April 17, 1996

TO: The Commissioners

FROM: James M. Taylor /s/
Executive Director for Operations

SUBJECT: ISSUANCE OF FINAL AMENDMENT TO 10 CFR § 50.55a TO INCORPORATE BY REFERENCE THE ASME BOILER AND PRESSURE VESSEL CODE (ASME CODE), SECTION XI, DIVISION 1, SUBSECTION IWE AND SUBSECTION IWL

PURPOSE:

This final rulemaking will amend 10 CFR 50.55a to incorporate by reference the 1992 Edition with the 1992 Addenda of the American Society of Mechanical Engineers (ASME) Code with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants," with specified modifications. The staff recommends that Subsections IWE and IWL be imposed as a compliance backfit on an expedited basis. As a result of the rate of occurrence of containment degradation, and the extent of containment degradation, the NRC staff concludes that there is a reasonable likelihood that licensees will not be in conformance with appropriate regulatory requirements and current licensing basis commitments with respect to structural integrity and leak-tightness throughout the term of the operating license. Subsection IWE provides criteria for visual inspection of the surface of metal containments, the steel liners of concrete containments, pressure-retaining bolts, and seals and gaskets. Subsection IWL provides criteria for visual inspection of concrete pressure-retaining shells, shell components, and for the examination of unbonded post-tensioning systems.

Licensees will be required to incorporate Subsections IWE and IWL into their inservice inspection (ISI) program. Licensees will also be required to implement visual inspections of containments and complete the inspections in accordance with Subsections IWE and IWL within five years of the effective date of this rule. This expedited inspection schedule is necessary to prevent a possible delay by as much as 20 years in the implementation of Subsection IWE, and a possible delay by as much as 15 years in the implementation of Subsection IWL. The potential delays could result from Subsection IWE which permits deferral of some of the required inspections until the end of the 10-year inspection interval. Adding the 10 years that could pass before some utilities are required to update their ISI plans, a period of 20 years could pass before these inspections would take place. Provisions have also been included that would prevent unnecessary duplication of Subsections IWE and IWL inspections and routine ISI inspections.

Subsections IWE and IWL have not been previously incorporated by reference into the NRC regulations. This rule incorporates the inspection criteria of Subsections IWE and IWL to assure that the critical areas of containments are periodically inspected to detect and take corrective action for defects that could compromise a containment's structural integrity.

SUMMARY:

On January 7, 1994 (59 FR 979), the NRC published in the Federal Register, a proposed amendment to 10 CFR 50.55a. Comments were received from 25 separate sources. The response to public comments is contained in Attachment 6. As a result of the public comments received, there is one editorial change and one clarification to the provisions which were included in the proposed rule. As a result of other public comments, four modifications and a clarification have been added to the final rule. The four modifications to the rule are: (1) § 50.55a(b)(2)(x)(A) expands the evaluation of inaccessible areas of concrete containments to metal containments and the liners of concrete containments; (2) § 50.55a(b)(2)(x)(B) permits alternative lighting and resolution requirements for remote visual inspection of the containment; (3) § 50.55a(b)(2)(x)(C) makes the examination of pressure retaining welds and pressure retaining dissimilar metal welds optional; and (4) an alternative sampling plan has been added (§ 50.55a(b)(2)(x)(D)). The clarification (§ 50.55a(b)(2)(x)(E)) more clearly defines the frequency of the Subsection IWE general visual examination.

The first modification, § 50.55a(b)(2)(x)(A), which expands the visual inspection of inaccessible areas of concrete containments to metal containments and the liners of concrete containments, was the result of a comment received on § 50.55a(b)(2)(ix)(E) of the proposed rule. The commenter believed that given the number of occurrences of corrosion in containments, the proposed provision (which only addressed concrete containments) should be expanded in the final rule to include metal containments and the liners of concrete containments. The staff agrees and the revised final rule addresses inaccessible areas of metal containments and the metal liners of concrete containments.

The second modification, § 50.55a(b)(2)(x)(B), provides alternative lighting and resolution requirements for remote visual inspection of the containment. Subsection IWE references the lighting and resolution requirements contained in ASME Section XI Subsection IWA, "General Requirements," (specifically IWA-2200). Commenters believe that the lighting and resolution requirements contained in IWA-2200 were intended for component flaw examination and are impractical for remote containment inspection. The staff agreed and revised the final rule to permit alternative lighting and resolution requirements.

The third modification, § 50.55a(b)(2)(x)(C), makes the Subsection IWE pressure retaining welds and Subsection IWE pressure retaining dissimilar metal welds inspections optional. The NRC staff concludes that requiring these inspections is not appropriate. There is no evidence of problems associated with welds of this type in operating plants. Therefore, the occupational radiation exposure that would be incurred while performing these inspections cannot be justified. It is estimated that the total occupational exposure that would be incurred yearly in the performance of the containment weld inspections would be 440 person-rems.

The fourth modification, § 50.55a(b)(2)(x)(D), provides an alternative to the ASME Section XI requirements for "additional" examinations (note: additional examinations are required during the same outage when acceptance criteria are exceeded). The alternative would allow licensees to determine the number of additional components to be examined based on an evaluation to determine the extent and nature of the degradation. Five commenters believe that the requirements for additional examinations used in other subsections of Section XI are inappropriate for containment components. Additional examinations are incorporated into Section XI to determine the extent to which degradation found in one component exists in other similar components. In some instances, a large number

of additional examinations could be required. The commenters believe that a review of the operational history of containment components shows that degradation is limited to the area in question and is not widespread. This makes the Section XI sampling plan burdensome and inappropriate for application to containments. The staff agreed and revised the final rule to permit the alternative to the Section XI requirements for additional examinations.

A clarification, § 50.55a(b)(2)(x)(E), was added to make it clear that the required frequency for performing general visual inspections is three times in each 10-year interval. The clarification will assure consistency between the amendment to 10 CFR Part 50, Appendix J (60 FR 49495), and the provisions of ASME Section XI, Subsection IWE.

The staff believes that the above changes in response to public comment improve the final rule and will improve the containment inspection program set forth by Subsections IWE and IWL. The comments leading to the proposed modifications have been transmitted to the ASME for its consideration for future addenda of Section XI. The ASME Code is considering the issues which resulted in the four modifications, and it is anticipated that a resolution of the issues will be published in future addenda.

On September 26, 1995 (60 FR 49495), the NRC published in the Federal Register, a final amendment to 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The final revision to Appendix J does not change the current containment visual inspection frequency requirements. Regulatory Guide 1.163, "Performance-Based Containment Leakage-Test Program," which accompanies the final revision to Appendix J, specifically addresses maintaining the current frequency of three times in 10 years. The final revision to Appendix J would permit longer intervals (up to 10 years) between Type A tests. The Subsections IWE and IWL rulemaking package is consistent with the final Appendix J containment visual inspection frequency requirement of three times in 10 years. This package, the Subsections IWE and IWL rulemaking, clarifies the Appendix J visual inspection requirement by providing licensees with criteria to meet this requirement.

DISCUSSION:

The primary function of the containment is to provide an essentially leak-tight barrier against the uncontrolled release of radioactivity into the environment should an accident occur as discussed in General Design Criterion (GDC) 16, "Containment Design." The role of an inservice inspection program is to detect evidence of deterioration that may affect either the containment structural integrity or leak-tightness. GDC 53, "Provisions for Testing and Inspection," requires the examinations of the containment. GDCs 16 and 53 are satisfied by implementing Subsections IWE and IWL.

The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. There have been 66 separate occurrences of degradation in operating containments (some plants have had more than one occurrence of degradation). One-fourth of all containments have experienced corrosion, and nearly one-half of the concrete containments have reported degradation related to the concrete or the post-tensioning system. Since 1986, 32 occurrences of corrosion in steel containments or the liners of concrete containments have been reported. (At the time the proposed rule was under development, the number of reported occurrences was 21.) In two cases, thickness measurements of the walls revealed areas where the wall thickness was below the minimum design thickness. There have been four cases where extensive corrosion of the liner had reduced the liner thickness locally by nearly one-half. Since the early 1970s, 34 occurrences of containment degradation related to post-tensioning systems of concrete containments have been reported. Four of these occurrences involved grease leakage from tendons. Upon investigation, in addition to grease leakage, these incidents showed signs of leaching of the concrete. Table 3 of Attachment 2 lists many of these occurrences of degradation, and Table 3A specifically lists occurrences associated with metal containments and the liners of concrete containments.

Of the 32 occurrences of metal corrosion, only four were detected through current containment inspection programs conducted prior to Type A testing. Nine of these occurrences were first identified by the NRC through its inspections or audits of plant structures. Eleven were detected by licensees while performing an unrelated activity or, after they were alerted to a degraded condition at another site. Occurrences of degradation not found by licensees, but initially detected at plants through NRC inspections, include (1) corrosion of the steel containment shell in the drywell sand cushion region, resulting in reduced wall thickness to below the minimum design thickness; (2) corrosion of the torus of the steel containment shell (wall thickness reduced to below the minimum design thickness); (3) corrosion of the liner of a concrete containment, resulting in the liner being reduced to one-half thickness in some locations; (4) grease leakage from the tendons of prestressed concrete containments; and (5) leaching and excessive cracking of the concrete.

The NRC Regional Offices surveyed licensees in 1990 to determine the type of inspections being performed on containment structures and to determine the effectiveness of the visual inspections then being performed by licensees. Based on this survey and on the results of inspections, the staff has determined that there is a large variation with regard to the performance and the effectiveness of containment inspections. Inspections are being performed, but in general, these inspections are not being conducted in a manner which will detect many of the types of degradations which have been reported. Therefore, the staff has determined that it is necessary to endorse more detailed requirements for the periodic visual inspection of containment structures and codify these by regulation to assure that the critical areas of containments are periodically inspected to detect and take corrective action for defects that could compromise the containment's pressure-retaining and leak-tight capability.

The staff believes that this action is an important step, not unlike the maintenance rule, that would strengthen the management of aging degradation for the containment structures and be entirely consistent with the aims of the license renewal of 10 CFR Part 54. As indicated in the license renewal rule, if the requirements of Subsections IWE and IWL are effective in managing the effects of aging through the renewal term, the basis would be established for further limiting the license renewal review.

It was recognized at the proposed rulemaking stage that an argument could be made for justifying the rule under each of the backfit justifications (i.e., the compliance exception, 10 CFR 50.109(a)(4)(i), adequate protection exception, 10 CFR 50.109(a)(4)(ii), justified safety enhancement in accordance with the criteria of 10 CFR 50.109(a)(3)). As a result of CRGR recommendations and consultation with OGC on this specific point, it was determined that, on balance, the compliance exception was the appropriate backfit justification in 1994 when the rule was published for public comment.

SECY-93-328, "Issuance of Proposed Amendment to 10 CFR § 50.55a to Incorporate by Reference the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, Division 1, Subsection IWE and Subsection IWL," dated December 1, 1993, discussed the similarity between the proposed action and the Pressurized Thermal Shock issue with respect to adequate protection. In that case, known time-dependent mechanisms could have ultimately threatened the design margins of an important barrier in the overall defense-in-depth protection, if appropriate actions to reduce the threat were not taken by the NRC. At that time, however, the staff believed that the safety concern (significant degradation of containment integrity) did not rise to a level that warranted identification in the category of a most compelling safety concern (i.e., adequate protection).

Because the inspection requirements contained in Subsections IWE and IWL had not been previously endorsed by the NRC, SECY-93-328 (the proposed amendment) contained an ISSUE section to highlight the fact that endorsement of these ASME subsections would be an expansion of the scope of 10 CFR 50.55a. The proposed rule was issued for publication without comment from the Commission. Based on this, after carefully considering the public comments, and given the number of occurrences of containment degradation and the extent of the containment degradation, the NRC staff believes that there is no longer a question as to whether the NRC should endorse these subsections. Therefore, the draft final SECY Paper does not contain an ISSUE section.

The draft final rule was presented to the ACRS on February 10, 1995. The ACRS concurred with the staff position. The draft final rule was presented to the CRGR at Meeting No. 272, on April 25, 1995 (Minutes attached). Questions raised by OGC during office concurrence regarding the proposed backfit justification for the draft final rule were not resolved at that meeting. The CRGR requested further consideration by OGC of the questions involved and deferred further discussion of the backfit issue. Extensive discussions were held between OGC, the CRGR, and staff. Bases for backfit justification under the compliance exception, adequate protection exception, and justified safety enhancement were discussed. A determination was made that justification under the compliance exception continues to be appropriate.

In analyzing the occurrences of containment degradation, it is apparent that all containments are subject to a certain type (s) of degradation depending on the particular design. Information gathered by the staff indicates that many licensees still have not reacted to this serious safety concern and have not initiated comprehensive containment inservice inspections. As a result of the rate of occurrence of containment degradation, and the extent of containment degradation, the NRC concludes that there is a reasonable likelihood that licensees will not be in conformance with appropriate regulatory requirements and current licensing basis commitments with respect to structural integrity and leak-tightness throughout the term of the operating license. The NRC rejects the argument that the NRC may not impose a compliance backfit in circumstances where licensees may be in compliance today, but there is reasonable likelihood that licensees will not be in compliance in the future.

A discussion of the staff's consideration of this issue as a justified safety enhancement, including a cost benefit analysis, is provided in Attachment 7. The staff believes that the final action would also result in a substantial safety increase and that the direct and indirect costs of implementation are justified in view of the significant safety benefit to be gained. The staff believes, therefore, that the final action is both a compliance backfit and a safety enhancement backfit.

PUBLIC COMMENTS:

Comments were received from 25 separate sources which included 15 utilities, Entergy Operations, Nuclear Energy Institute (NEI), Nuclear Utility Backfitting and Reform Group (NUBARG), Stone & Webster, a public citizens group, a consultant, and three individuals. Four of the commenters believe the NRC should endorse Subsections IWE and IWL. The BWROG submitted a program to be used in lieu of the ASME Code. The utilities generally endorsed the comments submitted by NEI. Some of the commenters were simply seeking clarification of the proposed requirements.

Six general comments were received separately from NEI and NUBARG, but were similar in nature. The first comment is that the application of the compliance exception is inappropriate, and that the proposed rule constitutes a backfit for which a cost-benefit analysis should be performed. The NRC staff agrees that the rulemaking is a backfit. However, as detailed above, the staff believes that invoking the compliance exception to the backfit rule is appropriate.

The second comment was a citation of a paragraph from the Statement of Considerations to the 1985 final backfit rule which addressed the compliance exception. That paragraph addressed "Section 50.109(a)(4) which creates exceptions for modifications necessary to bring a facility into compliance or to ensure through immediately effective regulatory action that a licensee meets a standard of no undue risk to public health and safety." Both NEI and NUBARG assert that the proposed rule is a new interpretation of how to demonstrate compliance with existing standards and therefore constitutes a backfit under 10 CFR 50.109(a)(1). The NRC staff does not believe that the use of the compliance exception must be confined only to the situation addressed in the Statement of Consideration to the 1985 final backfit rule - "omission or mistake of fact." In any event, the current unsatisfactory status of containment inservice inspections can be characterized fairly as, in retrospect, a mistake about and omission from the necessary elements of a satisfactory inspection program.

The third comment is that containments are not experiencing corrosion or degradation that is unanticipated and excessive. NUBARG expressed the idea somewhat differently, as whether "there is a broad-based concern with the operability of containment structures through the industry," and concluded that there was not. The NRC concludes that there is a "broad-based concern" regarding the structural integrity of containment structures. The NRC's approach focuses on two questions: (1) is the corrosion such that there is a reasonable basis for concluding that additional instances of noncompliance with the relevant GDCs, Appendix J, and licensee commitments of the majority of plants; and (2) whether there is a reasonable basis for believing that the corrosion would have been identified and properly addressed by the licensees in the absence of regulatory requirements. When the following factors are considered: (1) the number of occurrences of containment degradation; (2) the increasing rate of containment degradation; (3) the locations of the degradation; (4) two occurrences where containment wall thicknesses were below minimum design wall thickness, and four occurrences where corrosion reduced the thickness of the liner by one-half; (5) the diversity of corrosion paths which have been reported; and (6) the higher than anticipated corrosion rates in many of the occurrences, the NRC staff believes that containments are experiencing corrosion or degradation that is unanticipated and excessive. Further, based upon factors (1) to (6) above, the staff concludes that additional criteria are necessary to ensure that compliance with existing requirements for minimum accepted design wall thicknesses and prestressing forces are maintained and, thereby, the ability of the containment to continue to perform its intended safety function.

The fourth comment is that it is part of the anticipated process for the industry to rely upon NRC inspections and audits to identify problems and then alert the industry through NRC documents such as information notices and generic letters. During the presentation to the ACRS on February 10, 1995, NEI asserted that "[i]t really doesn't matter how the utilities identify these instances of degradation." The staff believes that inspections conducted by licensees should be adequate to ensure that containment degradation is identified without reliance upon NRC inspections.

The fifth comment is that to ensure compliance, the NRC could take individual enforcement action rather than endorse ASME standards. The staff believes that the best approach is to adopt the industry consensus standard (i.e., endorse ASME Section XI Subsections IWE and IWL). Containment corrosion and degradation have been reported since 1986. The patterns of degradation and the corrective actions were not immediately obvious. Given the number and the extent of occurrences, and the variability among plants with regard to the performance and the effectiveness of containment inspections, the staff believes that the best course of action is to endorse ISI requirements so that licensees carry out a consistent approach to ensure that containments comply with design wall thicknesses and prestressing forces.

The sixth comment is that GDC 16 required containments to be designed and constructed with an allowance for corrosion or degradation of the containment wall over the projected design life of the plant. NEI and NUBARG assert that "[i]t is therefore hardly surprising that, as noted in the Statement of Considerations, "[o]ver one-third of the operating containments have experienced corrosion or other degradation. Therefore, they state that there is not a broad-based concern with operability of containment structures. The NRC staff rejects that argument that because containments have corrosion allowances and corrosion was expected to occur that, ipso facto, further inspections are not necessary and the compliance exception is inappropriate. The staff notes that containments have corrosion allowances based upon a 40-year design life. In many cases, the corrosion rate has been found to be greater than that for which the containment was designed (in some cases the rate was twice that predicted or assumed in the design). A corrosion allowance simply increases the tolerance (time period) for corrosion. Once the allowance is eroded, then concern with compliance becomes relevant. Based upon the staff's finding of the number and extent of occurrences of corrosion to date, and the lack of activities to manage the degradation by many licensees, the staff concludes that it is likely that those licensees will be in violation of applicable requirements for containment structural integrity and leak-tightness during the OL term, absent the imposition of Subsections IWE and IWL.

The staff presented the proposed final amendment to the ACRS on February 10, 1995. As previously indicated, NEI also made a presentation to the ACRS. During its presentation, NEI stated that the rule is overly prescriptive and contrary to other initiatives toward performance-based approaches. The staff believes that the best approach is to adopt the industry criteria. By memorandum from Dr. T.S. Kress, Chairman, ACRS, to Mr. James M. Taylor, EDO, dated February 17, 1995, the ACRS concurred with the staff position to incorporate by reference Subsections IWE and IWL into § 50.55a. In this memorandum, the ACRS stated "However, a suitable "metric," which could be used as the basis for a performance-based inspection for the assurance of the structural integrity of the containment, seems to be difficult to identify."

With regard to alternative approaches such as drafting a simple rule and endorsing the ASME Code in a regulatory guide, the staff considered and rejected such alternatives at the proposed rulemaking stage because of the apparent ineffectiveness of licensee containment inspection programs. Licensees were made aware of containment degradation through several industry notices, and yet the staff is still detecting many occurrences of degradation. Information compiled by the staff indicates that many licensee containment programs are not sufficiently comprehensive to detect many of the types of degradation being reported. A rule is needed to ensure that the detailed guidance in Subsections IWE and IWL is implemented uniformly and timely by all licenses to reliably detect such degradation for correction.

RECOMMENDATION:

That the Commission:

1. Note that it is my intention to approve a notice of final rulemaking (Attachment 1) incorporating by reference ASME Code rules for inservice inspection and inservice testing of nuclear power plant containment structures within 10 working days from the date of this paper unless instructed otherwise by the Commission.

2. This rule will not have a significant economic impact on a substantial number of small entities.

3. The amendments to 10 CFR Part 50.55a will be published in the Federal Register, and will become effective 60 days after publication.

   a. No environmental impact statement, negative declaration, or environmental impact appraisal need be prepared in connection with the amendments because the action taken by the amendments will not significantly affect the quality of the human environment.

   b. The reporting and recordkeeping requirements contained in this regulation have been approved by the Office of Management and Budget, OMB approval No. 3150-AC93.

   c. The Office of Public Affairs concurs that a public announcement is not needed.

   d. The NRC staff will inform the Subcommittee on Energy and Power of the House Committee on Commerce and the Subcommittee on Clean Air, Wetlands, Private Property and Nuclear Safety of the Senate Committee on Environment and Public Works of this action by letter such as Attachment 5.

   e. The Federal Register Notice of rulemaking will be distributed by ADM to power reactor licensees/permit holders, applicants for a construction permit for a power reactor, public interest groups, and nuclear steam system suppliers.

COORDINATION:

The Offices of Nuclear Reactor Regulation, Analysis and Evaluation of Operational Data, Administration, and Information Resources Management concur in this draft final rule to amend 10 CFR 50.55a. OGC has reviewed the backfit justifications considered for the final rulemaking and has no legal objection to the conclusion that the rule is subject to the compliance section under 10 CFR 50.109(a)(4)(i).

James M. Taylor
Executive Director for Operations

Contact: Wallace Norris, RES
415-6796

Attachments:

1. Federal Register Notice
2. Documented Evaluation
3. Environmental Assessment and Finding of No Significant Environmental Impact
4. Supporting Statement for Information Collection Requirements
5. Congressional Committee Correspondence
6. Resolution of Public Comments
7. Justification of Action as Safety Enhancement

ATTACHMENT 2

DOCUMENTED EVALUATION

SUMMARY:

The NRC is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's pressure-retaining and structural integrity. As a result of the rate of occurrence of containment degradation, and the extent of containment degradation, the NRC concludes that there is a reasonable likelihood that licensees will not be in conformance with appropriate regulatory requirements and current licensing basis commitments with respect to structural integrity and leak-tightness throughout the term of the operating license, and that imposition of these containment inservice inspection requirements under the compliance exception to 10 CFR 50.109(a)(4)(i) is appropriate.

Based on the results of inspections and audits of plant structures, as well as plant operational experiences, the NRC has determined that many licensee containment examination programs have not detected degradation that could result in a compromise of pressure-retaining capability and structural integrity. The location and extent of corrosion or degradation in a containment can be critical to the containments behavior during an accident.

The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. According to studies of major components ranked according to their relevance to plant safety and effect on plant life, the containment is ranked first in BWRs and second to the reactor pressure vessel in PWRs. There have been 32 reported occurrences of corrosion in metal containments and the liners of concrete containments. This is one-fourth of all operating nuclear power plants. Only four of the 32 occurrences were detected by current containment inspection programs. Nine of these occurrences were first identified by the NRC through its inspections or structural audits. Eleven occurrences were detected by licensees after they were alerted to a degraded condition at another site or through activity other than containment inspection. It is clear that current licensee containment inspection programs have not proved to be adequate to detect the types of degradation which have been reported. There have been 34 reported occurrences of degradation of the concrete or of the post-tensioning systems of concrete containments. This is nearly one-half of these types of containments. Examples of degradation not found by licensees, but initially detected at plants through NRC inspections include: 1) corrosion of steel containment shells in the drywell sand cushion region, resulting in wall thickness reduction to below the minimum design thickness; 2) corrosion of the torus of the steel containment shell (wall thickness below minimum design thickness); 3) corrosion of the liner of a concrete containment to approximately half-depth; 4) grease leakage from the tendons of prestressed concrete containments; and 5) leaching as well as excessive cracking in concrete containments.

The NRC believes that more specific ISI requirements, which expand upon existing requirements for the examination of containment structures in accordance with General Design Criterion (GDC) 53, Appendix A to 10 CFR Part 50, and Appendix J to 10 CFR Part 50, are needed and are justified for the purpose of ensuring that containments continue to maintain minimum design wall thicknesses and prestressing forces as provided for in industry standards used to design containments (e.g., Section III and Section VIII of the ASME Code, and the American Concrete Institute Standard ACI-318), as reflected in license conditions, technical specifications, or licensee commitments (e.g., the Final Safety Analysis Report). The NRC also believes that the occurrences of corrosion and other degradation discussed above would have been detected by licensees implementing the comprehensive periodic examinations set forth in Subsection IWE and Subsection IWL of the ASME Code which are being incorporation by reference into 10 CFR 50.55a.

Recent changes and additions to the ASME Code include provisions to address the concerns outlined above; and the NRC is making these provisions mandatory by amending 10 CFR 50.55a to incorporate by reference these additional portions of the ASME Code (Subsection IWE and Subsection IWL).

STATEMENT OF THE OBJECTIVES:

In view of the increasing rate of occurrences of degradation in containments, the unacceptability of some occurrences, and the variability of present containment examinations, the NRC has determined that it is necessary to include more detailed requirements for the periodic examination of containment structures in the regulations to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise the containment's pressure-retaining integrity. Specifically, this final rule incorporates by reference the 1992 Edition with the 1992 Addenda of both Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants," with specified modifications and a limitation. Subsection IWE provides rules for inspecting the surface of metal containments, the steel liners of concrete containments, pressure-retaining bolting, seals and gaskets, containment vessel welds, and pressure-retaining dissimilar metal welds. Subsection IWL provides rules for the examination of concrete pressure-retaining shells and shell components, and for unbonded post-tensioning systems.

REASONS FOR THE MODIFICATIONS:

An increase in the rate of reported incidents of significant corrosion and degradation of containments has been detected at operating nuclear power plants. Since 1986, 32 instances of corrosion in steel containments have been reported. In two cases, thickness measurements of the walls revealed areas where the wall thickness was below the minimum design thickness. There have been four occurrences reported where the liner of a concrete containment had been reduced locally by corrosion to one-half of its original thickness. Since the early 1970s, 34 incidents of containment degradation related to the post-tensioning systems or the concrete of concrete containments have been reported. Four of these reported incidents which involved grease leakage from tendons also showed signs of leaching of the concrete (See Table 3 and Table 3A).

There are several GDC criteria and ASME Code sections which establish minimum requirements for the design, fabrication, construction, testing, and performance of structures, systems, and components important to safety in water-cooled nuclear power plants (See Table 1). The GDC serve as fundamental underpinnings for many of the most safety important commitments in licensee design and licensing bases. Criterion 16, "Containment design," requires the provision of reactor containment and associated systems to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity in to the environment and to ensure that the containment design conditions important to safety are not exceeded for as long as required for postulated accident conditions. Section III and Section VIII of the ASME Code, and the American Concrete Institute provide design specifications for minimum wall thicknesses and prestressing forces of containments, and these are reflected in license conditions, technical specifications, and licensee commitments for the operating plants.

Criterion 53, "Provisions for containment testing and inspection," requires that the reactor containment design permit: (1) appropriate periodic inspection of all important areas, such as penetrations; (2) an appropriate surveillance program; and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows. Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," of 10 CFR Part 50 contains specific rules for leakage testing of containments. Paragraph III. A. of Appendix J requires that a general inspection of the accessible interior and exterior surfaces of the containment structures and components be performed prior to any Type A test to uncover any evidence of structural deterioration that may affect either the containment structural integrity or leak-tightness (Type A test means tests intended to measure the primary reactor containment overall integrated leakage rate: (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter). None of these existing requirements, however, provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation in licensee containment examination programs, such that there have been cases of noncompliance with GDC 16. Based on the results of inspections and audits, and plant operational experiences, it is clear that many licensee containment examination programs have not detected degradation that could ultimately result in a compromise to the pressure-retaining capability and structural integrity.

Among other things, the GDCs address designing to permit periodic inspection and an appropriate surveillance program. 10 CFR 50.55a implements certain GDCs by requiring inspections of specific components. Criterion 32, "Inspection of reactor coolant pressure boundary," is implemented by Section XI, Subsection IWB, Class 1 components. Criterion 36, "Inspection of emergency core cooling system," is implemented by Section XI, Subsection IWC, Class 2 components. Criterion 45, "Inspection of cooling water system," is implemented by Section XI, Subsection IWD, Class 3 components. Implementation of certain provisions of Criterion 16, "Containment design," and Criterion 53, "Provisions for testing and inspection," will be accomplished by requiring the examinations of the containment as provided for in Subsection IWE and Subsection IWL.

Because of the staff's concern arising from operating experience with regard to the vulnerabilities of highly stressed post-tensioning systems of prestressed concrete containments (PCCs), the staff developed some periodic inspections to monitor the status of these components during the design lives of these containments. Prior to the issuance of Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," in 1973, these inspection provisions were promulgated on a case-by-case basis. This regulatory guide was issued to assure that PCCs were properly monitored and to assure some consistency with regard to industry inspection. Revisions of the regulatory guide were developed and issued to incorporate increased experience in the examination of post-tensioning systems. Revision 2 (1976) contained technical updates to Revision 1, and the costs of implementation of Rev. 2 compared to Rev. 1 were nearly equal. Revision 3 (1990) changed the tendon detensioning and sampling requirements such that a considerable cost savings to the industry would be realized (See Tables 5 - 7). At present, twenty-seven licensees have voluntarily adopted Rev. 3. Five licensees are committed to Rev. 2, six licensees to Rev. 1, and of the five remaining post-tensioned containments, one will start using Rev. 3 with their next scheduled surveillance, three would rather use the industry standard (Subsection IWL) than a regulatory guide, and are therefore waiting for the rule to become effective, and the last PCC is a grouted tendon containment to which these rules would not apply. As pointed out in NUREG/CR-4712, "Regulatory Analysis of Regulatory Guide 1.35 (Revision 3, Draft 2)," the implementation of Rev. 3 of Regulatory Guide 1.35 (Subsection IWL with modifications) will have very little effect on cost, but a positive impact on safety.

The five modifications which were contained in one paragraph of the proposed rule (§ 50.55a(b)(2)(ix)(A)-(E)) to address two concerns of the NRC are retained in the final rule. The first concern was that certain recommendations for tendon examinations that are included in Regulatory Guide 1.35, Rev. 3, are not addressed in Subsection IWL (this involves four of the modifications). The ASME Code has considered these four issues and is in the process of adopting them in Subsection IWL. These issues will be published in future Addenda. The second concern was that if there is visible evidence of degradation of the concrete (e.g., leaching, surface cracking) there may also be degradation of inaccessible areas. This fifth modification contains a provision which would require an evaluation of inaccessible areas when visible conditions exist that could result in degradation of these areas.

The limitation specifies the 1992 Edition with 1992 Addenda of Subsection IWE and Subsection IWL as the earliest version of the ASME Code the staff finds acceptable. This edition and addenda combination incorporates the concept of base metal examinations and will provide a comprehensive set of rules for the examination of post-tensioning systems. The final rule incorporates a provision for an expedited examination schedule. This expedited examination schedule is necessary to prevent a delay in the implementation of Subsection IWE and Subsection IWL (Table 4 list each plant and the delay in implementation which would be encountered without an expedited implementation schedule. Provisions have been incorporated in the final rule so that the expedited examination which would be required 5 years after the effective date of the rule and the routine 120-month examinations are not duplicated.

The proposed rule was published in the Federal Register on January 7, 1994 (59 FR 979). As a result of public comments, four modifications have been added to the final rule. The four modifications are: (1) § 50.55a(b)(2)(x)(A) has been added to the final rule which expands the evaluation of inaccessible areas of concrete containments (Class CC) to metal containments and the liners of concrete containments (Class MC). A commenter believed that given the operating history of Class MC containments, the proposed provision (which only addressed concrete containments) should be expanded in the final rule to include metal containments and the liners of concrete containments; (2) § 50.55a(b)(2)(x)(B) has been added to the final rule to permit alternative lighting and resolution requirements for remote visual examination of the containment which would be precluded on a practical basis by IWA-2200; (3) § 50.55a(b)(2)(x)(C) has been added to the final rule which gives the licensee the option to perform the examinations of Subsection IWE, Examination Category E-B (pressure retaining welds) and Subsection IWE, Examination Category E-F (pressure retaining dissimilar metal welds). The NRC staff concludes that a basis for requiring these examinations does not currently exist; and (4) an optional sampling plan (§ 50.55a(b)(2)(x)(D)) has been added to the final rule which will allow licensees to determine the number of additional components to be examined based on an evaluation to determine the extent and nature of the degradation. Five commenters indicated that the sampling plan used in other subsections of Section XI (on which the Subsection IWE sampling plan is based) is inappropriate for containment components. The Section XI sampling plan was developed to detect degradation such as cracks in piping systems where the degradation can be widespread. In order to determine the extent of the problem, a large percentage of pipe welds must be examined. The commenters believe that a review of the operational history of containment components shows that these components degrade or fail by mechanisms unique to each component which makes the current Section XI sampling program burdensome.

BASIS FOR INVOKING THE COMPLIANCE EXCEPTION:

The NRC is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL to assure that the critical areas of containments are routinely inspected to detect and take corrective action for defects that could compromise a containment's pressure-retaining and structural integrity. As a result of the rate of occurrence of containment degradation, and the extent of containment degradation, the NRC concludes that there is a reasonable likelihood that licensees will not be in conformance with appropriate regulatory requirements and current licensing basis commitments with respect to structural integrity and leak-tightness throughout the term of the operating license, and that imposition of these containment inservice inspection requirements under the compliance exception to 10 CFR 50.109(a)(4)(i) is appropriate.

The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. There have been 32 reported occurrences of corrosion in metal containments and the liners of concrete containments. This is one-fourth of all operating nuclear power plants. Only four of the 32 occurrences were detected by current containment inspection programs. Nine of these occurrences were first identified by the NRC through its inspections or structural audits. Eleven occurrences were detected by licensees after they were alerted to a degraded condition at another site or through activity other than containment inspection. There have been 34 reported occurrences of degradation of the concrete or of the post-tensioning systems of concrete containments. This is nearly one-half of these types of containments. It is clear that current licensee containment inspection programs have not proved to be adequate to detect the types of degradation which have been reported. Examples of degradation not found by licensees, but initially detected at plants through NRC inspections include: 1) corrosion of steel containment shells in the drywell sand cushion region, resulting in wall thickness reduction to below the minimum design thickness; 2) corrosion of the torus of the steel containment shell (wall thickness below minimum design thickness); 3) extensive corrosion of the liner of a concrete containment with local degradation at many locations to approximately half-depth; 4) grease leakage from the tendons of prestressed concrete containments; and 5) leaching as well as excessive cracking in concrete containments.

None of the existing requirements for containment inspection provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation with regard to the performance and the effectiveness of licensee containment examination programs. Based on the results of inspections and audits, and plant operational experiences, it is clear that many licensee containment examination programs have not detected degradation that could result in a compromise of pressure-retaining capability.

The NRC believes that more specific ISI requirements are necessary for the purpose of identifying additional occurrences of noncompliance in containment structures. The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. There have been 66 reported occurrences of containment degradation. Most of those occurrences were first identified by the NRC through its inspections or audits of plant structures, or by licensees while performing an unrelated activity or, after they were alerted to a degraded condition at another site. Given the widespread degradation, the types of degradation, and the location of the degradation, it is clear that all containments are subject to certain type(s) of degradation depending on the design. Information gathered by the staff indicates that many licensees still have not reacted to this serious safety concern and have not initiated comprehensive containment inservice inspection. As a result of the rate of occurrence of containment degradation, the extent of containment degradation, the NRC believes that there is a basis for reasonably concluding that such degradation is widespread and affects virtually all plants. Further, the occurrences of degradation cited should have been detected by licensees when conducting visual inspections of the containment. Based upon the staff's finding of the number and extent of occurrences of corrosion to date, and the lack of activities to manage the degradation by many licensees, the staff concludes that it is likely that those licensees will be in violation of applicable requirements for containment structural integrity and leak-tightness during the OL term, absent the imposition of Subsections IWE and IWL. Therefore, the NRC has determined that imposition of these containment inservice inspection requirements under the compliance exception to 10 CFR 50.109(a)(4)(i) is appropriate.

The staff believes that the final action would also result in a substantial safety increase and that the direct and indirect costs of implementation are justified in view of the significant safety benefit to be gained. The NRC believes that the inspections contained in Subsections IWE and IWL will improve significantly the ability to detect degradation and take timely action to correct degradation of containment structures. A review of early implementation of the maintenance rule (10 CFR 50.65) at nine nuclear power plants, which is documented in NUREG-1526, indicates that most licensees assigned a low priority to the monitoring of structures. Several licensees incorrectly assumed that many of their structures are inherently reliable. This is true so long as there is no degradation. However, the degradation of structures can reduce high margins of safety to a low or negligible margin of safety. The final rule will provide for improved periodic examination of containment structures assuring that the critical areas of containment are periodically inspected to detect and take corrective action for defects that could compromise the containment's pressure-retaining and leak-tight capability. The justification of this action as a safety enhancement is contained in a separate attachment to the package. The staff believes, therefore, that the final action is both a compliance backfit and a safety enhancement backfit.

Recent changes and additions to the ASME Code include provisions to address the concerns outlined above. The NRC is making these provisions mandatory by amending 10 CFR 50.55a to incorporate by reference these additional portions of the ASME Code (Subsection IWE and Subsection IWL). The NRC believes that the occurrences of corrosion and other degradation discussed above would have been detected by licensees implementing the comprehensive periodic examinations set forth in Subsection IWE and Subsection IWL of the ASME Code being incorporated into 10 CFR 50.55a.

ADDITIONAL CONSIDERATION:

The Documented Evaluation for the proposed rulemaking contained the following discussion. The Nuclear Management and Resources Council (NUMARC) has developed a number of industry reports to address license renewal issues. Two of those, one for PWR containments and the other for BWR containments, were developed for the purpose of managing age-related degradation of containments on a generic basis. The NUMARC plan for containments relies on the examinations contained in Subsection IWE and Subsection IWL to manage age-related degradation, and this plan assumes that these examinations are "in current and effective use." In the BWR Containment Industry Report that, NUMARC concluded "On account of these available and established methods and techniques to adequately manage potential degradation due to general corrosion of freestanding metal containments, no additional measures need to be developed and, as such, general corrosion is not a license renewal concern if the containment minimum wall thickness is maintained and verified." Similarly, in the PWR Containment Industry Report, NUMARC concluded that potentially significant degradation of concrete surfaces, the post-tensioning system, and the liners of concrete containments could be managed effectively if periodically examined in accordance with the requirements contained in Subsection IWE and Subsection IWL.

NEI (formerly NUMARC) commented that identification of Subsection IWE and Subsection IWL in the industry reports "[s]hould not be interpreted as supporting the imposition of new requirements on operating plants during their operating license term." "The industry reports provide guidance that can be used by license renewal applicants to address age-related degradation of key plant components. The documents are formatted to identify the aging degradation mechanisms with respect to their potential safety significance during the license renewal term. Potentially significant degradation mechanisms are evaluated to determine if they are addressed by inspection, testing, maintenance or surveillance programs. If the mechanism is adequately managed by an effective program, then the degradation is not considered to be an issue for license renewal."

In Chapter 5.0, Component Life Evaluation, "BWR Containments License Renewal Industry Report," it is stated, "In this section, these potentially significant degradation mechanisms are evaluated further, considering accepted maintenance, inservice inspection, testing and analytical assessment procedures, with the objective of demonstrating that degradation can be managed within established limits." Section 5.1, Life Evaluation Techniques, states, "This section evaluates the potentially significant age-related degradation mechanisms on the basis of determining which mechanisms are effectively managed by currently accepted inspection, testing, maintenance, or analytical techniques." The techniques which are then described are: 1) Section III and Section VIII of the ASME Code for design, Section XI for inservice conditions and flaw evaluation; 2) Appendix J Leakage Rate Testing (Type A, B, and C tests but not the general visual examination); 3) Subsections IWE, IWF, and IWL of Section XI; 4) R.G. 1.35 for the inspection of ungrouted tendons in prestressed concrete containments; 5) BWR suppression chamber coating inspections (as defined in plant Technical Specifications); 6) ultrasonic thickness measurements which a few licensees have instituted due to excessive wall loss; 7) settlement monitoring; and 8) the BWROG Model Containment Inspection Plan. In the Conclusion to Chapter 5.0, it is stated, "On the basis of these currently accepted programs, the components listed below are considered to be managed by currently accepted programs with respect to general corrosion and need not be further evaluated beyond assurance by the license renewal applicant that no plant design or operating features exist that would preclude the applicant from verifying the conclusion and that the programs identified (or their equivalents) are in current and effective use." There are several points to be discussed in regard to various statements made in the industry report. The first point is that one of the identified accepted programs or its equivalent must clearly be in current and effective use before the license renewal application is received in order to verify that potential degradation is adequately managed. The second point to be discussed is which of these identified programs is considered acceptable. Addressing them in the order in which they appear above: 1) design and flaw evaluation - the NRC staff does not consider "refined analyses in accordance with the ASME Boiler and Pressure Vessel Code to demonstrate a wall thickness above the minimum, to be designated as corrosion allowance." as managing age-related degradation mechanisms. Likewise, Section XI flaw evaluation is an after-the-fact determination of degradation; 2) Appendix J Leakage Rate Testing - the NRC staff believes that Type A, B, and C tests only "provide a method of detecting degradation mechanisms in the early stages." for containment components such as seals and gaskets. A significant reduction in structural integrity would have to occur before a leakage rate test would detect the degradation; 3) Subsections IWE, IWF, and IWL - of the comprehensive containment ISI programs, as detailed in the rulemaking package, these are the only programs the staff believes would detect the types of occurrences of degradation being reported; 4) R.G. 1.35 - this regulatory guide has proven effective in detecting degradation of ungrouted tendons in prestressed concrete containments. However, the regulatory guide does not address the inspection of concrete containments which is why Subsection IWL was developed by the ASME; 5) BWR suppression chamber coating inspections - the inspection has generally proven to be effective. However, the NRC staff has found that lack of acceptance standards has resulted in pitting and corrosion not being repaired in many instances; 6) ultrasonic thickness measurements - currently, only a few licensees have performed measurements on either a periodic or comprehensive basis, and generally the measurements were performed in response to the detection of significant wall thinning; 7) settlement monitoring - the monitoring generally is performed by all new plants for approximately the first 5 years. Although a recommended and needed practice, settlement monitoring is not performed to age-related degradation; and 8) the BWROG Model Containment Inspection Plan - as greatly detailed in the response to public comments, this plan only addresses two of the occurrences of corrosion detected in metal containments and dismisses examination of other areas, in which corrosion have been reported, as being not justifiable. The NRC staff's conclusion is that Subsection IWE and Subsection IWL are the only acceptable comprehensive programs for detecting degradation in Class MC and Class CC components.

The final license renewal (10 CFR Part 54) rule was clarified to indicate that "only passive, long-lived structures and components are subject to an aging management review for license renewal." Further, "the second and equally important principle of license renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term." The requirements of the final rule reflect a greater reliance on existing licensee programs that manage the detrimental effects of aging on functionality, including those activities implemented to meet the requirements of the maintenance rule. On page 25 of the License Renewal Rule, *(vii) Future Exclusion of Structures and Components Based on NRC Requirements*, states the Commission's intent to evaluate whether Subsection IWE and Subsection IWL, if implemented, can be considered effective in continuing to manage the effects of aging through any renewal term, and that a positive conclusion could establish the bases for further limiting scope of review for license renewal.

A review of early implementation of the maintenance rule (10 CFR 50.65) at 9 nuclear power plants, which is documented in NUREG-1526, indicates that most licensees assigned a low priority to the monitoring of structures. Several licensees incorrectly assumed that many of their structures are inherently reliable. This is true so long as there is no degradation. However, the degradation of structures can reduce high margins of safety to a low or negligible margin of safety. The performance criteria for monitoring some structures were not predictive and did not include adequate examination/inspection of structures. Hence, the staff has determined that it is necessary to endorse more detailed requirements for the periodic examination of containment structures and codify these by regulations to assure that the critical areas of containments are periodically inspected to detect and take corrective action for defects that could compromise the containment's pressure-retaining capability and structural integrity.

APPENDIX A

Details from Some of the Instances of Degradation

OCCURRENCES OF DEGRADATION

Specifics on some of the reported occurrences of corrosion and degradation are given below. Tables 3 and 3A (which are attached) are a more complete list of reported instances.

Oyster Creek

Steel containment shell corrosion was found in the sand cushion region (elevation 51 feet) of the drywell of Oyster Creek (BWR Mark I) in 1986 during an NRC inspection. The drywell steel shell thickness had been reduced from 1.115 inch to an average thickness of 0.85 inch, with some local spots reduced to about 0.75 inch. The drywell pressure vessel stresses do not exceed the ASME Code allowable stresses for a minimum local thickness of 0.425 inch provided the mean thickness remains above 0.591 inch. This evaluation of stresses took advantage of actual material properties obtained from Certified Mill Test Reports. The root cause of the corrosion is the presence of very corrosive water solution in the sand cushion. The water leaked through a defective rubber gasket from the refueling pool on the top of the drywell into the air gap between the drywell and the concrete shield, then through the gap-forming material, which contains chlorides and sulfides, to the sand cushion. Corrosion also occurred in the upper portion of the drywell where gap-forming material is present.

After the discovery of corrosion in the sand cushion region of the steel drywell at Oyster Creek Plant in 1986, the licensee had committed to conduct ultrasonic test (UT) thickness measurements of the drywell shell at outages of opportunity (whenever a drywell entry is planned or required). On this basis, measurements were performed in September, 1989 and again in February, 1990. An evaluation by the licensee of the data collected indicated a corrosion rate of 9.5 to 10 mils per year. This rate is twice that established on the basis of previous UT measurements at the drywell elevation 51' region. During a recent licensee examination, some local spots were found to have further degraded having lost up to ¾ inch of the wall thickness.

Nine Mile Point 1

Corrosion of the torus at Nine Mile Point 1 (BWR Mark I) was discovered during an NRC special announced team inspection. The torus was designed and constructed uncoated. The UT of the torus shell showed several areas where the thickness of steel was at or near the minimum specified wall thickness. Following the discovery of corrosion at Nine Mile Point 1, a survey of BWRs in NRC Region I was conducted. Although all plants perform periodic visual inspections per technical specification requirements, the extent of this inspection varies (e.g., some plants only examine the torus above water line, and other plants use divers or cameras to inspect the torus underwater). From these limited inspections, Fitzpatrick, Millstone 1, Pilgrim, and Oyster Creek have found degradation of the coatings and have had to clean and recoat the surfaces. Fitzpatrick had some pitting, and Pilgrim experienced flaking of the coating which led to some rusting of the torus.

Fort Calhoun, TMI-1, & Trojan

Extensive grease leakage from tendon sheathings or at joints between conduit lengths has been found at Fort Calhoun, Three Mile Island 1, and Trojan plants. In addition to grease leakage, Three Mile Island 1 and Trojan showed signs of leaching of the concrete in the tendon gallery.

Farley 2

During a visual inspection of the Unit 2 containment at Farley, it was discovered that the grease cap of the shop-end of a vertical tendon was deformed. When the field-end of the tendon in the tendon gallery was opened for inspection, it was found that the anchor head was broken, allowing the tendon to detension completely. An extensive investigation conducted to understand the cause and extent of the occurrence showed that two additional anchor heads were broken, and the tendons were completely detensioned. The anchor heads of 23 other tendons (17 in Unit 2 and 6 in Unit 1) were found to have cracks. The factors contributing to this event can be summarized as (1) high hardness material of the anchor heads, (2) free water in the grease caps, and (3) high stresses in the anchor heads. Similar incidents have been reported at two other plants.

Turkey Point 3

Turkey Point Unit 3 performed their twentieth year tendon surveillance of the containment post-tensioning system. Lift-off forces below the predicted lower limit were found in three of the hoop surveillance tendons and in six additional adjacent tendons. The average tendon lift-off force at one buttress was slightly below the minimum required prestress force. Subsequent evaluation showed that higher local internal containment temperatures were responsible. Even though it was shown that all of the tendon groups provide adequate prestress force to maintain containment integrity at this time, the NRC staff is concerned because the relaxation losses are greater than those that were predicted. The most conservative estimate was that the hoop tendons will provide adequate prestress force to maintain design basis requirements until 1997. Although it is recognized that this was a conservative estimate, the results, should this happen, are that the minimum prestress forces for Unit 3 would be realized some fourteen years earlier than previously predicted.

McGuire 2

On August 24, 1989, the licensee for McGuire Unit 2 (PWR Ice Condenser) reported base metal corrosion on the outside of the steel containment vessel (SCV) at plant elevation 725 feet which was discovered during a pre-integrated leak rate test inspection. The shell is enclosed in a reinforced concrete shield building, with a 6 foot annulus. The failure of the coating has led to the corrosion of the base metal up to 0.123 inch. The degradation of the shell, which has a nominal thickness of 1 inch at elevation 725', is limited to a 37 foot section no higher than 1-1/2 inches above the floor. Corrosion that is up to 0.03 inch deep was also found in areas below the level of the annulus floor, where concrete was removed to expose the shell surface. The below-floor corrosion is due to a lack of sealant at the interface between the shell and the floor. The condition that led to the SCV corrosion is attack by condensed boric acid coolant leaking from some of the instrumentation lines. Drains are provided in the floor, but they are widely separated, and the floor is not sufficiently graded to prevent pooling of the condensate between the drain locations. The design is considered adequate to handle large water spills, but leaking instrumentation lines was not considered. This is a potential problem in the annular area at any BWR or PWR with a steel containment.

McGuire 1, & Catawba 1&2

Subsequently, the McGuire licensee found similar but less extensive corrosion in Unit 1. The licensee also has two units at Catawba which are PWR Ice Condensers with SCV's similar to those at McGuire. An inspection of the Catawba SCV's found corrosion, but it was less extensive than that at the McGuire plants being limited to about a 15 foot section 1 inch above the annulus floor.

Using ASME Code calculations (NE-3200), the minimum required SCV wall thickness at the base for the Catawba containments was found to be 0.6875 inch as governed by external pressure. Given that the nominal wall thickness is 1 inch at all four units, and using the worst case corrosion rate at McGuire, which is 38.5 mils per year, the time frame for corrective action at McGuire would be 17 months, and would be 24 months for Catawba.

Information Notice 89-79 states that the degradation of the containment shells at the McGuire and Catawba plants is considered significant for several reasons. The fact that the corrosion affects four different units indicates that other steel containments with similar configurations may be susceptible to the same problem. Furthermore, the observed rate of corrosion far exceeds the allowance made for corrosion in the containment design. This condition leads to the concern that such corrosion could result in undetected wall thinning to less than the minimum design thickness, accompanied by a loss of leaktightness or structural integrity. This problem can be prevented by a containment inservice inspection program that is adequate to ensure early detection and the maintenance of the intended licensing bases through proper corrosion control.

On April 18, 1990, additional areas of degradation of the steel containment at McGuire Unit 1 were found during an inspection by the licensee. The degradation consists of general coating failures and localized pits having a depth of up to 45 mils. The corrosion is located on the inside surface at the floor level between the upper and the lower containment compartments, in the vicinity of the ice condenser. The corrosion occurs in a 2-inch floor gap filled with cork that interfaces with the coated steel containment. The cork contains moisture originating most likely from the ice condenser or from condensation or both. General surface corrosion, which is presently of no significance, appears throughout the areas accessible for inspection. The licensee had done UT on inaccessible areas, and the worst pitted area still meets minimum thickness requirements.

NRC Information Notice Number 89-79, Supplement 1, specifies that although corrosion of the containment shell at the cork interface of the floor expansion joint has been discovered only at McGuire Unit 1, it is expected that such corrosion will likely occur at other plants with the same design details for the floor expansion joint. There are indications that cork may have been used in foundation level expansion joints in other plants. The additional corrosion in McGuire Unit 1 has occurred at locations previously considered not susceptible to corrosion.

The information notice further states that the detection of corroded steel plate material in the drywells and wet wells of BWR plants and corroded steel containments of PWR plants has resulted in the concern that degradation caused by corrosion may be generic to all types of containments.

Haddam Neck

During the 1987 Integrated Leak Rate Test (ILRT), the licensee observed cracking on the outside of the containment dome. During the 1991 ILRT, the cracking was considered significant to require repair. It was determined by the licensee that the cracks were due to shrinkage, construction joints opening up, and subsequent crack enlargement because of the freeze-thaw cycle. Non-conformance reports and repair procedures were written. A civil engineering firm was hired to evaluate the damage and initiate repairs. Some significant repair of the concrete was performed. The dome is two feet thick, and 8-9" of concrete was removed in some places to find structurally acceptable concrete. Rebar was exposed at this level, but it was determined that the rebar was undamaged. New concrete was poured in the removal areas, and the entire dome was sealed with water proofing. The dome is considered fully restored in function and capacity, but the degradation mechanisms and the extent of repairs needed at the 25 year mark after containment construction are noteworthy.

ATTACHMENT 3

ENVIRONMENTAL ASSESSMENT AND FINDING OF
NO SIGNIFICANT IMPACT

Environmental Assessment:

Identification of Final Action:

The Nuclear Regulatory Commission (NRC) is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants," of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) with specified modifications and a limitation. Subsection IWE of the ASME Code provides rules for inservice inspection, repair, and replacement of Class MC pressure retaining components and their integral attachments and of metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments in light-water cooled power plants. Subsection IWL of the ASME Code provides rules for inservice inspection and repair of the reinforced concrete and the post-tensioning systems of Class CC components. Licensees will be required to incorporate Subsection IWE and Subsection IWL into their inservice inspection (ISI) program. Licensees will also be required to expedite implementation of the containment examinations and to complete the expedited examination in accordance with Subsection IWE and Subsection IWL within 5 years of the effective date of this rule. Provisions have been included that will prevent unnecessary duplication of examinations between the expedited examination and the routine 120-month ISI examinations. Subsection IWE and Subsection IWL have not been previously incorporated by reference into the NRC regulations. The final rule specifies requirements to assure that the critical areas of containments are routinely inspected to detect defects that could compromise a containment's pressure-retaining integrity.

The Need for the Final Action:

The NRC is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL to ensure that the critical areas of containments are periodically inspected to detect and take corrective action for defects that could compromise a containment's structural integrity. As a result of the rate of occurrence of containment degradation, and the extent of containment degradation, the NRC concludes that there is a reasonable likelihood that licensees will not be in conformance with appropriate regulatory requirements and current licensing basis commitments with respect to structural integrity and leak-tightness throughout the term of the operating license. This will ensure that containment structures continue to maintain or exceed minimum accepted design wall thicknesses and prestressing forces as provided for in industry standards used to design containment structures, as reflected in license conditions, technical specifications, and licensee commitments.

The rate of occurrence of corrosion and degradation of containment structures has been increasing at operating nuclear power plants. Over one-third of operating containment structures have experienced corrosion or other degradation. Almost one-half of the occurrences were first identified by the NRC through its inspections or structural audits, or by licensees because they were alerted to a degraded condition at another site.

GDCs 16 and 53, Appendix J, and several ASME Code sections establish minimum requirements for the design, fabrication, construction, testing, and performance of structures, systems, and components important safety in water-cooled nuclear power plants. The requirements for minimum design wall thicknesses and prestressing forces as provided in these industry standards used to design containment structures are reflected in license conditions, technical specifications, and licensee commitments (e.g., the Final Safety Analysis Report). None of the existing requirements, however, provide specific guidance on how to perform the necessary containment examinations. This lack of guidance has resulted in a large variation with regard to the performance and the effectiveness of licensee containment examination programs, such that there have been cases of noncompliance with GDC 16. Based on the results of inspections and audits, and plant operational experiences, it is clear that many licensee containment examination programs have not detected degradation that could result in a compromise of pressure-retaining capability.

The location and extent of corrosion or degradation in a containment can be critical to the containment's behavior during an accident.

The NRC believes that more specific ISI requirements, that expand upon existing requirements for the examination of containment structures in accordance with GDCs 16 and 53, and Appendix J, are needed and are justified to ensure that containment structures continue to maintain at least minimum accepted design wall thicknesses and prestressing forces as reflected in license conditions, technical specifications, or licensee commitments. Based on operating experience, a serious concern exists regarding continued compliance by the operating plants with existing requirements for ensuring containment minimum design wall thicknesses and prestressing forces if the proposed action is not taken. The NRC also believes that the occurrences of corrosion and other degradation would have been detected by licensees when conducting the comprehensive periodic examinations set forth in Subsection IWE and Subsection IWL of the ASME Code, with modifications, as incorporated by reference into 10 CFR 50.55a.

Environmental Impacts of the Final Action:

Performance of those examinations contained in Subsection IWE and Subsection IWL will reduce the risk uncertainties by ensuring that containment margins are maintained. The staff believes that the occurrences of corrosion and other degradation which have been reported would have been detected by licensees implementing the comprehensive periodic examinations set forth in Subsection IWE and Subsection IWL. By managing corrosion and degradation, performance of these examinations will ensure that containments continue to maintain or exceed minimum design wall thicknesses and prestressing forces as provided for in industry standards used to design containments.

Occupational Exposure:

Summary:

The performance of containment examinations, as set forth by the provisions of this final rule is not expected to result in significant occupational radiation exposure for any one individual. The basis for this finding is provided below.

The following examples provide the exposures which were incurred by personnel performing various examinations and tests within the containment. The containment liner examination at the Monticello Plant, which was conducted during plant life extension studies, resulted in 20 millirems exposure compared with a total of 935 millirems for all testing and surveillance activities conducted during the study at this facility. This exposure was incurred not only in conducting the liner examination but also while removing concrete at the liner-to-shell interface in order to ascertain the condition of the liner below the floor level.

Information received from industry inservice inspection specialists performing Regulatory Guide 1.35 examinations indicate that 20 millirems is the highest exposure received during any one inspection, and that over a 10-year period the highest cumulative exposure for any one individual was only 112 millirems.

SECY-94-283 lists exposures for personnel performing containment integrated leakage rate tests. One utility indicated that the total occupational exposure incurred during the performance of an integrated leakage rate test (ILRT) is approximately 0.4 person-rems. An industry inservice inspection specialist indicated that the highest exposure incurred while setting up for an Appendix J test was 50 millirems. It takes approximately 3,500 person-hours to perform the system alignments, drainings, rigging of containment, inspections and walkdowns, and the post-test restoration of the containment.

The total occupational exposure during the performance of local leak rate testing (LLRT) is approximately 2.4 person-rems. It takes approximately 2,500 hours for a complete battery of Type B and C tests. Type B testing is measurement of the leak rate across each pressure-containing or leakage-limiting boundary (electrical penetrations) and presently occurs once every 2 years. Type C testing is a measurement of the containment isolation valves leak rate and also occurs once every 2 years. Type B tests of are estimated to account for about 15 percent of the estimated hours (375 hours). In order to make some estimate of exposure per penetration, an equal exposure per penetration test will be assumed. Taking 15 percent of 2.4 person-rems (the total for all LLRT testing) yields 360 millirems as the exposure for all Type B testing. Using 75 as the average number of Type B penetrations in a plant yields 4.8 millirems as the average exposure per Type B penetration tested. Approximately 110 Type C tests must be performed which would be approximately 2,125 hours. Information received from industry indicates that the testing of a typical Type C penetration takes a 4.5 person crew one shift to complete, or 36 labor hours. Type C testing accounts for 85 percent or 2,040 millirems of the total for LLRT testing. Again assuming a uniform exposure for each valve tested, the average exposure for a crew would be 18.6 millirems per valve tested or 4 millirems per person.

The following information was received on exposures incurred while performing BWR reactor pressure vessel weld examinations. The occupational radiation exposure associated with equipment setup and examination of the two beltline welds of a BWR reactor pressure vessel has been measured to be 1.80 person-rems. The occupational radiation exposure associated with equipment setup and examination of essentially 100% of the length of BWR reactor vessel shell welds is estimated to be 8.6 person-rems.

Information received from another industry inservice inspection specialist indicated that 10 millirems is the highest exposure received while performing Regulatory Guide 1.35 examinations, and that occurred only if the one of the tendons randomly selected was located in the auxiliary building. Based on the Regulatory Guide 1.35 examinations and the Appendix J Type A testing performed, this specialist estimates that the exposure which would be incurred during the general containment inspection (i.e., visual examination to detect corrosion or signs of flaking, blistering, or peeling of coatings) of a PWR containment would be between 10 - 20 millirems because nearly all of the examination can be done remotely. Typically, this visual examination would be performed by one individual. This exposure would be small because the time required to visually examine around penetrations is limited (0.5 hour), and a dose would only be incurred while performing examinations adjacent to a few penetrations.

For Mark II and Mark III containments, personnel entry into the steam tunnel and other design features make the estimate of exposure somewhat higher than for PWRs. The estimate for a typical Mark II or Mark III containment is about 100 millirems for the general visual examination (for one individual).

In response to the discovery of excessive corrosion in six bays of the torus at Nine Mile Point 1, the licensee committed to ultrasonic testing at specific locations to determine if any further areas had corroded to a point below the required minimum wall thickness. These examinations are scheduled to be performed once every 6 months. Sixty-five measurements are taken in each of the one foot by three foot grids (6 grids total - one in each bay at location of thinnest wall). Information received from the licensee indicates that for the four personnel performing the ultrasonic examinations, the total dose is 80 millirems.

The following information was received concerning recent containment examinations of the wetwell performed at a Mark I containment. The licensee contracted to have the sludge vacuumed out of the torus, and a 100% visual examination performed below the waterline. The visual examination was equivalent to a VT-3. Three divers, who are qualified coating inspectors, removed the sludge and performed the visual examination. There was also a number of technicians involved. During a five-week period, it took over 2,000 person-hours to perform the examinations. This included the preparation, such as lighting, ventilation, staging, and the vacuuming and visual examinations). The total exposure incurred during these activities was 4.3 person-rems. It should be noted that the visual examination of the wetwell could have been performed with robots. The licensee chose to use divers for two reasons. One was that the sump drain areas needed to be vacuumed. The second was that the licensee is more comfortable with the divers ability to determine the location (i.e., relative location with respect to a specific plate) of any degradation than with determining the location remotely by camera.

The Tennessee Valley Authority (TVA) initiated a program at Browns Ferry Unit 3 to decontaminate the drywell in an effort to reduce the costs associated with protective clothing (i.e., laundry processing fees, replacement costs, radwaste burial). Up to 30 personnel worked around the clock handwiping nearly all of the surface areas and using special cleaning equipment for the other surfaces. In addition to the 30 people performing the decontamination, carpenters provided scaffolding, and health physics personnel monitored overall radiological conditions. The decontamination project was completed in 8,000 person-hours and resulted in a total accumulation of 8 person-rem. The total cost of the project was approximately $480,000, which was recovered within three months of the project's completion due to easier access into the drywell area.

NRC Staff Estimates of Exposures Which Would Be Incurred:

Based on the 20 millirems exposure incurred while performing the liner examination at Monticello, and estimates from other industry ISI specialists, the estimate of exposure which would be incurred when performing the detailed examinations of PWR containments (free-standing steel and containments with liners is 1.54 person-rems (77 units X 20 millirems). There are three such examinations performed in a 10-year interval. Thus, the estimated exposure over an interval would be 4.6 person-rems. Since these examinations occur approximately once every 3 years, about 23 such examinations would be performed each year. Thus, the total estimated industry exposure for these containments would be approximately 0.460 person-rems or 460 millirems per year.

As detailed previously, in most instances, personnel performing examination of the post-tensioning systems do not incur any exposure. The highest exposure reported while performing these examinations is 20 millirems. Assuming an average crew of six, and assuming that each individual received 20 millirems, than the total exposure would be 120 millirems per examination. There are 43 units with post-tensioning systems, and these examinations occur once every 5 years or twice in an interval. Based on the information received, a dose is received in only about 5% of the examinations. The total estimated industry exposure incurred while performing post-tensioning system examinations would then be 0.48 person-rems or 480 millirems each interval (2 units x 2 examinations per interval x 120 millirems per examination). On the average in any one year, there would be approximately 9 examinations of post-tensioning systems performed. Assuming that exposure was incurred during one of these examinations, the total estimated industry exposure for the examination of post-tensioning systems would be 0.12 person-rems or 120 millirems per year.

Using information received from industry ISI specialists, it is estimated that the exposure which would be incurred performing a detailed visual of the containment in a BWR Mark II or a Mark III containment would be 100 millirems. The estimated exposure for the industry which would be incurred while performing these examinations is 3.6 person-rems each 10-year interval (12 units X 100 millirems x 3 examinations). On the average in any one year, there would be 4 examinations conducted of Mark II and Mark III containments. The total estimated industry exposure for Mark II and Mark III containments would be 0.4 person-rems or 400 millirems per year.

For BWR Mark I reactors, based on information received from industry ISI specialists and from utility personnel, the estimated industry exposure which would be incurred while performing the detailed visual containment examination, including ultrasonic testing, is estimated to be 100.8 person-rems (24 units x 4.2 person-rems). The total estimated industry exposure would be 302.4 person-rems each 10-year interval (24 units x 4.2 person-rems x 3 examinations). This exposure could be decreased by about 30% if robots are used to perform part of the underwater visual examinations. On the average in any one year, there would be 7 examinations conducted of Mark I containments. The total estimated industry exposure for the examination of Mark I containments would be 29.4 rems per year. However, this exposure could also be decreased by about 30% with the use of robots.

The total estimated industry exposure for all containments would be 30.4 person-rems per year. The total estimated industry exposure for containment examinations over an ISI interval (10 years) would be 304 person-rems. Assuming that each unit would perform the examinations required during two full intervals (20 years) plus the examinations required during the five-year expedited implementation (required examinations for the first period of the first interval), the estimated life-time industry exposure would be 911.4 person-rems.

To put the above exposure levels in context with other exposures incurred at nuclear power plants, NUREG-0713, Vol. 14, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 1992," indicates that the total occupational radiation exposure incurred by workers for all functions at light-water reactor nuclear power plants in the United States during 1992 (i.e., the latest year for which occupational radiation exposures have been compiled and documented) was approximately 29,309 person-rems. This represented 13,309 person-rems for BWRs (based upon 37 reactors, each having been in commercial operation at least one full year by the end of 1992), and 16,000 person-rem for PWRs (based upon 73 reactors). Approximately 9.2% (2,697 person-rem) of this total exposure was the direct result of inservice inspection functions at these facilities. Inservice inspection functions at BWRs imposed 8.0% (1,065 person-rem total or 29 person-rem per plant averaged over 37 plants) and at PWRs imposed 10.3% (1,648 person-rem total or 23 person-rem per plant averaged over 73 plants) of this annual total for the year 1992.

For BWR containments, the estimated exposure incurred while performing the containment examinations in any one year (29.4 person-rems or 0.8 person-rems per reactor averaged over 37 plants) would be approximately 0.22% (1992 total exposure at BWRs was 13,309 person-rems) of that presently incurred while conducting all activities at BWRs, and 2.7% of the total exposure as a result of ISI activities (1992 total exposure at BWRs for ISI activities was 1,065 person-rems). For PWR containments, the estimated exposure incurred while performing the containment examinations in any one year compared to the exposure presently incurred while conducting ISI activities at PWRs (1.0 person-rems or 0.013 person-rems averaged over 77 reactors) would be approximately 0.06% (1992 total exposure at PWRs while performing ISI activities was 1,648 person-rems).

The performance of containment examinations, as set forth by the provisions of this final rule, for PWRs, Ice Condensers, and BWR Mark IIs and IIIs is not expected to result in significant occupational radiation exposure (1.0 person-rems per year or 0.05 person-rems per unit averaged over 27 examinations each year). The above categories of plants, for which the occupational radiation exposure is insignificant, represent the vast majority of units (89). For BWR Mark I containments, the occupational radiation exposure which would be incurred by any one individual is insignificant. The estimated occupational radiation exposure which would be incurred per year while performing BWR Mark I containment examination is 29.4 person-rems per year or 4.2 person-rems per unit averaged over 7 examinations per year. However, the estimated occupational radiation exposure per unit does not provide an accurate representation of the actual radiological exposure that would be incurred by any one individual. 10 CFR § 20.101, "Radiation dose standards for individuals in restricted areas" only permits a whole body dose of 1.25 rem per calendar quarter. As a practical matter, licensees carefully manage the exposure incurred by any one individual by practicing and applying "as low as reasonably achievable" (ALARA) principles to protect the health and safety of personnel. In the performance of the examination of BWR Mark I containments, this is accomplished by having several individuals perform the examinations to "spreadout" the exposure. In this manner, no one individual will suffer any significant health effects. It also must be kept in mind that these containment examinations are scheduled to occur at the interval of once every 3 years. This provides licensees ample time for planning the examinations, and scheduling personnel in accord with ALARA considerations. Therefore, the occupational radiation exposure is insignificant given the relatively low exposure on a unit basis and the licensees programs for controlling the impact of exposure for any one individual.

Alternatives to the Proposed Action:

Because of the significant safety issue involved, the Commission has concluded that a comprehensive containment examination program must be instituted. The alternative of taking no action is not an option.

If the NRC did not take action to endorse Subsection IWE and Subsection IWL, the NRC position on examination practices for steel containment structures, and steel liners of concrete containments, and concrete containments and the reinforcing systems of concrete containments would have to be established on a case-by-case basis. If the NRC does not take action to include the Subsection IWE and Subsection IWL rules by reference, improved examination practices for steel, reinforced and post-tensioned concrete containment structures might not be implemented. This will result in containment examinations being performed solely to the present licensee containment programs. This is not a viable alternative as current licensee containment examination programs have proved to be inadequate in detecting the types of degradation which has been reported.

Another alternative to incorporating by reference the requirements of Subsection IWE and Subsection IWL is for the NRC staff to develop rules for inservice inspection 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 in-depth input from industry experts that is achieved through development of the ASME Code by the consensus process.

Alternative Use of Resources:

This action does not involve the use of any resources not previously considered in other rulemakings which incorporated by reference ASME Code requirements into 10 CFR 50.55a.

Agencies and Persons Consulted:

In the preparation of the final rule, the NRC staff obtained information regarding the time required for performing containment ISI and the estimated exposures from several ISI specialists at different firms as well as ISI personnel from several utilities.

FINDING OF NO SIGNIFICANT IMPACT:

Based upon the environmental assessment, the Commission concludes that the final rule will not have a significant effect on the quality of the human environment. Accordingly, the Commission has determined not to prepare an environmental impact statement for the final rule.

ATTACHMENT 5

The Honorable Dan Schaefer, Chairman
Subcommittee on Energy and Power
Committee on Commerce
United States House of Representatives
Washington, DC 20515

Dear Mr. Chairman:

Enclosed for the information of the Subcommittee are copies of a notice of a final rulemaking to amend § 50.55a of 10 CFR Part 50 which will incorporate by reference national codes and standards for the inservice inspection of nuclear power plant components.

This section of the regulations incorporates by reference 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). The Nuclear Regulatory Commission (NRC) is amending these regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants," of Section XI, Division 1, of the ASME Code. These subsections have not been previously endorsed by the NRC. This final rule continues the NRC process of reviewing and, as appropriate, incorporating by reference ASME Code rules for the inservice inspection of components, which until now has been limited to Class 1, Class 2, and Class 3 components. Endorsement of these subsections at this time is considered necessary because significant corrosion and degradation of containments has occurred increasingly at operating nuclear power plants as evidenced by the number of reported incidents.

The final rule will:

- For the first time, incorporate by reference Subsection IWE and Subsection IWL, of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code. The NRC has reviewed the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL of Section XI of the ASME Code and has found that with specified modifications these subsections of Section XI provide an acceptable method for detecting degradation of metal and concrete containments before structural integrity is compromised. Existing regulatory requirements contain general requirements applicable to containment inspection and surveillance, but these regulations do not provide sufficiently specific guidance on how to perform the necessary containment examinations. This has resulted in a large variation in licensee containment examination programs. In spite of present requirements, some containment structures have undergone unacceptable degradation which was not detected by the mandated tests and examinations.
- Require licensees to expedite implementation of the Subsection IWE and Subsection IWL containment examinations by completing the first examination within 5 years of the effective date of this rule. This expedited examination schedule is necessary to prevent a delay in the implementation of Subsection IWE and Subsection IWL and to establish an early baseline for future examinations.

- Include modifications to the endorsement of Subsection IWL to address four issues that are addressed in NRC Regulatory Guide 1.35, Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," but are not currently addressed in Subsection IWL. Because of the importance the NRC attributes to these issues, each issue has been addressed in the final rulemaking in a modification to the endorsement of Subsection IWL.
- Include four modifications which resulted from public comments received on the proposed rule. The four modifications are: (1) licensees will be required to evaluate the acceptability of inaccessible areas of Class MC components when conditions exist in accessible areas that could result in degradation to inaccessible areas; (2) alternative lighting and resolution requirements for remote visual examination of the containment have been added; (3) examination of pressure retaining welds and pressure retaining dissimilar metal welds of Class MC are optional; and (4) an optional sampling plan for determining the number of additional components to be examined if degradation is detected.

In view of the routine nature of the amendment, we do not consider that a public announcement is warranted.

Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure: As stated

cc: Representative Frank Pallone

The Honorable Lauch Faircloth, Chairman
Subcommittee on Clean Air, Wetlands, Private
Property and Nuclear Safety
Committee on Environment and Public Works
United States Senate
Washington, DC 20510

Dear Mr. Chairman:

Enclosed for the information of the Subcommittee are copies of a notice of a final rulemaking to amend § 50.55a of 10 CFR Part 50 which will incorporate by reference national codes and standards for the inservice inspection of nuclear power plant components.

This section of the regulations incorporates by reference 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). The Nuclear Regulatory Commission (NRC) is amending these regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants," of Section XI, Division 1, of the ASME Code. These subsections have not been previously endorsed by the NRC. This final rule continues the NRC process of reviewing and, as appropriate, incorporating by reference ASME Code rules for the inservice inspection of components, which until now has been limited to Class 1, Class 2, and Class 3 components. Endorsement of these subsections at this time is considered necessary because significant corrosion and degradation of containments has occurred increasingly at operating nuclear power plants as evidenced by the number of reported incidents.

The final rule will:

- For the first time, incorporate by reference Subsection IWE and Subsection IWL, of Section XI, Division 1, of the ASME Boiler and Pressure Vessel Code. The NRC has reviewed the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL of Section XI of the ASME Code and has found that with specified modifications these subsections of Section XI provide an acceptable method for detecting degradation of metal and concrete containments before structural integrity is compromised. Existing regulatory requirements contain general requirements applicable to containment inspection and surveillance, but these regulations do not provide sufficiently specific guidance on how to perform the necessary containment examinations. This has resulted in a large variation in licensee containment examination programs. In spite of present requirements, some containment structures have undergone unacceptable degradation which was not detected by the mandated tests and examinations.
- Require licensees to expedite implementation of the Subsection IWE and Subsection IWL containment examinations by completing the first examination within 5 years of the effective date of this rule. This expedited examination schedule is necessary to prevent a delay in the implementation of Subsection IWE and Subsection IWL and to establish an early baseline for future examinations.
- Include modifications to the endorsement of Subsection IWL to address four issues that are addressed in NRC Regulatory Guide 1.35, Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," but are not currently addressed in Subsection IWL. Because of the importance the NRC attributes to these issues, each issue has been addressed in the final rulemaking in a modification to the endorsement of Subsection IWL.
- Include four modifications which resulted from public comments received on the proposed rule. The four modifications are: (1) licensees will be required to evaluate the acceptability of inaccessible areas of Class MC components when conditions exist in accessible areas that could result in degradation to inaccessible areas; (2) alternative lighting and resolution requirements for remote visual examination of the containment have been added; (3) examination of pressure retaining welds and pressure retaining dissimilar metal welds of Class MC are optional; and (4) an optional sampling plan for determining the number of additional components to be examined if degradation is detected.

In view of the routine nature of the amendment, we do not consider that a public announcement is warranted.

Sincerely,

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosure: As stated

cc: Senator Bob Graham

ATTACHMENT 7

DISCUSSION OF JUSTIFICATION AS A SAFETY ENHANCEMENT

The NRC is amending its regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsection IWE and Subsection IWL for the purpose of identifying additional occurrences of noncompliance in containment structures. Based on a preponderance of reliable information, the NRC concludes that this backfitting action will result in detection of degraded containments in noncompliance with relevant Commission requirements and licensing commitments for affected nuclear power plants. These degradations would not likely be detected by licensees under existing requirements and licensee practices. Therefore, the Commission concludes that this backfit is necessary to assure compliance, and that a backfit analysis for this rule is not necessary pursuant to 10 CFR 50.109(a)(4)(i). However, because the Commission believes that this final action would result in a substantial safety increase and that the direct and indirect costs of implementation are justified in view of the increased protection, a backfit analysis in support of a finding that this rule constitutes a substantial increase in public health and safety has been performed and is contained below.

SUBSTANTIAL SAFETY IMPROVEMENT RATIONALE

The rate of occurrence and degradation of containment has been increasing at operating nuclear power plants. To date, there have been 66 occurrences of degradation of containment structures. Since 1986, 32 instances of corrosion in steel containment have been reported (one-fourth of all containment). In two cases, thickness measurements of the walls revealed areas where the wall thickness was below the minimum design thickness. There have been four occurrences reported where the liner of a concrete containment had been reduced locally by corrosion to one-half of its original thickness. Since the early 1970s, 34 incidents of containment degradation related to the concrete or to the post tensioning systems of concrete containment have been reported (nearly one-half of these containment types). Four of these incidents which involved grease leakage from tendons also showed signs of leaching of the concrete. Tables 3 and 3A of Attachment 2 list these occurrences of degradation.

Over one-third of the operating containment have experienced corrosion or other degradation. Thirty per cent of the reported occurrences in metal containment and the liners of concrete containment were first identified by the NRC through its inspections or structural audits. Another forty per cent was identified by licensees because they were alerted to a degraded condition at another site or were detected through an activity unrelated to containment inspection. Examples of degradation not found by licensees, but initially detected at plants through NRC inspections include: steel containment shell corrosion in the dry well sand cushion region (wall thickness reduced to below minimum design thickness); steel containment shell torus corrosion (wall thickness reduced to below minimum design thickness); liners of concrete containments (wall thickness reduced to one-half original thickness); grease leakage from the tendons of prestressed concrete containment; and water seepage, as well as concrete cracking in concrete containment.

The staff surveyed the NRC Regional Offices to determine the type of inspections being performed on containment structures and to determine the effectiveness of the visual inspection as is currently required in 10 CFR Part 50, Appendix J. Based on the survey of licensees (performed by NRC regional inspectors) and on experiences of NRC regional inspectors, the NRC has determined that there are great differences among plants with regard to the Performance and the effectiveness of containment inspections.

The NRC believes that more specific ISI requirements, which expand upon existing requirements for the examination of containment structures in accordance with General Design Criterion (GDC) 16 and 53, Appendix A to 10 CFR Part 50, and Appendix J to 10 CFR Part 50, will improve significantly the ability to detect degradation and take timely action to correct degradation of containment structures.

One of the containment's functions is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity into the environment should an accident occur. The role of an in service inspection program is to uncover any evidence of structural deterioration that may affect either the containment structural integrity or leak-tightness. PRA analyses performed for aging and license renewal studies show that when the major components are ranked according to their relevance to plant safety that the containment is ranked first in BWRs and second to the reactor pressure vessel for PWRs. There is no backup to the containment. In the event of an accident, should containment structural integrity or leak-tightness be adversely affected, then there would be a release into the environment. Licensees have been unable to provide assurance that their present containment inspection programs can detect the types of degradation which have been reported. The final rule will provide for improved periodic examination of containment structures and better assure that the critical areas of containment are periodically inspected to detect defects that could compromise the containment's pressure-retaining and leak-tight integrity.

At present, 10 CFR 50.55a specifies requirements for the in service inspection and in service testing of ASME Class 1, Class 2, and Class 3 components. While this includes principal components within the nuclear steam supply system, it does not include metal or concrete containment. The final rule will expand the scope of 10 CFR 50.55a to include inspection of Class MC (metal) and Class CC (concrete) containment. The NRC has found that these new subsections of Section XI, which are national consensus standards that are developed with NRC participation, will provide an acceptable method for detecting degradation of metal and concrete containment before margins in containment are seriously compromised. It is anticipated that § 50.55a will be amended, as appropriate, to include later editions and addenda of Class MC and Class CC similar to the current process for ASME Code Class 1, Class 2, and Class 3 components.

Enclosure 6 of SECY-93-328 included a discussion of the proposed rule as a cost-justified safety enhancement. NUBARG and NEI submitted several comments on this analysis even though the NRC did not rely upon the safety enhancement rationale for the proposed rule. NUBARG commented, "But the analysis does not address, either quantitatively or qualitatively, the extent that the proposed rule may lower containment failure probabilities." NEI goes further in suggesting that an evaluation of the safety benefits must show that there is an improvement in containment failure probabilities in order for the rule to be justified as a substantial increase in safety.

Based on the results of inspections and audits of plant structures, as well as plant operational experiences, the NRC has determined that many licensee containment examination programs have not detected degradation that could result in a compromise of pressure-retaining capability. As stated previously, the NRC has also determined that there are great differences among plants with regard to the performance and the effectiveness of containment inspections. The Subsection IWE and Subsection IWL containment examination program will reduce the risk to the public, but the change is not quantifiable. The change is not quantifiable since the extent of existing containment degradation is not known because of the limited previous examinations of containment. In PRA studies, it would be easy to conclude that the identified dominant accident sequences provide a complete and unchanging characterization of plant risk, but this is not the case. Risk can be decreased in time by modifying systems and procedures. More importantly, degraded systems, poorly implemented procedures, inadequate maintenance, or ineffective management (to name a few) could drastically alter both the dominant contributors and the absolute magnitude of risk at a plant. Therefore, it is very important to provide assurance that the estimate of risk provided by the PRA is actually realized or even bettered in practice. The NRC believes that the Subsection IWE and Subsection IWL containment examination program will provide this assurance, with respect to containment performance.

The NRC has determined that many licensee containment examination programs have not detected degradation that could result in a compromise of pressure-retaining capabilities. The industry response to the growing number of reports of adverse, unanticipated degradation has not been effective. There is a large variation with regard to the performance and effectiveness of containment inspections, and there is no evidence that even a significant minority of licensees have instituted comprehensive containment examinations in response to the reported occurrences of degradation. A review of early implementation of the maintenance rule (10 CFR 50.65) at 9 nuclear power plants, which is documented in NUREG-1526, indicates that most licensees assigned a low priority to the monitoring of structures. Several licensees incorrectly assumed that many of their structures are inherently reliable. This is true so long as there is no degradation. However, the degradation of structures can reduce high margins of safety to a low or negligible margin of safety. The performance criteria for monitoring some structures were not predictive and did not include adequate examination/inspection of structures.

The final license renewal (10 CFR Part 54) rule was clarified to indicate that "only passive, long-lived structures and components are subject to an aging management review for license renewal." Further, "the second and equally important principle of license renewal holds that the plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term." The requirements of the final rule reflect a greater reliance on existing licensee programs that manage the detrimental effects of aging on functionality, including those activities implemented to meet the requirements of the maintenance rule.

NUBARG commented that the safety enhancement analysis minimized the impact of differences in containment types and design on the need for the proposed rule as required under Section 50.109(c). NUBARG believes that such an assessment leads to the conclusion that the most significant cases of degradation have been the result of unique factors, and do not indicate a generic, industry-wide problem. The Commission disagrees. There are many paths for water to reach the areas of Mark I containments which experienced the excessive corrosion. Thus, even though minor unique design characteristics contributed to the occurrence of degradation, the scenarios are by no means unique. The staff believes that every containment will eventually experience corrosion at the floor to shell interface, for example. In addition, based on reported occurrences at other units, there are other areas in the containment which are susceptible to degradation. In fact, during subsequent licensee and NRC inspections in response to those reported occurrences, 30% of the Mark I containment have been found to have varying levels of corrosion. Many of the reported occurrences are not limited to one specific containment type and could occur in several containment types (e.g., at any free-standing steel containment shell or the liners of concrete containment). This indicates a generic, industry-wide problem. Given the increasing rate of occurrence of degradation, the determination that current licensee containment inspection programs generally have not detected the types of degradation being reported, and that degradation has been discovered in over one-third of the containment, the Commission believes that it is reasonable to perform a visual examination of a sampling of the containment surfaces in all containment on a periodic basis.

DISCUSSION OF COSTS

Incorporating by reference Subsection IWE and Subsection IWL, of Section XI, Division 1, of the ASME Code will establish the Commission's requirements with respect to the examination of steel containment structures and metal liners of concrete containment, and concrete containment and reinforcing systems of concrete containment on a generic basis for applicants and licensees, thereby minimizing the need for case-by-case evaluations and reducing the time and effort required for submittal preparations and licensee reviews.

The Commission believes that industry objections to NRC endorsement of Subsection IWE and Subsection IWL must be viewed in the context of the industry's participation in the process of development of these subsections. The American National Standards Institute (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 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 objectives 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 technical committee for resolution.

A number of public commenters on the proposed rulemaking believed that the NRC underestimated the resources required to develop the ISI plan and implement the periodic inspections. However, none of these commenters submitted their own estimates. Another commented that submitted detailed comments, with estimates, stated that the staff's resource estimates were an accurate average based on their analysis. The NRC believes that the resource estimates are accurate as they were developed with information received from utility inservice inspection specialists. In fact, a commenter stated that his organization found the NRC estimates to be about average.

NUBARG believes exemption requests will be numerous because the different types of containment designs were not adequately considered by the NRC. NUBARG also believes that the impacts of the five-year expedited examination schedule were not considered nor were the costs of supplemental augmented inspections. The NRC does not expect a large number of exemption requests to be submitted. Subsection IWE and Subsection IWL were specifically developed taking into consideration differences in containment design. One example is containment components exempted from examination because of inaccessibility. Another example is the augmented examination category where the Owner is responsible for determining the surface areas to be included based on considerations such as differences in design, susceptibility to accelerated degradation, and other mitigating factors. With regard to the impact of the five-year implementation schedule, this has been clarified in the final rule. Some commenters misinterpreted the provision in the proposed rule as a requirement to perform all of the examinations which would normally be required during a 10-year ISI interval to be performed in 5 years for the expedited examination. The intent of the provision is to require only the examinations which would be performed during the first period of the first in service inspection interval to be conducted during the expedited examination. These are examinations which would be performed approximately every 31/2 years. The expedited examination will not require accelerating the examinations as was interpreted by some commenters.

Summary of Costs:

The Estimated Resource Burden on the NRC:

It costs $15,000 - $20,000 to review an entire in service inspection (ISI) program. Because the Subsection IWE and Subsection IWL programs are new, it is estimated that the costs to the NRC would be $4,000 - $5,000 for each Subsection IWE/IWL review. Taking into account the augmented examination schedule (submittal of an ISI plan by each licensee within 5 years of the effective date of the rule) and basing the estimation of twenty reviews for the first year, costs to the NRC would be $80,000 - $100,000 per year for the first five years. Also, based on the expected questions of intent and interpretation which always accompany any new ISI program, another man-year might be required (taking into account regional personnel, writing of Safety Evaluation Reports, etc). This would be a total of 2 man-years for the first 5 years. After Subsection IWE and Subsection IWL have been implemented by all of the licensees, it is anticipated that subsequent ISI review and staff time would be approximately 1 man-year per year.

In order to alleviate this burden on the NRC staff, the final rule will not require submittal of the initial Subsection IWE and Subsection IWL ISI plans, but instead, the Owners will be required to keep these on site for audit.

Impacts

Industry Impacts:

The final rule will require each licensee to:

1. Develop and implement an initial Inservice Inspection Plan (ISI);
2. Develop and implement 10-year updates to ISI plan.

Following is a description of and cost estimate for the requirements identified above.

Develop and Implement an Initial Inservice Inspection Plan

The tasks associated with the initial ISI plan development are identified below for a representative facility. For most of the tasks, the cost is a direct function of the manpower involved. Engineering, drafting, and consulting labor has been costed at $66 per hour and clerk labor at $28 per hour. These hourly rates are based on 1984 base wage rates adjusted by a factor of 1.8 for fringe benefits and plant management, with escalation to 1994 dollars based on the GNP Implicit Price Deflator.[(1)] Employees are assumed to work 167 hours per month (i.e., based on a 2004 hour per year).

The following cost estimates assume no containment inspections are presently taking place, and are therefore conservative. Estimates are based on information received from utility in service inspection specialists minus the estimated hours to perform the visual examinations of containment welds required by Subsection IWE Examination Category E-B and the surface examination of containment welds required by Subsection IWE Examination Category E-F. The performance of the examinations required by these categories were made optional in the final rule based on public comments received on the proposed rule. Based on ASME committee discussions, it is anticipated that the ASME will delete these examinations from Subsection IWE in future Code addenda.

ISI Plan Development

Drawing Update- Includes preparation of ISI drawings for the containment structure, numbering ISI components (penetrations, supports, etc.) and performing field as-builts as required (7 person-months for draftsman)
$66 per hr x 167 hrs per month x 7 months... $77K

Computer Database Preparation - Entering components into the ISI computer database program for tracking purposes (1 Person-Month for Clerk)
$28 per hr x 145 hours..$4K

Video Mapping Containment - For job planning $67K

Inspection Plan Preparation - Preparation of the ISI plan to include initial containment inspection, review of as-built data, update the ISI program, prepare code exemption relief requests, review construction data, etc. (6 person-months for engineer)
$66 per hour x 167 his per month x 6 months...$66K

Clerical Assistance - To assist engineer for review of construction records, typing, archival search, preparation of ISI program updates, etc. (3 person-months for clerk)
$28 per hr x 167 his per month x 2 months...$10K

Additional Engineer and Consultant Work (4 person-months)
$53 per hr x 167 his per month x 4 months...$44K

TOTAL ONE-TIME COST...$268K

Implementation of Initial Inservice Inspection

The final rule will result in detailed examinations of containment and post-tensioning systems in accordance with the containment ISI plan.

It is estimated that the required Subsection IWE in service inspection (ISI) of the containment each ten year ISI interval will require 7000 hours of technicians' time (e.g., NDE examiner, Level II technician). For the purposes of this analysis, the staff has assumed that 75% (5300 hours) of this inspection effort will occur at the end of each 10 year ISI interval, and the remaining 25% (1700 hours) will be distributed every three years corresponding to the 3-year cycle for Appendix

J inspections.

These assumptions result in a distribution of examination effort over a 10 year period of 567 hours each in years 3, 6, and 9, and 5300 hours in year 10. Applying an hourly labor rate of $66 results in a total cost per reactor on a 1994 present value basis of about $106K to $195K for a 10% and 5% real discount rate, respectively.

Regulatory Guide 1.35, Revision 2, "In service Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," was published in January 1976, and many licensees voluntarily adopted its provisions. Regulatory Guide 1.35, Revision 3, was issued on July 11, 1990. Twenty-six licensees have voluntarily adopted this regulatory guide, and some of these provisions have been incorporated into their Technical Specifications. For these licensees, there would be incremental costs of $5K to $12K associated with adoption of the Subsection IWL tendon surveillance program due to specified modification in the rule.

Five licensees are committed to Revision 2, six licensees are committed to Revision 1, and six licensees have their own tendon surveillance program. For these licensee's, following is a description of and cost estimate for adopting the provisions of Revision 3 of R.G. 1.35.

If the assumption were made that a licensee currently was not performing any examination of the post-tensioning system, the costs to perform this examination per Revision 3 of R.G. 1.35 is approximately $300,000. (Revision 2 contained technical updates to Revision 1, and the costs of implementation of Rev. 2 compared to Rev. 1 were nearly equal. Revision 3 changed the tendon detensioning and sampling requirements such that a considerable cost savings would be realized (See Tables 5 - 7 of Attachment 2). Thus, implementation of Subsection IWL would result in a cost savings to the industry.).

Tendon detensioning requirements are changed in Revision 3 relative to Revision 2 because detensioning of only one tendon in a group is required as compared with the previous requirement that all tendons selected for inspection were to be detensioned. When the effect of this change is combined with the change in tendon sampling requirements, a considerable cost savings is realized by industry. Tables 5 through 8 compare this reduction in tendon sampling requirements and the associated cost savings. Using a typical Type III[(2)] containment (nine inspections), the total cost saving to industry per plant resulting from implementation of the sampling and tendon detensioning requirements in Revision 3 is estimated to be $320,000 (assuming a discount rate of 5% per year). Assuming a 10% discount rate, the cost savings would be approximately $278,000[(3)].

Periodic Updates to Inservice Inspection Plan

Updates to the initial ISI plan would be for subsequent 10-year ISI intervals. Updates would be required in accordance with provisions in § 50.55a. Based on a remaining useful life of 30 years, two succeeding 10-year ISI intervals will occur following the initial in service inspection for which updates will be necessary. For the purposes of the analysis, implementation of the update is assumed to occur at the midpoint of these 10-year intervals which corresponds to impacts occurring 15 years and 25 years into the future.

Industry in service inspection specialists have estimated an 1800-hour engineering effort (mainly ISI engineers) per reactor to perform plan updates during a given 10-year ISI interval. Based on an engineer labor rate of $66 per hour, this results in a cost of $120K cost (1994 dollars) per reactor per 10-year in service interval. This cost is assumed to occur in the year 2010 and again in the year 2020; corresponding to 15 years and 25 years into the future, respectively. Assuming a 10% real discount rate, the 1994 present value of two ISI plan updates is estimated at $40K per reactor. If a 5% real discount rate is assumed. the 1994 Present value cost is about $95K.

Implementation of Periodic Inservice Inspections

The following analysis is based on the following assumptions. The required Subsection IWE in service inspection (ISI) of the containment each ten year ISI interval will require 7000 hours of technicians' time, and that 75% of this inspection effort will occur at the end of each 10 year ISI interval, and the remaining 25% will be distributed every three years corresponding to the 3-year cycle for Appendix J inspections. These assumptions result in a distribution of examination effort over a 20 year period of 567 hours each in years 12, 15, 18, 21, 24, and 27, and 5300 hours each in years 20 and 30. Applying an hourly labor rate of $66 results in a total cost per reactor on a 1994 present value basis of about $210K to $385K for a 10% and 5% real discount rate, respectively.

The costs estimates for Regulatory Guide 1.35, Rev. 3, initial in service inspection will be the same as the costs for the periodic in service inspections. Refer to pages 7-8 and 7-9 for those costs.

Industry Cost - Summary

Based on the foregoing analysis, high and low estimates of lifetime costs are summarized below. Results are presented on a per reactor basis. The low estimates assume a 10% real discount rate, whereas the high estimate assumes a 5% real discount rate. All costs are expressed in 1994 dollars and all future costs are present valued.

Summary - Lifetime Costs for a Facility Presently Using Regulatory Guide 1.35, Rev. 3 (1994 dollars)

	High Estimate	Low Estimate
Cost Per Reactor	$1108K	$735K

Summary - Lifetime Costs for a Facility Presently Using Regulatory Guide 1.35, Rev. 2 (1994 dollars)

	High Estimate	Low Estimate
Cost Per Reactor	$789K	$455K

Occupational Exposure

Summary:

The performance of containment examinations, as set forth by the provisions of this final rule, is not expected to result in significant occupational radiation exposure for any one individual. The basis for this finding is provided below.

The following examples provide the exposures which were incurred by personnel performing various examinations and tests within the containment. The containment liner examination at the Monticello Plant, which was conducted during plant life extension studies, resulted in 20 millirems exposure compared with a total of 935 millirems for all testing and surveillance activities conducted during the study at this facility. This exposure was incurred not only in conducting the liner examination but also while removing concrete at the liner-to-shell interface in order to ascertain the condition of the liner below the floor level.

Information received from industry inservice inspection specialists performing Regulatory Guide 1.35 examinations indicate that 20 millirems is the highest exposure received during any one inspection, and that over a 10-year period the highest cumulative exposure for any one individual was only 112 millirems.

SECY-94-283 lists exposures for personnel performing containment integrated leakage rate tests. One utility indicated that the total occupational exposure incurred during the performance of an integrated leakage rate test (ILRT) is approximately 0.4 person-rems. An industry inservice inspection specialist indicated that the highest exposure incurred while setting up for an Appendix J test was 50 millirems. It takes approximately 3,500 person-hours to perform the system alignments, drainings, rigging of containment, inspections and walkdowns, and the post-test restoration of the containment.

The total occupational exposure during the performance of local leak rate testing (LLRT) is approximately 2.4 person-rems. It takes approximately 2,500 hours for a complete battery of Type B and C tests. Type B testing is measurement of the leak rate across each pressure-containing or leakage-limiting boundary (electrical penetrations) and presently occurs once every 2 years. Type C testing is a measurement of the containment isolation valves leak rate and also occurs once every 2 years. Type B tests of are estimated to account for about 15 percent of the estimated hours (375 hours). In order to make some estimate of exposure per penetration, an equal exposure per penetration test will be assumed. Taking 15 percent of 2.4 person-rems (the total for all LLRT testing) yields 360 millirems as the exposure for all Type B testing. Using 75 as the average number of Type B penetrations in a plant yields 4.8 millirems as the average exposure per Type B penetration tested. Approximately 110 Type C tests must be performed which would be approximately 2,125 hours. Information received from industry indicates that the testing of a typical Type C penetration takes a 4.5 person crew one shift to complete, or 36 labor hours. Type C testing accounts for 85 percent or 2,040 millirems of the total for LLRT testing. Again assuming a uniform exposure for each valve tested, the average exposure for a crew would be 18.6 millirems per valve tested or 4 millirems per person.

The following information was received on exposures incurred while performing BWR reactor pressure vessel weld examinations. The occupational radiation exposure associated with equipment setup and examination of the two beltline welds of a BWR reactor pressure vessel has been measured to be 1.80 person-rems. The occupational radiation exposure associated with equipment setup and examination of essentially 100% of the length of BWR reactor vessel shell welds is estimated to be 8.6 person-rems.

Information received from another industry inservice inspection specialist indicated that 10 millirems is the highest exposure received while performing Regulatory Guide 1.35 examinations, and that occurred only if the one of the tendons randomly selected was located in the auxiliary building. Based on the Regulatory Guide 1.35 examinations and the Appendix J Type A testing performed, this specialist estimates that the exposure which would be incurred during the general containment inspection (i.e., visual examination to detect corrosion or signs of flaking, blistering, or peeling of coatings) of a PWR containment would be between 10 - 20 millirems because nearly all of the examination can be done remotely. Typically, this visual examination would be performed by one individual. This exposure would be small because the time required to visually examine around penetrations is limited (0.5 hour), and a dose would only be incurred while performing examinations adjacent to a few penetrations.

For Mark II and Mark III containments, personnel entry into the steam tunnel and other design features make the estimate of exposure somewhat higher than for PWRs. The estimate for a typical Mark II or Mark III containment is about 100 millirems for the general visual examination (for one individual).

In response to the discovery of excessive corrosion in six bays of the torus at Nine Mile Point 1, the licensee committed to ultrasonic testing at specific locations to determine if any further areas had corroded to a point below the required minimum wall thickness. These examinations are scheduled to be performed once every 6 months. Sixty-five measurements are taken in each of the one foot by three foot grids (6 grids total - one in each bay at location of thinnest wall). Information received from the licensee indicates that for the four personnel performing the ultrasonic examinations, the total dose is 80 millirems.

The following information was received concerning recent containment examinations of the wetwell performed at a Mark I containment. The licensee contracted to have the sludge vacuumed out of the torus, and a 100% visual examination performed below the waterline. The visual examination was equivalent to a VT-3. Three divers, who are qualified coating inspectors, removed the sludge and performed the visual examination. There was also a number of technicians involved. During a five-week period, it took over 2,000 person-hours to perform the examinations. This included the preparation, such as lighting, ventilation, staging, and the vacuuming and visual examinations). The total exposure incurred during these activities was 4.3 person-rems. It should be noted that the visual examination of the wetwell could have been performed with robots. The licensee chose to use divers for two reasons. One was that the sump drain areas needed to be vacuumed. The second was that the licensee is more comfortable with the divers ability to determine the location (i.e., relative location with respect to a specific plate) of any degradation than with determining the location remotely by camera.

The Tennessee Valley Authority (TVA) initiated a program at Browns Ferry Unit 3 to decontaminate the drywell in an effort to reduce the costs associated with protective clothing (i.e., laundry processing fees, replacement costs, radwaste burial). Up to 30 personnel worked around the clock handwiping nearly all of the surface areas and using special cleaning equipment for the other surfaces. In addition to the 30 people performing the decontamination, carpenters provided scaffolding, and health physics personnel monitored overall radiological conditions. The decontamination project was completed in 8,000 person-hours and resulted in a total accumulation of 8 person-rem. The total cost of the project was approximately $480,000, which was recovered within three months of the project's completion due to easier access into the drywell area.

NRC Staff Estimates of Exposures Which Would Be Incurred:

Based on the 20 millirems exposure incurred while performing the liner examination at Monticello, and estimates from other industry ISI specialists, the estimate of exposure which would be incurred when performing the detailed examinations of PWR containments (free-standing steel and containments with liners is 1.54 person-rems (77 units X 20 millirems). There are three such examinations performed in a 10-year interval. Thus, the estimated exposure over an interval would be 4.6 person-rems. Since these examinations occur approximately once every 3 years, about 23 such examinations would be performed each year. Thus, the total estimated industry exposure for these containments would be approximately 0.460 person-rems or 460 millirems per year.

As detailed previously, in most instances, personnel performing examination of the post-tensioning systems do not incur any exposure. The highest exposure reported while performing these examinations is 20 millirems. Assuming an average crew of six, and assuming that each individual received 20 millirems, than the total exposure would be 120 millirems per examination. There are 43 units with post-tensioning systems, and these examinations occur once every 5 years or twice in an interval. Based on the information received, a dose is received in only about 5% of the examinations. The total estimated industry exposure incurred while performing post-tensioning system examinations would then be 0.48 person-rems or 480 millirems each interval (2 units x 2 examinations per interval x 120 millirems per examination). On the average in any one year, there would be approximately 9 examinations of post-tensioning systems performed. Assuming that exposure was incurred during one of these examinations, the total estimated industry exposure for the examination of post-tensioning systems would be 0.12 person-rems or 120 millirems per year.

Using information received from industry ISI specialists, it is estimated that the exposure which would be incurred performing a detailed visual of the containment in a BWR Mark II or a Mark III containment would be 100 millirems. The estimated exposure for the industry which would be incurred while performing these examinations is 3.6 person-rems each 10-year interval (12 units X 100 millirems x 3 examinations). On the average in any one year, there would be 4 examinations conducted of Mark II and Mark III containments. The total estimated industry exposure for Mark II and Mark III containments would be 0.4 person-rems or 400 millirems per year.

For BWR Mark I reactors, based on information received from industry ISI specialists and from utility personnel, the estimated industry exposure which would be incurred while performing the detailed visual containment examination, including ultrasonic testing, is estimated to be 100.8 person-rems (24 units x 4.2 person-rems). The total estimated industry exposure would be 302.4 person-rems each 10-year interval (24 units x 4.2 person-rems x 3 examinations). This exposure could be decreased by about 30% if robots are used to perform part of the underwater visual examinations. On the average in any one year, there would be 7 examinations conducted of Mark I containments. The total estimated industry exposure for the examination of Mark I containments would be 29.4 rems per year. However, this exposure could also be decreased by about 30% with the use of robots.

The total estimated industry exposure for all containments would be 30.4 person-rems per year. The total estimated industry exposure for containment examinations over an ISI interval (10 years) would be 304 person-rems. Assuming that each unit would perform the examinations required during two full intervals (20 years) plus the examinations required during the five-year expedited implementation (required examinations for the first period of the first interval), the estimated life-time industry exposure would be 911.4 person-rems.

To put the above exposure levels in context with other exposures incurred at nuclear power plants, NUREG-0713, Vol. 14, "Occupational Radiation Exposure at Commercial Nuclear Power Reactors and Other Facilities 1992," indicates that the total occupational radiation exposure incurred by workers for all functions at light-water reactor nuclear power plants in the United States during 1992 (i.e., the latest year for which occupational radiation exposures have been compiled and documented) was approximately 29,309 person-rems. This represented 13,309 person-rems for BWRs (based upon 37 reactors, each having been in commercial operation at least one full year by the end of 1992), and 16,000 person-rem for PWRs (based upon 73 reactors). Approximately 9.2% (2,697 person-rem) of this total exposure was the direct result of inservice inspection functions at these facilities. Inservice inspection functions at BWRs imposed 8.0% (1,065 person-rem total or 29 person-rem per plant averaged over 37 plants) and at PWRs imposed 10.3% (1,648 person-rem total or 23 person-rem per plant averaged over 73 plants) of this annual total for the year 1992.

For BWR containments, the estimated exposure incurred while performing the containment examinations in any one year (29.4 person-rems or 0.8 person-rems per reactor averaged over 37 plants) would be approximately 0.22% (1992 total exposure at BWRs was 13,309 person-rems) of that presently incurred while conducting all activities at BWRs, and 2.7% of the total exposure as a result of ISI activities (1992 total exposure at BWRs for ISI activities was 1,065 person-rems). For PWR containments, the estimated exposure incurred while performing the containment examinations in any one year compared to the exposure presently incurred while conducting ISI activities at PWRs (1.0 person-rems or 0.013 person-rems averaged over 77 reactors) would be approximately 0.06% (1992 total exposure at PWRs while performing ISI activities was 1,648 person-rems).

The performance of containment examinations, as set forth by the provisions of this final rule, for PWRs, Ice Condensers, and BWR Mark IIs and IIIs is not expected to result in significant occupational radiation exposure (1.0 person-rems per year or 0.05 person-rems per unit averaged over 27 examinations each year). The above categories of plants, for which the occupational radiation exposure is insignificant, represent the vast majority of units (89). For BWR Mark I containments, the occupational radiation exposure which would be incurred by any one individual is insignificant. The estimated occupational radiation exposure which would be incurred per year while performing BWR Mark I containment examination is 29.4 person-rems per year or 4.2 person-rems per unit averaged over 7 examinations per year. However, the estimated occupational radiation exposure per unit does not provide an accurate representation of the actual radiological exposure that would be incurred by any one individual. 10 CFR § 20.101, "Radiation dose standards for individuals in restricted areas" only permits a whole body dose of 1.25 rem per calendar quarter. As a practical matter, licensees carefully manage the exposure incurred by any one individual by practicing and applying "as low as reasonably achievable" (ALARA) principles to protect the health and safety of personnel. In the performance of the examination of BWR Mark I containments, this is accomplished by having several individuals perform the examinations to "spreadout" the exposure. In this manner, no one individual will suffer any significant health effects. It also must be kept in mind that these containment examinations are scheduled to occur at the interval of once every 3 years. This provides licensees ample time for planning the examinations, and scheduling personnel in accord with ALARA considerations. Therefore, the occupational radiation exposure is insignificant given the relatively low exposure on a unit basis and the licensees programs for controlling the impact of exposure for any one individual.

CONCLUSION:

As discussed in the preceding, the NRC has determined that there is a substantial increase in the overall protection of the public health and safety and that the direct and indirect costs of implementation for that facility are justified in view of this increased protection. In addition, the final rule is consistent with the Staff Requirement Memorandum, SECY-93-086 - Backfit Considerations (i.e., the substantial increase criterion are flexible enough to allow for qualitative arguments, and flexible enough to allow for arguments that consistency with national and international standards, or the incorporation of widespread industry practices, contributes either directly or indirectly to a substantial increase in safety.

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1. NRC analysis of industry labor rates is available in NUREG/CR-4627, Generic Cost Estimates; abstract 6.3, "Industry Labor Rates," June 1986.

2. See Table 5.

3. Estimated cost savings for industry for a Type I containment, using discount rates of 5% and 10%, are estimated to be $356,000 and $293,000, respectively, over the life of the plant. For a Type II containment, the estimated cost savings, are estimated to be $874,000 and $468,000, respectively.