AD05-1
PDR

FEB 21 1990

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: REQUEST FOR CRGR REVIEW OF PROPOSED AMENDMENT TO 10 CFR 50.55a THAT WOULD INCORPORATE BY REFERENCE THE 1986 ADDENDA, 1987 ADDENDA, 1988 ADDENDA, AND 1989 EDITION OF SECTION III, DIVISION 1, AND SECTION XI, DIVISION 1, OF THE ASME BOILER AND PRESSURE VESSEL CODE, AND WOULD IMPOSE AN AUGMENTED EXAMINATION FOR REACTOR VESSELS

Enclosed for review and approval by the CRGR is a proposed rule to amend 10 CFR 50.55a. The proposed rule was submitted for Division review on June 28, 1989. Comments received were resolved and, as appropriate, incorporated into the proposed rule. A summary of the resolution of Division level comments is provided in Enclosure 3.

The proposed rule would:

- o Incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, and the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure vessel Code.
- o Specify that, when using the 1988 Addenda or 1989 Edition of Section XI, supplemental rules must be followed for analysis of leakage rates and corrective actions for containment isolation valves that do not provide a reactor coolant system pressure isolation function.
- o Impose a one-time augmented examination of reactor vessel shell welds to expedite implementation of the expanded reactor vessel examination specified in the 1988 Addenda and 1989 Edition of Section XI.
- o Separate the requirements in § 50.55a for inservice inspection and inservice testing.
- o Make an editorial correction in the use of "shall" and "must" throughout § 50.55a to be consistent with the convention of the Office of the Federal Register.

Enclosure 1 contains the proposed rule. Enclosure 2 contains the regulatory analysis in support of the proposed rule. Enclosure 2 contains three appendices. Appendix A provides the documented evaluation for backfitting required by § 50.109, "Backfitting." Appendix B provides comparative text to show the manner in which the new paragraph 50.55a(f), "Inservice testing requirements," was developed from paragraph 50.55a(g), "Inservice inspection requirements." Appendix C provides the recordkeeping analysis required by OMB to comply with the Paperwork Reduction Act. Enclosure 4 contains the information required in submittals to the CRGR and/or provides references to that information in other parts of this submittal package. Enclosure 5 provides copies of Part 6 and Part 10 (as contained in the ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987), which replace Subsections IWP and IWV, respectively, in the 1988 Addenda and 1989 Edition of Section XI. The supplemental rules noted above for testing of containment isolation valves apply to implementation of Part 10. Copies of Section III and Section XI are not included, herein, because of their large size, and their general availablity from the staff.

RES has prepared the supporting regulatory analysis based upon the following criteria:

o Incorporating by reference the latest addenda and edition of the ASME Code is not a backfit (see OGC opinion Enclosure 2, P. 12, Section (3)). A cost-benifit analysis is not required because the ASME Code is developed under an established consensus process that ensures a proper balance between interests (see Enclosure 2, P.6, Section 4.A.). The CRGR did not require a cost-benefit analysis for the last amendment to § 50.55a, which endorsed the ASME Code without exception, recognizing that the proposed rule would be published for public comment. No significant comments were received for that proposed rule, and on that basis the CRGR determined that it was not necessary for the CRGR to review the final rule. We believe that CRGR position continues to be appropriate for the portion of this proposed amendment that updates the reference to ASME Code editions and addenda.

o The proposed supplemental requirement (see Enclosure 1, P. 22, paragraph 50.55a(b)(2)(vii) that would be applied to the 1988 Addenda and 1989 Edition of Section XI is not a backfit because the proposed requirement is presently part of the 1987 Addenda (and earlier) (see Enclosure 2, P. 7).

o The proposed augmented examination of the reactor vessel shell welds (see Enclosure 1, P. 32) is a backfit that is required to ensure that the public health and safety is adequately protected (see Enclosure 2, Appendix A for the documented evaluation required by § 50.109).

o The separation of inservice inspection and inservice testing requirements is essentially editorial and does not impose any new requirements. Incorporating words into new paragraph 50.55a(f)(6) (see Enclosure 1, P. 27) regarding the intent of the Commission to impose augmented testing requirements, if deemed necessary to ensure operational readiness, is administrative since it does not impose any new requirement. A regulatory analysis, with documented evaluation, if necessary, would be prepared to demonstrate the need and value of any augmented test program proposed in the future.

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By concurrence in this memorandum, the proposed rule has been approved by AEOD and ARM; the OGC has indicated that it has no legal objection to the proposed rule; and IRM has approved the supporting statement for information collection requirements (Enclosure 2, Attachment C). NRR has indicated that it has no technical objection to the proposed rule, but has noted a concern that the draft regulatory analysis does not include a cost-benefit analysis to support the proposed amendment. The criteria used by RES to prepare the regulatory analysis are delineated above for the various elements of the proposed rule. RES believes that the draft regulatory analysis, which was developed on the basis of these criteria, properly supports the proposed amendment and the concept of backfitting without the need for a specific cost-benefit analysis.

The contact for this proposed rule is Gilbert C. Millman, Electrical & Mechanical Engineering Branch (x23848).

ORIGINAL SIGNED BY

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosures:

1. Draft Proposed Rule
2. Draft Regulatory Analysis
3. Resolution of Division Comments
4. CRGR Information
5. O&M Parts 6 and 10

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NOTE TO CONCURRING OFFICES:

Separate concurrence packages have been sent to NRR, AEOD, ARM, IRM, and OGC for the purpose of expediting the concurrence process. The individual concurrences and any comments will be integrated into a composite package prior to submittal to the D:RES for signature.

SEE PREVIOUS CONCURRENCE*********************************

| EMEB:DE* | EMEB:DE* | DE:RES* | DE:RES* | RES* | NRR | AEOD * |
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| 02/ /90 | 02/ /90 | 02/02/90 | 02/21/90 | | | |

** Concurrence noted in attached memorandum.

*** IRM has reviewed and approves enclosed OMB submittal, but reserves concurrence for final, composite OMB transmittal package.

[7590-01]

Enclosure 1

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AD05

Codes and Standards for Nuclear Power Plants

: Nuclear Regulatory Commission.

N: Proposed rule.

:RY: The Commission proposes to amend its regulations to incorporate by rence the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of :ion III, Division 1, of the American Society of Mechanical Engineers ler and Pressure Vessel Code (ASME Code), and the 1986 Addenda, 1987 :enda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, of the ASME de, with a specified modification. The proposed amendment would impose gmented examination of reactor vessel shell welds, and would separate in the gulations the requirements for inservice testing from those for inservice nspection by placing the requirements for inservice testing in a separate



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10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which incorporated by reference a new edition and new addenda to the ASME Code. This amendment revised § 50.55a to incorporate by reference the Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition for Division 1 rules of Section III, "Rules for the Construction of Nuclear Power Plant Components," and the Winter 1983 Addenda, Summer 1984 Addenda, Winter 1984 Addenda, Summer 1985 Addenda, Winter 1985 Addenda, and 1986 Edition for Division 1 rules of Section XI, "Rules for the Inservice Inspection of Nuclear Power Plant Components," of the ASME Code.

The Commission proposes to amend § 50.55a to incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, of the ASME Code, and the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, of the ASME Code, with a specified modificaton. (In 1986, the ASME Code initiated a once-a-year addenda system and dropped the Summer/Winter designator). Also, the proposed amendment would impose augmented examination of reactor vessel shell welds, and would separate in the regulations the requirements for inservice testing from those for inservice inspection by placing the requirements for inservice testing in a separate paragraph.

Subsection IWP, "Inservice Testing of Pumps," and Subsection IWV, "Inservice Testing of Valves," as contained in the 1988 Addenda and 1989 Edition of Section XI, incorporate by reference, respectively, Part 6, "Inservice Testing of Pumps in Light-Water Reactor Power Plants," and Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants," of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987, "Operation and Maintenance of

Nuclear Power Plants." The contents of Subsections IWP and IWV in the 1988 Addenda and 1989 Addenda are replaced in their entirety by the referenced rules of Part 6 and Part 10, respectively. The NRC has determined that certain requirements in Part 10 represent unacceptable changes from present requirements in Subsection IWV of Section XI editions and addenda that have been incorporated by reference into § 50.55a. Therefore, the proposed amendment would incorporate by reference the 1988 Addenda and 1989 Edition of Section XI, Division 1, with a specified modification to Subsection IWV.

Paragraph IWV-3420 of Subsection IWV of Section XI editions and addenda presently incorporated by reference in § 50.55a require all Category A valves, except those that function in the course of plant operation in a manner that demonstrates functionally adequate leak tightness, to undergo individual valve leakage rate testing. Subsection IWV paragraphs IWV-3426 and IWV-3427, respectively, require analysis of leakage rates and implementation of corrective actions dependent upon results of the leakage rate analysis. Subsection IWV in the 1988 Addenda and 1989 Edition of Section XI, which reference Part 10 for the inservice testing of valves, provide rules for testing containment isolation valves (CIVs) (i.e., paragraph 4.2.2.2 of Part 10 of the ASME/ANSI OMa-1988 Addenda). These rules specify that Category A CIVs be tested in accordance with 10 CFR Part 50, Appendix J, and that CIVs which also provide a reactor coolant system pressure isolation function additionally be tested in accordance with Part 10, paragraph 4.2.2.3, "Leakage Rate for Other Than Containment Isolation Valves." Paragraph 4.2.2.3(e) of Part 10 requires analysis of leakage rates and paragraph 4.2.2.3(f) of Part 10 specifies requirements for corrective action for Category A CIVs that also provide a reactor coolant system pressure isolation function.

Subsection IWV in the 1988 Addenda and 1989 Addenda eliminate the present requirement to analyze leakage rates and to take corrective action in the event of abnormally high leakage rates for those CIVs that do not provide a reactor coolant system pressure isolation function. The NRC is concerned that this could significantly reduce the ability to detect degraded valves and, thereby, could permit an unacceptable reduction in the safety margin associated with the leak tight integrity of those CIVs that do not provide a reactor coolant system pressure isolation function. Therefore, the NRC proposes to incorporate by reference the 1988 Addenda and 1989 Edition of Section XI with a modification that would be specified in a new § 50.55a(b)(2)(vii). The modification would preserve the existing requirements for analysis of leakage rates and corrective actions that exist in Subsection IWV prior to the 1988 Addenda. Specifically, the modification would require licensees to implement the requirements of paragraph 4.2.2.3(e), "Analysis of Leakage Rates," of Part 10 and paragraph 4.2.2.3(f), "Corrective Action," of Part 10, in addition to the requirements of paragraph 4.2.2.2 of Part 10, for all Category A valves that are CIVs, regardless of whether they provide a reactor coolant system pressure isolation function.

The 1988 Addenda to Section XI modifies the 1986 Edition to require in the 2nd, 3rd, and 4th inspection intervals examination of essentially 100 percent of the length of all reactor vessel shell welds (i.e., Item B1.10, "Shell Welds," of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB, "Requirements for Class 1 components of Light-Water Cooled Power Plants"). Since the 1989 Edition is identical to the 1986 Edition as modified by the 1986 Addenda, 1987

Addenda, and 1988 Addenda, this revision also appears in the 1989 Edition of Section XI. The 1986 Edition of Section XI (the most current Section XI rules presently incorporated by reference into § 50.55a) requires examination of only one longitudinal weld and one circumferential weld from the beltline region during the 2nd, 3rd, and 4th inspection intervals. The requirement to examine essentially 100 percent of the length of all reactor vessel shell welds during the 1st inspection interval has been in Section XI since the 1975 Winter Addenda to the 1974 Edition.

In view of recent information which shows the susceptibility of reactor vessel materials to degradation and the limited examinations performed to date on some reactor vessels, the NRC is concerned with the length of time which might elapse before a licensee would be required to implement the reactor vessel shell weld examinations specified in the 1988 Addenda and the 1989 Edition of Section XI through routine updating of its inservice inspection program. Section 50.55a(g)(4)(ii) requires that inservice inspection programs be updated to reflect the latest edition and addenda of Section XI identified in § 50.55a(b)(2) 12 months prior to the start of the next 120-month inspection interval. Routine updating in accordance with this requirement could result in the 1989 Edition not being implemented for as long as 240 months (20 years). For example, a plant just entering the first period in the 2nd, 3rd, or 4th inspection interval when this rule becomes effective would not have to implement the reactor vessel examinations specified in the 1989 Edition for 20 years, because that inspection interval would be covered by a previous Section XI edition/addenda and because under existing Section XI rules, the reactor vessel examinations in the succeeding interval, which would implement the 1989 Edition or later, could be deferred another 10 years until the end of that interval. Similarly, a plant just entering the second or third period in the 2nd, 3rd or 4th inspection interval would not be required to implement the 1989 Edition, or subsequent addenda, for 200 months (16 years, 8 months) or 160 months (13 years, 4 months), respectively.

Consistent with the existing updating requirements of § 50.55a(g)(4)(ii) and the changing requirements of Section XI, some inservice inspection programs based on certain editions and addenda of Section XI may have resulted in very limited reactor vessel examinations. For example, if examinations of the beltline welds during the 1st inspection interval were performed to the 1974 edition of Section XI, 5 percent of the circumferential welds and 10 percent of the longitudinal welds would have been examined. If, for the same plant, examinations during the 2nd inspection interval were performed to the 1980 Edition, including subsequent addenda, one circumferential weld and one longitudinal weld would have been required to be examined. [The 1974 Edition of Section XI (with addenda through the 1975 Winter Addenda) through the 1986 Edition (with addenda through the 1987 Addenda) require that all reactor vessel shell welds be examined volumetrically during the 1st inspection interval, and that one circumferential and one longitudinal beltline weld be examined volumetrically in succeeding inspection intervals; whereas the 1971 Edition through the 1974 Edition (with addenda through the 1975 Summer Addenda) require that 10 percent of the length of each longitudinal weld and 5 percent of the length of each circumferential weld be examined volumetrically each inspection interval.]

Degradation of reactor vessel materials has become more of a concern recently, because (1) results from irradiation surveillance material tests show, contrary to earlier expectations, that certain vessel materials undergo significant radiation damage, (2) indications from operational data show that stress corrosion cracking of BWR reactor vessels is more probable than was thought several years ago, and (3) evidence that PWR steam generator shell materials, which are essentially identical to unclad PWR reactor vessel materials, are susceptible to stress corrosion cracking.

The NRC is concerned that the inherent delay in implementing the expanded reactor vessel examinations is inconsistent with the importance of the reactor vessel examinations as evidenced by recent new information regarding degradation of reactor vessel materials, the limited examination of shell welds previously performed on many reactor vessels, and the need to ensure that the failure probability of the reactor vessel remains extremely low. It is the judgment of the NRC that because of new information and limited previous reactor vessel examinations, there may exist a substantially greater potential for reactor vessel degradation than previously considered and that restoration of the level of protection presumed by the regulations requires more than compliance to existing regulatory requirements.

The NRC has determined that augmented reactor vessel examinations are necessary to provide "adequate protection" as addressed in § 50.109, "Backfitting." Section 50.109(a)(4) specifies that a backfit analysis is not required when, consistent with § 50.109(a)(4)(ii), the action is necessary to ensure that the facility provides adequate protection to the health and safety of the public. Section 50.109 does require, however, a "documented evaluation" which is provided in the regulatory analysis that supports this proposed rule.

Section 50.55a(g)(6)(ii) addresses augmented inservice inspection programs for those systems and components for which the Commission deems that added assurance of structural reliability is necessary. For that purpose, and consistent with the above discussion, it is proposed that § 50.55a(g)(6)(ii)(A) be added to require expedited implementation of the reactor vessel shell weld examinations specified in the 1989 Edition of Section XI, Division 1, in Item B1.10, "Shell Welds," of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table 2500-1 of Subsection IWB, "Requirements for Class 1 Components of Light-Water Cooled Power Plants." Proposed § 50.55a(g)(6)(ii)(A) was developed with two primary considerations in mind. First, the proposed rule must require implementation of the provisions for reactor vessel shell weld examinations provided in the 1989 Edition as quickly as practicable. Second, to minimize unnecessary impact on licensees, the implementation requirements for the augmented examination should be integrated as closely as possible with existing examination requirements and practices.

The NRC has structured the proposed requirement for augmented examination of reactor vessel shell welds recognizing that plants will be on different schedules for their 120-month inservice inspection interval. Section 50.55a(g)(6)(ii)(A)(1) requires all licensees to implement the specified augmented reactor vessel examination during the inspection interval in force when this proposed rule becomes effective, subject to conditions specified in proposed § 50.55a(g)(6)(ii)(A)(2) and (3). Section 50.55a(g)(6)(ii)(A)(1) specifically permits the use of the augmented examination as a substitute for the reactor vessel shell weld examinations scheduled for the inspection interval in effect when this proposed rule becomes effective.

The NRC recognizes that plants with fewer than 40 months remaining in the inspection interval when this proposed rule becomes effective may find it impractical to implement the augmented reactor vessel examination during that inspection interval. Therefore, proposed § 50.55a(g)(6)(ii)(A)(2) permits plants with fewer than 40 months remaining in the inspection interval when this rule becomes effective to defer the augmented examination until the first period of the next inspection interval. However, this same paragraph specifically prohibits use of the augmented examination as a substitute for reactor vessel examinations scheduled for this next inspection interval. If such a substitution were permitted, it could result in a 160 month time span before the next reactor vessel examination, which would be contrary to the basic intent of the augmented examination. For example, a 160 month time span would result if a licensee just entering the third period deferred the augmented reactor vessel examination to the first period of the next inspection interval and used that examination as a substitute for the normally scheduled reactor vessel shell weld examination for that interval, and then deferred the reactor vessel examination to the end of the interval as permitted by Section XI.

Proposed § 50.55a(g)(6)(ii)(A)(3) specifies that a licensee that has either completed or has scheduled an inspection of essentially 100 percent of the length of all Examination Category B-A shell welds during the current inservice inspection interval does not have to implement the proposed requirement for augmented examination of the reactor vessel shell welds. Primarily, this proposed paragraph is intended to permit licensees who would be in the 1st inspection interval to use the essentially 100 percent reactor vessel shell weld examination required for that interval by Section XI to satisfy the requirement for the proposed augmented reactor vessel examination. The technical objective of the augmented examination would have been accomplished under such conditions. These licensees would continue to apply the current requirements of § 50.55a(g)(4) until the next inspection interval when future examinations must be performed based on ASME Section XI, 1989 Edition, or later Code edition specified in § 50.55a(b).

The proposed amendment to § 50.55a would separate the requirements for inservice testing from those for inservice inspection by moving the requirements for inservice testing to a separate paragraph. Presently, § 50.55a(g), "Inservice inspection requirements," specifies the requirements for (1) preservice and inservice examinations for Class 1, Class 2, and Class 3 components and their supports, (2) system pressure tests for Class 1, Class 2, and Class 3 components, and (3) inservice testing of Class 1, Class 2, and Class 3 pumps and valves. In order to emphasize the importance of inservice testing and to more clearly distinguish its requirements from those of inservice inspection, the proposed rule would move the present requirement for inservice testing from existing § 50.55a(g), "Inservice inspection requirements," to a separate (presently reserved) § 50.55a(f), which would be titled "Inservice testing requirements." All existing requirements for inservice examination and system pressure testing would be retained in § 50.55a(g).

Two editorial revisions, relative to existing § 50.55a(g), are included in the proposed new § 50.55a(f). These editorial revisions (1) reserve § 50.55a(f)(3)(i) and (ii) so that the structure of § 50.55a(f) would parallel that of § 50.55a(g) for the purpose of promoting easier cross-referencing between the two paragraphs, (2) modify reference to 120-month inspection interval in § 50.55a(g) to 120-month interval in proposed § 50.55a(f), because inspection interval, as used in Section XI, is used only in the context of inservice inspection. (The term "test interval" was not used because, unlike inspection interval, the 120-month time frame does not designate a period of required actions for the testing program. The 120-month interval used in § 50.55a(f) and the 120-month inspection interval used in § 50.55a(g) are considered by the staff to be coincident for the purpose of 120-month updating requirements.)

In addition, two administrative changes have been made in the development of proposed § 50.55a(f) relative to existing § 50.55a(g). First, § 50.55a(f)(6)(ii) has been added to indicate intent by the Commission to impose an augmented inservice testing program if added assurance of operational readiness is deemed necessary. This proposed paragraph only indicates intent and does not impose a specific requirement. It does parallel the existing § 50.55a(g)(6)(ii) which specifies that the Commission may require an augmented inservice inspection program for systems and components for which it deems that added assurance of structural reliability is necessary.

Second, the proposed amendment includes the addition of introductory text to § 50.55a(g) which states that the requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in § 50.55a(f). This change is necessary because the proposed placement of inservice testing requirements into a separate § 50.55a(f) would cause administrative inconsistencies with regard to existing references to § 50.55a(g) for inservice testing in documents such as technical specifications, safety analysis reports, procedures, and records. With the proposed change, existing references to § 50.55a(g) for inservice testing would refer the user to § 50.55a(f) where the specific requirements for inservice testing would be located. The NRC recommends that as the governing documents are updated, the direct reference to § 50.55a(f) be incorporated, as appropriate.

Section 50.55a(g) provides requirements for selecting the ASME Code edition and addenda of Section XI to be complied with during the preservice inspection (§ 50.55a(g)(3), for plants whose construction permit was issued on or after July 1, 1974); the initial 10-year inspection interval (§ 50.55a(g)(4)(i)); and successive 10-year inspection intervals (§ 50.55a(g)(4)(ii)). As noted in the Supplementary Information to the final rule of the most recent amendment to § 50.55a (May 5, 1988; 53 FR 16051), paragraph IWA-2400 of Section XI (as revised by the Winter 1983 Addenda) incorporated rules for selecting the applicable edition and addenda of Section XI during the preservice inspection (IWA-2411), the initial 10-year inspection interval (IWA-2412), and successive 10-year inspection intervals (IWA-2413). The criteria provided in the regulations and Section XI are effectively the same for the preservice inspection and the successive 10-year inspection intervals, but differ for the initial 10-year inspection interval. In general, use of the Commission requirements will result in the selection of a more recent edition and addenda than will use of the Section XI rules. Satisfying the requirements of § 50.55a(g)(4)(i) for the inital 10-year inspection interval will, in general, also satisfy the rules of Section XI. Although the Section XI requirements for selecting editions and addenda remain unchanged in the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition, the Commission is reaffirming its intent that in all cases the existing requirements in § 50.55a(g) be the basis for selecting the edition and addenda of Section XI to be complied with during the preservice inspection, the initial 10-year inspection interval, and the successive 10-year inspection intervals.

The proposed amendment would make a number of editorial changes to § 50.55a for the purpose of adopting a standard convention for imposing an obligation or expressing a prohibition. In this convention "shall" is used to impose an obligation on an individual or legal entity capable of performing the required action, "must" is used as the mandatory form when the subject of the sentence is an inanimate object, and "may not" is used to impose a prohibition. The following paragraphs were amended solely to be consistent with this convention: the introductory paragraph to the section; paragraphs (a)(1), (a)(3), (b)(2)(iii), (b)(2)(iv), (g)(1), (g)(3)(ii), (g)(3)(iii), (g)(3)(iv), introductory paragraph to (g)(4), (g)(4)(i), (g)(4)(ii), (g)(5)(i), (g)(5)(iv), (g)(6)(i), (h), and footnote 8. Other paragraphs were revised for the same editorial reason, but they also contain technical revisions relevant to other parts of this proposed amendment. Section 50.55a(f) has been developed consistent with the noted convention.

Subsection IWE, "Requirements for Class MC Components of Light-Water-Cooled Power Plants," was added to Section XI, Division 1, in the Winter 1981 Addenda. However, 10 CFR 50.55a presently incorporates Section XI inservice inspection requirements for only Class 1, 2, and 3 components and their supports. The regulation does not currently address the inservice inspection of containments. Because this amendment is only intended to update current regulatory requirements to include the latest ASME Code edition and addenda, the requirements of Subsection IWE would not be imposed upon Commission licensees by this amendment. The incorporation by reference of Subsection IWE into § 50.55a is presently the subject of a separate rulemaking action. Section 50.55a(b)(2)(vi) is reserved for that action.

The NRC previously alerted all holders of operating licenses or construction permits for nuclear power reactors, through NRC Information Notice No. 88-95 (IN 88-95), "Inadequate Procurement Requirements Imposed by Licensees on Vendors," to the potential that inadequate licensee procurement requirements or implementation by vendors in supplying components under the ASME Code could result in failure by these vendors to fully implement 10 CFR Part 50, Appendix B (Quality Assurance Criteria). The problem, which was revealed during routine NRC inspections of vendors, resulted from the belief by some vendors that if an item was exempted by the ASME Code from Code requirements, the item was exempt from all other regulatory requirements. The apparent belief of some vendors was that since NRC endorses the ASME Code in its regulations and has accepted the various exemptions, there are, therefore, no other applicable regulatory requirements. This belief is not consistent with the NRC position. The NRC reaffirms its position which, as previously put forth in IN 88-95, states that all safety-related items, even those exempted from ASME Code requirements, are required to be manufactured under a quality assurance program that meets
10 CFR Part 50, Appendix B requirements.

## Environmental Impact: Categorical Exclusion

The NRC has determined that this proposed rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(3). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for thi proposed rule.

## Paperwork Reduction Act Statement

This proposed rule would amend information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This proposed rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

Public reporting burden for this collection of information is estimated to average 70 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington, DC 20503.

## Regulatory Analysis

The Commission has prepared a regulatory analysis for this proposed amendment to the regulations. The analysis examines the costs and benefits of the alternatives considered by the Commission. Interested persons may examine a copy of the regulatory analysis at the NRC Public Document Room, 2120 L Street NW. (Lower Level), Washington, DC. Single copies of the analysis may be obtained from Mr. G. C. Millman, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone (301) 492-3848.

## Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this proposed rule does not fall within the purview of the Act.

## Backfit Analysis

The NRC has concluded, on the basis of the documented evaluation required by 10 CFR 50.109(a)(4), that the backfit requirements contained in this proposed amendment are necessary to ensure that the facility provides adequate protection to the public health and safety, and, therefore, that a backfit analysis is not required and the cost-benefit standards of 10 CFR 50.109 (a)(3) do not apply. The documented evaluation contained in the regulatory analysis includes a statement of the objectives of and reasons for the backfits that would be required by the proposed rule and sets forth the basis for the NRC's conclusion that these backfits are not subject to the cost-benefit standards of 10 CFR 50.109 (a)(3).

## List Of Subjects In 10 CFR Part 50

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

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 50.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

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

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235); sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, 50.54(dd) and 50.103 also issued under Sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a and Appendix Q also issued under sec. 102, Pub. L 91-190, 83 Stat. 853 (42 U.S.C 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844). Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 122, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80-50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

2. In § 50.55a, the introductory text, paragraphs (a), (b)(1), the introductory text of (b)(2), (b)(2)(iii), (b)(2)(iv), (g)(1), (g)(2), (g)(3)(i), (g)(3)(ii), (g)(4), (g)(5)(i), (g)(5)(iv), (h), and footnote 8 are revised; paragraphs (g)(3)(iii) and (g)(3)(iv) are removed and reserved; paragraph (b)(vi) is added and reserved; and paragraphs (b)(2)(vii), (f), introductory text to (g), and

(g)(6)(ii)(A) are added to read as follows:

§ 50.55a Codes and standards.

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

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

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

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

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

(b) * * *

(1) As used in this section, references to Section III of the ASME Boiler and Pressure Vessel Code refer to Section III, Division 1, and include addenda through the 1988 Addenda and editions through the 1989 Edition.

(2) As used in this section, references to Section XI of the ASME Boiler and Pressure Vessel Code refer to Section XI, Division 1, and include addenda through the 1988 Addenda and editions through the 1989 Edition, subject to the following limitations and modifications:

*     *     *     *     *

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

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

*    *    *    *    *

(vi) [Reserved]

(vii) Inservice testing of containment isolation valves. When using Subsection IWV in the 1988 Addenda or the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, leakage rates for Category A containment isolation valves that do not provide a reactor coolant system pressure isolation function must be analyzed in accordance with paragraph 4.2.2.3(e) of Part 10, and corrective actions for these valves must be taken in accordance with paragraph 4.2.2.3(f) of Part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987.

*    *    *    *    *

(f) Inservice testing requirements.

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

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

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

(i) [Reserved]

(ii) [Reserved]

(iii) Pumps and valves which are classified as ASME Code Class 1 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[4] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

(iv) Pumps and valves which are classified as ASME Code Class 2 and Class 3 must be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[4] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

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

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

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

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

(iii) [Reserved]

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

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

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

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

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

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

(ii) The Commission may require the licensee to follow an augmented inservice test program for pumps and valves for which the

Commission deems that added assurance of operational readiness is necessary.

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

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

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 must be designed and be provided with access to enable the performance of inservice examination of such components (including supports) and must meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda' in effect six months prior to the date of issuance of the construction permit. The components (including supports) may meet

the requirements set forth in subsequent editions of this code and addenda which are incorporated by reference in paragraph (b) of this section, subject to the limitation and modifications listed therein.

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

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

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

(iii) [Reserved]

(iv) [Reserved]

* * * * *

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

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

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

(iii) [Reserved]

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

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

* * * * *

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

(6) * * *

(ii) * * *

(A) Augmented examination of reactor vessel

(1) All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on ________ (effective date of rule will be inserted), the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in § 50.55a(g)(6)(ii)(A)(2) and (3). The augmented examination may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on ________(effective date of rule will be inserted).

(2) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on ________ (effective date of rule will be inserted) may defer the augmented reactor vessel examination specified in § 50.55a(g)(6)(ii)(A)(1) to the first period of the next inspection interval. The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice

inspection interval in effect on _________ (effective date of rule will be inserted). The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed.

(3) The requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100 percent of the reactor vessel shell welds specified in § 50.55a(g)(6)(ii)(A)(1) that has been completed, or is scheduled for implementation with a written commitment, or is required by § 50.55a(g)(4)(i), during the inservice inspection interval in effect on _________ (effective date of rule will be inserted).

(h) Protection systems. For construction permits issued after January 1, 1971, protection systems must meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," (IEEE-279) in effect[7] on the formal docket date[4] of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE-279 which become effective.

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[4] ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability -- ASME Section III Division 1," NRC

Regulatory Guide 1.85, "Materials Code Case Acceptability -- ASME Section III Division 1," and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptabilty -- ASME Section III Division 1." The use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to § 50.55a(a)(3).

' For purposes of this regulation the proposed IEEE 279 became "in effect" on August 30, 1968, and the revised issue IEEE 279--1971 became "in effect" on June 3, 1971. Copies may be obtained from the Institute of Electrical and Electronics Engineer's, United Engineering Center, 345 East 47th St., New York, NY 10017. Copies are available for inspection at the Commission's Technical Library, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland.

• Where an application for a construction permit is submitted in four parts pursuant to the provisions of § 2.101(a-1) and Subpart F of this chapter, "the formal docket date of the application for a construction permit" for purposes of this section must be the date of docketing of the information required by § 2.101(a-1) (2) or (3), whichever is later.

*     *     *     *     *

Dated at Rockville, Maryland this ______day of ______________1990.

For the Nuclear Regulatory Commission.

_______________________________

James M. Taylor,

Executive Director for Operations.

PDR
Certified Original:__________
Date:__________

Enclosure 2

Draft Regulatory Analysis

Amendment to 10 CFR § 50.55a
Codes and Standards

Executive Summary

Section 50.55a of the NRC regulations requires that nuclear power plant owners construct Class 1, Class 2, and Class 3 components in accordance with Division 1 rules of Section III, "Rules for Construction of Nuclear Power Plant Components", of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and that they provide for and perform inservice inspection (ISI) and inservice testing (IST) of those components in accordance with Division 1 rules of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", of the ASME Code. NRC has been mandating this requirement since shortly after publication of Section III and Section XI of the ASME Code in 1971. The preamble to the August 24, 1972, final rule amending § 50.55a (37 FR 17021) states that "As new or amended editions of applicable codes, code cases, or addenda are issued, the Commission will review them and amend the provisions of § 50.55a ... as appropriate."

The mechanism for endorsement, which has been used since the first endorsement in 1971, has been to incorporate by reference Section III, Division 1, and Section XI, Division 1, of the ASME Code into § 50.55a. The regulation identifies which editions and addenda of the ASME Code have been approved by the NRC for use. At present, the NRC endorses for Section III, Division 1, all addenda through the Winter 1985 Addenda and all editions through the 1986 Edition, and for Section XI, Division 1, all addenda through the Winter 1985 Addenda and all editions through the 1986 Edition, subject to certain limitations and modifications.

The proposed amendment would incorporate by reference, additionally, the 1986 Addenda, 1987 Addenda, 1988 Addenda and 1989 Edition of Section III, Division 1, and the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI, Division 1, with a specified modification. (In 1986 the ASME implemented a once-a-year addenda system and dropped the Summer/Winter designator). Also, the proposed amendment would impose an augmented examination of reator vessel shell welds, and would separate the requirements in the regulation for inservice testing from those for inservice inspection by placing the requirements for inservice testing in a separate paragraph. The proposed amendment is of particular importance to operating plants because § 50.55a requires that licensees update their inservice examination and in-service testing programs every ten years to comply with the requirements of the latest edition and addenda of Section XI, Division 1 endorsed by the NRC 12 months prior to the start of the next 120-month inspection interval. The proposed amendment is also of importance to plants preparing for their initial inspection interval, for these plants must comply with the latest edition and addenda of Section XI, Division 1, endorsed by the NRC 12 months prior to the date of issuance of the operating license.

The ASME Code is developed through the American National Standards Institute consensus process. This ensures that the various technical interests (e.g., utility, manufacturing, insurance, regulatory) are represented on the standards development committees and their viewpoints are addressed fairly in the standards writing process. In general, revisions are made to improve the ASME Code by providing more detailed rules where experience indicates greater guidance is necessary, or relaxing the rules where experience shows equivalent operational safety can be maintained with a reduced burden on the licensees. The consensus process ensures that the cost and benefit of revisions to the ASME Code are properly considered from all sides and are reasonably balanced. Should the NRC feel that at any time in the process safety is being compromised, it can and has taken exception in § 50.55a to the rules provided in the ASME Code. This option is not used often, but is available when necessary. The proposed amendment does implement this option by modifying a technical requirement contained in the 1988 Addenda and 1989 Edition of Subsection IWV of Section XI.

The proposed amendment would incorporate by reference the 1988 Addenda and 1989 Edition of Section XI, Division 1. This addenda and edition retain Subsection IWP "Inservice Testing of Pumps" and Subsection IWV "Inservice Testing of Valves", but the contents of these subsections are replaced, through the mechanism of referencing, with Part 6 and Part 10, respectively, of the ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987. The NRC has determined that certain requirements in Part 10 represent unacceptable changes from the previous requirements of Subsection IWV. Therefore, the NRC has identified in the proposed amendment a specific modification that must be applied when using Subsection IWV in the 1988 Addenda and 1989 Edition of Section XI. The modification will help ensure that safety margins on valve operability are retained.

The proposed amendment would require operating plants to implement, prior to the time required by normal updating of the inservice inspection program, the provision in the 1989 Edition to examine essentially 100% of the length of all reactor vessel shell welds during the 2nd, 3rd, and 4th inspection intervals. The NRC has determined that such an augmented examination is necessary because of recent information that shows reactor vessel materials to be more vulnerable to degradation than previously thought, and because many licensees have performed only very limited examinations of their reactor vessels. The augmented examination is necessary to ensure that the reactor vessel maintains a very low probability of failure.

The proposed amendment does not conflict with any existing or proposed regulatory action. Incorporating into the proposed rule the augmented examination of reactor vessel shell welds would eliminate the need for an NRC Generic Letter that has been drafted on the subject.

It is concluded that the proposed amendment, when implemented, would result in a net increase in the overall protection of public health and safety, because improved rules would be used in new and subsequent inservice inspection programs (i.e., obsolete requirements would not be continued from one 120-month ispection interval to another), and the augmented examination of reactor vessel shell welds would help to ensure a low probability of failure for those vessels.

1. Statement of the Problem

The proposed amendment updates the § 50.55a reference to Section III, Division 1, and Section XI, Division 1, editions and addenda, with a specified modification; imposes an augmented examination of the reactor vessel shell welds; and separates the ISI and IST requirements in the regulation. These proposed actions are addressed individually, below, in this regulatory analysis.

Update of Reference to Section III and Section XI Edition and Addenda

The General Design Criteria (Appendix A of 10 CFR Part 50) require that structures, systems, and components of light-water-reactor's be designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with the importance of the safety function performed. Without a set of specific rules to implement these quality standards, it would be necessary for each applicant and licensee to develop its own program for submittal to the NRC. Each program would have to be reviewed by the staff on a case-by-case basis. This would increase significantly the licensing review time and would make inspections by the staff more difficult because of the nonstandard nature of each program.

To provide a consistent set of rules, which the industry has participated in developing, § 50.55a mandates use of Section III, Division 1, of the ASME Code for construction of Class 1, 2, 3 components, and Section XI, Division 1, of the ASME Code for inservice inspection and inservice testing of these components. Section III and Section XI are implemented, respectively, by applicants and licensees of all light-water-cooled reactors. The NRC first endorsed the ASME Code by reference in 10 CFR § 50.55a in 1971. The ASME publishes a new edition of the Code every three years. In the past, new addenda have been published for Section III, Division 1, and for Section XI, Division 1, every six months. Since 1986, new addenda have been published once a year. It has been a continuing policy of the Commission to update this section of the regulations to keep the references current. In those cases where an item in the ASME Code is inconsistent with NRC criteria, an exception may be taken to endorsing that portion of the Code or supplementary criteria may be incorporated to make the item consistent with NRC requirements.

Section 50.55a presently endorses all addenda through the Winter 1985 Addenda, and all editions through the 1986 Edition for Section III, Division 1, and all addenda through the Winter 1985 Addenda, and all editions through the 1986 Edition for Section XI, Division 1, subject to certain limitations and modifications. The purpose of this proposed rulemaking is to incorporate by reference into the regulations the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition for both Section III, Division 1 and Section XI, Division 1.

The ASME Code is developed through the consensus process, which ensures that the various technical interests (e.g., utility, manufacturer, insurance, regulatory) are represented on the standards development committees and that their viewpoints are considered in the standards writing process. Endorsement of the ASME Code by the NRC provides a method of incorporating rules into the regulatory process that are acceptable to the NRC and have received industry participation in their development.

If the NRC did not take action to endorse the ASME Code, the NRC position on methods for construction, inservice inspection, and inservice testing would have to be established on a case-by-case basis. If the NRC did not take action to update the ASME Code references, improved methods for construction, inservice inspection, and inservice testing might not be implemented.

As noted previously, the NRC has determined that certain requirements in the ASME/ANSI Oma-1988 Addenda Part 10, which is referenced in the 1988 Addenda and 1989 Edition of Subsection IWV of Section XI, represent unacceptable changes from previous requirements in Subsection IWV. Therefore, the proposed amendment would incorporate by reference the 1988 Addenda and 1989 Edition of Section XI with a specific modification. This modification would require implementation of certain additional requirements for data analysis and corrective actions that are included in Subsection IWV prior to the 1988 Addenda.

Augmented Examination of Reactor Vessel

The 1988 Addenda to Section XI modifies the 1986 Edition to require in the 2nd, 3rd, and 4th inspection intervals examination of essentially 100% of the length of all reactor vessel shell welds (i.e. Item B1.10 of Examination Category B-A in Table IWB-2500-1 of Section XI Subsection IWB). Since the 1989 Edition is identical to the 1986 Edition as modified by the 1986 Addenda, 1987 Addenda, and 1988 Addenda, this requirement also appears in the 1989 Edition of Section XI. However, the 1986 Edition of Section XI (the latest rules presently incorporated by reference into § 50.55a) requires examination of only one longitudinal weld and one circumferential weld from the beltline region during the 2nd, 3rd, and 4th inspection intervals. The requirement to examine essentially 100% of the length of all reactor vessel shell welds during the 1st inspection interval has been in Section XI since the Winter 1975 Addenda to the 1974 Edition.

In view of recent information which shows the susceptibility of reactor vessel materials to dgradation, and the limited examinations performed to date on some reactor vessels, the NRC is concerned with the length of time (i.e., as long as 20 years) before a licensee is required to implement the reactor vessel shell weld examinations specified in the 1988 Addenda and 1989 Edition of Section XI through routine updating of its inservice inspection program. The NRC is concerned that the inherent delay in implementing the expanded reactor vessel examinations is inconsistent with the importance of the reactor vessel, with recent new information regarding degradation of reactor vessel materials, with the limited examination of shell welds previously performed on many reactor vessels, and with the need to ensure that the failure probability of the reactor vessel remains extremely low. For these reasons, the NRC has determined that augmented examinations of reactor vessel shell welds are necessary to provide "adequa protection" as addressed in § 50.109.

See Appendix A, "Documented Evaluation for Backfitting," for a detailed description of the basis for the proposed augmented examination of the reactor vessel.

Separation of ISI and IST Requirements

The requirements in § 50.55a for inservice inspection and inservice testing are presently intermingled in § 50.55a(g), "Inservice inspection requirements." This paragraph specifies the requirements for preservice and inservice examinations, and system pressure tests for Class 1, Class 2, and Class 3 components and their supports, and the requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves. In order to emphasize the importance of inservice testing, and to more clearly distinguish the requirements for inservice testing from those for inservice inspection, it is proposed that the present requirement for inservice testing be moved from existing paragraph (g) to a separate, presently reserved, paragraph (f), which would be titled "Inservice testing requirements." It was originally intended that the proposed revision be strictly editorial with only existing requirements for inservice testing in paragraph (g) transferred to paragraph (f), and that all existing requirements for inservice examination and system pressure testing be retained in paragraph (g). However, one noneditorial change in paragraph (f) was determined to be necessary to indicate Commission intent for requiring augmented inservice testing. Thus, similar to the present expression of intent for requiring augmented inservice inspections that is contained in § 50.55a(g)(6)(ii), proposed § 50.55a(f)(6)(ii) specifies that the Commission may require augmented inservice testing programs for pumps and valves for which it is deems that added assurance of operational readiness is necessary. It should be noted that this change does not in and of itself impose a specific new requirement, but states the Commission's intent to do so if it deems necessary.

Comparative text that shows the new § 50.55a(f) against the existing § 50.55a(g) is provided in Appendix B.

2. Objectives

The proposed rule would amend § 50.55a to:

o Incorporate by reference into 10 CFR § 50.55a, the 1986 Addenda, 1987 Addenda, 1988 Addenda and 1989 Edition of Section III, Division 1, of the ASME Code, and the 1986 Addenda, 1987 Addenda, and the 1988 Addenda and 1989 Edition (with a specified modification) of Section XI, Division 1, of the ASME Code.

o Require augmented examination of reactor vessel shell welds.

o Separate in the regulation the requirements for inservice inspection and inservice testing.

## 3. Alternatives

Update of Reference to Section III and Section XI Edition and Addenda

One alternative to incorporating by reference into NRC regulations the updated requirements of Section III, Division 1, and Section XI, Division 1, would be to take no action. This would mean that the NRC position on the methods for construction and inservice inspection contained in recent addenda and the latest edition of the ASME Code would have to be provided on a case-by-case basis, and that would not be desirable.

An alternative to incorporating by reference these later requirements of Section III, Division 1, and Section XI, Division 1, is to incorporate the entire text of these sections of the ASME Code into the NRC regulations. Because of the volume of these sections, this approach is not practicable.

Augumented Examination of Reactor Vessel

An alternative to imposing the augmented examination program for the reactor vessel would be to permit implementation of the requirements for the reactor vessel shell welds in the 1989 Edition of Section XI through the normal ISI program update path specified in § 50.55a(g)(4)(ii). This alternative (i.e., status quo), however, is unacceptable because it could result in some plants not implementing the revised reactor vessel shell weld examination requirements for 20 years after the effective date of this rule.

Separation of ISI and IST Requirements

An alternative to separating the ISI and IST requirements would be to maintain the status quo. This alternative is not desirable because it would do nothing to clarify the existing requirements, and to emphasize the individual importance of both the inservice inspection and inservice testing programs.

## 4. Consequences

### A. Costs and Benefits

Update of Reference to Section III and Section XI Edition and Addenda

Incorporating by reference the latest addenda of the ASME Code will establish the NRC staff position on these Code rules on a generic basis for applicants and licensees. This would minimize the need for case-by-case evaluations and would reduce the time and effort required for submittal preparations and license reviews.

The cost/benefit of ASME Code revisions is balanced by the manner in which these revisions are achieved through the American National Standards Institute (ANSI) consensus process. The ANSI consensus process ensures that participation in ASME Code development is open to all persons and organizations that might reasonably be expected to be directly and materially affected by the activity, and ensures that such persons and organizations shall have the opportunity for fair and equitable participation without dominance by any single interest. Consensus is established when substantial agreement has been achieved by the interests involved. Consensus requires that all views and objections be considered and that a concerted effort be made toward resolution. ASME Code proposed revisions are published for public comment in the ASME Mechanical Engineering and ANSI Reporter publications prior to being submitted for final ASME and ANSI approval. Adverse public comments are referred to the appropriate ASME technical committee for resolution.

The consensus process ensures a proper balance between utility, manufacturing, enforcement, regulatory and other interests concerned with revisions to the ASME Code, and ensures that the cost of implementing any Code revision is consistent with its benefit.

The proposed amendment would incorporate by reference the 1988 Addenda and 1989 Edition with a specific modification. The modification requires implementation of requirements for data analysis and corrective actions that are included in Subsection IWV prior to the 1988 Addenda. Specifically, the modification provided in § 50.55a(b)(vii) would require that when using Subsection IWV in the 1988 Addenda or 1989 Edition of Section XI, Division 1, of the ASME Code, the licensee shall analyze leakage rates for Category A containment isolation valves that do not provide a reactor coolant system pressure isolation function in accordance with paragraph 4.2.2.3(e), and shall take corrective action in accordance with paragraph 4.2.2.3(f) of Part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987.

The proposed modification is not a backfit because it imposes a requirement that licensees now are required to implement in accordance with requirements in Section XI Subsection IWV editions and addenda through the 1987 Addenda. It does, however, represent a deviation from a "clean" endorsement of the Section XI 1988 Addenda and 1989 Edition, and its impact is, therefore, discussed below.

Paragraph 4.2.2.3(e), "Analysis of Leakage Rates," of Part 10, specifies that "Leakage rate measurments shall be compared with the permissible leakage rates specified by the plant Owner for a specific valve or valve combination. If leakage rates are not specified by the Owner, the following rates shall be permissible: ... " Paragraph 4.2.2.3(f), "Corrective Action," of Part 10, specifies that "Valves or valve combinations with leakage rates exceeding the values specified by the Owner in (e) above shall be declared inoperable and either repaired or replaced. A retest demonstrating acceptable operation shall be performed following any required corrective action before the valve is returned to service."

Subsection IWV paragraph IWV-3426, "Analysis of Leakage Rates," of Section XI editions and addenda through the 1987 Addenda specifies that "Leakage rate measurements shall be compared with previous measurements and with the permissible leakage rates specified by the Plant Owner for a specific valve. If leakage rates are not specified by the Owner, the following rates shall be permissible: ... " Additionally, Subsection IWV paragraph IWV-3427, "Corrective Action," presently specifies that "(a) Valves with leakage rates exceeding either the values specified by the Owner or those rates given in IWV-3426 shall be replaced or repaired. (b) For valves NPS 6 and larger, if a leakage rate exceeds the rate determined by the previous test by an amount that ... the test frequency shall be doubled; ... "

The difference between the proposed requirement (OMa-1988 Part 10) and the existing requirement (Section XI Subsection IWV through the 1987 Addenda) is:

(1) Analysis of leakage rates: Part 10 requires leakage rates to be compared with permissible leakage rates for a specific valve or valve combination, whereas existing Section XI requirements apply only to a specific valve.

(2) Corrective action: Part 10 requires valve or valve combinations that exceed specified leak rates to be declared inoperable and to be repaired or replaced, whereas the existing Section XI requirement again relates only to a specific valve and, although requires the valve to be repaired or replaced, makes no statement about declaring a valve inoperable. In addition, Part 10 does not include the IWV-3427(b) trending requirement for valves NPS 6 and larger.

The proposed modification to the 1988 Addenda and 1989 Edition of Section XI that is specified in § 50.55a(b)(2)(vii) would require that leakage rates continue to be analyzed for containment isolation valves that do not provide a reactor coolant system pressure isolation function. However, because paragraph 4.2.2.3(e) of Part 10 is specified in the proposed modification, rather than existing IWV-3426, the existing requirement is relaxed by permitting valve combinations rather than specific valves to be analyzed. This recognizes that in the past requests for relief have been granted where design constraints necessitate testing combinations of valves. The proposed modification would also require that valves, or groups of valves, that exceed permissible leakage rates be declared inoperable and be repaired or replaced. Since the testing of containment isolation valves is generally performed during plant shutdown, having to declare a defective valve inoperable prior to repair or replacement is not considered significant in that it would not require a technical specification action statement to be entered. The proposed modification would not require the present practice of trending of NPS 6 and larger valves, because that requirement has not been carried from IWV-3427(b) to Part 10.

<u>Augmented Examination of Reactor Vessel</u>

The NRC has concluded, on the basis of the documented evaluation required by 10 CFR 50.109(a)(4), that the backfit requirements associated with the augmented examination of the reactor vessel are necessary to ensure that the facility provides adequate protection to the public health and safety, and, therefore, that a backfit analysis is not required and the cost-benfit standards of 10 CFR 50.109 (a)(3) do not apply. The documented evaluation provided in Appendix A of this regulatory analysis includes a statement of the objectives of and reasons for the backfit that would be required by the proposed rule and sets forth the basis for the NRC's conclusion that the backfit is not subject to the cost-benefit standard of 10 CFR 50.109 (a)(3).

Separation of ISI and IST Requirements

Separation of the ISI and IST requirements in the regulation is essentially editorial with no specific new requirements and no additional costs accrued to the licensees. Inclusion of a provision in new § 50.55a(f)(6)(ii) that expresses the intent of the Commission to require augmented inservice testing if deemed necessary does not impose a specific new requirement or cost at this time. The need for a cost/benefit analysis would be determined when, and if, an augmented inservice test program were required in the future.

B. Impacts on Other Requirements

(1) Effect of Proposed Amendment on Existing NRC Requirements

Update of Reference to Section III and Section XI Edition and Addenda

In the past, Section III and Section XI have been revised twice a year. These revisions have been published in two addenda each year (i.e., Summer Addenda and Winter Addenda). Starting in 1986, there has been only one addenda for each of these sections and it is called the 19XX Addenda (e.g., 1988 Addenda). The revisions are the result of consensus participants meeting 4-5 times a year for the purpose of improving the existing rules. The revisions take into account the many lessons learned in a specific area since the development of a particular Code rule. The revisions generally fall into three categories: (i) technical revisions that incorporate new rules in technical areas not previously addressed by the Code; (ii) technical revisions to existing rules; and (iii) editorial revisions. When a technical revision is made, it may make the existing set of rules more or less restrictive, or may simply clarify the existing rule without changing its intent. There are numerous revisions in each addenda. In general, technical revisions are made to improve the ASME Code by providing more detailed rules where experience indicates greater guidance is necessary, or relaxing the rules where experience shows equivalent operational safety can be maintained with a reduced burden on the licensee.

Relative to implementation of Section III, Division 1, § 50.55a specifies that the ASME Code edition and Addenda to be applied to reactor coolant pressure boundary (i.e., Class 1), and Quality Group B (i.e., Class 2) and Quality Group C (i.e., Class 3) components must be determined by the provisions of paragraph NCA-1140 of Subsection NCA of Section III, Division 1, of the ASME Code, but the applicable edition and addenda must be those which are incorporated by reference in § 50.55a. NCA-1140 specifies that the owner (or his designee) shall establish the ASME Code edition and addenda to be included in the Design Specifications, but that in no case shall the Code edition and addenda dates established in the Design Specifications be earlier than three years prior to the date that the nuclear power plant construction permit is docketed. NCA-1140 further states that later ASME Code editions and addenda may be used by mutual consent of the Owner (or his designee) and Certificate Holder. Plants may implement the improved rules on a voluntary basis as they are incorporated by reference into § 50.55a, but unless they make that choice, there is no additional burden associated with incorporating the proposed Section III edition and addenda.

Relative to implementation of Section XI, Division 1, § 50.55a specifies that:

(a) Inservice examinations of components, inservice tests of pumps and valves, and system pressure tests conducted during the initial 120-month inspection interval shall comply with the requirements in the latest edition and addenda of the ASME Code incorporated by reference on the date 12 months prior to the date of issuance of the operating license, subject to any limitations noted (§ 50.55a(4)(i)).

(b) Similar to (a.), above, for successive 120-month inspection intervals, it is necessary to comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference 12 months prior to the start of the 120-month inspection interval, subject to any limitations noted (§ 50.55a(4)(ii)).

(c) If a licensee determines that conformance with certain Code requirements is impractical for his facility, the licensee shall notify the Commission and submit information to support his determination (§ 50.55a(g)(5)(iii)). The Commission will evaluate licensee determinations that Code requirements are impractical and may grant such relief and may impose alternative requirements giving due consideration to the burden on the licensee (§ 50.55a(g)(6)(i)).

The existing requirements in § 50.55a specified in the items (a) and (b), above, ensure that all plants perform inservice inspection and inservice test programs in conformance with updated versions of Section XI of the ASME Code. The proposed amendment would update the editions and addenda that are endorsed by the NRC staff, and would thereby cause these later editions and addenda to be implemented by licensees consistent with the time constraints identified in Items (a) and (b), above.

The existing requirement specified in item (c), above, provides for the submittal of relief requests by licensees. It ensures that in those cases where the generic requirements of Section XI are impractical, or are overly burdensome for a specific facility, that facility may obtain relief from the particular requirement, provided the licensee demonstrates to the Commission that omission of the Section XI requirement believed to be impractical will not have an adverse affect on public health and safety.

The proposed revisions to the ASME Code generally improve plant safety by incorporating new rules to cover areas not previously addressed, or by revising the rules consistent with experience to reduce the number of areas where the Code has been found to be impractical, inadequate, or insufficiently clear. The revisions were developed through the consensus process and, therefore, have been thoroughly reviewed by all elements of the nuclear industry, and the NRC staff. It is fair to say that the consensus process ensures that the burden of any revision to the ASME Code is balanced by its value to the industry and to protecting the public health and safety.

As noted above, § 50.55a presently requires that licensees update their inservice inspection programs every 10 years to the Section XI rules that were endorsed by the NRC 12 months prior to the start of the next 120-month inspection interval. There will be a substantial increase in safety through the endorsement of the later addenda because it will be these addenda that will be used in subsequent inservice inspection programs. Obsolete requirements will not be continued from one 120-month inspection interval to another.

NRC Generic Letter No. 89-04, "Guidance on Developing Acceptable Inservice Testing Programs" provides staff positions that should be incorporated into inservice test programs developed in accordance with Section XI Subsections IWP and IWV. The generic letter was developed consistent with Section XI references prior to the 1988 Addenda. In the 1988 Addenda and 1989 Edition of Section XI, the requirements in Subsections IWP and IWV were replaced in their entirety by references to Part 6 and Part 10, respectively, of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987. The staff positions delineated in the generic letter have been reviewed by the staff and found to be applicable to and not inconsistent with the rules provided in Part 6 and Part 10.

Augmented Examination of Reactor Vessel

The 1988 Addenda and 1989 Edition of Section XI, Division 1, require examination of essentially 100% of the length of all reactor vessel shell welds during the 2nd, 3rd, and 4th inspection intervals (Section XI has required examination of essentially 100% of the length of reactor vessel shell welds during the 1st interval since the 1974 Edition as modified by addenda through the 1975 Addenda). Section 50.55a(g)(4)(ii) requires that inservice examinations during successive (i.e., 2nd, 3rd, and 4th) inspection intervals comply with the requirements of the latest edition and addenda of Section XI incorporated by reference in § 50.55a(b) 12 months prior to the start of the 120-month inspection interval. Therefore, through routine updating of the individual plant inservice inspection programs, each licensee would ultimately implement the new requirements for reactor vessel shell weld examinations. However, as noted in paragraph 1(b), above, and described fully in Section 1 of Appendix A, implementation through routine updating could take as long as 20 years.

The proposed augmented examination would require all plants to implement the reactor vessel shell weld examinations specified in the 1989 Edition of Section XI during the inspection interval in force when the proposed rule becomes effective, but would permit plants with fewer than 40 months remaining in their present inspection interval to implement the examination requirements during the first period of the next inspection interval. The deferred examination may not be used as a substitute for the routine reactor vessel shell weld examination scheduled for that next interval. The augmented examination may be used as a substitute for the reactor vessel shell weld examinations scheduled for implementation during the inspection interval in effect at the time the proposed rule becomes effective. This would result in all plants implementing the revised reactor vessel shell weld examinations specified in the 1989 Edition of Section XI of the ASME Code within 120 months (i.e., the length of one inspection interval) of the effective date of the rule.

Separation of ISI and IST Requirements

The proposed amendment includes an editorial revision to § 50.55a that would remove the requirements for inservice testing from § 50.55a(g) and would place those requirements in § 50.55a(f) which is presently reserved. Section

50.55a(g) then would contain requirements for only inservice inspection (i.e., inservice examinations, and system pressure tests). Section 50.55a(f) would be titled, "Inservice testing requirements" and § 50.55a(g) would retain its present title, "Inservice inspection requirements." This revision is essentially editorial and, therefore, will have no impact on any NRC requirement. Comparative text that shows the new § 50.55a(f) against the existing
§ 50.55a(g) change is provided in Appendix B.

(2) Effect of Proposed Amendment on Other NRC Regulatory Actions

Update of Reference to Section III and Section XI Editions and Addenda

No effect on other NRC regulatory actions.

Augmented Examination of Reactor Vessel

Eliminates need for NRC Generic Letter which has been drafted on subject.

Separation of ISI and IST Requirements

No effect on other NRC regulatory actions.

(3) Application of Backfit Rule (§ 50.109)

Update of Reference to Section III and Section XI Editions and Addenda

It is the opinion of the Office of the General Counsel that proposed amendments to § 50.55a that simply update the existing reference to edition and addenda of Section III and Section XI of the ASME Code and do not impose limitations and modifications should not be subjected to the backfit provisions in 10 CFR § 50.109. The rationale is that, (1) the Section III, Division 1, update applies only to new construction (i.e., the edition and addenda to be used in the construction of a plant are selected based upon the date of the construction permit and are not changed thereafter, except voluntarily by the licensee), (2) licensees are fully aware that § 50.55a requires that they update their inservice inspection program every 10 years to the latest edition and addenda of Section XI that were incorporated by reference in § 50.55a 12 months prior to the start of the next inspection interval, and (3) endorsing and updating references to the ASME Code, a national consensus standard developed by participants (including the NRC) with broad and varied interests, is consistent with both the intent and spirit of the backfit rule (i.e., NRC provides for the protection of the public health and safety, and does not unilaterally impose an undue burden on applicants or licensees).

This proposed amendment does impose a modification on the use of Subsection IWV as contained in the 1988 Addenda and 1989 Edition of Section XI. However, as discussed in Section 4.A, above, the modifcation is not a backfit because it is essentially a continuation of a present requirement with certain relaxations.

Augmented Examination of Reactor Vessel

The NRC has concluded that the backfit requirements contained in this proposed amendment for augmented examination of the reactor vessel shell welds are necessary to ensure that the facility provides adequate protection to the public health and safety, and, therefore, that a backfit analysis is not required and the cost-benefit standards of 10 CFR 50.109 (a)(3) do not apply. § 50.109(c) identifies 9 factors to be considered, as appropriate, in evaluating the proposed backfit. Each of the 9 factors is addressed, as appropriate, in the documented evaluation provided in Appendix A.

Separation of ISI and IST Requirements

No backfit.

(4) Impact on Requirements of Other Government Agencies

Implementation of the new ASME Code rules imposes certain additional information collection requirements. The Supporting Statement for Information Collection Requirements in 10 CFR § 50.55a is provided in Appendix C.

This amendment to § 50.55a affects only the licensing and operating of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act in the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this amendment does not fall in the province of this Act. The proposed rule would have no significant effect on a substantial number of small companies.

(5) Decision Rationale

From the above analysis, the NRC staff concludes that this proposed amendment to 10 CFR § 50.55a to update the reference to incorporate recent addenda of the ASME Code, to impose an augmented examination of reactor vessel shell welds, and to separate the requirements for inservice inspection and inservice testing, would result in a net increase in the overall protection of public health and safety. This is because the improved rules would be used in new and subsequent inservice inspection programs and would save applicants, licensees, and the NRC staff both time and effort by providing uniform detailed criteria against which the staff could review any single submission; the augmented reactor vessel examination would help ensure the continued low probability of failure for reactor vessels; and the separation of inspection and testing requirements would emphasize the importance that NRC places on both inservice inspection and inservice testing programs.

(6) Implementation

Update of Reference to Section III and Section XI Edition and Addenda

No implementation problems are anticipated. The framework for implementation is well established in both the industry and the NRC.

Augmented Examination of Reactor Vessel

Programmatic implementation (e.g., development of plans, preparation of procedures, documentation of results) of the augmented reactor vessel examination should be much like that of other inservice examination requirements and no problems are anticipated. However, the difficulty associated with performing the actual examinations will vary greatly between plant designs. In general, most PWRs will be able to implement the proposed augmented examination without appreciable difficulty, whereas, the various BWR designs will have significantly greater difficulty because of accessibility problems. In certain cases, it may be necessary to grant exemptions from the full requirement of the proposed augmented reactor vessel examination. See Appendix A, Section 7 and 8.

Separation of ISI and IST Requirements

No implementation problems are anticipated.

## Appendix A

## Documented Evaluation
## for
## Backfitting

Paragraph (a)(4) of Section 50.109, "Backfitting," specifies that a backfit analysis is not required where the Commission or staff, as appropriate, finds and declares, with a documented evaluation for its finding, that the regulatory action is necessary (i.e., consistent with § 50.109(a)(4)(ii)) to ensure that the facility provides adequate protection to the health and safety of the public and is in accord with the common defense and security.

The NRC has concluded that the backfit requirement contained in this proposed amendment for augmented examination of the reactor vessel shell welds is necessary to ensure that the facility provides adequate protection to the public health and safety, and, therefore, that a backfit analysis is not required and the cost-benefit standards of 10 CFR 50.109 (a)(3) do not apply. Section 50.109(c) identifies 9 factors that shall be considered, as appropriate, in evaluating the proposed backfit. The response to each of the 9 factors is provided, below, in this documented evaluation.

1. Statement of the specific objectives that the proposed backfit is designed to achieve.

a. Statement of Objective

The NRC staff is concerned that because of new information which shows the susceptibility of reactor vessel materials to degradation, and the limited examinations performed to date on some reactor vessels, the time-frame to routinely implement (i.e., through routine ten-year updating of licensees' inservice inspection programs) the expanded reactor vessel shell weld examination that has been incorporated into the 1988 Addenda and 1989 Edition of Section XI of the ASME Code is inconsistent with the importance of the reactor vessel. It is the staff's judgement that restoration of the level of protection presumed by the regulations requires more than compliance to existing requirements. The proposed augmented examination would ensure that all plants examine essentially 100% of the length of all reactor vessel shell welds during the inservice inspection interval in effect when this proposed rule becomes effective, or at the latest during the first period of the following inspection interval.

b. Basis for Required Objective

(1) Summary of Basis

The 1988 Addenda and 1989 Edition of Section XI incorporate a provision that requires examination of essentially 100% of the length of all reactor vessel shell welds during the 2nd, 3rd, and 4th inspection intervals. The staff's concerns regarding the need for revising Section XI to include such expanded reactor vessel examination requirements was previously transmitted to the ASME Subcommittee on Inservice Inspection in a letter from T. E. Murley (Director, NRR) to S. H. Bush (Chairman, ASME Subcommittee on Inservice Inspection), dated March 14, 1988. The desirablility of implementing an examination of all accessible shell welds in BWR reactor vessels to maintain "the appropriate defense-in-depth" was expressed in a letter from W. Kerr (Chairman, ACRS) to the Honorable L. W. Zech, Jr. (Chairman, USNRC), dated June 7, 1988. In this letter Mr. Kerr noted that "Although we believe that catastrophic failure of a BWR pressure vessel should continue to remain outside the design basis, recent experience demonstrates that flaws can grow from the coolant side of the pressure vessel into the steel pressure boundary."

Section XI has required examination of essentially 100% of the length of all reactor vessel shell welds during the 1st inspection interval since the Winter 1974 Addenda. The requirements of the 1989 Edition of Section XI would be implemented as a routine update to the inservice inspection program in accordance with § 50.55a(g)(4)(ii). This could result in plants not implementing the revised reactor vessel examination from 10 years and 4 months to as long as 20 years from the effective date of this rule. The staff considers this time-frame for routine implementation of the revised reactor vessel examination to be inconsistent with the importance of the reactor vessel, with recent new information regarding degradation of reactor vessel materials, with the limited examination of shell welds previously performed on many reactor vessels, and the need to ensure that the failure probability of the reactor vessel remains extremely low.

The proposed augmented inspection would impose the reactor vessel examination requirements in the 1989 Edition for 2nd, 3rd, and 4th intervals on all plants. However, plants with fewer than 40 months remaining in their present inspection interval would be permitted to defer the augmented examination until the first period of the next inspection interval. Other plants would implement the examination as part of their next inspection interval.

(2) Detailed Basis

The 1988 Addenda and 1989 Edition of Section XI, Division 1, require examination of essentially 100% of the length of all reactor vessel shell welds during the 2nd, 3rd, and 4th inspection intervals (Section XI has required examination of essentially 100% of the length of all reactor vessel shell welds during the 1st interval since the 1974 Edition as modified by addenda through the 1975 Addenda). Section 50.55a(g)(4)(ii) requires that inservice examinations during successive (i.e., 2nd, 3rd, and 4th) inspection intervals comply with the requirements of the latest edition and addenda of Section XI incorporated by reference in § 50.55a(b) 12 months prior to the start of the 120-month inspection interval. Therefore, through routine updating of the individual plant inservice inspection programs, each licensee would ultimately implement the new requirements for reactor vessel shell weld examinations. However, as described in the tables, below, implementation through routine updating could take as long as 20 years.

§ 50.55a(b)(2). The following table shows the maximum allowable times for plants to implement the revised reactor vessel inspections based upon the routine update requirements of § 50.55a(g)(4)(ii).

Time-Frames Associated With Implementing Reactor Vessel Examinations Specified In 1989 Edition Through Routine Updating

| Inspection Period Plant Just Entering | Maximum Allowable Time to Perform Revised Reactor Vessel Examination[1][2] |
|---|---|
| 1st | 240 months |
| 2nd | 200 months |
| 3rd | 160/240[3] months |

[1] for 2nd, 3rd, or 4th inspection intervals
[2] based on deferring examination until end of interval, as permitted by Subsection IWB Table IWB-2500-1.
[3] a plant with less than 12 months remaining would not have to implement the 1989 Edition until the inservice inspection interval that follows the next interval.

As can be seen from the above table, depending upon which inspection period a particular plant happens to be in when this proposed rule becomes effective, the maximum allowable time to perform the expanded reactor vessel examinations ranges from 20 years (240 months) to 13 years and 4 months (160 months). The NRC is concerned that these time-frames are generally inconsistent with the importance of the reactor vessel, the limited examinations performed to date on many reactor vessels, and recent evidence of the susceptibility of reactor vessel materials to degradation.

From the standpoint of safety, the reactor vessel is arguably the most important single component in the nuclear steam supply system. All PRA studies assume a very low probability for failure of the reactor vessel. An effective inse vice inspection program is necessary, in view of evidence that reactor vessel materials are subject to degradation mechanisms, to ensure that the failure rate of the reactor vessel remains very low. These degradation mechanisms are evidenced by: 1) Results from irradiation surveillance material tests which show that, contrary to earlier predictions, certain vessel materials will undergo significant radiation damage (this information was used in the development of Revision 2 to Regulatory Guide 1.99; and 2) The following information, which shows stress corrosion cracking of the reactor vessel material in BWRs to be more probable now than it was several years ago:

a. Stress corrosion cracking initiated in Inconel weld metal has propagated into reactor vessel low alloy steels in BWRs.

b. Radiation damage to metals appears to increase the suscepti-bility to stress corrosion cracking. The increase in strength

The 1988 Addenda of Section XI revises the extent of reactor vessel shell weld examinations required by Item B1.10, "Shell Welds" in Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel" of Table 2500-1 of Subsection IWB "Requirements for Class 1 components of Light-Water Cooled Power Plants." The revision, which is also contained in the 1989 Edition, extends the existing requirement to perform a volumetric examination on all reactor vessel shell welds during the 1st inspection interval to successive intervals (i.e., 2nd, 3rd, and 4th intervals).

Section XI requirements for examination of the reactor vessel shell welds prior to the 1988 Addenda are as follows:

- o 1974 Edition (with addenda through the 1975 Winter Addenda) through the 1986 Edition (with addenda through the 1987 Addenda):
  - 1st inspection interval: Volumetric examination of all reactor vessel shell welds.
  - Successive inspection intervals: Volumetric examination of one circumferential and one longitudinal beltline weld.
- o 1971 Edition through the 1974 Edition (with addenda through the 1975 Summer Addenda)
  - 1st inspection interval: Volumetric examination of 5% of the length of each circumferential weld and 10% of the length of each longitudinal weld.
  - Successive inspection intervals: Same as for the 1st inspection interval.

Sections 50.55a(g)(4)(i) and (ii), respectively, require that licensees use the latest editions and addenda of the ASME Code approved by the NRC 12 months prior to the initial 120-month inspection interval, and 12 months prior to the start of successive 120-month inspection intervals.

Consistent with the updating requirements of the regulations, and the changing requirements of Section XI, some inservice inspection programs have resulted in very limited examinations of the reactor vessel shell welds. For example, if the first 120-month examinations were performed to the 1974 Edition of Section XI, 5% of the length of each circumferential weld and 10% of the length of each longitudinal weld would have been examined. If examinations during the second 120-month inspection interval were performed to the 1980 Edition (with addenda through the Winter 1982 Addenda), one circumferential weld and one longitudinal weld would have been examined.

The update requirement for successive inspection intervals (i.e., 50.55a(g)(4)(ii)) would result in routine implementation of the provisions for reactor vessel examinations in the 1989 Edition the next time a utility updated its inservice inspection program, if, at the time of selecting the applicable edition/addenda (i.e., 12 months prior to start of the next inspection interval), the 1989 Edition had been incorporated by reference into level and hardness alone would be expected to cause this, but other more subtle changes may also increase susceptibility.

c. Tests on irradiated, nonsensitized stainless steel show that the steel is susceptible to stress corrosion cracking in BWR water; tests also show that material tested while being exposed to radiation is more susceptible.

d. Intergranular stress corrosion cracking of the reactor vessel cladding is generally considered to be present; such cracks, by a crevice effect could increase the susceptibility of the vessel material to stress corrosion or corrosion fatigue.

e. Cracking has occurred in the Inconel 182 attachment weld in a BWR reactor vessel.

f. Stress corrosion cracking has occurred in PWR steam generator shell material that is essentially identical to that in reactor vessels.

The NRC is concerned that the importance of the reactor vessel and the availability of degradation mechanisms is inconsistent with the time frame, associated with routine updating of the inservice inspection program, to implement the examination of essentially 100% of the length of all reactor vessel shell welds as specified in the 1988 Addenda and 1989 Edition of Section XI. Therefore, the NRC is proposing an augmented examination of the reactor vessel, as addressed in § 50.55a(g)(6)(ii). The proposed augmented examination would require that all licensees examine essentially 100% of the length of all reactor vessel shell welds in their present inspection interval. However, those plants with fewer than 40 months remaining in their present inspection interval would be permitted to defer the augmented examination until the first period of the next inspection interval. The following table shows the significance of the proposed expedited implementation procedure:

Time-Frames Associated With Implementing
Reactor Vessel Examinations
Through Augmented Examination

| Inspection Period Plant Just Entering | Maximum Allowable Time To Perform Augmented Reactor Vessel Examinatons[1] |
|---|---|
| 1st | 120 months |
| 2nd | 80 months |
| 3rd | 40/80[2] months |

[1] based on deferring examination until end of interval, as permitted by Subsection IWB Table IWB-2500-1
[2] with augmented examination deferred to first period of next inspection interval

An alternative to imposing the augmented examination program for the reactor vessel would be to permit implementation of the requirements for the reactor vessel shell welds in the 1989 Edition of Section XI through the normal ISI program update path specified in § 50.55a(g)(4)(ii). This alternative (i.e., status quo), however, is unacceptable because it could result in some plants not implementing the revised reactor vessel examination requirements for 20 years after the effective date of this rule.

2. <u>General description of the activity that would be required by the licensee or applicant in order to complete the backfit.</u>

The proposed augmented examination of reactor vessel shell welds would apply to all 112 nuclear power plants with operating licenses. The activity required by each licensee would depend upon which provisions of the proposed rule for augmented examination would apply. The proposed rule would result in three catgories of implementation and thereby timing of associated activities. These categories are:

- o Licensee has not previously examined essentially 100% of the length of all reactor vessel shell welds and has 40 or more months remaining in the inservice inspection interval in effect when the proposed rule becomes effective ---- Such licensees would be required to modify their inservice inspection program and to implement the augmented examination during the ongoing inspection interval.

- o Licensee has not previously examined essentially 100% of the length of all reactor vessel shell welds and has fewer than 40 months remaining in the inservice inspection interval in effect when the proposed rule becomes effective ---- Such licensees could perform the augmented examination during the ongoing inspection interval, but would likely defer the examination, as permitted by the proposed rule, until the first period of the next inspection interval. Such licensees would be required to modify their inservice inspection program and implement the augmented examination in the appropriate time-frame.

- o Licensee has previously examined essentially 100% of the length of the reactor vessel shell welds during the inspection interval in effect when the proposed rule becomes effective, or is scheduled to implement such an examination (e.g., written commitment, or 1st inspection interval examination) ---- No further action is required by such licensees to satisfy the requirements of the proposed augmented examination.

As appropriate, each licensee would be required to revise their plant's inservice inspection program to include examination of essentially 100% of the length of all reactor vessel shell welds in accordance with Section XI Subsection IWB, Table 2500-1, Examination Category B-A, Item B 1.10 and to implement that examination. A report documenting the results of the augmented examination would be submitted to the appropriate Regional Administrator.

Potential difficulties assciated with implementing the proposed augmented examination for reactor vessels are addressed in Item 8, below.

In view of the fact that certain licensees have previously requested, and received, relief from Section XI requirements for examining reactor vessel shell welds, it is reasonable to assume that some of the same licensees will request an exemption from the proposed augmented reactor vessel examination, in accordance with § 50.12, "Specific exemptions." See Item 7, below, for further discussion on this item.

3. <u>Potential change in the risk to the public from the accidental off-site release of raioactive material.</u>

The potential change in risk is not quantifiable, since the extent of existing reactor vessel degradation is not known because of limited previous examinations on operating reactor vessels. However, implementation of the proposed augmented examination for reactor vessels is important for maintaining the validity of results for the quantification of reactor vessel rupture accidents quoted in Appendix V of WASH-1400 (NUREG-75/014) "Reactor Safety Study," October 1975, which were abstracted from WASH-1286 "Report on the Integrity of Reactor Vessels for Light-Water Power Reactors" by the Advisory Committee on Reactor Safeguards (ACRS), January 1974. The ACRS report (Section 5.9, "Probability of Disruptive Failure of a Reactor Vessel -- Committee Appraisal") states, in part, that:

> "It is the opinion of the Committee that the disruptive failure probability of nuclear reactor vessels is significantly lower than that of the non-nuclear vessels evaluated in the preceding sections. This is based on the following:
>
> > " 1. ... the Committee has concluded that the failure rate of Section III reactor vessels is lower than that of Section I boiler drums. This conclusion is based on differences in design, fabrication and inspection, materials, and operating conditions ... Of particular importance are the preoperational and inservice inspection measures required by Section XI of the Code.
>
> " ... Accordingly the Committee concludes that there is a reasonable assurance that the disruptive failure rate of reactor vessels designed, constructed and operated in accordance with Code Sections III and XI is less than $1x10^{-6}$ per vessel year..."

The proposed augmented examination, which could expedite by as much as 10 years the implementation of the requirement in the 1989 Edition of Section XI, to examine essentially 100% of all reactor vessel shell welds would contribute significantly in maintaining the estimated low probability for disruptive failure of the reactor vessel.

4. <u>Potential impact on radiological exposure of facility employees.</u>

The 1974 Edition of Section XI, with addenda through the Winter 1975 Addenda, through the 1986 Edition, with addenda through the 1987 Addenda, require volumetric examination of one circumferential and one longitudinal beltline weld of the reactor vessel during the 2nd, 3rd, and 4th inspection intervals. The occupational radiation exposure associated with equipment setup and examination of the two beltline welds has been measured to be 1.80 person-rem (i.e., 1.48 person-rem for the circumferential and 0.32 person-rem for the longitudinal weld) in a BWR-4 facility. The occupational radiation exposure associated with equipment setup and examination of essentially 100% of the length of the reactor vessel shell welds, as specified in the 1988 Addenda and 1989 Edition of Section XI, and required by the proposed augmented examination, is estimated to be 8.6 person-rem (i.e., 3.63 and 4.97 person-rem for all circumferential and longitudinal welds, respectively) for the same facility. The preceding estimate of occupational radiation exposure for the proposed augmented examination is based upon radiation measurements made during present examinations and extrapolations from these measurements to the expanded reactor vessel shell weld examinations.

The estimated occupational radiation exposure of 8.6 person-rem to implement the proposed augmented reactor vessel examination should be bounding for BWR-4/5/6 designs. In a BWR-6, essentially all reactor vessel welds are designed to be accessible for examination from the outside diameter. ALARA considerations have been factored into the inspection concept. Therefore, the occupational exposure should be significantly below the estimate for a BWR-4 design.

A new examination tool is currently under development by the industry to provide more comprehensive coverage of BWR reactor vessels. For older BWRs, the occupational exposure could be below the estimated value of 8.6 person-rem for a BWR-4, or could be somewhat higher, depending upon economic decisions made by the licensee regarding plant specific access to the reactor vessel. Examination from the vessel inside diameter could impact existing refueling outage schedules. If a licensee elects to remove insulation and manually perform examinations of accessible welds, the occupational exposure will be high. This concept would minimize the impact on the refueling schedule. If the licensee elects to use the remote inspection tool to the maximum extent practical, the occupational exposure would be minimized, however, the inservice inspection may increase the length of the refueling outage.

For PWRs, the additional radiation exposure that would be incurred as a result of implementing the proposed augmented reactor vessel examination should be extremely small. PWRs are defueled and their internals are removed to permit installation of a remote positioning tool. These operations must be performed whether two or all the reactor vessel welds are examined. The incremental increase in occupational radiation exposure primarily would be the result of additional instrumentation placements and calibrations.

As indicated above, a reasonable estimate of the maximum occupational radiation exposure that is incurred at a BWR facility to perform the present two weld beltline examination is 1.8 person-rem, and to perform the proposed augmented examination of essentially 100% of the length of all reactor vessel shell welds would be 8.6 person-rem. It must be recognized that the value of 8.6 person-rem is an estimate, and although considered to be close to a bounding value, actual exposures at specific facilities could vary widely based upon facility design, examination concepts, and actual procedures. With this in mind, these occupational radiation exposures are not considered to be insignificant (i.e., 10 CFR § 20.101, "Radiation dose standards for individuals in restricted areas" permits a whole body dose of 1.25 rem per calendar quarter), but are considered reasonable recognizing that no individual would receive more than the allowable dose and that the benefit would be a considerably better awareness of the structural integrity of the reactor vessel.

To put the above exposure levels in context with other exposures incurred at nuclear power plants, it should be noted that the total occupational radiation exposure incurred by workers for all functions at light-water reactor nuclear power plants in the United States during 1986 (i.e., the latest year for which occupational radiation exposures have been compiled and documented in NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1986," Vol. 8, August 1989) was approximately 43,000 person-rems. This represented 19,000 person-rem for BWRs (based upon 30 reactors, each having been in commercial operation at least one full year by the end of 1986) and 24,000 person-rem for PWRs (based upon 59 reactors). Approximately 8.3% (3569) person-rem) of this total exposure was the direct result of inservice inspection functions at these facilities. Inservice inspection functions at BWRs imposed 6.06% (1151 person-rem total or 39 person-rem per plant averaged over 30 plants) and at PWRs imposed 10.04% (2410 person-rem total or 41 person-rem per plant averaged over 59 plants) of this annual total for the year 1986. For reasons explained above, the proposed augmented examination is not expected to have a significant effect on the occupational exposure incurred at PWRs during inservice inspection functions. If 6.8 person-rem is taken as the average difference in occupational exposure between that incurred during the proposed augmented examination and the two-weld beltline examinations that are presently being performed (i.e., [8.6 - 1.8] person-rems), it is seen that the one-time averaged supplemental exposure of 6.8 person-rem would represent approximately 17% of the exposure incurred at BWRs for inservice inspection functions in the year 1986, and 1% of the total occupational exposures for that year..

5. <u>Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay.</u>

Not applicable -- cost is not a basis for evaluation. The NRC has concluded that the backfit requirement contained in this proposed amendment for augmented examination of the reactor vessel shell welds is necessary to ensure that the facility provides adequate protection to the public health and safety, and, therefore, that a backfit analysis is not required and the cost-benefit standards of § 50.109(a)(3) do not apply.

6. <u>The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements.</u>

No potential safety impact. The proposed augmented examination, which would be performed during a scheduled plant shutdown, would not result in any significant change in plant or operational complexity.

7. <u>The estimated resource burden on the NRC associated with the proposed backfit and the availability of such resources</u>.

The augmented examination for reactor vessels would be incorporated into the ongoing Section XI inservice inspection program. The proposed examination could, under certain circumstances, be used as a substitute for the reactor vessel examination for the inservice inspection interval in effect at the time the proposed rule becomes effective. The results of the augmented examination would be submitted along with the results of the routine inservice inspection to the appropriate Regional Administrator.

If a licensee implements the proposed augmented examination without exception, there would be no additional burden to the NRC staff unless an evaluation of results associated with the examination required, in accordance with Section XI, review by the NRC.

In the past, the staff has, upon request by licensees, granted certain plants relief from beltline weld examinations on the basis of impracticality, early plant life, cost, and the fact that at that time no known degradation mechanism existed that could affect the safety of the reactor vessel. The staff now has technical reasons for concern regarding the radiation induced degradation of reactor vessel materials. Many of the same plants that, in the past, requested relief from reactor vessel shell weld examinations will likely request relief from the proposed augmented examination for the same reasons. Because this is considered an issue of "adequate protection" in accordance with 10 CFR 50.109, cost cannot be a consideration, but the issue of accessibility could be. Since the issue of accessibility is intertwined with cost, the staff will have to make a determination on when a reactor vessel shell weld is truly inaccessible for examination.

Previous requests by licensees for relief from ASME Code provisions were made in accordance with procedures defined in § 50.55a(a)(3), which permit the Director of the Office of Nuclear Reactor Regulation to authorize such relief. Requests for exemption from the proposed augmented examination would have to be made in accordance with § 50.12, "Specific Exemptions," which provides the special circumstances under which such exemptions would be granted.

8. <u>The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed backfit.</u>

The difficulty associated with implementing the examination will vary extensively from plant to plant. The weld configuration of reactor vessels differs with each manufacturer and nuclear steam supply system vendor. In spite of 10 CFR 50, Appendix A, Criterion 32, "Inspection of reactor coolant pressure boundary," which states that the reactor coolant pressure boundary components shall be designed to permit periodic inspection and testing of important areas and features to assess their structure and leak-tight integrity, many BWRs will have considerable difficulty implementing the proposed augmented examination for reason of accessibility. During the original licensing process, serious questions arose regarding whether the BWR design met the requirements of Criterion 32. It was determined that, although inconvenient and expensive, access could be provided to examine the shell welds. Therefore, a finding could be made that the plants would be in conformance with Criterion 32 and could be licensed.

PWRs are designed to facilitate examination of reactor vessel shell welds, as internal components can be removed to allow remote, automatic examination of all longitudinal and circumferential shell welds from the inside of the vessel. Some PWRs also have access for examining such welds from the outside of the vessel. All PWRs (except Yankee Rowe) have complied with Section XI requirements for examination of reactor vessel shell welds, with only minor exceptions for small portions of welds obstructed by such items as surveillance capsule brackets.

Most BWRs are not designed to provide easy access for such examinations. BWRs, with the exception of BWR-5 and 6 plants, only have convenient access to examine 5% to 10% of the core beltline welds. Access for examination of the core beltline shell welds in most BWRs is difficult for two reasons. First, access from the outside of the reactor vessel is restricted because the concrete biological shield is very close to the vessel, and the space between them contains insulation that is not designed to be removable. Second, access from the inside is restricted primarily by the jet pumps and other obstructions not readily removable.

Because of these difficulties, all licensees of BWRs (except for some recent BWR 5 and 6 plants) have asked for relief from Section XI 1st interval requirements to examine these welds. Relief requests for the 2nd inspection interval have also been received. Some BWR plants have already been granted such relief. For these plants, the reactor vessels will not be examined for about 30 years as Section XI permits the examinations to be performed near the end of the 10-year inspection interval.

Equipment and techniques are presently being developed for examining essentially 100% of the length of all reactor vessel shell welds. The schedule of this development program is consistent with the implementation schedule for the proposed augmented examination program.

9. <u>Whether the proposed backfit is interim or final and, if interim, the justificaton for imposing the proposed backfit on an interim basis.</u>

The proposed augmented examination would become effective wher the proposed rule becomes effective, which would follow resolution of any comments received during the 60-day public comment period. The augmented examination would be implemented only once. Thereafter, when licensees update their plant's inservice inspection program and proceed to the next inservice inspection interval, the examination of essentially 100% of the length of all reactor vessel shell welds would become mandatory as part of the updated Section XI inservice inspection program.

## Appendix B

## Comparative Text
## for
## Proposed Amendment to § 50.55a Paragraphs (f) and (g)

Proposed paragraph (f) relative to existing paragraph (g)

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

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

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

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

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

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

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

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

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

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

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

(iii) [Reserved]

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

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

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

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

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

(6) (i) The Commission will evaluate determinations under paragraph ~~(g)~~ (f)(5) of this section that code requirements are impractical.

The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

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

Proposed Paragraph (g) relative to existing paragraph (g)

(g) Inservice inspection requirements. Requirements for inservice testing of Class 1, Class 2, and Class 3 pumps and valves are located in paragraph 50.55a(f). (1) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) ~~shall~~ must meet the requirements of paragraphs (g)(4) and (5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary and their supports ~~shall~~ must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves ~~shall~~ must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

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

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

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

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

(iii) ~~Pumps-and-valves-which-are-classified-as-ASME-Code-Class-1-shall-be-designed-and-be-provided-with-access-to-enable-the-performance-of-inservice-testing-of-the-pumps-and-valves-for-assessing-operational-readiness-set-forth-in-Section-XI-of-editions-of-the-ASME-Boiler-Vessel-Code-and-Addenda[6]-applied-to-the-construction-of-the-particular-pump-or-valve-or-the-Summer-1973-Addenda,-whichever-is-later.~~ [Reserved]

(iv) ~~Pumps-and-valves-which-are-classified-as-ASME-Code-Class-2-and-Class-3-shall-be-designed-and-be-provided-with-access-to-enable-the-performance-of-inservice-testing-of-the-pumps-and-valves-for-assessing-operational-readiness-set-forth-in-Section-XI-of-editions-of-the-Boiler-and-Pressure-Vessel-Code-and-Addenda[6]-applied-to-the-construction-of-the-particular-pump-or-valve-or-the-Summer-1973-Addenda,-whichever-is-later.~~ [Reserved]

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

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

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

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

(iii) [Reserved]

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

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

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

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

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

(6)

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

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

(A) Augmented examination of reactor vessel

(1) All licensees shall augment their reactor vessel examination by implementing once, as part of the inservice inspection interval in effect on ________ (effective date of rule will be inserted), the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB of the 1989 Edition of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, subject to the conditions specified in paragraphs 50.55a(g)(6)(A)(2) and (3). The augmented examination may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on ______________ (effective date of rule will be inserted).

(2) Licensees with fewer than 40 months remaining in the inservice inspection interval in effect on _________ (effective date of rule will be inserted) may defer the augmented reactor vessel examination specified in paragraph 50.55a(g)(6)(ii)(A)(1) to the first period of the next inspection interval. The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the inservice inspection interval in effect on _________ (effective date of rule will be inserted). The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed.

(3) The requirement for augmented examination of the reactor vessel may be satisfied by an examination of essentially 100% of the reactor vessel shell welds specified in paragraph 50.55a(g)(6)(A)(1) that has been completed, or is scheduled for implementation with a written commitment, or is required by paragraph 50.55a(g)(4)(i), during the inservice inspection interval in effect on ________ (effective date of rule will be inserted).

## Appendix C

## Supporting Statement for Information Collection Requirements in 10 CFR § 50.55

A. JUSTIFICATION

1. Need for the Collection of Information

NRC Regulations in 10 CFR § 50.55a incorporate by reference Division 1 rules of Section III, "Rules for Construction of Nuclear Power Plant Components," and Division 1 rules of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). These sections of the ASME Code set forth the requirements to which nuclear power plant components are designed, constructed, tested and inspected. Section III, Division 1, and Section XI, Division 1, each contain existing recordkeeping requirements. In general, Section III records are needed to provide documentation that construction procedures have been properly implemented, and Section XI records are needed to document the plans for and results of inservice inspection and inservice test programs. The scope of presently required Section III lifetime and nonpermanent records, and Section XI records is summarized in paragraph A.13.d, below. The scope of the existing recordkeeping requirements is not affected by the proposed amendment. The proposed amendment does, however, include revisions that require implementation of the existing recordkeeping requirements. The records developed are generally not collected, but are retained by the licensee to be made available to the NRC in the event of an NRC audit.

2. Agency Use of Information

The records are generally historical in nature and provide data on which future activities and actions can be based. The practical utility of the information collection for NRC is that appropriate records are available for auditing by NRC inspection and enforcement personnel to determine if ASME Code provisions for construction, inservice inspection, and inservice testing are being properly implemented in accordance with § 50.55a of the NRC regulations, or whether specific enforcement actions are necessary.

3. Reduction of Burden Through Information Technology

The information being collected represents the documentation for the various plant specific construction, inservice inspection, and inservice testing programs. The NRC has no objection to the use of new information technologies and generally encourages their use.

4. Effort to Identify Duplication

ASME Code requirements are incorporated by reference into the NRC regulations to avoid the need for writing equivalent NRC requirements. This amendment will not duplicate the information collection requirements contained in any other generic regulatory requirement.

5. Effort to Use Similar Information

The NRC is using the information reporting requirements specified in the ASME Code in lieu of developing its own equivalent requirements.

6. Effort to Reduce Small Business Burden

This amendment to § 50.55a affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act in the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, the proposed amendment does not fall in the province of this Act. The proposed rule would have no significant effect on a substantial number of small companies.

7. Consequences of Less Frequent Collection

The information is generally not collected, but is retained by the licensee to be made available to the NRC in the event of an NRC audit.

8. Circumstances Which Justify Variation from OMB Guidelines

The record retention periods for information requested is frequently for the service lifetime of the applicable component. Such lifetime retention of records is necessary to ensure adequate historical information on the design and examination of components and systems to provide a basis for evaluating degradation of these components and systems at any time during their service lifetime.

9. Consultations Outside the NRC

There are no consultations outside the NRC.

10. Confidentiality of Information

NRC provides no pledge of confidentiality for this collection of information.

11. Justification for Sensitive Questions

No sensitive questions are involved. Information collected is simply a documentation of construction procedures, inservice inspections, and inservice testing.

12. Estimated Annualized Cost to the Federal Government

NRC inspection personnel who audit plant quality assurance records would include in their audit verification that the above records are being properly prepared and maintained. The time associated with NRC inspectors verifying these records would be extremely small when the activity is performed as part of a normal quality assurance audit.

13. Estimate of Burden

a. Number and Type of Respondents

In general, the recordkeeping requirements incurred by § 50.55a through incorporation by reference of the ASME Code could apply to the owners of the 11 nuclear power plants with construction permits and to the owners of the 112 nuclear power plants with operating licenses. The actual number of plants that would implement the addenda incorporated by the proposed revision, and thereby be affected by the information collection requirements, is dependent on a variety of factors. These factors include whether the application is for Section III or Section XI, the class and type of components involved, the date of the construction permit application, the schedule of the inservice inspection program, and whether the plant voluntarily elects to implement updated editions and addenda of the ASME Code. However, conservatively, it is assumed that the recordkeeping requirements imposed by the proposed amendment apply to the 123 nuclear power plants presently under construction or in operation.

b. Estimated Hours

As noted in Item A.1, above, the scope of the existing recordkeeping requirements is not affected by the proposed amendment. However, incorporation by reference of the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, and Section XI, Division 1, of the ASME Code revises the following provisions which affect implementation of the existing recordkeeping requirements.

Section III

- 1986 Addenda

  NCA-4134.17: Retention time for rebar splice test reports and joint-welder identification records increased from 3 years to 10 years.

  NCA-4134.18(c): Specifies that audit frequencies for N-type Certificate Holders be commensurate with schedule of activities.

  NB/C/D-4322.1: Allows multiple fillet welds, socket welds, welds of specially designed seals, weld metal cladding, and hard surfacing to have group identification for the purpose of verifying that all welders and welding operators were properly qualified.

  NB/C/D-4424(d): Allows for the engineering evaluation of the concavity on the root side of a single welded circumferential butt weld.

  NC/D-2130/4122: Relaxes requirements for Certified Material Test Report for pressure retaining material defined as small products.

  Appendix V: Certain Data Report Forms have been editorially revised to clarify reporting requirements.

- 1987 Addenda

  NB/C/D-2121: Exempts brazing material, and some cladding, from the requirements of NB-2000, including quality assurance requirements.

  NC-7512: Requires that the Certificate Holder for the safety valve confirm, by test on each production main steam safety valve at the stamped set pressure, that the lift and blowdown requirements of NC-7500 are met.

- 1988 Addenda

  NC/D-6114.1(b): Requires that the Design Specification and N-5 Data Report Form identify the portions of piping that are exempted from pressure testing.

  Appendix V: Certain Data Report Forms have been editorially revised to clarify reporting requirements.

- 1989 Edition

  The 1989 Edition of Section III is identical to the 1986

Edition, as modified by the 1986 Addenda, 1987 Addenda, and 1988 Addenda. The 1986 Edition has been incorporated by reference into § 50.55a by a previous amendment. Information collection requirements for the 1986 Addenda, 1987 Addenda, and 1988 Addenda are discussed above.

Section XI

- 1986 Addenda

  IWB-3700, Appendix E: Requires an engineering evaluation to be performed to determine the effects of an out-of-limit condition on the structural integrity of the reactor vessel beltline region. Procedures and criteria are provided in new, nonmandatory appendix. Evaluation procedures subject to acceptance by the regulatory authorities having jurisdiction at the plant site.

  NIS-1 Form: Certification statement on the form (Owner's Report for Inservice Inspections) to be signed by the Inspector was clarified to specifically reference the Owner's Inspection Plan.

- 1987 Addenda

  IWA-2420, Appendix F: New, nonmandatory appendix provides guidance to the Owner on how to prepare the required Inspection Plan.

  IWA-6220(d): Duplication is eliminated by deleting from the Inservice Inspection Summary Reports those items that are addressed in the NIS-1 and NIS-2 Owner's Reports.

- 1988 Addenda

  IWA-2300: The qualification requirements for nondestructive examination personnel have been editorially revised and clarified.

  IWA-4000: Expanded to consolidate and clarify identical requirements that appear in the various subsections.

  Table IWB-2500-1: The reactor vessel examination requirements are revised to require the same extent of examination in the second and successive intervals as is required for the first interval.

  Subsections IWP and IWV: The rules for inservice testing of pumps and valves are deleted and reference is made to ASME/ANSI OMa-1988 Addenda to OM-1987 Part 5 and Part 10 for inservice testing of pumps and valves, respectively. Appendix IV: The revisions permit digital recording of steam generator examination results.

Appendix VII: This new mandatory appendix provides for the qualification of nondestructive examination personnel for ultrasonic examination.

- 1989 Edition

  The 1989 Edition of Section XI is identical to the 1986 Edition, as modified by the 1986 Addenda, 1987 Addenda, and 1988 Addenda. The 1986 Edition has been incorporated by reference into § 50.55a by a previous amendment. Information collection requirements for the 1986 Addenda, 1987 Addenda, and 1988 Addenda are discussed above.

Implementation of Section III

Section 50.55a specifies that the Code Edition, Addenda, and optional Code Cases to be applied to reactor coolant pressure boundary, and Quality Group B and Quality Group C components must be determined by the provisions of paragraph NCA-1140 of Subsection NCA of Section III of the ASME Code. NCA-1140 specifies that the owner (or his designee) shall establish the ASME Code edition and addenda to be included in the Design Specifications, but that in no case shall the Code edition and addenda dates established in the Design Specifications be earlier than 3 years prior to the date that the nuclear power plant construction permit is docketed. NCA-1140 further states that later ASME Code editions and addenda may be used by mutual consent of the Owner (or his designee) and Certificate Holder. The earliest Section III addenda being addressed in the proposed rule is the 1986 Addenda. Since the last plant docketed in October 1974 (Palo Verde Units 1, 2, 3), there is no plant under construction for which implementation of the Section III addenda specified in the proposed rule would be a requirement. It is permissible for individual plants to implement these improved rules on a voluntary basis, but unless they make that choice, there is no additional paperwork burden associated with incorporating the proposed Section III addenda.

Implementation of Section XI

Nuclear power plants are required to update their inservice inspection and inservice test programs by incorporating into successive 120-month inspection intervals requirements of the latest edition and addenda of Section XI that have been incorporated by reference as of 12 months prior to the start the next 120-month inspection interval. On this basis, many plants may at one time be required to implement the Section XI, Division 1, addenda specified in the proposed rule. The number of plants that could be implementing the specified addenda will grow gradually as each plant updates its inservice inspection program at the 10-year interval.

Therefore, conservatively, the total number of plants that may ultimately be required to implement the specified edition and addenda is 123 (i.e., the 112 plants operating licenses and the 11 plants with construction permits).

The Section XI revisions contained in the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition that affect reporting provisions are identified above. Of the 10 revisions identified, seven could serve to slightly reduce the overall reporting burden by clarifying the reporting requirements, eliminating duplication, or in the case of the OM Part 6 and Part 10 standards, reducing relief requests and associated records by providing requirements that are compatible with present day plant capabilities. The three remaining revisions (i.e., IWB-3700, Table IWB-2500-1, and Appendix VII) will result in some additional recordkeepings which are detailed below.

Section XI, Articles IWP and IWV

The revisions to Subsection IWP and Subsection IWV to reference the OM Part 6 and Part 10 standards are conservatively estimated to save 80 person-hours (p-hrs) per plant per 10-year inspection inverval through reduced time for preparation and review of relief requests. An average of 12 plants per year move from one interval into the next interval, and usually relief requests are updated at this time. The expected saving per year is therefore 960 p-hrs (i.e., 80 p-hrs/plant x 12 plants/year).

Section XI, Paragraph IWB-3700

IWB-3700, Analytical Evaluation of Plant Operating Events, requires an engineering evaluation when an operating event causes an excursion outside the normal operating pressure and temperature limits defined in the plant Technical Specifications. (The new nonmandatory Appendix E provides procedures and criteria that may be used to evaluate the integrity of the reactor vessel beltline for the out-of-limit conditions.) It is estimated that a plant required to implement the IWB-3700 evaluation procedures would expend approximately 200 p-hrs. to review the plant's operational data, establish acceptance criteria, collect data for a plant/event specific analysis, perform an engineering evaluation, and prepare a final report. Not all plants will implement IWB-3700, but it is conservatively estimated that 5 percent of the total number of operating plants would be required to prepare an engineering evaluation and report. The expected additional burden per year is approximately 1200 p-hrs (i.e., 200 p-hrs/plant x .05 x 123 plants/year).

Section XI, Paragraph IWB-2500

A revision to Table IWB-2500-1 increases the extent of reactor vessel shell weld examinations in the second and successive 10-year intervals. The data from these examinations is automatically recorded and processed, but it is estimated that about 200 p-hrs would be expended in reviewing, assembling, and summarizing the additional data that would be collected once during each 10-year inspection interval. An average of 12 plants per year have reactor vessel weld examinations performed during a refueling outage. Therefore, the estimated additional recordkeeping burden per year is 2400 p-hrs (i.e., 200 hrs/plant x 12 plants/year).

Section XI, Augmented Examination of Reactor Vessel

The proposed rule would impose a one-time augmented examination of reactor vessel shell welds that follows the Table IWB-2500-1 examinations addressed above. The augmented examination would result in all plants implementing the examinations within 80 months of the effective date of the proposed rule, which is, on average, earlier than required by the present regulations. However, because the above burden calculation for the Table IWB-2500-1 revision assumes immediate implementation of the reactor vessel examination, the calculated burden of 2400 p-hrs effectively includes the recordkeeping burden associated with the proposed augmented reactor vessel examination.

Section XI, Appendix VII

Appendix VII will add to the training records due to expanded training requirements for ultrasonic examiners. It is estimated that it would take 65 p-hrs per plant per year to maintain the additional training records. This appendix will eventually apply to all operating plants. Therefore, the estimated additional recordkeeping burden per year is 7995 p-hrs (i.e., 65 hrs/plant/year x 123 plants/year).

Total Recordkeeping Burden

As noted above, there is no requirement for existing applicants/licensees to implement the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, but there is an increase in the recordkeeping burden associated with implementing the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section XI for required inservice inspection and inservice testing activities, as summarized below.

Recordkeeping Burden

| ASME Code Reference | No. of Plants/ Year | Annual Rcrdkping Hrs/Plant | Total Annual Hours | Retention Period |
|---|---|---|---|---|
| Subsections IWP/IWV | 12 | ·80 | ·960 | Lifetime |
| Paragraph IWB-3700 | 6 | 200 | 1200 | Lifetime |
| Table IWB-2500 (EC B-A) | 12 | 200 | 2400 | Lifetime |
| Mandatory Appendix VII | 123 | 65 | 7995 | Lifetime |

Number of plants per year responsible for specific recordkeeping varies in accordance with above table.

Total annual recordkeeping burden for all plants is 10,635 hrs.

c. Estimated Cost Required to Respond to the Collection

Based upon the hours specified in Item A.13.b, above, and a rate of $95/hr., it is estimated that the total industry cost of responding to the information collection required by the proposed amendment to § 50.55a is a total of $1,010,325/year (10,635 hours x $95/hour).

The table below shows the elemental burdens associated with this computation.

| ASME Code Reference | P-HRS/YR | K$/YR |
|---|---|---|
| Subsections IWP and IWV | ·960 | ·91.2 |
| Paragraph IWB-3700 | 1200 | 114.0 |
| Table IWB-2500 (EC B-A) | 2400 | 228.0 |
| Mandatory Appendix VII | 7995 | 759.5 |
| Totals | 10,635 | 1010.3 |

d. Record Retention Period

Section III, Division 1, recordkeeping requirements are provided in NCA-4134.17, "Quality Assurance Records". Lifetime quality assurance records are identified in Table NCA-4134.17-1. The specified lifetime records are:

Section III Lifetime Records

- o Index to lifetime records
- o Code Data Reports
- o Design Specification
- o Design Documents
- o Overpressure Protection Report
- o Construction Specification
- o As-built Drawings
- o Certified Material Test Reports
- o Heat Treatment Records
- o Qualification Test Reports
- o Final Hydrostatic and Pneumatic Test Results
- o Final Nondestructive Examination Reports
- o Repair Records when required by Code
- o Weld Procedures
- o Construction Report

The Section III, Division 1, nonpermanent quality assurance records and required retention periods are provided in Table NCA-4134.17-2. The specified nonpermanent records are:

Section III Nonpermanent Records

| Record | Retention Period |
|---|---|
| o QA Program Manual | 3 years [1] |
| o Design Procurement and QA Procedures | 3 years [1] |
| o Installation and NDE Procedures | 10 years [1] |
| o Personnel Qualification Records | 3 years [1] |
| o Purchase Orders | 10 years [1] |
| o Audit and Survey Reports | 3 years [2] |
| o Final Radiographs | 10 years [2] |
| o Calibration Records | Until recalibration |
| o Process Sheets, Travellers, or Checklists | 10 years [2] |
| o Joint-welder identification records when such records are used in lieu of physical marking of welds | 10 years [2] |

[1] After superseded or invalidated
[2] After completion

Section XI, Division 1, requirements for inservice inspection records and reports are provided in IWA-6000, "Records and Reports". These records and reports must be maintained for the service lifetime of the component or system. As a minimum, the records and reports that must be prepared and maintained are:

Section XI Lifetime Records

- o Index to record file
- o Preservice and inservice inspection plans
- o Preservice and inservice inspection reports
- o Repair records and reports
- o Replacement records and reports
- o Nondestructive examination procedures
- o Nondestructive examination records
- o Pump records and reports
- o Valve records and reports
- o Pressure test procedures
- o Pressure test records

The retention period for information is in accordance with a schedule provided in Tables NCA-4134.17-1 and -2, and paragraph IWA-6300 of the ASME Code. The record retention periods for information specified in Item A.13.b, above, are generally for the service lifetime of the component. Lifetime retention of the above records is necessary to ensure adequate historical information on the design and examination of components and systems to provide a basis for evaluating degradation of these components and systems at any time during their service lifetime.

The record retention periods for the specific information specified in Item 13.b, above, are as follows:

| Record | Retention Period |
|---|---|
| o Rebar splice test reports | 10 years[2] |
| o Joint-welder identification | 10 years[2] |
| o Group indentification of welds | 10 years[2] |
| o Engineering evaluation of root-side weld concavity | Lifetime |
| o Certified Material Test Report | Lifetime |
| o Data report forms | Lifetime |
| o Confirmation of steam safety valve settings | Lifetime |
| o Contents of Design Specification and N-5 Data Report Form | Lifetime |
| o Engineering evaluation of an out-of-limit condition | Lifetime |
| o NIS-1 and NIS-2 forms | Lifetime |
| o Qualification records of NDE personnel | 3 years[1] |

[1] After surperseded or invalidated
[2] After completion

14. Reasons for Change in Burden

The change in burden results from a change in ASME Code recordkeeping requirements effected by the addenda and edition that are being incorporated by reference through this proposed amendment into the NRC regulation, and by a proposed augmented examination of reactor vessel shell welds.

15. Publication for Statistical Use

This information will not be published for statistical use.

B. COLLECTION OF INFORMATION EMPLOYING STATISTICAL METHODS

Statistical methods are not used in the collection of the required information.

Enclosure 3

**Resolution Of Division Comments For Proposed Amendment To Incorporate By Reference Into 10 CFR 50.55a The 1986 Addenda, 1987 Addenda, 1988 Adenda, and 1989 Edition Of Section III, Division 1, and Section XI, Division 1, Of The American Society of Mechanical Engineers Boiler And Pressure Vessel Code**

| Source | Comment | Resolution |
|---|---|---|
| NRR | 1. Revise proposed augmented examination to, among other things, include a provision that if augmented examination is deferred to first period of next inspection interval it may not be used as a substitute for previously scheduled examination for that interval. | Incorporated |
| | 2. Editorial changes to proposed new paragraph 50.55a(f). | Incorporated |
| | 3. Supplementary Information should be modified to add a statement describing expected licensee action regarding written commitments to 50.55a(g) when IST requirements are moved to new paragraph 50.55a(f). | Incorporated |
| | 4. Supplementary Information should note that while the NRC does endorse the Code exemptions from the Code requirements, items exempt from the Code must meet other applicable regulatory requirements. | Incorporated |
| | 5. "NRR considers the proposed amendment to be a "backfit" to enhance existing adequate safety, rather than a necessity to ensure adequate safety as | RES considers the proposed augmented reactor vessel examination to be a backfit and has |

| Source | Comment | Resolution |
| --- | --- | --- |
| | in the regulatory analysis." | treated it to be such in the proposed rule and supporting regulatory analysis. Consistent with an OGC position, RES does not believe the incorporation by reference of updated ASME Code editions and addenda is a backfit and has so documented in the regulatory analysis. |
| AEOD | No comments. | |
| OGC | 1. "Documented evaluation" required by para. 50.109(a)(4) should be a separate document from the Regulatory Analysis and should address all items listed in para. 50.109(c)(1) - (c)(9) except (c)(5) which deals with cost to the licensee. | Separate "documented evaluation" prepared ---- provided in Appendix A of Regulatory Analysis. |
| | 2. Certain discrepancies exist between what is "proposed" in the comparative text and what is contained in the text of the proposed rule. | Discrepancies eliminated. |
| | 3. Editorial comments. | Incorporated. |
| IRM | 1. Second paragraph of Paperwork Reduction Act Statement should be changed in accordance with specified sample. | Incorporated |
| | 2. Supporting Statement for information collection requirements should be modified to address the practical utility of the new information collections, the retention periods for records addressed under Section III, Division 1, | Incorporated |

| Source | Comment | Resolution |
| --- | --- | --- |
| | and the burden table should be expanded to reflect the number of response/burden for each collection. | |
| ARM | 1. Editorial comments. | Incorporated |
| | 2. Use of "shall" and "must" throughout § 50.55a should adhere to Federal Register convention. | Incorporated |

## Enclosure 4

## CRGR Information

The CRGR Charter requires that specific information be included in submittal packages. For this submittal package, some of the required information is included in the Proposed Rule (Enclosure 1), the Regulatory Analysis (Enclosure 2), and the the Documented Evaluation (Enclosure 2, Appendix A) required by § 50.109. Required information that is not included in these documents is included in this enclosure. For convenience, information that is contained in the aforementioned documents is cross-referenced herein.

The following information requests have been extracted from the CRGR Charter.

1. "The proposed generic requirement or staff position as it is proposed to be sent out to licensees."

   The proposed rule is provided in its entirety in Enclosure 1. The proposed augmented reactor vessel examination may be found on P. 32 of that enclosure.

2. "Draft staff papers or other underlying staff documents supporting the requirements or staff positions. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request CRGR staff to obtain a copy of any reference material for his or her use.)"

   a. Proposed Generic Letter titled, "Inservice Inspection of the Pressure-Retaining Welds in the Reactor Vessel."

   b. Letter from W. Kerr to Honorable L. W. Zech, regarding expanded examination requirements for BWR reactor vessels, dated June 7, 1988.

   c. Letter from T. E. Murley to S. H. Bush, regarding expanded examination requirements for the reactor vessel, dated March 14, 1988.

   d. Memorandum from C. Y. Cheng to J. E. Richardson, "Action Plan for Reactor Vessel Inspection," dated September 23, 1987.

   e. Memorandum from L. C. Shao to R. W. Starostecki, "Examination of BWR Reactor Vessel Shell Welds," dated September 1, 1987.

3. "Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase requirements, or staff positions, implement existing requirements or staff positions, or would relax or reduce existing requirements or staff positions."

   The proposed rule is comprised of two technical actions. The first, updates the reference to the editions and addenda that are incorporated by reference into § 50.55a, "Codes and standards." The second, imposes an augmented examination on reactor vessel shell welds.

With regard to updating ASME Code references in the regulations, it must be recognized that the affected edition and addenda consist of numerous revisions. These revisions are generally made to improve the ASME Code by providing more detail where experience indicates greater guidance is necessary for existing rules (e.g., often as the result of Inquiries submitted to the ASME), to relax the rules where experience shows equivalent operational safety can be maintained with a reduced burden on licensees and contractors, and to add new rules in specific areas based upon demonstrated need.

In general, the most significant actions affected by the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III and Section XI, are:

- Revision to Section XI Subsection IWB Table IWB-2500-1, Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," Item B1.10, "Shell Welds" which requires examinations in the 2nd and successive intervals to be the same as in the 1st interval (i.e., examination requirement went from one longitudinal and one beltline welds to essentially 100% of the length of all reactor vessel shell welds). This represents an increase in Section XI requirements. (1988 Addenda and 1989 Edition)

- Replacement (by referencing) of pump and valve testing rules in Section XI Subsection IWP and Subsection IWV with the rules in ASME Operation & Maintenance (O&M) Committee standards Part 6 (pumps) and Part 10 (valves), respectively. This represents an increase in Section XI requirements, but will also negate the need for some relief requests that have been generically granted by the staff. (1988 Addenda and 1989 Edition)

- Addition of Section XI mandatory Appendix VII, which provides rules for qualifying nondestructive examination personnel for ultrasonic examination. This represents an increase in Section XI requirements. (1988 Addenda and 1989 Edition)

It must be remembered that the ASME Code is developed (and revised) through the American National Standards Institute consesus process. The consensus process ensures that the various technical interests (e.g., utility, manufacturing, insurance, regulatory) are represented on the standards development committees and their viewpoints are addressed fairly in the standards writing process. The consensus process ensures that the cost and benefit of revisions to the ASME Code, such as those noted above, are properly considered from all sides and are reasonably balanced.

The second technical action associated with the proposed rule is the proposed augmented examination of reactor vessel shell welds. Although the proposed augmented examination does represent an increase in requirements, it is, perhaps, more accurately characterized as a on-time effort to expedite implementation of the reactor vessel shell weld examination requirements incorporated into the 1989 Edition of Section XI in a time-frame consistent with the importance of the reactor vessel.

4. "The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed."

Licensees would implement the provisions of the later edition and addenda of Section XI that would be incorporated by reference, as part of their initial 120-month inspection interval (i.e., in accordance with existing paragraph 50.55a(g)(4)(i) and proposed paragraph 50.55a (f)(4)(iii)), or during successive 120-month intervals (i.e., in accordance with updating requirements specified in existing paragraph 50.55a(g)(4)(ii) or in proposed paragraph 50.55a(f)(4)(iv)). Applicant applying for new construction permits would be required to implement the provisions of the later edition and addenda of Section III. However, licensees and existing applicants could use the provisions of the later edition and addenda of Section III on a voluntary basis.

Licensees would be required to implement the one-time augmented examination of reactor vessel shell welds in accordance with the provisions of the proposed paragraph 50.55a(g)(6)(ii)(A) (see Enclosure 1, P. 32).

The OGC has no legal objection to the method of implementation specified in the proposed rule.

5. "Regulatory analyses as specified in NUREG/BR-0058, Revision 1, May 1984, Regulatory Analysis Guidelines of the U.S. NRC."

The draft regulatory analysis is provided in Enclosure 2.

6. "Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.)."

The proposed rule would apply to all operating PWRs and BWRs.

7. "For each category of reactor plants, the evaluation should also demonstrate how the action should be prioritized and scheduled in light of other ongoing regulatory activities."

The proposed rule has two components that require implementation. In both cases the provisions of the proposed rule would be integrated into the timing and requirements of the ongoing inservice inspection and inservice test programs. The first component relates to the selection of the Section XI edition/addenda for use during the initial 120-month interval and during successive 120-month interval. This part of the implementation process is routine. The § 50.55a paragraphs specified in Item 4, above, require that the edition/addenda of Section XI to be used in the inservice inspection and inservice test programs for the intial 120-month interval, and for successive 120-month intervals comply, respectively, with the latest edition/addenda incorporated by reference 12 months prior to the date of issuance of the operating license, and 12 months prior to the start of the next 120-month interval. This procedure is well established within the industry, and no implementation problems are anticipated.

The second component of the proposed rule that would require implementation is the augmented examination of reactor vessel shell welds. The proposed augmented examination requires licensees to implement the expanded examination during the inspection interval in effect when the rule becomes effective. Recognizing that plants close to the end of the inspection interval might have difficulty implementing the examination in the time remaining, the proposed rule permits, for those plants with fewer than 40 months remaining in the effective interval, the ability to defer the augmented examination until the first period of the next interval.

"The evaluation is to consider information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

a. Statement of the specific objectives that the proposed action is designed to achieve;

b. General decscription of the activity that would be required by the licensee or applicant in order to complete the action;

c. Potential change in the risk to the public from the accidental offsite release of radioactive material;

d. Potential impact on radiological exposure of facility employees;

e. Installation and continuing costs associated with the action, including the cost of facility downtime or the cost of construction delay;

f. The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements and staff positions;

g. The estimated resource burden on the NRC associated with the proposed action and the availability of such resources;

h. The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed action;

i. Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis."

Items a - i, above, are all addressed in the Documented Evaluation (Enclosure 2, Appendix A), which was prepared to comply with the backfitting requirements specified in § 50.109.

8. "For each evaluation conducted pursuant to 10 CFR 50.109, the proposing office director shall determine based on the considerations of paragraphs 1 through 7 above, whether:

a. the proposal would result in a substantial increase in the overall protection of public health and saftey or the common defense and security; and

b. the direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection."

A determination has been made, based upon the documented evaluation required by § 50.109, that the backfit requirements imposed by the augmented examination of reactor vessel shell welds contained in this proposed amendment are necessary to ensure that the facility provides adequate protection to the public helath and safety, and, therefore, that a backfit analysis is not required and the cost-benefit standards of 10 CFR 50.109(a)(3) do not apply.

9. "For each evaluation conducted for proposed relaxation or decreases in current staff positions, the proposing office director shall determine, based on the considerations of paragraphs 1 -7 above, whether:

a. the proposal would result in any decrease in plant safety; and

b. the proposal would result in substantial cost savings for the industry."

The proposed rule is not expected to result in any decrease in plant safety, nor result in an overall substantial cost savings for the industry.

Enclosure 5

1. O&M Part 6, "Inservice Testing of Pumps in Light-Water Reactor Power Plants"

2. O&M Part 10, "Inservice Testing of Valves in Light-Water Reactor Power Plants"

PART 6 (a)



Tables

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# PART 6

## Inservice Testing of Pumps in Light-Water Reactor Power Plants

### 1 INTRODUCTION

#### 1.1 Scope

This Part establishes the requirements for preservice and inservice testing to assess the operational readiness of certain centrifugal and positive displacement pumps used in nuclear power plants.

The pumps covered are those, provided with an emergency power source, which are required in shutting down a reactor to the cold shutdown condition, maintaining the cold shutdown condition, or mitigating the consequences of an accident.

This Part establishes test intervals, parameters to be measured and evaluated, acceptance criteria, corrective actions, and records requirements.

#### 1.2 Exclusions

The following are excluded from this Part:

(a) drivers, except where the pump and driver form an integral unit and the pump bearings are in the driver;

(b) pumps that are supplied with emergency power solely for operating convenience.

#### 1.3 Terminology

The following are provided to ensure a uniform understanding of select terms used in this Part.

*inservice test* — a test to determine the operational readiness of a pump

*instrument accuracy* — the allowable inaccuracy of an instrument loop based on the square root of the sum of the squares of the inaccuracies of each instrument or component in the loop

*instrument loop* — two or more instruments or components working together to provide a single output (e.g., a vibration probe and its associated signal conditioning and readout devices)

*operational readiness* — the ability of a pump to perform its intended function

*preservice test period* — the period of time following completion of construction activities related to the pump, and prior to first electrical generation by nuclear heat, in which component and system testing takes place

*pump* — a mechanical device used to move liquid

*reference values* — one or more values of test parameters measured or determined when the equipment is known to be operating acceptably

*routine servicing* — the performance of planned, preventive maintenance (e.g., replacing or adjusting valves in reciprocating pumps, changing oil, flushing the cooling system, adjusting packing, adding packing rings or mechanical seal maintenance or replacement)

*system resistance* — the hydraulic resistance to flow in a system

### 2 REFERENCE INFORMATION

#### 2.1 Detection of Change

The hydraulic and mechanical condition of a pump relative to a previous condition can be determined by attempting to duplicate by test a set of reference values. Deviations detected are symptoms of changes and, depending upon the degree of deviation, indicate need for further tests or corrective action.

### 3 DESIGN REQUIREMENTS

#### 3.1 Owner's Responsibility

(a) It is the Owner's responsibility to include in both the pump and plant design all necessary valving, instrumentation, test loops, required fluid inventory, or other provisions which are required to fully comply with the rules of this Part.

(b) Each pump to be tested in accordance with the rules of this Part shall be identified by the Owner and listed in the plant records (see Section 7).

### 3.2 Bypass Loops

A bypass test loop may be used, provided the bypass is designed to recognize the pump manufacturer's operating conditions for minimum flow operation.

## 4 TESTING REQUIREMENTS

### 4.1 Preservice Testing

Each pump shall be tested during the preservice test period as required by this Part. This testing shall be conducted under conditions as near as practicable to those expected during subsequent inservice testing. Only one preservice test of each pump is required, except that the requirements of para. 4.4 shall be met.

### 4.2 Inservice Testing

Inservice testing in accordance with this Part shall commence when the pump(s) is required to be operable (see para. 1.1).

### 4.3 Reference Values

Reference values shall be determined from the results of preservice testing or from the results of the first inservice test. Reference values shall be at points of operation readily duplicated during subsequent tests. All subsequent test results shall be compared to these initial reference values or to new reference values established in accordance with paras. 4.4 and 4.5. Reference values shall only be established when the pump is known to be operating acceptably. If the particular parameter being measured or determined can be significantly influenced by other related conditions, then these conditions shall be analyzed.[1]

[1] Vibration measurements of pumps may be foundation, driver, and piping dependent. Therefore, if initial vibration readings are high and have no obvious relationship to the pump, then vibration measurements should be taken at the driver, at the foundation, and on the piping and analyzed to ensure that the reference vibration measurements are representative of the pump and that the measured vibration levels will not prevent the pump from fulfilling its function.

### 4.4 Effect of Pump Replacement, Repair, and Maintenance on Reference Values

When a reference value or set of values may have been affected by repair, replacement, or routine servicing of a pump, a new reference value or set of values shall be determined or the previous value reconfirmed by an inservice test run prior to declaring the pump operable. Deviations between the previous and new set of reference values shall be identified, and verification that the new values represent acceptable pump operation shall be placed in the record of tests (see Section 7).

### 4.5 To Establish an Additional Set of Reference Values

If it is necessary or desirable, for some reason other than stated in para. 4.4, to establish an additional set of reference values, an inservice test shall first be run at the conditions of an existing set of reference values and the results analyzed. If operation is acceptable per para. 6.1, a second test run at the new reference conditions shall follow as soon as practicable. The results of this test shall establish the additional set of reference values. Whenever an additional set of reference values is established, the reasons for so doing shall be justified and documented in the record of tests (see Section 7). The requirements of para. 4.3 apply.

### 4.6 Instrumentation

#### 4.6.1 General

**4.6.1.1 Quality.** Instrument accuracy shall be within the limits of Table 1. Station instruments meeting these requirements are acceptable.

**4.6.1.2 Range**

(a) The full-scale range of each analog instrument shall be not greater than three times the reference value.

(b) Digital instruments shall be selected such that the reference value shall not exceed 70% of the calibrated range of the instrument.

(c) Vibration instruments are excluded from the range requirements of (a) and (b) above.

**4.6.1.3 Instrument Location.** The sensor location shall be established by the Owner, documented in the plant records (see Section 7), and shall be appropriate for the parameter being measured. The same location shall be used for subsequent tests. Instruments that are position

TABLE 1
ACCEPTABLE INSTRUMENT ACCURACY

| Quantity | Percent [Note (1)] |
|---|---|
| Pressure | ±2 |
| Flow Rate | ±2 |
| Speed | ±2 |
| Vibration | ±5 |
| Differential Pressure | ±2 |

NOTE:
(1) Percent of full scale for individual analog instruments, percent of total loop accuracy for a combination of instruments, or over the calibrated range for digital instruments.

sensitive shall be either permanently mounted or provision shall be made to duplicate their position during each test.

**4.6.1.4 Calibration.** Instruments and instrument loops shall be calibrated in accordance with the Owner's quality assurance program. New or repaired instruments shall be calibrated prior to test use.

**4.6.1.5 Fluctuations.** Symmetrical damping devices or averaging techniques may be used to reduce instrument fluctuations. Hydraulic instruments may be damped by using gauge snubbers or by throttling small valves in instrument lines.

**4.6.1.6 Frequency Response Range.** The frequency response range of the vibration measuring transducers and their readout system shall be from one-third minimum pump shaft rotational speed to at least 1000 Hz.

**4.6.2 Pressure Measurement**

**4.6.2.1 Gage Lines.** If the presence or absence of liquid in a gage line could produce a difference of more than 0.25% in the indicated value of the measured pressure, means shall be provided to assure or determine the presence or absence of liquid as required for the static correction used.

**4.6.2.2 Differential Pressure.** When determining differential pressure across a pump, a differential pressure gauge, a differential pressure transmitter that provides direct measurement of pressure difference or the difference between the pressure at a point in the inlet pipe and the pressure at a point in the discharge pipe, may be used.

**4.6.3 Rotational Speed Measurements.** Rotational speed measurements of variable speed pumps shall be taken by a method which meets the requirements of para. 4.6.1.

**4.6.4 Vibration Measurements**

(*a*) On centrifugal pumps, measurements shall be taken in a plane approximately perpendicular to the rotating shaft in two orthogonal directions on each accessible pump bearing housing. Measurement also shall be taken in the axial direction on each accessible pump thrust bearing housing.

(*b*) On vertical line shaft pumps, measurements shall be taken on the upper motor bearing housing in three orthogonal directions, one of which is the axial direction.

(*c*) On reciprocating pumps, the location shall be on the bearing housing of the crankshaft, approximately perpendicular to both the crankshaft and the line of plunger travel.

(*d*) If a portable vibration indicator is used, the reference points must be clearly identified on the pump to permit subsequent duplication in both location and plane.

**4.6.5 Flow Rate Measurement.** When measuring flow rate, use a rate or quantity meter installed in the pump test circuit. If a meter does not indicate the flow rate directly, the record shall include the method used to reduce the data.

## 5 TESTING METHODS

### 5.1 Frequency of Inservice Tests

An inservice test shall be run on each pump, nominally every 3 months, except as provided in paras. 5.3, 5.4, and 5.5.

### 5.2 Test Procedure

An inservice test shall be conducted with the pump operating at specified test reference conditions. The test parameters shown in Table 2 shall be determined and recorded as directed in this paragraph. The test shall be conducted as follows.

(*a*) The pump shall be operated at nominal motor speed for constant speed drives and at a speed adjusted to the reference speed for variable speed drives.

(*b*) The resistance of the system shall be varied until the flow rate equals the reference value. The pressure shall then be determined and compared

TABLE 2
INSERVICE TEST PARAMETERS

| Quantity | Remarks |
|---|---|
| Speed $N$ | If variable speed |
| Differential Pressure $\Delta P$ | Centrifugal Pumps, including vertical line shaft pumps |
| Discharge Pressure $P$ | Positive Displacement Pumps |
| Flow Rate $Q$ | |
| Vibration: | |
| Displacement, $V_d$ | Peak-to-peak |
| Velocity, $V_v$ | Peak |

to its reference value. Alternatively, the flow rate can be varied until the pressure equals the reference value and the flow rate shall be determined and compared to the reference flow rate value.

(c) Where system resistance cannot be varied, flow rate and pressure shall be determined and compared to their respective reference values.

(d) Pressure, flow rate, and vibration (displacement or velocity) shall be determined and compared with corresponding reference values. All deviations from the reference values shall be compared with the limits given in Table 3 and corrective action taken as specified in para. 6.1.

Vibration measurements are to be broad band (unfiltered). If velocity measurements are used, they shall be peak. If displacement amplitudes are used, they shall be peak-to-peak.

### 5.3 Pumps in Regular Use

Pumps that are operated more frequently than every 3 months need not be run or stopped for a special test provided the plant records show each such pump was operated at least once every 3 months at the reference conditions, and the quantities specified were determined, recorded, and analyzed per Section 6.

### 5.4 Pumps in Systems Out of Service

For a pump in a system declared inoperable or not required to be operable, the test schedule need not be followed. Within 3 months prior to placing the system in an operable status, the pump shall be tested and the test schedule followed in accordance with the requirements of this Part. Pumps which can only be tested during plant operation shall be tested within 1 week following plant startup.

### 5.5 Pumps Lacking Required Fluid Inventory

Pumps lacking required fluid inventory, (e.g., pumps in dry sumps) need not be tested in accordance with this Part every 3 months. These pumps shall be tested at least once every 2 years except as provided in para. 5.4. The required fluid inventory shall be provided during this test.

### 5.6 Duration of Tests

After pump conditions are as stable as the system permits, each pump shall be run at least 2 min. At the end of this time at least one measurement or observation of each of the quantities required shall be made and recorded.

## 6 ANALYSES AND EVALUATION

### 6.1 Acceptance Criteria

If deviations fall within the alert range of Table 3, the frequency of testing specified in para. 5.1 shall be doubled until the cause of the deviation is determined and the condition corrected. If deviations fall within the required action range of Table 3, the pump shall be declared inoperable until the cause of the deviation has been determined and the condition corrected.

When a test shows deviations outside of the acceptable range of Table 3, the instruments involved may be recalibrated and the test rerun.

## TABLE 3
## RANGES FOR TEST PARAMETERS

TABLE 3a[1]

| Pump Type | Pump Speed | Test Parameter | Acceptable Range | Alert Range | Required Action Range |
|---|---|---|---|---|---|
| Centrifugal and vertical line shaft [Note (2)] | $< 600$ rpm | $V_v$ or $V_r$ | $\leq 2.5\ V_r$ | $> 2.5\ V_r$ to $6\ V_r$ or $> 10.5$ mils | $> 6\ V_r$ or $> 22$ mils |
| Centrifugal and vertical line shaft [Note (2)] | $\geq 600$ rpm | $V_r$ or $V_v$ | $\leq 2.5\ V_r$ | $> 2.5\ V_r$ to $6\ V_r$ or $> 0.325$ in./sec | $> 6\ V_r$ or $> 0.70$ in./sec |
| Reciprocating | | $V_v$ or $V_r$ | $\leq 2.5\ V_r$ | $> 2.5\ V_r$ to $6\ V_r$ | $> 6\ V_r$ |

NOTES:
(1) Vibration parameter per Table 2. $V_r$ is vibration reference value in the selected units.
(2) Refer to Fig. 1 to establish displacement limits for pumps with speeds $\geq 600$ rpm or velocity limits for pumps with speeds $< 600$ rpm.

TABLE 3b

| Test Parameter | Acceptable Range | Alert Range | | Required Action Range | |
|---|---|---|---|---|---|
| | | Low | High | Low | High |
| $P$ (Positive displacement pumps) | 0.93 to $1.10P_r$ | 0.90 to $< .93P_r$ | . . . | $< 0.90P_r$ | $> 1.10P_r$ |
| $\Delta P$ (Vertical line shaft pumps) | 0.95 to $1.10\Delta P_r$ | 0.93 to $< .95\Delta P_r$ | . . . | $< 0.93\Delta P_r$ | $> 1.10\Delta P_r$ |
| $Q$ (Positive displacement vertical line shaft pumps) | 0.95 to $1.10Q_r$ | 0.93 to $< .95Q_r$ | . . . | $< 0.93Q_r$ | $> 1.10Q_r$ |
| $\Delta P$ (Centrifugal pumps) | 0.90 to $1.10\Delta P_r$ | . . . | . . . | $< 0.90\Delta P_r$ | $> 1.10\Delta P_r$ |
| $Q$ (Centrifugal pumps) | 0.90 to $1.10Q_r$ | . . . | . . . | $< 0.90Q_r$ | $> 1.10Q_r$ |

GENERAL NOTE: The subscript $r$ denotes reference value.

### 6.2 Time Allowed for Analysis of Tests

All test data shall be analyzed within 96 hr after completion of a test.

## 7 RECORDS AND REPORTS

### 7.1 Pump Records

The Owner shall maintain a record which shall include the following for each pump covered by this Part:

(*a*) the manufacturer and the manufacturer's model and serial or other identification number;

(*b*) a copy or summary of the manufacturer's acceptance test report if available;

(*c*) a copy of the pump manufacturer's operating limits.

### 7.2 Inservice Test Plans

The Owner shall maintain a record of test plans and procedures which shall include the following:

(*a*) the hydraulic circuit to be used;

(*b*) the location and type of measurement for the required test parameters;

(*c*) the reference values;

(*d*) the method of determining reference values which are not directly measured by instrumentation.

### 7.3 Record of Tests

The Owner shall maintain a record of each test which shall include the following:

(*a*) pump identification;

(*b*) date of test;

(*c*) reason for test (e.g., post-maintenance, routine inservice test, establishing reference values);

(*d*) values of measured parameters;

(*e*) identification of instruments used;

(*f*) comparisons with allowable ranges of test values and analysis of deviations;

(*g*) requirement for corrective action;

(*h*) evaluation and justification for changes to reference values;

(*i*) signature of the person or persons responsible for conducting and analyzing the test.

### 7.4 Record of Corrective Action

The Owner shall maintain records of corrective action which shall include a summary of the corrections made, the subsequent inservice tests and confirmation of operational adequacy (see para. 4.4), and the signature of the individual responsible for corrective action and verification of results.

## PART 10

(a)



Table

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# PART 10

## Inservice Testing of Valves in Light-Water Reactor Power Plants

### 1 INTRODUCTION

#### 1.1 Scope

This Part establishes the requirements for preservice and inservice testing to assess the operational readiness of certain valves and pressure relief devices (and their actuating and position indicating systems) used in nuclear power plants.

The active or passive valves covered are those which are required to perform a specific function in shutting down a reactor to the cold shutdown condition, in maintaining the cold shutdown condition, or in mitigating the consequences of an accident. The pressure-relief devices covered are those for protecting systems or portions of systems which perform a required function in shutting down a reactor to the cold shutdown condition, in maintaining the cold shutdown condition, or in mitigating the consequences of an accident.

This Part establishes test intervals, parameters to be measured and evaluated, acceptance criteria, corrective action, and records requirements.

#### 1.2 Exclusions

(a) The following are excluded from this Part provided that the valves are not required to perform a specific function as specified in para. 1.1:

(1) valves used only for operating convenience such as vent, drain, instrument, and test valves;

(2) valves used only for system control, such as pressure regulating valves;

(3) valves used only for system or component maintenance.

(b) External control and protection systems responsible for sensing plant conditions and providing signals for valve operation are excluded from the requirements of this Part.

#### 1.3 Terminology

The following are provided to ensure a uniform understanding of select terms used in this Part.

*active valves* — valves which are required to change obturator position to accomplish the required function(s) as specified in para. 1.1

*exercising* — the demonstration based on direct visual or indirect positive indications that the moving parts of a valve function

*full-stroke time* — the time interval from initiation of the actuating signal to the indication of the end of the operating stroke

*plant operation* — the conditions of startup, operation at power, hot standby, and reactor cooldown, as defined by the plant technical specifications

*obturator* — valve closure member (disk, gate, plug, ball, etc.)

*passive valves* — valves which maintain obturator position and are not required to change obturator position to accomplish the required function(s), as specified in para. 1.1

*preservice test period* — the period of time following completion of construction activities related to the valve and prior to first electrical generation by nuclear heat in which component and system testing takes place

*reactor coolant system pressure isolation* — that function which prevents intersystem overpressurization between the reactor coolant system and connected low-pressure systems

*reference values* — one or more values of test parameters measured or determined when the equipment is known to be operating acceptably

*operational readiness* — the ability of a valve to perform its intended function

### 1.4 Categories of Valves

Valves within the scope of this Part shall be placed in one or more of the following categories. When more than one distinguishing category characteristic is applicable, all requirements of each of the individual categories are applicable, although duplication or repetition of common testing requirements is not necessary.

(*a*) *Category A* — valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s), as specified in para. 1.1

(*b*) *Category B* — valves for which seat leakage in the closed position is inconsequential for fulfillment of the required function(s), as specified in para. 1.1

(*c*) *Category C* — valves which are self-actuating in response to some system characteristic, such as pressure (relief valves) or flow direction (check valves) for fulfillment of the required function(s), as specified in para. 1.1

(*d*) *Category D* — valves which are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves

## 2 OWNER'S RESPONSIBILITY

The Owner shall identify, categorize (see para. 1.4), and list in the plant records (see Section 6) each valve to be tested in accordance with the rules of this Part, including Owner-specified acceptance criteria. The Owner shall specify test conditions.

## 3 TESTING REQUIREMENTS

### 3.1 Preservice Testing

Each valve shall be tested during the preservice test period as required by this Part. These tests shall be conducted under conditions as near as practicable to those expected during subsequent inservice testing. Only one preservice test of each valve is required except that:

(*a*) any valve which has undergone maintenance that could affect its performance after the preservice test shall be tested in accordance with para. 3.4;

(*b*) safety and relief valves and nonreclosing pressure relief devices shall meet the preservice test requirements of Part 1 of this Standard.

### 3.2 Inservice Testing

Inservice testing in accordance with this Part shall commence when the valves are required to be operable to fulfill their required function(s) (see para. 1.1).

### 3.3 Reference Values

Reference values shall be determined from the results of preservice testing or from the results of inservice testing. These tests shall be performed under conditions as near as practicable to those expected during subsequent inservice testing.

Reference values shall only be established when the valve is known to be operating acceptably. If the particular parameter being measured can be significantly influenced by other related conditions, then these conditions shall be analyzed.

### 3.4 Effect of Valve or Actuator Replacement, Repair, and Maintenance on Reference Values

When a valve or its control system has been replaced, repaired, or has undergone maintenance[1] that could affect the valve's performance, a new reference value shall be determined or the previous value reconfirmed by an inservice test run prior to the time it is returned to service or immediately if not removed from service, to demonstrate that performance parameters which could be affected by the replacement, repair, or maintenance are within acceptable limits. Deviations between the previous and new reference values shall be identified and analyzed. Verification that the new values represent acceptable operation shall be documented in the record of tests (see para. 6.3). Safety and Relief valves and nonreclosing pressure relief devices shall be tested as required by the replacement, repair, and maintenance requirements of Part 1.

### 3.5 To Establish an Additional Set of Reference Values

If it is necessary or desirable for some reason, other than stated in para. 3.4, to establish additional reference values, an inservice test shall first

[1] Adjustment of stem packing, limit switches, or control system valves, and removal of the bonnet, stem assembly, actuator, obturator, or control system components are examples of maintenance that could affect valve performance parameters.

TABLE 1
INSERVICE TEST REQUIREMENTS

| Category (See Para. 1.4) | Valve Function | Leakage Test Procedure | Exercise Test Procedure | Special Test Procedure [Note (1)] | Position Indication Verification |
|---|---|---|---|---|---|
| A | Active | See para. 4.2.2 | See para. 4.2.1 | None | See para. 4.1 |
| A | Passive | See para. 4.2.2 | None | None | See para. 4.1 |
| B | Active | None | See para. 4.2.1 | None | See para. 4.1 |
| B | Passive | None | None | None | See para. 4.1 |
| C (Safety and relief) | Active | None [Note (2)] | See para. 4.3.1 | None | See para. 4.1 |
| C (check) | Active | None [Note (2)] | See para. 4.3.2 | None | See para. 4.1 |
| D | Active | None | None | See para. 4.4 | None |

NOTES:
(1) Note additional requirement for fail-safe valves, para. 4.2.1.6.
(2) When more than one distinguishing category characteristic is applicable, all requirements of each of the individual categories are applicable, although duplication or repetition of common testing requirements is not necessary.

be run at the conditions of an existing set of reference values, or, if impractical, at the conditions for which the new reference values are required, and the results analyzed. If operation is acceptable in accordance with paras. 4.2.1.4(a) and 4.2.1.8, a second test shall be performed under the new conditions as soon as practical. The results of the second test shall establish the additional reference values. Whenever additional reference values are established, the reasons for doing so shall be justified and documented in the record of tests (see para. 6.3).

### 3.6 Inservice Test Requirements

Active and passive valves in the categories defined in para. 1.4 shall be tested in accordance with the paragraphs specified in Table 1.

## 4 TESTING METHODS

### 4.1 Valve Position Verification

Valves with remote position indicators shall be observed locally at least once every 2 years to verify that valve operation is accurately indicated. Where practicable, this local observation should be supplemented by other indications such as use of flow meters or other suitable instrumentation to verify obturator position. These observations need not be concurrent. Where local observation is not possible, other indications shall be used for verification of valve operation.

### 4.2 Inservice Tests for Category A and B Valves

#### 4.2.1 Valve Exercising Test

**4.2.1.1 Exercising Test Frequency.** Active Category A and B valves shall be tested nominally every 3 months, except as provided by paras. 4.2.1.2, 4.2.1.5, and 4.2.1.7.

**4.2.1.2 Exercising Requirements.** Valves shall be tested as follows:

(*a*) full-stroke during plant operation to the position(s) required to fulfill its function(s);

(*b*) if full-stroke exercising during plant operation is not practicable, it may be limited to part-stroke during plant operation and full-stroke during cold shutdowns;

(*c*) if exercising is not practicable during plant operation, it may be limited to full stroke exercising during cold shutdowns;

(*d*) if exercising is not practicable during plant operation and full-stroke during cold shutdowns is also not practicable, it may be limited to part-stroke during cold shutdowns, and full-stroke during refueling outages;

(*e*) if exercising is not practicable during plant operation or cold shutdowns, it may be limited to full-stroke during refueling outages;

(*f*) valves full-stroke exercised at cold shutdowns shall be exercised during each cold shutdown, except as specified in (g) below. Such exercise is not required if the time period since the previous full-stroke exercise is less than 3 months.

(g) valve exercising during cold shutdown shall commence within 48 hr of achieving cold shutdown, and continue until all testing is complete or the plant is ready to return to power. For extended outages, testing need not be commenced in 48 hr provided all valves required to be tested during cold shutdown will be tested prior to plant startup. However, it is not the intent of this Part to keep the plant in cold shutdown in order to complete cold shutdown testing.

(h) all valve testing required to be performed during a refueling outage shall be completed prior to returning the plant to operation.

**4.2.1.3 Valve Obturator Movement.** The necessary valve obturator movement shall be determined by exercising the valve while observing an appropriate indicator, such as indicating lights which signal the required change of obturator position, or by observing other evidence, such as changes in system pressure, flow rate, level, or temperature, which reflect change of obturator position.

**4.2.1.4 Power-Operated Valve Stroke Testing**

(a) The limiting value(s) of full-stroke time of each power-operated valve shall be specified by the Owner.

(b) The stroke time of all power-operated valves shall be measured to at least the nearest second.

(c) Any abnormality or erratic action shall be recorded (see para. 6.3), and an evaluation shall be made regarding need for corrective action.

**4.2.1.5 Valves in Regular Use.** Valves which operate in the course of plant operation at a frequency which would satisfy the exercising requirements of this Part need not be additionally exercised, provided that the observations otherwise required for testing are made and analyzed during such operation and are recorded in the plant record at intervals no greater than specified in para. 4.2.1.1.

**4.2.1.6 Fail-Safe Valves.** Valves with fail-safe actuators shall be tested by observing the operation of the actuator upon loss of valve actuating power in accordance with the exercising frequency of para. 4.2.1.1.

**4.2.1.7 Valves in Systems Out of Service.** For a valve in a system declared inoperable or not required to be operable, the exercising test schedule need not be followed. Within 3 months prior to placing the system in an operable status, the valves shall be exercised and the schedule followed in accordance with requirements of this Part.

**4.2.1.8 Stroke Time Acceptance Criteria.** Test results shall be compared to the initial reference values or reference values established in accordance with paras. 3.4 and 3.5.

(a) Electric-motor-operated valves with reference stroke times greater than 10 sec shall exhibit no more than ±15% change in stroke time when compared to the reference value.

(b) Other power-operated valves with reference stroke times greater than 10 sec shall exhibit no more than ±25% change in stroke time when compared to the reference value.

(c) Electric-motor-operated valves with reference stroke times less than or equal to 10 sec shall exhibit no more than a ±25% or ±1 sec change in stroke time, whichever is greater, when compared to the reference value.

(d) Other power-operated valves with reference stroke times less than or equal to 10 sec shall exhibit no more than ±50% change in stroke time when compared to the reference value.

(e) Valves that stroke in less than 2 sec may be exempted from (c) and (d) above. In such cases the maximum limiting stroke time shall be 2 sec.

**4.2.1.9 Corrective Action**

(a) If a valve fails to exhibit the required change of obturator position or exceeds the limiting values of full-stroke time [see para. 4.2.1.4(a)], the valve shall be immediately declared inoperable.

(b) Valves with measured stroke times which do not meet the acceptance criteria of para. 4.2.1.8 shall be immediately retested or declared inoperable. If the valve is retested and the second set of data also does not meet the acceptance criteria, the data shall be analyzed within 96 hr to verify that the new stroke time represents acceptable valve operation, or the valve shall be declared inoperable. If the second set of data meets the acceptance criteria, the cause of the initial deviation shall be analyzed and the results documented in the record of tests (see para. 6.3).

(c) Valves declared inoperable may be repaired, replaced, or the data may be analyzed to determine the cause of the deviation and the valve shown to be operating acceptably.

(d) Valve operability based upon analysis shall down, except as specified in (g) below. Such exercise is not required if the time period since the previous full-stroke exercise is less than 3 months.

(g) valve exercising during cold shutdown shall commence within 48 hr of achieving cold shutdown, and continue until all testing is complete or the plant is ready to return to power. For extended outages, testing need not be commenced in 48 hr provided all valves required to be tested during cold shutdown will be tested prior to plant startup. However, it is not the intent of this Part to keep the plant in cold shutdown in order to complete cold shutdown testing.

(h) all valve testing required to be performed during a refueling outage shall be completed prior to returning the plant to operation.

**4.2.1.3 Valve Obturator Movement.** The necessary valve obturator movement shall be determined by exercising the valve while observing an appropriate indicator, such as indicating lights which signal the required change of obturator position, or by observing other evidence, such as changes in system pressure, flow rate, level, or temperature, which reflect change of obturator position.

**4.2.1.4 Power-Operated Valve Stroke Testing**

(a) The limiting value(s) of full-stroke time of each power-operated valve shall be specified by the Owner.

(b) The stroke time of all power-operated valves shall be measured to at least the nearest second.

(c) Any abnormality or erratic action shall be recorded (see para. 6.3), and an evaluation shall be made regarding need for corrective action.

**4.2.1.5 Valves in Regular Use.** Valves which operate in the course of plant operation at a frequency which would satisfy the exercising requirements of this Part need not be additionally exercised, provided that the observations otherwise required for testing are made and analyzed during such operation and are recorded in the plant record at intervals no greater than specified in para. 4.2.1.1.

**4.2.1.6 Fail-Safe Valves.** Valves with fail-safe actuators shall be tested by observing the operation of the actuator upon loss of valve actuating power in accordance with the exercising frequency of para. 4.2.1.1.

**4.2.1.7 Valves in Systems Out of Service.** For a valve in a system declared inoperable or not required to be operable, the exercising test schedule need not be followed. Within 3 months prior to placing the system in an operable status, the valves shall be exercised and the schedule followed in accordance with requirements of this Part.

**4.2.1.8 Stroke Time Acceptance Criteria.** Test results shall be compared to the initial reference values or reference values established in accordance with paras. 3.4 and 3.5.

(a) Electric-motor-operated valves with reference stroke times greater than 10 sec shall exhibit no more than ±15% change in stroke time when compared to the reference value.

(b) Other power-operated valves with reference stroke times greater than 10 sec shall exhibit no more than ±25% change in stroke time when compared to the reference value.

(c) Electric-motor-operated valves with reference stroke times less than or equal to 10 sec shall exhibit no more than a ±25% or ±1 sec change in stroke time, whichever is greater, when compared to the reference value.

(d) Other power-operated valves with reference stroke times less than or equal to 10 sec shall exhibit no more than ±50% change in stroke time when compared to the reference value.

(e) Valves that stroke in less than 2 sec may be exempted from (c) and (d) above. In such cases the maximum limiting stroke time shall be 2 sec.

**4.2.1.9 Corrective Action**

(a) If a valve fails to exhibit the required change of obturator position or exceeds the limiting values of full-stroke time [see para. 4.2.1.4(a)], the valve shall be immediately declared inoperable.

(b) Valves with measured stroke times which do not meet the acceptance criteria of para. 4.2.1.8 shall be immediately retested or declared inoperable. If the valve is retested and the second set of data also does not meet the acceptance criteria, the data shall be analyzed within 96 hr to verify that the new stroke time represents acceptable valve operation, or the valve shall be declared inoperable. If the second set of data meets the acceptance criteria, the cause of the initial deviation shall be analyzed and the results documented in the record of tests (see para. 6.3).

(c) Valves declared inoperable may be repaired, replaced, or the data may be analyzed to determine the cause of the deviation and the valve shown to be operating acceptably.

(d) Valve operability based upon analysis shall have the results of the analysis recorded in the record of tests (see para. 6.3).

(e) Prior to returning a repaired or replacement valve to service, a test demonstrating satisfactory operation shall be performed.

4.2.2 Valve Seat Leakage Rate Test

4.2.2.1 Scope. Category A valves shall be leakage tested, except that valves which function in the course of plant operation in a manner that demonstrates functionally adequate seat leak-tightness need not be additionally leakage tested. In such cases, the valve record shall provide the basis for the conclusion that operational observations constitute satisfactory demonstration.

4.2.2.2 Containment Isolation Valves. Category A valves, which are containment isolation valves, shall be tested in accordance with Federal Regulation 10CFR50, Appendix J. Containment isolation valves which also provide a reactor coolant system pressure isolation function shall additionally be tested in accordance with para. 4.2.2.3.

4.2.2.3 Leakage Rate for Other Than Containment Isolation Valves. Category A valves, which perform a function other than containment isolation, shall be seat leakage tested to verify their leak-tight integrity. Valve closure prior to seat leakage testing shall be by using the valve operator with no additional closing force applied.

(a) *Frequency*. Tests shall be conducted at least once every 2 years.

(b) *Differential Test Pressure*. Valve seat leakage tests shall be made with the pressure differential in the same direction as when the valve is performing its function, with the following exceptions.

(1) Globe-type valves may be tested with pressure under the seat.

(2) Butterfly valves may be tested in either direction, provided their seat construction is designed for sealing against pressure on either side.

(3) Double-disk gate valves may be tested by pressurizing between the disks.

(4) Leakage tests involving pressure differentials lower than function pressure differentials are permitted in those types of valves in which service pressure will tend to diminish the overall leakage channel opening, as by pressing the disk into or onto the seat with greater force. Gate valves, check valves, and globe-type valves, having function pressure differential applied over the seat, are examples of valve applications satisfying this requirement. When leakage tests are made in such cases using pressures lower than function maximum pressure differential, the observed leakage shall be adjusted to the function maximum pressure differential value. This adjustment shall be made by calculation appropriate to the test media and the ratio between test and function pressure differential, assuming leakage to be directly proportional to the pressure differential to the one-half power.

(5) Valves not qualifying for reduced pressure testing as defined in (4) above shall be tested at full maximum functional pressure differential.

(c) *Seat Leakage Measurement*. Valve seat leakage shall be determined by one of the following methods:

(1) measuring leakage through a downstream telltale connection while maintaining test pressure on one side of the valve; or

(2) measuring the feed rate required to maintain test pressure in the test volume or between two seats of a gate valve, provided the total apparent leakage rate is charged to the valve or valve combination or gate valve seat being tested, and that the conditions required by (b) above are satisfied; or

(3) determining leakage by measuring pressure decay in the test volume, provided the total apparent leakage rate is charged to the valve or valve combination or gate valve seat being tested, and that the conditions required by (b) above are satisfied.

(d) *Test Medium*. The test medium shall be specified by the Owner.

(e) *Analysis of Leakage Rates*. Leakage rate measurements shall be compared with the permissible leakage rates specified by the plant Owner for a specific valve or valve combination. If leakage rates are not specified by the Owner, the following rates shall be permissible:

(1) for water 0.5$D$ gpm or 5 gpm, whichever is less, at function pressure differential;

(2) for air, at function pressure differential, 7.5$D$ standard cu ft/day

where

$D$ = nominal valve size, in.

(f) *Corrective Action*. Valves or valve combinations with leakage rates exceeding the values specified by the Owner in (e) above shall be declared inoperable and either repaired or replaced. A retest demonstrating acceptable operation shall be performed following any required corrective action before the valve is returned to service.

### 4.3 Inservice Tests for Category C Valves

**4.3.1 Safety Valve and Relief Valve Tests.** Safety and relief valves shall meet the inservice test requirements of Part 1.

**4.3.2 Exercising Tests for Check Valves**

**4.3.2.1 Exercising Test Frequency.** Check valves shall be exercised nominally every 3 months, except as provided by paras. 4.3.2.2, 4.3.2.3, 4.3.2.4, and 4.3.2.5.

**4.3.2.2 Exercising Requirements.** Valves shall be exercised as follows.

(*a*) During plant operation, each check valve shall be exercised or examined in a manner which verifies obturator travel to the closed, full-open or partially open position required to fulfill its function.

(*b*) If full-stroke exercising during plant operation is not practicable it may be limited to part-stroke during plant operation and full-stroke during cold shutdowns.

(*c*) If exercising is not practicable during plant operation, it may be limited to full-stroke exercising during cold shutdowns.

(*d*) If exercising is not practicable during plant operation and full-stroke during cold shutdowns is also not practicable, it may be limited to part-stroke during cold shutdowns, and full-stroke during refueling outages.

(*e*) If exercising is not practicable during plant operation or cold shutdowns, it may be limited to full-stroke during refueling outages.

(*f*) Valves full-stroke exercised at shutdowns shall be exercised during each shutdown, except as specified in (g) below. Such exercise is not required if the time period since the previous full-stroke exercise is less than 3 months.

(*g*) Valve exercising shall commence within 48 hr of achieving cold shutdown, and continue until all testing is complete or the plant is ready to return to power. For extended outages, testing need not be commenced in 48 hr provided all valves required to be tested during cold shutdown will be tested prior to plant startup. However, it is not the intent of this Part to keep the plant in cold shutdown in order to complete cold shutdown testing.

(*h*) All valve testing required to be performed during a refueling outage shall be completed prior to returning the plant to operation.

**4.3.2.3 Valves in Regular Use.** Check valves which operate in the course of plant operation at a frequency which would satisfy the exercising requirements of this Part need not be additionally exercised provided that the observations otherwise required for testing are made and analyzed during such operation and are recorded in the plant records at intervals no greater than specified in para. 4.3.2.1.

**4.3.2.4 Valve Obturator Movement**

(*a*) The necessary valve obturator movement shall be demonstrated by exercising the valve and observing that either the obturator travels to the seat on cessation or reversal of flow, or opens to the position required to fulfill its function, as specified in para. 1.1, or both.

Observation may be by observing a direct indicator such as a position indicating device, or by other indicator(s) such as changes in system pressure, flow rate, level, temperature, seat leakage testing or other positive means.

(*b*) If a manual mechanical exerciser is used to move the obturator, the force or torque required to initiate movement (breakaway) shall be measured and recorded. The breakaway force shall not vary by more than 50% from the established reference value. The reference value used shall be the value obtained when the valve is known to be operating properly and shall be taken under conditions as close as practicable to the conditions under which the valve will be tested, e.g., wet vs. dry, equivalent static head, etc.

(*c*) As an alternative to the testing in (a) or (b) above, disassembly every refueling outage to verify operability of check valves may be used.

**4.3.2.5 Valves in Systems Out of Service.** For a valve(s) in a system declared inoperable or not required to be operable, the exercising test schedule need not be followed. If the test schedule is not followed, within 3 months prior to placing the system in an operable status the valve(s) shall be exercised and the schedule followed in accordance with requirements of this Part.

**4.3.2.6 Corrective Action.** If a check valve fails to exhibit the required change of obturator position it shall be declared inoperable. A retest showing acceptable performance shall be run following any required corrective action before the valve is returned to service.

### 4.4 Inservice Tests for Category D Valves

**4.4.1 Explosively Actuated Valve Tests**

(*a*) A record of the service life of each charge in each valve shall be maintained. This record shall include the date of manufacture, batch number, installation date, and the date when service life expires based on manufacturer's recommendations. In no case shall the service life exceed 10 years.

(*b*) Concurrent with the first test and at least once every 2 years, the service life records of each valve shall be reviewed to verify that the service lives of the charges have not been exceeded and will not be exceeded before the next refueling. The Owner shall take appropriate actions to ensure charge service lives are not exceeded.

(*c*) At least 20% of the charges in explosively actuated valves shall be fired and replaced at least once every 2 years. If a charge fails to fire, all charges with the same batch number shall be removed, discarded, and replaced with charges from a different batch.

(*d*) Replacement charges shall be from batches from which a sample charge shall have been tested satisfactorily and with a service life such that the requirements of para. 4.4.1(b) are met for subsequent inspection periods.

**4.4.2 Rupture Disk Tests.** Rupture discs shall meet the requirements for nonreclosing pressure relief devices of Part 1.

## 5 ACCEPTANCE CRITERIA AND CORRECTIVE ACTION

Acceptance criteria for Category A, B, C, and D valve testing are contained in paras. 4.2, 4.3, and 4.4 of this Part. Corrective action for each category valve test showing discrepancies is contained in paras. 4.2, 4.3, and 4.4 of this Part.

## 6 RECORDS AND REPORTS

### 6.1 Valve Records

The Owner shall maintain a record which shall include the following for each valve covered by this Part:

(*a*) the manufacturer and manufacturer's model and serial or other unique identification number

(*b*) a copy or summary of the manufacturer's acceptance test report if available

(*c*) preservice test results

(*d*) limiting value of full stroke time specified in para. 4.2.1.4(a)

### 6.2 Test Plans

The Owner shall maintain a record of test plans which shall include the following:

(*a*) identification of valves subject to test

(*b*) category of each valve

(*c*) tests to be performed

(*d*) justification for deferral of stroke testing in accordance with paras. 4.2.1.2 or 4.3.2.2

### 6.3 Record of Tests

The Owner shall maintain a record of each test which shall include the following:

(*a*) valve identification

(*b*) date of test

(*c*) reason for test (e.g., post maintenance, routine inservice test establishing reference values, etc.)

(*d*) values of measured parameters

(*e*) identification of instruments used

(*f*) comparisons with allowable ranges of test values and analysis of deviations

(*g*) requirement for corrective action

(*h*) signature of the person or persons responsible for conducting and analyzing the test

### 6.4 Record of Corrective Action

The Owner shall maintain records of corrective action which shall include a summary of the corrections made and the subsequent inservice tests and confirmation of operation adequacy (see para. 3.4), and the signature of the individual responsible for corrective action and verification of results.