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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DEC 2 0 1990

MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

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

SUBJECT: PROPOSED AMENDMENT TO 10 CFR § 50.55a -- CODES AND STANDARDS

Enclosed for the information of the Advisory Committee on Reactor Safeguards is a proposed rule (and supporting regulatory analysis) that would amend § 50.55a of 10 CFR Part 50 which incorporates by reference national codes and standards for the construction, inservice inspection, and inservice testing of nuclear power plant components.

This section of the regulations incorporates by reference Division 1 rules of Section III, "Rules for the Construction of Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) and Division 1 rules of Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" of the ASME Code. ASME procedures provide that editions of the ASME Code be revised every 3 years and that addenda to the editions be issued on an annual basis. The present reference endorses all addenda and editions through the 1986 Edition for Section III, Division 1, and subject to certain limitations and modifications, addenda and editions through the 1986 Edition for Section XI, Division 1.

The proposed rule was discussed with the CRGR at Meeting No. 193, on October 24, 1990. At that time, the CRGR recommended that the staff move forward with the proposed rulemaking on the condition of making a number of editorial and administrative revisions to the proposed rule and supporting regulatory analysis. These revisions have been incorporated into the proposed rulemaking package.

The proposed rule would incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, and Section XI, Division 1. These have been reviewed by the staff and, with one proposed modification to Section XI, found to be acceptable. The proposed modification relates to inservice testing of containment isolation valves that do not provide a reactor coolant system pressure isolation function. The modification would, for these valves, substantially prese[illegible] the existing requirements for analysis of leakage rates and corrective act[illegible] that have been deleted in the later addenda and edition. The staff feels that the proposed modification is necessary to ensure that existing safety margins on valve leak-tightness are retained. Based upon a recommendation from the CRGR, the supplementary information to the proposed rulemaking contains a specific request for comments that might provide insight and justification, based upon plant operating experiences, relative to the need for revising or possibly eliminating the proposed modification.

In addition to incorporating the noted addenda and edition with the specified modification, the proposed rule includes an augmented reactor vessel examination. The proposed augmented reactor vessel examination 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 staff has determined that such an augmented examination is justified because many licensees have performed only very limited examinations of their reactor vessel, and because recent information shows reactor vessel materials to be more vulnerable to degradation than previously thought.

The enclosed notice of proposed rulemaking is presently being routed for Office concurrence. Following that concurrence, it will be transmitted to the Office of the Federal Register. The comment period for the proposed rule will be 75 days.

Eric S. Beckjord

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosures:
1. Federal Register Notice
2. Regulatory Analysis

[7590-01]

Enclosure 1

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150-AD05

Codes and Standards for Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Commission proposes to amend its regulations to incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (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 modification. 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. The ASME Code addenda and edition being incorporated by reference

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provide updated rules for the construction of light-water-cooled nuclear power plant components, and for the inservice inspection and inservice testing of those components. Adoption of this proposed amendment would permit the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power plant components; would require expedited implementation of the expanded reactor vessel shell weld examinations specified in the 1989 Edition of Section XI; and would more clearly distinguish in the regulations the requirements for inservice testing from those for inservice inspection.

DATES: Comment period expires (75 days after publication in the Federal Register). Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Send comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555. ATTN: Docketing and Service Branch. Deliver comments to: 11555 Rockville Pike, Rockville, Maryland, between 7:45 a.m. and 4:15 p.m. Monday through Friday. Examine comments received at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Mr. G. C. Millman, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-3848.

SUPPLEMENTARY INFORMATION: On May 5, 1988, the Nuclear Regulatory Commission published in the Federal Register (53 FR 16051) an amendment to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," which incorporated by reference new addenda and a new edition 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 Edition are replaced in their entirety by the referenced rules of Part 6 and Part 10, respectively. The NRC believes 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.

The NRC is particularly interested in receiving comments on the following discussed basis for and content of the proposed modification to Subsection IWV of the 1988 Addenda and 1989 Edition of Section XI, Division 1. 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 a valve leakage rate test. 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 Edition 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. The NRC's concern regarding the revision stems from the findings of two independent reviews of containment leakage rate failure experiences. Both reviews conclude from analysis of Appendix J leak test results, which included analysis of valve leakage, that containment leakage during operation would exceed plant technical specification limits approximately 30 percent of the time. This indicates a need to improve, rather than relax, the present requirements concerning containment test, leak monitoring, and maintenance programs, including the ASME Section XI requirement for valve leak rate analysis. It has yet to be demonstrated by analysis of more recent and comprehensive containment leakage test experiences that containment leakage integrity can be improved to an acceptable level without implementation of a rigorous valve leak rate test program in conjunction with the present Section XI requirement for leak rate analysis.

In proposing the following modification, the NRC specifically requests comments that would provide insight and justification, based upon plant experiences, relative to the need for revising or possibly eliminating the proposed modification. 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 proposed modification would

substantially 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.

Section XI Subsection IWP and Subsection IWV editions and addenda, published up through the 1987 addenda, address Class 1, Class 2, and Class 3 pumps and valves, respectively, that perform specific safety functions. The reference to Part 6 in Subsection IWP and to Part 10 in Subsection IWV in the 1988 Addenda and 1989 Edition expands the scope of these subsections to potentially include certain pumps and valves that are not classified as Class 1, Class 2, or Class 3. Because § 50.55a, at this time, only specifies requirements for pumps and valves that are designated Class 1, Class 2, or Class 3, this proposed amendment does not impose requirements on those pumps and valves that are not Class 1, Class 2, or Class 3, but would be included in the expanded scope of Subsection IWP and Subsection IWV in the 1988 Addenda and 1989 Edition. However, Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Program," addresses this issue and notes in Position 11 that "The intent of 10 CFR 50 Appendix A, GDC-1, and Appendix B, Criterion XI, is that all components, such as pumps and valves, necessary for safe operation are to be tested to demonstrate that they will perform satisfactorily in service. Therefore, while 10 CFR 50.55a delineates the testing requirements for ASME Code Class 1, 2, and 3 pumps and valves, the testing of pumps and valves is not to be limited to only those covered by

10 CFR 50.55a."

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.

Recent information from reactor vessel material surveillance programs, and observed flaws in certain operating reactor and steam generator vessels, reveal the potential susceptibility of reactor vessel materials to degradation. Because of these experiences and the limited examinations performed to date on some reactor vessels, the NRC is concerned with the length of time that 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 that certain reactor vessel materials undergo greater radiation damage than previously expected, (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) significant service induced cracking has occurred in large vessels (i.e., pressirizer, steam generators) designed and fabricated to the ASME Code.

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. 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 maintenance of the level of protection presumed by the regulations requires more than compliance to existing regulatory requirements.

The NRC has determined that the proposed augmented reactor vessel examination would result in a substantial increase in the overall protection of the public health and safety, and that the costs of implementation would be justified in view of the increased protection. The backfit analysis required by § 50.109, "Backfitting," is provided as part of 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.

In order to ensure the applicability of the proposed augmented examination to all licensees, § 50.55a(g)(6)(ii)(A)(1) would revoke all previously granted reliefs to licensees for reactor vessel shell weld examinations for the inservice inspection interval that would be in effect

when the rule becomes effective. This is consistent with the ongoing development schedule for equipment and techniques that would permit those licensees with limited accessibility to implement the proposed augmented examination. 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)(2) would require 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)(3) and (4). Section 50.55a(g)(6)(ii)(A)(2) would specifically permit 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)(3) would permit 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 would specifically prohibit the use of the deferred augmented examination as a substitute for reactor vessel shell weld examinations scheduled for the inspection interval in effect when the rule becomes effective. The intent is to ensure that the examinations are deferred only when necessary and not to have the proposed rule encourage a 40-month delay in reactor vessel shell weld examinations.

Section 50.55a(g)(6)(ii)(A)(3) would permit using the deferred examination, with a condition, as a substitute for reactor vessel shell weld examinations scheduled for the inspection interval in which the deferred examinations are performed. The condition is that subsequent reactor vessel shell weld examinations for successive inspection intervals be performed in the first period of the inspection interval. This condition is necessary to prevent a potential 160-month gap between reactor vessel shell weld examinations. This gap would occur if a plant used the deferred examination performed in the first period as a substitute for the scheduled examination and then deferred the examination for the next inspection interval to the end of that interval as permitted by Section XI.

Proposed § 50.55a(g)(6)(ii)(A)(4) 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 inservice inspection interval in effect when the proposed rule becomes effective 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 would be performed based on ASME Section XI, 1989 Edition, or later Code edition and addenda 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, Class 2, and Class 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.

Finding of No Significant Environmental Impact

The Commission has determined under the National Environmental Policy Act of 1969, as amended, and the Commission's regulations in Subpart A of 10 CFR Part 51, that this rule, if adopted would not be a major Federal action significantly affecting the quality of the human environment and therefore an environmental impact statement is not required.

The proposed rule is one part of a regulatory framework directed to ensuring pressure vessel integrity, and the operational readiness of pumps and valves. Therefore, in the general sense, the proposed rule would have a positive impact on the environment. The proposed rule would incorporate by reference into the NRC regulations improved rules contained in the ASME Code for the construction, inservice inspection and inservice testing of components used in nuclear power plants. In addition, the proposed rule would require an

augmented examination of reactor vessel shell welds to further ensure the structural integrity of the reactor vessel. Actions required of applicants and licensees to implement the proposed rule are of a routine nature that should not increase the potential for a negative environmental impact.

The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from Gilbert C. Millman, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-3848.

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 110 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, based on the analysis required by § 50.109(a)(3) which is provided in the regulatory analysis, that the backfit that would be imposed by the proposed augmented reactor vessel examination would result in a substantial increase in the overall protection of the public health and safety, and that the direct and indirect costs of implementation would be justified in view of the increased protection.

The incorporation by reference into the regulations of later editions and addenda of Section III and Section XI of the ASME Code is not a backfit because Section III requirements apply only to new construction, except as voluntarily implemented by licensees, and because updated Section XI requirements are an integral part of the longstanding § 50.55a(g)(4)(ii) requirement to update inservice inspection and inservice testing programs to 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, subject to specified limitations and modifications. The proposed modification to Part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987 is not a backfit because it simply retains a requirement that licensees now are required to implement in accordance with § 50.55a(g).

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 made in accordance with paragraph 4.2.2.3(f) of Part 10 of ASME/ANSI OMa-1988 Addenda to ASME/ANSI OM-1987.

* * * * *

(f) <u>Inservice testing requirements</u>.

(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* 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[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 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 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 the limitations and modifications listed in paragraph (b) of this section, and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met.

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

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

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

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

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

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

(g) <u>Inservice inspection requirements</u>. 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 vessels, 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[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 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[6] applied to the construction of the particular component.

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

(iii) [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 paragraphs (g)(2) and (g)(3) of this section and that are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

(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 previously granted reliefs under § 50.55a to licensees for the examination of reactor vessel shell welds specified in Item B1.10 of

Examination Category B-A, "Pressure Retaining Welds in Reactor Vessel," in Table IWB-2500-1 of Subsection IWB in applicable edition and addenda of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code, during the inservice inspection interval in effect on ________ (effective date of rule will be inserted) are hereby revoked.

(2) 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)(3) and (4). 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).

(3) 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)(2) to the first period of the next inspection interval. The deferred augmented examination may not be used as a substitute for the reactor vessel shell weld examination scheduled for implementation during the

inservice inspection interval in effect on ________ (effective date of rule will be inserted). The deferred augmented examination may be used as a substitute for the reactor vessel shell weld examination normally scheduled for the inspection interval in which the deferred examination is performed. If the deferred augmented examination is used as a substitute for the normally scheduled reactor vessel shell weld examination, subsequent reactor vessel shell weld examinations must be performed during the first period of successive inspection intervals.

<u>(4)</u> 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)<u>(2)</u> 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) <u>Protection systems.</u> 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[6] 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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[6] ASME Code cases that have been determined suitable for use by the Commission staff are listed in NRC Regulatory Guide 1.84, "Design and Code Case Acceptability -- ASME Section III Division 1," NRC Regulatory Guide 1.85, "Materials Code Case Acceptability -- ASME Section III Division 1," and NRC Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptabilty -- ASME Section XI Division 1." The use of other Code cases may be authorized by the Director of the Office of Nuclear Reactor Regulation upon request pursuant to § 50.55a(a)(3).

[7] 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 Engineers, 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 tc the provisions of § 2.101(a-1) and Subpart F of Part 2 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 ______________1991.

For the Nuclear Regulatory Commission.

____

James M. Taylor,

Executive Director for Operations.

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, addenda through the Winter 1985 Addenda and editions through the 1986 Edition, and for Section XI, Division 1, addenda through the Winter 1985 Addenda and 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 reactor 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 inservice 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 revisions to the ASME Code are properly considered from all sides and that the cost and benefit of each action are reasonably balanced. Should the NRC feel, in the interest of safety, that the ASME Code should be modified, 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. The NRC's concern regarding the revision stems from the findings of two independent reviews[1,2] of containment leakage rate failure experiences. Both reviews conclude from analysis of Appendix J leak test results that containment leakage during plant operation would exceed plant technical specification limits approximately 30 percent of the time. This indicates a need to improve, rather than relax, the present requirements concerning containment test, leak monitoring, and maintenance programs, including the ASME Section XI requirement for valve leak rate analysis. It has yet to be demonstrated by analysis of more recent and comprehensive containment leakage test experiences that containment leakage integrity can be improved to an acceptable level without implementation of a rigorous valve leak rate test program in conjunction with the present Section XI requirement for leak rate analysis. 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 the existing safety margin on valve leak-tightness is 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 justified because many licensees have performed only very limited examinations of their reactor vessels, and because of recent information that shows reactor vessel materials

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[1] Weinstein, M.B. "Primary Containment Leakage Integrity: Availability and Review of Failure Experience." Nuclear Safety 21(5):618-632 (1980).

[2] NUREG/CR-4220, "Reliability Analysis of Containment Isolation Systems," Pacific Northwest Laboratories, June 1985.

to be more vulnerable to degradation than previously thought. The augmented examination is necessary to ensure that the reactor vessel maintains a very low probability of failure. It has been concluded that the backfit that would be imposed by the proposed augmented reactor vessel examination would result in a substantial increase in the overall protection of the public health and safety, and that the cost of implementation would be justified in view of the increased protection.

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).

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-reactors 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, Class 2, and Class 3 components, and Section XI, Division 1, of the ASME Code for inservice inspection and inservice testing of these components. 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 not consistent with an NRC staff position, an exception may be taken to endorsing that portion of the Code or supplementary criteria may be incorporated to make the item consistent with the staff position.

Section 50.55a presently endorses addenda through the Winter 1985 Addenda, and editions through the 1986 Edition for Section III, Division 1, and addenda through the Winter 1985 Addenda, and 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 an unacceptable change 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 specified modification. This modification would require implementation of certain existing 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. 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 potential 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 (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. On the basis of a cost-benefit analysis provided in Appendix A, "Backfit Analysis," the NRC has concluded that the benefit to be gained in terms of public health and safety from the proposed augmented examination of reactor vessel shell welds is commensurate with the cost to affected utilities.

Certain licensees previously were granted relief, for accessibility reasons, from reactor vessel shell weld examinations for the inspection interval that would be in effect when the proposed rule becomes effective. Equipment and techniques are presently being developed, on a schedule consistent with the implementation schedule for the proposed augmented examination, that would permit implementation of the proposed augmented examination by those licensees that previously required relief from reactor vessel shell weld examinations. In order to ensure applicablity of the proposed augmented examination to all licensees and to avoid possible confusion regarding NRC intent regarding applicability of the proposed augmented examination to those licensees that previously had been granted relief for reactor vessel shell weld examinations, the proposed rule specifically revokes all previously granted reliefs for

reactor vessel shell weld examinations for the inspection interval when the rule becomes effective.

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 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:

- 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.

- Require augmented examination of reactor vessel shell welds.

- 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 technical and administrative improvements to the ASME Code would not be incorporated by reference into the regulations and 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.

Another alternative to incorporating by reference these later requirements of Section III, Division 1, and Section XI, Division 1, is for the NRC staff itself to develop updated rules for construction, inservice inspection, and inservice testing on a periodic basis, taking into account advances in the various technologies, and to incorporate these rules directly into the regulations. This alternative is not considered attractive because it would be very manpower intensive on the part of the staff, and would not include the comprehensive indepth input from industry experts that is achieved through development of the ASME Code by the consensus process.

Alternatives to each revision in Section III and Section XI of the ASME Code were considered from the perspective of whether the revision was acceptable as is, or should be modified. It was determined that only one revision required

modification. The proposed modification would cause the retention of an existing requirement to analyze leak rate data for containment isolation valves. Since the modification would impose an existing requirement, there would be no change in burden to licensees. This modification is discussed in detail in Section 4.A of this regulatory analysis.

Augmented 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. The time-frames associated with implementing the reactor vessel examination through routine updating are shown on a generic basis in Appendix A Table A-1. The time-frames associated with implementing the reactor vessel examination through the proposed augmented examination are shown on a generic basis in Appendix A Table A-2. The time-frames associated with implementing the reactor vessel examinations for specific plants on the basis of routine updating contrasted to the proposed augmented examination are shown in Appendix A Table A-3.

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 later addenda and edition of the ASME Code would 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 generally 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.

In general, the Section III revisions would improve the procurement process through the use of improved material testing procedures and would improve the consistency of component and component support design through redefinition of the jurisdictional boundaries. The Section XI revisions serve to make the rules for the testing of pumps, valves, and snubbers consistent with regulatory positions, and generally improve the procedures for the detection, analysis, and repair of flaws in the reactor vessel and other components and their supports.

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 fully in Section 4.B(3), below, the modification 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, based on the analysis required by 10 CFR 50.109(a)(3) and provided in Appendix A, that the backfit that would be imposed by the proposed augmented reactor vessel examination would result in a substantial increase in the overall protection of the public health and safety, and that the direct and indirect costs of implementation would be justified in view of the increased protection.

The backfit analysis 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 detailed basis for the NRC's conclusion that the backfit is justified in terms of cost/benefit.

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(g)(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(g)(4)(ii)).

(c) If a licensee determines that conformance with certain Code requirements is impractical for its facility, the licensee shall notify the Commission and submit information to support that 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 tne 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.

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 to be 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 Section 1, 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 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
No effect on other NRC regulatory actions.

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).

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 and representative permissible leakage rate limits are specified for valve groups. 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. This does not represent a difference from existing rules, since the Section XI testing requirements are called out in technical specifications and licensees are expected to declare a valve inoperable when the valve leakage rate exceeds the permissible leakage rate. 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.

Augmented Examination of Reactor Vessel

The NRC has concluded, based on the analysis required by § 50.109(a)(3) and described in § 50.109(c), that the proposed augmented reactor vessel examination would result in a substantial increase in the overall protection

of the public health and safety, and that the direct and indirect costs of implementation would be justified in view of the increased protection. Section 50.109(c) identifies 9 factors to be considered, as appropriate, in evaluating the proposed backfit. Each of the 9 factors is addressed in the backfit analysis provided in Appendix A.

Separation of ISI and IST Requirements

No backfit.

(4) Impact on Requirements of Other Government Agencies

The proposed rule would not be a major Federal action significantly affecting the quality of the human environment as determined under the National Environmental Policy Act of 1969, as amended, and the Commission regulations in Subpart A of 10 CFR Part 51. Therefore, an environmental impact statement is not required. A Finding of No Significant Environmental Impact, which is supported by an environmental assessment in Appendix C, is provided in the proposed rule.

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 D.

This amendment to § 50.55a affects only the construction 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 Administration at 13 CFR Part 121. Since these companies are dominant in their service areas, this amendment does not fall into 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 without undue burden on holders of construction permits and operating licenses. 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 result in a substantial increase in the overall protection of the public health and safety by ensuring comprehensive and timely reactor vessel examinations whose costs are justified by the increased protection; and the separation of inspection and testing requirements would emphasize the importance that NRC places on both inservice inspection and inservice testing programs, and would clarify the requirements in the individual 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 would be an extension of the existing reactor vessel examinations and other inservice examination requirements and, therefore, 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. The industry effort to develop inspection tools and procedures that would permit accessing a high percentage of BWR shell welds is consistent with the estimated effective date of the rule. However, in certain cases, it may be necessary to grant exemptions from the full requirement of the proposed augmented reactor vessel examination. See Appendix A, Sections 7 and 8.

Separation of ISI and IST Requirements

No implementation problems are anticipated.

Appendix A

Backfit Analysis

The NRC has concluded, based on the analysis required by § 50.109(a)(3) and described in § 50.109(c), that there would be a substantial increase in the overall protection of the public health and safety were the proposed augmented reactor vessel examination to be implemented and that the direct and indirect costs of implementation would be justified in view of the increased protection. The response to each of the 9 factors specified in § 50.109(c) is provided, below, in this backfit analysis for the proposed augmented reactor vessel examination.

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

a. Statement of Objective

The NRC staff is concerned that the limited examination performed to date on some reactor vessels and planned for others is inconsistent with the importance of the reactor vessel, and with information that shows the potential susceptibility of reactor vessel materials to degradation that could compromise the integrity of the reactor vessel. The proposed augmented examination would require all plants to perform an inservice examination of essentially 100% of the length of all reactor vessel shell welds during the inservice inspection interval in effect when the proposed rule becomes effective, or, at the latest, during the first period of the following inspection interval (i.e., plants that were within 40 months of the end of the inspection interval when this proposed rule became effective would be permitted to defer the augmented examination until the first period of the next inspection interval). The augmented examination would automatically be satisfied by those plants that previously had performed, during the inspection

interval in effect when the proposed rule becomes effective, or, previously had scheduled for that inspection interval an examination of essentially 100% of the reactor vessel shell welds.

The augmented examination would shorten the time by which utilities would be required to implement the inservice examination requirements for reactor vessel shell welds during the 2nd, 3rd, and 4th inspection intervals that have been incorporated into the 1988 Addenda and 1989 Edition of Section XI of the ASME Code. The reduction in time to implement the rules could be, depending upon plant, from 6.7 to 16.7 years. The proposed augmented examination would be a one-time requirement that would ensure that all plants would have performed a 100% examination of the reactor vessel shell welds within 10 years, and the majority of plants within 5 years, of the effective date of the rule.

b. Basis for Required Objective

(1) Summary of Basis

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 1975 Addenda, but since that addenda until the 1988 Addenda has required examination of only one circumferential and one longitudinal beltline weld during the 2nd, 3rd, and 4th inspection intervals. Plants that used Section XI prior to the Winter 1975 Addenda for the first interval, and subsequent addenda for the 2nd and 3rd intervals, would have been required to exam 5% of each circumferential weld and 10% of each longitudinal weld during the first interval, and essentially 100% of one circumferential weld and one longitudinal weld in the beltline area during the 2nd and successive intervals.

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 to revise 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."

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). However, this could, in general, result in plants not implementing the revised reactor vessel examination from 13 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 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 examination would expedite implementation of the reactor vessel examination requirements in the 1989 Edition for 2nd, 3rd, and 4th intervals on all plants. 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 the inspection interval that was in place at the time the proposed rule became effective.

(2) <u>Detailed Basis</u>

From the standpoint of safety, the reactor vessel is the most important single component in the nuclear steam supply system. There is no backup for the reactor vessel. All PRA studies assume a very low probability for failure of the reactor vessel, and failure of the reactor vessel is considered to be an incredible event that is not included as a design basis accident. An effective inservice inspection program is necessary to ensure that reactor vessel flaws do not go undetected. Discussed below, as they relate to the basis to support the proposed augmented examination, are: (i) the effects of implementing the reactor vessel examinations specified in the 1989 Edition of Section XI in accordance with routine update procedures; (ii) potential for reactor vessel degradation; and (iii) the effect of implementing the proposed augmented program.

(i) <u>Reactor Vessel Examinations in Accordance with Routine Updating</u>
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 IWB-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 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 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.

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 examination of essentially 100% of the reactor vessel shell welds. However, as detailed in Table A-1, below, implementation through routine updating could take as long as 20 years.

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 § 50.55a(b)(2). Table A-1, below, 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).

Table A-1

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 would, in general, range from 20 years (240 months) to 13 years and 4 months (160 months).

(ii) Potential for Reactor Vessel Degradation

The primary reason for the proposed augmented examination, which would expedite examination of 100% of the reactor vessel shell welds, is that with so few examinations having been performed to date on both BWR and PWR reactor vessel shell welds there is concern on the part of the staff regarding the existence of manufacturing flaws, and the initiation and propagation of flaws in these major welds during service. At present, there is no assurance that Section XI flaw acceptance criteria are being satisfied in those reactor vessel shell welds that never have been, or have been only partially, examined as part of a Section XI inservice inspection program.

Beyond the general belief that expedited implementation of the 100% reactor vessel shell weld examination is necessary to compensate for the limited reactor vessel shell examinations performed to date, there are specific concerns regarding potential degradation mechanisms for both BWR and PWR reactor vessels which support implementation of the proposed augmented examination. These concerns are discussed below.

The proposed augmented examination is needed for BWRs because of evidence that there exists a viable mechanism for initiating environmentally assisted cracks in the cladding; evidence that the environmentally assisted cracking in stainless steels (i.e., the cladding) can propagate into ferritic steel (i.e., the reactor vessel); and evidence that BWR pressure vessels are being embrittled more by neutron irradiation than would be predicted by R.G. 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." These points are addressed below.

o Viable cracking mechanism: BWR service environments are relatively severe due to the high dissolved oxygen content. In some cases, the severity of the environment has been exacerbated by poor control of the overall water chemistry. While stainless steel weld cladding is generally not susceptible to environmentally assisted cracking, the potential for stress corrosion cracking does exist in situations where welding or weld

repairs may have resulted in low ferrite content in the cladding, excessive cold working from grinding operations, and high residual stresses, which render the material susceptible to cracking in the BWR environment. Poor control of water chemistry may also aggravate this susceptibility. Cracks in cladding of BWR vessels have been known to initiate in the cladding and progress into the base metal in areas where excessive repair and grinding have been performed. There is reason to believe that a significant amount of weld repair has been made on many reactor vessels during the fabrication process, although records documenting such repairs are not always available or are difficult to obtain. Therefore, it is important that reactor vessel shell welds be examined as they may have been rendered susceptible to cracking by the manufacturing process. This susceptibility to cracking may have a long incubation period and may not manifest itself for many years. The potential for neutron irradiation to contribute to environmentally assisted cracking in the stainless steel cladding of reactor vessels cannot be dismissed.

Recently, stress corrosion cracking has occurred in the stainless steel weld cladding in the Quad Cities Unit 2 reactor vessel head[1]. Also, inspections of the Japanese Power Demonstration Reactor, conducted as part of the decommissioning of that reactor, have shown evidence of cracking in the cladding of the vessel head, nozzles, the beltline region, and the lower head.

o <u>Stress corrosion crack propagation into ferritic materials</u>: It has been assumed in the past that stress corrosion cracks that might initiate in the stainless steel cladding would not propagate into the ferritic steel because the ferritic steel was not a susceptible material. However, several recent incident of stress corrosion cracking show that it is possible for such cracks to

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[1] NRC Information Notice No. 90-29, "Cracking of Cladding and Its Heat Affected Zone in the Base Metal of a Reactor Vessel Head," April 30, 1990.

propagate into the ferritic base material. Specifically, as noted above, the stress corrosion cracking in the weld cladding in the Quad Cities Unit 2 reactor vessel head had propagated into the base metal. Additionally, there have been incidents of stress corrosion cracking in the feedwater nozzle safe-ends at Brunswick and the Chinshan plant in Taiwan that propagated from the safe-end material into the ferritic steel nozzle. While the cracking mechanism may change from intergranular stress corrosion cracking to environmentally assisted fatigue crack growth, or crevice corrosion cracking, the fact remains that growing stress corrosion cracks tend to continue to propagate into the ferritic materials. The normal service loads, coupled with the BWR environment, could be sufficient to sustain this growth and produce relatively large cracks over the life of the plant.

- Fracture Toughness of BWR reactor vessels: Recent surveillance capsule reports from U.S. BWRs indicate that the irradiation damage predicted by Regulatory Guide 1.99, Revision 2, does not conservatively predict actual irradiation damage. This is consistent with data obtained from the Japanese Power Demonstration Reactor surveillance program. Thus, predictions of BWR pressure vessel integrity made using the irradiation damage trends from the regulatory guide are not as conservative as anticipated. If there is an active cracking mechanism that could produce a crack significantly larger than the 1/4-t flaw considered in most regulatory analyses, there could be a serious safety concern for BWR reactor vessels.

A recent industry report[2] overviewed BWR reactor vessels and their degradation mechanisms from a license renewal perspective. Although, in that report, most degradation mechanisms were not considered significant for license renewal,

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[2] "Boiling Water Reactor Vessel License Renewal Industry Report," October 1989, Nuclear Management and Resources Council, Inc.

the report did identify the reactor vessel beltline[3] welds as an area that could require implementation of an aging management program for some plants. This same report noted when summarizing inspection and test requirements that: "Most older BWR plants were granted exemptions from the Section XI vessel beltline weld inspection requirements by the NRC. However, recent vessel nozzle cracking at two plants coupled with higher reference temperature shifts at a few U.S. BWR plants raise the priority of inspection. It is recommended that Section XI be followed."

Whereas the primary concern with BWRs is the growth of cracks during normal service conditions, the primary concern with PWRs is the existence of cracks that could be a concern during an operational transient, such as pressurized thermal shock (PTS) or a low temperature overpressurization (LTOP). The regulations for pressure vessel safety (e.g., 10 CFR 50 Appendix G, "Fracture Toughness Requirements " and 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events") presume that active service induced cracking does not occur. However, to ensure reactor vessel safety, the regulations (e.g., 10 CFR 50 Appendix G) and the ASME Code (e.g., Section III Appendix G, "Protection Against Nonductile Failure") require that a large crack be postulated and that an anlysis be performed to demonstrate that the reactor vessel would not fracture under severe accident loadings. To date, no reactor vessel has failed. However, there have been a number of events that raise some doubts about the presumption that active service induced cracking does not exist in PWRs.

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[3] The "beltline" is a broad area encompassing many shell welds, which is defined in 10 CFR 50 Appendix G as "the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."

Specifically, the steam generator shell cracks at Indian Point 2 and Zion[4,5,6], and the pressurizer cracks at Haddam Neck[7] show that combinations of service environment and unanticipated loads can lead to significant cracking even in large vessels that are designed to have low stresses based on ASME Code rules, are examined during fabrication, and are subjected to periodic inservice inspection. While neither the steam generator nor pressurizer cracking occurences indicate potential problems with any specific reactor vessel, the occurrences do show that unanticipated conditions can lead to unexpected cracking of large vessels. Periodic examinations can help detect such cracking long before the cracks pose a threat to reactor vessel integrity.

As vessels age, the number of anticipated and unanticipated loading cycles increases, and the need for periodic examinations to detect service induced cracking is heightened. Preexisting flaws produced by fabrication, such as underclad cracking, can grow under operational loadings to sizes that could be detected by inservice examinations. Because the fracture toughness of reactor vessel materials is continually being reduced by neutron irradiation embrittlement, the frequency of examinations should be increased to ensure cracking is detected before reaching a critical size -- a size that becomes smaller with continued operation because of the reduced fracture toughness.

In an industry report[8], similar to that for BWRs, PWR reactor vessels and their degradation mechanisms were overviewed from a license renewal

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[4] NRC Information Notice No. 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," January 26, 1990.

[5] NRC Information Notice No. 82-37, "Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating Pressurized Water Reactor," September 16, 1982.

[6] NRC Information Notice No. 85-65, "Crack Growth in Steam Generator Girth Welds," July 31, 1985.

[7] Presentation made to NRC staff on June 7, 1990 to provide results of recent inspections of Haddam Neck Pressurizer (Docket No. 50-213)

[8] "Pressurized Water Reactor Vessel License Renewal Industry Report," May 1990, Nuclear Mamgement and Resources Council, Inc.

perspective. In this report, neutron radiation embrittlement of the reactor vessel beltline region received significant emphasis in the discussion of degradation mechanisms. The issues addressed by the report relative to the management of neutron radiation embrittlement in the reactor vessel beltline region were: the reactor vessel material surveillance program, PTS, low upper shelf fracture toughness, and pressure-temperature limits. Without making a judgement relative to license renewal, it is important to note that reactor vessel integrity analyses that address these areas assume that active service induced cracking does not occur. Without more extensive examinations of the beltline and other areas of the reactor vessel, there can be no assurance that such cracking does not occur.

(iii) <u>Reactor Vessel Examinations in Accordance with Augmented Examination</u>
The NRC is concerned that the importance of the reactor vessel and the presence 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. Table A-2, below, shows the significance of the proposed expedited implementation procedure. Comparing the information in Table A-2 to that in Table A-1, it can be seen that the proposed augmented examination would, in general, have the effect of reducing the time to implement 100% examination of reactor vessels by about 10 years.

Table A-2

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

Table A-3 (see Page A-13A) shows the effect of the augmented examination on the implementation schedule for 100% reactor vessel shell weld examinations for specific plants (Table A-3 provides data for 112 plants, however, five of these plants were not included in the analyses that follow because of their ongoing extended downtimes). The effect of the proposed augmented examination is summarized in Table A-4, below, which shows the accelerated time-frames distributed by reactor type.

Table A-4

Summary of Table A-3 Estimates for Accelerating 100% Reactor Vessel Examinations

Acceleration (Years) (Augmented - Updated)	--No. of Plants Affected-- Total	PWRs	BWRs
0.0	41	27	14
6.7	22	15	7
10.0	37	28	9
16.7	7	3	4
Total	107	73	34

Table A-3

IMPACT OF UPDATED/AUGMENTED REACTOR VESSEL EXAMINATION ON PLANT IMPLEMENTATION SCHEDULE

NO.	Plant	Type	OL	Commercial	----Applicable Code Edition/Addenda---- 1st Interval	2nd Interval	3rd Interval	4th Interval	End Present Interval	(A) Update	(A) Augmented	Aug. - Update	Aug. - Effect	(B) 100% Exam
1	Big Rock Point	BWR/GE-1	05-01-64	03-29-63		77/S78			Mar-94	Mar-2004	Jul-97	-6.7	6.1	
2	Nine Mile Point 1	BWR/GE-2	12-26-74	12-01-69	74/S75	83/S83			Jun-96	Jun-2006	Jun-96	-10.0	5.0	
3	Oyster Creek	BWR/GE-2	08-01-69	12-01-69		74/S75			Oct-91	Oct-2011	Feb-95	-16.7	3.7	
4	Dresden 2	BWR/GE-3	12-22-69	06-09-70	74/S75	80/W81			Mar-92	Mar-2012	Jul-95	-16.7	4.1	
5	Dresden 3	BWR/GE-3	03-02-71	11-16-71	74/S75	77/S79			Mar-92	Mar-2012	Jul-95	-16.7	4.1	
6	Millstone 1	BWR/GE-3	10-31-86	03-01-71	71,74/S75	80/W80			Dec-90	Dec-2010	Dec-2000	-10.0	9.5	
7	Monticello	BWR/GE-3	01-09-81	06-30-71	74/S75	77/S78			May-92	May-2012	Sep-95	-16.7	4.3	
8	Pilgrim 1	BWR/GE-3	09-15-72	12-01-72	74/S75	80/W80			Dec-92	Dec-2002	Apr-96	-6.7	4.8	
9	Quad Cities 1	BWR/GE-3	12-14-72	02-18-73	74/S75	80/W80			Feb-93	Feb-2003	Jun-96	-6.7	5.0	
10	Quad Cities 2	BWR/GE-3	12-14-72	03-10-73	74/S75	80/W80			Mar-93	Mar-2003	Jul-96	-6.7	5.1	
11	Browns Ferry 1	BWR/GE-4	12-20-73	08-01-74	74/S75				Mar-87					
12	Browns Ferry 2	BWR/GE-4	08-02-74	03-01-75	74/S75				Mar-87					
13	Browns Ferry 3	BWR/GE-4	08-18-76	03-01-77	74/S75				Mar-87					
14	Brunswick 1	BWR/GE-4	11-12-76	03-18-77	77/S78	80/W81			Mar-97	Mar-2007	Mar-97	-10.0	5.8	*
15	Brunswick 2	BWR/GE-4	12-27-74	11-03-75	77/S78	80/W81			Nov-95	Nov-2005	Nov-95	-10.0	4.4	*
16	Cooper	BWR/GE-4	01-18-74	07-01-74	74/S75	80/W81			Jul-94	Jul-2004	Nov-97	-6.7	6.4	
17	Duane Arnold	BWR/GE-4	02-22-74	02-01-75	74/S75	80/W81			Nov-95	Nov-2005	Nov-95	-10.0	4.4	
18	Fermi 2	BWR/GE-4	07-15-85	01-23-88	80/W81				Jan-98	Jan-2008	(C) N/A	N/A	N/A	*
19	Fitzpatrick	BWR/GE-4	10-17-74	10-28-75	74/S75	80/W81			Jul-95	Jul-2005	Jul-95	-10.0	4.1	
20	Hatch 1	BWR/GE-4	10-13-74	12-31-75	74/S75	80/W81			Jan-96	Jan-2006	Jan-96	-10.0	4.6	
21	Hatch 2	BWR/GE-4	06-13-78	09-05-79	74/S75	80/W81			Jan-96	Jan-2006	Jan-96	-10.0	4.6	
22	Hope Creek 1	BWR/GE-4	07-25-86	12-20-86	83/S83				Dec-96	Dec-2006	N/A	N/A	N/A	*
23	Limerick 1	BWR/GE-4	08-08-85	02-01-86	80/W81				Feb-96	Feb-2006	N/A	N/A	N/A	*
24	Limerick 2	BWR/GE-4	08-25-89	01-08-90	86				Jan-2000	Jan-2010	N/A	N/A	N/A	*
25	Peach Bottom 2	BWR/GE-4	12-14-73	07-05-74	74/S75	80/W81			Jul-94	Jul-2004	Nov-97	-6.7	6.4	
26	Peach Bottom 3	BWR/GE-4	07-02-74	12-23-74	74/S75	80/W81			Dec-94	Dec-2004	Dec-94	-10.0	3.5	
27	Susquehanna 1	BWR/GE-4	11-12-82	06-08-83	80/W80				Jun-93	Jun-2003	N/A	N/A	N/A	*
28	Susquehanna 2	BWR/GE-4	06-27-84	02-12-85	80/W81				Feb-95	Feb-2005	N/A	N/A	N/A	*
29	Vermont Yankee	BWR/GE-4	02-28-73	11-30-72	74/S75	80/W80			Nov-92	Nov-2002	Mar-96	-6.7	4.8	
30	LaSalle 1	BWR/GE-5	12-03-82	01-01-84	80/W80				Jan-94	Jan-2004	N/A	N/A	N/A	*
31	LaSalle 2	BWR/GE-5	03-23-84	10-19-84	80/W80				Oct-94	Oct-2004	N/A	N/A	N/A	*
32	Nine Mile Point 2	BWR/GE-5	07-02-87		83/S83				Mar-98	Mar-2008	N/A	N/A	N/A	*
33	Washington Nuclear 2	BWR/GE-5	04-13-84	12-13-84	80/W80				Dec-94	Dec-2004	N/A	N/A	N/A	*
34	Clinton	BWR/GE-6	04-17-87	11-24-87	80/W81				Apr-97	Apr-2007	N/A	N/A	N/A	*
35	Grand Gulf 1	BWR/GE-6	11-01-84	07-01-85	77/S79				Jun-95	Jun-2005	N/A	N/A	N/A	*
36	Perry 1	BWR/GE-6	11-13-86	11-18-87	83/S83				Nov-97	Nov-2007	N/A	N/A	N/A	*
37	River Bend 1	BWR/GE-6	11-20-85	06-16-86	80/W81				Jun-96	Jun-2006	N/A	N/A	N/A	*
38	Arkansas 1	PWR/BW	05-21-74	12-19-74	74/S75	80/W80			Dec-94	Dec-2004	Dec-94	-10.0	3.5	
39	Crystal River 3	PWR/BW	01-28-77	03-13-77	74/S75				Mar-87					
40	Davis Besse 1	PWR/BW	04-22-77	07-31-78	77/S78				Sep-90	Sep-2010	Sep-2000	-10.0	9.3	*
41	Oconee 1	PWR/BW	02-06-73	07-15-73		80/W80			Mar-94	Mar-2004	Jul-97	-6.7	6.1	
42	Oconee 2	PWR/BW	10-06-73	09-09-74		80/W80			Mar-94	Mar-2004	Jul-97	-6.7	6.1	
43	Oconee 3	PWR/BW	07-19-74	12-16-74		80/W80			Mar-94	Mar-2004	Jul-97	-6.7	6.1	
44	Rancho Seco	PWR/BW	08-16-74	04-17-75	74/S75	86E			Sep-98	Sep-2008	Sep-98	-10.0	7.3	

(A) BASED ON RULE EFFECTIVE 6-1-91 (B) *: 100% RV SHELL EXAM SPECIFIED IN SXI EDITION/ADDENDA (C) N/A: NO AUGMENTATION

IMPACT OF UPDATED/AUGMENTED REACTOR VESSEL EXAMINATION ON PLANT IMPLEMENTATION SCHEDULE

NO.	Plant	Type	OL	Commercial	Applicable Code Edition/Addenda 1st Interval	2nd Interval	3rd Interval	4th Interval	End Present Interval	(A) Update	(A) Augmented	Aug. - Update	Aug. - Effect.	(B) 100% Exam
45	Three Mile Island 1	PWR/BW	04-19-74	09-02-74	74/S75				Sep-90	Sep-2010	Sep-2000	-10.0	9.3	
46	Arkansas 2	PWR/CE	09-01-78	03-26-80	74/S75	86E			Mar-2000	Mar-2010	Mar-2000	-10.0	8.8	
47	Calvert Cliffs 1	PWR/CE	07-31-74	05-08-75	74/S75	83/S83			Apr-97	Apr-2007	Apr-97	-10.0	5.8	
48	Calvert Cliffs 2	PWR/CE	11-30-76	04-01-77	74/S75	83/S83			Apr-97	Apr-2007	Apr-97	-10.0	5.8	
49	Fort Calhoun 1	PWR/CE	08-09-73	06-20-74	74/S75	80/W80			Sep-93	Sep-2003	Jan-97	-6.7	5.6	
50	Maine Yankee	PWR/CE	06-29-73	12-28-72		80/W80			Dec-92	Dec-2002	Apr-96	-6.7	4.8	
51	Millstone 2	PWR/CE	09-30-75	12-26-75	74/S75	80/W81			Dec-95	Dec-2005	Dec-95	-10.0	4.5	
52	Millstone 3	PWR/CE	01-31-86	04-23-86	83/S83				Apr-96	Apr-2006	Apr-96	-10.0	4.8	*
53	Palisades	PWR/CE	10-16-72	12-31-71	77/S78	83/S83			May-95	May-2005	May-95	-10.0	3.9	*
54	Palo Verde 1	PWR/CE	06-01-85	01-28-86	80/W81				Jan-96	Jan-2006	N/A	N/A	N/A	*
55	Palo Verde 2	PWR/CE	04-24-86	09-19-86	80/W81				Sep-96	Sep-2006	N/A	N/A	N/A	*
56	Palo Verde 3	PWR/CE	11-25-87	01-08-88	80/W81				Jan-98	Jan-2008	N/A	N/A	N/A	*
57	San Onofre 2	PWR/CE	09-07-82	08-08-83	77/S79				Aug-93	Aug-2003	N/A	N/A	N/A	*
58	San Onofre 3	PWR/CE	09-16-83	04-01-84	77/S79				Apr-94	Apr-2004	N/A	N/A	N/A	*
59	St. Lucie 1	PWR/CE	03-01-76	12-21-76	74/S75	83/S83			Feb-98	Feb-2008	Feb-98	-10.0	6.7	
60	St. Lucie 2	PWR/CE	06-10-83	08-08-83	80/W80				Mar-98	Mar-2008	N/A	N/A	N/A	*
61	Waterford 3	PWR/CE	03-16-85	09-24-85	80/W81				Sep-95	Sep-2005	N/A	N/A	N/A	*
62	Beaver Valley 1	PWR/W	07-02-76	10-01-76	74/S75	83/S83			Sep-97	Sep-2007	Sep-97	-10.0	6.3	
63	Beaver Valley 2	PWR/W	08-14-87	11-17-87	83/S83				Nov-97	Nov-2007	N/A	N/A	N/A	*
64	Braidwood 1	PWR/W	07-02-87	07-29-88	83/S83				Jul-98	Jul-2008	N/A	N/A	N/A	*
65	Braidwood 2	PWR/W	05-20-88	10-17-88	83/S83				Oct-98	Oct-2008	N/A	N/A	N/A	*
66	Byron 1	PWR/W	02-14-85	09-16-85	80/W81				Sep-95	Sep-2005	N/A	N/A	N/A	*
67	Byron 2	PWR/W	01-30-87	08-21-87	80/W81				Aug-97	Aug-2007	N/A	N/A	N/A	*
68	Callaway	PWR/W	10-18-84	12-19-84	80/W81				Dec-94	Dec-2004	N/A	N/A	N/A	*
69	Catawba 1	PWR/W	01-17-85	06-29-85	80/W81				Jun-95	Jun-2005	N/A	N/A	N/A	*
70	Catawba 2	PWR/W	05-15-86	08-19-86	80/W81				Aug-96	Aug-2006	N/A	N/A	N/A	*
71	Comanche Peak 1	PWR/W												
72	Cook 1	PWR/W	10-25-74	08-28-75	74/S75	83/S83			Jun-96	Jun-2006	Jun-96	-10.0	5.0	
73	Cook 2	PWR/W	12-23-77	07-01-78	74/S75	83/S83			Jun-96	Jun-2006	Jun-96	-10.0	5.0	
74	Diablo Canyon 1	PWR/W	11-02-84	05-07-85	77/S78				May-95	May-2005	May-95	-10.0	3.9	*
75	Diablo Canyon 2	PWR/W	08-26-85	03-13-86	77/S78				Mar-96	Mar-2006	Mar-96	-10.0	4.8	*
76	Farley 1	PWR/W	06-25-77	12-01-77	74/S75	83/S83			Dec-97	Dec-2007	Dec-97	-10.0	6.5	
77	Farley 2	PWR/W	03-31-81	07-30-81	74/S75	83/S83			Jul-91	Jul-2011	Nov-94	-16.7	3.4	
78	Ginna	PWR/W	12-10-84	07-01-70		74/S75	86E		Jan-2000	Jan-2010	Jan-2000	-10.0	8.6	
79	Haddam Neck	PWR/W	12-27-74	01-01-68	71,74/S75	80/W80	83/S83		Jan-98	Jan-2008	Jan-98	-10.0	6.6	
80	Harris 1	PWR/W	01-12-87	05-02-87	83/S83				May-97	May-2007	N/A	N/A	N/A	*
81	Indian Point 2	PWR/W	09-28-73	08-01-74	74/S75	80/W81			Jun-94	Jun-2004	Oct-97	-6.7	6.3	
82	Indian Point 3	PWR/W	04-05-76	08-30-76	74/S75	83/S83			Aug-96	Aug-2006	Aug-96	-10.0	5.2	
83	Kewaunee	PWR/W	12-21-73	06-16-74	74/S75	80/W81			Jun-94	Jun-2004	Oct-97	-6.7	6.3	
84	McGuire 1	PWR/W	07-08-81	12-01-81	80/W80				Dec-91	Dec-2011	N/A	N/A	N/A	*
85	McGuire 2	PWR/W	05-27-83	03-01-84	80/W80				Mar-94	Mar-2004	N/A	N/A	N/A	*
86	North Anna 1	PWR/W	04-01-78	06-06-78	74/S75	83/S83			Jun-98	Jun-2008	Jun-98	-10.0	7.0	
87	North Anna 2	PWR/W	08-21-80	12-14-80	74/S75				Dec-90	Dec-2010	Dec-2000	-10.0	9.5	
88	Point Beach 1	PWR/W	10-05-70	12-21-70	74/S75	77/S79			Dec-90	Dec-2010	Dec-2000	-10.0	9.5	

(A) BASED ON RULE EFFECTIVE 6-1-91 (B) *: 100% RV SHELL EXAM SPECIFIED IN SXI EDITION/ADDENDA (C) N/A: NO AUGMENTATION

IMPACT OF UPDATED/AUGMENTED REACTOR VESSEL EXAMINATION ON PLANT IMPLEMENTATION SCHEDULE

NDX	Plant	Type	OL	Commercial	----Applicable Code Edition/Addenda---- 1st Interval	2nd Interval	3rd Interval	4th Interval	End Present Interval	(A) Update	(A) Augmented	Aug. - Update	Aug. - Effect.	(B) 1002 Exam
89	Point Beach 2	PWR/W	03-08-73	10-01-72	74/S75	77/S79			Sep-92	Sep-2002	Jan-96	-6.7	6.6	
90	Prairie Island 1	PWR/W	04-05-74	12-16-73	74/S75	80/W81			Dec-93	Dec-2003	Apr-97	-6.7	5.8	
91	Prairie Island 2	PWR/W	10-29-74	12-21-74	74/S75	80/W81			Dec-94	Dec-2004	Dec-94	-10.0	3.5	
92	Robinson 2	PWR/W	09-23-70	03-07-71	71,74/S75	77/S78			Mar-91	Mar-2011	Mar-2001	-10.0	9.8	
93	Salem 1	PWR/W	12-01-76	06-30-77	74/S75	83/S83			Dec-97	Dec-2007	Dec-97	-10.0	6.5	
94	Salem 2	PWR/W	05-20-81	10-13-81	74/S75				Oct-91	Oct-2011	Feb-95	-16.7	3.7	
95	San Onofre 1	PWR/W	03-27-67	01-01-68	71,74/S75	74/S75			Jan-91	Jan-2011	Jan-2001	-10.0	9.6	
96	Seabrook 1	PWR/W	05-26-89		83/S83						N/A	N/A	N/A	*
97	Sequoyah 1	PWR/W	09-17-80	07-01-81	77/S78				Jul-91	Jul-2011	N/A	N/A	N/A	*
98	Sequoyah 2	PWR/W	09-15-81	06-01-82	77/S78				Feb-95	Feb-2005	N/A	N/A	N/A	*
99	South Texas 1	PWR/W	03-22-88	08-25-88	83/S83				Aug-98	Aug-2008	N/A	N/A	N/A	*
100	South Texas 2	PWR/W	03-28-89	06-19-89	83/S83				Jun-99	Jun-2009	N/A	N/A	N/A	*
101	Summer	PWR/W	11-12-82	01-01-84	77/S78				Nov-92	Nov-2002	N/A	N/A	N/A	*
102	Surry 1	PWR/W	05-25-72	12-22-72	74/S75	80/W80			Dec-92	Dec-2002	Apr-96	-6.7	4.8	
103	Surry 2	PWR/W	01-29-73	05-01-73	74/S75	80/W80			May-93	May-2003	Sep-96	-6.7	5.3	
104	Trojan	PWR/W	11-21-75	05-20-76	74/S75	83/S83			May-96	May-2006	May-96	-10.0	4.9	
105	Turkey Point 3	PWR/W	07-19-72	12-14-72	74/S75	80/W81			Feb-94	Feb-2004	Jun-97	-6.7	6.0	
106	Turkey Point 4	PWR/W	04-10-73	09-07-73	74/S75	80/W81			Apr-94	Apr-2004	Aug-97	-6.7	6.2	
107	Vogtle 1	PWR/W	03-16-87	06-01-87	83/S83				Jun-97	Jun-2007	N/A	N/A	N/A	*
108	Vogtle 2	PWR/W	03-31-89	05-20-89	83/S83						N/A	N/A	N/A	*
109	Watts Bar 1	PWR/W												
110	Watts Bar 2	PWR/W												
111	Wolf Creek 1	PWR/W	06-04-85	09-03-85	80/W81				Sep-95	Sep-2005	N/A	N/A	N/A	*
112	Yankee Rowe	PWR/W	12-24-63	07-01-61			77/S78		Jul-91	Jul-2011	Nov-94	-16.7	3.4	
113	Zion 1	PWR/W	06-25-76	12-31-73		80/W81			Dec-93	Dec-2003	Apr-97	-6.7	5.8	
114	Zion 2	PWR/W	11-14-73	09-17-74		80/W81			Sep-94	Jul-2004	Jan-98	-6.5	6.6	

(A) BASED ON RULE EFFECTIVE 6-1-91 (B) *: 100% RV SHELL EXAM SPECIFIED IN SXI EDITION/ADDENDA (C) N/A: NO AUGMENTATION

The augmented examination has no effect (indicated N/A in Table A-3) on those plants that would be implementing a 100% examination of the reactor vessel shell welds in the inspection interval in effect when the rule is estimated to become effective.

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

The proposed augmented examination of reactor vessel shell welds would apply to all operating nuclear power plants. 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 general catgories of implementation and thereby timing of associated activities. These categories are:

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

- Licensee has not previously examined essentially 100% of the length of all reactor vessel shell welds in the inservice inspection interval in effect when the proposed rule becomes effective. is not scheduled to do so, and has fewer than 40 months remaining in that interval ---- Such licensees could perform the augmented examination during the ongoing inspection interval, but would have the option of deferring 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. Licensees

that elect to defer the augmented examination to the first period of the next inspection interval may use that examination as a substitute for the reactor vessel shell weld examination scheduled for that interval, but must then perform reactor vessel shell weld examinations for successive inspection intervals during the first period of those inspection intervals

- 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.

Table A-5, below, summarizes the number of plants in the three general categories of implementation (i.e., 40 or more months remaining, fewer than 40 months remaining, and proposed augmented examination has no effect on implementation of 100% reactor vessel shell weld examination). Of the 107 plants included in this analysis, it is estimated that 39 plants would have 40 or more months remaining in the inspection interval when the rule becomes effective and would be required to perform the 100% examination in that interval; 27 plants would have fewer than 40 months remaining in the interval and could defer the proposed augmented examination to the next inspection interval; and 41 plants would be performing the 100% examination of the reactor vessel shell welds as a requirement of the effective edition and addenda of their present inspection interval). Table A-5 summarizes this information along with the distribution of reactor types for each implementation category. The effect of the proposed augmented examination is to expedite implementation of the 100% reactor vessel shell weld examination for 62% of the operating nuclear power plants [i.e., 63% for PWRs and 59% for BWRs].

Table A-5

Summary of Table A-3 Estimate of Plants in Implementation Categories

Implementation Category	--No. of Plants Affected-- Total	PWRs	BWRs
≥ 40 months	39	28	11
< 40 months	27	18	9
N/A	41	27	14
	---	--	--
Total	107	73	34

As appropriate, each affected 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 IWB-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.

Certain licensees have previously requested, and received, relief from Section XI requirements for examining reactor vessel shell welds. Although new equipment is being developed to work within the limited accessibility, it is reasonable to assume that some of the same licensees will request an exemption from at least a portion of the proposed augmented reactor vessel examination, in accordance with § 50.12, "Specific exemptions." See Section 7, below, for further discussion on this item. Potential difficulties associated with implementing the proposed augmented examination for reactor vessels are addressed in Section 8, below.

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

The proposed augmented reactor vessel examination will reduce the risk to the public, but the change in risk is not quantifiable. The change is not quantifiable since the extent of existing reactor vessel degradation is not

known because of limited previous examinations on operating reactor vessels. Implementation of the proposed augmented examination for reactor vessels is necessary to ensure 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 16.7 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. Potential impact on radiological exposure of facility employees.

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 actual 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 from the outside diameter, 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)[9] 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% (3559) 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 incremental 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>

The incremental cost associated with implementation of the proposed augmented examination for a specific plant is dependent on the reactor type, the reactor vessel examinations previously scheduled for the inspection interval when the proposed rule becomes effective, and the time remaining in that inspection

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[9] NUREG-0713, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities, 1986" Vol. 8, August 1989.

interval. Table A-6, below, summarizes the cost to examine two beltline welds (designated Partial Exam) and the cost to examine essentially 100% of the reactor vessel shell welds (dsignated Full Augmented). Table A-6 assumes that BWR 2, 3, 4 plants have no access from the outside diameter in the beltline region, and that BWR 5, 6 plants do. The range of days under Indirect Cost for BWR 5,6 is representative of the time penalty associated with performing examinations from the outside diameter for the purpose of keeping examination equipment out of the reactor vessel. The days listed under Indirect Cost represent the time required for both onsite setup and examination time.

Table A-6

Cost to Implement
Beltline and Proposed Augmented Reactor Vessel Examination

Plant Type	Extent of Examination (Effective Interval)	Cost: Direct	Cost: Indirect
PWR	Partial Exam (beltline)	$400K	1 day
	Full Augmented	$550K	3-4 days
BWR 2,3,4	Partial Exam (beltline)	$600K	10 days
	Full Augmented	$2M	15 days
BWR 5,6	Partial Exam (beltline)	$1M	10 days
	Full Augmented	$2M	15-28 days

The incremental cost to implement the proposed augmented examination is the cost in excess of performing the normally scheduled reactor vessel shell weld examinations. Because of the three implementation categories (see Table A-5, above), there are a number of possibilities for incremental cost. The incremental cost associated with each implementation category is discussed below and summarized in Table A-7.

A. Licensee required to perform proposed augmented examination in present interval (i.e., $\geq$ 40 months remaining in interval) and is normally scheduled to examine two beltline welds in that interval. In this case, the incremental cost to the licensee would be the difference between examining 100% of the reactor vessel shell welds

and examining the two beltline welds. Thirty seven plants (i.e., 28 PWRs and 9 BWRs) are affected this way. Based upon the costing information provided in Table A-6, above, and assuming the cost of downtime at $400K/day, the incremental cost associated with implementing the proposed augmented examination would be $1350K for an individual PWR; $3400K for an individual BWR 2/3/4; and $5800K for an individual BWR 5/6.

B. Licensee has fewer than 40 months remaining in the inspection interval when the proposed rule becomes effective. There are actually three possibilities for overall implementation in this category:

   1) Licensee chooses to implement proposed augmented examination during inspection interval when proposed rule becomes effective. This choice would result in the same incremental costs as for Item A, above).

   2) Licensee chooses to defer proposed augmented examination and has 12 or more 12 months remaining in present interval. By having 12 or more 12 months remaining in the present interval, the licensee would be required, in accordance with § 50.55a(g)(4), to implement the 100 % reactor vessel shell weld examination specified in the 1989 Edition during the next inspection interval, but could use the deferred examination as a substitute for that examination. This would result in a one-for-one replacement of the normally scheduled 100% examination with that of the proposed augmented examination. In this instance, the incremental cost to implement the proposed augmented examination would be zero.

   3) Licensee chooses to defer proposed examination and has fewer than 12 months remaining in present interval. By having fewer than 12 months remaining in the present interval, the licensee would be required, in accordance with § 50.55a(g)(4), to implement the 1986

Edition (which requires the examination of only two beltline welds during the 2nd, 3rd and 4th inspection intervals) during the next inspection interval. In this case, the proposed augmented examination would be used as a substitute for the normally scheduled two beltline shell weld examinations. In this instance, the incremental cost associated with implementing the proposed augmented examination would be the same as Item A, above, for an individual plant (i.e., $1350K for a PWR, $3400K for a BWR 2/3/4, and $5800K for a BWR 5/6).

C. Licensee is scheduled to exam 100% of the reactor vessel shell welds during the interval in effect when the proposed rule becomes effective. This would occur if the plant were in the first interval and were implementing the 1974 Edition (with addenda through the 1975 Winter Addenda) through the 1986 Edition (with addenda through the 1987 Addenda), which require volumetric examination of a reactor vessel shell welds in the first interval. In this case, the licensee would automatically satisfy the proposed augmented reactor vessel examination by implementing the normally scheduled reactor vessel examination. There would be no incremental cost associated with implementing the proposed augmented reactor vessel examination.

Table A-7

Incremental Cost Calculations for Options to Implement Proposed Augmented Reactor Vessel Examination

Implementation Category	Costs (K$) Augmented Examination Direct	Indirect	Costs (K$) Normally Scheduled Direct	Indirect	Cost (K$) Individual Incremental
A. ≥ 40 months			(2 beltline welds)		
o PWR	550	1600 (400K x 4 days)	400	400 (400K x 1 day)	1350
o BWR 2/3/4	2000	6000 (400K x 15 days)	600	4000 (400K x 10 days)	3400
o BWR 5/6	2000	8800 (400K x 22 days)	1000	4000 (400K x 10 days)	5800
B. < 40 months					
1. > 12 months (deferred)			(100% RV shell welds)		
o PWR	550	1600	550	1600 (400K x 4 days)	0
o BWR 2/3/4	2000	6000	2000	6000 (400K x 15 days)	0
o BWR 5/6	2000	8800	1000	8800 (400K x 22 days)	0
2. < 12 months (deferred)			(2 beltline welds)		
o PWR	550	1600	400	400 (400K x 1 day)	1350
o BWR 2/3/4	2000	6000	600	4000 (400K x 10 days)	3400
o BWR 5/6	2000	8800	1000	5000 (500K x 10 days)	5800
C. 100 % RV Shell Exam Required in Effective Interval					
o PWR	550	1600	550	1600	0
o BWR 2/3/4	2000	6000	2000	6000	0
o BWR 5/6	2000	8800	2000	8800	0

The industry wide incremental costs associated with the above options are summarized in Table A-8. The assumption is made in this table that a plant with fewer than 40 months remaining in the inspection interval in effect when the rule becomes effective would defer the proposed augmented examination to the first period of the next inspection interval as permitted by the proposed rule.

Table A-8

Summary of Incremental Cost to Implement Proposed Augmented Reactor Vessel Examination

Plant Type	Implementation Category	Number of Plants	Individual Plant Incremental Cost	Industry Wide Incremental Cost
PWR	≥ 40 months	28	$1.35M	$37.80M
	< 40 months	(18)		
	- > 12 months	15	0	0
	- < 12 months	3	$1.35M	$ 4.05M
	First Interval	27	0	0
BWR 2/3/4	≥ 40 months	11	$3.40M	$37.40M
	< 40 months	(9)		
	- > 12 months	6	0	0
	- < 12 months	3	$3.40M	$10.20M
	First Interval	6	0	0
BWR 5/6	≥ 40 months	0	$5.80M	0
	< 40 months	(0)		
	- > 12 months	0	0	0
	- < 12 months	0	$5.80M	0
	First Interval	8	0	0

Total Incremental Cost for the Industry: $89.45M
Averaged Over All Operating Plants: $0.84M/plant
Averaged Over Affected Plants: $1.99M/plant

The total incremental cost (in 1990 dollars) for the industry to implement the proposed augmented reactor vessel shell weld examination would be $89.45M. Recognizing that the augmented examinations would be implemented over a span of time from about 3 years to 10 years after the proposed rule becomes effective, an estimate of the discounted value was calculated assuming a discount rate of 5%, a distributed implementation based on the schedules in

Table A-3, and a discounting of the incremental cost effective when the augmented examination is performed. Based upon these assumptions, the discounted cost of the proposed augmented examination is estimated to be $43.15M.

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

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

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, 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. Some of the same plants that, in the past, requested relief from reactor vessel shell weld examinations will likely request relief from at least a portion of the proposed augmented examination for the same reasons. 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 augmented examination 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, some BWRs likely 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 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 been granted such relief. For these plants, the reactor vessels would not be examined for about 30 years (i.e., based upon updating) 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 BWR 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 when the proposed rule becomes effective, which would follow resolution of any comments received during the 75-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)~~ must meet the test requirements 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~~ must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related ~~pressure vessels, 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, pumps and valves ~~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 ~~(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 Sections XI of the ASME Boiler and Pressure Vessel Code and Addenda[6] in effect 6 months prior to the date of issuance of the construction permit. The pumps and valves ~~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 perservice 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 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 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 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 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 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 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-test,~~ 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,~~ ~~I~~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 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~~ test program for a boiling or pressurized water-cooled nuclear power facility shall 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 6 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 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. (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 requirement 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 vessels, 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 ~~(i)~~ inservice examination of such components (including supports) and ~~(ii) tests for operational readiness of pumps and valves,~~ 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 6 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 secion, 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 Calss 1 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[6] applied to the construction of the particular component.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 must be designed and be provided with access to enable the performance of inservice examination of such components and must meet the preservice examinaiton 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 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 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 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, ~~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 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 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 6 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 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.

Appendix C

Environmental Assessment
and
Finding of No Significant impact

Environmental Assessment

Identification of Proposed Action

The proposed rule would update existing references in the NRC regulations to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Specifically, the proposed rule would incorporate by reference the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition for Division 1 rules of Section III, "Rules for the Construction of Nuclear Power Plant Components," and Division 1 rules of Section XI, "Rules for the Inservice Inspection of Nuclear Power Plant Components" of the ASME Code. In addition, the proposed rule would impose on all licensees an augmented reactor vessel examination that would require expedited examination of essentially 100% of the reactor vessel shell welds.

The Need for the Proposed Action

The NRC regulations require that licensees preparing for their first inservice inspection interval 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, and that licensees update their inservice examination and inservice testing programs every 120 months 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. Further, the NRC regulations provide that the edition and addenda of Section III, Division 1, of the the ASME Code used in the design specification for a component be no earlier than three years prior to the date that the nuclear power plant construction permit is docketed. Therefore, because the ASME Code is constantly being improved to take into account new technologies and plant

operating experiences, it is in the interest of the public health and safety for NRC to adopt later versions of the ASME Code after the staff has performed a thorough review and made a determination of the need to modify or supplement the ASME Code rules.

Adoption of the proposed amendment would impose on applicants and licensees the use of improved methods for construction, inservice inspection, and inservice testing of nuclear power plant components. The Commission is imposing a modification on one Section XI revision to ensure that existing safety margins on operational readiness are retained for certain containment isolation valves.

The augmented reactor vessel examination would ensure that all reactor vessel shell welds are examined in an expedited manner. The 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.

Environmental Impacts of the Proposed Action

The Commission has completed its evaluation of the proposed rule and has concluded that the proposed rule would improve the structural reliability of Class 1, Class 2, and Class 3 components and their supports, and the operational readiness of Class 1, Class 2, and Class 3 pumps and valves. It is expected that the proposed rule would significantly reduce the probability or consequences of an accident. The Commission also concluded that the proposed rule would not affect the amounts and types of any effluents that may be released offsite and that there should be no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the Commission concludes that this proposed rule would result in no significant radiological environmental impact.

The proposed rule does not affect non-radiological plant effluents and has no other environmental impact. Therefore the Commission concludes that there are

no significant non-radiological environmental impacts associated with the proposed amendment.

Alternative to the Proposed Action

The principal alternatives to the technical actions in the proposed rule would be to not update the existing references in the NRC regulations to Section III and Section XI of the ASME Code and to not impose the proposed sugmented reactor vessel examination. Since the Commission has already concluded that no significant environmental effect would result from the proposed rule, the specified alternatives, which would have no safety benefit, would not reduce the environmental impact of plant construction, and inservice examination and inservice testing.

Agencies and Persons Consulted

The NRC staff prepared the proposed rule in consultation with personnel from the Idaho National Engineering Laboratory (Idaho Falls, ID), the Oak Ridge National Laboratory (Oak Ridge, TN), and the consulting firm of Reedy Associates (Los Gatos, CA).

Finding of No Significant Impact

The Commission has determined not to prepare an environmental impact statement for the proposed rulemaking.

Based upon the foregoing environmental assessment, the Commission concludes that the proposed action will not have a significant effect on the quality of the human environment. The environmental assessment and finding of no significant impact on which this determination is based are available for inspection at the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, DC. Single copies of the environmental assessment and the finding of no significant impact are available from Gilbert C. Millman, Division of Engineering, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-3848.

Appendix D

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

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 records developed are generally not collected by the NRC, 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 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. <u>Reduction of Burden Through Information Technology</u>
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. <u>Effort to Identify Duplication</u>
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. <u>Effort to Use Similar Information</u>
The NRC is using the information reporting requirements specified in the ASME Code in lieu of developing its own equivalent requirements.

6. <u>Effort to Reduce Small Business Burden</u>
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. <u>Consequences of Less Frequent Collection</u>
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, examination, and testing of components to provide a basis for evaluating degradation of these components and systems at any time during their service lifetime.

9. Consultations Outside the NRC

The NRC staff prepared the proposed rule in consultation with personnel from the Idaho National Engineering Laboratory (Idaho Falls, ID), the Oak Ridge National Laboratory (Oak Ridge, TN), and the consulting firm of Reedy Associates (Los Gatos, CA).

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 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 8 nuclear power plants with construction permits and to the owners of the 111 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 119 nuclear power plants presently under construction or in operation.

b. Estimated Hours

Section 50.55a specifies that the Code edition and addenda to 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 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 designee) and Certificate Holder. The earliest Section III addenda being addressed in

the proposed rule is the 1986 Addenda. Since the last plant was 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 Section III addenda.

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 revisions contained in the Section XI, Division 1, addenda and edition specified in the proposed rule. The number of plants that could implement 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 119 (i.e., the 111 plants with operating licenses and the 8 plants with construction permits). The revisions in the Section XI edition and addenda affected by the proposed rulemaking that significantly affect recordkeeping requirements are addressed below.

- 1986 Addenda

  IWB-3700: IWB-3700, "Analytical Evaluation of Plant Operating Events," requires a documented engineering evaluation when an operating event causes an excursion outside the normal operating pressure and temperature limits defined in the plant Technical Specifications. It is estimated that a plant implementing 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. It is estimated that, in a given year, 5 percent of the total number of operating plants would be required to prepare the specified engineering evaluation and report. Therefore, the additional burden per year resulting from this revision is estimated to be 1190 p-hrs (i.e., 200 p-hrs/plant x [.05 x 119] plants/year).

IWF-4000: This revision adds rules for the repair of Class 1, Class 2, Class 3, and Class MC component supports. These rules require documentation of repairs in accordance with IWA-6000, "Records and Reports." IWA-6000 specifies that the Owner is required to document the repairs in the inservice inspection summary reports on existing Form NIS-2, "Owner's Report for Repair or Replacements." Information to be included on Form NIS-2 includes identification of the component (i.e., name of component, name of manufacturer, manufacturer Serial No., National Board No., year built, whether ASME Code stamped) and system, the applicable construction code and Section XI edition and addenda, repair organization, and a description of the work performed.

Form NIS-2 expedites documentation of the required information. For the purpose of this burden calculation, it has been estimated that, on average, 20 component supports would be repaired in accordance with Section XI rules each year by each plant. It is estimated that it would take 2 hours to document the repair of an individual component support on Form NIS-2. Therefore, the additional recordkeeping burden associated with this revision is estimated to be 4760 p-hrs (i.e., 2 p-hrs/repair x 20 repairs/year/plant x 119 plants/year).

- <u>1987 Addenda</u>

  <u>IWF-5000</u>: This revision incorporates the rules of ASME/ANSI OM-1987, Part 4, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)" in place of the existing IWF rules for preservice and inservice examinations, and repairs and replacement of snubbers. Existing IWA-1400, "Owner's Responsibility" specifies that, among other things, the Owner is responsible for: preparing plans and schedules for preservice and inservice examinations and tests; recording examination and test results that provide a basis for evaluation and facilitate comparison with the results of subsequent examinations; maintenance of adequate inspection exmination, test, and repair and replacement records; retention of all inspection, examination, test, and replacement records for the service lifetime of the component or system. OM Part 4, Section 1.2, "Responsibility" and Section 4, "Records and Record Keeping" specify requirements for written procedures and records necessary to verify the result of the preservice and inservice inspection programs. This recordkeeping would result in essentially the same type of documentation that is presently required. Because most plants are already implementing major portions of the revision as part of the Standard Technical Specifications, it is not expected that the new requirements would, in general, significantly affect the present recordkeeping burden.

  However, one technical change is significant from the standpoint of affecting recordkeeping. That change is the deletion of the 50 Kip limit on snubbers to be tested. This will result in the need for some of the older plants (i.e., about 10% of all plants) to implement, and document, the testing of the larger

hydraulic snubbers. It is estimated that the burden associated with preparing written procedures for the implementation of tests on these larger snubbers, and the documentation and maintenance of results would be 100 p-hrs/affected plant. This would result in a total additional burden of 1190 p-hrs/year for all affected plants (i.e., 100 p-hrs/plant/year x [0.10 x 119] plants/year).

- 1988 Addenda

  Table IWB-2500-1: A revision to Table IWB-2500-1 increases the extent of reactor vessel shell weld examinations in the second and successive 10-year inspection intervals. Although the data from these examinations is generally automatically recorded and processed, it is estimated that about 200 p-hrs would be required to assemble, review, and summarize the additional data that would be collected once during each 10-year inspection interval. On average, about 10 percent of all operating plants perform the reactor vessel shell weld examinations each year. Therfore, the additional recordkeeping burden per year is estimated to be 2380 p-hrs (i.e., 200 p-hrs/plant x [.10 x 119] plants/year).

  Subsections IWP and IWV: This revision deletes specific rules contained in these subsections for inservice testing of pumps and valves and instead references rules contained in ASME/ANSI OMa-1988 Addenda to OM-1987 Part 6 (Inservice Testing of Pumps) and Part 10 (Inservice Testing of Valves). These OM standards provide specific rules for the maintainence of records associated with: the construction of pumps and valves; inservice test plans; record of tests; and records of corrective actions. Since these are essentially the same types of records presently being required by IWA-1400, "Owner's

Responsibility" (see comments on revision to IWF-5000, above), there should be no significant change in the basic recordkeeping requirements. However, the change in technical requirements associated with this revision would result in a reduction in the number of relief request submittals because the new rules are consistant with the positions contained in a number of commonly granted relief requests.

It is estimated that the revisions to Subsection IWP and Subsection IWV to reference the OM Part 6 and Part 10 standards would save 100 p-hours/plant/10-year inspection intverval, because of the reduced need to prepare and process relief requests. On average about 10 percent of the operating plants each year move from one interval into the next interval, and usually relief requests are updated at this time. Therefore, the expected reduction in burden/year is estimated to be 1190 p-hrs (i.e., 100 p-hrs/plant x [.10 x 119] plants/year).

<u>Appendix VII</u>: This new mandatory appendix specifies requirements for the training and qualification of ultrasonic nondestructive examination (NDE) personnel in preparation for Employer certification to perform NDE. Appendix VII specifies the requirements for qualification records. These records include those for precertification (e.g., name of individual, qualification level, educational background and experience, statement indicating satisfactory completion of prior training, record of annual supplemental training, results of vision examinations, current qualification examination results) and certification (e.g., in addition to those for precertification, includes date of current certification and expiration date, name and signature of certifying Employer representative, evidence of continued proficiency in the case of interrupted service).

It is estimated that it would take 65 p-hrs per plant per year to prepare and maintain the specified additional training records. Since Appendix VII will eventually apply to all operating plants, the additional recordkeeping burden is estimated to be 7735 p-hrs (i.e., 65 hrs/plant/year x 119 plants/year).

- <u>1989 Edition</u>
  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.

In addition to the revisions identified above, the proposed rule would impose an augmented examination of the reactor vessel shell welds. The augmented examination serves the purpose of expediting the reactor vessel shell weld examinations addressed above in the revision of Table IWB-2500-1. The augmented examination would result in all plants implementing the examination within approximately 80 months of the effective date of the rule, which is, on average, earlier than required by the present regulations. However, because the above burden calculation for Table IWB-2500-1 revision assumes immediate implementation of the reactor vessel examination, that calculated burden of 2380 p-hrs includes the recordkeeping burden associated with the proposed augmented reactor vessel examination.

<u>Total Recordkeeping Burden</u>
As noted above, there is no requirement for existing licensees to implement the 1986 Addenda, 1987 Addenda, 1988 Addenda, and 1989 Edition of Section III. There is, however, 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

discussed above and summarized below.

Recordkeeping Burden

Section XI Reference	No. of Plants/ Year	Annual Rcrdkping Hrs/Plant	Total Annual Hours[1]	Retention Period
IWB-3700	6	200	1200	Lifetime
IWF-4000	119	40	4760	Lifetime
IWF-5000	12	100	1200	Lifetime
Table IWB-2500	12	200	2400	Lifetime
Subsections IWP/IWV	12	-100	-1200	Lifetime
Appendix VII	119	65	7735	3 years[2]

[1]: Figures have been rounded for integer number of plants.
[2]: After superseded or invalidated.

The net annual increase in the recordkeeping burden is 16,095 p-hrs. This averages to approximately 135 p-hrs/plant/year.

c. Estimated Cost Required to Respond to the Collection

Based upon the hours specified in Item A.13.b, above, and a rate of \$92/hr., it is estimated that the cost to the industry for responding to the information collection required by the proposed amendment to § 50.55a is a total of \$1,481K/year (16 095 hours x \$92/hour).

The table below shows the individual costs associated with each revision that affects the burden.

Section XI Revision	P-HRS/YR	K\$/YR
IWB-3700	1200	110.4
IWF-4000	4760	437.9
IWF-5000	1200	110.4
Subsections IWP and IWV	-1200	-110.4
Mandatory Appendix VII	7735	711.6
Table IWB-2500	2400	220.8
Totals	16,095	1,480.7

d. Record Retention Period

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

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

Lifetime retention of the above records is necessary to ensure adequate historical information on the design, examination, and testing of components and systems to provide a basis for evaluating degradation of these components and systems at any time during their service lifetime.

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.