

2017 Final Rule (10 CFR 50.55a)

American Society of Mechanical Engineers Codes and Code Cases

Analysis of Public Comments

[NRC-2011-0088; RIN 3150-AI97]

The U.S. Nuclear Regulatory Commission (NRC) is amending Section 50.55a of Title 10 of the *Code of Federal Regulations* (10 CFR), to incorporate the following codes and standards by reference (with conditions on their use):

- the 2009 Addenda, 2010 Edition, 2011 Addenda, and 2013 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III, Division 1 and Section XI, Division 1
- the 2009 Edition, 2011 Addenda, and 2012 Edition of “Division 1: OM Code: Section IST” of the ASME Operation and Maintenance of Nuclear Power Plants (OM Code)¹
- ASME BPV Code Case N-729-4, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1,” ASME approval date: June 22, 2012
- ASME BPV Code Case N-770-2, “Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1,” ASME approval date: June 9, 2011
- ASME BPV Code Case N-824, “Ultrasonic Examination of Cast Austenitic Piping Welds From the Outside Surface Section XI, Division 1,” ASME approval date: October 16, 2012
- ASME BPV Code Case N-513-3, “Evaluation Criteria for Temporary Acceptance of Flaws in Moderate Energy Class 2 or 3 Piping Section XI, Division 1,” Mandatory Appendix I, “Relations for F_m , F_b , and F for Through-Wall Flaws,” ASME approval date: January 26, 2009
- ASME BPV Code Case N-852, “Application of the ASME NPT Stamp, Section III, Division 1; Section III, Division 2; Section III, Division 3; Section III, Division 5,” ASME approval date: February 9, 2015
- ASME OM Code Case OMN-20, “Inservice Test Frequency”
- ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications,” including several editions and addenda to ASME NQA-1 from previous years with

¹ ASME has applied different titles to the ASME Code for Operation and Maintenance of Nuclear Power Plants in previous editions and addenda. The NRC uses the term “OM Code” to identify the ASME document in this analysis of public comments.

slightly varying titles, as identified in 10 CFR 50.55a(a)(1)(v); more specifically, incorporating by reference the 1983 Edition through the 1994 Edition, the 2008 Edition, and the 2009-1a Addenda to the 2008 Edition of ASME NQA-1

The NRC published a proposed rule for public comment in the *Federal Register* (FR) on September 18, 2015 (80 FR 56820). The NRC considered the comments received on the proposed rule in developing the final rule. Public comment submissions are available online in the NRC Library at <http://www.nrc.gov/reading-rm/adams.html>. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which supplies text and image files of the NRC's public documents. If you do not have access to ADAMS, or if there are problems in accessing the documents located in ADAMS, contact the NRC's Public Document Room reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. In addition, public comments and supporting materials related to this final rule can be found at <http://www.regulations.gov> by searching for Docket ID NRC-2011-0088.

The NRC received comments from the individuals and groups shown in Table 1 (listed in order of receipt).

TABLE 1 COMMENTS FROM INDIVIDUALS AND GROUPS

Submission ID	Commenter	ADAMS Accession Number
1	Private Citizen, Edward Cavey	ML15321A419
2	Private Citizen, Dale Matthews	ML15321A418
3	Private Citizen, Ron Clow	ML15328A487
4	American Society of Mechanical Engineers (ASME)	ML15328A486
5	Iddeal Solutions, LLC	ML15334A379
6	Electric Power Research Institute (EPRI)	ML15335A151
7	Private Citizen, William Taylor	ML15335A523
8	ASME	ML15335A522
9	Private Citizen, Dan Nowakowski	ML15335A524
10	Wolf Creek Nuclear Operating Corporation (WCNOC)	ML15335A561
11	Northern States Power Company – Minnesota	ML15338A106
12	FirstEnergy Nuclear Operating Company	ML15338A105
13	PSEG Nuclear	ML15338A104
14	Dominion Resources Services, Inc.	ML15338A111
15	Private Citizen, Terence Chan	ML15338A110
16	Nuclear Energy Institute	ML15338A107
17	Electric Power Research Institute	ML15338A241
18	Duke Energy	ML15338A240
19	Private Citizen, William Taylor	ML15338A243
20	Dominion Engineering, Inc.	ML15338A242

Submission ID	Commenter	ADAMS Accession Number
21	Tennessee Valley Authority	ML15338A251
22	Southern Nuclear Operating Company	ML15352A092
23	Prairie Island Nuclear Plant	ML15355A158
24	Inservice Test Owners Group	ML16015A352
25	Exelon Generation Company	ML16019A286
26	Electric Power Research Institute	ML16027A236
27	Electric Power Research Institute	ML16027A235

The NRC received five letters after the close of the comment period (Submission IDs 22, 23, 24, 26, and 27) but before the NRC staff had begun to evaluate the other comments. Therefore, the staff considered it practical to consider these late comments as well.

In this document, the NRC has summarized each comment and placed it into one of several categories shown below. Many of the comments give essentially the same position, argument, rationale, or basis. In such cases, the NRC binned similar comments into a single comment summary and responded to the comment summary. At the end of each comment, the NRC refers to the specific public comment letter containing that comment in the form [XX-YY], where XX represents the Submission ID in Table 1 of this document and YY represents individual, sequential comments as noted in the margin of the annotated copy of the public comments (ADAMS Accession No. ML15331A021).

The NRC received one comment submission (from Northern States Power Company – Minnesota, Submission ID 11) that endorsed the comments made by another commenter, ASME. Therefore, the responses to ASME also respond to the comments of Northern States Power Company – Minnesota.

After the close of the public comment period, the NRC held a public meeting on March 2, 2016, to discuss the proposed rule and to answer questions about specific provisions. The NRC considered the feedback from this public meeting during the development of the final rule. The public meeting summary is available in ADAMS under Accession No. ML16069A408.

Public Comment Categories

- I. Responses to Specific Requests for Comments
- II. Documents Approved for Incorporation by Reference
 - a. 10 CFR 50.55a(a)(1)(i)
 - b. 10 CFR 50.55a(a)(1)(ii)
 - c. 10 CFR 50.55a(a)(1)(iii)
- III. ASME BPV Code, Section III
 - a. 10 CFR 50.55a(b)(1)(viii)
- IV. ASME BPV Code, Section XI
 - a. 10 CFR 50.55a(b)(2)
 - b. 10 CFR 50.55a(b)(2)(viii)(H) and (I)

- c. 10 CFR 50.55a(b)(2)(ix)(H)
- d. 10 CFR 50.55a(b)(2)(xii)
- e. 10 CFR 50.55a(b)(2)(xv)
- f. 10 CFR 50.55a(b)(2)(xvi)
- g. 10 CFR 50.55a(b)(2)(xvii)
- h. 10 CFR 50.55a(b)(2)(xviii)
- i. 10 CFR 50.55a(b)(2)(xx)
- j. 10 CFR 50.55a(b)(2)(xxi)
- k. 10 CFR 50.55a(b)(2)(xxiii)
- l. 10 CFR 50.55a(b)(2)(xxv)
- m. 10 CFR 50.55a(b)(2)(xxvi)
- n. 10 CFR 50.55a(b)(2)(xxx)
- o. 10 CFR 50.55a(b)(2)(xxxi)
- p. 10 CFR 50.55a(b)(2)(xxxii)
- q. 10 CFR 50.55a(b)(2)(xxxiii)
- r. 10 CFR 50.55a(b)(2)(xxxiv)
- s. 10 CFR 50.55a(b)(2)(xxxvii)

V. ASME OM Code

- a. 10 CFR 50.55a(b)(3)(ii)
- b. 10 CFR 50.55a(b)(3)(ii)(A)
- c. 10 CFR 50.55a(b)(3)(ii)(B)
- d. 10 CFR 50.55a(b)(3)(ii)(C)
- e. 10 CFR 50.55a(b)(3)(ii)(D)
- f. 10 CFR 50.55a(b)(3)(iii)(A)
- g. 10 CFR 50.55a(b)(3)(iii)(B)
- h. 10 CFR 50.55a(b)(3)(iii)(C)
- i. 10 CFR 50.55a(b)(3)(iii)(D)
- j. 10 CFR 50.55a(b)(3)(iv)
- k. 10 CFR 50.55a(b)(3)(vii)
- l. 10 CFR 50.55a(b)(3)(viii)
- m. 10 CFR 50.55a(b)(3)(x)
- n. 10 CFR 50.55a(b)(3)(xi)

VI. Inservice Testing

- a. 10 CFR 50.55a(f)
- b. 10 CFR 50.55a(f)(4)

VII. Inservice Inspection

- a. 10 CFR 50.55a(g)(1)
- b. 10 CFR 50.55a(g)(2)
- c. 10 CFR 50.55a(g)(3)
- d. 10 CFR 50.55a(g)(4)
- e. 10 CFR 50.55a(g)(6)(ii)

VIII. Other Comments

I. Responses to Specific Requests for Comments

In the notice on the proposed rule published in the *Federal Register* on September 18, 2015, the NRC asked three specific questions concerning process improvements for 10 CFR 50.55a

rulemakings. The NRC's goal is to make the process of incorporating by reference ASME BPV and OM Code editions and addenda into 10 CFR 50.55a more predictable and consistent.

NRC Question 1. The NRC is considering removing the references to versions of NQA-1 older than the 1994 Edition in § 50.55a(b)(1)(iv), § 50.55a(b)(2)(x), and § 50.55a(b)(3)(i). The NRC requests public comment on whether any applicant or licensee is committed to, and is using, a version of NQA-1 older than the 1994 Edition, and if so, what version the applicant or licensee is using.

Comment: ASME supports the incorporation by reference of NQA-1. [8-4]

NRC Response: No response is necessary.

NRC Question 2. Should ASME BPV Code Case N-824, as conditioned, be mandatory? What are the possible advantages and disadvantages of making N-824, as conditioned, mandatory?

Comment: If the use of ASME BPV Code Case N-824 is to be made mandatory, an implementation schedule should be established to allow licensees to prepare to implement the Code Case. [5-10]

NRC Response: The NRC agrees that allowing a delay to implement revised inspection plans would be needed if the agency were to mandate ASME BPV Code Case N-824. The NRC has determined that ASME BPV Code Case N-824 should be approved for use but is not making it mandatory at this time. The NRC recognizes that the 2015 Edition of ASME BPV Code, Section XI, incorporates ASME BPV Code Case N-824 into Appendix III and that the industry is working to make these provisions mandatory.

The NRC made no change to the final rule as a result of this comment.

Comment: The conditions placed on the use of ASME BPV Code Case N-824 will limit the use of the Code Case.

The condition to use encoded examinations in 10 CFR 50.55a(b)(2)(xxxvii)(A) would limit the use of the Code Case, as field conditions do not always allow for the collection of encoded ultrasonic data for offline analysis, due to permanent obstructions (walls, floors, whip restraints, branch connections, etc.).

Section 50.55a(b)(2)(xxxvii)(B) would require the use of phased array search units. While it is recognized that a great deal of the recent industry research for cast austenitic stainless steel applications has utilized ultrasonic phased array transducers, conventional search units have also been demonstrated to be effective, especially when the thickness and internal geometry of the component to be inspected are well understood.

The condition described in 10 CFR 50.55a(b)(2)(xxxvii)(C) would require that a center frequency of 500 kilohertz (kHz) to 1 megahertz (MHz) be used on piping less than or equal to 1.6 inches (41 mm) in thickness, instead of the requirements of ASME BPV Code Case N-824, Paragraph 1(c)(1)(-c)(-1). Pacific Northwest National Laboratories (PNNL) has published a report that shows effective ultrasonic results being obtained using from 800 kHz all the way up to 2.0 MHz probe frequencies (see "An Evaluation of Ultrasonic Phased Array Testing for Cast Austenitic Stainless Steel Pressurizer Surge Line Piping Welds" (NUREG/CR-7122, PNNL-19497)).

Finally, the condition given in 10 CFR 50.55a(b)(2)(xxxvii)(E) would require the use of a phased array search unit, which produces angles from 30 to 70 degrees, with a maximum increment of 5 degrees, instead of the requirements of Paragraph 1(c)(1)(-d). As previously discussed, only a small range of ultrasonic angles have been shown to be effective for cast austenitic stainless steel applications based on recent industry research activities. As such, the available research would support the use of conventional ultrasonic search units having fixed inspection angles. [6-6]

NRC Response: The NRC partially agrees with the comment on the conditions for the use of ASME BPV Code Case N-824. As a result, the NRC has modified some of the conditions in the proposed rule to align with those in NUREG/CR-6933, "Assessment of Crack Detection in Heavy-Walled Cast Stainless Steel Piping Welds Using Advanced Low-Frequency Ultrasonic Methods," issued March 2007, and NUREG/CR-7122, "An Evaluation of Ultrasonic Phased Array Testing for Cast Austenitic Stainless Steel Pressurizer Surge Line Piping Welds," issued March 2012. The NRC's action, if any, for each condition is described below.

The NRC disagrees with the comment asking the agency to remove the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(A) requiring encoding. The current technical basis for the examination of cast stainless steel components points to the need to spatially encode the examinations. While requiring spatial encoding makes the examination more complex, the NRC currently does not have confidence in non-encoded examinations of cast stainless steel materials. The NRC will reconsider this condition in a future rulemaking if non-encoded inspection techniques can be demonstrated to reliably detect flaws in cast stainless steel components.

The NRC disagrees with the comment asking the agency to remove the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(B) to use phased array search units. The current technical basis for the examination of cast stainless steel components used phased array search units. The NRC will reconsider this condition in a future rulemaking if conventional search units can be demonstrated to reliably detect flaws in cast stainless steel components.

The NRC agrees with the comment asking the agency to remove the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(C) restricting the allowed frequencies to below 1 MHz. The NRC recognizes that NUREG/CR-7122 used higher frequency phased array probes to successfully find flaws in thinner cast stainless steel piping welds. Therefore, the NRC has removed the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(C) from the final rule and has relocated the proposed conditions in paragraphs (D) and (E) to paragraphs (C) and (D), respectively.

The NRC partially agrees with the comment requesting that the agency modify the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(E) to more closely match the technical basis described in NUREG/CR-6933. Although the higher angles can be useful for measuring the depth of flaws, they are not needed for detection. Therefore, the NRC has revised the condition in the final rule to read, "Instead of Paragraph 1(c)(1)(-d), the phased array search unit must produce angles including, but not limited to, 30 to 55 degrees with a maximum increment of 5 degrees."

Comment: As proposed, 10 CFR 50.55a(b)(2)(xxxvii)(D) would require the use of a center frequency of 500 kHz, instead of the requirements of ASME BPV Code Case N-824, Paragraph 1(c)(1)(-c)(-2), when examining piping greater than 1.6 inches (41 mm) in thickness. A tolerance (e.g., +/- 20% to +/- 30%) on the center frequency of 500 kHz should be included in this condition. [8-25; 14-18]

NRC Response: The NRC agrees with these comments. The intent of the condition is to limit the frequency to the lower range of the allowed frequencies in the Code Case, not set an absolute value for the frequency. The NRC recognizes that without a tolerance range, licensees using a search unit with a center frequency of 499 kHz or 501 kHz would not be in full compliance. Therefore, the NRC has revised the condition in the final rule to read, "Instead of Paragraph 1(c)(1)(-c)(-2), licensees shall use a phased array search unit with a center frequency of 500 kHz with a tolerance of +/- 20 percent."

Comment: The conditions imposed in 10 CFR 50.55a(b)(2)(xxvii)(A), (B), (C), and (E) should be eliminated. [14-14]

NRC Response: The NRC disagrees with this comment. The conditions were added to align the Code Case with the technical basis found in NUREG/CR-6933 and NUREG/CR-7122. Although the conditions have been modified in the final rule for technical reasons, they are still required.

Comment: The condition to use encoded examinations in 10 CFR 50.55a(b)(2)(xxvii)(A) would limit the use of the Code Case, as field conditions do not always allow for the collection of encoded ultrasonic data for offline analysis, due to permanent obstructions (walls, floors, whip restraints, branch connections, etc.). [14-15]

NRC Response: The NRC disagrees with the comment asking the agency to remove the proposed condition in 10 CFR 50.55a(b)(2)(xxvii)(A) requiring encoding. The current technical basis for the examination of cast stainless steel components points to the need to spatially encode the examinations. While requiring spatial encoding makes the examination more complex, the NRC currently does not have confidence in non-encoded examinations of cast stainless steel materials. The NRC will reconsider this condition in a future rulemaking if non-encoded inspection techniques can be demonstrated to reliably detect flaws in cast stainless steel components.

The NRC made no change to the final rule as a result of this comment.

Comment: Section 50.55a(b)(2)(xxvii)(B) would require the use of phased array search units. While it is recognized that a great deal of the recent industry research for cast austenitic stainless steel applications has utilized ultrasonic phased array transducers, conventional search units have also been demonstrated to be effective, especially when the thickness and internal geometry of the component to be inspected are well understood. [14-16]

NRC Response: The NRC disagrees with the comment asking the agency to remove the proposed condition to use phased array search units in 10 CFR 50.55a(b)(2)(xxvii)(B). The current technical basis for the examination of cast stainless steel components used phased array search units. The NRC will reconsider this condition in a future rulemaking if conventional search units can be demonstrated to reliably detect flaws in cast stainless steel components.

The NRC made no change to the final rule as a result of this comment.

Comment: The condition described in 10 CFR 50.55a(b)(2)(xxvii)(C) would require that a center frequency of 500 kHz to 1 MHz be used on piping less than or equal to 1.6 inches (41 mm) in thickness, instead of the requirements of ASME BPV Code Case N-824, Paragraph 1(c)(1)(-c)(-1). PNNL has published a report which shows effective ultrasonic results being obtained using from 800 kHz all the way up to 2.0 MHz probe frequencies (see "An

Evaluation of Ultrasonic Phased Array Testing for Cast Austenitic Stainless Steel Pressurizer Surge Line Piping Welds” (NUREG/CR-7122, PNNL-19497)). [14-17]

NRC Response: The NRC agrees with the comment asking the agency to remove the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(C) restricting the allowed frequencies to below 1 MHz. The NRC recognizes that NUREG/CR-7122 used higher frequency phased array probes to successfully find flaws in thinner cast stainless steel piping welds.

The NRC has removed the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(C) from the final rule and has relocated the proposed conditions in paragraphs (D) and (E) to paragraphs (C) and (D), respectively.

Comment: The condition given in 10 CFR 50.55a(b)(2)(xxxvii)(E) would require the use of a phased array search unit which produces angles from 30 to 70 degrees, with a maximum increment of 5 degrees, instead of the requirements of Paragraph 1(c)(1)(-d). As previously discussed, only a small range of ultrasonic angles have been shown to be effective for cast austenitic stainless steel applications based on recent industry research activities. As such, the available research would support the use of conventional ultrasonic search units having fixed inspection angles. [14-19]

NRC Response: The NRC partially agrees with the comment asking that the proposed condition in 10 CFR 50.55a(b)(2)(xxxvii)(E) be modified to more closely match the technical basis described in NUREG/CR-6933. While the higher angles can be useful for measuring the depth of flaws, they are not needed for detection. Therefore, the NRC has revised the condition in the final rule to read, “Instead of Paragraph 1(c)(1)(-d), the phased array search unit must produce angles including, but not limited to, 30 to 55 degrees with a maximum increment of 5 degrees.”

Comment: The NRC should not require the use of ASME BPV Code Case N-824, as this would force inspectors to use outdated technology. Also, the center frequency limits do not account for tolerances in center frequency. As an example, a 500 kHz nominal frequency search unit could have a center frequency of 498 kHz. This condition clearly meets the required performance needed but would not meet literal wording of the proposed rule. Thus, 10 CFR 50.55a(b)(2)(xxxvii) should be removed from the final rule. [25-12]

NRC Response: The NRC partially agrees with this comment. The NRC has determined that ASME BPV Code Case N-824 should be approved for use, but that it does not need to be made mandatory at this time. The NRC recognizes that the 2015 Edition of ASME BPV Code, Section XI, incorporates ASME BPV Code Case N-824 into Appendix III and that the industry is working to make these provisions mandatory. The NRC asked for public comments about whether the provisions of ASME BPV Code Case N-824 should be made immediately mandatory. Although the NRC will not delete 10 CFR 50.55a(b)(2)(xxxvii) in the final rule, the use of ASME BPV Code Case N-824 will be allowed but not made mandatory.

Additionally, as discussed above in the responses to Comments 6-6 and 8-25, the NRC has revised the condition identified in the comment to read, “Instead of Paragraph 1(c)(1)(-c)(-2), licensees shall use a phased array search unit with a center frequency of 500 kHz with a tolerance of +/- 20 percent.”

II. Documents Approved for Incorporation by Reference

a. 10 CFR 50.55a(a)(1)(i)

Comment: The NRC proposes to clarify that Section III Nonmandatory Appendices are not incorporated by reference. This language was originally added in a final rule published on June 21, 2011 (76 FR 36232); however, it was omitted from the final rule published on November 5, 2014 (79 FR 65776). The NRC is correcting the omission by inserting “(excluding Non-mandatory Appendices)” in 10 CFR 50.55a(a)(1)(i).

- (1) The proposed change is unclear as to the impact to the industry. The proposed change, as written, implies that the NRC is not approving the Section III Nonmandatory Appendices for use. If the NRC chooses to exclude Section III Nonmandatory Appendices, the rule change should clarify how they are to be used.*
- (2) Prior to the proposed change, it appears that the Section III Nonmandatory Appendices were approved because they were not specifically excluded from the NRC’s incorporation of Section III by reference. Therefore, it seems that the proposed change to 10 CFR 50.55a(a)(1)(i) is retroactively removing the Nonmandatory Appendices from 10 CFR 50.55a. Licensees have, and in good faith, used the Section III Nonmandatory Appendices because their use was not prohibited in 10 CFR 50.55a, and Section III incorporation into 10 CFR 50.55a did not exclude them. The NRC should consider making the change effective only to edition/addenda that are added during and after the proposed change, or provide a basis and impact evaluation for rescinding the NRC’s approval of previously approved Section III Nonmandatory Appendices. [5-1]*

NRC Response: The proposed change is a clarification that has no impact on industry. The NRC has never approved the use of the Nonmandatory Appendices of ASME BPV Code, Section III. This language was originally added in a final rule published on June 21, 2011 (76 FR 36232); however, it was inadvertently omitted from the final rule published on November 5, 2014 (79 FR 65776). The NRC is correcting the omission by inserting the parenthetical clause “(excluding Nonmandatory Appendices)” in 10 CFR 50.55a(a)(1)(i). The Section III Nonmandatory Appendices were not incorporated by reference in 10 CFR 50.55a and, therefore, were never an NRC requirement. This change corrects the current error in 10 CFR 50.55a.

The NRC made no change to the final rule as a result of this comment.

b. 10 CFR 50.55a(a)(1)(ii)

Comment: The proposed revision to 10 CFR 50.55a(a)(1)(ii) implies that all nonmandatory appendices must be referenced now since there are only three instances of the word “nonmandatory” and zero instances of the word “non-mandatory” in the current 10 CFR 50.55a—these are things such as flaw evaluation, evaluating coverage, fracture toughness criteria, qualification of personnel written practice requirements, surface conditioning, analysis of flaws, forms, etc. These are not things that require prior approval currently, but they are not referenced in 10 CFR 50.55a, which references Section XI as a whole, and calls out exclusions. [19-1]

NRC Response: The NRC understands this comment to say that because the NRC proposed to exclude Nonmandatory Appendix U from the incorporation by reference this implied that all Nonmandatory Appendices must be referenced now. The NRC disagrees with this comment.

The NRC has reviewed all Section XI Nonmandatory Appendices and included them in the incorporation by reference. A licensee may voluntarily use these appendices subject to the conditions placed on them in the regulations. At the time of the proposed rule, the NRC was proposing to exclude Appendix U from the incorporation by reference. This would have meant a licensee could not use the methods in the appendix without submitting an alternative to the NRC in accordance with 10 CFR 50.55a(z) and obtaining NRC approval.

As discussed in the response to Comment 5-2, the NRC has changed the final rule to remove the exclusion of Nonmandatory Appendix U from the incorporation by reference.

c. 10 CFR 50.55a(a)(1)(iii)

Comment: The NRC should accept the NPT Code Symbol Stamp having the NPT letters arranged horizontally as an acceptable NPT Stamp to certify Code compliance for fabricated items that have already been stamped prior to receiving a replacement NPT Stamp from ASME and that the NRC include acceptance of ASME BPV Code Case N-852 in the final rule for this purpose. Within the context of the ASME Code rules, the NPT Code Symbol Stamp having the NPT letters arranged horizontally, although differing slightly in appearance from the NPT Code Symbol Stamp as illustrated in Section III, Table NCA-8100-1, of the ASME BPV Code, 2010 Edition and earlier editions and addenda, serves the same purpose of certifying Code compliance by the ASME NPT Certificate Holder with confirmation by the Authorized Nuclear Inspector and provides the same level of quality assurance.

On or after January 1, 2016, ASME will no longer authorize use of the NPT Code Symbol Stamp having the NPT letters arranged horizontally. Accordingly, on or after January 1, 2016, fabricated items will only be stamped with the NPT Code Symbol Stamp as illustrated in Section III, Table NCA-8100-1, of the ASME BPV Code, 2010 Edition and earlier editions and addenda. [4-1; 22-3]

NRC Response: The NRC agrees in general with this comment in which ASME asserts that the ASME NPT Code Symbol Stamp with the letters arranged horizontally is equivalent to the "N over PT" ASME NPT Code Symbol Stamp. Therefore, using either Code Symbol Stamp serves the same purpose of certifying code compliance by the ASME Certificate Holder with confirmation by the Authorized Nuclear Inspector and gives the same level of quality assurance. The NRC notes that the same administrative and technical requirements in the ASME BPV Code still apply whether an ASME NPT Code Symbol Stamp with the letters arranged horizontally or an "N over PT" ASME NPT Code Symbol Stamp is applied. However, because the NPT Code Symbol Stamp having the NPT letters arranged horizontally will only be applied to fabricated components from January 1, 2005, through December 31, 2015, the time period for applying this NPT Code Symbol Stamp to the component should be limited to these dates to prevent inadvertent fraudulent material. Therefore, the NRC agrees that ASME BPV Code Case N-852 is acceptable for the service life of a component that had the NPT Code Symbol stamp applied during the time period January 1, 2005, through December 31, 2015.

The NRC has added 10 CFR 50.55a(b)(1)(ix) to the final rule to indicate that licensees may use the NPT Code Symbol Stamp with the letters arranged horizontally as specified in ASME BPV Code Case N-852 for the service life of a component that had the NPT Code Symbol Stamp applied during the time period January 1, 2005, through December 31, 2015.

Comment: Section 50.55a(a)(1)(iii)(C) should be revised to incorporate by reference ASME BPV Code Case N-770-3 or N-770-4, in lieu of ASME BPV Code Case N-770-2. [8-2; 14-2]

NRC Response: The NRC disagrees with these comments. Neither ASME BPV Code Case N-770-3 nor Code Case N-770-4 were addressed in the proposed rule. Adoption of either version of the Code Case would require a significant delay in the completion of this rule, in order to provide an opportunity for comment on the incorporation by reference and approval for use of these versions of the Code Case. The NRC will consider incorporating by reference the latest approved version of ASME BPV Code Case N-770 as part of the future rulemaking addressing the 2015 Edition of the ASME BPV Code, Section XI.

The NRC made no change to the final rule as a result of these comments.

Comment: The use of ASME BPV Code Case N-824 should be allowed without the conditions of 10 CFR 50.55a(b)(2)(xxvii) (A), (B), (C), and (E). The proposed conditions reduce the likelihood of the use of this Code Case. [14-3]

NRC Response: The NRC agrees with this comment that these conditions would reduce the likelihood that the Code Case will be used. However, the NRC has retained the proposed conditions in 10 CFR 50.55a(b)(2)(xxvii)(A), (B), (D), and (E) because they are required to align the use of ASME BPV Code Case N-824 with the technical basis described in NUREG/CR-6933 and NUREG/CR-7122. The NRC has removed the proposed condition in 10 CFR 50.55a(b)(2)(xxvii)(C) for the reasons described in the NRC's response to Comment 6-6.

The NRC made no change to the final rule as a result of this comment.

Comment: The reference to ASME BPV Code Case N-824 should be deleted from 10 CFR 50.55a(a)(1)(iii)(D). The nuclear industry is striving to improve examination capability for cast austenitic piping welds and should be allowed to use the most current and technically appropriate methods available at the time of examination. ASME BPV Code Case N-824 should be listed as an approved Code Case in Regulatory Guide (RG) 1.147 without conditions that may restrict a better examination from being performed. [25-3]

NRC Response: The NRC partially agrees with this comment. ASME BPV Code Cases approved for use, with and without conditions, are normally listed in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," not 10 CFR 50.55a. ASME BPV Code Case N-824 was listed in 10 CFR 50.55a to expedite its use. The NRC will move the Code Case from 10 CFR 50.55a and list it in RG 1.147 in a future rulemaking.

The NRC made no change to the final rule as a result of this comment.

Comment: ASME OM Code Case OMN-20 should be incorporated by reference into 10 CFR 50.55a. This is a very important change as this will provide for flexibility for testing due to unforeseen circumstances without requiring regulatory approval and still provide for the detection and monitoring of degradation in a sufficient manner and under adequate frequency of testing and control. [8-3; 14-4]

NRC Response: No response is necessary.

III. ASME BPV Code, Section III

a. 10 CFR 50.55a(b)(1)(viii)

Comment: This condition, addressing the use of components stamped with either the ASME Code Symbol Stamp or the ASME Certification Mark, should be adopted. [8-5]

NRC Response: No response is necessary.

IV. ASME BPV Code, Section XI

a. 10 CFR 50.55a(b)(2)

Comment: Nonmandatory Appendix U was developed to incorporate the provisions of ASME BPV Code Cases N-513-3 and N-705, without any technical changes. It is not clear as to why 10 CFR 50.55a should exclude Nonmandatory Appendix U, because the NRC has approved the use of ASME BPV Code Case N-513-3 in Table 2 of RG 1.147 and has approved ASME BPV Code Case N-705 in Table 1 of RG 1.147. Based on this, the NRC should incorporate Nonmandatory Appendix U by reference in 10 CFR 50.55a. [5-2; 8-1; 8-6; 14-1; 14-5]

NRC Response: The NRC agrees with these comments. The NRC approved ASME BPV Code Cases N-513-3 and N-705 in RG 1.147, which allows licensees to use these Code Cases without prior permission from the NRC. The NRC has changed the final rule to remove the exclusion of Nonmandatory Appendix U from the incorporation by reference. However, the NRC has found it necessary to apply a new condition with two parts in 10 CFR 50.55a(b)(2)(xxxiv) related to the use of Nonmandatory Appendix U. In 10 CFR 50.55a(b)(2)(xxxiv)(A), the NRC requires that ASME BPV Code repair or replacement activity temporarily deferred under the provisions of Nonmandatory Appendix U to the 2013 Edition of the ASME BPV Code, Section XI, shall be performed during the next refueling outage. In 10 CFR 50.55a(b)(2)(xxxiv)(B), the NRC requires the use of the Mandatory Appendix in ASME BPV Code Case N-513-3 in lieu of the appendix referenced in paragraph U-S1-4.2.1(c) of Appendix U, which was inadvertently omitted from Appendix U.

Appendix U defines the evaluation period as the operational time for which the temporary acceptance criteria are satisfied but not exceeding 26 months from the initial discovery of the condition. Original versions of ASME BPV Code Case N-513 stated, in part, that certain flaws may be considered acceptable without performing a repair or replacement activity for a limited time, not exceeding the time to the next scheduled outage. The NRC found that the acceptance of ASME BPV Code Case N-513 was based on allowing continued plant operation with a monitored and evaluated low-safety-significant degraded condition for a limited time until plant shutdown. By allowing use of Appendix U, the NRC would allow this option rather than require an unnecessary plant shutdown to repair the degradation. However, the NRC has determined that once the plant is shut down, the degraded piping shall be repaired. This condition is consistent with the condition placed on ASME BPV Code Case N-513-3 in RG 1.147.

As a result of the public comments, the NRC has changed the final rule to remove the exclusion of Nonmandatory Appendix U from the incorporation by reference.

b. 10 CFR 50.55a(b)(2)(viii)(H) and (I)

Comment: The new condition is unnecessary and should be deleted. The intent of the proposed condition can be accomplished by extending the applicability of existing IWL

Condition E through the 2013 Edition of the ASME BPV Code. Furthermore, it is unclear whether the new condition is intended to apply to inaccessible areas identified as being suspect in accordance with IWL-2512(a), or whether this condition is intended to also apply to IWL-2512(b). Evaluations performed in accordance with IWL-2512(b) would not necessarily identify any inaccessible areas of concrete that would be considered suspect. If not deleted, the condition should only apply to IWL-2512(a). [8-7; 8-8; 10-1; 14-6; 14-7; 16-1; 16-3]

NRC Response: The NRC agrees with a portion of these comments. The NRC disagrees that the condition is unnecessary. The NRC has determined the information identified in the proposed condition must be reported, but the 2013 Edition of the ASME BPV Code does not require reporting the information. Although a similar result could be accomplished by extending the applicability of an existing condition, the NRC has determined that the proposed condition more accurately reflects the 2013 Edition of the ASME BPV Code.

However, the NRC agrees that the proposed condition is not clear about which portion of IWL-2512 it applies to. The condition should be applicable to concrete identified under IWL-2512(a) as well as to concrete that is identified as “susceptible to deterioration” under the IWL-2512(b) evaluation. The NRC understands that the required IWL-2512(b) evaluation may not identify any suspect areas; however, if the evaluation does identify suspect areas, the proposed condition should apply.

Therefore, the NRC has revised the condition to clarify that for each inaccessible area of concrete identified for evaluation under IWL-2512(a), or identified as susceptible to deterioration under IWL-2512(b), the licensee must provide the applicable information specified in paragraphs (b)(2)(viii)(E)(1), (2), and (3) of this section in the Inservice Inspection Summary Report required by IWA-6000.

Comment: Requiring examination of below-grade concrete when excavated for any reason regardless of environment, instead of only when an aggressive environment is present, is appropriate. Since these examinations are opportunistic, the condition does not represent a significant change. [16-2]

NRC Response: No response is necessary.

Comment: The 10-year frequency in the ASME BPV Code is appropriate. Operating experience does not warrant imposing a 5-year frequency for this evaluation and the proposed condition should be deleted. Furthermore, the 2013 Edition contains a provision to examine below-grade concrete when excavated for any reason, when an aggressive below-grade environment is present. The proposed condition may not be necessary for plants without an aggressive below-grade environment. [8-9; 14-8; 25-4]

NRC Response: The NRC agrees with portions of these comments. The NRC believes a 10-year frequency is appropriate for the evaluation of below-grade concrete for plants in the initial 40 years of operation. However, for plants in the period of extended operation (beyond 40 years), the NRC believes the evaluation should be conducted on a 5-year frequency. This aligns with the current NRC guidance for license renewal in NUREG-1801, Revision 2, “Generic Aging Lessons Learned (GALL) Report,” issued December 2010. In addition, the NRC has determined that below-grade concrete should be inspected when excavated for any reason, regardless of the below-grade environment, during the period of extended operation. These inspections are opportunistic (i.e., only required when the excavation is being done for another purpose) and do not place a significant burden on licensees.

The NRC made no change to the rule as a result of these comments.

c. 10 CFR 50.55a(b)(2)(ix)(H)

Comment: The condition may be confusing, as it could be interpreted to mean that a VT-3 examination is required each time the connection is disassembled when the intent is only to examine the connection once per interval even if it is disassembled more often. The NRC should revise the condition to state, "Containment bolted connections that are disassembled during the inspection interval shall be examined at least once with the connection disassembled 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. If the containment bolted connection is not disassembled during the inspection interval, the bolted connections shall be examined with the bolting in place at least once during the inspection interval." [5-3; 8-10; 14-9; 25-5]

NRC Response: The NRC disagrees with these comments. The condition in question has been in the regulations since 2002, and the NRC is not aware of any significant confusion about implementation of the condition. When this issue is considered in context with existing ASME BPV Code, Subsection IWE, Condition G, it is clear that the inspection needs to be done once an interval. If the connection will be disassembled during the interval, the examination needs to be completed with the connection disassembled.

The NRC made no change to the final rule as a result of these comments.

d. 10 CFR 50.55a(b)(2)(xii)

Comment: IWA-4660 was revised in the 2010 Edition of the ASME BPV Code to address this condition. ASME Record #09-1618 was approved by the Section XI Standards Committee on Letter Ballot #10-2158 with support from the NRC member on the standards committee. In light of these approvals, this condition should be revised such that it applies only to those editions and addenda earlier than the 2010 Edition. [8-11]

NRC Response: The NRC agrees that the condition needs to be revised. IWA-4660 was revised in the 2010 Edition of the ASME BPV Code to permit the welding of irradiated P-No. 8 materials containing less than 0.1 atomic parts per million measured or calculated helium content generated through irradiation. Therefore, for the 2010 Edition and later, there is an inconsistency with the proposed rule, which prohibits underwater welding of irradiated materials. In addition, the NRC has noted other inconsistencies for addressing welding on irradiated materials in the ASME BPV Code and in some Code Cases. As the comment implies, rules for welding of irradiated materials should be consistent. The NRC has added the following conditions for welding of irradiated materials to the final rule in 10 CFR 50.55a(b)(2)(xii):

1. Licensees must obtain NRC approval in accordance with 10 CFR 50.55a(z) regarding the welding technique to be used prior to performing welding on ferritic material exposed to fast neutron fluence greater than 1×10^{17} n/cm² ($E > 1$ MeV).
2. Licensees must obtain NRC approval in accordance with 10 CFR 50.55a(z) regarding the welding technique to be used prior to performing welding on austenitic material other than P-No. 8 material exposed to thermal neutron fluence greater than 1×10^{17} n/cm²

($E < 0.5$ eV). Licensees must obtain NRC approval in accordance with 10 CFR 50.55a(z) regarding the welding technique to be used prior to performing welding on P-No. 8 austenitic material exposed to thermal neutron fluence greater than 1×10^{17} n/cm² ($E < 0.5$ eV) and measured or calculated helium concentration of the material greater than 0.1 atomic parts per million.

The commenter's observation of the inconsistency forms part of the basis for revising the condition. The following technical rationale also forms part of the basis for revising the condition. For ferritic materials, a fast neutron fluence greater than 1×10^{17} neutrons per square centimeter (n/cm²) ($E > 1$ million electron volts [MeV]) is the regulatory threshold above which the effects of neutron irradiation damage must be considered. This threshold for ferritic materials is established in Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and also in Regulatory Issue Summary 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," dated October 14, 2014 (ADAMS Accession No. ML14149A165). For austenitic materials, a regulatory threshold for neutron irradiation damage has not been established. However, work documented in Boiling Water Reactor Vessel and Internals Project (BWRVIP) Report 1003020, "BWR Vessel and Internals Project, Guidelines for Performing Weld Repairs to Irradiated BWR Internals" (BWRVIP-97), issued November 2001, suggests that a thermal neutron fluence of 1×10^{17} n/cm² ($E < 0.5$ electron volts [eV]) is the threshold for austenitic materials above which the effects of neutron irradiation damage must be considered when welding. BWRVIP-97 also documents test data for P-No. 8 austenitic material that demonstrate that a measured or calculated helium concentration of 0.1 atomic parts per million can be used as a threshold above which neutron irradiation damage can detrimentally affect welding. The NRC completed its Safety Evaluation of BWRVIP-97 in May 2008 and concluded that implementation of the guidelines in the BWRVIP-97 report, with some modifications as documented in the NRC Safety Evaluation dated June 30, 2008 (ADAMS Accession No. ML081650458), will give an acceptable technical basis for the design of weld repairs based on the helium content of irradiated reactor vessel internals.

In summary, for the reasons set forth above, the NRC is adding two conditions to 10 CFR 50.55a(b)(2)(xii) to address the comment.

e. 10 CFR 50.55a(b)(2)(xv)

Comment: Implementing a large number of possible ASME BPV Code Section XI, Appendix VIII, performance demonstration programs will present severe administrative and logistical challenges to licensees and the Performance Demonstration Institute, which is used by licensees to meet the requirements of Appendix VIII testing. Allowing licensees to use the latest edition and addenda of ASME Code Section XI, Appendix VIII, incorporated by reference without the need for a relief request will allow licensees and PDI to coordinate effectively to maintain compliance with ASME Code and 10 CFR 50.55a requirements. [5-12; 6-2; 8-12; 14-10]

NRC Response: The NRC agrees with these comments. Licensees can currently request the use of later editions of the ASME BPV Code, in whole or in part, using 10 CFR 50.55a(g)(4)(iv) with NRC approval. With the modification in the rule to allow licensees to use the latest edition of ASME BPV Code, Section XI, Appendix VIII, this step would not be required for this specific use of a section of a later edition of ASME BPV Code, Section XI. Allowing licensees to use the

latest NRC-approved version of Appendix VIII will allow licensees to use the latest developments in Appendix VIII approved by the NRC without an essentially unnecessary request.

The NRC has revised 10 CFR 50.55a(g)(4)(ii) of the final rule to state, “Alternatively, licensees may, at any time in their 120-month [inservice inspection] interval, elect to use the Appendix VIII in the latest edition and addenda of the ASME BPV Code incorporated by reference in paragraph (a) of this section, subject to any applicable conditions listed in paragraph (b) of this section. Licensees using this option must also use the same edition and addenda of Appendix I as Appendix VIII, including any applicable conditions listed in paragraph (b) of this section.”

Comment: Implementing a large number of possible ASME Code Section XI, Appendix VIII, performance demonstration programs will present severe administrative and logistical challenges to licensees and the Performance Demonstration Institute, which is used by licensees to meet the requirements of Appendix VIII testing. 10 CFR 50.55a should be modified to contain a provision that requires licensees to use Appendix VIII from the latest edition and addenda of the ASME BPV Code that is incorporated by reference in 10 CFR 50.55a. [27-1]

NRC Response: The NRC disagrees with this comment. The NRC recognizes that allowing additional editions and addenda of ASME BPV Code, Section XI, Appendix VIII, may cause some logistical and administrative challenges for licensees and the Performance Demonstration Institute. However, the NRC does not find a sufficiently strong safety reason to justify requiring licensees to update to the latest edition and addenda of Appendix VIII as soon as they are incorporated by reference. The NRC will allow licensees to update to the latest edition and addenda of ASME BPV Code, Section XI, Appendix VIII, incorporated by reference on a voluntary basis without the use of a relief request to help address the administrative and logistical problems associated with the incorporation of new editions and addenda without placing a regulatory burden on licensees.

The NRC made no change to the final rule as a result of this comment.

Comment: The requirements of 10 CFR 50.55a(b)(2)(xv)(A)(2) are not in alignment with some piping configurations. ASME BPV Code Case N-695-1 and the 2015 Edition of ASME Code Section XI address this issue. To address this inconsistency, 10 CFR 50.55a(b)(2)(xv)(A)(2) should be revised to read, “Single side dissimilar metal weld qualifications shall be performed with specimen sets that contain a range of access restrictions. For components that have scan access from both the ferritic and austenitic sides, qualification shall be performed from the austenitic side of the weld only. For components with no austenitic side, or for which scan access is limited to the ferritic side only, qualification may be performed from the ferritic side. Dissimilar metal welds may be examined from either side of the weld.” [6-4]

NRC Response: The use of ASME BPV Code Case N-695-1 and the 2015 Edition of ASME BPV Code, Section XI, are not part of this rulemaking; therefore, this comment is outside the scope of this rulemaking. Addressing this matter would delay the completion of this rulemaking. The NRC will consider the issue raised in this comment in either the upcoming rulemaking effort related to updating the reference to RG 1.147 or the rulemaking addressing the 2015 Edition of ASME Code, Section XI.

The NRC made no change to the final rule as a result of this comment.

Comment: Paragraph (b)(2)(xv)(L), “Specimen set and qualification: Twelfth provision,” states, “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.” According to ASME BPV Code Section XI, FIG. IWB-2500-12, the examination volume for closure studs starts at the top of the edge of the nut, in the bolted position. The examination volume often does not start within one diameter of the end of the bolt or stud. The condition may result in incorrect placement of the notches. This condition should be modified to match ASME requirements. [26-1]

NRC Response: The NRC disagrees with this comment. The paragraph in question is outside the scope of this rulemaking. Additionally, this statement is optional, not mandatory, and only pertains to licensees using Appendix VIII in the 1995 Edition through the 2001 Edition of the ASME BPV Code.

The NRC made no change to the final rule as a result of this comment.

f. 10 CFR 50.55a(b)(2)(xvi)

Comment: The title of 10 CFR 50.55a(b)(2)(xvi)(A), “Ferritic and stainless steel piping examinations: First provision,” should be changed to, “Ferritic vessel examinations: First provision,” as that provision is only applicable to ferritic vessel examinations. [6-3]

NRC Response: The NRC disagrees with this comment. The NRC understands that the subsection does not cover austenitic welds. The heading of the subsection is consistent with the heading of the overall section and should remain so for clarity.

The NRC made no change to the final rule as a result of this comment.

g. 10 CFR 50.55a(b)(2)(xvii)

Comment: It appears that the NRC imposed this condition because the NRC may have believed that IWA-4222(a)(2) in the 1995 Addenda through the 1999 Addenda would have allowed a licensee to eliminate the reconciliation of applicable QA Program requirements (i.e., 10 CFR 50, App. B; NQA-1; NCA-3800). The ASME Code has never stated that an Owner could reconcile to a Quality Assurance Program not endorsed by the NRC (10 CFR 50, Appendix B; NQA-1; or ASME III NCA-4000). It is ASME’s position that IWA-4222(a)(2) only allows a user to reconcile between endorsed Quality Assurance Programs. To address this concern, an endnote was added to IWA-4222(a)(2) in the 2000 Addenda (ASME Record #99-491) to clarify that the reconciliation provisions regarding administrative requirements do not negate nor modify the Owner’s QA Program requirements. The intent of this change was to eliminate the concern that an Owner could misinterpret the Code to allow any exception to an Owner’s QA Program requirements. For these reasons, the existing condition in 10 CFR 50.55a(b)(2)(xvii) should be removed. [8-13; 25-6]

NRC Response: The NRC understands that the end note was added to address the NRC existing 10 CFR 50.55a(b)(2)(xvii) Section XI condition, as described in the comments. However, the existing condition clarifies the NRC’s position relative to reconciliation of quality assurance program requirements. The NRC is concerned that the end note could easily be misinterpreted by owners.

The NRC made no change to the final rule as a result of these comments.

h. 10 CFR 50.55a(b)(2)(xviii)

Comment: The proposed rule adds 10 CFR 50.55a(b)(2)(xviii)(D), which prohibits applicants and licensees from using the ultrasonic examination nondestructive examination (NDE) personnel certification requirements in Section XI, Appendix VII, and subarticle VIII-2200 of the 2011 Addenda and 2013 Edition of the ASME BPV Code. Specifically, the proposed rule would prohibit the use of laboratory training as a substitute for hours of experience in the field by not allowing the use of the 2011 and 2013 versions of Appendix VII. The industry is finding it challenging to recruit and retain NDE staff, and the condition will make it more difficult to quickly train NDE staff to meet reduced outage times. The use of laboratory training in lieu of experience hours will allow the industry to train staff in less time. The small number of flaws found in the field make it possible that laboratory testing will provide inspectors with more flaws during training. It is therefore requested that the condition be removed and allow for the substitution of laboratory time for experience hours. [5-4; 8-14]

NRC Response: The NRC disagrees with these comments. The primary reasons stated for the use of laboratory hours in lieu of experience are entirely related to the logistics and costs associated with training and recruiting NDE staff, and not to the relative skill of the inspectors and the safety provided by laboratory training. While the NRC is open to evidence related to a technical basis for the substitution of laboratory experience as a substitute for hours of work experience, the NRC cannot use cost and convenience alone as justifications for this substitution. The effects of the substitution of laboratory hours for field experience and nuclear power plant familiarization are unknown. The ASME BPV Code replaces field experience with training hours without defined technical bases, process details, or standardization.

The NRC made no change to the final rule as a result of these comments.

Comment: The NRC proposes to add 10 CFR 50.55a(b)(2)(xviii)(D), which references Section XI, Appendix VII, and Subarticle VIII-2200 of the 2011 Addenda and 2013 Edition of the ASME BPV Code. The NRC should clarify which appendix it wishes the proposed rule to affect, as the rule appears to incorrectly refer to two separate appendices. [19-2]

NRC Response: The NRC disagrees with this comment. Before the 2011 Addenda, Subarticle VIII-2200 required an inspector to be qualified to Appendix VII requirements as a prerequisite to obtaining an Appendix VIII qualification. The proposed rule language would prevent the use of the 2011 Addenda and 2013 Edition of Appendix VII and assure that inspectors qualified under Appendix VIII also meet the requirements of the 2008 Addenda to Appendix VII.

The NRC made no change to the final rule as a result of this comment.

Comment: The proposed rule would require licensees to use the 2010 Edition, Table VII-4110-1, training hour requirements for Levels I, II, and III ultrasonic examination personnel, and the 2010 Edition, subarticle VIII-2200 of Appendix VIII prerequisites for personnel requirements. Many licensees are using the 2008 Edition, and requiring them to use a later edition would be challenging for the licensees and an insufficient backfit analysis has been performed. [19-3]

NRC Response: The NRC disagrees with this comment. The proposed rule language would only restrict the use of the 2011 and 2013 Editions of Appendix VII and Subarticle VIII-2200 for licensees adopting the 2011 or 2013 Editions or Addenda at the beginning of a new 10-year

inservice inspection interval. The proposed rule would not affect licensees using the 2008 Addenda.

The NRC made no change to the final rule as a result of this comment.

i. 10 CFR 50.55a(b)(2)(xx)

Comment: The additional NDE imposed by the NRC condition in 10 CFR 50.55a(b)(2)(xx)(B) is unnecessary and implies that existing components are unsuitable. The NRC should remove the referenced condition from 10 CFR 50.55a(b)(2)(xx)(B). [8-15]

NRC Response: The NRC considers this matter to be outside the scope of this rulemaking, as the proposed rule contains no change to this requirement. However, the NRC also disagrees that the additional NDE requirements imposed by 10 CFR 50.55a(b)(2)(xx)(B) are unnecessary and imply that existing components are unsuitable. The NRC does agree that hydrostatic pressure testing or NDE alone do not ensure structural integrity. The original Construction Codes ensured structural integrity through a combination of many factors, including material testing, design formulas, design factors, qualification of procedures, qualification of personnel, NDE, and hydrostatic testing. Since the incorporation of ASME BPV Code Case N-416-4, Section XI would allow a system leakage test to be performed in lieu of (1) a hydrostatic pressure test before return to service of Class 1, 2, and 3 welded or brazed repairs, (2) fabrication welds or brazed joints for replacement parts and piping subassemblies, or (3) installation of replacement items by welding or brazing.

The NRC has determined that the rigorous NDE requirements of Section III should be performed when the hydrostatic pressure test is not performed. The reason for this condition is that some earlier Construction Codes have less stringent NDE requirements than Section III; however, they require a greater pressure than the Section XI required pressure test. Section III NDE requirements for Class 1, 2, and 3 components generally require either surface or volumetric examinations, or possibly both. The volumetric examination is generally required for full-penetration welds and the surface examination for partial-penetration welds. The NRC has determined that these NDE requirements along with a system leakage test give the same level of quality and safety as the higher pressure hydrostatic test and reduced NDE requirements of earlier Construction Codes.

No changes were made to the final rule as a result of this comment.

j. 10 CFR 50.55a(b)(2)(xxi)

Comment: The NRC proposes to revise paragraph (b)(2)(xxi)(A) to specify the standard for visual magnification resolution sensitivity and contrast for VT-1 tests, making the rule in line with ASME BPV Code Section IX requirements for VT-1 inspections. ASME supports the proposed change to this condition. [8-16]

NRC Response: No response is necessary.

Comment: The NRC proposes to revise paragraph (b)(2)(xxi)(A) to specify the standard for visual magnification resolution sensitivity and contrast for VT-1 tests, making the rule in line with ASME Code Section IX requirements for VT-1 inspections. The requirement for the nozzle inner radius examination for Class I pressurizers and steam generators should be removed entirely, as the inside surface examinations can result in significant personnel radiation dose,

increased probability of loose parts, and in some cases significantly increased time at elevated risk conditions. [14-11; 25-7]

NRC Response: The NRC partially agrees with these comments. The NRC is evaluating 10 CFR 50.55a to find sections that result in repetitive relief requests that may have a limited safety impact. One of the regulations the NRC is evaluating is the nozzle inner radius examination requirements in 10 CFR 50.55a(b)(2)(xxi)(A). While the NRC is evaluating the history and safety significance of the nozzle inner radius examinations, this work is not yet complete. If the NRC determines that there is an insufficient safety benefit to this requirement, it may be modified or eliminated in a future rulemaking.

The NRC made no change to the final rule as a result of these comments.

k. 10 CFR 50.55a(b)(2)(xxiii)

Comment: ASME Record #06-853 revised IWA-4461.4 and deleted paragraph IWA-4461.4.2. This action was approved by the Section XI Standards Committee and was published in the 2010 Edition. Accordingly, the NRC should revise this condition so that it applies only to the 2001 Edition through the 2009 Addenda. [8-17]

NRC Response: The NRC agrees that the condition should only apply to the 2001 Edition through the 2009 Addenda because IWA-4461.4 was revised in the 2010 Edition to delete paragraph IWA-4461.4.2, which permitted an application-specific evaluation of thermally cut surfaces in lieu of a thermal metal removal process qualification. In addition, IWA-4461.4 was revised to include appropriate requirements for thermal metal removal qualification.

The NRC has revised 10 CFR 50.55a(b)(2)(xxiii) in the final rule to apply to the 2001 Edition through the 2009 Addenda. The revised 10 CFR 50.55a(b)(2)(xxiii) Section XI condition regarding the evaluation of thermally cut surfaces states, "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 2009 Addenda, is prohibited."

l. 10 CFR 50.55a(b)(2)(xxv)

Comment: The NRC should revise the condition in 10 CFR 50.55a(b)(2)(xxv) so that it applies only to the 2001 Edition through the 2010 Edition. [8-18; 25-8]

NRC Response: The NRC disagrees with these comments. Section IWA-4340, through the 2013 Edition of Section XI, has not resolved the issue of corrosion rate determination and validation for a repair that could remain in service for the remaining life of the plant. The NRC staff continues to work with the ASME Code Committee to resolve issues with IWA-4340 for future versions of Section XI.

The NRC made no change to the final rule as a result of these comments.

m. 10 CFR 50.55a(b)(2)(xxvi)

Comment: The NRC should remove the condition set in (b)(2)(xxvi), which requires that the licensee must perform a system pressure test and a VT-2 for R/R activities of mechanical connections. [3-1; 8-19]

NRC Response: The NRC disagrees with these comments. The NRC's continued position is that it should be a requirement that Class 1, 2, and 3 mechanical joints be pressure tested in accordance with IWA-4540(c), as described in the 1998 Edition of Section XI. The NRC has determined that when a repair or replacement activity (which includes IWA-4132, "Items Rotated From Stock") is performed on a mechanical joint, an opportunity exists to create a condition that could disturb the joint and cause leakage. Although this leakage may not be an immediate challenge to the structural integrity of the joint, it could cause a condition that could become a structural integrity issue if not detected and corrected. The NRC has determined that this could be the case particularly in moderate- or high-energy systems and in systems containing fluids, such as boric acid, that could cause corrosion. Therefore, the NRC determined that there should be a requirement for a pressure test using VT-2 qualified individuals and that the Section XI acceptance standards and corrective action should apply.

The NRC made no change to the final rule as a result of these comments.

Comment: If the NRC determines not to delete the condition in 10 CFR 50.55a(b)(2)(xxvi), then the NRC should confirm that the existing 10 CFR 50.55a(b)(2)(xxvi) condition does not take exception to the requirement of IWA-4540(c) in the 1998 Edition (as clarified by Interpretation XI-1-10-20). If clarification of these requirements warrants revising 10 CFR 50.55a(b)(2)(xxvi), then the NRC should consider clarifying this condition. [8-20]

NRC Response: The NRC agrees with the comment in that if no replacement of pressure-retaining parts is performed, the activity is not an ASME BPV Code, Section XI, activity and no ASME BPV Code pressure test is required. This activity is a maintenance activity, and the owner would perform a post maintenance test to insure the leak tightness of the system. However, the NRC disagrees with the replacement of bolting not requiring a pressure test, as discussed in ASME Interpretation XI-1-10-20. As the NRC stated in response to Comment 8-19:

The NRC has determined that when a repair or replacement activity (which includes IWA-4132, "Items Rotated From Stock") is performed on a mechanical joint, an opportunity exists to create a condition that could disturb the joint and cause leakage. Although this leakage may not be an immediate challenge to the structural integrity of the joint, it could cause a condition that could become a structural integrity issue if not detected and corrected. The NRC has determined that this could be the case particularly in moderate- or high-energy systems and in systems containing fluids, such as boric acid, that could cause corrosion. Therefore, the NRC determined that there should be a requirement for a pressure test using VT-2 qualified individuals and that the Section XI acceptance standards and corrective action should apply.

The NRC made no change to the final rule as a result of this comment.

n. 10 CFR 50.55a(b)(2)(xxx)

Comment: The condition in 10 CFR 50.55a(b)(2)(xxx) of the proposed rule is unclear as to what kind of examination is required for the steam generator tube preservice inspections (PSIs), and what acceptance criteria apply. Since this requirement will replace the provisions of Section XI, something needs to be specified for examination type and acceptance criteria. Do the examinations required by NB-2000 satisfy this requirement? [2-1; 5-5; 8-21; 14-12; 15-1; 21-1; 25-9]

NRC Response: The NRC agrees with the need for clarification expressed in these comments; however, during the development of the final rule, the NRC determined that additional time was needed to evaluate this proposed condition. Therefore, to ensure that this rulemaking is concluded as timely as possible, the NRC is not including this condition in this final rule and will consider adding it in a future rulemaking. The NRC has concluded that omitting this condition does not present a health or safety concern because licensees are currently performing appropriate steam generator preservice inspections under existing programs.

o. 10 CFR 50.55a(b)(2)(xxxi)

Comment: For clarity, the proposed condition in 10 CFR 50.55a(b)(2)(xxxi) should cite the specific paragraphs of Section XI to which the NRC is taking exception. [15-2; 25-10]

NRC Response: The NRC agrees with these comments. The NRC created this condition when ASME changed from mandatory to nonmandatory the mechanical clamping device appendix that previously restricted the use of mechanical clamping devices on ASME Class 1 components. The NRC condition states that the use of mechanical clamping devices on Class 1 piping and portions of piping systems that form the containment boundary is prohibited. Although this statement is clear, mechanical clamping devices can be implemented in potentially different ways through the ASME Code, as discussed in Comment 15-2. To clarify the requirement for the implementation of mechanical clamps, the NRC has changed the condition in the final rule to require the use of Appendix W to Section XI when using mechanical clamps. Additionally, the NRC has prohibited the use of IWA-4131.1(c) of the 2010 Edition of Section XI and IWA-4131.1(d) of the 2011 Addenda to the 2010 Edition and later versions of Section XI. The condition maintains the previous regulatory requirement for the implementation of mechanical clamping devices on ASME BPV Code Class components. Therefore, it remains consistent with the relevant backfit discussion in the proposed rule.

The final rule at 10 CFR 50.55a(b)(2)(xxxi) reflects this change.

Comment: There is no sound basis for the conclusion of proposed condition 10 CFR 50.55a(b)(2)(xxxi) on the restriction on the use of mechanical clamping devices. These devices have been proven time and time again to be quite effective and very near permanent. [19-4]

NRC Response: The NRC disagrees with this comment. The NRC has found mechanical clamping devices are acceptable for use on ASME Code Class 2 and 3 systems and components when applied under the requirements of Mandatory Appendix IX to the 2008 Addenda to the 2007 Edition of Section XI of the ASME BPV Code. However, the NRC did not find sufficient technical basis was given to justify changing Appendix IX to a non-mandatory appendix (Appendix W). In making this change, the ASME BPV Code allows mechanical clamping devices to be used on previously prohibited ASME Class 1 systems and piping. Further, as identified in Comment 15-2, this change in the ASME BPV Code would allow mechanical clamping devices to be used on ASME Class 2 and 3 systems and piping without defined testing and design requirements.

The NRC made no change to the final rule as a result of this comment.

p. 10 CFR 50.55a(b)(2)(xxxii)

Comment: ASME recommends that this condition specifically reference paragraph IWA-6240 where the report submittal provisions are stated in Section XI. [8-22]

NRC Response: The NRC agrees with this comment. IWA-6240 is the paragraph of the ASME BPV Code that describes the completion of the summary report. Subparagraph (a) describes the completion of the preservice inspection report, subparagraph (b) describes the completion of the inservice inspection report, and subparagraph (c) describes submittal to the regulatory authority if required.

The NRC has changed the final rule to reference Section XI, paragraph IWA-6240, to state, "When using ASME BPV Code, Section XI, 2010 Edition through the latest edition and addenda incorporated by reference in paragraph (a)(1)(ii) of this section, Summary Reports described in IWA-6000 must be submitted to the NRC as described in IWA-6240(a) and IWA-6240(b)."

q. 10 CFR 50.55a(b)(2)(xxxiii)

Comment: Rather than prohibit its use, the NRC should provide approval with a condition that requires the licensee to obtain prior approval by the NRC of the methodology and results. The condition can also require the licensees to demonstrate that no surface breaking flaw exists within the IWB-2500 inspection volumes for the RPV beltline. These conditions would address the basis for the staff's negative vote on the proposed ASME action. [5-6; 8-23; 14-13]

NRC Response: The NRC disagrees with these comments. The analysis in MRP-250, "Technical Basis for Revision to ASME Code to Appendix G: Incorporate Risk Informed P-T Methodology," dated June 30, 2009, supporting the risk-informed alternative in ASME BPV Code, Appendix G, does not consider surface-breaking flaws and nozzle regions. Further, the NRC has not completed its review of MRP-250 and, therefore, cannot conclude that all technical issues associated with the risk-informed alternative have been identified.

The NRC made no change to the final rule as a result of these comments.

r. 10 CFR 50.55a(b)(2)(xxxiv)

Comment: The proposed condition in 10 CFR 50.55a(b)(2)(xxxiv) is no longer necessary and should be removed because ASME has addressed the typographical error in Table IWD-3401-1 by errata (Record #14-776). [5-7; 8-24; 25-11]

NRC Response: The NRC agrees with these comments. Because the error in the table was corrected by the published errata in ASME Records 14-1395 and 14-776, the NRC has revised the final rule to eliminate the proposed condition in 10 CFR 50.55a(b)(2)(xxxiv) regarding disposition of flaws in Class 3 components. As discussed in the response to Comment 5-2, the NRC has added a new condition regarding the use of Nonmandatory Appendix U to 10 CFR 50.55a(b)(2)(xxxiv).

s. 10 CFR 50.55a(b)(2)(xxxvii)

Refer to the discussion of ASME BPV Code Case N-824 in Section I of this document, Question 2.

V. ASME OM Code

a. 10 CFR 50.55a(b)(3)(ii)

Comment: The proposed revision to this condition requires implementation of Mandatory Appendix III for motor-operated valve (MOV) inservice testing (IST) and effectively codifies

existing NRC Generic Letter 96-05 requirements. This change is expected to add additional IST active MOVs into a licensee's MOV Diagnostic Test Program, depending on the plant. This change will result in less flexibility with waiving as-found MOV IST diagnostic testing since MOV diagnostic testing becomes the de facto MOV IST surveillance test of record. While this change will introduce additional burden, it is expected and is not a change from that approved under ASME OM Code, Appendix III. [8-26; 25-13]

NRC Response: No response is necessary.

b. 10 CFR 50.55a(b)(3)(ii)(A)

Comment: The proposed 10 CFR 50.55a(b)(3)(ii)(A), "MOV Diagnostic Test Interval," contains the following text:

...require that licensees evaluate the adequacy of the diagnostic test interval for each MOV [motor-operated valve] and adjust the interval as necessary, but not later than 5 years or three refueling outages (whichever is longer) from initial implementation of ASME OM Code, Appendix III.

For existing plants with mature MOV Programs that are utilizing the Joint Owners Group (JOG) Program (ADAMS Accession No. ML063490199), most plant MOVs are already on an established periodic verification test interval based on margin and risk per the JOG static test interval matrix. In many cases, these test intervals are longer than 5 years or three refueling outages. In the NRC safety evaluation report (ADAMS Accession No. ML061280315) for the JOG Program, page 21 contains the following text:

Condition J specified that MOVs with scheduled test frequencies beyond 5 years will need to be grouped with other MOVs that will be tested on frequencies less than 5 years in order to validate assumptions for the longer test intervals. This condition is superseded by the test intervals established by the long-term JOG program.

This new NRC rulemaking appears to require operating plants to limit the existing (mature program) MOV periodic verification test intervals to 5 years or three refueling outages maximum at the time of implementing Appendix III until "sufficient data exist" to justify longer test intervals. If the intent was for this to apply to new reactors or for new MOVs at existing plants, this clarification should be added. There is no benefit nor any evidence of a problem which would warrant this requirement being applied to existing plants with mature MOV programs. [1-1; 8-27; 10-2; 14-20; 21-2; 25-14]

NRC Response: The NRC agrees that the condition in 10 CFR 50.55a(b)(3)(ii)(A) should be clarified because the proposed language might have been interpreted as limiting the MOV periodic verification test intervals to 5 years or three refueling outages at the time of initial implementation of ASME OM Code, Appendix III.

In response to the comments, the NRC recognized that the references to "each MOV" and "adjust the interval as necessary" before the phrase "not later than 5 years or three refueling outages" in the proposed condition might be interpreted to imply that every MOV must be tested within 5 years or three refueling outages of the initial implementation of ASME OM Code, Appendix III. However, the condition is intended to allow grouping of MOVs to share test information in the evaluation of the MOV periodic verification intervals within 5 years or three refueling outages of the implementation of ASME OM Code, Appendix III.

During the public meeting on March 2, 2016, commenters indicated that the planned clarification of 10 CFR 50.55a(b)(3)(ii)(A) resolved their comments.

Therefore, the NRC has revised 10 CFR 50.55a(b)(3)(ii)(A) in the final rule to read, "Licensees shall evaluate the adequacy of the diagnostic test intervals established for MOVs within the scope of ASME OM Code, Appendix III, not later than 5 years or three refueling outages (whichever is longer) from initial implementation of ASME OM Code, Appendix III."

c. 10 CFR 50.55a(b)(3)(ii)(B)

Comment: This condition should be adopted with specific core damage frequency/large early release frequency criteria. [8-28]

NRC Response: The NRC does not agree with the comment that the regulation should include specific core damage frequency (CDF) and large early release frequency (LERF) criteria. The NRC provides specific criteria for evaluating CDF and LERF in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." As indicated in RG 1.174, the NRC considers acceptably small changes to be relative and to depend on the current plant CDF and LERF. For plants with total baseline CDF of 10^{-4} per year or less, acceptably small means CDF increases of up to 10^{-5} per year; and for plants with total baseline CDF greater than 10^{-4} per year, acceptably small means CDF increases of up to 10^{-6} per year. For plants with total baseline LERF of 10^{-5} per year or less, acceptably small LERF increases are considered to be up to 10^{-6} per year; and for plants with total baseline LERF greater than 10^{-5} per year, acceptably small LERF increases are considered to be up to 10^{-7} per year. The NRC considers it appropriate to provide the specific CDF and LERF criteria in RG 1.174.

The NRC made no change to the final rule as a result of this comment.

d. 10 CFR 50.55a(b)(3)(ii)(C)

Comment: The NRC should clarify whether the intent of this condition is to require that existing plants utilize only the two risk categories (High and Low) or allow the continued use of three risk categories (High, Medium, and Low) when establishing the periodic verification (inservice) test intervals per the JOG matrix. [8-29; 14-21; 25-15]

NRC Response: The intent of this condition is to indicate that when applying Appendix III to the ASME OM Code, licensees may use either a two-risk category approach or three-risk category approach, provided the risk-ranking method has been accepted by the NRC. The condition in 10 CFR 50.55a(b)(3)(ii)(C) is consistent with a similar condition in RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," that accepted with conditions the implementation of ASME OM Code Case OMN-1.

The NRC made no change to the final rule as a result of these comments.

e. 10 CFR 50.55a(b)(3)(ii)(D)

Comment: Quarterly stroke time testing should not be required in 10 CFR 50.55a. [1-2; 8-30; 10-3; 12-1; 12-2; 12-3; 14-22; 21-3; 25-16]

NRC Response: The NRC agrees with these comments that only those MOVs that have an isolation time limit to meet design-basis event assumptions in plant TS need to have their stroke

times verified during the valve exercise test specified in Appendix III. Therefore, the NRC has clarified the condition to indicate that it applies to MOVs specified in plant TS, as discussed below.

Currently, the TS at many operating nuclear power plants include a surveillance requirement for certain MOVs to verify that their isolation time is within limits specified in the plant safety analysis. The TS surveillance requirement frequency for some plants is listed as “In accordance with the Inservice Testing Program.” Other plants have frequencies of 92 days, or “In accordance with Surveillance Frequency Control Program.” This TS surveillance requirement is related to the analyses of a design-basis event that differs from the ASME OM Code stroke time measurement, which is evaluating potential valve degradation and verifying operational readiness as part of the IST program. ASME OM Code, Appendix III, replaces the MOV quarterly stroke time testing provision in the ASME OM Code with a 24-month exercise interval and diagnostic testing at longer intervals. Therefore, the proposed condition assures that licensees will verify the isolation time of MOVs identified in their TS at a frequency of “In accordance with the Inservice Test Program,” when those MOVs are exercised using the provisions of ASME OM Code, Appendix III.

Based on the discussion during the public meeting on March 2, 2016, the NRC clarified the condition in 10 CFR 50.55a(b)(3)(ii)(D) to apply to MOVs referenced in the plant TS. The participants at the public meeting indicated that the planned clarification resolved their concerns.

The NRC has revised the condition to indicate that when a licensee applies Paragraph III-3600, “MOV Exercising Requirements,” of Appendix III to the ASME OM Code, the licensee shall verify that the stroke time of MOVs specified in plant TS satisfies the assumptions in the plant safety analyses.

f. 10 CFR 50.55a(b)(3)(iii)(A)

Comment: The NRC should consider revising the proposed condition requiring design-bases verification on all power-operated valves (POVs) to require verification of the POV ability to perform the ASME OM Code-specified IST safety function, which would be more appropriate since many solenoid-operated valves, hydraulic-operated valves, and such do not have methods developed, other than ASME OM Code, to determine “operational readiness” at this time. The ASME OM Code is looking at establishing and identifying methods for these types of valves in the future, but there does not appear to be an “industrywide concern” identified as of yet regarding the necessity of being able to periodically verify design bases for these types of valve/actuators, as identified for MOVs and air-operated valves (AOVs). [8-31; 14-23; 21-4; 24-8]

NRC Response: The NRC disagrees with these comments because this condition is based on Commission policy that is being implemented during the ongoing licensing of new reactors.

The provisions in 10 CFR 50.55a(b)(3)(iii) of the proposed rule will apply to holders of operating licenses for nuclear power reactors that received construction permits under 10 CFR Part 50 on or after the date 12 months after the effective date of the final rule, and holders of combined licenses (COLs) issued under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” whose initial fuel loading occurs on or after the date 12 months after the effective date of the final rule. As such, these provisions will not apply to current holders of operating licenses under 10 CFR Part 50. Future construction permit applicants under 10 CFR Part 50 will need to address these provisions in their applications. With respect to current and future COL holders under 10 CFR Part 52, the NRC will require those licensees to

apply the provisions in 10 CFR 50.55a(b)(3)(iii) when their initial fuel loading occurs on or after the date 12 months after the effective date of the final rule.

NRC Commission Papers SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990; SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated April 2, 1993; SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994; and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," dated May 22, 1995, and their staff requirements memoranda (SRMs) discuss the Commission policy related to IST programs for new reactors. In the NRC Staff Memorandum, "Consolidation of SECY-94-084 and SECY-95-132," dated July 24, 1995 (ADAMS Accession No. ML003708048), the NRC staff consolidated the guidance in SECY-94-084, SECY-95-132, and their respective SRMs. The provisions in 10 CFR 50.55a(b)(3)(iii) are based on the Commission policy developed for new reactors and the NRC staff review of COL applications for new reactors to date.

As part of its review of COL applications under 10 CFR Part 52, the NRC evaluates the descriptions of IST programs submitted by COL applicants in their final safety analysis reports (FSARs), including compliance with the ASME OM Code as incorporated by reference into NRC regulations and implementation of the Commission policy on IST programs for new reactors. The NRC staff's review of the descriptions of IST programs in COL applications has included the provisions the NRC is proposed to incorporate into 10 CFR 50.55a(b)(3)(iii). For example, in NUREG-2124, "Final Safety Evaluation Report Related to the Combined Licenses for Vogtle Electric Generating Plant, Units 3 and 4," issued September 2012, the NRC staff describes its review of the IST program description submitted by the COL applicant in the FSAR for Vogtle Units 3 and 4. In NUREG-2124, the NRC staff found that the IST program description in the Vogtle Units 3 and 4 FSAR, including aspects incorporated by reference from the AP1000 Design Control Document (DCD), was acceptable. The NRC considers the IST provisions in the Vogtle Units 3 and 4 FSAR, including aspects incorporated by reference from the AP1000 DCD, to be acceptable to satisfy the conditions in 10 CFR 50.55a(b)(3)(iii).

With respect to paragraph (A) of 10 CFR 50.55a(b)(3)(iii), the proposed rule specifies that new reactor licensees shall periodically verify the capability of POVs to perform their design-basis safety functions. This provision is consistent with the Commission policy summarized in the NRC Staff Memorandum dated July 24, 1995, that a) the design capability of safety-related POVs should be demonstrated by a qualification test prior to installation; b) prior to initial startup, POV capability under design-basis differential pressure and flow should be verified by a pre-operational test; and c) during the operational phase, POV capability under design-basis differential pressure and flow should be verified periodically through a program similar to that developed for MOVs in Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," dated June 28, 1989.²

The proposed condition in 10 CFR 50.55a(b)(3)(iii)(A) specifies to the same level of detail the current condition in 10 CFR 50.55a(b)(3)(ii) that nuclear power plant licensees must establish a program to ensure that MOVs continue to be capable of performing their design-basis safety

² The NRC issued seven supplements to provide guidance for the implementation of the MOV testing program requested in Generic Letter 89-10. The supplements to Generic Letter 89-10 did not modify the substance of the MOV testing program requested in Generic Letter 89-10 to provide reasonable assurance in the capability of safety-related MOVs to perform their design-basis functions.

functions. When establishing the MOV periodic verification condition, the NRC gave guidance on developing programs that would satisfy the MOV periodic verification condition in the *Federal Register* notice (64 FR 51370; September 22, 1999) for rulemaking for licensees. Similarly, the NRC included guidance on developing acceptable programs to periodically verify the capability of POVs to perform their design-basis safety functions in the statement of considerations for this final rule for new reactor applicants and licensees.

In NUREG-2124, the NRC staff found acceptable the provisions established by the COL applicant for Vogtle Units 3 and 4 in its FSAR to periodically verify the capability of POVs (such as AOVs, solenoid-operated valves, and hydraulic-operated valves) to perform their design-basis safety functions. In particular, the Vogtle Units 3 and 4 FSAR specifies the following:

Power-operated valves other than active MOVs are exercised quarterly in accordance with ASME OM ISTC, unless justification is provided in the inservice testing program for testing these valves at other than Code mandated frequencies. Although the design basis capability of power-operated valves is verified as part of the design and qualification process, power-operated valves that perform an active safety function are tested again after installation in the plant, as required, to ensure valve setup is acceptable to perform their required functions, consistent with valve qualification. These tests, which are typically performed under static (no flow or pressure) conditions, also document the "baseline" performance of the valves to support maintenance and trending programs. During the testing, critical parameters needed to ensure proper valve setup are measured. Depending on the valve and actuator type, these parameters may include seat load, running torque or thrust, valve travel, actuator spring rate, bench set and regulator supply pressure. Uncertainties associated with performance of these tests and use of the test results (including those associated with measurement equipment and potential degradation mechanisms) are addressed appropriately. Uncertainties may be considered in the specification of acceptable valve setup parameters or in the interpretation of the test results (or a combination of both). Uncertainties affecting both valve function and structural limits are addressed. Additional testing is performed as part of the air-operated valve (AOV) program, which includes the key elements for an AOV Program as identified in the JOG AOV program document, Joint Owners Group Air Operated Valve Program Document, Revision 1, December 13, 2000 (References 203 and 204 [JOG AOV Program Document, Revision 1, December 13, 2000 (ADAMS Accession No. ML010950310), and NRC, Eugene V. Imbro, NRC, letter to Mr. David J. Modeen, Nuclear Energy Institute, "Comments on Joint Owners' Group Air Operated Valve Program Document," dated October 8, 1999 (ADAMS Accession No. ML020360077)]). The AOV program incorporates the attributes for a successful power-operated valve long-term periodic verification program, as discussed in Regulatory Issue Summary 2000-03, Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions, by incorporating lessons learned from previous nuclear power plant operations and research programs as they apply to the periodic testing of air- and other power-operated valves included in the IST program.

For example, key lessons learned addressed in the AOV program include:

- Valves are categorized according to their safety significance and risk ranking.
- Setpoints for AOVs are defined based on current vendor information or valve qualification diagnostic testing, such that the valve is capable of performing its design-basis function(s).
- Periodic static testing is performed, at a minimum on high risk (high safety significance) valves, to identify potential degradation, unless those valves are periodically cycled during normal plant operation, under conditions that meet or exceed the worst case operating conditions within the licensing basis of the plant for the valve, which would provide adequate periodic demonstration of AOV capability. If required based on valve qualification or operating experience, periodic dynamic testing is performed to re-verify the capability of the valve to perform its required functions.
- Sufficient diagnostics are used to collect relevant data (e.g., valve stem thrust and torque, fluid pressure and temperature, stroke time, operating and/or control air pressure, etc.) to verify the valve meets the functional requirements of the qualification specification.
- Test frequency is specified, and is evaluated each refueling outage based on data trends as a result of testing. Frequency for periodic testing is in accordance with References 203 and 204, with a minimum of 5 years (or 3 refueling cycles) of data collected and evaluated before extending test intervals.
- Post-maintenance procedures include appropriate instructions and criteria to ensure baseline testing is re-performed as necessary when maintenance on the valve, repair or replacement, have the potential to affect valve functional performance.
- Guidance is included to address lessons learned from other valve programs specific to the AOV program.
- Documentation from AOV testing, including maintenance records and records from the corrective action program are retained and periodically evaluated as a part of the AOV program.

The attributes of the AOV testing program described above, to the extent that they apply to and can be implemented on other safety-related power-operated valves, such as electro-hydraulic operated valves, are applied to those other power-operated valves.

The NRC considers that holders of operating licenses for nuclear power reactors that receive construction permits under 10 CFR Part 50 on or after the date 12 months after the effective date of the final rule, and holders of COLs issued under 10 CFR Part 52 whose initial fuel loading occurs on or after the date 12 months after the effective date of the final rule, may follow

the method described in the Vogtle Units 3 and 4 FSAR in satisfying 10 CFR 50.55a(b)(3)(iii)(A) or may establish a different method subject to evaluation by the NRC during licensing review or inspections.

In Section II.C, "OM Code," of the statement of considerations for the final rule, the NRC describes the Commission policy on IST programs for new reactors. In addition, the NRC indicates methods, such as those described in the Vogtle Units 3 and 4 FSAR, that are acceptable for satisfying 10 CFR 50.55a(b)(3)(iii)(A).

The NRC made no change to the rule as a result of this comment.

g. 10 CFR 50.55a(b)(3)(iii)(B)

Comment: This condition is unnecessary considering that the edition/addenda of the ASME OM Code that would be applicable to IST programs of new reactors already include the requirements for bidirectional testing of check valves in Subsection ISTC and Appendix II. [21-5]

NRC Response: The NRC agrees that the ASME OM Code requirements for bidirectional testing of check valves may be satisfied through the provisions of Subsection ISTC or Appendix II to the ASME OM Code. However, as discussed below, the NRC disagrees that this condition is unnecessary because this provision requires that new reactor designs provide design attributes to allow bi-directional testing of check valves.

The NRC response to the comments related to 10 CFR 50.55a(b)(3)(iii)(A) describes the basis for and applicability of the provisions in 10 CFR 50.55a(b)(3)(iii).

Proposed paragraph (B) in 10 CFR 50.55a(b)(3)(iii) states that new reactor licensees must perform bi-directional testing of check valves within the IST program where practicable. This proposed condition is based on the Commission policy specified in Commission Papers SECY-94-084 and SECY-95-132, and their respective SRMs, for the IST programs to be established for new reactors. At that time, the ASME OM Code did not clearly require bi-directional testing of check valves as part of the IST programs at nuclear power plants.

Although the ASME OM Code currently addresses check valve testing in both directions, the NRC included the provision for bi-directional testing of check valves for new reactors in 10 CFR 50.55a(b)(3)(iii)(B) to emphasize that new reactors should include the capability for bi-directional testing of check valves as part of their initial design. The NRC recognizes that bi-directional testing for many check valves in currently operating nuclear power plants might be practicable only to a limited extent because of the design of their plant systems. For example, the ASME OM Code includes provisions for testing of check valves depending on their safety direction in ISTC-5221, "Valve Obturator Movement."

The NRC agrees that verification of bi-directional capability of check valves could be accomplished by testing or as part of condition-monitoring activities allowed in ASME OM Code, Appendix II, "Check Valve Condition Monitoring Program." Therefore, the NRC revised Section II.C of the statement of considerations for the final rule to state that bi-directional testing of check valves in new reactors as required by 10 CFR 50.55a(b)(3)(iii)(B) can be accomplished by valve-specific testing or condition-monitoring activities in accordance with Appendix II to the ASME OM Code as accepted in 10 CFR 50.55a.

h. 10 CFR 50.55a(b)(3)(iii)(C)

Comment: It is not clear whether this condition will provide any improved method for detecting and monitoring for degradation of the valve by the use of IST. There presently is no guidance regarding this in the ASME OM Code and there does not appear to be a need. A more sound and readily available recommendation may be to provide an evaluation of any flow-induced vibration during the preservice test period and/or the post-maintenance test period, if the applicable flow-induced vibration is identified during this period of time. Perhaps then a test or method (outside of the IST scope) could be determined and included during the post-maintenance testing. [8-32; 10-4; 14-24; 21-6]

NRC Response: The NRC agrees with the comments that detection and monitoring for degradation of components caused by flow-induced vibration may be accomplished during preservice testing, IST, or post-maintenance testing. As discussed below, the NRC considers the condition on flow-induced vibration monitoring to be necessary in light of lessons learned from experience at operating nuclear power plants. In response to this public comment, the NRC has revised the condition in the final rule to allow the licensee to monitor flow-induced vibration of components from hydrodynamic loads and acoustic resonance during preservice testing or IST.

The NRC response to the comments related to 10 CFR 50.55a(b)(3)(iii)(A) describes the basis for and applicability of the provisions in 10 CFR 50.55a(b)(3)(iii).

In the final rule, 10 CFR 50.55a(b)(3)(iii)(C) states that new reactor licensees shall monitor flow-induced vibration from hydrodynamic loads and acoustic resonance during preservice testing or IST to identify potential adverse flow effects on components within the scope of the IST program. This provision is based on lessons learned from flow-induced vibration adverse effects on plant components during power uprate ascension at operating nuclear power plants. In the statement of considerations for this final rule, the NRC provided guidance for new reactor applicants and licensees to monitor flow-induced vibration from hydrodynamic loads and acoustic resonance during preservice testing or IST.

In the Vogtle Units 3 and 4 final safety evaluation report (FSER), the NRC staff found acceptable the provisions established by the COL applicant for Vogtle Units 3 and 4 in its FSAR to monitor flow-induced vibration from hydrodynamic loads and acoustic resonance during preservice testing or IST. In particular, the NRC staff stated the following in the Vogtle Units 3 and 4 FSER:

AP1000 DCD Tier 2, Section 3.9.2, "Dynamic Testing and Analysis," describes tests to confirm that piping, components, restraints, and supports have been designed to withstand the dynamic effects of steady-state [flow-induced vibration] and anticipated operational transient conditions. Section 14.2.9.1.7, "Expansion, Vibration and Dynamic Effects Testing," in AP1000 DCD Tier 2, Chapter 14, "Initial Test Program," states that the purpose of the expansion, vibration and dynamic effects testing is to verify that safety-related, high energy piping and components are properly installed and supported such that, in addition to other factors, vibrations caused by steady-state or dynamic effects do not result in excessive stress or fatigue to safety-related plant systems. Nuclear power plant operating experience has revealed the potential for adverse flow effects from vibration caused by hydrodynamic loads and acoustic resonance on reactor coolant, steam, and feedwater systems. ... In its response, SNC [Vogtle Units 3 and 4 COL applicant] stated that it intended to use the overall Initial Test

Program to demonstrate that the plant has been constructed as designed and the systems perform consistent with design requirements. SNC referenced the provisions in the AP1000 DCD for vibration monitoring and testing to be implemented at VEGP [Vogtle Electric Generating Plant]. For example, the applicant notes that AP1000 DCD Tier 2, Section 3.9.2.1, "Piping Vibration, Thermal Expansion and Dynamic Effects," specifies that the preoperational test program for ASME BPV Code, Section III, Class 1, 2, and 3 piping systems simulates actual operating modes to demonstrate that components comprising these systems meet functional design requirements and that piping vibrations are within acceptable levels. SNC indicates that the planned vibration testing program described in AP1000 DCD Tier 2, Sections 14.2.9 and 14.2.10, with the preservice and IST programs described in AP1000 DCD Tier 2, Sections 3.9.3.4.4 and 3.9.6, will confirm component installation in accordance with design requirements, and address the effects of steady-state (flow-induced) and transient vibration to ensure the operability of valves and dynamic restraints in the IST Program. The NRC staff considers the response by SNC clarifies its application of the provisions in the AP1000 DCD to ensure that potential adverse flow effects will be addressed at VEGP. Therefore, the staff considers Standard Content Open Item 3.9-5 to be resolved for the VEGP COL application.

In the statement of considerations for the final rule, the NRC indicates methods, such as described in the Vogtle Units 3 and 4 FSAR, that are acceptable for satisfying 10 CFR 50.55a(b)(3)(iii)(C).

In response to this public comment, the NRC has modified the final rule language to indicate that the flow-induced vibration may be monitored as part of preservice testing or inservice testing.

i. 10 CFR 50.55a(b)(3)(iii)(D)

Comment: The ASME OM Code should not be required to be applied to assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the Regulatory Treatment of Non-Safety Systems (RTNSS) for applicable reactor designs. [7-3; 8-33; 14-25]

NRC Response: The NRC agrees that the IST program or other justified approaches may be applied to assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the RTNSS program for new reactors with passive core cooling systems. As discussed below, this condition is based on Commission policy that is being implemented during the ongoing licensing of new reactors.

Proposed paragraph (D) in 10 CFR 50.55a(b)(3)(iii) states that new reactor licensees shall assess the operational readiness of pumps, valves, and dynamic restraints within the scope of the RTNSS program for applicable reactor designs. This provision is consistent with the Commission policy for RTNSS equipment summarized in the NRC Staff Memorandum dated July 24, 1995, that provided a consolidated list of the approved policy and technical positions associated with RTNSS equipment in passive plant designs discussed in SECY-94-084, SECY-95-132, and their associated SRMs.

The NRC Staff Memorandum dated July 24, 1995, summarizes the Commission's policy positions related to IST of RTNSS pumps and valves as follows:

The staff also concluded that additional inservice testing requirements may be necessary for certain pumps and valves in passive plant designs. The unique passive plant design relies significantly on passive safety systems, but also depends on non-safety systems (which are traditionally safety-related systems in current light-water reactors) to prevent challenges to passive systems. Therefore, the reliable performance of individual components is a very significant factor in enhancing the safety of passive plant design. The staff recommends that the following provisions be applied to passive ALWR plants to ensure reliable component performance.

1. Important non-safety-related components are not required to meet criteria similar to safety-grade criteria. However, the non-safety-related piping systems with functions that have been identified as being important by the RTNSS process should be designed to accommodate testing of pumps and valves to assure that the components meet their intended functions. Specific positions on the inservice testing requirements for those components will be determined as a part of the staff's review of plant-specific implementation of the regulatory treatment of non-safety systems for passive reactor designs.
2. ...The vendors for advanced passive reactors, for which the final designs are not complete, have sufficient time to include provisions in their piping system designs to allow testing at power. Quarterly testing is the base testing frequency in the Code and the original intent of the Code. Furthermore, the COL holder may need to test more frequently than during cold shutdowns or at every refueling outage to ensure that the reliable performance of components is commensurate with the importance of the safety functions to be performed and with system reliability goals. Therefore, to the extent practicable, the passive ALWR piping systems should be designed to accommodate the applicable Code requirements for the quarterly testing of valves. However, design configuration changes to accommodate Code-required quarterly testing should be done only if the benefits of the test outweigh the potential risk.
3. The passive system designs should incorporate provisions (1) to permit all critical check valves to be tested for performance, to the extent practicable, in both forward- and reverse-flow directions, although the demonstration of a non-safety direction test need not be as rigorous as the corresponding safety direction test, and (2) to verify the movement of each check valve's obturator during inservice testing by observing a direct instrumentation indication of the valve position such as a position indicator or by using nonintrusive test methods.
4. ...Similarly, to the extent practicable, the design of non-safety-related piping systems with functions under design-basis condition that have been identified as being important by the RTNSS process should incorporate provisions to periodically test power-operated valves in the system during operations to assure that the valves meet their intended functions under design-basis conditions.

5. ...Mispositioning may occur through actions taken locally (manual or electrical), at a motor control center, or in the control room, and includes deliberate changes of valve position to perform surveillance testing. The staff will determine if and the extent to which this concept should be applied to MOVs in important non-safety-related systems when the staff reviews the implementation of the regulatory treatment of non-safety systems.

Consistent with the Commission policy for RTNSS equipment, 10 CFR 50.55a(b)(3)(iii)(D) of this final rule specifies that new reactor licensees shall assess the operational readiness of pumps, valves, and dynamic restraints within the RTNSS scope. This regulatory requirement will allow licensees flexibility in developing programs to assess operational readiness of RTNSS components that satisfy the Commission policy. Guidance on the implementation of the Commission policy for RTNSS equipment is set forth in NRC Inspection Procedure 73758, "Part 52, Functional Design and Qualification, and Preservice and Inservice Testing Programs for Pumps, Valves and Dynamic Restraints," dated April 19, 2013.

As noted by the public commenters, ASME is preparing guidance for new reactor licensees to use in developing programs for the treatment of RTNSS equipment. The NRC staff is participating on the ASME OM Code committees to assist in developing guidance for the treatment of RTNSS equipment that is consistent with Commission policy.

In Section II.C of the statement of considerations for the final rule, the NRC describes the Commission policy related to RTNSS equipment to satisfy 10 CFR 50.55a(b)(3)(iii)(D).

The NRC made no change to the final rule as a result of these comments.

j. 10 CFR 50.55a(b)(3)(iv)

Comment: Based on the new requirements or modifications for Appendix II, "Check Valve Condition Monitoring Program," the proposed changes may still allow a group of four valves to be tested at the same time on a 16-year interval while testing each valve at an approximate equal interval (16 years) not to exceed the maximum interval. As an example, a licensee could test a group of four valves every fourth refueling outage, at the same time, based on a 16-year interval and 2-year fuel cycle and still be in compliance with the NRC's changes. Was the intent of this condition to require one valve from each group be inspected individually, with the remaining members of the group inspected in equal increments of the overall interval? In other words, a group of four valves on a 16-year interval, per Table II-4000-1, should have one valve from the group inspected every 4 years, on a staggered basis, based on a 2-year fuel cycle, i.e., A-B-C-D-A-B-C-D. [5-9]

NRC Response: The NRC agrees with the public commenter that the clarifications added to 10 CFR 50.55a(b)(3)(iv) could be misinterpreted. These clarifications are intended to ensure that licensees understand the purpose of Appendix II to allow sampling of check valves on a periodic basis such that all check valves in a group are evaluated over a maximum test or examination interval. Therefore, the NRC has revised the condition with the following clarification:

Trending and evaluation shall support the determination that the valve or group of valves is capable of performing its intended function(s) over the entire interval. At least one of the Appendix II condition monitoring activities for a valve group

shall be performed on each valve of the group at approximate equal intervals not to exceed the maximum interval shown in the following table:

Maximum Intervals for Use When Applying Interval Extensions		
Group Size	Maximum interval between activities of member valves in the groups (Years)	Maximum interval between activities of each valve in the group (Years)
≥4	4.5	16
3	4.5	12
2	6	12
1	Not applicable	10

The conditions specified for the use of Appendix II, 1995 Edition with the 1996 and 1997 Addenda and 1998 Edition through the 2002 Addenda, of the ASME OM Code in 10 CFR 50.55a have not been revised by this rulemaking.

Comment: The 2004 edition through the 2012 edition should not be included as a condition since the changes required by the regulators regarding check valve condition monitoring have been incorporated into the Subsection ISTC, Mandatory Appendix II, approved and incorporated by reference into 10 CFR 50.55a, with the 2001 Edition/2003 addenda and later.

Through this proposed condition, it appears that the NRC is interpreting the ASME OM Code in a manner inconsistent with its intent. The NRC is encouraged to seek clarifications through ASME's inquiry or revision process. Also, ASME record #14-12 has already addressed these concerns, so the NRC should withdraw these "clarifications."

This statement is confusing and may contradict other sections of 10 CFR 50.55a. For instance, per 10 CFR 50.55a(f)(4)(ii), for successive 120-month intervals, licensees are required to update to 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. However, the draft rulemaking implies that the licensee would have to update to a later edition and addenda of Appendix II of the OM Code every time the NRC incorporates later editions and addenda of the code. The NRC should clarify whether this is the intent of this condition.

This condition should be modified such that it does not apply to the 2004 Edition through the 2012 Edition. [8-34; 14-26; 21-7; 23-1]

NRC Response: The NRC disagrees with the assertion that the NRC does not need to include a clarification about the 2004 Edition through the 2012 Edition of the ASME OM Code. Appendix II was added to the ASME OM Code in the ASME OMa 1996 Addenda, which was incorporated by reference into the *Code of Federal Regulations* with conditions in 1999 (64 FR 51370; September 22, 1999). One condition, 10 CFR 50.55a(b)(3)(iv)(B), required that test intervals shall not be greater than 10 years. This condition is applicable to licensees currently testing under the requirements of 1995 Edition with 1996 and 1997 Addenda to the 1998 Edition through the 2002 Addenda to the ASME OM Code. After several years of obtaining test data, the industry desired to increase the test interval beyond 10 years. The NRC staff worked with the ASME OM Code committees and arrived at an acceptable solution. Beginning with the updated ASME OMb 2003 Addenda, Appendix II to the ASME OM Code was revised to include a table that specifies test interval requirements for valves and groups of

valves. However, the table is not clear in the ASME OM Code, including the 2004 Edition through 2012 Edition, because it could be interpreted to allow no monitoring of the check valve over the entire 10-year interval. The review of test programs has confirmed that the requirement is not being applied consistent with the intent of Appendix II. The NRC staff is aware that the ASME OM Code committees have added a clarification to the ASME OM Code, Appendix II, addressing this issue that is currently scheduled to be included in the 2017 Edition. Incorporation into the *Code of Federal Regulations* will occur a few years after the 2017 Edition of the ASME OM Code is published.

Further, the NRC agrees with the commenters that the statement in the proposed rule with respect to updating to a later edition and addenda of Appendix II was confusing. Therefore, the NRC removed the following statement from the final rule: "The NRC notes that ASME has provided additional improvements to Appendix II since issuance of the 2003 Addenda. Therefore, where a licensee plans to voluntarily implement Appendix II to the ASME OM Code, the licensee should apply Appendix II in the most recent addenda and edition of ASME OM Code incorporated by reference in § 50.55a."

k. 10 CFR 50.55a(b)(3)(vii)

Comment: The proposed draft rulemaking background information should be clarified that it was not an oversight that the pump periodic verification test was not added to the 2011 Addenda of the ASME OM Code. The revised upper limit for the comprehensive pump test was ultimately a separate ballot from the pump periodic verification test. The revised upper limit was approved by Board on Nuclear Codes and Standards ballot 10-1356, which was closed on July 14, 2010. This revision was approved in time to be published in the 2011 Addenda. The pump periodic verification test code revision was approved by ballot 11-2801, in which voting ended on December 19, 2011. This approval was obtained in time to be published in the 2012 Edition of the ASME OM Code. [8-35]

NRC Response: The NRC agrees with this comment that the omission of the pump periodic verification test from the ASME OM Code update did not reflect an oversight.

The NRC has revised the statement of considerations for the final rule to remove the assertion that the pump periodic verification test was not added to the 2011 Addenda to the ASME OM Code was the result of an oversight by ASME.

l. 10 CFR 50.55a(b)(3)(viii)

Comment: ASME Subsection ISTE is working to resolve NRC concerns so that endorsement of Subsection ISTE may be possible in the near future. [8-36]

NRC Response: No response is necessary.

m. 10 CFR 50.55a(b)(3)(x)

Comment: OM Code Case OMN-20 should be adopted, as it provides a resolution to the Task Interface Agreement issue identified regarding the use of TS Section 3.0.2 regarding "Grace Period." [8-37]

NRC Response: No response is necessary.

n. 10 CFR 50.55a(b)(3)(xi)

Comment: The NRC proposes to add 10 CFR 50.55a(b)(3)(xi) to require that licensees supplement the ASME OM Code provisions in Subsection ISTC-3700, "Position Verification Testing," as necessary to verify that valve operation is accurately indicated. ASME OM Code, Subsection ISTC-3700, requires valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated.

- (1) Because of the significance of implementing the condition, as stated, for 10 CFR 50.55a(b)(3)(xi), some licensees may need time to revise or create procedures to govern the implementation requirements for this "condition."*
- (2) What allowances will be given by the NRC for compliance, and does this "condition" apply to "all licensees" whether they are updating to the 2012 Edition, or does it only apply to plants "when" they update to the 2012 Edition of the ASME OM Code? If it is the NRC's intent to implement this condition in the licensee's current interval, there appears to be inadequate justification for such a radical change, and there are inadequate provisions for implementation considering the level of effort to implement and the impact to station activities.*
- (3) Because of the significance of implementing the condition of 10 CFR 50.55a(b)(3)(xi) and that some licensees may have already started their update process (see comment on 10 CFR 50.55a(f)(4)(ii) and 10 CFR 50.55a(g)(4)(ii)), for licensees that will be updating to the 2012 Edition of the ASME OM Code within 24 months of the rulemaking, it seems appropriate to allow additional time to meet the condition. It is suggested that licensees updating to the 2012 Edition of the ASME OM Code within 24 months of the rulemaking be given an additional 24 months to implement the condition of 10 CFR 50.55a(b)(3)(xi) or identify, prepare, and submit any necessary requests for alternatives in accordance with 10 CFR 50.55a(z).*
- (4) For passive valves, the condition of 10 CFR 50.55a(b)(3)(xi) will require verification of obturator movement while performing ISTC-3700 using flow, level indication, or temperature, etc., other than lights. Many systems contain passive valves that are out of service during refueling outages, which is when these tests are typically performed. The condition of 10 CFR 50.55a(b)(3)(xi) may require the system to be in service during testing and could affect normal operations of the plant, require abnormal system alignment or operation, and result in additional radiation exposure. Based on the industry's experience with incidents of stem and disc separation in passive valves, this added burden seems excessive without a compensating increase in safety. Operating experience from the failure at Browns Ferry Nuclear Plant, and other plants, indicates the failure is attributed to high flow conditions over long periods where the disc separated from the stem while the valve was being used to throttle flow. This failure has not been attributed to normally closed valves that are only opened under administrative control, as is the service condition for most passive valves, where the ISTC-3700 requirements would also apply in this case. Therefore, the proposed condition should be limited to active valves and excluded from passive valves. [5-8]*

NRC Response: The NRC partially agrees and partially disagrees with the assertions in this comment, as explained below.

- (1) The NRC agrees that additional time for implementation of the condition on valve position verification is appropriate. Therefore, the NRC has revised the condition to allow additional implementation time, as discussed below.
- (2) The NRC agrees that the implementation date of this condition should be clarified. The NRC has revised this condition to indicate that it is associated with implementation of the 2012 Edition of the ASME OM Code. As such, nuclear power plant licensees will be required to implement the condition when adopting the 2012 Edition of the ASME OM Code as their code of record for the 120-month IST interval.
- (3) The NRC disagrees that licensees should be given additional time, as proposed in the comments, to comply with this condition. Licensees that determine that they will need additional time to implement the 2012 Edition of the ASME OM Code, including the condition on valve position indication specified in 10 CFR 50.55a, may submit requests for alternatives in accordance with 10 CFR 50.55a(z).

The NRC made no change to the final rule as a result of this comment.

- (4) The NRC does not agree with the commenters that the condition should be limited to active MOVs. Passive valves require periodic verification of position indication. For example, operating experience from the valve failure at Browns Ferry referenced by the commenter involved a valve that had been classified as passive.

The NRC made no change to the final rule as a result of this comment.

Comment: ISTOG also has some concern about implementation requirements/costs for 10 CFR 50.55a(b)(3)(xi) (new), which deals with new requirements for position indication verification following the Browns Ferry MOV finding:

“OM condition: Valve Position Indication.” When implementing ASME OM Code, Subsection ISTC-3700, ‘Position Verification Testing,’ licensees shall develop and implement a method to verify that valve operation is accurately indicated by supplementing valve position indicating lights with other indications, such as flow meters or other suitable instrumentation, to provide assurance of proper obturator position.”

This will also require extensive effort by the licensee to ensure that testing of existing IST valves meets this new requirement. Any new IST SSCs that are added to the Program per the change to 10 CFR 50.55a(f)(4) will also need to meet this requirement. This may result in installation of new test connections or new procedures. [7-2]

NRC Response: The NRC disagrees with this comment because the ASME OM Code requires licensees to verify the accuracy of the remote position indication for all valves in the IST program that have remote position indication. Subsection ISTC-3700 states that where local observation is not possible, licensees shall use other indications to verify operation. Nuclear power plant operating experience has revealed that reliance on indicating lights and stem travel are not sufficient to satisfy the requirement in ISTC-3700 to verify that valve operation is accurately indicated for those valves where the integrity of the internal mechanism of the valve (such as the stem-to-disk connection) cannot be assured. Criterion V, “Instructions, Procedures, and Drawings,” of Appendix B to 10 CFR Part 50 requires that safety-related components that are subjected to test activities have appropriate instructions, procedures, or drawings and qualitative or quantitative acceptance criteria for determining that activities have

been successfully completed. Therefore, licensees are responsible for developing a method to verify that valve operation is accurately indicated to satisfy ISTC-3700 requirements.

As indicated in the response to Comment 5-8, the NRC has revised the condition in 10 CFR 50.55a for valve position verification to indicate that it will apply with the implementation of the 2012 Edition of the ASME OM Code for the 120-month IST interval to allow additional time for licensees to address this condition.

Comment: This is a new concern associated with the potential “stem to disk separation” of IST valves. The major burden here is the “shall statement” regarding the implementation of supplemental methods to verify obturator position and movement. The ASME Subsection ISTC is working to change the Code to alleviate the regulatory concern associated with the determination of obturator position or movement using ONLY stem position, especially in harsh or corrosive environments.

For these reasons, the NRC should remove this proposed condition from the final rule. [8-38; 8-42; 21-8; 22-1; 23-2]

NRC Response: The NRC disagrees with these comments because this condition is necessary to emphasize the current ASME OM Code requirements. The NRC recognizes that the ASME OM Code, Subsection ISTC, committee has been attempting to improve the ASME OM Code since the Browns Ferry valve failure occurred more than 5 years ago. The NRC staff has been working with the ASME OM Code committees to improve the ASME OM Code provisions to address this issue. However, after 5 years, the NRC does not believe there is a consensus on the ASME OM Code improvement. Moreover, the NRC does not believe there is a clear path on how consensus might be reached. If the ASME OM Code is revised to resolve this issue, then the NRC will evaluate whether the condition on valve position verification may be removed from 10 CFR 50.55a.

As indicated in the response to Comment 5-8, the NRC has revised the condition in 10 CFR 50.55a for valve position verification to indicate that it will apply with the implementation of the 2012 Edition of the ASME OM Code for the 120-month IST interval to allow additional time for licensees to address this condition.

Comment: The background information for the proposed rule with respect to ISTC-3700 indicates that this is only a “clarification of the intent of the existing ASME OM Code.” This statement is misleading and incorrect. The NRC is encouraged to seek clarifications through ASME’s inquiry or revision process. The existing code does not require supplemental indications to be performed with all position indication testing. This was confirmed through ASME OM Code Interpretation 12-01, which is consistent with how the industry approaches this testing. This NRC “clarification” of the code would result in a very significant new requirement for licensees. Finally, based on the NRC’s Backfit Rule, this “clarification” appears to be a new or different regulatory position that would require a backfit analysis.

For the reasons detailed above, the NRC should remove this proposed condition from the final rule. [8-39; 25-17]

NRC Response: The NRC partially agrees and partially disagrees with the assertions in this comment. ISTC-3530, “Valve Obturator Movement,” allows obturator movement to be determined by indicating lights in the control room when exercising a valve. The valve position verification testing required by ISTC-3700 provides confirmation on a 2-year frequency that the indicating lights reflect the actual valve operation. The proposed condition recognizes that the

vast majority of valves have no provision for verifying the obturator position by direct observation when implementing ISTC-3700 and, therefore, supplemental methods must be used. This long-held NRC position has been documented in multiple revisions of NUREG-1482, "Guidelines for Inservice Testing at Nuclear Power Plants," beginning in 1995, and is consistent with the ASME OM Code definition of exercising. For these reasons, the NRC disagrees with the assertion in the comment that the condition reflects a new regulatory position.

The NRC agrees with the comment that the condition is not clarifying the current requirements for valve position indication in the ASME OM Code; rather, the condition is emphasizing those requirements. Therefore, the NRC modified the discussion in the final rule to remove the description of this condition as a clarification of the ASME OM Code requirements and to indicate that the condition is emphasizing the ASME OM Code requirements. In addition, the NRC expanded the discussion in the final rule to indicate that ISTC-3700 allows flexibility to licensees in verifying that operation of valves with remote position indicators is accurately indicated. For example, NUREG-1482 (Revision 2) in paragraph 4.2.7, "Verification of Remote Position Indication for Valves by Methods Other Than Direct Observation," refers to various methods to verify valve operation, such as nonintrusive techniques, flow initiation or absence of flow, leak testing, and pressure testing. The extent of verification necessary for valve operation to satisfy ISTC-3700 will depend on the type of valve, the sophistication of the diagnostic equipment used in testing the valve, possible failure modes of the valve, and the operating history of the valve and similar valve types. To satisfy ISTC-3700, the licensee is responsible for developing and implementing a method to provide reasonable assurance that valve operation is accurately indicated.

In response to this comment, the NRC simplified the condition to specify that when implementing ASME OM Code, 2012 Edition, Subsection ISTC-3700, licensees shall verify that valve operation is accurately indicated by supplementing valve position indication lights with other indications, such as flow meters or other suitable instrumentation, to provide assurance of proper obturator position.

As indicated in the response to Comment 5-8, the NRC has revised the condition in 10 CFR 50.55a for valve position verification to indicate that it will apply with the implementation of the 2012 Edition of the ASME OM Code for the 120-month IST interval to allow additional time for licensees to address this condition.

Comment: To impose this condition on every IST component with a position indication test would be overly burdensome to the licensees with little to no benefit in return. Reviewing one plant's program resulted in 214 components that require a position indication test. In order to implement this proposal, a total of 428 valve positions (214 open and 214 closed) would have to be validated through other supplementary methods every 2 years. Implementing this proposal would take the licensee several months in order to research the proper test methods, revise procedures, schedule the new testing, and submit new relief requests, as necessary. The additional testing requirements may lengthen the licensee's outage durations and increase personnel dose. The initial and follow-up costs to implement this proposal would be very significant throughout the entire life of the plant. With this in mind, no studies have concluded that imposing this additional testing would result in any safety benefits in return. On the contrary, however, an extensive study of MOV failure data over the last 30 years, performed by the MOV subgroup, concluded that disc/stem separation events are rare and occur approximately only once per year throughout the industry. Of these failures, 80–90 percent of them were identified at or near the time of failure under normal plant processes and procedures.

Reference the white paper for ASME code change record 14-877. Therefore, ASME does not understand the justification for imposing this new condition.

For the reasons detailed above, the NRC should remove this proposed condition from the final rule. [8-40; 8-41; 12-4; 12-5; 14-27]

NRC Response: The NRC disagrees with this comment. The ASME OM Code in ISTC-3700 requires verification of valve position indication every 2 years. In the 10 CFR 50.55a condition, the NRC is reminding licensees of the ASME OM Code requirements in ISTC-3700. The NRC has emphasized these Code requirements for valve position indication in ISTC-3700 for over 20 years. For example, the NRC discussed these requirements in ISTC-3700 for valve position indication in the initial issuance of NUREG-1482 in 1995. As indicated in NUREG-1482, licensees have flexibility in satisfying the valve position indication requirements in ISTC-3700 based on a wide variety of system indications. Where a licensee has been implementing ISTC-3700, there will be no additional resources associated with this condition in 10 CFR 50.55a. If necessary, a licensee may submit a relief request where the ISTC-3700 requirements result in a hardship without a compensating increase in safety. The NRC has relaxed the implementation schedule for this 10 CFR 50.55a condition to allow additional time for licensees to verify that their IST program satisfies the valve position indication requirements in ISTC-3700. This 10 CFR 50.55a condition is not a backfit because the condition does not alter the ASME OM Code requirements for valve position indication in ISTC-3700 as known for many years.

The NRC made no change to the final rule as a result of the comment.

Comment: The NRC should not adopt the proposed rule's condition on ISTC-3700 because it goes against the recognized authority of the OM Code interpretation and change processes. The ASME OM Code should be revised to detect that the obturator has not been separated from the stem by the ASME OM Code consensus process. [24-2; 24-4; 24-10; 24-14]

NRC Response: The NRC disagrees that the condition reflects a new interpretation of ISTC-3700 by the NRC that goes against ASME authority. ISTC-3700 requires, on a 2-year frequency, that the indicating lights reflect the actual valve operation. The NRC alerted licensees to this fact in multiple revisions of NUREG-1482 beginning in 1995. In addition, ISTC-3700 is consistent with the ASME OM Code definition in ISTA-2000 that exercising is a demonstration based on direct visual or indirect positive indications that the moving parts of a component function. The NRC does agree that future ASME OM Code changes could improve the requirements to identify stem separation.

The NRC made no change to the final rule as a result of these comments.

Comment: ISTA-1100 of the ASME OM Code states, "Section IST establishes the requirements for preservice and inservice testing and examination of certain components to assess their operational readiness." IST requirements do not verify operability but provide a measure of reasonable assurance of the ability of the component to perform its intended function. It appears as though the proposed rulemaking approach is using a provision of the code for another purpose without recognizing other more efficient and targeted techniques. Normal plant processes may provide some means of verifying that the obturator is attached to the valve stem for a portion of the population of power-operated valves. Using these processes as a new part of the IST program requirement to create a new program element extends the purpose of IST beyond what was intended. The new feature, especially when this new program feature has to

be extended to the valves that are not currently subject to this verification process, is a burden to plants. [24-3; 24-6; 24-11; 24-13; 24-16]

NRC Response: The NRC disagrees with the statement that the IST program only provides a “measure” of reasonable assurance of the ability of the valve to perform its intended function. The ASME OM Code as incorporated by reference in 10 CFR 50.55a requires implementation of Section IST of the ASME OM Code which establishes requirements for preservice and inservice testing and examination of pumps, valves, and snubbers to assess their operational readiness. In part by ISTC-3700, the operational readiness of a valve is assessed by verifying that the valve will actually open and close. Unless the actuator is disassembled to inspect the obturator, the open/close function of the valve is determined through monitoring system parameters that reflect the change in obturator position. The ASME OM Code allows valves that operate during plant operation at a frequency that would satisfy the ASME OM Code exercising requirements to not be additionally exercised (see ISTC-3550). The use of normal plant processes to verify valve operation is consistent with existing ASME OM Code provisions. Therefore, licensees may apply system information from normal plant processes to satisfy ISTC-3700 as discussed in NUREG-1482.

Licensees could apply lessons learned from check valve exercising to satisfy the ISTC-3700 requirement. Prevention, detection, and correction lessons learned can also be used to help satisfy ISTC-3700. In addition, lessons learned help identify valves that are susceptible.

The NRC made no change to the final rule as a result of these comments.

Comment: Publication of the proposed rule “as-is” would place all of the Owner Inservice Testing Programs in immediate noncompliance that is not readily resolved within the 30-day timeframe. Using a level of effort of in-house staff, it is estimated that it would take upwards of 12 months per unit to incorporate these changes, with an additional 6 to 12 months to validate the added testing activities. Many of the test activities will be performed during a refueling outage. It is expected that plants will also require relief from this testing. This multi-person group would include procedure writers, operations, engineering, work management, outage planning, licensing and schedulers to “credit” those valves that can be bi-directionally tested and to then develop activities for those valves that require new means for one or both directions. This magnitude of a supplemental indications change as outlined in the proposed rulemaking should be coordinated with the plant's 10 Year Program Update as opposed to implementation within 30 days of the publication of the final rule. [24-5; 24-7; 24-12; 24-15; 24-17]

NRC Response: The NRC agrees with these comments. The NRC revised the condition for valve position verification in the final rule to indicate that it will apply with the implementation of the 2012 Edition of the ASME OM Code for the 120-month IST interval to allow time for licensees to address this condition. Licensees that need additional time may submit requests for alternatives in accordance with 10 CFR 50.55a(z).

VI. Inservice Testing

a. 10 CFR 50.55a(f)

Comment: The proposed change to address a previous weakness in the rule where preservice testing was not specifically addressed is welcome, but referring to preservice and inservice testing collectively as inservice testing may cause confusion. In addition, more should be done to clarify the applicable code of record for preservice testing—perhaps in (f)(1) through (3). [21-9]

NRC Response: The NRC agrees with the comment that 10 CFR 50.55a needed to be clarified to include preservice testing as part of the IST program. In the proposed rule, the NRC proposed revising the first sentence of 10 CFR 50.55a(f) to specify that systems and components of boiling and pressurized water-cooled nuclear power reactors must meet the requirements for preservice and inservice testing (referred to collectively as inservice testing) of the ASME BPV Code and ASME OM Code. In the statement of considerations for the proposed rule, the NRC explained that the proposed change clarifies that the ASME OM Code includes provisions for preservice testing of components as part of its overall provisions for IST programs (80 FR 56834; September 18, 2015). The NRC noted that no expansion of the IST program scope is intended by this clarification. The NRC agrees with the comment that the clarification addresses a previous weakness in the regulatory language in 10 CFR 50.55a(f).

Regarding the suggestion to revise the subparagraphs in 10 CFR 50.55a(f) to address the applicable code of record for preservice testing, the NRC notes that the other subparagraphs in 10 CFR 50.55a(f) refer to the code of record for nuclear power plants constructed over various time periods. The NRC believes that nuclear power plant applicants and licensees have typically established the same code of record for both the preservice testing and IST programs for the pumps and valves in their nuclear power plants. Therefore, the NRC is concerned that addressing each code of record for the preservice testing and IST programs individually in the subparagraphs of 10 CFR 50.55a(f) might cause confusion among the users of 10 CFR 50.55a. The NRC has determined that any instances in which a nuclear power plant applicant or licensee might select a different code of record for its preservice testing and IST programs can be addressed on a plant-specific basis. Therefore, the NRC considers that the proposed change to the first sentence of 10 CFR 50.55a(f) will provide appropriate clarification of the preservice and inservice testing requirements that are collectively addressed as inservice testing in 10 CFR 50.55a(f) and has retained that language in the final rule.

The NRC made no change to the rule as a result of this comment.

b. 10 CFR 50.55a(f)(4)

Comment: It is unclear whether the removal of "ASME Code Class 1, Class 2, and Class 3" from this paragraph is intended to expand the scope of this paragraph to also apply to components other than Class 1, 2, or 3. The NRC should clarify that the proposed change is not intended to expand the scope of this paragraph to include pumps and valves other than Class 1, 2, or 3. Expanding the scope of this paragraph would have a significant impact on Licensees. Components other than Class 1, 2, and 3 that meet the scope of ISTA-1100 or have been given some safety significance by the plant have typically been treated as augmented IST by owners or have been tested in a manner (outside of the IST program) that is commensurate with their safety function. Increasing the scope of this paragraph would require licensees to perform some or all of the following:

- *Reevaluate the scope of the components subject to OM Code requirements.*
- *Update IST program documents and procedures.*
- *Seek relief for components that cannot fully comply with these new requirements.*

A backfit analysis would be required if the scope of this paragraph is extended to include components other than Class 1, 2, and 3. The NRC should revise this condition to clarify that it applies only to Class 1, 2, and 3 components. [7-1; 8-43; 12-6; 14-28; 21-10; 22-2; 23-3; 24-1; 24-9; 25-18]

NRC Response: The NRC agrees with the comments that the revision to 10 CFR 50.55a(f)(4) should be clarified to avoid an unintended paperwork burden from the alignment of the scopes of the ASME OM Code and 10 CFR 50.55a.

The intent of the revision to 10 CFR 50.55a(f)(4) is to align the scope of the IST program described in 10 CFR 50.55a with the scope of the ASME OM Code for testing of pumps and valves that are required to perform a specific function in shutting down a reactor to the safe-shutdown condition, in maintaining the safe-shutdown condition, or in mitigating the consequences of an accident.

The concern raised by public comments for aligning the scopes of the ASME OM Code and 10 CFR 50.55a relates to a potential paperwork burden for the submittal of relief or alternative requests for safety-related pumps and valves that are not classified as ASME Code Class 1, 2, or 3 components. The NRC did not intend that the alignment of the scopes of the ASME OM Code and 10 CFR 50.55a would cause a paperwork burden.

Therefore, the NRC has included the following additional statement in 10 CFR 50.55a(f)(4):

The inservice test requirements for pumps and valves that are within the scope of the ASME OM Code but are not classified as ASME BPV Code Class 1, Class 2, or Class 3 may be satisfied as an augmented IST program in accordance with paragraph (f)(6)(ii) without requesting relief under paragraph (f)(5) or alternatives under paragraph (z) of this section. This use of an augmented IST program may be acceptable provided the basis for deviations from the ASME OM Code, as incorporated by reference in this section, demonstrates an acceptable level of quality and safety, or that implementing the Code provisions would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, where documented and available for NRC review.

During the public meeting on March 2, 2016, commenters indicated that this additional provision in 10 CFR 50.55a(f)(4) resolves their comments.

Comment: Section 50.55a(f)(4)(i), (f)(4)(ii), (g)(4)(i), and (g)(4)(ii) should be revised from 12 months before the start of the 120-month interval to 24 months before the start of the 120-month interval. [5-11]

NRC Response: The NRC considers this matter to be outside the scope of this rulemaking, as the proposed rule contains no change to this longstanding requirement. However, the NRC will consider addressing this subject in a future rulemaking.

The NRC made no change to the final rule as a result of these comments.

VII. Inservice Inspection

a. 10 CFR 50.55a(g)(1)

Comment: Many plants have used 10 CFR 50.55a(c)(2)(i) and RG 1.26 as guidance for classifying Class 1, 2, and 3 components. However, NRC regional inspector input has indicated that 10 CFR 50.55a(c)(2)(i) may not be used as guidance for determining inservice inspection Class 1 components, and a recent NRC non-cited violation (ADAMS Accession No. ML15218A371) indicates that the NRC may be taking a position that all safety-related pressure vessels, piping, pumps and valves, and their supports are required to be classified as

Class 2 or 3 regardless of function or classification guidance provided in RG 1.26. These perceived NRC positions would result in older plants being required to include systems and segments of systems in the Section XI inservice inspection program that are not required for newer plants and would essentially negate use of the licensing basis as a tool for classifying components as required by IWA-1400. The NRC should clarify the language in 10 CFR 50.55a(g)(1). [25-19]

NRC Response: The NRC disagrees with this comment. The NRC position has not changed on this requirement, and a rule change to address the perceived position change is not justified. The component classification, as defined by the plant licensing basis, remains valid. The guidance in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," offers an acceptable means of system classification to be incorporated into the inservice inspection program. A licensee may also opt to deviate from the guidance in the RG by means of a license amendment.

The NRC made no change to the final rule as a result of this comment.

b. 10 CFR 50.55a(g)(2)

Comment: For plants whose construction permit was issued on or after January 1, 1971, but before the effective date of this final rule, this provision applies only to components affected by repair/replacement activities. As such, the provisions in 10 CFR 50.55a(g)(2)(i) and (ii) are not necessary because ASME BPV Code Section XI specifies requirements for owners to provide component accessibility for examinations and tests. The proposed requirements of 10 CFR 50.55a(g)(2)(i) and (ii) should be removed. Alternatively, these requirements could be revised to clarify that they apply only to design and access of components affected by repair/replacement activities. If it is the intent of this revised condition to apply retroactively to plants that are already constructed, then the proposed change should be evaluated as a backfit. [8-44; 8-45; 14-29; 14-30; 25-20]

NRC Response: The NRC disagrees with the comments that the requirements of 10 CFR 50.55a(g)(2)(i) and (ii) should be removed. The requirements of 10 CFR 50.55a(g)(2)(i) and (ii) clarify the level of accessibility that must be designed into and provided for the subject components throughout the lifetime of the plant. The NRC agrees with the comments in that these provisions also apply to components affected by repair and replacement activities, and this same level of accessibility must be designed into and provided for the subject components unless changes are allowed by 10 CFR 50.55a(g)(2)(iii). While Section XI, IWA-1400(b), requires that the owner's responsibility include "design and arrangement of system components to include allowances for adequate access and clearances for conduct of the examination and tests," it is not clear to the NRC that Section XI requires the original level of accessibility to be maintained. The proposed changes to 10 CFR 50.55a(g)(2)(i) and (ii) are not a change in requirements; the changes merely place the requirements for accessibility for inservice and preservice examinations for all plants with a construction permit issued in 1971 or later under paragraph 10 CFR 50.55a(g)(2). Therefore, it is the NRC's position that the changes in the language of 10 CFR 50.55a(g)(2)(i) and (ii) are not a change in NRC position or requirement and do not fall within the definition of backfitting.

The NRC made no change to the final rule as a result of these comments.

c. 10 CFR 50.55a(g)(3)

Comment: For plants whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, this provision applies only to components affected by repair/replacement activities. As such, the provision in 10 CFR 50.55a(g)(3)(i) is not necessary because ASME BPV Code Section XI already specifies requirements for performing preservice inspection. Therefore, this condition should be removed from the final rule. [8-46; 14-31]

For plants whose construction permit was issued after July 1, 1974, but before the effective date of this final rule, this provision will apply only to components affected by repair/replacement activities. As such, the provision in 10 CFR 50.55a(g)(3)(ii) is not necessary for these plants because ASME BPV Code Section XI already specifies requirements for performing preservice inspection. Therefore, this condition should be removed from the final rule. [8-47; 14-32]

NRC Response: The NRC disagrees with these comments. The requirements of 10 CFR 50.55a(g)(3) are only applicable before commercial operation and, therefore, have no effect on the current operating fleet. For example, 10 CFR 50.55a(g)(3)(i) is applicable to Watts Bar Unit 2 (construction permit issued 1973) until startup. Similarly, 10 CFR 50.55a(g)(3)(ii) is applicable to Bellefonte Units 1 and 2 (construction permit issued 1978) and all current 10 CFR Part 52 combined license holders until startup. Once the plant begins commercial operation, repair and replacement activities and their associated preservice exams are considered part of the inservice inspection program and are covered under 10 CFR 50.55a(g)(4), which requires compliance with ASME BPV Code Section XI.

The NRC made no change to the final rule as a result of these comments.

d. 10 CFR 50.55a(g)(4)

Comment: Section 50.55a(g)(4)(i) and (g)(4)(ii) should be revised from 12 months before the start of the 120-month interval to 18 or 24 months before the start of the 120-month interval. [8-48; 25-21; 25-22]

NRC Response: The NRC considers this matter to be outside the scope of this rulemaking, as the proposed rule contains no change to this longstanding requirement. However, the NRC will consider addressing this subject in a future rulemaking.

The NRC made no change to the final rule as a result of these comments.

Comment: Implementing a large number of possible ASME Code Section XI, Appendix VIII, performance demonstration programs will present severe administrative and logistical challenges to licensees and the Performance Demonstration Institute, which is used by licensees to meet the requirements of Appendix VIII testing. Delaying the date for mandatory implementation of Appendix VIII, as contained in the 2009 Addenda through the 2013 Edition of Section XI, for a minimum of 18 months in order to allow time to make all the necessary program and procedure revisions and to communicate these changes to the industry would alleviate these concerns. [6-1]

NRC Response: The NRC agrees with this comment in that it takes time to update and implement a revised Appendix VIII program. To address this practical consideration in implementing the final rule's requirement in this regard, the NRC has added the following sentence to 10 CFR 50.55a(g)(4)(ii): "However, a licensee whose inservice inspection interval commences during the 12 through 18-month period after [the effective date of the final rule],

may delay the update of their Appendix VIII program by up to 18 months after [the effective date of the final rule].”

e. 10 CFR 50.55a(g)(6)(ii)

Comment: A final rule typically becomes effective 30 days after being published in the Federal Register. Implementation of ASME BPV Code Case N-729-4 and conditions would be required by the first refueling outage following the effective date of the final rule. [10-5]

NRC Response: The NRC agrees with this comment. The requirements of ASME BPV Code Case N-729-4 with NRC conditions are implemented during scheduled refueling outages. Although the rule will become effective 30 days after being published in the *Federal Register*, no specific action would be expected until the first scheduled refueling outage begins following the effective date of the final rule.

The NRC made no change to the final rule as a result of this comment.

Comment: The comment states that 10 CFR 50.55a(g)(6)(ii)(D)(3) should not be applied until the first refueling outage at least 6 months after the final rule becomes effective in order to give utilities time to implement the changes (i.e., prepare for the Bare Metal Visual exam). [10-6]

NRC Response: The NRC disagrees with this comment. Proposed condition 10 CFR 50.55a(g)(6)(ii)(D)(3) would only require a bare metal visual examination if a volumetric examination was not being performed. Under the current regulatory requirements, Note 4 of ASME BPV Code Case N-729-1 requires that if a bare metal visual examination was not being performed, then an IWA-2212 VT-2 visual examination of the head, under the insulation through multiple access points, must be performed. This requirement necessitates similar access to the upper head to perform a bare metal visual examination. As stated in response to Comment 10-5, there will be a delay of 30 days after publication of the rule before its implementation, which will provide some flexibility for licensees currently in an outage. Additionally, if a licensee believes this requirement creates a sufficient hardship, it can request relief on a case-by-case basis.

The NRC made no change to the final rule as a result of this comment.

Comment: The proposed condition 10 CFR 50.55a(g)(6)(ii)(D)(3) should not be included in the final rulemaking. To support this request, the comments provided responses to the NRC concerns regarding susceptibility of cold-leg temperature heads to primary water stress-corrosion cracking (PWSCC) and the potential for boric acid corrosion due to leakage through the weld. The comments state that no through-wall cracking has been observed in the United States after the first inservice volumetric or surface examination was performed. Further, there have been relatively few instances of PWSCC of leaking welds that have not been accompanied by PWSCC in the nozzle. The comments explain that approximately only 13 heads are expected to be affected by this proposed condition. Each of these plants has heads that operate at cold-leg temperature, which has a reduction factor of 3.1 to 4.6 versus hot-leg temperature heads. Since bare metal visual examination of the hot-leg temperature heads is required every outage, bare metal visual inspection every 5 years for cold-leg temperature heads is sufficient to address the potential for boric acid corrosion. Additionally, the testing performed by EPRI on the visual evidence of boric acid from a leaking penetration nozzle is sufficient to demonstrate the effectiveness of the required VT-2 examination performed each outage a bare metal visual exam is not performed. The comments state that these points were outlined in Section 5 of MRP-395, “Materials Reliability Program: Reevaluation of

Technical Basis for Inspection of Alloy 600 PWR Reactor Vessel Top Head Nozzles,” which is freely available on www.epri.com. [17-1; 18-1; 20-1; 25-23]

NRC Response: The NRC disagrees with these comments. The need for this proposed condition, as given in the statement of considerations, is a result of the operational experience of recent PWSCC identified in 5 of 20 cold-leg temperature heads in pressurized-water reactors and additional cracking in two bottom head penetration nozzles operating at cold-leg temperature. In one of the five cold-leg temperature heads, new indications of cracking were identified in three subsequent refueling outages. The previous bare metal visual inspection requirement was based on the expectation of a low susceptibility of PWSCC at cold-leg temperature locations. However, operational experience has shown sufficient cracking at cold-leg temperature locations to demonstrate that factors affecting PWSCC susceptibility go beyond the time-at-temperature model that the current inspection program is based on. Additionally, because there is no qualified method to volumetrically inspect the J-groove weld, and no surface examination of the J-groove weld is required, there is no method to ensure the absence of PWSCC in the J-groove weld. As such, it is necessary to perform an effective NDE each outage to identify any potential leakage and, thereby, minimize the impact of any PWSCC in a J-groove weld.

The condition includes a provision to allow extension of the bare metal visual examination requirement for cold-leg temperature heads with no previous indications of PWSCC if a surface examination is performed of the J-groove welds to ensure no PWSCC has initiated. If no indication of cracking is identified, then a hypothetical flaw would have to initiate and grow through-weld to cause leakage. For some J-groove welds in hot-leg temperature heads, this growth through-weld can occur in one operating cycle, hence the bare metal visual examination requirement each outage. Given that basis, use of the cold-leg temperature PWSCC crack growth rate reduction factor of 3.1 to 4.6, as stated by the commenters, will provide reasonable assurance of the structural integrity of the J-groove weld in a cold-leg temperature head until the next volumetric or bare metal visual inspection.

The NRC notes that operational experience also shows several issues with the commenters' basis. There have been indications of cracking through-weld without volumetric indication of the cracking in the nozzle material. This demonstrates that a fully qualified volumetric examination of the nozzle material could miss significant PWSCC in the weld material that could cause leakage during subsequent cycles of operation. While the VT-2 examination would be able to identify large amounts of boric acid on the head, operational experience has shown that indications of leakage can be as small as an aspirin-sized deposit tightly adhering to a nozzle-to-head annulus. Although the EPRI boric acid program showed significant boric acid deposits being visible in only 30-day leakage tests, operational experience has not shown similar deposits of boric acid on heads with leaking penetration nozzles or welds. The reasons for these differences may be due to ventilation or other factors; however, operational experience has shown that the indication of boric acid leakage on an upper head surface is sometimes difficult to identify. For example, even bare metal visual examinations of two leaking upper head nozzle penetrations failed to identify boric acid deposits at one plant. At another plant, regulatory relief from the requirements of completing an effective bare metal visual examination was required. As such, the NRC has greater confidence in the precise requirements of a bare metal visual examination versus a generic VT-2 exam for leakage, given the potential detrimental effects of leakage on this specific component in the reactor coolant system.

Given the available options and the increased occurrence of PWSCC in cold-leg temperature heads, the requirement in this condition, viz., that bare metal visual examinations be performed

for each outage when a volumetric or surface examination is not performed, provides reasonable assurance of structural integrity of the upper head and associated penetration nozzles, while minimizing the effect of any potential boric acid leakage on the head surface. The condition's option to allow a surface examination of the J-groove weld to extend the bare metal visual inspection frequency gives reasonable assurance of the structural integrity and leak-tightness of the upper head.

The NRC made no change to the final rule as a result of these comments.

Comment: If the NRC chooses to impose the acceptance criteria of NB-5352, then it would be clearer if the NRC either specified the year and addenda of Section III or stated that any edition or addenda in (a)(1) is acceptable. If the NRC does not adopt this proposed approach, then the condition should contain the acceptance criteria. [5-13; 8-49]

NRC Response: The NRC agrees with these comments. Although the wording for Paragraph NB-5352 is generally consistent over the years, for clarity the NRC finds including the requirement to use the 2013 Edition of Section III is reasonable. The NRC has changed 10 CFR 50.55a(g)(6)(ii)(D)(4) in the final rule to use the 2013 Edition of Section III to clarify the use of Paragraph NB-5352 as the acceptance criteria for surface examination.

Comment: The comment suggests that rounded indications greater than allowed in Paragraph NB-5352 in size on the partial-penetration or associated fillet weld in Alloy 52/152 materials may be accepted without the need for repair when monitored in the future by a bare metal inspection and evaluated by the Responsible Engineer to confirm leakage has not occurred to date. [14-33]

NRC Response: The NRC disagrees with this comment. The commenter does not provide sufficient technical basis for this proposed alternative. Additionally, the commenter does not define the frequency of bare metal inspection and the type of evaluation. Currently, when alloy 52/152 materials are used to repair individual control rod drive mechanism nozzles and associated J-groove welds, the surface examination acceptance criteria of Paragraph NB-5352 are required to be met.

The NRC made no change to the final rule as a result of this comment.

Comment: The NRC should revise the proposed condition in 10 CFR 50.55a(g)(6)(ii)(D)(4) so that the surface examination acceptance criteria applies only to dye-penetrant surface examinations. The basis to not apply the NB-5352 acceptance criteria to eddy current surface examinations is that the acceptance criteria are at the threshold of eddy current's detection capabilities and give no credit to the subsurface detection capabilities of the eddy current examination technique. [17-2; 18-2]

NRC Response: The NRC disagrees with these comments. Over the past 4 years, the NRC has worked with the ASME Code committees to establish eddy current acceptance criteria for use in conjunction with various versions of ASME BPV Code Cases N-729 and N-770 that differ from the requirements found in Paragraph NB-5352 of Section III of the ASME BPV Code. However, no new requirement was developed to replace these acceptance criteria, and no agreed-upon credit could be gained in evaluating the effectiveness of the subsurface detection capabilities of the eddy current examination technique. This lack of an alternative acceptance criterion specifically for eddy current examination does not alleviate the need to define acceptance criteria for eddy current surface examinations. The minimum detection requirement of a relevant condition of 1/16-inch is within the Appendix IV qualification criteria for eddy

current examinations and, therefore, should be within the technique's detection capabilities. Operational experience has shown that fabrication defects, which may show up only as rounded indications on a surface examination, are potentially high-stress zones that allow the initiation of PWSCC. Therefore, a surface examination technique for these welds should be able to identify relevant linear indications as well as potential rounded indications of sufficient size or location as identified in Paragraph NB-5352, as are the required acceptance criteria for liquid dye penetrant surface examinations. In the future, if acceptance criteria for the eddy current technique that give credit for the detection of subsurface indications are developed and agreed upon, the NRC will consider revision of this condition in future rulemakings.

The NRC made no change to the final rule as a result of these comments.

Comment: The proposed condition 10 CFR 50.55a(g)(6)(ii)(D)(4) appears to be based upon the NRC taking an ASME interpretation out of context and does not consider that all plants have a corrective action program to address such conditions. This proposed condition should be removed. [25-24]

NRC Response: The NRC disagrees with this comment. The condition is sustaining the current requirements for acceptance criteria for surface examinations of rounded indications. The ASME BPV Code interpretation, as discussed in the statement of considerations for the proposed rule, was clear in allowing rounded indications of any size as part of the acceptance criteria for surface examinations. The NRC disagreed with this interpretation during the ASME BPV Code development process. The ASME Code committee passed the interpretation over the NRC's objections. The NRC notes that operational experience has shown that fabrication defects may show up only as rounded indications on a surface examination. These rounded indications can be high-residual-stress zones that could allow initiation of PWSCC. Therefore, a surface examination technique for these welds should be able to identify relevant rounded indications as identified in Paragraph NB-5352, as are the NRC-required acceptance criteria for surface examinations. The NRC expects the licensee to use the corrective action program to address any unacceptable indications in accordance with ASME BPV Code Case N-729-4, as conditioned by the NRC.

The NRC made no change to the final rule as a result of this comment.

Comment: The proposed language in 10 CFR 50.55a(a)(1)(iii)(C) should be revised to incorporate by reference ASME BPV Code Case N-770-3 or N-770-4, in lieu of ASME BPV Code Case N-770-2. [8-51; 14-35]

NRC Response: The NRC disagrees with these comments. Neither ASME BPV Code Case N-770-3 nor N 770-4 was finalized at the start of this rulemaking process. Adoption of either version would require a significant delay in the processing of this rule. The NRC will consider adoption of the latest approved version of ASME BPV Code Case N-770 in the review of the 2015 Edition of Section XI incorporation into 10 CFR 50.55a.

The NRC made no change to the final rule as a result of these comments.

Comment: The proposed condition in 10 CFR 50.55a(g)(6)(ii)(F)(1) should be revised to require implementation of the requirements of ASME BPV Code Case N 770-2 by the first refueling outage starting 18 months following the effective date of the final rule. [8-50; 10-7; 14-34; 19-5; 25-25]

NRC Response: The NRC disagrees with these comments. The NRC considered the impact of delaying the implementation of this change in version from ASME BPV Code Case N-770-1 to N-770-2. Neither the commenter nor the NRC has identified any significant change to the inspection frequencies of the welds to warrant a delay of 18 months. If an individual licensee identifies an issue that cannot be met under the new requirements of ASME BPV Code Case N-770-2, any licensee has the option to propose an alternative under 10 CFR 50.55a(z)(1) and 10 CFR 50.55a(z)(2).

The NRC made no change to the final rule as a result of these comments.

Comment: There are still significant technical issues that need to be addressed before cast austenitic stainless steel can be examined effectively. The proposed date of having a qualification process in place by the year 2020 is considered to be very challenging. The comment requests that the reference to ASME Code Section XI, Appendix VIII, Supplement 9, be removed from the rulemaking in paragraph (g)(6)(ii)(F)(11). [6-5; 8-53; 14-37; 15-3; 21-11; 25-27]

NRC Response: The NRC partially agrees with these comments. Although the inspection of cast austenitic stainless steel can be challenging, advances in ultrasonic examination technology over the past 10 years have significantly improved the ability of ultrasonic inspection techniques to find cracks in cast stainless steel components. The NRC technical position is that performance demonstration testing should be possible using current technology. The NRC agrees that the 2020 implementation date would be challenging and that delaying the implementation date to 2022 in light of the rulemaking timeline would allow sufficient time for the technical and administrative obstacles to be overcome and Supplement 9 to be implemented effectively.

The NRC changed the implementation date to 2022 in the final rule as a result of these comments.

Comment: The specific content being added in 10 CFR 50.55a(g)(6)(ii)(F)(12) appears to be exactly what is required by -2500(b) of the ASME Code Case. [5-15]

NRC Response: The NRC disagrees with this comment. Paragraph -2500(b) of ASME BPV Code Case N-770-2 does not require the implementation of a qualified inspection of cast stainless steel or through the cast stainless steel material. Therefore, the requirements in 10 CFR 50.55a(g)(6)(ii)(F)(12) must be adopted to address cast stainless steel.

The NRC made no change to the final rule as a result of this comment.

Comment: The proposed condition in 10 CFR 50.55a(g)(6)(ii)(F)(13) would require licensees to perform encoded examinations when required to perform volumetric examinations of all non-mitigated and cracked mitigated butt welds in accordance with ASME BPV Code Case N-770-2. This condition is stricter than the "Guideline for Conducting Ultrasonic Examinations of Dissimilar Metal Welds," Revision 1, that includes an NEI-03-08 "needed" element that requires encoding of some examinations of dissimilar metal welds. Specifically, the condition gives less credit to mitigated and cracked components and does not account for configurations that cannot be examined using current encoding technology. This requirement should be dropped, and the NEI-03-08 guidelines on the encoding of dissimilar metal weld examination should be considered sufficient. [9-1; 13-1; 18-3; 25-28]

NRC Response: The NRC disagrees with these comments. Recent experiences with mitigated and unmitigated welds show that human factors issues related to field conditions and deviations from the demonstrated procedures can degrade the effectiveness of non-encoded conventional ultrasonic examination to the point of questioning the full effectiveness of the examination. As ASME BPV Code Case N-770-2 covers welds in Class 1 piping that are susceptible to PWSCC, the NRC determined that a regulatory requirement to use encoding was necessary to give reasonable assurance of public health and safety. Further, the volumetric inspection frequency for mitigated welds with cracking is typically 10 years. A purpose of these examinations is to verify no growth in the original crack and that no new cracking has occurred. Encoded examinations ensure detailed documentation of any indication during the previous volumetric inspection to compare and contrast with the current volumetric inspection data. The purpose of this examination, in part, is to validate the continued effectiveness of the mitigation method to address cracking that was in the reactor coolant pressure boundary.

The NRC made no change to the final rule as a result of these comments.

Comment: The last sentence of the proposed condition 10 CFR 50.55a(g)(6)(ii)(F)(2) involves additional scope in the transition from ASME BPV Code Case N-770-1, as conditioned by the NRC, to ASME BPV Code Case N-770-2, as conditioned in the proposed rule. As such, a backfit analysis should be prepared. [25-26]

NRC Response: The NRC disagrees with this comment. As stated in the section-by-section analysis of the statement of considerations for the proposed rule, the last sentence of the proposed condition is included to clarify the previous NRC position on the use of Paragraph -1100(e) of ASME BPV Code Case N-770-1 or N-770-2 to exempt an ASME Class I butt weld from the categorization process outlined in 10 CFR 50.55a(g)(6)(ii)(F)(2). This is not a new requirement, but rather emphasizes the current requirement. The NRC's current position was discussed in a public meeting on July 12, 2011. A summary of that public meeting is available in ADAMS under Accession No. ML112240818. The purpose the meeting was to discuss pressurized water reactor licensees' implementation of ASME BPV Code Case N-770-1. During that meeting, the public feedback on the condition added by the previous rulemaking (76 FR 36232; June 21, 2011; and correction 77 FR 3073; January 23, 2012) indicated that the wording was not satisfactorily clear on the meaning of this condition. As documented in Questions 26 and 27 in Enclosure 2 of the meeting summary, stakeholders asked if the exemption identified in Paragraph -1100(e) would allow a licensee to exempt welds under those conditions from the inspection program. The NRC response, in accordance with the current rule language, stated that Paragraph -1100(e) does not allow a licensee to override the language of the NRC scoping requirement of 10 CFR 50.55a(g)(6)(ii)(F)(2). In order to emphasize this position, the NRC is simply including a statement noting that Paragraph -1100(e) cannot be used to exempt welds from the inspection requirements. Therefore, it is the NRC's position that the change in the language of 10 CFR 50.55a(g)(6)(ii)(F)(2) is not a change in NRC position or requirement and does not fall within the definition of backfitting.

The NRC made no change to the final rule as a result of this comment, and a backfit analysis was not prepared to support 10 CFR 50.55a(g)(6)(ii)(F)(2).

Comment: The proposed condition in 10 CFR 50.55a(g)(6)(ii)(F)(4) should be reworded to state, "essentially 100 percent of the required volumetric examination coverage for circumferential..." in lieu of "essentially 100 percent volumetric examination coverage requirement for circumferential...." The current proposed condition could lead the reader to examine 100 percent of the weld volume, rather than the required weld volume. [19-6]

NRC Response: The NRC agrees with this comment. The NRC has revised 10 CFR 50.55a(g)(6)(ii)(F)(4) of the final rule to state, "When implementing Paragraph -2500(a) of ASME BPV Code Case N-770-2, essentially 100 percent of the required volumetric examination coverage shall be obtained, including greater than 90 percent of the volumetric examination coverage for circumferential flaws."

Comment: The NRC should put a limit on the use of the last sentence of -2500(c), and not the entire section. [19-7]

NRC Response: The NRC disagrees with this comment. The proposed condition states, "Licensees are prohibited from using Paragraph -2500(c) and -2500(d) of ASME BPV Code Case N-770-2 to meet examination requirements." Paragraph -2500(c) of ASME BPV Code Case N-770-2 gives an alternative method to obtain acceptable inspection coverage if less-than-acceptable axial flaw coverage is obtained in paragraph -2500(a). The last sentence of Paragraph -2500(c) states, "The examination coverage requirements shall be considered to be satisfied." The proposed condition does not prohibit the licensee from performing the actions listed in Paragraph -2500(c); it only prohibits using those actions to meet the examination requirements. Conditioning just the final sentence in Paragraph -2500(c) would perform the same function. Therefore, no change in the wording is required.

The NRC made no change to the final rule as a result of this comment.

Comment: Paragraph -2500(d) of ASME BPV Code Case N-770-2 is understandably restricted in the proposed condition 10 CFR 50.55a(g)(6)(ii)(F)(4). [19-8]

NRC Response: No response is necessary.

Comment: It may help the reader understand the purpose of the limitation, proposed condition 10 CFR 50.55a(g)(6)(ii)(F)(9), if it is clarified that the limitation applies to plants that have extended their reactor pressure vessel examination interval to something longer than 10 years. [5-14]

NRC Response: The NRC disagrees with this comment. The condition does not require a licensee to have extended its reactor pressure vessel examination interval. The purpose of the proposed condition is simply to not allow the deferral of the first examination for welds mitigated with optimized weld overlays.

The NRC made no change to the final rule as a result of this comment.

Comment: The condition in 10 CFR 50.55a(g)(6)(ii)(F)(9) seems to contradict the condition in 10 CFR 50.55a(g)(6)(ii)(F)(8) which allows the initial inservice examination to be performed between the third refueling outage and no later than 10 years after the application. If the overlay is installed at some point within the interval, deferring the initial inservice examination to the end of the interval will be less than 10 years. The condition should be clarified to address this. [8-52; 14-36]

NRC Response: The NRC disagrees with these comments. This condition, which does not allow the deferral of the first inspection to the end of the interval, works in conjunction with 10 CFR 50.55a(g)(6)(ii)(F)(8). If deferral were allowed, then depending on when the 10-year point for the required inspection of 10 CFR 50.55a(g)(6)(ii)(F)(8) occurred within a 10-year inservice inspection interval, the deferral of the inspection to the end of the 10-year inservice inspection interval could allow the first inspection to not occur until up to 20 years after the

installation of the optimized weld overlay. This was not the intent of the original 10-year inspection requirement for the first inspection. The potential for deferral of the first inspection following installation of the optimized weld overlay only became possible in ASME BPV Code Case N-770-2 because of the change in categorization of an optimized weld overlay to Item Number C-2. This condition removes the deferral possibility and ensures the inspection frequency of item 10 CFR 50.55a(g)(6)(ii)(F)(8) will be met.

The NRC made no change to the final rule as a result of these comments.

VIII. Other Comments

Comment: The proposed rule is overly restrictive. The ASME Code and Code Cases are sound. The proposed conditions are not merited. [25-1]

NRC Response: No response is necessary. This general statement is supported by more specific comments later in the comment submission, and the NRC responds to those comments in separate sections of this comment response document.

The NRC made no change to the final rule as a result of this comment.

Comment: The NRC should evaluate previous conditions and not automatically extend to later editions. [25-2]

NRC Response: No response is necessary. This general statement is supported by more specific comments later in the comment submission, and the NRC responds to those comments in separate sections of this comment response document.

The NRC made no change to the final rule as a result of this comment.