PPR

Standard Form 83
(Rev. September 1983)

Request for OMB Review

DESIGNATED ORIGINAL
Certified By P. Smith

Important

Read instructions before completing form. Do not use the same SF 83 to request both an Executive Order 12291 review and approval under the Paperwork Reduction Act.

Answer all questions in Part I. If this request is for review under E.O. 12291, complete Part II and sign the regulatory certification. If this request is for approval under the Paperwork Reduction Act and 5 CFR 1320, skip Part II, complete Part III and sign the paperwork certification.

Send three copies of this form, the material to be reviewed, and for paperwork—three copies of the supporting statement, to:

Office of Information and Regulatory Affairs
Office of Management and Budget
Attention: Docket Library, Room 3201
Washington, DC 20503

PART I.—Complete This Part for All Requests.

1. Department/agency and Bureau/office originating request

U.S. Nuclear Regulatory Commission

2. Agency code

3 1 5 0

3. Name of person who can best answer questions regarding this request

Gilbert Millman

Telephone number

(301) 492-3848

4. Title of information collection or rulemaking

10 CFR 50, Domestic Licensing of Production and Utilization Facilities

5. Legal authority for information collection or rule (cite United States Code, Public Law, or Executive Order)

42 USC 2201(o) or

6. Affected public (check all that apply)

1 ☐ Individuals or households
2 ☐ State or local governments
3 ☐ Farms
4 ☒ Businesses or other for-profit
5 ☐ Federal agencies or employees
6 ☐ Non-profit institutions
7 ☐ Small businesses or organizations

PART II.—Complete This Part Only if the Request is for OMB Review Under Executive Order 12291

7. Regulation Identifier Number (RIN)

___ ___ ___ ___ — ___ ___ ___ ___ or, None assigned ☐

8. Type of submission (check one in each category)

Classification
1 ☐ Major
2 ☐ Nonmajor

Stage of development
1 ☐ Proposed or draft
2 ☐ Final or interim final, with prior proposal
3 ☐ Final or interim final, without prior proposal

Type of review requested
1 ☐ Standard
2 ☐ Pending
3 ☐ Emergency
4 ☐ Statutory or judicial deadline

9. CFR section affected

_______ CFR _______

10. Does this regulation contain reporting or recordkeeping requirements that require OMB approval under the Paperwork Reduction Act and 5 CFR 1320? ☐ Yes ☐ No

11. If a major rule, is there a regulatory impact analysis attached? 1 ☐ Yes 2 ☐ No

If "No," did OMB waive the analysis? 3 ☐ Yes 4 ☐ No

Certification for Regulatory Submissions

In submitting this request for OMB review, the authorized regulatory contact and the program official certify that the requirements of E.O. 12291 and any applicable policy directives have been complied with.

Signature of program official | Date

Signature of authorized regulatory contact | Date

12. *(OMB use only)*

Previous editions obsolete
NSN 7540-00-634-4034

83-108

Standard Form 83 (Rev. 9-83)
Prescribed by OMB
5 CFR 1320 and E.O. 12291

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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. Reduction of Burden Through Information Technology

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

4. Effort to Identify Duplication

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

5. Effort to Use Similar Information

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

6. Effort to Reduce Small Business Burden

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

7. Consequences of Less Frequent Collection

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

8. Circumstances Which Justify Variation from OMB Guidelines

The record retention periods for information requested is frequently for the service lifetime of the applicable component. Such lifetime retention of records is necessary to ensure adequate historical information on the design, 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.

- <u>1986 Addenda</u>

  <u>IWB-3700</u>: 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).

- 1987 Addenda

IWF-5000: This revision incorporates the rules of ASME/ANSI OM-1987, Part 4, "Examinatio J Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)" in place of the existing rules for preservice and inservice examinati d repairs and replacement of snubbers. Existing IWA- 3, "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 summa ̀e 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. <u>Record Retention Period</u>

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:

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

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. <u>Reasons for Change in Burden</u>

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. <u>Publication for Statistical Use</u>

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.

[7590-01]

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

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 or not 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 Devel ing 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 comply with 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 comply with 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., pressurizer, 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 requiremonts 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: Availability

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 135 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).

For the purposes of sec. 223, 68 Stat. 958, as amemded (42 U.S.C. 2273); §§ 50.46(a) and (b), and 50.54(c) are issued under sec. 161b, 161i and 161o, 68 Stat. 948 as amended (42 U.S.C. 2201(b); §§ 50.7(a), 50.10(a)-(c), 50.34(a) and (e), 50.44(a)-(c), 50.46(a) and (b), 50.47(b), 50.48(a), (c), (d), and (e), 50.49(a), 50.54(a), (i), (i)(1), (l)-(n), (p), (q), (t), (v), and (y), 50.55(f), 50.55a(a), (c)-(e), (g), and (h), 50.59(c), 50.60(a), 50.62(c), 50.64(b), and 50.80(a) and (b) are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.49(d), (h), and (j), 50.54(w), (z), (bb), (cc), and (dd), 50.55(e), 50.59(b), 50.61(b), 50.62(b), 50.70(a), 50.71(a)-(c) and (e), 50.72(a), 50.73(a) and (b), 50.74, 50.78, and 50.90 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

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)(2)(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:

§ <u>50.55a Codes and standards.</u>

Each operating license for a boiling or pressurized water-cooled nuclear power facility must be subject to the conditions in paragraphs (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 [illegible] 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, Category C-F). (A) Appropriate Code Class 2 pipe welds in Residual Heat Removal Systems, Emergency Core Cooling Systems, and Containment Heat Removal Systems, must be examined. When applying editions and addenda up to the 1983 Edition through the Summer 1983 Addenda of Section XI of the ASME Code, the extent of examination for these systems must be determined by the requirements of paragraph IWC-1220, Table IWC-2520 Category C-F and C-G, and paragraph IWC-2411 in the 1974 Edition and Addenda through the Summer 1975 Addenda.

*   *   *   *   *

(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) Inservice testing requirements.

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

(2) For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued on or after January 1, 1971,

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

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

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

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

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

(h) Protection systems. For construction permits issued after January 1, 1971, protection systems must meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," (IEEE-279) in effect[7] on the formal docket date[8] 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 th⸱s 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.

[8] Where an application for a construction permit is submitted in four parts pursuant to the provisions of § 2.101(a-1) and Subpart F of 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.