reduce the commitments must be submitted to NRC and receive NRC approval before implementation, as follows:

(i) Changes to the Safety Analysis Report must be submitted for review as specified in § 50.4. Changes made to NRC-accepted quality assurance topical report descriptions must be submitted as specified in § 50.4.

(ii) The submittal of a change to the Safety Analysis Report quality assurance program description must include all pages affected by that change and must be accompanied by a forwarding letter identifying the change, the reason for the change, and the basis for concluding that the revised program incorporating the change continues to satisfy the criteria of appendix B of this part and the Safety Analysis Report quality assurance program description commitments previously accepted by the NRC (the letter need not provide the basis for changes that correct spelling, punctuation, or editorial items).

(iii) A copy of the forwarding letter identifying the changes must be maintained as a facility record for three years.

(iv) Changes to the quality assurance program description included or referenced in the Safety Analysis Report shall be regarded as accepted by the Commission upon receipt of a letter to this effect from the appropriate reviewing office of the Commission or 60 days after submittal to the Commission, whichever occurs first.

[21 FR 355, Jan. 19, 1956, as amended at 32 FR 4055, Mar. 15, 1967; 35 FR 11461, July 17, 1970; 35 FR 19661, Dec. 29, 1970; 36 FR 11424, June 12, 1971; 37 FR 6460, Mar. 30, 1972; 38 FR 1272, Jan. 11, 1973; 41 FR 16446, Apr. 19, 1976; 42 FR 43385, Aug. 29, 1977; 48 FR 1029, Jan. 10, 1983; 51 FR 40309, Nov. 6, 1986; 56 FR 36091, July 31, 1991; 59 FR 14087, Mar. 25, 1994]

§ 50.55a Codes and standards.

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

(a)(1) Structures, systems, and components must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

(2) Systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements of the ASME Boiler and Pressure Vessel Code specified in paragraphs (b), (c), (d), (e), (f), and (g) of this section. Protection systems of nuclear power reactors of all types must meet the requirements specified in paragraph (h) of this section.

(3) Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

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

(b) The ASME Boiler and Pressure Vessel Code, and the ASME Code for Operation and Maintenance of Nuclear Power Plants, which are referenced in the following paragraphs, were approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the FEDERAL REGISTER. Copies of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants may be purchased from the American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016. They are also available for inspection at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland 20852–2738. Copies are also available at the Office of the Federal Register, 800 N. Capitol Street, Suite 700, Washington, DC.

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include editions

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

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

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

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

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

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

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

(i) *Limitations on specific editions and addenda*. When applying the 1974 Edition, only the addenda through the Summer 1975 Addenda may be used. When applying the 1977 Edition, all of the addenda through the Summer 1978 Addenda must also be used. Addenda and editions subsequent to the Summer 1978 Addenda, that are incorporated by reference in paragraph (b)(2) of this section are not affected by these limitations.

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

(iii) *Steam generator tubing (modifies Article IWB–2000)*. If the technical specifications of a nuclear power plant include surveillance requirements for steam generators different than those in Article IWB–2000, the inservice inspection program for steam generator tubing is governed by the requirements in the technical specifications.

(iv) *Pressure-retaining welds in ASME Code Class 2 piping (applies to Tables IWC–2520 or IWC–2520–1, Category C–F)*. (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, must be examined. When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for these systems must be determined by the requirements of paragraph IWC–1220, Table IWC–2520 Category C–F and C–G, and paragraph IWC–2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda.

(B) For a nuclear power plant whose application for a construction permit was docketed prior to July 1, 1978, when applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of section XI of the ASME Code, the extent of examination for Code Class 2 pipe welds may be determined by the requirements of paragraph IWC–1220, Table IWC–2520 Category C–F and C–G and paragraph IWC–2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda of Section XI of the ASME Code or other requirements the Commission may adopt.

(v) Evaluation procedures and acceptance criteria for austenitic piping (applies to IWB–3640). When applying the Winter 1983 Addenda and Winter 1984 Addenda, the rules of paragraph IWB–3640 may be used for all applications permitted in that paragraph, except those associated with submerged arc welds (SAW) or shielded metal arc welds (SMAW). For SAW or SMAW, use paragraph IWB–3640, as modified by the Winter 1985 Addenda.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(A) For Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas.

For each inaccessible area identified, the licensee shall provide the following in the ISI Summary Report as required by IWA–6000:

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(xii) [Reserved]

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

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

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

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

(*2*) Leakage through a flange gasket;

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

(*4*) Degraded socket welds.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(*4*) Notches are not permitted.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

TABLE VIII–S7–1—MODIFIED

Flaw Locations and Orientations		
	Parallel to weld	Perpendicular to weld
Inner 15 percent	X	X
OD Surface	X	....
Subsurface	X	....

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

(3) The Code Edition, Addenda, and optional Code Cases[6] to be applied to components of the reactor coolant pressure boundary must be determined

See footnotes at end of section.

by the provisions of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but (i) the edition and addenda applied to a component must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the pressure vessel may be dated no earlier than the Summer 1972 Addenda of the 1971 edition, (iii) the ASME Code provisions applied to piping, pumps, and valves may be dated no earlier than the Winter 1972 Addenda of the 1971 edition, and (iv) ASME Code Cases[6] must have been determined suitable for use by the NRC.

(4) For a nuclear power plant whose construction permit was issued prior to May 14, 1984 the applicable Code Edition and Addenda for a component of the reactor coolant pressure boundary continue to be that Code Edition and Addenda that were required by Commission regulations for such component at the time of issuance of the construction permit.

(d) *Quality Group B components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group B[9] must meet the requirements for Class 2 Components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code Edition, Addenda, and optional Code Cases[6] to be applied to the systems and components identified in paragraph (d)(1) of this section must be determined by the rules of paragraph NCA–1140, Subsection NCA of Section III of the ASME Boiler Vessel and Pressure Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases[6] must have been determined suitable for use by the NRC.

(e) *Quality Group C components.* (1) For a nuclear power plant whose application for a construction permit is docketed after May 14, 1984 components classified Quality Group C[9] must meet the requirements for Class 3 components in Section III of the ASME Boiler and Pressure Vessel Code.

(2) The Code Edition, Addenda, and optional Code Cases[6] to be applied to the systems and components identified in paragraph (e)(1) of this section must be determined by the rules of paragraph NCA–1140, subsection NCA of Section III of the ASME Boiler and Pressure Vessel Code, but (i) the edition and addenda must be those which are incorporated by reference in paragraph (b)(1) of this section, (ii) the ASME Code provisions applied to the systems and components may be dated no earlier than the 1980 Edition, and (iii) the ASME Code Cases[6] must have been determined suitable for use by the NRC.

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

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

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, pumps and valves which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice tests for operational readiness set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect 6 months prior to the date of issuance of

the construction permit. The pumps and valves may meet the inservice test requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

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

(i)–(ii) [Reserved]

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

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

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

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

(v) All pumps and valves may meet the test requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

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

(i) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice tests to verify operational readiness of pumps and valves, whose function is required for safety, conducted during successive 120-month intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) [Reserved]

(iv) Inservice tests of pumps and valves may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

(5)(i) The inservice test program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (f)(4) of this section.

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

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determination.

(iv) Where a pump or valve test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice test program as permitted by paragraph (f)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the test is determined to be impractical.

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

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

(g) *Inservice inspection requirements.* Requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in § 50.55a(f).

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

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions of section XI of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect six months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed in paragraph (b) of this section.

(3) For a boiling or pressurized water-cooled nuclear power facility whose

construction permit was issued on or after July 1, 1974:

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

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examination requirements set forth in section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular component.

(iii)–(iv) [Reserved]

(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which are incorporated by reference in paragraph (b) of this section, subject to the limitations and modifications listed therein.

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

(i) Inservice examinations of components and system pressure tests conducted during the initial 120-month inspection interval must comply with the requirements in the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section on the date 12 months prior to the date of issuance of the operating license, subject to the limitations and modifications listed in paragraph (b) of this section.

(ii) Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.

(iii) Licensees may, but are not required to, perform the surface examinations of High Pressure Safety Injection Systems specified in Table IWB–2500–1, Examination Category B–J, Item Numbers B9.20, B9.21, and B9.22.

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

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

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

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

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

(5)(i) The inservice inspection program for a boiling or pressurized water-cooled nuclear power facility must be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

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

(iii) If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in §50.4, information to support the determinations.

(iv) Where an examination requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination must be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination is determined to be impractical.

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

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

(A) Augmented examination of reactor vessel.

(*1*) All previously granted reliefs under §50.55a to licensees for the extent of volumetric examination of reactor vessel shell welds specified in Item B1.10 of Examination Category B–A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB–2500–1 of subsection IWB in applicable edition and addenda of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, during the inservice inspection interval in effect on September 8, 1992 are hereby revoked, subject to the specific modification in §50.55a(g)(6)(ii)(A)*(3)(iv)* for licensees that defer the augmented examination in accordance with §50.55a(g)(6)(ii)(A)*(3)*.

(*2*) All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell

See footnotes at end of section.

welds specified in Item B1.10 of Examination Category B–A, ''Pressure Retaining Welds in Reactor Vessel,'' in Table IWB–2500–1 of subsection IWB of the 1989 Edition of section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in § 50.55a(g)(6)(ii)(A) (*3*) and (*4*). The augmented examination, when not deferred in accordance with the provisions of § 50.55a(g)(6)(ii)(A)(*3*), shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inservice inspection interval in effect on September 8, 1992, and may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992. For the purpose of this augmented examination, ''essentially 100% as used in Table IWB–2500–1 means more than 90 percent of the examination volume of each weld, where the reduction in coverage is due to interference by another component, or part geometry.

(*3*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may defer the augmented reactor vessel examination specified in § 50.55a(g)(6)(ii)(A)(*2*) to the first period of the next inspection interval under the following conditions:

(*i*) The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on September 8, 1992.

(*ii*) The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed.

(*iii*) If the deferred augmented examination is used as a substitute for the normally scheduled reactor vessel shell weld examination, subsequent reactor vessel shell weld examinations must be performed during the first period of successive inspection intervals.

(*iv*) Licensees that defer the augmented examination, as permitted herein, may retain all previously granted reliefs that otherwise would be revoked by § 50.55a(g)(6)(ii)(A)(*1*) for the inservice inspection interval in effect on September 8, 1992.

(*v*) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on September 8, 1992 may extend that interval in accordance with the provisions of section XI (1989 Edition) IWA–2430(d) for the purpose of implementing the augmented examination during that interval.

(*vi*) The deferred augmented examination shall be performed in accordance with the related procedures specified in the section XI edition and addenda applicable to the inspection interval in which the augmented examination is performed.

(*4*) The requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100 percent of the reactor vessel shell welds specified in § 50.55a(g)(6)(ii)(A)(*2*) that has been completed, or is scheduled for implementation with a written commitment, or is required by § 50.55a(g)(4)(i), during the inservice inspection interval in effect on September 8, 1992.

(*5*) Licensees that make a determination that they are unable to completely satisfy the requirements for the augmented reactor vessel shell weld examination specified in § 50.55a(g)(6)(ii)(A) shall submit information to the Commission to support the determination and shall propose an alternative to the examination requirements that would provide an acceptable level of quality and safety. The licensee may use the proposed alternative when authorized by the Director of the Office of Nuclear Reactor Regulation.

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

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

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

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

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

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

(h) Protection and safety systems. (1) IEEE Std. 603–1991, including the correction sheet dated January 30, 1995, which is referenced in paragraphs (h)(2) and (h)(3) of this section, is approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of IEEE Std. 603–1991 may be purchased from the Institute of Electrical and Electronics Engineers Service Center, 445 Hoes Lane, Piscataway, NJ 08855. The standard is also available for inspection at the NRC Library, 11545 Rockville Pike, Rockville, Md; and at the Office of the Federal Register, 800 North Capitol Street, NW., Suite 700, Washington, DC IEEE Std. 279, which is referenced in paragraph (h)(2) of this section, was approved for incorporation by reference by the Director of the Office of the Federal Register in accordance with 5 U.S.C. 552(a) and 1 CFR Part 51. Copies of IEEE Std. 279 are also available as indicated for IEEE Std. 603–1991.

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

(3) Safety systems. Applications filed on or after May 13, 1999 for preliminary and final design approvals (10 CFR Part 52, Appendix O), design certifications, and construction permits, operating licenses and combined licenses that do not reference a final design approval or design certification, must meet the requirements for safety systems in IEEE Std. 603–1991 and the correction sheet dated January 30, 1995.

Footnotes to § 50.55a:

[1–3] [Reserved]

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

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

[6] ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, ''Design and Code Case Acceptability—ASME Section III Division 1,'' NRC Regulatory Guide 1.85, ''Materials Code Case Acceptability—ASME Section III Division 1,'' and NRC Regulatory Guide 1.147, ''Inservice Inspection Code Case Acceptability—ASME Section XI Division 1.'' The use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to § 50.55a(a)(3).

[7–8] [Reserved]

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

[36 FR 11424, June 12, 1971]

EDITORIAL NOTE: For FEDERAL REGISTER citations affecting § 50.55a, see the List of CFR Sections Affected, which appears in the Finding Aids section of the printed volume and on GPO Access.

§ 50.56 Conversion of construction permit to license; or amendment of license.

Upon completion of the construction or alteration of a facility, in compliance with the terms and conditions of the construction permit and subject to any necessary testing of the facility for health or safety purposes, the Commission will, in the absence of good cause shown to the contrary issue a license of the class for which the construction permit was issued or an appropriate amendment of the license, as the case may be.

[21 FR 355, Jan. 19, 1956, as amended at 35 FR 11461, July 17, 1970]

§ 50.57 Issuance of operating license. [1]

(a)Pursuant to § 50.56, an operating license may be issued by the Commission, up to the full term authorized by § 50.51, upon finding that:

(1) Construction of the facility has been substantially completed, in conformity with the construction permit and the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and

(3) There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations in this chapter; and

(4) The applicant is technically and financially qualified to engage in the activities authorized by the operating license in accordance with the regulations in this chapter. However, no finding of financial qualification is necessary for an electric utility applicant for an operating license for a utilization facility of the type described in § 50.21(b) or § 50.22.

(5) The applicable provisions of part 140 of this chapter have been satisfied; and

(6) The issuance of the license will not be inimical to the common defense and security or to the health and safety of the public.

[1] The Commission may issue a provisional operating license pursuant to the regulations in this part in effect on March 30, 1970, for any facility for which a notice of hearing on an application for a provisional operating license or a notice of proposed issuance of a provisional operating license has been published on or before that date.