NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

[NRC–2009–0359]

RIN 3150–AI72

Approval of American Society of Mechanical Engineers' Code Cases

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is proposing to amend its regulations to incorporate by reference the latest revisions of three regulatory guides (RGs) approving new and revised Code Cases published by the American Society of Mechanical Engineers (ASME). This proposed action would allow nuclear power plant licensees, and applicants for construction permits (CPs), operating licenses (OLs), combined licenses (COLs), standard design certifications, standard design approvals and manufacturing licenses, to use the Code Cases listed in these RGs as alternatives to engineering standards for the construction, inservice inspection (ISI), and inservice testing (IST) of nuclear power plant components.

This rulemaking also includes consideration of a petition for rulemaking (PRM), PRM–50–89, submitted by Mr. Raymond West. This rulemaking also proposes resequencing NRC's requirements governing Codes and standards in order to comply with the Office of the Federal Register's (OFR) guidelines for incorporation by reference.

DATES: Submit comments by September 9, 2013. Comments received after this date will be considered if it is practical to do so, but the NRC is able to ensure consideration only of comments received on or before this date.

ADDRESSES: You may submit comments by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- *Federal Rulemaking Web site:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2009–0359. Address questions about NRC dockets to Carol Gallagher; telephone: 301–492–3668; email: *Carol.Gallagher@nrc.gov.* For technical questions, contact the individuals listed in the **FOR FURTHER INFORMATION CONTACT** section of this proposed rule.
- *Email comments to: Rulemaking.Comments@nrc.gov.* If you do not receive an automatic email reply confirming receipt, contact the NRC directly at 301–415–1677.
- *Fax comments to:* Secretary, U.S. Nuclear Regulatory Commission at 301–415–1101.
- *Mail comments to:* Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, ATTN: Rulemakings and Adjudications Staff.
- *Hand deliver comments to:* 11555 Rockville Pike, Rockville, Maryland 20852, between 7:30 a.m. and 4:15 p.m. (Eastern Time) Federal workdays; telephone: 301–415–1677.

You may submit comments on the information collections by the methods indicated in the Paperwork Reduction Act Statement.

For additional direction on accessing information and submitting comments, see ''Accessing Information and Submitting Comments'' in the **SUPPLEMENTARY INFORMATION** section of this document.

FOR FURTHER INFORMATION CONTACT: Manash K. Bagchi, Office of Nuclear Reactor Regulation, telephone: 301–415–2905; email: *Manash.Bagchi@nrc.gov;* or Wallace Norris, Office of Nuclear Regulatory Research, telephone: 301–251–7506; email: *Wallace.Norris@nrc.gov*; U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

SUPPLEMENTARY INFORMATION

Executive Summary

The NRC is proposing to amend its regulations to incorporate by reference the latest revisions of three NRC RGs approving new and revised Code Cases published by the ASME. The three RGs that would be incorporated by reference are RG 1.84, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,'' Revision 36, (DG–1230 for this proposed rule); RG 1.147, ''Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,'' Revision 17, (DG–1231 for this proposed rule); and RG 1.192, ''Operation and Maintenance [OM] Code Case Acceptability, ASME OM Code,'' Revision 1 (DG–1232 for this proposed rule). This proposed action would allow nuclear power plant licensees, and applicants for CPs, OLs, COLs, standard design certifications, standard design approvals, and manufacturing licenses, to use the Code Cases listed in these RGs as alternatives to engineering standards for the construction, ISI, and IST of nuclear power plant components.

This rulemaking also includes consideration of PRM–50–89, submitted by Mr. Raymond West, requesting that the NRC amend its regulations to allow consideration of alternatives to the ASME Boiler and Pressure Vessel [BPV] and OM Code Cases. Lastly, this rulemaking proposes resequencing the order of NRC's requirements, governing Codes and standards in order to comply with the OFR guidelines for incorporating by reference.

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID NRC–2009–0359 when contacting the NRC about the availability of information for this proposed rule. You may access information related to this proposed rule, which the NRC possesses and is publicly available, by any of the following methods:

- *Federal Rulemaking Web site:* Go to *http://www.regulations.gov* and search for Docket ID NRC–2009–0359.
- *NRC's Agencywide Documents Access and Management System (ADAMS):* You may access publicly available documents online in the NRC Library at *http://www.nrc.gov/reading-rm/adams.html.* To begin the search, select ''*ADAMS Public Documents,''* and then select ''*Begin Web-based ADAMS Search.''* For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1–800–397–4209, 301–415–4737, or by email to *PDR.Resource@nrc.gov.* The

ADAMS Accession Number for each document referenced in this proposed rule (if that document is available in ADAMS) is provided the first time that a document is referenced. In addition, for the convenience of the reader, the ADAMS Accession Numbers are provided in a table in Section IX, ''Availability of Documents,'' of this document.

- *NRC's PDR:* You may examine and purchase copies of public documents at the NRC's PDR, O1–F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID NRC–2009–0359 in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC will post all comment submissions at *http://www.regulations.gov* as well as enter the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

II. Background

The ASME develops and publishes the ASME Boiler and Pressure Vessel Code (BPV Code), which contains requirements for the design, construction, and ISI of nuclear power plant components, and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), which contains requirements for IST of nuclear power plant components. In response to BPV and OM Code user requests, the ASME develops ASME Code Cases that provide alternatives to BPV and OM Code requirements under special circumstances.

The NRC approves and/or mandates the use of the ASME BPV and OM Code in § 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR) through the process of incorporation by reference. As such, each provision of the ASME Codes incorporated by reference into, and mandated by, 10 CFR 50.55a, ''Codes and standards,'' constitutes a legally-binding NRC requirement imposed by rule. As noted previously, ASME Code Cases, for the most part, represent alternative approaches for complying with provisions of the ASME BPV and OM Codes.

The NRC periodically amends 10 CFR 50.55a to incorporate by reference NRC RGs listing approved ASME Code Cases that may be used as alternatives to the BPV Code and the OM Code. See **Federal Register** notice (FRN), ''Incorporation by Reference of ASME BPV and OM Code Cases'' (68 FR 40469; July 8, 2003).

This rulemaking is the latest in a series of rulemakings that incorporate by reference new versions of several RGs identifying new and revised [1] unconditionally or conditionally acceptable ASME Code Cases that are approved for use. In developing these RGs, the NRC staff reviews ASME BPV and OM Code Cases, determines the acceptability of each Code Case, and publishes its findings in RGs. The RGs are revised periodically as new Code Cases are published by the ASME. The NRC incorporates by reference the RGs listing acceptable and conditionally acceptable ASME Code Cases into 10 CFR 50.55a. Currently, NRC RG 1.84, Revision 35, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III''; RG 1.147, Revision 16, ''Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1''; and RG 1.192, Revision 0, ''Operation and Maintenance Code Case Acceptability, ASME OM Code,'' are incorporated into the NRC's regulations at 10 CFR 50.55a. A request for comment on the draft RGs is published elsewhere in today's **Federal Register** (Docket ID NRC–2009–0359).

This rulemaking also addresses PRM–50–89 that was submitted to the NRC on December 14, 2007, and revised on December 19, 2007, by Mr. Raymond West (ADAMS Accession No. ML073600974). The petition requests that the NRC amend 10 CFR 50.55a to allow NRC authorization of alternatives to NRC-approved ASME BPV and OM Code Cases. This rulemaking includes proposed provisions that address the PRM. A detailed discussion of the PRM is provided in Section IV, ''Petition for Rulemaking (PRM–50–89),'' of this document.

[1] ASME Code Cases can be categorized as one of two types: new or revised. A new Code Case provides for a new alternative to specific ASME Code provisions or addresses a new need. A revised Code Case is a revision (modification) to an existing Code Case to address, for example, technological advancements in examination techniques or to address NRC conditions imposed in one of the regulatory guides that have been incorporated by reference into 10 CFR 50.55a.

III. Discussion

This proposed rule would incorporate by reference the latest revisions of the NRC regulatory guides that list ASME BPV and OM Code Cases the NRC finds to be acceptable or ''conditionally acceptable'' (i.e., NRC-specified conditions). Draft Regulatory Guide (DG)-1230, Regulatory Guide 1.84, Revision 36, (ADAMS Accession No. ML102590003) would supersede the incorporation by reference of Revision 35; DG–1231, RG 1.147, Revision 17, (ADAMS Accession No. ML102590004) would supersede the incorporation by reference of Revision 16; and DG–1232, RG 1.192, Revision 1, (ADAMS Accession No. ML102600001) would supersede the incorporation by reference of Revision 0.

This proposed rule addresses two categories of ASME Code Cases. The first category of Code Cases are the new and revised Section III and Section XI Code Cases listed in Supplements 1 through 10 to the 2007 Edition of the BPV Code, and the OM Code Cases published with the 2002 Addenda through the 2006 Addenda. The second category is the Code Cases that were not addressed in the final rule published on October 5, 2010 (75 FR 61321). The 2010 final rule addressed the new and revised Section III and Section XI Code Cases listed in Supplements 2 through 11 to the 2004 Edition and Supplement 0 to the 2007 Edition of BPV Code. Public comments were received during the proposed rule stage (June 2, 2009; 74 FR 26303) requesting that the NRC include certain revised Code Cases in the final guides that were not listed in the draft guides. The NRC determined that the revised Code Cases represented changes significant enough to warrant broader public participation prior to the NRC making a final determination of them. Accordingly, the NRC is requesting comment on these Code Cases in this proposed rule.

The latest editions and addenda of the ASME BPV and OM Codes that the NRC has approved for use are referenced in 10 CFR 50.55a. The ASME also publishes Code Cases that provide alternatives to existing Code requirements developed and approved by the ASME. The proposed rule would incorporate by reference RGs 1.84, 1.147, and 1.192. The NRC, by incorporating by reference these three RGs, would allow nuclear power plant licensees and applicants for standard design certifications, standard design approvals, manufacturing licenses, applicants for OIs, CPs, and COLs under the regulations that govern license certifications, to use the Code Cases listed in these RGs as suitable alternatives to the ASME BPV and OM Codes for the construction, ISI, and IST of nuclear power plant components. This action would be consistent with the provisions of the National Technology Transfer and Advancement Act of 1995, Public Law 104–113, which encourages Federal regulatory agencies to consider adopting industry consensus standards as an alternative to *de novo* agency development of standards affecting an industry. This action would also be consistent with the NRC policy of evaluating the latest versions of consensus standards in terms of their suitability for endorsement by regulations or regulatory guides.

The NRC follows a three-step process to determine the acceptability of new and revised Code Cases and the need for regulatory positions on the uses of these Code Cases. This process was employed in the review of the Code Cases in Supplements 1 through 10 to the 2007 Edition of the BPV Code and the 2002 Addenda through the 2006 Addenda of the OM Code. The Code Cases in these supplements are the subject of this proposed rule. First, the ASME develops Code Cases through a consensus development process, as administered by the American National Standards Institute (ANSI), which ensures that the various technical interests (e.g., utility, manufacturing, insurance, regulatory) are represented on standards development committees and that their viewpoints are addressed fairly. This process includes development of a technical justification in support of each new or revised Code Case. The ASME committee meetings are open to the public, and attendees are encouraged to participate. Task groups, working groups, and subgroups report to a standards committee. The standards committee is the decisive consensus committee and ensures that the development process fully complies with the ANSI consensus process. The NRC actively participates through full involvement in discussions and technical debates of the task groups, working groups, subgroups, and standards committee regarding the development of new and revised standards.

Second, the standards committee transmits to its members a first consideration letter ballot requesting comment or approval of new and revised Code Cases. To be approved, Code Cases from the first consideration letter ballot must receive the following: (1) Approval votes from at least two thirds of the eligible consensus committee membership, (2) no disapprovals from the standards committee, and (3) no substantive comments from ASME oversight committees such as the Technical Oversight Management Committee (TOMC). The TOMC's duties, in part, are to oversee various standards committees to ensure technical adequacy and provide recommendations in the development of codes and standards, as required. The Code Cases that are disapproved or receive substantive comments from the first consideration ballot are reviewed by the working level group(s) responsible for their development to consider the comments received. These Code Cases may be approved by the standards committee on second consideration with an approval vote by at least two thirds of the eligible consensus committee membership, with no more than three disapprovals from the consensus committee.

Third, the NRC reviews new and revised Code Cases to determine their acceptability for incorporation by reference in 10 CFR 50.55a through the subject RGs. This rulemaking process, when considered together with the ANSI process for developing and approving ASME codes and standards and ASME Code Cases, constitutes the NRC's basis that the Code Cases (with conditions as necessary) provide reasonable assurance of adequate protection to public health and safety.

The NRC reviewed the new and revised Code Cases identified in this proposed rule and concluded, in accordance with the process previously described, that the Code Cases are technically adequate (with conditions as necessary) and consistent with current NRC regulations. Thus, the new and revised Code Cases listed in the subject RGs are approved for use subject to any specified conditions.

A. Code Cases Approved for Unconditional Use

The NRC determined, in accordance with the process previously described for review of ASME Code Cases, that each ASME Code Case listed in Table I is appropriate for incorporation by reference without conditions into the NRC's regulations.

TABLE I

Code Case No.	Supplement	Title
Boiler and Pressure Vessel Code Section III (Addressed in DG–1230/RG 1.84, Table 1)		
N–4–13	5 (07 Edition)	Special Type 403 Modified Forgings or Bars Class 1 and CS, Section III, Division 1.
N–570–2	7 (07 Edition)	Alternative Rules for Linear Piping and Linear Standard Supports for Classes 1, 2, 3, and MC, Section III, Division 1.
N–580–2	4 (07 Edition)	Use of Alloy 600 With Columbium Added, Section III, Division 1.
N–655–1	2 (07 Edition)	Use of SA–738, Grade B, for Metal Containment Vessels, Class MC, Section III, Division 1.
N–708	2 (07 Edition)	Use of JIS G–4303, Grades SUS304, SUS304L, SUS316, and SUS316L, Section III, Division 1.
N–759–2	4 (07 Edition)	Alternative Rules for Determining Allowable External Pressure and Comprehensive Stress for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1.
N–760–2	7 (07 Edition)	Welding of Globe Valve Disks to Valve Stem Retainers, Classes 1, 2, and 3, Section III, Division 1.
N–767	4 (07 Edition)	Use of 21 Cr-6Ni-9Mn (Alloy UNS S21904) Grade GXM–11 (Conforming to SA-182/SA–182M and SA–336/SA–336M), Grade TPXM–11 (Conforming to SA-312/SA–312M) and Type XM–11 (Conforming to SA–666) Material, for Class 1 Construction, Section III, Division 1.
N–774	7 (07 Edition)	Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4,540 kg) and Otherwise conforming to the Requirements of SA–336/SA–336M for Class 1, 2, and 3 Construction, Section III, Division 1.
N–782	9 (07 Edition)	Use of Editions, Addenda, and Cases, Section III, Division 1.

TABLE I—Continued

Code Case No.	Supplement	Title
N–801	4 (10 Edition)	Rules for Repair of N-Stamped Class 1, 2, and 3 Components by Organization Other Than the N Certificate Holder That Originally Stamped the Component Being Repaired, Section III, Division 1.
N–802	4 (10 Edition)	Rules for Repair of Stamped Components by the N Certificate Holder That Originally Stamped the Component, Section III, Division 1.
Boiler and Pressure Vessel Code Section XI (Addressed in DG–1231/RG 1.147, Table 1)		
N–532–5 ..	5 (10 Edition)	Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA–4000 and IWA–6000, Section XI, Division 1.
N–716–1 ..	1 (13 Edition)	Alternative Piping Classification and Examination Requirements, Section XI Division 1.
N–747	9 (04 Edition)	Reactor Vessel Head-to-Flange Weld Examinations Section XI, Division 1.
N–762	1 (07 Edition)	Temper Bead Procedure Qualification Requirements for Repair/Replacement Activities Without Postweld Heat Treatment, Section XI, Division 1.
N–765	8 (07 Edition)	Alternative to Inspection Interval Scheduling Requirements of IWA–2430, Section XI, Division 1.
N–769	8 (07 Edition)	Roll Expansion of Class 1 In-Core Housing Bottom Head Penetrations in BWR's, Section XI, Division 1.
N–773	8 (07 Edition)	Alternatives Qualification Criteria for Eddy Current Examinations of Piping Inside Surfaces, Section XI Division 1.
Code for Operation and Maintenance (Addressed in DG–1232/RG 1.192, Table 1).		
OMN–6	2006 Addenda	Alternate Rules for Digital Instruments.
OMN–8	2006 Addenda	Alternative Rules for Preservice and Inservice Testing of Power-Operated Valves That Are Used for System Control and Have a Safety Function per OM–10.
OMN–14 ..	2004 Addenda	Alternative Rules for Valve Testing Operations and Maintenance, Appendix I, Boiling Water Reactor (BWR) Control Rod Drive Rupture Disk Exclusion.
OMN–16 ..	2006 Addenda	Use of a Pump Curve for Testing.

B. Code Cases Approved for Use With Conditions

The NRC has determined that certain Code Cases, as issued by the ASME, are generally acceptable for use, but that the alternative requirements specified in those Code Cases must be supplemented to provide an acceptable level of quality and safety. Accordingly, the NRC proposes to impose conditions on the use of these Code Cases to modify, limit or clarify their requirements. For each applicable Code Case, the conditions would specify the additional activities that must be performed, the limits on the activities specified in the Code Case, and/or the supplemental information needed to provide clarity. These ASME Code Cases are included in Table 2 of the following: DG–1230 (RG 1.84), DG–1231 (RG 1.147), and DG–1232 (RG 1.192). The NRC's evaluation of the Code Cases and the reasons for the NRC's proposed conditions are discussed in the following paragraphs. Notations have been made to indicate the conditions duplicated from previous versions of the RGs.

The NRC requests public comment on these Code Cases and the proposed conditions. It should also be noted that the following paragraphs only address those Code Cases for which the NRC proposes to impose a condition or conditions that are listed in the RG for the first time (e.g., the conditions on OMN–4, 2004 are identical to those listed in Revision 0 to RG 1.192 on OMN–4, 1999 Addenda).

Section III Code Cases (DG–1230/RG 1.84)

NRC-proposed changes to Tables 1 and 2 of DG–1230/RG 1.84 for Code Cases N–520–2, N–655–1, N–757–1, N–759–2, and N–782, are discussed in this notice under the heading, *NRC Proposals for Code Cases on which NRC Received Public Comments in the 2009 Proposed ASME Code Case Rulemaking.*

Code Case N–60–5

Type: Revised
Title: *Material for Core Support Structures, Section III, Division 1*
Published: Supplement 12, 2001 Edition

The NRC proposes to reinstate a condition on the use of ASME Code Case N–60–5, which in a previous publication was inadvertently deleted. Code Case N–60–5 was originally listed in RG 1.85, ''Materials Code Case Acceptability, ASME Section III, Division 1.'' Two conditions were listed in RG 1.85 for Code Case N–60–5: 1) welding of age-hardenable Alloy SA–453 Grade 660 and SA–637 Grade 688 should be performed when the material is in the solution-treated condition, and 2) the maximum yield strength of strain-hardened austenitic stainless steel should not exceed 90,000 psi in view of the susceptibility of this material to environmentally assisted cracking. Revision 31 of RG 1.85 was last published in May 1999. In June 18, 2004 (69 FR 34202), RG 1.85 was merged into RG 1.84. The combined RG 1.84 now lists all Section III Code Cases, and RG 1.85 is no longer published. When RG 1.85 was merged into RG.1.84, the NRC inadvertently dropped the two conditions applicable to Code Case N–60–5. The NRC is now proposing to reinstate the second of the two conditions by reinstating Code Case N–60–5 in DG–1230/RG 1.84, Table 2, ''Conditionally Acceptable Section III Code Cases.''

The NRC has determined that the first condition, regarding age-hardenable Alloy SA–453 Grade 660 and SA–637 Grade 688, is no longer needed. These alloy materials are used for bolting and pins that are not typically subjected to welding.

The second condition was instituted because operating experience and laboratory testing showed that strain hardened (also known as cold-worked), austenitic stainless steel in excess of 90,000 psi yield strength, is susceptible to environmentally induced cracking. The caution regarding the limit on the maximum yield strength of strain-hardened austenitic stainless steels has been addressed in the Standard Review Plan (SRP) for over 30 years and has been used as guidance by the NRC staff in its review of reactor coolant pressure boundary materials in all new reactors since the condition was inadvertently dropped in RG 1.84. Specifically, the limit is addressed in NUREG–0800, ''Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants SRP Section 4.5.1, Control Rod Drive Structural Materials, and Section 5.2.3, Reactor Coolant Pressure Boundary Materials. In Section II, SRP Acceptance Criteria state the need for such a limitation: ''Laboratory stress corrosion tests and service experience provide the basis for the criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit on the yield strength of 620 MPa (90,000 psi).''

Thus, the technical basis for the condition is well-established and continues to be valid because these materials are used in current reactor designs and may be used in future reactor designs. Accordingly, the NRC proposes to reinstate this condition on Code Case N–60–5 in Table 2 of DG–1230/RG 1.84. A licensee that implemented Code Case N–60–5 after RG 1.84 and RG 1.85 were combined (i.e., Code Case N–60–5 unconditionally approved) would not have to comply with the reinstated condition limiting the maximum yield strength. Two of the five new reactor designs, Economic Simplified Boiling Water Reactor [ESBWR] and US–EPR, specified the use of Code Case N–60–5 during the time period that no conditions were listed in RG 1.84. These new reactor design certifications were reviewed by the NRC staff for conformance with this condition using the guidelines of the SRP. The condition is included in the Design Control Document for each of these two designs. Operating reactor licensees, who specified Code Case N–60–5 during the time that it was unconditionally approved, are required to meet the ISI examinations in ASME Code Section XI, to ensure detection of environmentally assisted cracking that might result from using strain hardened austenitic stainless steels with yield strength in excess of 90,000 psi.

Reinstatement of this condition would not impact combined license applications that are currently under review by the NRC or have been approved. The condition would only apply to those applicants or licensees in the future that implement Code Case N–60–5 in accordance with Revision 36 (or later) of the final RG 1.84.

Code Case N–208–2

Type: Revised

Title: *Fatigue Analysis for Precipitation Hardening Nickel Alloy Bolting Material to Specification SB–637 N07718 for Class 1 Construction, Section III, Division 1*

Published: Supplement 4, January 4, 2008

Figure A, ''Design Fatigue Curve for Nickel-Chromium Alloy 718,'' Code Case N–208–2, presents maximum mean stress curves. The upper-most curve is labeled ''No mean stress or $\sigma_{max} < 100$ ksi.'' The words ''No mean stress'' may be confusing to users and should be implemented with the condition that this means ''Maximum mean stress.'' In addition, the lower-most curve is labeled as ''$\sigma_{y.}$'' which may also be confusing to users. The σ_y should be implemented with the condition that it means $\sigma_{max.}$ Therefore, the NRC proposes to add two conditions to Code Case N–208–2 in Table 2 of DG–1230/RG 1.84 that would provide definitions for ''no mean stress'' and ''σ_{max}'' with respect to Figure A.

Section XI Code Cases (DG–1231/RG 1.147)

NRC-proposed changes to Tables 1 and 2 of DG–1231/RG 1.147 for Code Cases N–508–4, N–597–2, N–619, N–648–1, and N–702, are discussed in this notice under the heading, *NRC Proposals for Code Cases on which NRC Received Public Comments in the 2009 Proposed ASME Code Case Rulemaking.*

Code Case N–561–2 [Supplement 1]

Type: Revised

Title: *Alternative Requirements for Wall Thickness Restoration of Class 2 and High Energy Class 3 Carbon Steel Piping, Section XI, Division 1*

Published: Supplement 1, 2007 Edition

The original version and first version of this Code Case were not approved by the NRC for use. The NRC's basis for not approving the use of this Code Case was that: 1) no criteria for determining the rate or extent of degradation of the repair of the wall thickness restoration or the surrounding base metal were provided, and 2) re-inspection requirements were not provided to verify structural integrity since the root cause may not be mitigated. The ASME made significant technical revisions to previous versions of this Code Case by applying the findings from a very similar application (i.e., Code Case N–661, ''Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service'').

A request to apply Code Case N–661 at the Edwin I. Hatch Nuclear Power Plant (Hatch Plant) was conditionally approved by the NRC in the Hatch Safety Evaluation Report (SER) (ADAMS Accession No. ML033280037). Code Case N–661 was subsequently approved with the same conditions in RG 1.147, Revision 15. The ASME used these same conditions in revising Code Case N–561–1 resulting in Code Case N–561–2. Based on the NRC staff's review of Code Cases N–561–2 and N–661, and on its experience applying Code Case N–661 at the Hatch Plant, the NRC proposes to approve Code Case N–561–2 with certain conditions. This is reflected in Table 2 of DG–1231. Five proposed conditions on this Code Case will be listed in Table 2 of DG–1231/RG 1.147. The proposed conditions are discussed in this section.

The provisions of Code Cases N–561–2 and N–661–1 are similar in that the Code Cases apply to similar systems (i.e., Class 2 and High Energy Class 3 Carbon Steel Piping, Class 3 Moderate Energy Carbon Steel Piping, and Carbon Steel Piping for Raw Water Service). The provisions were developed by the ASME to perform an alternative repair of degraded components, which involves the application of weld metal overlay on the exterior of the piping system to restore the wall thickness of the component. Accordingly, the conditions identified in the SER regarding Code Case N–661 also apply to Code Case N–561–2.

One of the conditions in the SER addressed the time period for which the repair would be considered acceptable. The definition established by the NRC was modified when added to Code Case N–561–2. In Code Case N–661, the repair is only acceptable until the ''next refueling outage.'' In contrast, Code Case N–561–2 states that the repair would be acceptable for ''one fuel cycle.'' The NRC believes that it is unclear in Code Case N–561–2 what one fuel cycle actually infers if a repair is performed at mid-cycle.

It could be interpreted that the repair is acceptable for the remainder of the current fuel cycle plus the subsequent fuel cycle. This interpretation could double the time period. The NRC established this limitation on the acceptable life of the repair of the five because the Code Case does not require that the root cause of the degradation be determined. If the root cause of the degradation has not been determined, a suitable reinspection frequency cannot be established. In addition, the Code Case would allow repairs to be made by welding on surfaces that are wet or exposed to water. Performing through-wall weld repairs on surfaces that are wet or exposed to water would greatly increase the chances of producing welds that include weld defects such as porosity, lack of fusion, and cracks. It is highly unlikely that a weld can be made on an open root joint with water present on the backside of the weld without having several weld defects. These types of weld defects can, and many times do, lead to premature failure of a weld joint.

Accordingly, the NRC is proposing on Code Case N–561–2 two of the five conditions (identified as Conditions 1 and 3) in the DG–1231 to address these concerns. The first proposed condition addresses those situations where welds are fabricated with water present on the backside, defects are likely, and the service life time would be expected to be greatly reduced: ''Paragraph 5(b): [of Code Case N–561–2] for repairs performed on a wet surface, the overlay is only acceptable until the next refueling outage.'' A second proposed condition is being added on Code Case N–561–2 that would not allow the exemption in Paragraph (6)(c)(1). Paragraph (6)(c)(1) states that ''Class 3 weld overlays are exempt from volumetric examination when the Construction Code does not require that full-penetration butt welds in the same location be volumetrically examined.'' Many licensees are mitigating stress corrosion cracking through the addition of a weld overlay on the outside of the piping. The purpose of the overlay is to restore wall thickness. The NRC has approved this mitigation technique provided that the full thickness of the weld overlay as well as a certain portion of the base material can be volumetrically examined. The exemption in Paragraph (6)(c)(1) conflicts with the NRC position on this matter, and thus the third condition is proposed requiring the performance of a volumetric examination of the weld overlay.

The third proposed condition on Code Case N–561–2 is: ''Paragraph 7(c): if the cause of the degradation has not been determined, the repair is only acceptable until the next refueling outage.''

The fourth condition on Code Case N–561–2 is proposed to address the NRC's concern that a preexisting flaw could grow through-wall after application of a weld overlay: ''The area where the weld overlay is to be applied must be examined using ultrasonic methods to demonstrate that no crack-like defects exist.'' The basis for this proposed condition is discussed in detail here. Weld overlays have been used as a mitigation method and as a repair method to address stress corrosion cracking in piping butt welds. The basis for applying a weld overlay is that it will result in compressive residual stresses on the inside surface of the pipe, thus preventing a flaw from growing. Analytical modeling has been used to predict post-weld repair residual stress distributions for common piping configurations. Many times, however, weld records are not available or are not complete with regard to weld repairs made during construction. The investigations using modeling to predict the residual stresses resulting from weld repairs have used various assumptions to address the lack of data from weld records.

This raises a question whether a model can accurately predict residual stresses if the extent of repairs is unknown. Factors such as the number of weld passes, welding sequence, and heat input can greatly influence stress patterns. Thus, analytical modeling of typical piping weld configurations with a weld overlay has been used to determine whether application of a weld overlay would result in compressive residual stresses and impede the growth of a preexisting flaw. Because of the many assumptions that might be required, configurations have been analyzed with up to a 75 percent through thickness flaw.

While the results of the analyses performed have shown that a weld overlay could produce compressive stresses on the inside diameter of the piping for repairs as great as 75 percent through-wall, the NRC continues to be concerned regarding the lack of repair information. For example, an investigation into a leak that occurred several years ago showed that at least four weld repairs had been performed. This case is not believed to be unique. Thus, the NRC does not believe that the analyses that have been conducted to date are bounding, nor that the analyses have demonstrated that a preexisting flaw would not continue to grow circumferentially and perhaps through-wall after application of a weld overlay. Accordingly, the NRC proposes that it must be shown, using ultrasonic methods that no flaws exist in the area where the weld overlay is to be applied.

The fifth and last condition being proposed on Code Case N–561–2 is ''Paragraph 4(b): All systems must be depressurized before welding.'' The need for this condition is the same as that for the first proposed condition, i.e., the Code Case would allow repairs to be made by welding on surfaces that is wet or exposed to water. As previously discussed, it is highly unlikely that a weld can be made on an open root joint with water present on the backside of the weld without having several weld defects, and these types of weld defects can lead to premature failure of a weld joint. Thus, depressurizing the system would decrease the chances of producing a suspect weld.

Code Case N–562–2

Type: Revised

Title: *Alternative Requirements for Wall Thickness Restoration of Class 3 Moderate Energy Carbon Steel Piping, Section XI, Division 1*

Published: Supplement 1, 2007 Edition

Code Case N–562–2 is nearly identical to Code Case N–561–2, which is discussed separately herein. The principal difference between the Code Cases is that N–562–2 addresses lower energy piping. However, the same concerns previously discussed regarding Code Case N–561–2 also apply to Code Case N–562–2. Accordingly, the same five conditions are being proposed for Code Case N–562–2.

Code Case N–661–2

Type: Revised

Title: *Alternative Requirements for Wall Thickness Restoration of Classes 2 and 3 Carbon Steel Piping for Raw Water Service, Section XI, Division 1*

Published: Supplement 1, 2007 Edition

As previously discussed with respect to Code Case N–561–2, Code Case N–661–2 is very similar to the other two Code Cases addressing restoration of wall thickness (namely N–561–1 and N–562–2), except that N–661–2 addresses raw water service systems.

Conditions (1) and (3) in draft Revision 17 to RG 1.147 for Code Case N–661–2 were listed in Revision 16 to RG 1.147. Those conditions are: (1) Paragraph 4(b): for repairs performed on a wet surface, the overlay is only acceptable until the next refueling outage; and (3) paragraph 7(c): if the cause of the degradation has not been determined, the repair is only acceptable until the next refueling outage. As previously indicated in the discussion addressing Code Case N–561–2, the ASME made significant technical revisions to Code Cases N–561–1, N–562–1, and N–661–1. Consistent with the technical justification addressing the proposed conditions for Code Case N–561–2, the NRC is proposing three new conditions for Code Case N–661–2. Those conditions are listed in draft Revision 17 to RG 1.147 as following: (2) Paragraph 6(c)(1): this exemption is not permitted; (4) The area where the weld overlay is to be applied must be examined using ultrasonic methods to demonstrate that no crack-like defects exist; and (5) All systems must be depressurized before welding.

Code Case N–739–1 [Supplement 1]

Type: Revised

Title: *Alternative Qualification Requirements for Personnel Performing Class CC Concrete and Post-Tensioning System Visual Examinations, Section XI, Division 1*

Published: Supplement 1, 2007 Edition

The original version of this Code Case was not approved by the NRC for use. The NRC had concerns regarding the lack of detail provided on the instructional material to be covered in the qualification of personnel performing these inspections. The revised Code Case includes detailed instructional material regarding requirements for training. The NRC finds the added requirements to be acceptable. However, the reference in the Code Case to the American Concrete Institute (ACI) standard has been printed incorrectly. To ensure that the correct instructional material is used, the NRC is proposing to conditionally approve Code Case N–739–1 to indicate that the correct ACI reference is 201.1.

OM Code Cases (DG–1232/RG 1.192)

Code Case OMN–1

Type: Revised

Title: *Alternative Rules for Preservice and Inservice Testing of Active Electric Motor-Operated Valve Assemblies in Light-Water Reactor Power Plants*

Published: 2006 Addenda

Proposed Revision 1 to RG 1.192 does not modify the conditions imposed on the implementation of Code Case OMN–1 that were listed in Revision 0 to RG 1.192, issued June 2003. The following discussion is included in the proposed rule to emphasize that caution is required when using risk insights to evaluate the performance of MOVs that have exercise intervals extended from quarterly to every refueling outage.

In 1996, ASME issued Code Case OMN–1 that allows quarterly stroke-time testing of motor-operated valves (MOVs) in the IST program to be replaced by a program of exercising on a refueling outage frequency and periodic diagnostic testing at intervals up to 10 years. In 1999, the NRC accepted the use of Code Case OMN–1 with conditions in 10 CFR 50.55a(b)(3)(iii) as an alternative to the requirement in 10 CFR 50.55a(b)(3)(ii) that licensees shall comply with the provisions for MOV stroke-time testing in the OM Code and shall establish a program to ensure that MOVs continue to be capable of performing their design-basis safety functions.

In June 2003, the NRC staff developed RG 1.192 and transferred the acceptance of Revision 0 to Code Case OMN–1 from 10 CFR 50.55a(b)(3)(iii) to RG 1.192 with the following conditions. Those conditions are:

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

(2) When extending exercise test intervals for high risk MOVs beyond a quarterly frequency, licensees must ensure that the potential increase in Core Damage Frequency (CDF) and risk associated with the extension is small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(3) When applying risk insights as part of the implementation of OMN–1, licensees must categorize MOVs according to their safety significance using the methodology described in Code Case OMN–3, ''Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants,'' with the conditions discussed in RG 1.192 or use other MOV risk-ranking methodologies accepted by the NRC on a plant-specific or industry-wide basis with the conditions in the applicable safety evaluations.

Licensees may use Code Case OMN–1 in lieu of the provisions for stroke-time testing in Subsection ISTC of the 1995 Edition up to and including the 2000 Addenda of the ASME OM Code when applied in conjunction with the provisions for leakage rate testing in, as applicable, ISTC 4.3 (1995 Edition with the 1996 and 1997 Addenda) and ISTC–3600 (1998 Edition through the 1999 and 2000 Addenda). In addition, licensees who continue to implement Section XI of the ASME BPV Code as their Code of Record may use OMN–1 in lieu of the provisions for stroke-time testing specified in Paragraph 4.2.1 of ASME/ANSI OM Part 10 as required by 10 CFR 50.55a(b)(2)(vii) subject to the conditions in this regulatory guide. Licensees who choose to apply OMN–1 must apply all its provisions.

It should be noted that ASME issued Code Case OMN–11, ''Risk-Informed Testing for Motor-Operated Valves,'' in the 2001 Edition to provide more specific provisions for the application of risk insights as part of the MOV diagnostic testing alternative allowed in Code Case OMN–1. The NRC accepted the use of OMN–11 in Revision 0 of RG 1.192 with conditions related to determination of acceptable MOV test intervals based on diagnostic data, evaluation of test results for grouped low-risk MOVs, and extension of the exercise interval for high-risk MOVs similar to the condition in RG 1.192 for Code Case OMN–1.

In the 2006 Addenda to the ASME OM Code, ASME issued an updated version of Code Case OMN–1 to clarify the guidance for users of the code case. In its updated version, Code Case OMN–1 incorporates the provisions of Code Case OMN–11 for applying risk insights as well as the conditions specified in the June 2003 version of RG 1.192 for the use of Code Case OMN–11.

The NRC staff is not proposing to modify the conditions for the acceptability of Code Case OMN–1 based on the incorporation of provisions for applying risk insights from OMN–11. However, based on operating experience at nuclear power plants, the NRC emphasizes the importance of evaluating the performance of MOVs that have exercise intervals extended from quarterly to every refueling outage. As discussed in **Federal Register** Notice 51370 (dated September 22, 1999) on page 51386, and which the NRC finds is still applicable when using the 2006 version of Code Case OMN–1, the licensee should 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. This information may be obtained by grouping similar MOVs and staggering the exercising of the MOVs in the group equally over the refueling interval. Licensees are cautioned that, when implementing 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.

Appendix B, ''Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,'' to 10 CFR part 50 requires nuclear power plant licensees to evaluate deficiencies in the performance of safety-related MOVs. Where degradation in the performance of a high-risk MOV is identified when exercised or tested at an extended interval, licensees should reapply the quarterly frequency for the exercise test interval for all high-risk MOVs and implement diagnostic testing of those MOVs at an interval that provides assurance of their design-basis capability throughout the test interval. Licensees should also incorporate the performance results for all MOVs into the probabilistic risk analysis to determine whether the risk ranking of

MOVs should be modified based on those results.

For additional information on OMN–1, see the discussion on OMN–4 and OMN–12 below.

Code Case OMN–3

Type: Revised

Title: *Requirements for Safety Significance Categorization of Components Using Risk Insights for Inservice Testing of LWR Power Plants*

Published: 2004 Edition

The NRC initially issued RG 1.192 in June 2003 accepting several ASME OM Code Cases, including Code Case OMN–3. Subsequently, on December 18, 2003, the Commission issued Staff Requirements Memorandum (SRM) COMNJD–03–0002, ''Stabilizing the PRA Quality Expectations and Requirements'' (ADAMS Accession No. ML033520457), which approved implementation of a phased approach to achieving an appropriate quality for probabilistic risk assessments (PRAs) for the NRC's risk-informed decisionmaking. In SECY–04–0118 dated July 13, 2004 (ADAMS Accession No. ML041470505), the NRC staff described its action plan to implement the SRM, which the Commission subsequently approved in an SRM dated October 6, 2004 (ADAMS Accession No. ML042800369).

The central concept of the action plan specifies the development of consensus PRA standards and associated industry guidance documents, as discussed in RG 1.200 (March 2009), ''An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.'' RG 1.200 clarifies that the staff anticipates that current good practice, (i.e., Capability Category II (CCII)) as explained in the appendices of RG 1.200, is the level of technical adequacy that is sufficient for the majority of applications. RG 1.200 provides that licensees evaluate all deviations from CCII or higher and document why the PRA is sufficient for the proposed application.

In a related action, the Commission published Section 69, ''Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors'' in 10 CFR part 50 on November 22, 2004. RG 1.201 (May 2006), ''Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to their Safety Significance,'' describes one acceptable method to categorize the safety significance of active components. Section 50.69 specifies high level treatment requirements for low risk SSCs whereas SSC treatment is prescribed in more detail in several risk-informed ASME OM Code Cases.

Based on a consideration of the information in Section 69 in 10 CFR part 50 and in the RG 1.201, the NRC proposes Conditions (5), (6) and (7) in RG 1.192 to its acceptance of Code Case OMN–3 included in the 2004 Edition of the ASME OM Code. Licensees applying Code Case OMN–3, 2004 Edition, will need to apply the conditions specified in the previous version of RG 1.192 issued in June 2003, and new Conditions (5), (6) and (7) discussed in this section. As stated in RG 1.192, if a licensee implements a Code Case and a later version of the Code Case is incorporated by reference into 10 CFR 50.55a and listed in Tables 1 and 2 during the licensee's present 120-month IST program interval, that licensee may use either the later version or the previous version. An exception to this provision would be the inclusion of a limitation or condition on the use of Code Case that is necessary, for example, to enhance safety. The NRC staff has determined that a licensee currently using Code Case OMN–3 must use the later version of the Code Case listed in Table 2 of RG 1.192, Revision 1, after it is incorporated by reference into 10 CFR 50.55a.

Condition (5) specifies that the implementation of Section 3.2, ''Plant Specific PRA,'' in Code Case OMN–3 must be consistent with the guidance that the Owner is responsible for demonstrating and justifying the technical adequacy of the PRA analyses used as the basis to perform component risk ranking and for estimating the aggregate risk impact. Condition (5) references RG 1.200 and 1.201 for guidance in satisfying this condition. For example, RG 1.200 includes descriptions of technical adequacy of PRA analyses beyond those modeling only internal initiating events, (e.g., for seismic and internal fire initiating events). RG 1.201 endorses the guidance described by the Nuclear Energy Institute (NEI) in Revision 0 to NEI 00–04, ''10 CFR 50.69 SSC Categorization Guideline,'' dated July 2005. This document describes how the importance of components relied on for seismic, fires, and other initiating events (and operating modes) should be included in the categorization process, including if no plant-specific PRA is available for the hazard.

Condition (6) specifies that paragraph (b) in Section 4.2.4, ''Reconciliation,'' in Code Case OMN–3 is not endorsed. Condition (6) states that the expert panel may not classify components that are ranked as a High Safety Significant Component (HSSC) by the results of a qualitative or quantitative PRA evaluation (excluding the sensitivity studies) or the defense-in-depth assessment to a Low Safety Significant Component (LSSC). RG 1.201 clarifies that a component, identified as high safety significant by any of the PRA (excluding the sensitivity studies) or defense-in-depth evaluations may not be re-categorized to low safety significant by the expert panel. The position in RG 1.201 that an expert panel may not decide the PRA or defense-in-depth evaluations are in error and lower the safety significance assigned according to these evaluations is applicable to OMN–3 deliberations. Rather, the expert panel should provide information regarding its views to the PRA analysts so that the evaluations can be re-performed, if appropriate, to address the expert panel issue or document the appropriateness of the current analysis results.

Condition (7) specifies that implementation of Section 3.3, ''Living PRA,'' in Code Case OMN–3 must be consistent with the following: (1) to account for potential changes in the failure rates and other changes that could affect the PRA, changes to the plant must be reviewed, and, as appropriate, the PRA updated; (2) when the PRA is updated, the SSC categorization must be reviewed and changed if necessary to remain consistent with the categorization process; and (3) the review of plant changes must be performed in a timely manner and must be performed once every two refueling outages or as required by 50.71(h)(2) for COL holders. Changes to the plant, including potential changes in failure rates, might affect the PRA evaluations, and changes to the PRA evaluations might affect the safety significance of the components developed from these evaluations. Therefore, the PRA must be periodically updated and the risk categorization reviewed when the PRA is updated. The period of two refueling outages as the maximum period between determinations of whether a PRA update is needed is consistent with the time span in 10 CFR 50.69.

Code Case OMN–3 addresses safety significance categorization of components using risk insights as applied to inservice testing. Several new conditions are proposed with respect to Code Case OMN–3 (discussed earlier) that reflect current NRC regulatory positions on determining PRA technical adequacy when using risk insights in regulatory applications. Code Cases OMN–1, OMN–4, 2004 Edition, and OMN–12, 2004 Edition, also address the use of risk insights for inservice testing. Accordingly, to ensure consistent implementation among these Code Cases, a note has been added to Code Case OMN–4 and OMN–12. Paragraph 3.1 of Code Case OMN–12 states that ''Valve assemblies shall be classified as either high safety significant or low safety significant in accordance with Code Case OMN–3.'' However, given the interdependence of Code Cases OMN–1, OMN–3, OMN–4, and OMN–12, Note 3 has been added to Code Case OMN–12 as a reminder of the dependence on Code Case OMN–3 (i.e., paragraph 3.1). In addition, Note 2 has been added to Code Case OMN–4 as a reminder that the conditions with respect to allowable methodologies for OMN–3 risk ranking specified for the use of OMN–1 also apply to OMN–4.

C. NRC Proposals for Code Cases on Which NRC Received Public Comments in the 2009 Proposed ASME Code Case Rulemaking

On June 2, 2009, the NRC published a proposed rule (74 FR 26303) and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comments on incorporating by reference draft RG 1.84, Revision 35, and draft RG 1.147, Revision 16. The NRC received public comments on draft Revision 35 to RG 1.84 and draft Revision 16 to RG 1.147 requesting that certain revised Code Cases that were not listed in those draft guides be approved in the final guides. These revised Code Cases that were the subject of comment in 2009 are N–520–2, N–655–1, N–757–1, N–759–2, and N–782 for RG 1.84; and Code Cases N–508–4, N–597–2, N–619, N–648–1, and N–702 for RG 1.147. In that earlier rulemaking, the NRC determined that the revised Code Cases represented changes significant enough to warrant broader public participation prior to the NRC making a final determination of them. Therefore, the final RG 1.84 and RG 1.147 associated with the 2010 final rule (75 FR 61321; October 5, 2010) did not include these Code Cases.

The NRC has reviewed these Code Cases, and now proposes to approve those Code Cases, in some cases with conditions. These Code Cases are discussed in this section, under the applicable draft regulatory guide.

Section III Code Cases (DG–1230/RG 1.84)

Code Case N–520–2

Type: Revised

Title: *Alternative Rules for Renewal of Active or Expired N-type Certificates for Plants Not in Active Construction*

Published: Supplement 4, 2007 Edition

Code Case N–520–1, the predecessor of Code Case N–520–2, was unconditionally approved in Revision 34 to RG 1.84. The objective of Code Case N–520–1 was to address situations where construction was halted on a nuclear power plant, interrupting ASME Code activities, but the Certificate Holder had maintained its certificate. Code Case N–520–1 provided guidance on what a Certificate Holder had to do to document and stamp the completed construction work. On June 2, 2009, the NRC published a proposed rule (74 FR 26303) and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comment on draft RG 1.84, Revision 35. The NRC received a public comment requesting that the NRC approve Code Case N–520–2 for inclusion in final Revision 35, noting that Code Case N–520–2 had been approved by the ASME on November 1, 2007, and published in Supplement 4 to the 2007 Edition. Code Case N–520–2 was developed to allow an organization with an expired certificate to secure an ASME Temporary Certificate of Authorization. Because Code Case N–520–2 was not part of the June 2009 proposed rule and the changes reflected in N–520–2 were significant, the NRC did not adopt the public comment to list Code Case N–520–2 in final Revision 35 to RG 1.84 (incorporated by reference in the final rule published on October 5, 2010 (75 FR 61321)).

The NRC has now determined that the provisions of Code Case N–520–2 are adequate for addressing a situation where a Certificate Holder has let its N-type certificates expire. The basis for this determination is that all completed in-process work must be clearly documented to ensure that remaining activities and Code responsibilities are readily identifiable. In addition, the ASME Temporary Certificate of Authorization is for the sole purpose of completing the required documentation and component stamping. Finally, this work must be completed under a contract with an Authorized Nuclear Inspection Agency (ANIA).

The NRC is proposing to conditionally approve Code Case N–520–2 because it believes that the wording of the Code Case may create confusion regarding the relationship between the ANIA and the Authorized Nuclear Inspector (ANI). The purpose of the condition in Table 2 of DG1230/RG 1.84, Revision 36, is to clearly indicate that the ANIA employs the ANI.

Code Cases N–655–1, N–757–1, N–759–2, N–782

A comment responding to the June 2, 2009, proposed rule (74 FR 26303) and a parallel notice of availability of draft RGs (74 FR 26440), requested that the following four Code Cases used in the AP–1000 design that were not included in draft Revision 35 of RG 1.84 be included in the final guide: Code Case N–655–1, ''Use of SA–738, Grade B, for Metal Containment Vessels, Class MC, Section III, Division 1;'' Code Case N–757–1, ''Alternative Rules for Acceptability for Class 2 and 3 Valves, NPS 1 (DN25) and Smaller with Welded and Nonwelded End Connections other than Flanges, Section III, Division 1;'' Code Case N–759–2, ''Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1;'' and Code Case N–782, ''Use of Code Editions, Addenda, and Cases Section III, Division 1.'' Draft Revision 35 of RG 1.84 considered Code Cases published up to Supplement 0 to the 2007 Edition. Code Cases N–655–1 and N–757–1 were published in Supplement 2 to the 2007 Edition. Code Case N–759–2 was published in Supplement 4 to the 2007 Edition. Code Case N–782 was published in Supplement 9 to the 2007 Edition. These four Code Cases were beyond the scope of the draft RG and thus had not been considered for inclusion in the draft RG.

The NRC did not include these four Code Cases in final Revision 35 of RG 1.84 because it would have been inappropriate to include them in the final RG without providing the public an opportunity for comment. In addition, these Code Cases were not referenced in the latest AP–1000 Design Control Document.

Code Cases N–655–1, N–759–2, and N–782 have been reviewed by the NRC and have been found to be acceptable. Accordingly, these Code Cases are listed in Table 1 of DG–1230/RG 1.84, Revision 36, and the NRC proposes to unconditionally approve them, as presented in Table I under ''Code Cases Approved for ''Unconditional Use''.

Code Case N–757–1 was reviewed and found to be conditionally acceptable. It is listed in Table 2 of the DG–1230/RG 1.84. The proposed condition for Code Case N–757–1 is discussed in the following discussion.

Code Case N–757–1 [Supplement 2]

Type: Revised

Title: *Alternative Rules for Acceptability for Class 2 and 3 Valves NPS 1 (DN 25) and Smaller with Welded and Nonwelded End Connections Other than Flanges, Section III, Division 1*

Published: Supplement 2, 2007 Edition

The NRC proposes to impose a condition on Code Case N–757–1 in Table 2 of RG 1.84 to prohibit the use of the design provisions in ASME Section III, Division 1, Appendix XIII, for Class 3 valves. This would be accomplished by adding the condition to Table 2 of DG–1230/RG 1.84. The Code Case addresses the use of instrument, control, and sampling line valves, NPS 1 (DN 25) and smaller, with nonwelded end connections other than ASME B16.5 flanges for Section III, Division 1, Class 2 and Class 3 construction. The Code Case provides three options for the design of Class 2 and Class 3 valves that do not meet the minimum thickness requirements in ASME B16.34. These options include the following: 1) the pressure design rules of Section III, paragraphs NC–3324 and ND–3324; 2) the experimental stress analysis rules in Section III, Appendix II; or 3) design based on the stress analysis rules in Section III, Appendix XIII.

The NRC finds that the first option provides an acceptable alternative basis for the design of ASME Class 2 and Class 3 valves because it provides adequate design margin by using the vessel design rules accepted by the NRC in 10 CFR 50.55a. The second option is also acceptable for the design of ASME Class 2 and Class 3 valves because it allows the designer to use experimental stress analysis techniques to establish that the design provides acceptable ASME Code margins for parts in which theoretical stress analysis might not be possible or practical. The third option, however, is not acceptable to the NRC.

Option 3 would allow a designer to use the criteria provided in Section III, Division 1, Appendix XIII. As defined by the scope of Appendix XIII, these Code rules are only applicable to the design of Class 2 vessels meeting the requirements of NC–3200. Further, Appendix XIII provides for design based on a stress analysis that uses criteria similar to that used for the design of ASME Class 1 components (including the ASME Class 1 stress intensity allowable limits). The stress intensity values in the acceptance criteria are greater than the allowable stress intensity values specified for the design of ASME Class 3 components. The NRC concludes that the criteria in Appendix XIII are not intended for the design of ASME Class 3 components, including the valves within the scope of N–757, and that a condition should be added to Table 2 of DG–1230/RG 1.84 that prohibits the use of these design provisions for Class 3 valves.

It should be noted that the NRC staff approved this Code Case as it was considered by the cognizant ASME committees. However, upon further consideration as Code Cases were reviewed for inclusion in the subject RGs, the NRC determined that use of the Code Case was inappropriate for ASME Class 3 components. Therefore, the NRC proposes to impose a condition that would prohibit the use of the design provisions in ASME Section III, Division 1, Appendix XIII, for Class 3 valves.

Section XI Code Cases (DG–1231/RG 1.147)

Code Case N–508–4

Type: New

Title: *Rotation of Snubbers and Pressure Retaining Items for the Purpose of Testing or Preventive Maintenance, Section XI, Division 1*

Published: Supplement 8, 2007 Edition

Code Case N–508–3, the predecessor of Code Case N–508–4, was unconditionally approved in Revision 15 to RG 1.147. The objective of Code Case N–508–3 was to provide guidance on rotating snubbers and relief valves from stock for the purpose of testing or preventive maintenance. On June 2, 2009, the NRC published a proposed rule (74 FR 26303) and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comment on draft RG 1.147, Revision 16. The NRC received a public comment noting that Code Case N–508–4 had been approved by the ASME on January 26, 2009, and published in Supplement 8 to the 2007 Edition, and requesting that the NRC approve Code Case N–508–4 in final Revision 16 rather than cease approval at Code Case N–508–3. Code Case N–508–4 significantly expands the list of components that may be rotated from stock for the purpose of testing or preventive maintenance (adds pumps, control rod drive mechanisms, and pump seal packages).

Because Code Case N–508–4 was not part of the June 2009 proposed rule and the changes reflected in N–508–4 were significant, the NRC did not adopt the public comment to list Code Case N–508–4 in final Revision 16 to RG 1.147 (incorporated by reference in the final rule published on October 5, 2010 (75 FR 61321)). Instead, this Code Case is addressed in draft Revision 17 to RG 1.147.

The NRC has not identified any technical reasons why additional components may be considered for the purpose of testing or preventive maintenance as described in the Code Case N–508–4. However, the NRC has identified an issue and proposes to condition Code Case N–508–4 to ensure that there is no conflict regarding the application of this Code Case. When Section XI is used to govern snubber examination and testing, Footnote 1 (which was later added to the Code Case) conflicts with Subsection IWF, Section XI, up to and including the 2004 Edition through the 2005 Addenda. Footnote 1 directs the user to implement the OM Code for snubber examination and testing. The OM Code was developed in order to have a separate Code for the development and maintenance of provisions for the IST of pumps and valves. In 1990, the ASME published the initial edition of the OM Code, thereby transferring responsibility for these provisions from Section XI to the OM Code Committee. While the use of the OM Code is an option under paragraph (b)(3)(v)(A), the examination and testing requirements for snubbers are also provided in the 2005 Addenda and earlier editions and addenda of Section XI. Thus, there is a conflict for editions and addenda up to the 2005 Addenda of Section XI, but there is no conflict for licensees who have adopted the 2006 Addenda or later editions and addenda of Section XI.

To resolve the conflict, the NRC is proposing to include in DG–1231/RG 1.147, Revision 17, a condition to Code Case N–508–4 stating that Footnote 1 to the Code Case would not apply when the ISI Code of record is earlier than Section XI, 2006 Addenda, and Section XI requirements are used to govern the examination and testing of snubbers.

Code Case N–597–2

Type: Revised

Title: *Requirements for Analytical Evaluation of Pipe Wall Thinning*

Listed: Revision 15 to RG 1.147

Published: November 18, 2003

Code Case N–597–2 was conditionally approved in Revision 15 to RG 1.147. Two comments responding to the proposed rule published on June 2, 2009 (74 FR 26303), and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comment on draft RG 1.147, Revision 16, suggested that the method in Code Case N–513–2, ''Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping,'' used to evaluate local degradation, should be approved by the NRC for application to Code Case, N–597–2. The comments argued that the NRC has conditionally approved Code Case N–513–2 with an evaluation methodology to allow licensees to temporarily accept flaws in moderate energy Class 2 or 3 piping, whereas condition (2) on Code Case N–597–2 requires NRC approval for any amount of local degradation beyond that calculated by the hoop stress equation.

Because Code Case N–513–2 was not part of the June 2, 2009, proposed rule and the changes reflected in N–513–2 are significant, the NRC did not adopt the public comments to allow the Code Case N–513–2 evaluation to also be used with respect to Code Case N–597–2. While the NRC agrees that the flaw evaluation methodology for analyzing piping degradation contained in Code Case N–513–2 could under certain circumstances be applied for a Code Case N–597–2 evaluation (i.e., both Code Cases address the analytical evaluation of pipe wall thinning), the NRC disagrees with the comments that through-wall leakage should be included in the scope of such an evaluation. Code Case N–597–2 was not developed to address leakage; it is focused only on analytical evaluation of wall thinning. The comments discuss local degradation up to and including through-wall leakage and believe it would be appropriate to allow such leakage for all ASME Code class components. This implies that such leakage from high temperature, high pressure systems is no different from leakage from low temperature, low pressure systems. Permitting degradation up to and including through-wall leakage in certain systems would violate 10 CFR part 50, appendix A, Criterion General Design Criteria (GDC) 14, ''Reactor coolant pressure boundary,'' and/or similar provisions in the licensing basis for these facilities, which require that the reactor coolant pressure boundary be tested to ensure an extremely low probability of abnormal leakage, of propagating failure, and of gross rupture. In addition, there have been pipe breaks and leakage in high temperature, high pressure lines throughout the world and some have been sudden and catastrophic. Code Case N–597–2 is applicable to all ASME Code class piping, including high energy piping; whereas, Code Case N–513–2 is limited to Class 2 and 3 moderate energy piping. The NRC has only approved temporary acceptance of flaws for moderate energy Class 2 or 3 piping (maximum operating temperature does not exceed 200 °F (93 °C) and maximum operating pressure does not exceed 275 psig (1.9 MPa)). The comments' requested change would redefine the defense-in-depth concept. Rather than performing inspections to detect flaws before structural integrity is compromised, degradation would be managed in effect after leakage is discovered.

The NRC agrees, however, that it should be permissible under certain circumstances for licensees to evaluate local thinning using the acceptance criteria of the Code Case without NRC review and acceptance. Thus, a sixth condition is being proposed for Code Case N–597–2 in DG–1231/RG 1.147, Revision 17. The condition would propose that, on moderate-energy Class 2 and 3 piping, wall thinning acceptance criteria may be used on a temporary basis based on the provisions of Code Case N–513–2, and that Code Case N–597–2 cannot be used to evaluate through-wall leakage conditions.

Code Cases N–619

Type: Conditionally approved
Title: *Alternative Requirements for Nozzle Inner Radius Inspections for Class 1 Pressurizer and Steam Generator Nozzles Published*
Published: April 8, 2002

Code Case N–648–1

Type: Conditionally approved for the first time
Title: *Alternative Requirements for Inner Radius Examination of Class 1 Reactor Pressure Vessel Nozzles*
Published: September 18, 2001

A comment on the proposed rule published on June 2, 2009 (74 FR 26303), and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comment on draft RG 1.147, Revision 16, requested that the NRC reconsider the conditions placed on Code Case N–619, ''Alternative Requirements for Nozzle Inner Radius Inspections for Class 1 Pressurizer and Steam Generator Nozzles,'' and Code Case N–648–1, ''Alternative Requirements for Inner Radius Examination of Class 1 Reactor Pressure Vessel Nozzles.'' The comment states that the conditions on the two Code Cases requiring a wire standard to demonstrate the resolution capability of remote visual examination systems should be changed to the ASME 0.044-inch characters because those characters have been recognized to be a better resolution standard than the wire standard.

Because Code Case N–619 and Code Case N–648–1 were not part of the June 2, 2009, proposed rule and the changes reflected in N–619 and N–648–1 are significant; the NRC did not adopt the public comment to use characters rather than the wire standard.

The NRC is addressing the comment as part of this rulemaking. The NRC agrees with the 2009 comment that characters have been demonstrated to be a better resolution standard than the wire standard. However, the NRC believes that the shift to characters should be part of broader changes to the visual testing standards. Visual examinations are used in certain situations as alternatives to volumetric and/or surface examination tests where it is not possible to conduct volumetric examination (e.g., where there are limitations due to access or geometry) or to reduce occupational exposure in high radiation fields. Visual testing experts had believed that if the camera and lighting were sufficient to see a 12 μm (0.0005 in.) diameter wire, then the camera system had a resolution sufficiently high for the inspection. Subsequent investigation of the effectiveness and reliability of visual examinations has shown that the wire resolution standard is not sufficient to determine the visual acuity of a remote system (i.e., there are important differences between visually detecting a wire and a crack). Research conducted at the Pacific Northwest National Laboratory showed that other calibration standards should be adapted for visual testing such as reading charts and resolution targets. Results supporting this recommendation were published in NUREG/CR–6943, ''A Study of Remote Visual Methods to Detect Cracking in Reactor Components'' (ADAMS Accession No. ML073110060). As also discussed in the report, other parameters such as crack size, lighting conditions, camera resolution, and surface conditions were assessed. The NRC concluded from the investigation that a significant fraction of the cracks that have been reported in nuclear power plant components are at the lower end of the capabilities of the visual testing equipment currently being used. Code Case N–619 addresses the examination of the nozzle inner radius of Class 1 pressurizers and steam generators. Code Case N–648–1 provides an alternative for examining the inner radius of Class 1 reactor vessel nozzles. The NRC investigation of crack opening dimensions of service-induced cracks in nuclear components included thermal fatigue, mechanical fatigue, and stress corrosion cracks. The NRC concluded that current visual testing systems may not reliably detect a significant number of these cracks, and the research results showed that detection of these cracks under field conditions is strongly dependent on camera magnification, lighting, inspector training, and inspector vigilance. While this research supports the use of characters in lieu of a wire standard, the research also showed that other changes should be considered to visual testing as related to these two Code Cases. The NRC and the Electric Power Research Institute (EPRI) are currently conducting a collaborative research project investigating these parameters. The results of the collaborative research will be assessed by the NRC and the industry to determine what changes should be made to visual testing requirements in the future.

The comment also indicated that it is unclear how allowable flaw lengths would be determined from Table IWB–3512–1. The condition on the two Code Cases states that ''licensees may perform a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria of Table IWB–3512–1 with limiting assumptions on the flaw aspect ratio.'' Table IWB–3512–1 does not specifically provide allowable flaw length criteria. The commenter recommended that the acceptance criteria be modified as following: ''Crack-like surface flaws exceeding the acceptance criteria of Table IWB–3510–3 are unacceptable for continued service unless the vessel meets the requirements of IWB–2142.2, IWB–3142.3, or IWB–3142.4. The component thickness, t, to be applied in calculating the allowable surface flaw, I, in Table IWB–3510–3 shall be selected as specified in Table IWB–3512–2.''

The NRC does not agree with the suggestion. Table IWB–3512–1 was selected because it is the only table that considers the inside corner region. In determining an acceptable flaw size, the limiting aspect ratio is assumed, which is 0.5. The surface flaw allowable size divided by the limiting aspect ratio yields the limiting surface flaw size in terms of the l/t. In the case of wall thickness sizes provided in Table IWB–3512–1, the acceptance criteria are the same as those in Table IWB–3510–3. The NRC does not intend to make any changes to the table referred to for acceptance criteria, because Table IWB–3512–1 is the only table to refer to the inside corner region.

Finally, the commenter believes that the condition on Code Case N–648–1 describing the surfaces to be examined is unnecessary because the Code Case describes the same examination surfaces. The NRC agrees and proposes to eliminate this condition in Table 2 of DG–1231/RG 1.147, Revision 17.

Code Case N–702

Type: New

Title: *Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1*

Published: Supplement 12, 2001 Edition

Two comments on the proposed rule published on June 2, 2009 (74 FR 26303), and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comment on draft RG 1.147, Revision 16, requested that Code Case N–702, ''Alternative Requirements for Boiling Water Reactor (BWR) Nozzle Inner Radius and Nozzle-to-Shell Welds, Section XI, Division 1,'' be conditionally approved in the final guide. Code Case N–702 had been listed in draft RG 1.193, Revision 3, ''ASME Code Cases Not Approved for Use,'' because at the time that draft Revision 16 to RG 1.147 was published (October 2007), the NRC staff was considering the industry response to the NRC staff's request for additional information relative to the acceptability of ''BWRVIP–108: BWR Vessel and Internals Project (VIP), Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii,'' EPRI Technical Report 1003557, October 2002 (ADAMS Accession No. ML023330203). BWRVIP–108 provides the technical basis supporting Code Case N–702. Subsequently, the NRC conditionally approved a licensee's request to use the Code Case on the basis of the NRC's Safety Evaluation (ADAMS Accession No. ML073600374; December 18, 2007).

The Safety Evaluation discussed the NRC's review of BWRVIP–108 and the conditions under which it could be used. The commenters believed that the conditions in the Safety Evaluation provided a basis for the NRC to conditionally approve Code Case N–702 in final RG 1.147, Revision 16. The NRC did not adopt the public comment to approve the Code Case in final Revision 16 to RG 1.147. Code Case N–702 is an alternative to provisions in the ASME Code to reduce the inspection requirements of BWR reactor vessel nozzle-to-shell welds and nozzle blend radii. BWRVIP–108 discusses the probabilistic fracture mechanics evaluation that was performed to demonstrate that the probability of failure considering these inspection changes meets NRC requirements. While the NRC believes that the Safety Evaluation and BWRVIP report provide a basis for conditionally approving the Code Case on a generic basis, the NRC did not believe that it would have been appropriate to move the Code Case from RG 1.193 to RG 1.147 without first having sought public comment. Thus, the NRC is proposing to conditionally approve Code Case N–702 in DG–1231/RG 1.147, Revision 17, based on the conditions that were discussed in the Safety Evaluation. The applicability of Code Case N–702 must be shown by demonstrating that the criteria in Section 5.0 of the NRC Safety Evaluation regarding BWRVIP–108 dated December 18, 2007, are met. The evaluation demonstrating the applicability of the Code Case must be reviewed and approved by the NRC prior to the application of the Code Case.

Code Case N–747

Type: New

Title: *Reactor Vessel Head-to Flange Weld Examinations, Section XI, Division 1*

Published: Supplement 9, 2004 Edition

A comment on the proposed rule published on June 2, 2009 (74 FR 26303), and a parallel notice of availability of draft RGs (74 FR 26440) seeking public comment on draft RG 1.147, Revision 16, suggested that the basis for listing Code Case N–747, ''Reactor Vessel Head-to Flange Weld Examinations, Section XI, Division 1,'' in draft RG 1.193 was flawed, and that the Code Case should be unconditionally accepted in final Revision 16. Additional technical information to support approval of the Code Case was provided in the comment letter (ADAMS Accession No. ML092190138). The NRC did not adopt the public comment to list Code Case N–747 in final Revision 16 to RG 1.147 (incorporated by reference in the final rule published on October 5, 2010, (75 FR 61321)), because the NRC determined that the public should have an opportunity to comment on the additional information that was submitted by the commenter.

The NRC has reviewed the information provided in the comment, which deals with the expected fluence levels of reactor vessel head-to-flange welds. Based on this information, the NRC believes that an adequate technical basis has been provided to support a conclusion that the fracture toughness will remain high. The key points discussed in the additional information are that calculations show that the fluence in the upper head region will be low, even after 60 years of service. Therefore, there will be no irradiation induced change in RT_{NDT}. In addition, the industry has calculated RT_{NDT} for the upper head region for early Westinghouse plant designs using the Standard Review Plan (NUREG–0800) and determined that the fracture toughness is high. Therefore, the NRC proposes to unconditionally approve

Code Case N–747 in Table 1 of DG–1231/RG 1.147, Revision 17.

D. ASME Code Cases Not Approved for Use

The ASME Code Cases that are currently issued by the ASME but not approved for generic use by the NRC are listed in RG 1.193, ''ASME Code Cases Not Approved for Use.'' In addition to ASME Code Cases that the NRC has found to be technically or programmatically unacceptable, RG 1.193 includes Code Cases on reactor designs for high-temperature gas-cooled reactors and liquid metal reactors, reactor designs not currently licensed by the NRC, and certain requirements in Section III, Division 2, for submerged spent fuel waste casks, that are not endorsed by the NRC. Regulatory Guide 1.193 complements RGs 1.84, 1.147, and 1.192. It should be noted that RG 1.193 is not part of this rulemaking because the NRC is not proposing to adopt any of the Code Cases listed in that RG. Comments have been submitted in the past, however, on certain Code Cases listed in RG 1.193 where the commenter believed that additional technical information was available that might not have been considered by the NRC in its determination not to approve the use of these Code Cases. While the NRC will consider those comments, any changes in the NRC's non-approval of such Code Cases will be the subject of an additional opportunity for public comment.

IV. Petition for Rulemaking (PRM–50–89)

On December 14, 2007, Mr. Raymond West (the petitioner) submitted a PRM requesting the NRC to amend 10 CFR 50.55a to allow consideration of alternatives to the ASME BPV and OM Code Cases. The petitioner submitted an amended petition on December 19, 2007 (ADAMS Accession No. ML073600974). The petition was docketed by the NRC as PRM–50–89. The petitioner requested that the regulations be amended to provide applicants and licensees a process for requesting NRC approval of changes or modifications to ASME Code Cases that are listed in the relevant NRC-approved RGs cited in the current regulations. The petitioner stated that the current requirements do not allow changes or modifications to be proposed as alternatives to NRC-approved ASME Code Cases, and asserted that such changes or modifications should be allowed as alternatives to NRC Code Cases. Overall, the petitioner requested that the regulations be amended to allow applicants and licensees to request authorization of NRC-approved Code Cases with proposed modifications directly through § 50.55a(a)(3).

The NRC believes that Code Cases often provide alternatives that have technical merit and, in many instances, are incorporated into future ASME Code editions. The ASME Code Case process itself constitutes a method of how an applicant or licensee can seek to obtain ASME approval for a variation of a previously-approved Code provision. Section 50.55a(a)(3) currently provides specific approaches for obtaining NRC authorization of alternatives to ASME Code provisions. Inasmuch as ASME Code Cases are analogous to ASME Code provisions, it is not unreasonable to provide an analogous regulatory approach for obtaining NRC authorization of alternatives to ASME Code Cases. For these reasons, the NRC determined that the issues raised in this PRM should be considered in the NRC's rulemaking process, and the NRC published a FRN with this determination on April 22, 2009 (74 FR 18303). Accordingly, the NRC is addressing PRM–50–89 in this proposed rule.

On the basis of the previous discussion, the NRC is proposing to include language in proposed 10 CFR 50.55a(z) (existing 50.55a(a)(3)) that would allow applicants and licensees to request authorization of alternatives for changes to conditions on NRC-approved ASME Code Cases in current paragraphs (b)(4), (b)(5), and (b)(6) of § 50.55a. In addition, the NRC proposes extending the scope of the petitioner's request for allowing alternatives to NRC-approved Code Case conditions to allow applicants and licensees to request authorization of alternatives for changes to conditions on Section III and XI of the ASME BPV Code and OM Code in current paragraphs (b)(1), (b)(2), and (b)(3).

V. Changes Addressing Office of the Federal Register Guidelines on Incorporation by Reference

This proposed rule includes changes to 10 CFR 50.54, 50.55, and 50.55a. These changes were made in accordance with the guidance for incorporation by reference of multiple standards that is included in Chapter 6 of the OFR's ''Federal Register Document Drafting Handbook,'' January 2011 Revision. This latest revision of the OFR's guidance provides several options for incorporating by reference multiple standards into regulations.

The NRC proposes to incorporate by reference, in a single paragraph, the multiple standards mentioned in 10 CFR 50.55a. For the least disruption to the existing structure of the section, the NRC proposes to incorporate by reference the multiple standards into 10 CFR 50.55a(a), the first paragraph of the section. Each national consensus standard that is being incorporated by reference in 10 CFR 50.55a has been listed separately. Accordingly, the regulatory language of 10 CFR 50.54, 50.55, and 50.55a has been reorganized by moving existing paragraphs, creating new paragraphs, and revising introductory and regulatory texts.

The NRC has made conforming changes to references throughout 10 CFR 50.55a to reflect this reorganization. A detailed discussion of the affected paragraphs, other than the aforementioned reference changes, is provided in Section VII, ''Paragraph-by-Paragraph Discussion,'' of this document. The regulatory text of 10 CFR 50.55a has been set out in its entirety for the convenience of the reader. The staff has also developed reader aids to help users understand these changes (see Section VI of this document.)

VI. Addition of Headings to Paragraph

The NRC is proposing to add headings (explanatory titles) to paragraphs and all lower-level subparagraphs of 10 CFR 50.55a. These headings are intended to enhance the readers' ability to identify the paragraphs (e.g., paragraphs (a), (b), (c)) and subparagraphs with the same subject matter. The NRC's proposal addresses longstanding complaints by external and internal stakeholders on the readability and complex structure of the requirements in 10 CFR 50.55a. To address this concern, the NRC evaluated a range of solutions, including the creation of new regulations and relocation of existing requirements from 10 CFR 50.55a to the new regulations.

Some alternatives the NRC considered were a new regulation adjacent to 10 CFR 50.55a (e.g., §§ 50.55b, 50.55c, 50.55d), a new subpart containing a new series of regulations at the end of 10 CFR part 50 (e.g., subpart B beginning at § 50.200, and continuing with §§ 50.201, 50.202, 50.203), or a new part (designated for codes and standards) containing a new series of regulations addressing codes and standards approved for incorporation by reference by the OFR. The relocation of each existing requirement to a new regulation (or set of regulations) would follow a set of organizing principles established by the NRC after consideration of stakeholder's views.

Upon consideration of these alternatives, the NRC decided that these alternatives should not be adopted—at least not at this time without further stakeholder input—and instead that the

NRC should develop and adopt headings for paragraphs and subparagraphs. The primary reason for the NRC's decision is external stakeholders' objections to a previous attempt by the NRC to re-designate paragraphs in § 50.55a (75 FR 24324; May 4, 2010). As the NRC understands it, many nuclear power plant licensees' procedures reference specific paragraphs and subparagraphs of § 50.55a. It would require substantial rewriting of these procedures and documents to correct the references to the old (superseded) section, paragraphs and subparagraphs. In addition, currently-approved design certification rules may require conforming amendments to be made to correct references to ASME Code provisions on design (and possibly ISI and IST).

The NRC requests public comment on whether the NRC should adopt one of these approaches, either as a follow-on activity to the addition of headings, or as a substitute for the addition of headings. The most helpful comments would identify a specific approach, and set forth the reasons why the proposed approach should be adopted, taking into account the factors considered by the NRC in selecting the headings approach.

NRC's Proposal: Convention for Headings and Subheadings

The NRC is proposing to add headings to all first, second, third, fourth, and some fifth-level paragraphs for certain sections of 10 CFR 50.55a to add clarity and a user-friendly method for following sublevel contents within a regulation. The proposed heading for a fourth-level would follow the same convention, but may designate the provision number only. Fifth-level paragraphs are proposed for only newly incorporated Code Cases. Each first-level paragraph (designated using letters, (e.g., (a), (b), (c))) would have a heading that concisely describes the general subject matter addressed in that paragraph. Each second-level paragraph (designated using numbers, e.g., (1), (2), (3)) would have a heading comprised of a summary of the first-level paragraph's heading and a semicolon ('';''), followed by a concise description of the subject matter addressed in the second paragraph. The proposed heading for a third-level paragraph would follow the same convention (i.e., a heading comprised of a summary level of the higher-level paragraph's title and a semicolon, followed by a concise description of the subject matter addressed in that subparagraph). The proposed heading for a fourth-level paragraph would follow the same convention, but may designate the provision number only. The proposed fifth-level paragraph is applied to only paragraph (a) for incorporation by reference of approved editions and addenda to the ASME BPV and OM Codes.

Reader Aids

The staff has developed a table showing the proposed structure of 10 CFR 50.55a. This table, ''Proposed Reorganization of Paragraphs and Subparagraphs in 10 CFR 50.55a, Codes and standards'' (ADAMS Accession No. ML12289A121) is available in a separate document and outlines the section showing all paragraph designations, including the new paragraph headings. The staff has also developed Cross-Reference tables showing the current designations for 10 CFR 50.54, 50.55, and 50.55a regulations and the proposed designations for these sections. These tables contain the new headings and a description of each change and are available in a separate document (ADAMS Accession No. ML12289A114).

VII. Paragraph-by-Paragraph Discussion

Section 50.54

In § 50.54, the introductory statement would be revised to include a reference to § 50.55a. This revision would clarify that nuclear power plant licensees, as described in the introductory paragraph of § 50.54, also are subject to the applicable requirements delineated in § 50.55a. In addition, the NRC proposes to revise the introductory text of this section, add and reserve paragraph (ii), and add paragraph (jj) to include a condition of every license. This requirement is currently contained in § 50.55a(a)(1), and no change to the requirement is intended by the transfer of this requirement from § 50.55a(a)(1) to § 50.54(jj).

Section 50.55

In § 50.55, the introductory text would be revised to include references to existing § 50.55a, and paragraphs (g) and (h) would be added and reserved for future use. Further, existing § 50.55a(a)(1) would be moved to a newly created § 50.55(i).

Section 50.55a

In § 50.55a, the current introductory statement would be relocated to § 50.54(jj), 50.55(i), and 50.55a.

Paragraph (a)

A new paragraph (a) would be created in § 50.55a to incorporate by reference the multiple standards currently identified in existing § 50.55a. The heading would be revised to read ''Documents approved for incorporation by reference.''

Paragraph (a)(1): This paragraph ''American Society of Mechanical Engineers (ASME)'' would be added to group all ASME Sections.

Paragraph (a)(1)(i): This paragraph, *''ASME Boiler and Pressure Vessel Code, Section III,''* would be added to discuss the availability of standards referenced in current paragraph (b)(1). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(i)(A): This paragraph, *''Rules for Construction of Nuclear Vessels,''* would be added to group all the individual standards referenced regarding the subject matter included in current paragraph (b)(1). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(i)(B): This paragraph, *''Rules for Construction of Nuclear Power Plant Components,''* would be added to group all the individual standards referenced regarding the subject matter included in current paragraph (b)(1). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(i)(C): This paragraph, ''*Division I Rules for Construction of Nuclear Power Plant Components,''* would be added to group all the individual standards referenced regarding the subject matter included in current paragraph (b)(1). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(i)(D): This paragraph, *''Rules for Construction of Nuclear Power Plant Components—Division 1,''* would be added to group all the individual standards referenced regarding the subject matter included in current paragraph (b)(1). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(i)(E): This paragraph, *''Rules for Construction of Nuclear Facility Components—Division 1,''* would be added to group all the individual standards referenced regarding the subject matter included in current paragraph (b)(1). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(ii)(A): This paragraph, *''Rules for Inservice Inspection of Nuclear Reactor Coolant Systems,''* would be added to discuss the availability of individual standards referenced regarding the subject matter included in current paragraph (b)(2). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(ii)(B): This paragraph, *''Rules for Inservice Inspection of Nuclear Power Plant Components,''* would be added to discuss the availability of individual standards referenced regarding the subject matter included in current paragraph (b)(2). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(ii)(C): This paragraph, *''Rules for Inservice Inspection of Nuclear Power Plant Components—Division 1,''* would be added to discuss the availability of individual standards referenced regarding the subject matter included in current paragraph (b)(2). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(iii): This paragraph, *''ASME Code Cases: Nuclear Components,''* would be added to discuss the newly approved Code Cases referenced regarding the subject matter in current paragraph (b). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(iii)(A): This paragraph, *''ASME Code Case N–722–1,''* would be added to discuss the newly approved Code Case referenced regarding the subject matter in current paragraph (b). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(iii)(B): This paragraph, ''*ASME Code Case N–729–1,''* would be added to discuss the newly approved Code Case referenced regarding the subject matter in current paragraph (b). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(iii)(C): This paragraph, ''*ASME Code Case N–770–1,''* would be added to discuss the newly approved Code Case referenced regarding the subject matter in current paragraph (b). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(iv): This paragraph, ''*ASME Operation and Maintenance Code,''* would be added to group all the individual standards referenced in current paragraph (b). This change would bring the NRC's requirements into compliance with the OFR's revised guidelines for incorporating by reference consensus standards in regulations.

Paragraph (a)(1)(iv)(A): This paragraph, ''*Code for Operation and Maintenance of Nuclear Power Plants,''* would be added to group all the individual standards referenced in current paragraph (b).

Paragraph (a)(1)(iv)(B): This paragraph would be added and reserved for future use.

Paragraph (a)(2): This paragraph, ''*Institute of Electrical and Electronics Engineers (IEEE) Service Center,''* would be added to list all IEEE sections.

Paragraph (a)(2)(i): This paragraph, ''*IEEE Standard 279–1971,''* would be added to discuss the availability of standards referenced in current paragraph (h)(2). This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (a)(2)(ii): This paragraph, ''*IEEE Standard 603–1991,''* would be added to discuss the availability of the standard referenced in current paragraph (h)(2) and (h)(3). This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (a)(2)(iii): This paragraph, ''*IEEE Standard 603–1991 correction sheet,''* would be added to discuss the availability of the standard referenced in current paragraphs (h)(2) and (h)(3). This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (a)(3): This paragraph, *''U.S. Nuclear Regulatory Commission (NRC) Reproduction and Distribution Services Section,''* lists all regulatory guides being incorporated by reference. This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (a)(3)(i): This paragraph, ''*NRC Regulatory Guide 1.84, Revision 36,''* would be added to discuss the availability of the standard. This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (a)(3)(ii): This paragraph, ''*NRC Regulatory Guide 1.147, Revision 17,''* would be added to discuss the availability of the standard. This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (a)(3)(iii): This paragraph, ''*NRC Regulatory Guide 1.192, Revision 1,''* would be added to discuss the availability of the standard. This would be done in compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (b): The paragraph heading would be revised to ''*Use and conditions on the use of standards.''* The contents would be moved, in part, to 50.55a(a) for compliance with OFR revised guidelines for incorporation by reference standards in regulations.

Paragraph (c): Introductory text would be added to the existing paragraph (c). Explanatory headings would be added for subparagraphs.

Paragraph (d): The new paragraph would add introductory text to ''*Quality Group B components,''* as part of the NRC initiative of adding headings and providing clarity. Explanatory headings would be added for subparagraphs.

Paragraph (e): The new paragraph would add introductory text to ''*Quality Group C components,''* as part of the NRC initiative of adding headings and providing clarity. Explanatory headings would be added for subparagraphs.

Paragraph (f): Introductory text would be revised and expanded in ''*Inservice testing requirements,''* as part of the NRC initiative of adding headings and providing clarity. Explanatory headings would be added for subparagraphs.

Paragraph (g): Introductory text would be revised and expanded in ''*Inservice inspection requirements,''* as part of the NRC initiative of adding headings and providing clarity. Explanatory headings would be added for subparagraphs.

Paragraphs (b)(5), (f)(2), (f)(3)(iii)(A), (f)(3)(iv)(A), (f)(4)(ii), (g)(2), (g)(3)(i), (g)(3)(ii), (g)(4)(i), and (g)(4)(ii): References to the revision number for RG 1.147 would be changed from ''Revision 16'' to ''Revision 17.''

Paragraph (h)(1): This paragraph would be designated as reserved because the informational content from current (h)(1) would be moved to proposed paragraph (a)(2).

Paragraphs (i)–(y): The paragraphs would be added and reserved for future use.

Paragraph (z): This paragraph would be added to contain information that would be relocated from the introductory text of current paragraph (a)(3) and current subparagraphs (a)(3)(i)–(ii) as a result of the NRC's compliance with the OFR's revised guidelines for incorporating by reference. Paragraph (z) would also be revised to allow applicants and licensees to request alternatives to the requirements in paragraph (b) of this section.

Overall Considerations on the Use of ASME Code Cases

This rulemaking would amend 10 CFR 50.55a to incorporate by reference RG 1.84, Revision 36, which would supersede Revision 35; RG 1.147, Revision 17, which would supersede Revision 16; and RG 1.192, Revision 1, which would supersede Revision 0. The following general guidance applies to the use of the ASME Code Cases approved in the latest versions of the RGs that are incorporated by reference into 10 CFR 50.55a as part of this rulemaking.

The approval of a Code Case in the NRC RGs constitutes acceptance of its technical position for applications that are not precluded by regulatory or other requirements or by the recommendations in these or other RGs. The applicant and/or licensee are responsible for ensuring that use of the Code Case does not conflict with regulatory requirements or licensee commitments. The Code Cases listed in the RGs are acceptable for use within the limits specified in the Code Cases. If the RG states an NRC condition on the use of a Code Case, then the NRC condition supplements and does not supersede any condition(s) specified in the Code Case, unless otherwise stated in the NRC condition.

The ASME Code Cases may be revised for many reasons, (e.g., to incorporate operational examination and testing experience and to update material requirements based on research results). On occasion, an inaccuracy in an equation is discovered or an examination, as practiced, is found not to be adequate to detect a newly discovered degradation mechanism. Hence, when an applicant or a licensee initially implements a Code Case, 10 CFR 50.55a requires that the applicant or the licensee implement the most recent version of that Code Case as listed in the RGs incorporated by reference. Code Cases superseded by revision are no longer acceptable for new applications unless otherwise indicated.

Section III of the ASME BPV Code applies only to new construction (i.e., the edition and addenda to be used in the construction of a plant are selected based on the date of the construction permit and are not changed thereafter, except voluntarily by the applicant or the licensee). Hence, if a Section III Code Case is implemented by an applicant or a licensee and a later version of the Code Case is incorporated by reference into 10 CFR 50.55a and listed in the RGs, the applicant or the licensee may use either version of the Code Case (subject, however, to whatever change requirements apply to its licensing basis, (e.g., 10 CFR 50.59)).

A licensee's ISI and IST programs must be updated every 10 years to the latest edition and addenda of Section XI and the OM Code, respectively, that were incorporated by reference into 10 CFR 50.55a and in effect 12 months prior to the start of the next inspection and testing interval. Licensees who were using a Code Case prior to the effective date of its revision may continue to use the previous version for the remainder of the 120-month ISI or IST interval. This relieves licensees of the burden of having to update their ISI or IST program each time a Code Case is revised by the ASME and approved for use by the NRC. Code Cases apply to specific editions and addenda, and Code Cases may be revised if they are no longer accurate or adequate, so licensees choosing to continue using a Code Case during the subsequent ISI or IST interval must implement the latest version incorporated by reference into 10 CFR 50.55a and listed in the RGs.

The ASME may annul Code Cases that are no longer required, are determined to be inaccurate or inadequate, or have been incorporated into the BPV or OM Codes. If an applicant or a licensee applied a Code Case before it was listed as annulled, the applicant or the licensee may continue to use the Code Case until the applicant or the licensee updates its construction Code of Record (in the case of an applicant, updates its application) or until the licensee's 120-month ISI or IST update interval expires, after which the continued use of the Code Case is prohibited unless NRC authorization is given under the current 10 CFR 50.55a(a)(3). If a Code Case is incorporated by reference into 10 CFR 50.55a and later annulled by the ASME because experience has shown that the design analysis, construction method, examination method, or testing method is inadequate; the NRC will amend 10 CFR 50.55a and the relevant RG to remove the approval of the annulled Code Case. Applicants and licensees should not begin to implement such annulled Code Cases in advance of the rulemaking.

A Code Case may be revised, for example, to incorporate user experience. The older or superseded version of the Code Case cannot be applied by the licensee or applicant for the first time.

If an applicant or a licensee applied a Code Case before it was listed as superseded, the applicant or the licensee may continue to use the Code Case until the applicant or the licensee updates its construction Code of Record (in the case of an applicant, updates its application) or until the licensee's 120-month ISI or IST update interval expires, after which the continued use of the Code Case is prohibited unless NRC authorization is given under proposed 10 CFR 50.55a(z). If a Code Case is incorporated by reference into 10 CFR 50.55a and later a revised version is issued by the ASME because experience has shown that the design analysis, construction method, examination method, or testing method is inadequate; the NRC will amend 10 CFR 50.55a and the relevant RG to remove the approval of the superseded Code Case. Applicants and licensees should not begin to implement such superseded Code Cases in advance of the rulemaking.

VIII. Plain Writing

The NRC has written this document to be consistent with the Plain Writing Act as well as the Presidential Memorandum, "Plain Language in Government Writing," published June 10, 1998 (63 FR 31883). The NRC requests comment on the proposed rule with respect to the clarity and effectiveness of the language used.

IX. Availability of Documents

The NRC is making the documents identified in Table II available to interested persons through one or more of the following methods, as indicated. To access documents related to this action, see the **ADDRESSES** section of this document.

TABLE II

Document	PDR	WEB	NRC Library.
Proposed Rule—Regulatory Analysis	X	X	ML103060189.
Proposed Rule **Federal Register** Notice	X	X	ML103060003.
Proposed Reorganization of Paragraphs and Subparagraphs	X	X	ML12289A121.
Cross-Reference Tables	X	X	ML12289A114.
RG 1.84, Revision 36 (DG–1230)	X	X	ML102590003.
RG 1.147, Revision 17 (DG–1231)	X	X	ML102590004.
RG 1.192, Revision 1 (DG–1232)	X	X	ML102600001.
RG 1.200, Revision 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-informed Activities.	X	X	ML090410014.
RG 1.201, Revision 1, Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance.	X	X	ML061090627.
2007/12/19—Petition for Rulemaking PRM–50–89 submitted by Ray West regarding, ''To Amend CFR 5–.55a—Codes and Standards—Revision 1''.	X	X	ML073600974.
Hatch Plant Report—Hatch, Units 1 & 2, Farley, Units 1 & 2, Vogtle, Units 1 & 2, Safety Evaluation Re. Request to Use ASME Code Case N–661.	X	X	ML033280037.
EPRI Technical Report—Project No. 704—BWRVIP–108: BWR Vessel & Internals Project, Technical Basis for Reduction of Inspection Requirements for Boiling Water Reactor Nozzle-to-Vessel Shell Welds & Nozzle Blend Radii.	X	X	ML023330203.
Safety Evaluation of Proprietary EPRI Report—BWR Vessel and Internals Project, Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Inner Radius (BWRVIP–108).	X	X	ML073600374.
Comment Letter—Comment (4) of Bryan A. Erler on Behalf of ASME Supporting Draft Regulatory Guides DG–1191, DG–1192, DG–1193, and the Proposed Rule Incorporating the Final Revisions of these Regulatory Guides into 10 CFR 50.55a.	X	X	ML092190138.
SRM–COMNJD–03–0002—Stabilizing the PRA Quality Expectations and Requirements	X	X	ML033520457.
SECY–04–0118—Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality.	X	X	ML041470505.
SRM–SECY–04–0118—Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality.	X	X	ML042800369.
NUREG–0800—Chapter 4, Section 4.5.1, Revision 3, Control Rod Drive Structural Materials, dated March 2007.	X	X	ML070230007.
NUREG–0800—Chapter 5, Section 5.2.3, Revision 3, Reactor Coolant Pressure Boundary Materials, dated March 2007.	X	X	ML063190006.
NUREG/CR–6943—A Study of Remote Visual Methods to Detect Cracking in Reactor Components	X	X	ML073110060.

X. Voluntary Consensus Standards

Section 12(d)(3) of the National Technology Transfer and Advancement Act (NTTAA) of 1995, Public Law 104–113, and implementing guidance in U.S. Office of Management and Budget (OMB) Circular A–119 (February 10, 1998), require each Federal government agency (should it decide that regulation is necessary) to use a voluntary consensus standard instead of developing a government-unique standard. An exception to using a voluntary consensus standard is allowed where the use of such a standard is inconsistent with applicable law or is otherwise impractical. The NTTAA requires Federal agencies to use industry consensus standards to the extent practical; it does not require Federal agencies to endorse a standard in its entirety. Neither the NTTAA nor OMB Circular A–119 prohibit an agency from adopting a voluntary consensus standard while taking exception to specific portions of the standard, if those provisions are deemed to be ''inconsistent with applicable law or otherwise impractical.'' Furthermore, taking specific exceptions furthers the Congressional intent of Federal reliance on voluntary consensus standards because it allows the adoption of substantial portions of consensus standards without the need to reject the standards in their entirety because of limited provisions that are not acceptable to the agency.

In this rulemaking, the NRC is continuing its existing practice of approving the use of ASME BPV and OM Code Cases, which are ASME-approved alternatives to compliance with various provisions of the ASME BPV and OM Code. The NRC's approval of the ASME Code Cases is accomplished by amending the NRC's regulations to incorporate by reference the latest revisions of the following, which are the subject of this rulemaking, into 10 CFR 50.55a: RG 1.84, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,'' Revision 36; RG 1.147, ''Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,'' Revision 17; and RG 1.192, ''Operation and Maintenance Code Case Acceptability, ASME Code,'' Revision 1. These RGs list the ASME Code Cases that the NRC has approved for use. The ASME Code Cases are national consensus standards as defined in the NTTAA and OMB Circular A–119. The ASME Code Cases constitute voluntary consensus standards, in which all interested parties (including the NRC and licensees of nuclear power plants) participate. Therefore, the NRC's approval of the use of the ASME Code Cases identified in RGs 1.84, Revision 36; RG 1.147, Revision 17; and RG 1.192, Revision 1, which are the subject of this rulemaking, is consistent with the overall objectives of the NTTAA and OMB Circular A–119.

The NRC reviews each Section III, Section XI, and OM Code Case published by the ASME to ascertain whether it is consistent with the safe operation of nuclear power plants. The Code Cases found to be generically acceptable are listed in the RGs that are incorporated by reference in 10 CFR 50.55a. The Code Cases found to be unacceptable are listed in RG 1.193, but licensees may still seek the NRC's approval to apply these Code Cases through the processes in 10 CFR 50.55a for requesting the approval of alternatives or for relief. Code Cases that the NRC finds to be conditionally acceptable are also listed in RGs 1.84, 1.147, and 1.192, which are the subject of this rulemaking, together with the conditions that must be used if the Code Case is applied. The NRC believes that this rule complies with the NTTAA and OMB Circular A–119 despite these conditions. If the NRC did not conditionally accept ASME Code Cases, it would disapprove these Code Cases entirely. The effect would be that licensees and applicants would submit a larger number of requests for use of alternatives under the current 10 CFR 50.55a(a)(3), requests for relief under 10 CFR 50.55a(f) and (g), or requests for exemptions under 10 CFR 50.12 and/or 10 CFR 52.7. For these reasons, the treatment of ASME Code Cases and any conditions proposed to be placed on them in this proposed rule do not conflict with any policy on agency use of consensus standards specified in OMB Circular A–119.

The NRC did not identify any other voluntary consensus standards developed by the United States voluntary consensus standards bodies for use within the United States that the NRC could approve instead of the ASME Code Cases.

The NRC also did not identify any voluntary consensus standards developed by multinational voluntary consensus standards bodies for use on a multinational basis that the NRC could incorporate by reference instead of the ASME Code Cases. This is because no other multinational voluntary consensus body would develop alternatives to a voluntary consensus standard (i.e., either the ASME BPV Code or the ASME OM Code) for which they did not develop and do not maintain.

In summary, this proposed rule satisfies the requirements of Section 12(d)(3) of the NTTAA and OMB Circular A–119.

XI. Finding of No Significant Environmental Impact: Environmental Assessment

This proposed action stems from the Commission's practice of incorporating by reference the RGs listing the most recent set of NRC-approved ASME Code Cases. The purpose of this proposed action is to allow licensees to use the Code Cases listed in the RGs as alternatives to requirements in the ASME BPV and OM Codes for the construction, ISI, and IST of nuclear power plant components. This proposed action is intended to advance the NRC's strategic goal of ensuring adequate protection of public health and safety and the environment. It also demonstrates the agency's commitment to participate in the national consensus standards process under the National Technology Transfer and Advancement Act of 1995, Pub. L.104–113.

The National Environmental Policy Act (NEPA), as amended, requires Federal government agencies to study the impacts of their ''major Federal actions significantly affecting the quality of the human environment'' and prepare detailed statements on the environmental impacts of the action and alternatives to the action (United States Code (U.S.C), Volume 42, Section 4332(C) [42 U.S.C. Sec. 4332(C)]; NEPA Sec. 102(C)).

The Commission has determined under NEPA, as amended, and the Commission's regulations in subpart A of 10 CFR part 51, that this proposed rule would not be a major Federal action significantly affecting the quality of the human environment. Therefore, an environmental impact statement is not required.

As alternatives to the ASME Code, NRC-approved Code Cases provide an equivalent level of safety. Therefore, the probability or consequences of accidents is not changed. There are also no significant, non-radiological impacts associated with this action because no changes would be made affecting non-radiological plant effluents and because no changes would be made in activities that would adversely affect the environment. The determination of this environmental assessment is that there will be no significant offsite impact to the public from this proposed action.

XII. Paperwork Reduction Act Statement

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

Type of submission, new or revision: Revision.

The title of the information collection: Domestic Licensing of Production and Utilization Facilities: Updates to Incorporation by Reference and Regulatory Guides.

The form number if applicable: Not applicable.

How often the collection is required: On occasion.

Who will be required or asked to report: Power reactor licensees and applicants for power reactors under construction.

An estimate of the number of annual responses: -185.

The estimated number of annual respondents: 109.

An estimate of the total number of hours needed annually to complete the requirement or request: A reduction of 14,800 reporting hours.

Abstract: This proposed rule is the latest in a series of rulemakings that incorporate by reference the latest versions of several Regulatory Guides identifying new and revised unconditionally or conditionally acceptable ASME Code Cases that are approved for use. The incorporation by reference of these Code Cases will reduce the number of alternative requests submitted by licensees under proposed 10 CFR 50.55a(z) by an estimated 185 requests annually.

The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the information collections contained in this proposed rule (or proposed policy statement) and on the following issues:

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

The public may examine and have copied for a fee publicly available documents, including the draft supporting statement at the NRC's PDR, One White Flint North, 11555 Rockville Pike, Room O–1 F21, Rockville, Maryland 20852. The OMB clearance requests are available at the NRC's Web site: *http://www.nrc.gov/public-involve/doc-comment/omb/*. The document will be available on the NRC home page site for 60 days after the signature date of this notice.

Send comments on any aspect of these proposed information collections, including suggestions for reducing the burden and on the above issues, by July 24, 2013 to the Information Services Branch (T–5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, or by email to *INFOCOLLECTS.RESOURCE@NRC.GOV* and to the Desk Officer, Chad Whiteman, Office of Information and Regulatory Affairs, NEOB–10202, (3150–0011), Office of Management and Budget, Washington, DC 20503. Comments on the proposed information collections may also be submitted via the Federal eRulemaking Portal *http://www.regulations.gov,* docket # NRC–2009–0359. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given to comments received after this date. Comments can also be emailed to *Chad_S_Whiteman@omb.eop.gov* or submitted by telephone at 202–395–4718.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection unless the requesting document displays a currently valid OMB control number.

XIII. Regulatory Analysis

The ASME Code Cases listed in the RGs to be incorporated by reference provide voluntary alternatives to the provisions in the ASME BPV and OM Codes for design, construction, ISI, and IST of specific structures, systems, and components used in nuclear power plants. Implementation of these Code Cases is not required. Licensees and applicants use NRC-approved ASME Code Cases to reduce unnecessary regulatory burden or gain additional operational flexibility. It would be difficult for the NRC to provide these advantages independently of the ASME Code Case publication process without expending considerable additional resources. The NRC has prepared a regulatory analysis addressing the qualitative benefits of the alternatives considered in this proposed rulemaking and comparing the costs associated with each alternative (ADAMS Accession No. ML103060189). The NRC invites public comment on this draft regulatory analysis. Copies of the regulatory analysis are available to the public as indicated in Section IX, ''Availability of Documents,'' of this document.

In addition to the general opportunity to submit comments on the proposed rule, the NRC also requests comments on the NRC's cost and benefit estimates as shown in the proposed rule regulatory analysis.

XIV. Regulatory Flexibility Certification

Under the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this proposed rule would not impose a significant economical impact on a substantial number of small entities. This proposed rule would affect only the licensing and operation of nuclear power plants. The companies that own these plants are not ''small entities'' as defined in the Regulatory Flexibility Act or the size standards established by the NRC (10 CFR 2.810).

XV. Backfitting and Issue Finality

The provisions in this proposed rulemaking would allow licensees and applicants to voluntarily apply NRC-approved Code Cases, sometimes with NRC-specified conditions. The approved Code Cases are listed in three regulatory guides that are incorporated by references into 10 CFR 50.55a.

An applicant's and/or licensees voluntary application of an approved Code Cases does not constitute backfitting, inasmuch as there is no imposition of a new requirement or new position. Similarly, voluntary application of an approved Code Case by a part 52 applicant or licensee does not represent NRC imposition of a requirement or action which is inconsistent with any issue finality provision in part 52. For these reasons the NRC finds that this proposed rule does not involve any provisions requiring the preparation of a backfit analysis or documentation demonstrating that one or more of the issue finality criteria in part 52 are met.

List of Subjects in 10 CFR Part 50

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

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

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

Authority: Atomic Energy Act secs. 102, 103, 104, 105, 147, 149, 161, 181, 182, 183, 186, 189, 223, 234 (42 U.S.C. 2132, 2133, 2134, 2135, 2167, 2169, 2201, 2231, 2232, 2233, 2236, 2239, 2273, 2282); Energy Reorganization Act secs. 201, 202, 206 (42 U.S.C. 5841, 5842, 5846); Nuclear Waste Policy Act sec. 306 (42 U.S.C. 10226); Government Paperwork Elimination Act sec. 1704 (44 U.S.C. 3504 note); Energy Policy Act of 2005, Pub. L. No. 109–58, 119 Stat. 194 (2005). Section 50.7 also issued under Pub. L. 95–601, sec. 10, as amended by Pub. L. 102–486, sec. 2902 (42 U.S.C. 5851). Section 50.10 also issued under Atomic Energy Act secs. 101, 185 (42 U.S.C. 2131, 2235); National Environmental Policy Act sec. 102 (42 U.S.C. 4332). Sections 50.13, 50.54(dd), and 50.103 also issued under Atomic Energy Act sec. 108 (42 U.S.C. 2138).

Sections 50.23, 50.35, 50.55, and 50.56 also issued under Atomic Energy Act sec. 185 (42 U.S.C. 2235). Appendix Q also issued under National Environmental Policy Act sec. 102 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97–415 (42 U.S.C. 2239). Section 50.78 also issued under Atomic Energy Act sec. 122 (42 U.S.C. 2152). Sections 50.80—50.81 also issued under Atomic Energy Act sec. 184 (42 U.S.C. 2234).

■ 2. In § 50.54, revise the introductory text of the section, add and reserve paragraph (ii), and add paragraph (jj) to read as follows:

§ 50.54 Conditions of licenses.

The following paragraphs of this section, with the exception of paragraphs (r) and (gg), and the applicable requirements of 10 CFR 50.55a, are conditions in every nuclear power reactor operating license issued under this part. The following paragraphs with the exception of paragraph (r), (s), and (u) of this section are conditions in every combined license issued under part 52 of this chapter, provided, however, that paragraphs (i), (i–1), (j), (k), (l), (m), (n), (w), (x), (y), and (z) of this section are only applicable after the Commission makes the finding under § 52.103(g) of this chapter.

* * * * *

(ii) [Reserved]

(jj) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

■ 3. In § 50.55, revise the introductory text of the section, add and reserve paragraphs (g) and (h), and add paragraph (i) to read as follows:

§ 50.55 Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses.

Each construction permit is subject to the following terms and conditions and the applicable requirements of 10 CFR 50.55a; each early site permit is subject to the terms and conditions in paragraph (f) of this section; each manufacturing license is subject to the terms and conditions in paragraphs (e) and (f) of this section and the applicable requirements of 10 CFR 50.55a; and each combined license is subject to the terms and conditions in paragraphs (e) and (f) of this section and the applicable requirements of 10 CFR 50.55a until the date that the Commission makes the finding under § 52.103(g) of this chapter:

* * * * *

(g) [Reserved]

(h) [Reserved]

(i) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

■ 4. Revise § 50.55a to read as follows:

§ 50.55a Codes and standards.

(a) *Documents approved for incorporation by reference.* The standards listed in this paragraph have been approved for incorporation by reference by the Director of the Federal Register pursuant to 5 U.S.C. 552(a) and 1 CFR Part 51. The standards are available for inspection at the NRC Technical Library, 11545 Rockville Pike, Rockville, Maryland 20852; or at the National Archives and Records Administration (NARA). For information on the availability of this material at NARA, call 202–741–6030 or go to *http://www.archives.gov/federal-register/cfr/ibr-locations.html.*

(1) *American Society of Mechanical Engineers* (*ASME*), Three Park Avenue, New York, NY 10016 (telephone 800–843–2763), *http://www.asme.org/Codes/*.

(i) *ASME Boiler and Pressure Vessel Code, Section III.* The editions and addenda for Section III of the ASME Boiler and Pressure Vessel Code are listed below, but limited to those provisions identified in paragraph (b)(1) of this section.

(A) ''*Rules for Construction of Nuclear Vessels:*''

(1) 1963 Edition,
(2) Summer 1964 Addenda,
(3) Winter 1964 Addenda,
(4) 1965 Edition
(5) 1965 Summer Addenda,
(6) 1965 Winter Addenda,
(7) 1966 Summer Addenda,
(8) 1966 Winter Addenda,
(9) 1967 Summer Addenda,
(*10*) *1967 Winter Addenda,*
(*11*) *1968 Edition,*
(*12*) *1968 Summer Addenda,*
(*13*)*1968 Winter Addenda,*
(*14*) *1969 Summer Addenda,*
(*15*) *1969 Winter Addenda,*
(*16*) *1970 Summer Addenda,* and
(*17*) *1970 Winter Addenda.*

(B) ''*Rules for Construction of Nuclear Power Plant Components:*''

(*1*) *1971 Edition,*
(*2*) *1971 Summer Addenda,*
(*3*) *1971 Winter Addenda,*
(*4*) *1972 Summer Addenda,*
(*5*) *1972 Winter Addenda,*
(*6*) *1973 Summer Addenda,* and
(*7*) *1973 Winter Addenda.*

(C) ''*Division 1 Rules for Construction of Nuclear Power Plant Components:*''

(*1*) *1974 Edition,*
(*2*) *1974 Summer Addenda,*
(*3*) *1974 Winter Addenda,*
(*4*) *1975 Summer Addenda,*
(*5*) *1975 Winter Addenda,*
(*6*) *1976 Summer Addenda,* and
(*7*) *1976 Winter Addenda;*

(D) ''*Rules for Construction of Nuclear Power Plant Components—Division 1;*''

(*1*) *1977 Edition,*
(*2*) *1977 Summer Addenda,*
(*3*) *1977 Winter Addenda,*
(*4*) *1978 Summer Addenda,*
(*5*) *1978 Winter Addenda,*
(*6*) *1979 Summer Addenda,*
(*7*) *1979 Winter Addenda,*
(*8*) *1980 Edition,*
(*9*) *1980 Summer Addenda,*
(*10*) *1980 Winter Addenda,*
(*11*) *1981 Summer Addenda,*
(*12*) *1981 Winter Addenda,*
(*13*) *1982 Summer Addenda,*
(*14*) *1982 Winter Addenda,*
(*15*) *1983 Edition,*
(*16*) *1983 Summer Addenda,*
(*17*) *1983 Winter Addenda,*
(*18*) *1984 Summer Addenda,*
(*19*) *1984 Winter Addenda,*
(*20*) *1985 Summer Addenda,*
(*21*) *1985 Winter Addenda,*
(*22*) *1986 Edition,*
(*23*) *1986 Addenda,*
(*24*) *1987 Addenda,*
(*25*) *1988 Addenda,*
(*26*) *1989 Edition,*
(*27*) *1989 Addenda,*
(*28*) *1990 Addenda,*
(*29*) *1991 Addenda,*
(*30*) *1992 Edition,*
(*31*) *1992 Addenda,*
(*32*) *1993 Addenda,*
(*33*) *1994 Addenda,*
(*34*) *1995 Edition,*
(*35*)*1995 Addenda,*
(*36*)*1996 Addenda,* and
(*37*) *1997 Addenda.*

(E) ''*Rules for Construction of Nuclear Facility Components—Division 1:*''

(*1*) *1998 Edition,*
(*2*) *1998 Addenda,*
(*3*) *1999 Addenda,*
(*4*) *2000 Addenda,*
(*5*) *2001 Edition,*
(*6*) *2001 Addenda,*
(*7*) *2002 Addenda,*
(*8*) *2003 Addenda,*
(*9*) *2004 Edition,*
(*10*) *2005 Addenda,*
(*11*) *2006 Addenda,*
(*12*) *2007 Edition,* and
(*13*) *2008 Addenda.*

(ii) *ASME Boiler and Pressure Vessel Code, Section XI.* The editions and addenda for Section XI of the ASME Boiler and Pressure Vessel Code are listed below, but limited to those provisions identified in paragraph (b)(2) of this section.

(A) ''*Rules for Inservice Inspection of Nuclear Reactor Coolant Systems:*''

(1) 1970 Edition,
(2) 1971 Edition,
(*3*) *1971 Summer Addenda,*
(*4*) *1971 Winter Addenda,*
(*5*) *1972 Summer Addenda,*
(*6*) *1972 Winter Addenda,*
(*7*) *1973 Summer Addenda,* and
(*8*) *1973 Winter Addenda.*

(B) ''*Rules for Inservice Inspection of Nuclear Power Plant Components:*''

(*1*) *1974 Edition,*
(*2*) *1974 Summer Addenda,*
(*3*) *1974 Winter Addenda,* and
(*4*) *1975 Summer Addenda.*

(C) ''*Rules for Inservice Inspection of Nuclear Power Plant Components—Division 1:*''

(*1*) *1977 Edition,*
(*2*) *1977 Summer Addenda,*
(*3*) *1977 Winter Addenda,*
(*4*) *1978 Summer Addenda,*
(*5*) *1978 Winter Addenda,*
(*6*) *1979 Summer Addenda,*
(*7*) *1979 Winter Addenda,*
(*8*) *1980 Edition,*
(*9*) *1980 Winter Addenda,*
(*10*) *1981 Summer Addenda,*
(*11*) *1981 Winter Addenda,*
(*12*) *1982 Summer Addenda,*
(*13*) *1982 Winter Addenda,*
(*14*) *1983 Edition,*
(*15*) *1983 Summer Addenda,*
(*16*) *1983 Winter Addenda,*
(*17*) *1984 Summer Addenda,*
(*18*) *1984 Winter Addenda,*
(*19*) *1985 Summer Addenda,*
(*20*) *1985 Winter Addenda,*
(*21*) *1986 Edition,*
(*22*) *1986 Addenda,*
(*23*) *1987 Addenda,*
(*24*) *1988 Addenda,*
(*25*) *1989 Edition,*
(*26*) *1989 Addenda,*
(*27*) *1990 Addenda,*
(*28*) *1991 Addenda,*
(*28*) *1992 Edition,*
(*30*) *1992 Addenda,*
(*31*) *1993 Addenda,*
(*32*) *1994 Addenda,*
(*33*) *1995 Edition,*
(*34*) *1995 Addenda,*
(*35*) *1996 Addenda,*
(*36*) *1997 Addenda,*
(*37*) *1998 Edition,*
(*38*) *1998 Addenda,*
(*39*) *1999 Addenda,*
(*40*) *2000 Addenda,*
(*41*) *2001 Edition,*
(*42*) *2001 Addenda,*
(*43*) *2002 Addenda,*
(*44*) *2003 Addenda,*

(*45*) *2004 Edition,*
(*46*) *2005 Addenda,*
(*47*) *2006 Addenda,*
(*48*) *2007 Edition,* and
(*49*) *2008 Addenda.*

(iii) *ASME Code Cases: Nuclear Components*

(A) *ASME Code Case N–722–1.* ASME Code Case N–722–1, ''Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1'' (Approval Date: January 26, 2009), with the conditions in paragraph (g)(6)(ii)(E) of this section.

(B) *ASME Code Case N–729–1.* ASME Code Case N–729–1, ''Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1'' (Approval Date: March 28, 2006), with the conditions in paragraph (g)(6)(ii)(D) of this section.

(C) *ASME Code Case N–770–1.* ASME Code Case N–770–1, ''Additional Examinations for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials, Section XI, Division 1'' (Approval Date: December 25, 2009), with the conditions in paragraph (g)(6)(ii)(F) of this section.

(iv) *ASME Operation and Maintenance Code.* The editions and addenda for the ASME Code for Operation and Maintenance of Nuclear Power Plants are listed below, but limited to those provisions identified in paragraph (b)(3) of this section.

(A) ''*Code for Operation and Maintenance of Nuclear Power Plants:''*
(*1*) *1995 Edition,*
(*2*) *1996 Addenda,*
(*3*) *1997 Addenda,*
(*4*) *1998 Edition,*
(*5*) *1999 Addenda,*
(*6*) *2000 Addenda,*
(*7*) *2001 Edition,*
(*8*) *2002 Addenda,*
(*9*) *2003 Addenda,*
(*10) 2004 Edition,*
(*11*) *2005 Addenda,* and
(*12*) *2006 Addenda.*

(B) [Reserved]

(2) *Institute of Electrical and Electronics Engineers (IEEE) Service Center,* 445 Hoes Lane, Piscataway, NJ 08855.

(i) *IEEE standard 279–1971.* (IEEE Std 279–1971), ''Criteria for Protection Systems for Nuclear Power Generating Stations'' (Approval Date: June 3, 1971), referenced in paragraphs (h)(2) of this section.

(ii) *IEEE Standard 603–1991.* (IEEE Std 603–1991), ''Standard Criteria for Safety Systems for Nuclear Power Generating Stations'' (Approval Date: June 27, 1971), referenced in paragraphs (h)(2) and (h)(3) of this section. All other standards that are referenced in IEEE Std 603–1991 are not approved incorporation by reference.

(iii) *IEEE standard 603–1991, correction sheet.* (IEEE Std 603–1991 correction sheet), ''Standard Criteria for Safety Systems for Nuclear Power Generating Stations, Correction Sheet, Issued January 30, 1995, '' referenced in paragraphs (h)(2) and (h)(3) of this section. (Copies of this correction sheet may be purchased from Thomson Reuters, 3916 Ranchero Dr., Ann Arbor, MI 48108, *http://www.techstreet.com.*)

(3) *U.S. Nuclear Regulatory Commission (NRC) Reproduction and Distribution Services Section,* Washington, DC 20555– 0001; fax: 301–415–2289; email: Distribution.Resource@nrc.gov.

(i) *NRC Regulatory Guide 1.84, Revision 36.* NRC Regulatory Guide 1.84, Revision 36, ''Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,'' [INSERT DATE OF FINAL RULE PUBLICATION IN THE **Federal Register**], with the requirements in paragraph (b)(4) of this section.

(ii) *NRC Regulatory Guide 1.147, Revision 17.* NRC Regulatory Guide 1.147, Revision 17, ''Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1,'' [INSERT DATE OF FINAL RULE PUBLICATION IN THE **Federal Register**], which lists ASME Code Cases that the NRC has approved in accordance with the requirements in paragraph (b)(5) of this section.

(iii) *NRC Regulatory Guide 1.192, Revision 1.* NRC Regulatory Guide 1.192, Revision 1, ''Operation and Maintenance Code Case Acceptability, ASME OM Code,'' [INSERT DATE OF FINAL RULE PUBLICATION IN THE **Federal Register**], which lists ASME Code Cases that the NRC has approved in accordance with the requirements in paragraph (b)(6) of this section.

(b) *Use and conditions on the use of standards.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code (BPV Code) and the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) as specified in this paragraph. Each combined license for a utilization facility is subject to the following conditions.

(1) *Conditions on ASME BPV Code Section III.* Each manufacturing license, standard design approval, and design certification under Part 52 of this chapter is subject to the following conditions. As used in this section, references to Section III refer to Section III of the ASME Boiler and Pressure Vessel Code and include the 1963 Edition through 1973 Winter Addenda and the 1974 Edition (Division 1) through the 2008 Addenda (Division 1), subject to the following conditions:

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

(ii) *Section III condition: Weld leg dimensions.* When applying the 1989 Addenda through the latest edition and addenda, applicants or licensees may not apply subparagraphs NB–3683.4(c)(1) and NB–3683.4(c)(2) or Footnote 11 from the 1989 Addenda through the 2003 Addenda, or Footnote 13 from the 2004 Edition through the 2008 Addenda to Figures NC–3673.2(b)–1 and ND–3673.2(b)–1 for welds with leg size less than 1.09 $t_{n.}$

(iii) *Section III condition: Seismic design of piping.* Applicants or licensees may use Subarticles NB–3200, NB–3600, NC–3600, and ND–3600 for seismic design of piping, up to and including the 1993 Addenda, subject to the condition specified in paragraph (b)(1)(ii) of this section. Applicants or licensees may not use these subarticles for seismic design of piping in the 1994 Addenda through the 2005 Addenda incorporated by reference in paragraph (a)(1) of this section, except that Subarticle NB–3200 in the 2004 Edition through the 2008 Addenda may be used by applicants and licensees, subject to the condition in paragraph (b)(1)(iii)(A) of this section. Applicants or licensees may use Subarticles NB–3600, NC–3600, and ND–3600 for the seismic design of piping in the 2006 Addenda through the 2008 Addenda, subject to the conditions of this paragraph corresponding to those subarticles.

(A) *Seismic design of piping: first provision.* When applying Note (1) of Figure NB–3222–1 for Level B service limits, the calculation of P_b stresses must include reversing dynamic loads (including inertia earthquake effects) if evaluation of these loads is required by NB–3223(b).

(B) *Seismic design of piping: second provision.* For Class 1 piping, the material and D_o/t requirements of NB–3656(b) must be met for all Service Limits when the Service Limits include reversing dynamic loads, and the alternative rules for reversing dynamic loads are used.

(iv) *Section III condition: 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 1994 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) *Section III condition: Independence of inspection.* Applicants or licensees may not apply NCA–4134.10(a) of Section III, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1) of this section.

(vi) *Section III condition: Subsection NH.* The provisions in Subsection NH, ''Class 1 Components in Elevated Temperature Service,'' 1995 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1) of this section, may only be used for the design and construction of Type 316 stainless steel pressurizer heater sleeves where service conditions do not cause the components to reach temperatures exceeding 900 °F.

(vii) *Section III condition: Capacity certification and demonstration of function of incompressible-fluid pressure-relief valves.* When applying the 2006 Addenda through the 2007 Edition up to and including the 2008 Addenda, applicants and licensees may use paragraph NB–7742, except that paragraph NB–7742(a)(2) may not be used. For a valve design of a single size to be certified over a range of set pressures, the demonstration of function tests under paragraph NB–7742 must be conducted as prescribed in NB–7732.2 on two valves covering the minimum set pressure for the design and the maximum set pressure that can be accommodated at the demonstration facility selected for the test.

(2) *Conditions on ASME BPV Code Section XI.* As used in this section, references to Section XI refer to Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, and include the 1970 Edition through the 1976 Winter Addenda and the 1977 Edition through the 2007 Edition with the 2008 Addenda, subject to the following conditions:

(i) [Reserved]

(ii) *Section XI condition: Pressure-retaining welds in ASME Code Class 1 piping (applies to Table IWB–2500 and IWB–2500–1 and Category B–J).* If the facility's application for a construction permit was docketed prior to July 1, 1978, the extent of examination for Code Class 1 pipe welds may be determined by the requirements of Table IWB–2500 and Table IWB–2600 Category B–J of Section XI of the ASME BPV Code in the 1974 Edition and Addenda through the Summer 1975 Addenda or other requirements the NRC may adopt.

(iii) [Reserved]

(iv) [Reserved]

(v) [Reserved]

(vi) *Section XI condition: Effective edition and addenda of Subsection IWE and Subsection IWL.* Applicants or 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 conditioned by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section, when implementing the initial 120-month inspection interval for the containment inservice inspection requirements of this section. Successive 120-month interval updates must be implemented in accordance with paragraph (g)(4)(ii) of this section.

(vii) *Section XI condition: 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 (a)(1)(iv) of this section.

(viii) *Section XI condition: Concrete containment examinations.* Applicants or licensees applying Subsection IWL, 1992 Edition with the 1992 Addenda, must apply paragraphs (b)(2)(viii)(A) through (b)(2)(viii)(E) of this section. Applicants or licensees applying Subsection IWL, 1995 Edition with the 1996 Addenda, must apply paragraphs (b)(2)(viii)(A), (b)(2)(viii)(D)(*3*), and (b)(2)(viii)(E) of this section. Applicants or licensees applying Subsection IWL, 1998 Edition through the 2000 Addenda, must apply paragraphs (b)(2)(viii)(E) and (b)(2)(viii)(F) of this section. Applicants or licensees applying Subsection IWL, 2001 Edition through the 2004 Edition, up to and including the 2006 Addenda, must apply paragraphs (b)(2)(viii)(E) through (b)(2)(viii)(G) of this section. Applicants or licensees applying Subsection IWL, 2007 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, must apply paragraph (b)(2)(viii)(E) of this section.

(A) *Concrete containment examinations: first provision.* 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) *Concrete containment examinations: second provision.* 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) *Concrete containment examinations: third provision.* 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) *Concrete containment examinations: fourth provision.* The applicant or licensee must 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; and

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

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

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

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

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

(F) *Concrete containment examinations: sixth provision.* Personnel that examine containment concrete surfaces and tendon hardware, wires, or strands must meet the qualification provisions in IWA–2300. The ''owner-defined'' personnel qualification provisions in IWL–2310(d) are not approved for use.

(G) *Concrete containment examinations: seventh provision.* Corrosion protection material must be restored following concrete containment post-tensioning system repair and replacement activities in accordance with the quality assurance program requirements specified in IWA–1400.

(ix) *Section XI condition: Metal containment examinations.* Applicants or licensees applying Subsection IWE, 1992 Edition with the 1992 Addenda, or the 1995 Edition with the 1996 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A) through (b)(2)(ix)(E) of this section. Applicants or licensees applying Subsection IWE, 1998 Edition through the 2001 Edition with the 2003 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(I) of this section. Applicants or licensees applying Subsection IWE, 2004 Edition, up to and including the 2005 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A), (b)(2)(ix)(B), and (b)(2)(ix)(F) through (b)(2)(ix)(H) of this section. Applicants or licensees applying Subsection IWE, 2004 Edition with the 2006 Addenda, must satisfy the requirements of paragraphs (b)(2)(ix)(A)(*2*) and (b)(2)(ix)(B) of this section. Applicants or licensees applying Subsection IWE, 2007 Edition through the latest addenda incorporated by reference in paragraph (a)(1)(ii) of this section, must satisfy the requirements of paragraphs (b)(2)(ix)(A)(*2*), (b)(2)(ix)(B), and (b)(2)(ix)(J) of this section.

(A) *Metal containment examinations: first provision.* For Class MC applications, the following apply to inaccessible areas.

(*1*) The applicant or licensee must evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or could result in degradation to such inaccessible areas.

(*2*) For each inaccessible area identified for evaluation, the applicant or licensee must provide the following in the ISI Summary Report as required by IWA–6000:

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

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

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

(B) *Metal containment examinations: second provision.* 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) *Metal containment examinations: third provision.* The examinations specified in Examination Category E–B, Pressure Retaining Welds, and Examination Category E–F, Pressure Retaining Dissimilar Metal Welds, are optional.

(D) *Metal containment examinations: fourth provision.* This paragraph (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 that exceeds acceptance standards, the applicant or licensee must 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) *Metal containment examinations: fifth provision.* A general visual examination as required by Subsection IWE must be performed once each period.

(F) *Metal containment examinations: sixth provision.* VT–1 and VT–3 examinations must be conducted in accordance with IWA–2200. Personnel conducting examinations in accordance with the VT–1 or VT–3 examination method must be qualified in accordance with IWA–2300. The ''owner-defined'' personnel qualification provisions in IWE–2330(a) for personnel that conduct VT–1 and VT–3 examinations are not approved for use.

(G) *Metal containment examinations: seventh provision.* The VT–3 examination method must be used to conduct the examinations in Items E1.12 and E1.20 of Table IWE–2500–1, and the VT–1 examination method must be used to conduct the examination in Item E4.11 of Table IWE–2500–1. An examination of the pressure-retaining bolted connections in Item E1.11 of Table IWE–2500–1 using the VT–3 examination method must be conducted once each interval. The ''owner-defined'' visual examination provisions in IWE–2310(a) are not approved for use for VT–1 and VT–3 examinations.

(H) *Metal containment examinations: eighth provision.* Containment bolted connections that are disassembled during the scheduled performance of the examinations in Item E1.11 of Table IWE–2500–1 must be examined using the VT–3 examination method. Flaws or degradation identified during the performance of a VT–3 examination must be examined in accordance with the VT–1 examination method. The criteria in the material specification or IWB–3517.1 must be used to evaluate containment bolting flaws or degradation. As an alternative to performing VT–3 examinations of containment bolted connections that are disassembled during the scheduled performance of Item E1.11, VT–3 examinations of containment bolted connections may be conducted whenever containment bolted connections are disassembled for any reason.

(I) *Metal containment examinations: ninth provision.* The ultrasonic examination acceptance standard specified in IWE–3511.3 for Class MC pressure-retaining components must also be applied to metallic liners of Class CC pressure-retaining components.

(J) *Metal containment examinations: tenth provision.* In general, a repair/replacement activity such as replacing a large containment penetration, cutting a large construction opening in the containment pressure boundary to replace steam generators, reactor vessel heads, pressurizers, or other major equipment; or other similar modification is considered a major containment modification. When applying IWE–5000 to Class MC pressure-retaining components, any major containment modification or repair/replacement must be followed by a Type A test to provide assurance of both containment structural integrity and leaktight integrity prior to returning to service, in accordance with 10 CFR Part 50, Appendix J, Option A or Option B on which the applicant's or licensee's Containment Leak-Rate Testing Program is based. When applying IWE–5000, if a Type A, B, or C Test is performed, the test pressure and acceptance standard for the test must be in accordance with 10 CFR Part 50, Appendix J.

(x) *Section XI condition: 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) [Reserved]

(xii) *Section XI condition: Underwater welding.* The provisions in IWA–4660, ''Underwater Welding,'' of Section XI, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, are not approved for use on irradiated material.

(xiii) [Reserved]

(xiv) *Section XI condition: Appendix VIII personnel qualification.* All personnel qualified for performing ultrasonic examinations in accordance with Appendix VIII must receive 8 hours of annual hands-on training on specimens that contain cracks. Licensees applying the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section may use the annual practice requirements in VII–4240 of Appendix VII of Section XI in place of the 8 hours of annual hands-on training provided that the supplemental practice is performed on material or welds that contain cracks, or by analyzing prerecorded data from material or welds that contain cracks. In either case, training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

(xv) *Section XI condition: Appendix VIII specimen set and qualification requirements.* Licensees using Appendix VIII in the 1995 Edition through the 2001 Edition of the ASME Boiler and Pressure Vessel Code may elect to comply with all of the provisions in paragraphs (b)(2)(xv)(A) through (b)(2)(xv)(M) of this section, except for paragraph (b)(2)(xv)(F) of this section, which may be used at the licensee's option. Licensees using editions and addenda after 2001 Edition through the 2006 Addenda must use the 2001 Edition of Appendix VIII and may elect to comply with all of the provisions in paragraphs (b)(2)(xv)(A) through (b)(2)(xv)(M) of this section, except for paragraph (b)(2)(xv)(F) of this section, which may be used at the licensee's option.

(A) *Specimen set and qualification: first provision.* When applying Supplements 2, 3, and 10 to Appendix VIII, the following examination coverage criteria requirements must be used:

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

(*2*) Where examination from both sides is not possible, full coverage credit may be claimed from a single side for ferritic welds. Where examination from both sides is not possible on austenitic welds or dissimilar metal welds, full coverage credit from a single side may be claimed only after completing a successful single-sided Appendix VIII demonstration using flaws on the opposite side of the weld. Dissimilar metal weld qualifications must be demonstrated from the austenitic side of the weld, and the qualification may be expanded for austenitic welds with no austenitic sides using a separate add-on performance demonstration. Dissimilar metal welds may be examined from either side of the weld.

(B) *Specimen set and qualification: second provision.* The following conditions 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) *Specimen set and qualification: third provision.* When applying Supplement 4 to Appendix VIII, the following conditions must be used:

(*1*) A depth sizing requirement of 0.15 inch RMS must be used in lieu of the requirements in Subparagraphs 3.2(a) and 3.2(c), and a length sizing requirement of 0.75 inch RMS must be used in lieu of the requirement in Subparagraph 3.2(b).

(*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 must 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) *Specimen set and qualification: fourth provision.* The following conditions 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 that 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) *Specimen set and qualification: fifth provision.* When applying Supplement 6 to Appendix VIII, the following conditions 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) *Specimen set and qualification: sixth provision.* The following conditions 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 must apply all of the provisions of Supplements 4 and 6 including the following conditions:

(*1*) For detection and sizing, the total number of flaws must be at least 10. A minimum of 5 flaws must 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) *Specimen set and qualification: seventh provision.* When applying Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6 qualification, the following additional conditions 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), must 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 paragraph (b)(2)(xv)(G)(*1*) of this section 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 conditions in paragraph (b)(2)(xv)(G)(*2*) are met.

(H) *Specimen set and qualification: eighth provision.* 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 must be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches.

(I) *Specimen set and qualification: ninth provision.* When applying Supplement 5, Paragraph (a), to Appendix VIII, the number of false calls allowed must be D/10, with a maximum of 3, where D is the diameter of the nozzle.

(J) [Reserved]

(K) *Specimen set and qualification: eleventh provision.* When performing nozzle-to-vessel weld examinations, the following conditions 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 conditions 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 notches, fabrication flaws, or cracks. Seventy-five (75) percent of the flaws must be cracks or fabrication flaws. Flaw locations and orientations must be selected from the choices shown in paragraph (b)(2)(xv)(K)(*4*) of this section, Table VIII–S7–1—Modified, with the exception that flaws in the outer eighty-five (85) percent of the weld need not be perpendicular to the weld. There may be no more than two flaws from each category, and at least one subsurface flaw must be included.

(*ii*) For length sizing, a minimum of four flaws as in paragraph (b)(2)(xv)(K)(*1*)(*i*) of this section 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 paragraph (b)(2)(xv)(E)(*3*) of this section.

(*iii*) For depth sizing, a minimum of four flaws as in paragraph (b)(2)(xv)(K)(*1*)(*i*) of this section 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 paragraphs (b)(2)(xv)(C)(*1*), (b)(2)(xv)(E)(*1*), and (b)(2)(xv)(F)(*3*) of this section.

(*2*) For examination of reactor pressure vessel nozzle-to-vessel welds conducted from the inside of the vessel, the following conditions are required:

(*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 conditioned by paragraphs (b)(2)(xv)(B) and (b)(2)(xv)(C) of this section.

(*ii*) When the examination volume defined in paragraph (b)(2)(xv)(K)(*2*)(*i*) of this section 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 paragraph (b)(2)(xv)(K)(*1*) of this section.

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

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

(*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 conditioned by paragraphs (b)(2)(xv)(B) and (b)(2)(xv)(C) of this section, for examinations performed in the radial direction, and Supplement 5 to Appendix VIII, as conditioned by paragraph (b)(2)(xv)(J) of this section, for examinations performed in the circumferential direction.

(*ii*) The examination volume not addressed by paragraph (b)(2)(xv)(K)(*3*)(*i*) of this section 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 conditioned by paragraphs (b)(2)(xv)(D) through (b)(2)(xv)(G) of this section.

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

TABLE VIII–S7–1—MODIFIED

Flaw locations and orientations		
	Parallel to weld	Perpendicular to weld
Inner 15 percent	X	X
Outside Diameter Surface ..	X	....
Subsurface	X	....

(L) *Specimen set and qualification: twelfth provision.* As a condition to the requirements of Supplement 8, Subparagraph 1.1(c), to Appendix VIII, notches may be located within one diameter of each end of the bolt or stud.

(M) *Specimen set and qualification: thirteenth provision.* When implementing Supplement 12 to Appendix VIII, only the provisions related to the coordinated implementation of Supplement 3 to Supplement 2 performance demonstrations are to be applied.

(xvi) *Section XI condition: Appendix VIII single side ferritic vessel and piping and stainless steel piping examinations.* When applying editions and addenda prior to the 2007 Edition of Section XI, the following conditions apply.

(A) *Ferritic and stainless steel piping examinations: first provision.* 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 conditioned by this paragraph and paragraphs (b)(2)(xv)(B) through (b)(2)(xv)(G) of this section, on specimens containing flaws with non-optimum sound energy reflecting characteristics or flaws similar to those in the vessel being examined.

(B) *Ferritic and stainless steel piping examinations: second provision.* 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 conditioned by this paragraph and paragraph (b)(2)(xv)(A) of this section.

(xvii) *Section XI condition: Reconciliation of quality requirements.* When purchasing replacement items, in addition to the reconciliation provisions of IWA–4200, 1995 Addenda through 1998 Edition, the replacement items must be purchased, to the extent necessary, in accordance with the licensee's quality assurance program description required by 10 CFR 50.34(b)(6)(ii).

(xviii) *Section XI condition: NDE personnel certification.*

(A) *NDE personnel certification: first provision.* Level I and II nondestructive examination personnel must be recertified on a 3-year interval in lieu of the 5-year interval specified in the 1997 Addenda and 1998 Edition of IWA–2314, and IWA–2314(a) and IWA–2314(b) of the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

(B) *NDE personnel certification: second provision.* When applying editions and addenda prior to the 2007 Edition of Section XI, paragraph IWA–2316 may only be used to qualify personnel that observe leakage during system leakage and hydrostatic tests conducted in accordance with IWA 5211(a) and (b).

(C) *NDE personnel certification: third provision.* When applying editions and addenda prior to the 2005 Addenda of Section XI, licensee's qualifying visual examination personnel for VT–3 visual examination under paragraph IWA–2317 of Section XI must demonstrate the proficiency of the training by administering an initial qualification examination and administering subsequent examinations on a 3-year interval.

(xix) *Section XI condition: Substitution of alternative methods.* The provisions for substituting alternative examination methods, a combination of methods, or newly developed techniques in the 1997 Addenda of IWA–2240 must be applied when using the 1998 Edition through the 2004 Edition of Section XI of the ASME BPV Code. The provisions in IWA–4520(c), 1997 Addenda through the 2004 Edition, allowing the substitution of alternative methods, a combination of methods, or newly developed techniques for the methods specified in the Construction Code, are not approved for use. The provisions in IWA–4520(b)(2) and IWA–4521 of the 2008 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, allowing the substitution of ultrasonic examination for radiographic examination specified in the Construction Code, are not approved for use.

(xx) *Section XI condition: System leakage tests.*

(A) *System leakage tests: first provision.* When performing system leakage tests in accordance with IWA–5213(a), 1997 through 2002 Addenda, the licensee must maintain a 10-minute hold time after test pressure has been reached for Class 2 and Class 3 components that are not in use during normal operating conditions. No hold time is required for the remaining Class 2 and Class 3 components provided that the system has been in operation for at least 4 hours for insulated components or 10 minutes for uninsulated components.

(B) *System leakage tests: second provision.* The NDE provision in IWA–4540(a)(2) of the 2002 Addenda of Section XI must be applied when performing system leakage tests after repair and replacement activities performed by welding or brazing on a pressure retaining boundary using the 2003 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

(xxi) *Section XI condition: Table IWB–2500–1 examination requirements.*

(A) *Table IWB–2500–1 examination requirements: first provision.* The provisions of Table IWB–2500–1, Examination Category B–D, Full Penetration Welded Nozzles in Vessels, Items B3.40 and B3.60 (Inspection Program A) and Items B3.120 and B3.140 (Inspection Program B) of the 1998 Edition must be applied when using the 1999 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section. A visual examination with magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria in Table IWB–3512–1, 1997 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, with a limiting assumption on the flaw aspect ratio (i.e., a/l = 0.5), may be performed instead of an ultrasonic examination.

(B) [Reserved]

(xxii) *Section XI condition: Surface examination.* The use of the provision in IWA–2220, ''Surface Examination,'' of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, that allows use of an ultrasonic examination method is prohibited.

(xxiii) *Section XI condition: Evaluation of thermally cut surfaces.* The use of the provisions for eliminating mechanical processing of thermally cut surfaces in IWA–4461.4.2 of Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, is prohibited.

(xxiv) *Section XI condition: Incorporation of the performance demonstration initiative and addition of ultrasonic examination criteria.* The use of Appendix VIII and the supplements to Appendix VIII and Article I–3000 of Section XI of the ASME BPV Code, 2002 Addenda through the 2006 Addenda, is prohibited.

(xxv) *Section XI condition: Mitigation of defects by modification.* The use of the provisions in IWA–4340, ''Mitigation of Defects by Modification,'' Section XI, 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section are prohibited.

(xxvi) *Section XI condition: Pressure testing Class 1, 2 and 3 mechanical joints.* The repair and replacement activity provisions in IWA–4540(c) of the 1998 Edition of Section XI for pressure testing Class 1, 2, and 3 mechanical joints must be applied when using the 2001 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section.

(xxvii) *Section XI condition: Removal of insulation.* When performing visual examination in accordance with IWA–5242 of Section XI of the ASME BPV Code, 2003 Addenda through the 2006 Addenda, or IWA–5241 of the 2007 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, insulation must be removed from 17–4 PH or 410 stainless steel studs or bolts aged at a temperature below 1100 °F or having a Rockwell Method C hardness value above 30, and from A–286 stainless steel studs or bolts preloaded to 100,000 pounds per square inch or higher.

(xxviii) *Section XI condition: Analysis of flaws.* Licensees using ASME BPV Code, Section XI, Appendix A, must use the following conditions when implementing Equation (2) in A–4300(b)(1):

For $R < 0$, ΔK_I depends on the crack depth (a), and the flow stress (σ_f). The flow stress is defined by $\sigma_f = 1/2(\sigma_{ys} + \sigma_{ult})$, where σ_{ys} is the yield strength and σ_{ult} is the ultimate tensile strength in units ksi (MPa) and (a) is in units in. (mm). For $-2 \leq R \leq 0$ and $K_{max} - K_{min} \leq 0.8 \times 1.12\ \sigma_f \sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = K_{max}$. For $R < -2$ and $K_{max} - K_{min} \leq 0.8 \times 1.12\ \sigma_f \sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = (1 - R) K_{max}/3$. For $R < 0$ and $K_{max} - K_{min} > 0.8 \times 1.12\ \sigma_f \sqrt{(\pi a)}$, $S = 1$ and $\Delta K_I = K_{max} - K_{min}$.

(xxix) *Section XI condition: Nonmandatory Appendix R.* Nonmandatory Appendix R, ''Risk-Informed Inspection Requirements for Piping,'' of Section XI, 2005 Addenda through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, may not be implemented without prior NRC authorization of the proposed alternative in accordance with paragraph (z) of this section.

(3) *Conditions on ASME OM Code.* As used in this section, references to the OM Code refer to the ASME Code for Operation and Maintenance of Nuclear Power Plants, Subsections ISTA, ISTB, ISTC, ISTD, Mandatory Appendices I and II, and Nonmandatory Appendices A through H and J, including the 1995 Edition through the 2006 Addenda, subject to the following conditions:

(i) *OM condition: Quality assurance.* When applying editions and addenda of the OM Code, the requirements of NQA–1, ''Quality Assurance Requirements for Nuclear Facilities,'' 1979 Addenda, are acceptable as permitted by ISTA 1.4 of the 1995 Edition through 1997 Addenda or ISTA–1500 of the 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, provided the licensee uses its 10 CFR Part 50, Appendix B, quality assurance program in conjunction with the OM Code requirements. Commitments contained in the licensee's quality assurance program description that are more stringent than those contained in NQA–1 govern OM Code activities. If NQA–1 and the OM Code do not address the commitments contained in the licensee's Appendix B quality assurance program description, the commitments must be applied to OM Code activities.

(ii) *OM condition: Motor-Operated Valve (MOV) testing.* Licensees must comply with the provisions for MOV testing in OM Code ISTC 4.2, 1995 Edition with the 1996 and 1997 Addenda, or ISTC–3500, 1998 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, and must establish a program to ensure that motor-operated valves continue to be capable of performing their design basis safety functions.

(iii) [Reserved]

(iv) *OM condition: Check valves (Appendix II).* Licensees applying Appendix II, ''Check Valve Condition Monitoring Program,'' of the OM Code, 1995 Edition with the 1996 and 1997 Addenda, must satisfy the requirements of paragraphs (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(C) of this section. Licensees applying Appendix II, 1998 Edition through the 2002 Addenda, must satisfy the requirements of paragraphs (b)(3)(iv)(A), (b)(3)(iv)(B), and (b)(3)(iv)(D) of this section.

(A) *Check valves: first provision.* Valve opening and closing functions must be demonstrated when flow testing or examination methods (nonintrusive, or disassembly and inspection) are used;

(B) *Check valves: second provision.* 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) *Check valves: third provision.* If the Appendix II condition monitoring program is discontinued, then the requirements of ISTC 4.5.1 through 4.5.4 must be implemented.

(D) *Check valves: fourth provision.* The applicable provisions of subsection ISTC must be implemented if the Appendix II condition monitoring program is discontinued.

(v) *OM condition: Snubbers ISTD.* Article IWF–5000, ''Inservice Inspection Requirements for Snubbers,'' of the ASME BPV Code, Section XI, must be used when performing inservice inspection examinations and tests of snubbers at nuclear power plants, except as conditioned in paragraphs (b)(3)(v)(A) and (b)(3)(v)(B) of this section.

(A) *Snubbers: first provision.* Licensees may use Subsection ISTD, ''Preservice and Inservice Examination and Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants,'' ASME OM Code, 1995 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(iv) of this section, in place of the requirements for snubbers in the editions and addenda up to the 2005 Addenda of the ASME BPV Code, Section XI, IWF–5200(a) and (b) and IWF–5300(a) and (b), by making appropriate changes to their technical specifications or licensee-controlled documents. Preservice and inservice examinations must be performed using the VT–3 visual examination method described in IWA–2213.

(B) *Snubbers: second provision.* Licensees must comply with the provisions for examining and testing snubbers in Subsection ISTD of the ASME OM Code and make appropriate changes to their technical specifications or licensee-controlled documents when using the 2006 Addenda and later editions and addenda of Section XI of the ASME BPV Code.

(vi) *OM condition: Exercise interval for manual valves.* Manual valves must be exercised on a 2-year interval rather than the 5-year interval specified in paragraph ISTC–3540 of the 1999 through the 2005 Addenda of the ASME OM Code, provided that adverse conditions do not require more frequent testing.

(4) *Conditions on Design, Fabrication, and Materials Code Cases.* Each manufacturing license, standard design approval, and design certification application under Part 52 of this chapter is subject to the following conditions. Licensees may apply the ASME BPV Code Cases listed in NRC Regulatory Guide 1.84, Revision 36, without prior NRC approval, subject to the following conditions:

(i) *Design, Fabrication, and Materials Code Case condition: Applying Code Cases.* When an applicant or licensee initially applies a listed Code Case, the applicant or licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.

(ii) *Design, Fabrication, and Materials Code Case condition: Applying different revisions of Code Cases.* If an applicant or licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the applicant or licensee may continue to apply the previous version of the Code Case as authorized or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use, until it updates its Code of Record for the component being constructed.

(iii) *Design, Fabrication, and Materials Code Case condition: Applying annulled Code Cases.* Application of an annulled Code Case is prohibited unless an applicant or licensee applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.84. If an applicant or licensee has applied a listed Code Case that is later listed as annulled in Regulatory Guide 1.84, the applicant or licensee may continue to apply the Code Case until it updates its Code of Record for the component being constructed.

(5) *Conditions on inservice inspection Code Cases.* Licensees may apply the ASME BPV Code Cases listed in Regulatory Guide 1.147, Revision 17, without prior NRC approval, subject to the following:

(i) *ISI Code Case condition: Applying Code Cases.* When a licensee initially applies a listed Code Case, the licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.

(ii) *ISI Code Case condition: Applying different revisions of Code Cases.* If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case, as authorized, or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. Licensees who choose to continue use of the Code Case during subsequent 120-month ISI program intervals will be required to implement the latest version incorporated by reference into 10 CFR 50.55a as listed in Tables 1 and 2 of Regulatory Guide 1.147, Revision 17.

(iii) *ISI Code Case condition: Applying annulled Code Cases.* Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.147. If a licensee has applied a listed Code Case that is later listed as annulled in Regulatory Guide 1.147, the licensee may continue to apply the Code Case to the end of the current 120-month interval.

(6) *Conditions on Operation and Maintenance of Nuclear Power Plants Code Cases.* Licensees may apply the ASME Operation and Maintenance Code Cases listed in Regulatory Guide 1.192, Revision 1, without prior NRC approval, subject to the following:

(i) *OM Code Case condition: Applying Code Cases.* When a licensee initially applies a listed Code Case, the licensee must apply the most recent version of that Code Case incorporated by reference in paragraph (a) of this section.

(ii) *OM Code Case condition: Applying different revisions of Code Cases.* If a licensee has previously applied a Code Case and a later version of the Code Case is incorporated by reference in paragraph (a) of this section, the licensee may continue to apply, to the end of the current 120-month interval, the previous version of the Code Case, as authorized, or may apply the later version of the Code Case, including any NRC-specified conditions placed on its use. Licensees who choose to continue use of the Code Case during subsequent 120-month ISI program intervals will be required to implement the latest version incorporated by reference into 10 CFR 50.55a as listed in Tables 1 and 2 of Regulatory Guide 1.192, Revision 1.

(iii) *OM Code Case condition: Applying annulled Code Cases.* Application of an annulled Code Case is prohibited unless a licensee previously applied the listed Code Case prior to it being listed as annulled in Regulatory Guide 1.192. If a licensee has applied a listed Code Case that is later listed as annulled in Regulatory Guide 1.192, the licensee may continue to apply the Code Case to the end of the current 120-month interval.

(c) *Reactor coolant pressure boundary.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code as specified in this paragraph. Each manufacturing license, standard design approval, and design certification application under Part 52 of this chapter and each combined license for a utilization facility is subject to the following conditions:

(1) *Standards requirement for reactor coolant pressure boundary components.* Components that are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III[4,5] of the ASME BPV Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

(2) *Exceptions to reactor coolant pressure boundary standards requirement.* Components that are connected to the reactor coolant system and are part of the reactor coolant pressure boundary as defined in § 50.2 need not meet the requirements of paragraph (c)(1) of this section, provided that*:*

(i) *Exceptions: Shutdown and cooling capability.* In the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system; or

(ii) *Exceptions: Isolation capability.* The component is or can be isolated from the reactor coolant system by two valves in series (both closed, both open, or one closed and the other open). Each open valve must be capable of automatic actuation and, assuming the other valve is open, its closure time must be such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only.

(3) *Applicable Code and Code Cases and conditions on their use.* The Code edition, addenda, and optional ASME Code Cases to be applied to components of the reactor coolant pressure boundary must be determined by the provisions of paragraph NCA–1140, Subsection NCA of Section III of the ASME BPV Code, subject to the following conditions:

(i) *Reactor coolant pressure boundary condition: Code edition and addenda.* The edition and addenda applied to a component must be those that are incorporated by reference in paragraph (a)(1)(i) of this section;

(ii) *Reactor coolant pressure boundary condition: Earliest edition and addenda for pressure vessel.* The ASME Code provisions applied to the pressure vessel may be dated no earlier than the summer 1972 Addenda of the 1971 Edition;

(iii) *Reactor coolant pressure boundary condition: Earliest edition and addenda for piping, pumps, and valves.* The ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 Edition; and

(iv) *Reactor coolant pressure boundary condition: Use of Code Cases.* The optional Code Cases applied to a component must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (a)(3)(i) of this section.

(4) *Standards requirement for components in older plants.* For a nuclear power plant whose construction permit was issued prior to May 14, 1984, the applicable Code edition and addenda for a component of the reactor coolant pressure boundary continue to be that Code edition and addenda that were required by Commission regulations for such a component at the time of issuance of the construction permit.

(d) *Quality Group B components.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code as specified in this paragraph. Each manufacturing license, standard design approval, and design certification application under Part 52 of this chapter, and each combined license for a utilization facility is subject to the following conditions:

(1) *Standards requirement for Quality Group B components.* For a nuclear power plant whose application for a construction permit under this part, or a combined license or manufacturing license under Part 52 of this chapter, docketed after May 14, 1984, or for an application for a standard design approval or a standard design certification docketed after May 14, 1984, components classified Quality Group B[9] must meet the requirements for Class 2 Components in Section III of the ASME BPV Code.

(2) Quality Group B: *Applicable Code and Code Cases and conditions on their use.* The Code edition, addenda, and optional ASME Code Cases to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA–1140, Subsection NCA of Section III of the ASME BPV Code, subject to the following conditions:

(i) *Quality Group B condition: Code edition and addenda.* The edition and addenda must be those that are incorporated by reference in paragraph (a)(1)(i) of this section;

(ii) *Quality Group B condition: Earliest edition and addenda for components.* The ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition; and

(iii) *Quality Group B condition: Use of Code Cases.* The optional Code Cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (a)(3)(i) of this section.

(e) *Quality Group C components.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code as specified in this paragraph. Each manufacturing license, standard design approval, and design certification application under Part 52 of this chapter and each combined license for a utilization facility is subject to the following conditions.

(1) *Standards requirement for Quality Group C components.* For a nuclear power plant whose application for a construction permit under this part, or a combined license or manufacturing license under Part 52 of this chapter, docketed after May 14, 1984, or for an application for a standard design approval or a standard design certification docketed after May 14, 1984, components classified Quality Group C[9] must meet the requirements for Class 3 components in Section III of the ASME BPV Code.

(2) Quality Group C a*pplicable Code and Code Cases and conditions on their use.* The Code edition, addenda, and optional ASME Code Cases to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA–1140, subsection NCA of Section III of the ASME BPV Code, subject to the following conditions:

(i) *Quality Group C condition: Code edition and addenda.* The edition and addenda must be those incorporated by reference in paragraph (a)(1)(i) of this section;

(ii) *Quality Group C condition: Earliest edition and addenda for components.* The ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition; and

(iii) *Quality Group C condition: Use of Code Cases.* The optional Code Cases must be those listed in NRC Regulatory Guide 1.84 that is incorporated by reference in paragraph (a)(3)(i) of this section.

(f) *Inservice testing requirements.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code and ASME Code for Operation and Maintenance of Nuclear Power Plants as specified in this paragraph. Each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions. Each combined license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions, but the conditions in paragraphs (f)(4), (f)(5), and (f)(6) of this section must be met only after the Commission makes the finding under § 52.103(g) of this chapter. 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) *Inservice testing requirements for older plants (pre-1971 CPs).* 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 that are part of the reactor coolant pressure boundary must meet the requirements applicable to components that 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 BPV or OM Code) must meet the test requirements applicable to components that are classified as ASME Code Class 2 or Class 3.

(2) *Design and accessibility requirements for performing inservice testing in plants with CPs issued between 1971 and 1974.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, pumps and valves that are classified as ASME Code Class 1 and Class 2 must be designed and provided with access to enable the performance of inservice tests for operational readiness set forth in editions and addenda of Section XI of the ASME BPV incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, or Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraphs (a)(3)(ii) and (a)(3)(iii) of this section, respectively) in effect 6 months before the date of issuance of the construction permit. The pumps and valves may meet the inservice test requirements set forth in subsequent editions of this Code and addenda that are incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17; or Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraphs (a)(3)(ii) and (a)(3)(iii) of this section, respectively), subject to the applicable conditions listed therein.

(3) *Design and accessibility requirements for performing inservice testing in plants with CPs issued after 1974.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit under this part or design approval, design certification, combined license, or manufacturing license under Part 52 of this chapter was issued on or after July 1, 1974:

(i)–(ii) [Reserved]

(iii) *IST design and accessibility requirements: Class 1 pumps and valves.*

(A) *Class 1 pumps and valves: first provision.* In facilities whose construction permit was issued before November 22, 1999, pumps and valves that are classified as ASME Code Class 1 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, or Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraphs (a)(3)(ii) and (a)(3)(iii) of this section, respectively) applied to the construction of the particular pump or valve or the summer 1973 Addenda, whichever is later.

(B) *Class1 pumps and valves: second provision.* In facilities whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter, issued on or after November 22, 1999, pumps and valves that are classified as ASME Code Class 1 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraph (a)(3)(iii) of this section), incorporated by reference in paragraph (a)(1)(iv) of this section at the time the construction permit, combined license, manufacturing license, design certification, or design approval is issued.

(iv) *IST design and accessibility requirements: Class 2 and 3 pumps and valves.*

(A) *Class 2 and 3 pumps and valves: first provision.* In facilities whose construction permit was issued before November 22, 1999, pumps and valves that are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, that are incorporated by reference in paragraph (a)(3)(ii) of this section) applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(B) *Class 2 and 3 pumps and valves: second provision.* In facilities whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter, issued on or after November 22, 1999, pumps and valves that are classified as ASME Code Class 2 and 3 must be designed and provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in editions and addenda of the ASME OM Code (or the optional ASME OM Code Cases listed in NRC Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraph (a)(3)(iii) of this section), incorporated by reference in paragraph (a)(1)(iv) of this section at the time the construction permit, combined license, or design certification is issued.

(v) *IST design and accessibility requirements: Meeting later IST requirements.* All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed in paragraph (b) of this section.

(4) *Inservice testing standards requirement for operating plants.* Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, pumps and valves that 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 (a)(1)(iv) of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

(i) *Applicable IST Code: Initial 120-month interval.* Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the OM Code incorporated by reference in paragraph (a)(1)(iv) of this section on the date 12 months before the date of issuance of the operating license under this part, or 12 months before the date scheduled for initial loading of fuel under a combined license under Part 52 of this chapter (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.192, Revision 1, that is incorporated by reference in paragraph (a)(3)(iii) of this section, subject to the conditions listed in paragraph (b) of this section.

(ii) *Applicable IST Code: Successive 120-month intervals.* Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the OM Code incorporated by reference in paragraph (a)(1)(iv) of this section 12 months before the start of the 120-month interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, or Regulatory Guide 1.192, Revision 1, that are incorporated by reference in paragraphs (a)(3)(ii) and (a)(3)(iii) of this section, respectively), subject to the conditions listed in paragraph (b) of this section.

(iii) [Reserved]

(iv) *Applicable IST Code: Use of later Code editions and addenda.* Inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (a)(1)(iv) of this section, subject to the conditions listed in paragraph (b) of this section, and subject to NRC approval. Portions of editions or addenda may be used, provided that all related requirements of the respective editions or addenda are met.

(5) *Requirements for updating IST programs.*

(i) *IST program update: Applicable IST Code editions and addenda.* The inservice test program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (f)(4) of this section.

(ii) *IST program update: Conflicting IST Code requirements with technical specifications.* If a revised inservice test program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program. The licensee must submit this application, as specified in § 50.4, at least 6 months before the start of the period during which the provisions become applicable, as determined by paragraph (f)(4) of this section.

(iii) *IST program update: Notification of impractical IST Code requirements.* If the licensee has determined that conformance with certain Code requirements is impractical for its facility, the licensee must notify the Commission and submit, as specified in § 50.4, information to support the determination.

(iv) *IST program update: Schedule for completing impracticality determinations.* Where a pump or valve test requirement by the Code or addenda is determined to be impractical by the licensee and is not included in the revised inservice test program (as permitted by paragraph (f)(4) of this section), the basis for this determination must be submitted for NRC review and approval not later than 12 months after the expiration of the initial 120-month interval of operation from the start of facility commercial operation and each subsequent 120-month interval of operation during which the test is determined to be impractical.

(6) *Actions by the Commission for evaluating impractical and augmented IST Code requirements.*

(i) *Impractical IST requirements: Granting of relief.* The Commission will evaluate determinations under paragraph (f)(5) of this section that code requirements are impractical. The Commission may grant relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) *Augmented IST requirements.* The Commission may require the licensee to follow an augmented inservice test program for pumps and valves for which the Commission deems that added assurance of operational readiness is necessary.

(g) *Inservice inspection requirements.* Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME BPV Code as specified in this paragraph. Each operating license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions. Each combined license for a boiling or pressurized water-cooled nuclear facility is subject to the following conditions, but the conditions in paragraphs (g)(4), (g)(5), and (g)(6) of this section must be met only after the Commission makes the finding under § 52.103(g) of this chapter. Requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in § 50.55a(f).

(1) *Inservice inspection requirements for older plants (pre-1971 CPs).* 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 that are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components that 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 that are classified as ASME Code Class 2 or Class 3.

(2) *Design and accessibility requirements for performing inservice inspection in plants with CPs issued between 1971 and 1974.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) that are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions and addenda of Section III or Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1) of this section (or the optional ASME Code

Cases listed in NRC Regulatory Guide 1.147, Revision 17, that are incorporated by reference in paragraph (a)(3)(ii) of this section) in effect 6 months before the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of this Code that are incorporated by reference in paragraph (a) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, that are incorporated by reference in paragraph (a)(3)(ii) of this section), subject to the applicable limitations and modifications.

(3) *Design and accessibility requirements for performing inservice inspection in plants with CPs issued after 1974.* For a boiling or pressurized water-cooled nuclear power facility, whose construction permit under this part, or design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter, was issued on or after July 1, 1974, the following are required:

(i) *ISI design and accessibility requirements: Class 1 components and supports.* Components (including supports) that are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section III or Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, that are incorporated by reference in paragraph (a)(3)(ii) of this section) applied to the construction of the particular component.

(ii) *ISI design and accessibility requirements: Class 2 and 3 components and supports.* Components that are classified as ASME Code Class 2 and Class 3 and supports for components that are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and provided with access to enable the performance of inservice examination of these components and must meet the preservice examination requirements set forth in the editions and addenda of Section XI of the ASME BPV Code incorporated by reference in paragraph (a)(1)(ii) of this section (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, that are incorporated by reference in paragraph (a)(3)(ii) of this section) applied to the construction of the particular component.

(iii)–(iv) [Reserved]

(v) *ISI design and accessibility requirements: Meeting later ISI requirements.* All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed therein.

(4) *Inservice inspection standards requirement for operating plants.* Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that 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 and addenda of the ASME BPV Code (or ASME OM Code for snubber examination and testing) 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 (a)(1)(ii) or (a)(1)(iv) for snubber examination and testing of this section, to the extent practical within the limitations of design, geometry, and materials of construction of the components. Components that are classified as Class MC pressure retaining components and their integral attachments, and components that 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 BPV Code and addenda that are incorporated by reference in paragraph (a)(1)(ii) of this section, subject to the condition listed in paragraph (b)(2)(vi) of this section and the conditions listed in paragraphs (b)(2)(viii) and (b)(2)(ix) of this section, to the extent practical within the limitation of design, geometry, and materials of construction of the components.

(i) *Applicable ISI Code: Initial 120-month interval.* Inservice examination of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (a) of this section on the date 12 months before the date of issuance of the operating license under this part, or 12 months before the date scheduled for initial loading of fuel under a combined license under Part 52 of this chapter (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, when using Section XI, or Regulatory Guide 1.192, Revision 1, when using the OM Code, that are incorporated by reference in paragraphs (a)(3)(ii) and (a)(3)(iii) of this section, respectively), subject to the conditions listed in paragraph (b) of this section.

(ii) *Applicable ISI Code: Successive 120-month intervals.* Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (a) of this section 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, when using Section XI, or Regulatory Guide 1.192, Revision 1, when using the OM Code, that are incorporated by reference in paragraphs (a)(3)(ii) and (a)(3)(iii) of this section), subject to the conditions listed in paragraph (b) of this section. However, a licensee whose inservice inspection interval commences during the 12 through 18-month period after July 21, 2011, may delay the update of their Appendix VIII program by up to 18 months after July 21, 2011.

(iii) *Applicable ISI Code: Optional surface examination requirement.* When applying editions and addenda prior to the 2003 Addenda of Section XI of the ASME BPV Code, 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.

(iv) *Applicable ISI Code: Use of subsequent Code editions and addenda.* Inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (a) of this section, subject to the conditions listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used, provided that all related requirements of the respective editions or addenda are met.

(v) *Applicable ISI Code: Metal and concrete containments.* For a boiling or pressurized water-cooled nuclear power facility whose construction permit under this part or combined license under Part 52 of this chapter was issued after January 1, 1956, the following are required:

(A) *Metal and concrete containments: first provision.* Metal containment pressure retaining components and their integral attachments must meet the inservice inspection, repair, and replacement requirements applicable to components that are classified as ASME Code Class MC;

(B) *Metal and concrete containments: second provision.* Metallic shell and penetration liners that are pressure retaining components and their integral attachments in concrete containments must meet the inservice inspection, repair, and replacement requirements applicable to components that are classified as ASME Code Class MC; and

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

(5) *Requirements for updating ISI programs.*

(i) *ISI program update: Applicable ISI Code editions and addenda.* The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

(ii) *ISI program update: Conflicting ISI Code requirements with technical specifications.* If a revised inservice inspection program for a facility conflicts with the technical specifications for the facility, the licensee must apply to the Commission for amendment of the technical specifications to conform the technical specifications to the revised program. The licensee must submit this application, as specified in § 50.4, at least six months before the start of the period during which the provisions become applicable, as determined by paragraph (g)(4) of this section.

(iii) *ISI program update: Notification of impractical ISI Code requirements.* If the licensee has determined that conformance with a Code requirement is impractical for its facility the licensee must notify the NRC and submit, as specified in § 50.4, information to support the determinations. Determinations of impracticality in accordance with this section must be based on the demonstrated limitations experienced when attempting to comply with the Code requirements during the inservice inspection interval for which the request is being submitted. Requests for relief made in accordance with this section must be submitted to the NRC no later than 12 months after the expiration of the initial or subsequent 120-month inspection interval for which relief is sought.

(iv) *ISI program update: Schedule for completing impracticality determinations.* Where the licensee determines that an examination required by Code edition or addenda is impractical, the basis for this determination must be submitted for NRC review and approval not later than 12 months after the expiration of the initial or subsequent 120-month inspection interval for which relief is sought.

(6) *Actions by the Commission for evaluating impractical and augmented ISI Code requirements.*

(i) *Impractical ISI requirements: Granting of relief.* The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines are authorized by law, will not endanger life or property or the common defense and security, and are otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

(ii) *Augmented ISI program.* The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

(A) [Reserved]

(B) *Augmented ISI requirements: Submitting containment ISI programs.* Licensees do not have to submit to the NRC for approval of their containment inservice inspection programs that were developed to satisfy the requirements of Subsection IWE and Subsection IWL with specified conditions. The program elements and the required documentation must be maintained on site for audit.

(C) *Augmented ISI requirements: Implementation of Appendix VIII to Section XI.*

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

(*2*) Licensees implementing the 1989 Edition and earlier editions and addenda of IWA–2232 of Section XI, Division 1, of the ASME BPV Code must implement the 1995 Edition with the 1996 Addenda of Appendix VIII and the supplements to Appendix VIII of Section XI, Division 1, of the ASME BPV Code.

(D) *Augmented ISI requirements: Reactor vessel head inspections.*

(*1*) All licensees of pressurized water reactors must augment their inservice inspection program with ASME Code Case N–729–1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(*2*) through (*6*) of this section. Licensees of existing operating reactors as of September 10, 2008, must implement their augmented inservice inspection program by December 31, 2008. Once a licensee implements this requirement, the First Revised NRC Order EA–03–009 no longer applies to that licensee and must be deemed to be withdrawn.

(*2*) Note 9 of ASME Code Case N–729–1 must not be implemented.

(*3*) Instead of the specified ''examination method'' requirements for volumetric and surface examinations in Note 6 of Table 1 of Code Case N–729–1, the licensee must perform volumetric and/or surface examination of essentially 100 percent of the required volume or equivalent surfaces of the nozzle tube, as identified by Figure 2 of ASME Code Case N–729–1. A demonstrated volumetric or surface leak path assessment through all J-groove welds must be performed. If a surface examination is being substituted for a volumetric examination on a portion of a penetration nozzle that is below the toe of the J-groove weld [Point E on Figure 2 of ASME Code Case N–729–1], the surface examination must be of the inside and outside wetted surface of the penetration nozzle not examined volumetrically.

(*4*) By September 1, 2009, ultrasonic examinations must be performed using personnel, procedures, and equipment that have been qualified by blind demonstration on representative mockups using a methodology that meets the conditions specified in paragraphs (g)(6)(ii)(D)(*4*)(*i*) through (g)(6)(ii)(D)(*4*)(*iv*), instead of the qualification requirements of Paragraph –2500 of ASME Code Case N–729–1. References herein to Section XI, Appendix VIII, must be to the 2004 Edition with no addenda of the ASME BPV Code.

(*i*) The specimen set must have an applicable thickness qualification range of +25 percent to −40 percent for nominal depth through-wall thickness. The specimen set must include geometric and material conditions that normally require discrimination from primary water stress corrosion cracking (PWSCC) flaws.

(*ii*) The specimen set must have a minimum of ten (10) flaws that provide an acoustic response similar to PWSCC indications. All flaws must be greater than 10 percent of the nominal pipe wall thickness. A minimum of 20 percent of the total flaws must initiate from the inside surface and 20 percent from the outside surface. At least 20 percent of the flaws must be in the depth ranges of 10–30 percent through-wall thickness and at least 20 percent within a depth range of 31–50 percent through-wall thickness. At least 20 percent and no more than 60 percent of the flaws must be oriented axially.

(*iii*) Procedures must identify the equipment and essential variables and settings used for the qualification, in accordance with Subarticle VIII–2100 of Section XI, Appendix VIII. The procedure must be requalified when an essential variable is changed outside the demonstration range as defined by Subarticle VIII–3130 of Section XI, Appendix VIII, and as allowed by Articles VIII–4100, VIII–4200, and VIII–4300 of Section XI, Appendix VIII. Procedure qualification must include the equivalent of at least three personnel performance demonstration test sets. Procedure qualification requires at least one successful personnel performance demonstration.

(*iv*) Personnel performance demonstration test acceptance criteria must meet the personnel performance demonstration detection test acceptance criteria of Table VIII—S10–1 of Section XI, Appendix VIII, Supplement 10. Examination procedures, equipment, and personnel are qualified for depth sizing and length sizing when the RMS error, as defined by Subarticle VIII–3120 of Section XI, Appendix VIII, of the flaw depth measurements, as compared to the true flaw depths, do not exceed ⅛ inch (3 mm) and the root mean square (RMS) error of the flaw length measurements, as compared to the true flaw lengths, do not exceed ⅜ inch (10 mm), respectively.

(*5*) If flaws attributed to PWSCC have been identified, whether acceptable or not for continued service under Paragraphs –3130 or –3140 of ASME Code Case N–729–1, the re-inspection interval must be each refueling outage instead of the re-inspection intervals required by Table 1, Note (8), of ASME Code Case N–729–1.

(*6*) Appendix I of ASME Code Case N–729–1 must not be implemented without prior NRC approval.

(E) *Augmented ISI requirements: Reactor coolant pressure boundary visual inspections.*

(*1*) All licensees of pressurized water reactors must augment their inservice inspection program by implementing ASME Code Case N–722–1, subject to the conditions specified in paragraphs (g)(6)(ii)(E)(*2*) through (g)(6)(ii)(E)(*4*) of this section. The inspection requirements of ASME Code Case N–722–1 do not apply to components with pressure retaining welds fabricated with Alloy 600/82/182 materials that have been mitigated by weld overlay or stress improvement.

(*2*) If a visual examination determines that leakage is occurring from a specific item listed in Table 1 of ASME Code Case N–722–1 that is not exempted by the ASME Code, Section XI, IWB–1220(b)(1), additional actions must be performed to characterize the location, orientation, and length of a crack or cracks in Alloy 600 nozzle wrought material and location, orientation, and length of a crack or cracks in Alloy 82/182 butt welds. Alternatively, licensees may replace the Alloy 600/82/182 materials in all the components under the item number of the leaking component.

(*3*) If the actions in paragraph (g)(6)(ii)(E)(*2*) of this section determine that a flaw is circumferentially oriented and potentially a result of primary water stress corrosion cracking, licensees must perform non-visual NDE inspections of components that fall under that ASME Code Case N–722–1 item number. The number of components inspected must equal or exceed the number of components found to be leaking under that item number. If circumferential cracking is identified in the sample, non-visual NDE must be performed in the remaining components under that item number.

(4) If ultrasonic examinations of butt welds are used to meet the NDE requirements in paragraphs (g)(6)(ii)(E)(*2*) or (g)(6)(ii)(E)(*3*) of this section, they must be performed using the appropriate supplement of Section XI, Appendix VIII, of the ASME BPV Code.

(F) *Augmented ISI requirements: Examination requirements for Class 1 piping and nozzle dissimilar-metal butt welds.*

(*1*) Licensees of existing, operating pressurized-water reactors as of July 21, 2011, must implement the requirements of ASME Code Case N–770–1, subject to the conditions specified in paragraphs (g)(6)(ii)(F)*(2)* through (g)(6)(ii)(F)*(10)* of this section, by the first refueling outage after August 22, 2011.

(*2*) Full structural weld overlays authorized by the NRC staff may be categorized as Inspection Items C or F, as appropriate. Welds that have been mitigated by the Mechanical Stress Improvement Process (MSIP™) may be categorized as Inspection Items D or E, as appropriate, provided the criteria in Appendix I of the Code Case have been met. For ISI frequencies, all other butt welds that rely on Alloy 82/182 for structural integrity must be categorized as Inspection Items A–1, A–2 or B until the NRC staff has reviewed the mitigation and authorized an alternative Code Case Inspection Item for the mitigated weld, or until an alternative Code Case Inspection Item is used based on conformance with an ASME mitigation Code Case endorsed in Regulatory Guide 1.147 with conditions, if applicable, and incorporated by reference in this section.

(*3*) Baseline examinations for welds in Table 1, Inspection Items A–1, A–2, and B, must be completed by the end of the next refueling outage after January 20, 2012. Previous examinations of these welds can be credited for baseline examinations if they were performed within the re-inspection period for the weld item in Table 1 using Section XI, Appendix VIII, requirements and met the Code required examination volume of essentially 100 percent. Other previous examinations that do not meet these requirements can be used to meet the baseline examination requirement, provided NRC approval of alternative inspection requirements in accordance with paragraphs (z)(1) or (z)(2) of this section is granted prior to the end of the next refueling outage after January 20, 2012.

(*4*) The axial examination coverage requirements of Paragraph—2500(c) may not be considered to be satisfied unless essentially 100 percent coverage is achieved.

(*5*) All hot-leg operating temperature welds in Inspection Items G, H, J, and K must be inspected each inspection interval. A 25 percent sample of Inspection Items G, H, J, and K cold-leg operating temperature welds must be inspected whenever the core barrel is removed (unless it has already been inspected within the past 10 years) or 20 years, whichever is less.

(*6*) For any mitigated weld whose volumetric examination detects growth of existing flaws in the required examination volume that exceed the previous IWB–3600 flaw evaluations or new flaws, a report summarizing the evaluation, along with inputs, methodologies, assumptions, and causes of the new flaw or flaw growth is to be provided to the NRC prior to the weld being placed in service other than modes 5 or 6.

(*7*) For Inspection Items G, H, J, and K, when applying the acceptance standards of ASME BPV Code, Section XI, IWB–3514, for planar flaws contained within the inlay or onlay, the thickness ''t'' in IWB–3514 is the thickness of the inlay or onlay. For planar flaws in the balance of the dissimilar metal weld examination volume, the thickness ''t'' in IWB–3514 is the combined thickness of the inlay or onlay and the dissimilar metal weld.

(*8*) Welds mitigated by optimized weld overlays in Inspection Items D and E are not permitted to be placed into a population to be examined on a sample basis and must be examined once each inspection interval.

(*9*) Replace the first two sentences of Extent and Frequency of Examination for Inspection Item D in Table 1 of Code Case N–770–1 with, ''Examine all welds no sooner than the third refueling outage and no later than 10 years following stress improvement application.'' Replace the first two sentences of Note (11)(b)(2) in Code Case N–770–1 with, ''The first examination following weld inlay, onlay, weld overlay, or stress improvement for Inspection Items D through K must be performed as specified.''

(*10*) General Note (b) to Figure 5(a) of Code Case N–770–1 pertaining to alternative examination volume for optimized weld overlays may not be applied unless NRC approval is authorized under paragraphs (z)(1) or (z)(2) of this section.

(h) *Protection and safety systems.* Protection systems of nuclear power reactors of all types must meet the requirements specified in this paragraph. Each combined license for a utilization facility is subject to the following conditions.

(1) [Reserved]

(2) *Protection systems.* For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, ''Criteria for Protection Systems for Nuclear Power Generating Stations,'' or in IEEE Std. 603–1991, ''Criteria for Safety Systems for Nuclear Power Generating Stations,'' and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

(3) *Safety systems.* Applications filed on or after May 13, 1999, for construction permits and operating licenses under this part, and for design approvals, design certifications, and combined licenses under Part 52 of this chapter, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

(i) through (y) [Reserved]

(z) *Alternatives to codes and standards requirements.* Alternatives to the requirements of paragraphs (b), (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:

(1) *Acceptable level of quality and safety.* The proposed alternative would provide an acceptable level of quality and safety; or

(2) *Hardship without a compensating increase in quality and safety.* Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Footnotes to § 50.55a:

[1] For inspections to be conducted once per interval, the inspections must be performed in accordance with the schedule in Section XI, paragraph IWB–2400, except for plants with inservice inspection programs based on a Section XI edition or addenda prior to the 1994 Addenda. For plants with inservice inspection programs based on a Section XI edition or addenda prior to the 1994 Addenda, the inspection must be performed in accordance with the schedule in Section XI, paragraph IWB–2400, of the 1994 Addenda.

[2–3] [Reserved]

[4] USAS and ASME Code addenda issued prior to the winter 1977 Addenda are considered to be ''in effect'' or ''effective'' 6 months after their date of issuance *and* after they are incorporated by reference in paragraph (a) of this section. Addenda to the ASME Code issued after the summer 1977 Addenda are considered to be ''in effect'' or ''effective'' after the date of publication of the addenda *and* after they are incorporated by reference in paragraph (a) of this section.

[5] For ASME Code editions and addenda issued prior to the winter 1977 Addenda, the Code edition and addenda applicable to the component is governed by the order or contract date for the component, not the contract date for the nuclear energy system. For the winter 1977 Addenda and subsequent editions and addenda the method for determining the applicable Code editions and addenda is contained in Paragraph NCA 1140 of Section III of the ASME Code.

[6–8] [Reserved]

[9] Guidance for quality group classifications of components that are to be included in the safety analysis reports pursuant to § 50.34(a) and § 50.34(b) may be found in Regulatory Guide 1.26, ''Quality Group Classifications and Standards for Water-, Steam-, and Radiological-Waste-Containing Components of Nuclear Power Plants,'' and in Section 3.2.2 of NUREG–0800, ''Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants.''

Dated at Rockville, Maryland, this 7th day of June 2013.

For the Nuclear Regulatory Commission.

Jennifer L. Uhle,

Deputy Director, Reactor Safety Programs, Office of Nuclear Reactor Regulation.

[FR Doc. 2013–15022 Filed 6–21–13; 8:45 am]

BILLING CODE 7590–01–P