5.2.3 Application of Subparagraph IWB–3740, Appendix L.

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

2.5.3 OM Code (Voluntary Implementation).

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

2.5.3.1 Code Case OMN–1.

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

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

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

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

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

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

Seven commenters responded to the proposed voluntary use of Code Case

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

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

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

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

2.5.3.2 Appendix II.

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

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

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

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

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

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

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

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

2.5.3.3 Subsection ISTD.

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

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

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

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

Section 2.5.3.3, ''Subsection ISTD,'' of the Statement of Considerations for the proposed rule (62 FR 63903; December 3, 1997) stated that inservice testing of dynamic restraints or snubbers is governed by plant technical specifications and, thus, has never been included in 10 CFR 50.55a. It was apparent from comments received on this section that this statement was confusing and needed to be clarified. First, it is true that 10 CFR 50.55a never directly required inservice testing of snubbers although the language in the current rule would appear to indicate otherwise. The language in the current rule states in § 50.55a(f)(4), ''Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements * * * set forth in section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda * * *'' (emphasis added). Although the language clearly states that ''components (including supports)'' are within the scope of inservice testing, and it appears that inservice testing of snubbers is included under this statement, this statement was an editorial error. In the 1992 final rule amending 10 CFR 50.55a to more clearly distinguish the requirements for inservice testing from those for inservice inspection (57 FR 34666; August 6, 1992), paragraph (g) was split into two separate paragraphs—paragraph (f) for inservice testing and paragraph (g) was retained for inservice inspection. In the 1992 final rule, similar requirements that applied to both inservice inspection and inservice testing were carried over from paragraph (f) to paragraph (g). The terminology, ''components (including supports),'' which existed in paragraph (g) was changed in paragraph (f) to read, ''pumps and valves,'' except in this one instance. Therefore, the Commission views this error as an editorial oversight. In the final rule, the language in paragraph (f)(4) has been corrected to read, ''pumps and valves,'' instead of ''components (including supports).''

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

2.5.3.4 Containment Isolation Valves.

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

2.6 ASME Code Interpretations.

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

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

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

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

2.7 Direction Setting Issue 13.

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

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

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

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

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

2.8 Steam Generators.

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

2.9 Future Revisions of Regulatory Guides Endorsing Code Cases.

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

3. Voluntary Consensus Standards

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

4. Finding of No Significant Environmental Impact

Based upon an environmental assessment, the Commission has determined, under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in subpart A of 10 CFR part 51, that this rule will not have a significant effect on the quality of the human environment and therefore an environmental impact statement is not required.

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

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

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

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

The Commission has determined, in accordance with the National Environmental Policy Act of 1969, as amended and the Commission's regulations in subpart A of 10 CFR part 51, that this rulemaking is not a major action significantly affecting the quality of the human environment, and, therefore, an environmental impact statement is not required. This final rule amends the NRC regulations pertaining to ISI and IST requirements for nuclear power plant components. The current regulations in 10 CFR 50.55a incorporates by reference the 1989 Edition of the ASME BPV Code, Section III, Division 1; the 1989 Edition of the ASME BPV Code, Section XI, Division 1, for Class 1, Class 2, and Class 3 components; the 1992 Edition with the 1992 Addenda of the ASME BPV Code, Section XI, Division 1, for Class MC and Class CC components; and the 1989 Edition of the ASME BPV Code, Section XI, Division 1, for Class 1, Class 2, and Class 3 pumps and valves. The Commission is amending its regulations to incorporate by reference the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section III, Division 1, of the ASME BPV Code with five limitations; the 1989 Addenda, 1990 Addenda, 1991 Addenda, 1992 Edition, 1992 Addenda, 1993 Addenda, 1994 Addenda, 1995 Edition, 1995 Addenda, and 1996 Addenda of Section XI, Division 1, of the ASME BPV Code with three limitations; and the 1995 Edition and 1996 Addenda of the ASME OM Code with one limitation and one modification. The final rule imposes an expedited implementation of performance demonstration methods for ultrasonic examination systems. The final rule permits the optional implementation of the ASME Code, Section XI, provisions for surface examinations of High Pressure Safety Injection Class 1 piping welds. The final rule also permits the use of evaluation criteria for temporary acceptance of flaws in ASME Code Class 3 piping (Code Case N–523–1); mechanical clamping devices for ASME Code Class 2 and 3 piping (Code Case N–513); the 1992 Edition including the 1992 Addenda of Subsections IWE and IWL in lieu of updating to the 1995 Edition and 1996 Addenda; alternative rules for preservice and inservice testing of certain motor-operated valve assemblies (OMN–1) in lieu of stroke-time testing; a check valve monitoring program in lieu of certain requirements in Subsection ISTC of the ASME OM Code (Appendix II to the OM Code); and guidance in Subsection ISTD of the OM Code as part of meeting the ISI requirements of Section XI for snubbers. This final rule deletes a previous modification for inservice testing of containment isolation valves. The editions and addenda of the ASME BPV Code and OM Code incorporated by reference provide updated rules for the construction of components of light-water-cooled nuclear power plants, and for the inservice inspection and inservice testing of those components. This final rule permits the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power plant components. For these reasons, the Commission concludes that this rule should have no significant adverse impact on the operation of any licensed facility or the environment surrounding these facilities.

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

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

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

5. Paperwork Reduction Act Statement

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

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

Public Protection Notification

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

6. Regulatory Analysis

The Commission has prepared a regulatory analysis on this final regulation. The analysis examines the costs and benefits of the alternatives considered by the Commission. The analysis is available for inspection in the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington DC. Single copies of the analysis may be obtained from Thomas G. Scarbrough, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–2794, or Robert A. Hermann, Division of Engineering, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, Telephone: 301–415–2768.

7. Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission certifies that this rule will not have a significant economic impact on a substantial number of small entities. This final rule involves the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of ''small entities'' set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR part 121. Public comment received on this section suggested that the implementation of Appendix VIII of ASME BPV Code, Section XI, on performance qualification for ultrasonic testing might negatively impact small entities that contract their examination personnel to nuclear power plants. However, the final rule permits licensees to implement either Appendix VIII as contained in the 1995 Edition with the 1996 Addenda of the ASME Code, or Appendix VIII as implemented by the industry's PDI program. As a result, the NRC is unaware of any small entities in this area of expertise that are adversely affected such that they cannot satisfy either Appendix VIII as written or as implemented by PDI and endorsed in the rule.

8. Backfit Analysis

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

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

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

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

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

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

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

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

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

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

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

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

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

Appendix VIII to ASME BPV Code, Section XI, or Appendix VIII as modified during the development of the PDI program will be used to demonstrate the qualification of personnel and procedures for performing nondestructive examination of welds in components of systems that include the reactor coolant system and the emergency core cooling systems in nuclear power facilities. These performance demonstration programs will greatly increase the reliability of detection and sizing of cracks and flaws. Current requirements have been demonstrated not to be able to consistently and accurately identify and size cracks and flaws and thus are not effective. The Appendix delineates a method for qualification of the personnel and procedures. Appendix VIII changes the Code rules from a prescriptive set of requirements to a performance based approach that allows for implementation of improved technology without changes to the regulations. Performance demonstration would normally be imposed by the 120-month update requirement but, because of its importance, implementation of Appendix VIII is being expedited by the rulemaking. Because of the fundamental change in the nature of the qualification requirements, Appendix VIII is being considered a backfit. The proposed rule would have required licensees to implement Appendix VIII, including the modifications, for all examinations of the pressure vessel, piping, nozzles, and bolts and studs which occur after 6 months from the date of the final rule. However, based on public comment, the final rule adopts a phased implementation approach for Appendix VIII, ranging from 6 months to 3 years, depending on the supplement. The final rule will not require any change to a licensee's ISI schedule for examination of these components, but will require that the provisions of Appendix VIII as contained in the 1995 Edition with the 1996 Addenda (as supplemented by the final rule) or Appendix VIII as modified during the development of the PDI program (as supplemented by the final rule) be used for all examinations after that date rather than the UT procedures and personnel requirements presently being utilized by licensees.

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

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

9. Small Business Regulatory Enforcement Fairness Act

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

List of Subjects in 10 CFR Part 50

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 552 and 553, the NRC is adopting the following amendments to 10 CFR part 50.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

Authority: Sections 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246 (42 U.S.C. 5841, 5842, 5846).

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

2. Section 50.55a is amended as follows:

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

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

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

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

§ 50.55a Codes and standards.

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

* * * * *

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

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

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

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

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

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

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

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

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

* * * * *

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

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

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

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

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

(C) When the elongation corresponding to a specific load (adjusted for effective wires or strands) during retensioning of tendons differs by more than 10 percent from that recorded during the last measurement, an evaluation must be performed to determine whether the difference is related to wire failures or slip of wires in anchorage. A difference of more than 10 percent must be identified in the ISI Summary Report required by IWA–6000.

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

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

(*2*) The absolute difference between the amount removed and the amount replaced exceeds 10 percent of the tendon net duct volume;

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

(E) For Class CC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report required by IWA–6000:

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

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

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

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

(A) For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA–6000:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(xii) Reserved.

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

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

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

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

(*2*) Leakage through a flange gasket;

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

(*4*) Degraded socket welds.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(*4*) Notches are not permitted.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(*ii*) When the examination volume defined in § 50.55a(b)(2)(xv)(K)(*2*)(*i*) cannot be effectively examined in all four directions, the examination must be augmented by examination from the nozzle bore using a procedure and personnel qualified in accordance with § 50.55a(b)(2)(xv)(K)(*1*).

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

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

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

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

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

TABLE VIII–S7–1—MODIFIED

Flaw Locations and Orientations

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(v) *Subsection ISTD.* Article IWF–5000, ''Inservice Inspection Requirements for Snubbers,'' of the ASME BPV Code, Section XI, provides inservice inspection requirements for examinations and tests of snubbers at nuclear power plants. Licensees may use Subsection ISTD, ''Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants,'' ASME OM Code, 1995 Edition up to and including the 1996 Addenda, in lieu of the requirements for snubbers in Section XI, IWF–5200(a) and (b) and IWF–5300(a) and (b), by making appropriate changes to their technical specifications or licensee controlled documents. Preservice and inservice examinations shall be performed using the VT–3 visual examination method described in IWA–2213.

* * * * *

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

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

* * * * *

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

* * * * *

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

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

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

(B) Pumps and valves, in facilities whose construction permit is issued on or after November 22, 1999, which are classified as ASME Code Class 2 and 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code referenced in paragraph (b)(3) of this section at the time the construction permit is issued.

* * * * *

(4) Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves which are classified as ASME Code Class 1, Class 2 and Class 3 must meet the inservice test requirements, except design and access provisions, set forth in the ASME OM Code and addenda that become effective subsequent to editions and addenda specified in paragraphs (f)(2) and (f)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

* * * * *

(g) * * *

(1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued before January 1, 1971, components (including supports) must meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves, and their supports must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

* * * * *

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

(i) Components (including supports) which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda [6] applied to the construction of the particular component.

* * * * *

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

* * * * *

(iii) Licensees may, but are not required to, perform the surface examinations of High Pressure Safety

Injection Systems specified in Table IWB–2500–1, Examination Category B–J, Item Numbers B9.20, B9.21, and B9.22.

* * * * *

(v) * * *

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

* * * * *

(6) * * *

(ii) * * *

(B) *Expedited examination of containment.*

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

(*2*) Licensees of all operating nuclear power plants shall implement the inservice examinations which correspond to the number of years of operation which are specified in Subsection IWL of the 1992 Edition with the 1992 Addenda in conjunction with the modifications specified in § 50.55a(b)(2)(viii) by September 9, 2001. The first examination performed must serve the same purpose for operating plants as the preservice examination specified for plants not yet in operation. The first examination of concrete must be performed prior to September 10, 2001, and the date of the examination need not comply with the requirements of IWL–2410(a) or IWL–2410(b). The date of the first examination of concrete must be used to determine the 5-year schedule for subsequent examinations subject to the provisions of IWL–2410(c).

* * * * *

(C) *Implementation of Appendix VIII to Section XI.* (*1*) The Supplements to Appendix VIII of Section XI, Division 1, 1995 Edition with the 1996 Addenda of the ASME Boiler and Pressure Vessel Code must be implemented in accordance with the following schedule: Supplements 1, 2, 3, and 8—May 22, 2000; Supplements 4 and 6—November 22, 2000; Supplement 11—November 22, 2001; and Supplements 5, 7, 10, 12, and 13—November 22, 2002.

* * * * *

Dated at Rockville, MD this 26th day of August, 1999.

For the Nuclear Regulatory Commission.

William D. Travers,

Executive Director for Operations.

[FR Doc. 99–24256 Filed 9–21–99; 8:45 am]

BILLING CODE 7590–01–P