XI.S1

ASME Section XI, Subsection IWE

INTRODUCTION

10 CFR 50.55a imposes the examination requirements of ASME B&PV Code, Section XI for steel containments and metallic components of concrete containments. Examination requirements for ASME Class MC pressure retaining components and their integral attachments, and metallic shell and penetration liners of Class CC pressure retaining components and their integral attachments are defined in Subsection IWE, 1992 Edition with 1992 Addenda or later edition, as approved in 10 CFR 50.55a.

ASME Code Section XI, Subsection IWE and additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program for managing aging of steel containments, steel liners of concrete containments, and other containment components described above for the period of extended operation.

Evaluation of 10 CFR 50.55a and Subsection IWE is presented below. An applicant needs to ensure that its implementation of 10 CFR 50.55a and Subsection IWE for containment steel elements is consistent with this evaluation.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Program: Subsection IWE-1000 specifies the components of steel containments and steel liners of concrete containments within its scope. The components within the scope of Subsection IWE (1992 Edition with 1992 Addenda) are Class MC pressure retaining components (steel containments) and their integral attachments; metallic shell and penetration liners of Class CC containments and their integral attachments; containment seals and gaskets; containment pressure retaining bolting; and metal containment surface areas, including welds and base metal. The concrete portions of containments are inspected in accordance with Subsection IWL. Subsection IWE exempts from examination (1) components that are outside the boundaries of the containment as defined in the plant-specific design specification; (2) embedded or inaccessible portions of containment components that met the requirements of the original construction code of record; (3) components that become embedded or inaccessible as a result of vessel repair or replacement, provided IWE-1230 and IWE-5220 are met; and (4) piping, pumps, and valves that are part of the containment system, or which penetrate or are attached to the containment vessel (governed by IWB or IWC). 10 CFR 50.55a(b)(2)(ix) specifies additional requirements for inaccessible areas. It states that the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. NUREG-1611 indicates that the management of potential corrosion of inaccessible areas of structural steel liners and steel containment shells when conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas needs to be addressed on a plant-specific basis. Examination requirements for containment supports are not within the scope of Subsection IWE.

(2) Preventive Action: No preventive actions are specified; Subsection IWE is a monitoring program.

(3) Parameters Monitored or Inspected: Table IWE-2500-1 specifies seven categories for examination. The categories, parts examined, and examination methods are presented in the following table.

CATEGORY	PARTS EXAMINED	EXAMINATION METHOD**
E-A	Containment Surfaces	General Visual, Visual VT-3
E-B*	Pressure Retaining Welds	Visual VT-1
E-C	Containment Surfaces Requiring Augmented Examination	Visual VT-1, Volumetric
E-D	Seals, Gaskets, and Moisture Barriers	Visual VT-3
E-F*	Pressure Retaining Dissimilar Metal Welds	Surface
E-G	Pressure Retaining Bolting	Visual VT-1, Bolt torque or tension test
E-P	All Pressure Retaining Components (Pressure retaining boundary, penetration bellows, airlocks, seals and gaskets)	10 CFR Part 50, Appendix J (Containment Leak Rate Testing)

* These two categories are optional, in accordance with 10 CFR 50.55a(b)(2)(ix)(C).

** The applicable examination method (where multiple methods are listed) depends on the particular subcategory within each category.

Table IWE-2500-1 references the applicable section in IWE-3500 that identifies the aging effects that are evaluated. The parameters monitored or inspected depend on the particular examination category. For Examination Category E-A, as an example, metallic surfaces (without coatings) are examined for evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities. For Examination Category E-D; seals, gaskets, and moisture barriers are examined for wear, damage, erosion, tear, surface cracks, or other defects that may violate the leak-tight integrity.

(4) ***Detection of Aging Effects:*** The frequency and scope of examination requirements specified in 10 CFR 50.55a and Subsection IWE ensure that aging effects would be detected before they would compromise the design basis requirements. As indicated in IWE-2400, inservice examinations and pressure tests must be performed in accordance with one of two Inspection Programs, A or B, on a specified schedule. Under Inspection Program A, there are four inspection intervals (at 3, 10, 23, and 40 years) for which 100% of the required examinations must be completed. Within each interval, there are various inspection periods for which a certain percentage of the examinations must be performed to reach 100% at the end of that interval. In addition, a general visual examination is performed once each inspection period. After 40 years of operation, any future examinations must be performed in accordance with Inspection Program B. Under Inspection Program B, from the time the plant is placed into service, there is an initial interval of 10 years and successive intervals of 10 years each, during which 100% of the required examinations must be completed. An expedited examination of containment is required by 10 CFR 50.55a in which an inservice (baseline) examination must be performed by September 9, 2001. Thereafter, subsequent examinations are performed every 10 years from the baseline examination. Regarding the extent of examination, all accessible surfaces receive a visual examination such as General Visual, VT-1, or VT-3 (see table presented in attribute (3) above). Selected areas, such as containment surfaces requiring augmented examination (E-C) require volumetric examination. All pressure retaining components (E-P) require system leakage testing in accordance with 10 CFR Part 50, Appendix J.

(5) ***Monitoring and Trending:*** With the exception of inaccessible areas, all surfaces are monitored by virtue of the examination requirements on a scheduled basis as described above. When component examination results require evaluation of flaws, evaluation of areas of degradation, or repairs, and the component is found to be acceptable for continued service, the areas containing such flaws, degradation, or repairs shall be reexamined during the next inspection period, in accordance with Examination Category E-C (containment surfaces requiring augmented examination). When these reexaminations reveal that the flaws, areas of degradation, or repairs remain essentially unchanged for three consecutive inspection periods, these areas no longer require augmented examination in accordance with Examination Category E-C. IWE-2430 requires that (a) examinations performed during any one inspection that reveal flaws or areas of degradation exceeding the acceptance standards shall be extended to include an additional number of examinations within the same category approximately equal to the initial number of examinations, and (b) when additional flaws or areas of degradation that exceed the acceptance standards are revealed, all of the remaining examinations within the same category must be performed to the extent specified in Table 2500-1 for the inspection interval. Alternatives to these examination requirements are provided in 10 CFR 50.55a(b)(2)(ix)(D).

(6) ***Acceptance Criteria:*** IWE-3000 provides acceptance standards for components of steel containments and liners of concrete containments. Table IWE-3410-1 presents criteria to evaluate the acceptability of the containment components for service following the preservice examination and each inservice examination. This table specifies the acceptance standard for each Examination Category (E-A, E-B, E-C, etc.). Most of the acceptance standards rely upon visual examinations. Areas that are suspect require an engineering evaluation or require correction by repair or replacement. For some examinations such as Augmented Examinations, numerical values are specified for the acceptance standards. For the containment steel shell or liner, material loss exceeding 10% of the nominal containment wall thickness, or material loss that is projected to exceed 10% of the nominal containment wall thickness prior to the next examination, shall be documented. Such areas shall be accepted by engineering evaluation or corrected by repair or replacement in accordance with IWE-3122.

(7) ***Corrective Actions:*** Subsection IWE states that components whose examination results indicate flaws or areas of degradation that do not meet the acceptance standards listed in Table-3410-1 can be considered acceptable if an engineering evaluation indicates that the flaw or area of degradation is nonstructural in nature or has no effect on the structural integrity of the containment. Except as permitted by 10 CFR 50.55a(b)(ix)(D), components that do not meet the acceptance standards are required to satisfy additional examination requirements and the flaw or area of degradation must be removed by mechanical methods or the component repaired. For repair of components within the scope of Subsection IWE, IWE-4000 and IWE-3124 state that repairs and reexaminations shall comply with the requirements of IWA-4000. IWA-4000 provides rules and requirements for the repair of pressure retaining components including metal containments and metallic liners of concrete containments. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** When areas of degradation are identified, an evaluation is required to determine if repair or replacement is necessary. If the evaluation determines that repair or replacement is necessary, Subsection IWE requires confirmation to ensure that appropriate corrective actions have been completed and are effective. Subsection IWE states that repairs and reexaminations shall comply with the requirements of IWA-4000. Reexaminations are required to be conducted in accordance with the requirements of IWA-2000 and the recorded results must demonstrate that the repair meets the acceptance standards set forth in Table IWE-3410-1. Additional confirmation of leak tightness is achieved through the pressure tests required by 10 CFR Part 50, Appendix J. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** An approved site QA Program would be applicable to Subsection IWE. The licensee is responsible for preparation of plans, schedules, and inservice inspection summary reports. The licensee is also responsible for the preparation of written examination instructions and procedures, verification of qualification level of personnel who perform the examinations, and documentation of a Quality Assurance Program. IWA-6000 specifically covers the requirements for the preparation, submittal, and retention of records and reports. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) ***Operating Experience:*** Since ASME Section XI, Subsection IWE was only recently incorporated into 10 CFR 50.55a, long term experience with Subsection IWE for managing aging of containment components needs to be established. Prior to incorporation of Subsection IWE into 10 CFR 50.55a, operating experience has been demonstrated by implementation of 10 CFR Part 50, Appendix J containment inspection requirements. The license renewal applicant should provide plant-specific operating experience related to inservice inspection of containment and occurrences of degradation. IN 97-10 identifies specific locations where concrete containments are susceptible to liner plate corrosion. Applicants should consider the liner plate corrosion concerns described in IN 97-10 and review plant-specific operating experience to determine applicability.

REFERENCES

10 CFR Part 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Office of the Federal Register, National Archives and Records Administration, 2000.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWA, *General Requirements*, edition approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWB*, Requirements for Class 1 Components of Light-Water Cooled Power Plants*, 1989 or later edition, as approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWC, *Requirements for Class 2 Components of Light-Water Cooled Power Plants*, 1989 or later edition, as approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants,* 1992 Edition with 1992 Addenda or later edition, as approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,* 1992 Edition with 1992 Addenda or later edition, as approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

NRC Information Notice 97-10, *Liner Plate Corrosion in Concrete Containment*, March 13, 1997.

NUREG-1611, *Aging Management of Nuclear Power Plant Containments for License Renewal*, U.S. Nuclear Regulatory Commission, September 1997.

XI.S2

ASME Section XI, Subsection IWL

INTRODUCTION

10 CFR 50.55a imposes the examination requirements of ASME B&PV Code, Section XI for reinforced and prestressed concrete containments. Examination requirements for concrete components of ASME Class CC concrete containments are defined in Subsection IWL, 1992 Edition with 1992 Addenda or later edition, as approved in 10 CFR 50.55a.

ASME Code Section XI, Subsection IWL and additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program for managing aging of concrete containments for the period of extended operation.

Evaluation of 10 CFR 50.55a and Subsection IWL is presented below. An applicant needs to ensure that its implementation of 10 CFR 50.55a and Subsection IWL for containment concrete elements and prestressing systems is consistent with this evaluation.

EVALUATION AND TECHNICAL BASIS

(1) Scope of Program: Subsection IWL-1000 specifies the components of concrete containments within its scope. The components within the scope of Subsection IWL (1992 Edition with 1992 Addenda) are reinforced concrete and unbonded post-tensioning systems of Class CC containments, as defined by CC-1000. Subsection IWL exempts from examination portions of the concrete containment that are inaccessible (e.g., concrete covered by liner, foundation material, or backfill, or obstructed by adjacent structures or other components). 10 CFR 50.55a(b)(2)(viii) specifies additional requirements for inaccessible areas. It states that the licensee shall evaluate the acceptability of concrete in inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. NUREG-1611 indicates that the management of aging effects associated with leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel in inaccessible areas when conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas needs to be addressed on a plant-specific basis. Steel liners for concrete containments and their integral attachments are not within the scope of Subsection IWL, but are included within the scope of Subsection IWE.

(2) Preventive Action: No preventive actions are specified; Subsection IWL is a monitoring program. The staff notes that a coating program currently credited for managing the effects of aging of concrete surfaces should be continued during the extended period of operation.

(3) Parameters Monitored or Inspected: Table IWL-2500-1 specifies two categories for examination of concrete surfaces: Category L-A for all concrete surfaces and Category L-B for concrete surfaces surrounding tendon anchorages. Both of these categories rely upon visual examination methods. Concrete surfaces are examined for evidence of damage or degradation such as concrete cracks. IWL-2510 specifies that concrete surfaces are examined for conditions indicative of degradation such as those defined in ACI 201.1R. Table IWL-2500-1 also specifies Category L-B for test and examination requirements for unbonded post tensioning systems. Tendon anchorage and wires or strands are visually examined for cracks, wear, corrosion, and mechanical damage. Tendon wires or strands are also tested for yield strength, ultimate strength, and elongation. Tendon corrosion protection medium is tested by analysis for alkalinity, water content, and soluble ion concentrations.

(4) ***Detection of Aging Effects:*** The frequency and scope of examination requirements specified in 10 CFR 50.55a and Subsection IWL ensure that aging effects would be detected before they would compromise the design basis requirements. The frequency of inspection is specified in IWL-2400. Concrete inspections are performed in accordance with Examination Category L-A. Under Subsection IWL, inservice inspections for concrete and unbonded post-tensioning systems are required at 1, 3, and 5 years following the structural integrity test. Thereafter, inspections are performed at 5 year intervals. In the case of tendons, only a sample of the tendons of each tendon type requires examination at each inspection. The tendons to be examined during an inspection are selected on a random basis. Table IWL-2521-1 specifies the number of tendons to be selected for each type (e.g. hoop, vertical, dome, helical, and inverted U) for each inspection period. The required minimum number of each tendon type selected for inspection varies from 2 to 4 percent. Regarding detection methods for aging effects, all concrete surfaces receive a visual VT-3C examination. Selected areas, such as those that indicate suspect conditions and areas surrounding tendon anchorages receive a more rigorous VT-1 or VT-1C examination. Prestressing forces in sample tendons are measured. In addition, one sample tendon of each type is detensioned. A single wire or strand is removed from each detensioned tendon for examination and testing. The visual examination methods and testing described above would identify the aging effects of accessible concrete components and the aging effects of prestressing systems in concrete containments.

(5) ***Monitoring and Trending:*** With the exception of inaccessible areas, all concrete surfaces are monitored on a regular basis by virtue of the examination requirements as described above. For prestressed containments, trending of prestressing forces in tendons is required. In addition to the random sampling used for tendon examination, one tendon of each type is selected from the first year inspection sample and designated as a common tendon. Each common tendon is then examined during each inspection. This provides monitoring and trending information over the life of the plant. 10 CFR 50.55a and Subsection IWL also require that prestressing forces in all inspection sample tendons be measured by lift-off tests and compared to acceptance standards based on the predicted force for that type of tendon over its life.

(6) ***Acceptance Criteria:*** IWL-3000 provides acceptance criteria for concrete containments. For concrete surfaces, the acceptance criteria rely on the determination of the "Responsible Engineer" (as defined by the ASME Code) regarding whether there is any evidence of damage or degradation sufficient to warrant further evaluation or repair. Although the acceptance criteria are qualitative, guidance is provided in IWL-2510, which references ACI 201.1R for identification of concrete degradation. In addition, IWL-2320 requires the Responsible Engineer to be a registered professional engineer experienced in evaluating the inservice condition of structural concrete and knowledgeable of the design and construction codes and other criteria used in design and construction of concrete containments. The staff notes that ACI 201.1R does not provide acceptance criteria. However, acceptance criteria based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R are acceptable. This document provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects; and specifies criteria when further evaluation is needed. ACI 349.3R is a recently published report written to evaluate existing nuclear safety-related concrete structures and reflects operating experience with reinforced concrete structures. The acceptance standards for the unbonded post-tensioning system are quantitative in nature. For the post-tensioning system, quantitative acceptance criteria are given for tendon force, tendon wire or strand samples, and corrosion protection medium. 10 CFR 50.55a and Subsection IWL do not define the method for calculating predicted tendon prestressing forces for comparison to the measured tendon lift-off forces. The predicted tendon forces are to be calculated in accordance with Regulatory Guide 1.35.1, which provides an acceptable methodology for use through the period of extended operation.

Note: for aging effects associated with freeze-thaw, leaching of calcium hydroxide, aggressive chemical attack, reaction with aggregates, and corrosion of embedded steel, Chapter III A of this report provides acceptance criteria, which if met, can demonstrate that the particular aging effect/mechanism is not significant.

(7) ***Corrective Actions:*** Subsection IWL specifies that items for which examination results do not meet the acceptance standards shall be evaluated in accordance with IWL-3300 "Evaluation". Items that do not meet the acceptance standards are to be evaluated and described in an engineering evaluation report. The report needs to include an evaluation of whether the concrete containment is acceptable without repair of the item and if repair is required, the extent, method, and completion date of the repair or replacement. The report also identifies the cause of the condition and the extent, nature, and frequency of additional examinations. Subsection IWL also provides repair procedures to follow in Article IWL-4000. This includes requirements for the concrete repair, repair of reinforcing steel, and repair of the post-tensioning system. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** When areas of degradation are identified, an evaluation is performed to determine if repair or replacement is necessary. As part of this evaluation, IWL-3300 requires that the engineering evaluation report include the extent, nature, and frequency of additional examinations. IWL-4000 contains requirements for examination of areas that are repaired. Pressure test requirements following repair or modifications are presented in IWL-5000. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** IWA-1400 provides requirements for the preparation of plans, schedules, and inservice inspection summary reports. In addition, written examination instructions and procedures, verification of qualification level of personnel who perform the examinations, and documentation of a Quality Assurance Program are required. IWA-6000 specifically covers the requirements for the preparation, submittal, and retention of records and reports. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) ***Operating Experience:*** Since ASME Section XI, Subsection IWL was only recently incorporated by 10 CFR 50.55a, long term experience with Subsection IWL for managing aging of containment concrete components needs to be established. Prior to incorporation of Subsection IWL into 10 CFR 50.55a, operating experience has been demonstrated by implementation of 10 CFR 50, Appendix J containment inspection requirements. The license renewal applicant should provide plant-specific operating experience related to inservice inspection of containment and occurrences of degradation.

REFERENCES

10 CFR Part 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Office of the Federal Register, National Archives and Records Administration, 2000.

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, 2000.

ACI Standard 201.1R, *Guide for Making a Condition Survey of Concrete in Service*, American Concrete Institute.

ACI Standard 349.3R-96, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*, American Concrete Institute.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWA, *General Requirements*, editions approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda or later edition, as approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda or later edition, as approved in 10 CFR 50.55a, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

NUREG-1557, *Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal*, U.S. Nuclear Regulatory Commission, October 1996.

NUREG-1611, *Aging Management of Nuclear Power Plant Containments for License Renewal*, U.S. Nuclear Regulatory Commission, September 1997.

XI.S3

ASME Section XI, Subsection IWF

INTRODUCTION

10 CFR 50.55a imposes the inservice inspection requirements of ASME B&PV Code Section XI for Class 1, 2, 3 and MC piping and components and their associated supports. Inservice inspection of supports is covered in Subsection IWF, 1989 edition or later edition as approved in 10 CFR 50.55a. Therefore, ASME Code Section XI, Subsection IWF constitutes an existing mandated program that can be credited for managing aging of ASME Class 1, 2, 3 and MC supports for license renewal.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Program:*** For Class 1 piping and component supports, Subsection IWF (1989 edition) refers to Subsection IWB for the inspection scope and schedule. Based on Table IWB-2500-1, Examination Category B-J "Pressure Retaining Welds in Piping," Note (1)(d), only 25% of non-exempted supports are subject to examination. Supports exempt from examination are the supports for piping systems that are exempt from examination, based on pipe diameter or service. The same supports are inspected in each 10-year inspection interval. For Class 2, 3 and MC piping and component supports, Subsection IWF (1989 edition) refers to Subsections IWC, IWD, and IWE for the inspection scope and schedule. Based on Table IWC-2500-1, Examination Categories C-F-1 and C-F-2, 7.5% of non-exempted supports are subject to examination for Class 2 systems. The same supports are inspected in each 10-year inspection interval. No specific numerical percentages are identified in Subsections IWD and IWE for Class 3 and Class MC, respectively. Starting with the 1990 addenda to the 1989 edition, the scope of Subsection IWF was revised. The required percentages of each type of non-exempt support subject to examination were incorporated into Table IWF-2500-1. The revised percentages are 25% of Class 1 non-exempted piping supports, 15% of Class 2 non-exempted piping supports, 10% of Class 3 non-exempted piping supports, and 100% of supports other than piping supports (Class 1, 2, 3, and MC). For pipe supports, the total sample is comprised of supports from each system (such as Main Steam, Feedwater, RHR), where the individual sample sizes are proportional to the total number of non-exempt supports of each type and function within each system. For multiple components other than piping, within a system of similar design, function, and service, the supports of only one of the multiple components are required to be examined. To the extent practical, the same supports selected for examination during the first inspection interval are examined during each successive inspection interval.

(2) ***Preventive Action:*** No preventive actions are specified; IWF is a monitoring program.

(3) ***Parameters Monitored or Inspected:*** As part of the visual examination (VT-3), general corrosion, which is an indication of loss of material, is noted during the inspection. Cracking is not explicitly noted in Subsection IWF; the visual inspection (VT-3) would be expected to identify only relatively large cracks. Table IWF-2500-1 (1989 edition) specifies examination of the following: (F1.10) Mechanical connections to pressure retaining components and building structure; (F1.20) Weld connections to building structure; (F1.30) Weld and mechanical connections at intermediate joints in multi-connected integral and non-integral supports; (F1.40) Clearances of guides and stops, alignment of supports, assembly of support items; (F1.50) Spring supports and constant load supports; (F1.60) Sliding Surfaces; (F1.70) Hot or cold position of spring supports and constant load supports.(Starting with the 1990 addenda, these items are listed in paragraph IWF-2500.) The parameters monitored or inspected include corrosion; deformation; misalignment;

improper clearances; improper spring settings; damage to close tolerance machined or sliding surfaces; and missing, detached, or loosened support items.

(4) ***Detection:*** VT-3 visual examination is specified in Table IWF-2500-1. The complete inspection scope is repeated every 10 year inspection interval. The qualified VT-3 inspector uses judgment in assessing general corrosion; it is not documented unless loss of structural capacity is suspected.

(5) ***Monitoring and Trending:*** There is no requirement to monitor or report progressive, time-dependent degradation. Unacceptable conditions, per IWF-3400 are noted for correction or further evaluation. Since the same supports are monitored during each inspection interval, trending is possible.

(6) ***Acceptance Criteria*:** The acceptance standards for visual examination are specified in IWF-3400. In IWF-3410(b)(5), "roughness or general corrosion which does not reduce the load bearing capacity of the support" is given as an example of a "non-relevant condition", which requires no further action. IWF-3410(a) identifies the following conditions as unacceptable: (i) deformations or structural degradations of fasteners, springs, clamps, or other support items; (ii) missing, detached, or loosened support items; (iii) arc strikes, weld spatter, paint, scoring, roughness, or general corrosion on close tolerance machined or sliding surfaces; (iv) improper hot or cold positions of spring supports and constant load supports; (v) misalignment of supports; (vi) improper clearances of guides and stops. Identification of unacceptable conditions triggers an expansion of the inspection scope, in accordance with IWF-2430, and re-examination of the supports requiring corrective action during the next inspection period, in accordance with IWF-2420(b).

(7) ***Corrective Actions:*** In accordance with IWF-3122, supports containing unacceptable conditions are evaluated or tested, or corrected prior to returning to service. Corrective actions are delineated in IWF-3122.2. IWF-3122.3 provides an alternative for evaluation or testing, to substantiate integrity for intended purpose. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) ***Operating Experience:*** To date, IWF sampling inspections appear to be effective in managing aging effects for ASME Class 1, 2, 3 and MC supports. If the current sample size proves to be inadequate in the future, as plants age, then revisions to the Subsection IWF inspection scope would be expected. There is reasonable assurance that the Subsection IWF inspection program will be effective through the period of extended operation.

REFERENCES

10 CFR 50.55a, *Codes and Standards*, Office of the Federal Register, National Archives and Records Administration, January 2000.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWB, *Requirements for Class 1 Components of Light-Water Cooled Power Plants,* 1989 Edition. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWC, *Requirements for Class 2 Components of Light-Water Cooled Power Plants,* 1989 Edition. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWD, *Requirements for Class 3 Components of Light-Water Cooled Power Plants,* 1989 Edition.

The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsection IWF, *Requirements for Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants*, 1989 Edition. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

XI.S4

10 CFR Part 50, Appendix J

INTRODUCTION

As described in 10 CFR Part 50, Appendix J, containment leak rate tests are required "to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment." A containment leak rate test (LRT) program performed in accordance with 10 CFR Part 50, Appendix J can ensure that these intended functions and the integrity of the containment structure are maintained through the period of extended operation.

Currently Appendix J provides two options, Option A and Option B, either of which can be chosen to meet the requirements of a containment LRT program. Under Option A, all of the testing must be performed on a periodic interval. Option B is a performance-based approach. Some of the differences between these options are discussed below, and more detailed information for Option B is provided in NRC Regulatory Guide 1.163 and NEI 94-01, Rev. 0.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Program:*** The scope of the containment LRT program includes all pressure retaining components. Two types of tests shall be implemented. Type A tests are performed to measure the overall primary containment integrated leakage rate, which is obtained by summing leakage through all potential leakage paths including containment welds, valves, fittings, and components that penetrate containment. Type B tests are performed to measure local leakage rates across each pressure containing or leakage limiting boundary for containment penetrations. Type A and Type B tests described in 10 CFR Part 50, Appendix J are acceptable methods for performing these leak rate tests. Leakage testing for containment isolation valves (normally performed under Type C tests), if not included under this program, would be included under leakage rate test programs for systems containing the isolation valves.

(2) ***Preventive Action:*** The containment LRT program is a monitoring program; no preventive actions are specified.

(3) ***Parameters Monitored or Inspected:*** The parameters to be monitored are leakage rates through containment shells, containment liners, and associated welds, penetrations, fittings, and other access openings.

(4) ***Detection of Aging Effects:*** A containment LRT program is effective in detecting degradation of containment shells, liners, and components that compromise the containment pressure boundary, including seals and gaskets. While the calculation of leakage rates demonstrates the leak-tightness and structural integrity of the containment, it does not by itself provide information that would indicate that aging degradation has initiated or that the capacity of the containment may have been reduced for other types of loads such as seismic loading. This would be achieved with the additional implementation of an acceptable containment inservice inspection program as described in Sections XI.S1 and XI.S2 of this report**.**

(5) ***Monitoring and Trending:*** Since the LRT program must be repeated throughout the operating license period, the entire pressure boundary is being monitored over time. The frequency of these tests depends on which option (A or B) is selected. With Option A, testing is performed on a regular fixed time interval as defined in 10 CFR Part 50, Appendix J. In the case of Option B, the interval for testing may be increased based on acceptable performance of meeting leakage limits in prior tests. Additional details for implementing Option B are provided in NRC R.G. 1.163 and NEI 94-01, Rev.0.

(6) Acceptance Criteria: Acceptance criteria for leakage rates are defined in plant technical specifications. These acceptance criteria meet the requirements in 10 CFR Part 50, Appendix J and are part of each plant's current licensing basis (CLB). The CLB carries forward to the period of extended operation.

(7) Corrective Actions: Corrective actions are taken in accordance with 10 CFR Part 50, Appendix J and NEI 94-01. When leakage rates do not meet the acceptance criteria, then an evaluation is required to identify the cause of the unacceptable performance and appropriate corrective actions must be taken. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) Confirmation Process: When corrective actions are implemented to repair a condition causing excessive leakage, confirmation by additional leak rate testing is required to confirm that the deficiency has been corrected. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) Administrative Controls: Results of the LRT program must be documented as described in 10 CFR Part 50, Appendix J to demonstrate that the acceptance criteria for leakage have been satisfied. The records are required to be available for inspection at the plant site. The test results that exceed the performance criteria must be assessed under 10 CFR 50.72 and 10 CFR 50.73. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) Operating Experience: To date, the 10 CFR Part 50, Appendix J leak rate testing program has been effective in preventing unacceptable leakage through the containment pressure boundary. Implementation of Option B for testing frequency must be consistent with plant-specific operating experience.

REFERENCES

10 CFR Part 50, Appendix J, *Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors*, Office of the Federal Register, National Archives and Records Administration, 2000.

10 CFR 50.72, *Immediate Notification Requirements for Operating Nuclear Power Reactors*, Office of the Federal Register, National Archives and Records Administration, 1997.

10 CFR 50.73, *Licensee Event Report System*, Office of the Federal Register, National Archives and Records Administration, 1997.

NEI 94-01, Rev. 0, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J*, Nuclear Energy Institute, July 26, 1995.

NRC Regulatory Guide 1.163, *Performance-Based Containment Leak-Test Program,*" September 1995.

XI.S5

Masonry Wall Program

INTRODUCTION

NUREG-1557 identifies IE Bulletin (IEB) 80-11, "Masonry Wall Design," and Information Notice (IN) 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," as an acceptable basis for a masonry wall aging management program. IEB 80-11 required identification of masonry walls in close proximity to, or having attachments from, safety related systems or components, and evaluation of design adequacy and construction practice. IN 87-67 recommended plant-specific condition monitoring of masonry walls and administrative controls to ensure that the evaluation basis developed in response to IEB 80-11 is not invalidated by (1) deterioration of the masonry walls (e.g., new cracks not considered in the reevaluation), (2) physical plant changes such as installation of new safety-related systems or components in close proximity to masonry walls, or (3) reclassification of systems or components from non-safety-related to safety-related.

Important elements in the evaluation of many masonry walls during the IEB 80-11 program were (1) the installation of steel edge supports to provide a sound technical basis for boundary conditions used in seismic analysis and (2) the installation of steel bracing to ensure containment of unreinforced masonry walls during a seismic event. Consequently, in addition to the development of cracks in the masonry walls, loss of function of the structural steel supports and bracing would also invalidate the evaluation basis.

After completion of the IEB 80-11 program, additional masonry walls that perform intended functions were subsequently identified and evaluated during implementation of the USI A-46 program. Therefore, the masonry walls included in the scope of license renewal should reflect not only the scope of the IEB 80-11 program, but also those walls subsequently identified as performing an intended function and evaluated during implementation of the USI A-46 program.

The objective of the masonry wall program is to ensure that the evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Program:*** The scope includes all masonry walls identified as performing an intended function in the IEB 80-11 program and the USI A-46 program.

(2) ***Preventive Action:*** No specific preventive actions are required. However, IN 87-67 identified that administrative controls should be implemented to ensure that the evaluation basis is either maintained or updated, when making plant modifications or re-classifying the safety significance of plant systems.

(3) ***Parameters Monitored or Inspected:*** The primary parameter monitored is wall cracking that could potentially invalidate the evaluation basis. Degradation of steel edge supports and wall bracing to the point where there is loss of intended function would also invalidate the evaluation basis. This latter condition is addressed in Chapter III B5.

(4) ***Detection of Aging Effects:*** Visual examination of the walls by qualified inspection personnel is sufficient. The frequency of inspection must be sufficient to ensure there is no loss of intended function between inspections. The inspection frequency may vary from wall to wall, depending on the significance of cracking in the evaluation basis. Unreinforced masonry walls that have not been contained by bracing require the most

frequent inspection because the development of cracks may invalidate the evaluation basis. These walls are to be inspected at least every refueling outage.

(5) ***Monitoring and Trending:*** Trending is not required. Monitoring is achieved by periodic examination.

(6) ***Acceptance Criteria:*** For each masonry wall, the extent of observed cracking of masonry and degradation of steel edge supports and bracing must not invalidate the evaluation basis established in response to IEB 80-11 or during implementation of USI A-46. If the extent of cracking and steel degradation is sufficient to invalidate the evaluation basis, corrective action is required. Another option is to develop a new evaluation basis that accounts for the degraded condition of the wall; i.e., acceptance by further evaluation.

(7) ***Corrective Actions:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) Operating Experience: IN 87-67 identified a number of conditions at several nuclear plants that invalidated the evaluation basis developed to resolve IEB 80-11, as well as instances in which walls performing intended functions were not addressed in the licensee's IEB 80-11 program. Implementation of the lessons learned from the IEB 80-11 program, as delineated in IN 87-67, should ensure that the structural integrity of all masonry walls within the scope of license renewal is adequately managed for the period of extended operation.

NOTE: In 1996, the NRC promulgated 10 CFR 50.65, the Maintenance Rule. Aging management for masonry walls may be incorporated in the Structures Monitoring Program (see Chapter XI.S6). However, details pertaining to masonry walls must incorporate the attributes of the program described above.

REFERENCES

10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, January 1997.

NUREG-1557, *Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal*, U.S. Nuclear Regulatory Commission, October 1996.

NRC Generic Letter 87-02, *Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46, February 19,1987.*

NRC IE Bulletin 80-11, *Masonry Wall Design*, May 8, 1980.

NRC Information Notice 87-67, Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11, December 31, 1987.

XI.S6

Structures Monitoring Program

INTRODUCTION

Implementation of structures monitoring under 10 CFR 50.65 (referred to as the Maintenance Rule) is addressed in Regulatory Guide 1.160, Rev. 2 and NUMARC 93-01, Rev. 2. These two (2) documents provide guidance for development of licensee-specific programs to monitor the condition of structures and structural components within the scope of the Maintenance Rule (MR), such that there is no loss of structure or structural component intended function.

For the purpose of license renewal, this aging management program (AMP) applies only to structures and structural components and applicable aging effects that are not addressed by the AMPs described in Sections XI.S1 thru XI.S4 and XI.S7; i.e., this AMP cannot be substituted for any of the five (5) specified AMPs.

Because structures monitoring programs are licensee-specific, the Evaluation and Technical Basis for this AMP is based on the implementation guidance provided in Regulatory Guide 1.160, Rev. 2 and NUMARC 93-01, Rev. 2.

If protective coatings are relied upon to manage the effects of aging for any structures included in the scope of this AMP, then the structures monitoring program needs to include requirements to address protective coating monitoring and maintenance in accordance with the guidance provided in Regulatory Guide 1.54, Rev. 1.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Program:*** Subject to the exclusions cited in the Introduction, the License Renewal applicant specifies the structure/aging effect combinations that are managed by its structures monitoring program.

(2) ***Preventive Action:*** No preventive actions are specified.

(3) ***Parameters Monitored/Inspected:*** For each structure/aging effect combination, the specific parameters monitored or inspected are selected to ensure that aging degradation leading to loss of intended functions will be detected and the extent of degradation can be determined. Parameters monitored or inspected must be commensurate with standard industry practice, as embodied in applicable codes, standards, and recommended existing practice, and must also consider industry and plant-specific operating experience. For concrete structural elements, parameters to be monitored or inspected include cracking, spalling, scaling, erosion, corrosion of reinforcing steel, settlements, and deformations. A more complete description of parameters for inclusion in this AMP is presented in ACI 349.3R-96. For steel liners and for joints, coatings, and waterproofing membranes (if any of these three items are relied upon to manage the effects of aging), ACI 349.3R-96 also specifies a description of the parameters to be monitored or inspected. For structural steel elements (including connections), parameters to be monitored or inspected include corrosion, cracking, erosion, discoloration, wear, pitting, gouges, dents, and other signs of surface irregularities. ANSI/ASCE 11-90 provides details for some of these parameters to be monitored or inspected. For welds, additional details on parameters to be monitored or inspected are provided in EPRI NP-5380. The licensee's plant-specific structures monitoring program must contain sufficient detail on parameters monitored or inspected so that a staff technical audit can reach the conclusion that Attribute 3 is satisfied.

(4) ***Detection:*** For each structure/aging effect combination, the inspection methods, inspection schedule, and inspector qualifications are selected to ensure that aging degradation will be detected and quantified before there is loss of intended functions. Inspection methods, inspection schedule, and inspector qualifications must be commensurate with standard industry practice, as embodied in applicable codes, standards, and recommended existing practice, and must also consider industry and plant-specific operating experience. Most detection methods rely upon visual inspections for identifying degradation. As specified in ACI 349.3R-96, "the visual inspection should include all exposed surfaces of the structure, joints and joint material, interfacing structures and materials (e.g., abutting soil), embedments, and attached components such as base plates and anchor bolts." ANSI/ASCE 11-90 specifies that inspection of the physical condition may sometimes require the use of simple physical assistance such as cleaning, scraping, and sounding. Details on detection methods for concrete; steel liners; and joints, coatings, and waterproofing material (if relied upon to manage the effects of aging) are specified in ACI 349.3R-96. Details on detection methods for structural steel (including connections) are specified in ASNI/ASCE 11-90. Additional details on detection methods for welds are specified in EPRI NP-5380. The frequency for the inspection of structures shall be dependent upon the structure, environment, and past performance; however, the frequency shall be no more than ten years. This frequency is in agreement with inspection intervals specified in ACI 349.3R-96 for concrete structures and recommendations given in NUREG-1522. The licensee's plant-specific structures monitoring program must contain sufficient detail on detection so that a staff technical audit can reach the conclusion that Attribute 4 is satisfied.

(5) ***Monitoring and Trending:*** Regulatory Position 1.5, "Monitoring of Structures" in Regulatory Guide 1.160, Rev. 2 provides an acceptable basis for meeting Attribute 5. Structures are monitored in accordance with 10 CFR 50.65 (a)(2) provided there is no significant degradation of the structure. A structure would be monitored in accordance with 10 CFR 50.65 (a)(1) if the extent of degradation is such that the structure may not meet its design basis or, if allowed to continue uncorrected until the next normally scheduled assessment, may not meet its design basis.

(6) ***Acceptance Criteria*****:** For each structure/aging effect combination, the acceptance criteria are selected to ensure that the need for corrective action will be identified prior to loss of intended functions. Acceptance criteria must be commensurate with standard industry practice, as embodied in applicable codes, standards, and recommended existing practice, and must also consider industry and plant-specific operating experience. For concrete structures (including steel liners and joints, coatings, and waterproofing material, if relied upon to manage the effects of aging), Chapter 5 of ACI 349.3R-96 specifies acceptance criteria. Acceptance criteria are specified for 1) acceptance without further evaluation, 2) acceptance after review, and 3) conditions requiring further evaluation. For example, acceptance without further evaluation for concrete is passive cracks in concrete less than 0.4 mm (0.015 in.) in maximum width. Acceptance criteria for visual examination of welds are specified in EPRI NP-5380. The licensee's plant-specific structures monitoring program must contain sufficient detail on acceptance criteria so that a staff technical audit can reach the conclusion that Attribute 6 is satisfied.

(7) ***Corrective Actions:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) ***Operating Experience:*** In many plants, structures monitoring programs have only recently been implemented. An applicant's plant-specific program that includes the attributes described above will be an effective AMP for License Renewal.

REFERENCES

10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Office of the Federal Register, National Archives and Records Administration, 2000.

ACI Standard 349.3R-96, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*, American Concrete Institute.

ANSI/ASCE 11-90, *Guideline for Structural Condition Assessment of Existing Buildings*, American Society of Civil Engineers.

EPRI NP-5380, *Visual Weld Acceptance Criteria*, prepared for Nuclear Construction Issues Group and Electric Power Research Institute, September, 1987.

NRC Regulatory Guide 1.54, Rev. 1, *Quality Assurance Requirements for Protective Coatings Applied to Water-Controlled Nuclear Power Plants*, July 2000.

NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, March 1997.

NUMARC 93-01, Rev. 2, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants [Line-In/Line-Out Version]*, Nuclear Energy Institute, April 1996.

NUREG-1522, *Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures*, U.S. Nuclear Regulatory Commission, June 1995.

XI.S7

RG 1.127 Inspection of Water-Control Structures Associated with Nuclear Power Plants

INTRODUCTION

USNRC Regulatory Guide 1.127 (RG 1.127), entitled "Inspection of Water-Control Structures Associated with Nuclear Power Plants," describes a basis acceptable to the NRC staff for developing an appropriate in-service inspection and surveillance program for dams, slopes, canals and other water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The RG 1.127 program addresses age-related deterioration, degradation due to extreme environmental conditions, and the effects of natural phenomena that may affect water-control structures. The RG 1.127 program recognizes the importance of periodic monitoring and maintenance of water-control structures so that the consequences of age-related deterioration and degradation can be prevented or mitigated in a timely manner.

RG 1.127 provides detailed guidance for the licensee's inspection program for water-control structures, including guidance on engineering data compilation, inspection activities, technical evaluation, inspection frequency, and the required content of inspection reports. Water-control structures, covered by the RG 1.127 program, include a) concrete structures; b) embankment structures; c) spillway structures and outlet works; d) reservoirs; e) cooling water channels and canals, and intake and discharge structures; and f) safety and performance instrumentation. The positions and requirements delineated reflect current NRC staff practice in evaluating in-service inspection programs for water-control structures.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Program:*** RG 1.127 applies to water-control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The water-control structures included in the RG 1.127 program are a) concrete structures; b) embankment structures; c) spillway structures and outlet works; d) reservoirs; e) cooling water channels and canals, and intake and discharge structures; and f) safety and performance instrumentation.

(2) ***Preventive Action:*** No preventive actions are specified; RG 1.127 is a monitoring program.

(3) ***Parameters Monitored/Inspected:*** RG 1.127 identifies the parameters to be monitored and inspected for water-control structures. The parameters vary depending on the particular structure. Parameters to be monitored and inspected for concrete structures include cracking, movements (e.g., settlement, heaving, deflection), conditions at junctions with abutments and embankments, erosion, cavitation, seepage, and leakage. Parameters to be monitored and inspected for earthen embankment structures include settlement, depressions, sink holes, slope stability (e.g., irregularities in alignment and variances from originally constructed slopes), seepage, proper functioning of drainage systems, and degradation of slope protection features. Further details of parameters to be monitored and inspected for these and other water-control structures are specified in Section C.2 of RG 1.127.

(4) ***Detection:*** Visual inspections are primarily used to detect degradation of water-control structures. In some cases, instruments have been installed to measure the behavior of water-control structures. RG 1.127 specifies that the available records and readings of installed instruments should be reviewed to detect any unusual performance or distress

that may be indicative of degradation. RG 1.127 requires periodic inspections at intervals of no more than 5 years. Similar intervals of 5 years are specified in ACI 349.3R for inspection of structures continually exposed to fluids or retaining fluids. Such intervals have been shown to be adequate to detect degradation of water-control structures before they have a significant effect on plant safety. RG 1.127 also identifies the need for special inspections immediately following the occurrence of significant natural phenomena such as large floods, earthquakes, hurricanes, tornadoes and intense local rainfalls.

(5) ***Monitoring and Trending:*** Monitoring of water-control structures is achieved by implementation of periodic inspection as specified in RG 1.127. In addition to monitoring the aging effects identified in Attribute (3) above, inspections also monitor the adequacy and quality of maintenance and operating procedures. RG 1.127 does not specify that trending is needed.

(6) ***Acceptance Criteria*****:** Acceptance criteria to evaluate the need for corrective action are not specified in RG 1.127. However, acceptance criteria based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R are acceptable. This document provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects; and specifies criteria when further evaluation is needed. ACI 349.3R is a recently published report written to evaluate existing nuclear safety-related concrete structures and reflects operating experience with reinforced concrete structures. Acceptance criteria for earthen structures such as dams, canals, and embankments must be consistent with programs falling within the regulatory jurisdiction of FERC or the Army Corps of Engineers.

(7) ***Corrective Actions:*** RG 1.127 requires that the licensee's in-service inspection and surveillance program include periodic inspections of water-control structures to identify deviations in structural conditions due to age-related deterioration and degradation from the original design basis. When findings indicate that significant changes have occurred, an evaluation of the conditions is required. This includes a technical assessment of the causes of distress or abnormal conditions, an evaluation of the behavior or movement of the structure, and recommendations for remedial or mitigating measures. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** As indicated in Section C.2 of RG 1.127, particular attention needs to be given to verify the adequacy and quality of maintenance and operating procedures. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** Inspections are documented in the form of a technical report which is retained on-site for reference purposes and NRC audits. The technical reports present the results of each inspection, along with conclusions and recommendations for additional investigations, remedial and mitigating measures, where appropriate, and abnormalities affecting the facility safety. As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) ***Operating Experience:*** Degradation of water-control structures has been detected, through RG 1.127 programs, at a number of nuclear power plants, and in some cases has required remedial action. No loss of intended functions has resulted from these occurrences. Therefore, it can be concluded that the inspections implemented in accordance with the guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs.

NOTE: Dam inspection and maintenance programs under the jurisdiction of FERC or the Army Corps of Engineers, continued through the period of the license renewal, will be adequate for the purpose of aging management. For programs not falling under the regulatory jurisdiction of FERC or the Army Corps of Engineers, the staff will evaluate the effectiveness of the aging management program based on compatibility to the common practices of the FERC and Corps programs.

REFERENCES

ACI Standard 349.3R-96, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*, American Concrete Institute.

NRC Regulatory Guide 1.127, Rev. 1, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*, Revision 1, March 1978.

XI.S8

Protective Coating Monitoring and Maintenance Program

INTRODUCTION

Proper maintenance of protective coatings inside containment (defined as Service Level I in Regulatory Guide (RG) 1.54, Rev. 1) is essential to ensure operability of post-accident safety systems that rely on water recycled through the containment sump/drain system. Degradation of coatings can lead to clogging of strainers, which causes reduction in flow through the sump/drain system. This has been previously addressed in GL98-04.

Maintenance of Service Level I coatings applied to carbon steel surfaces inside containment (e.g., steel liner, steel containment shell, penetrations, hatches) also serve to prevent or minimize loss of material due to corrosion. Regulatory Position C4 in RG 1.54, Rev. 1 defines an acceptable technical basis for a Service Level I coatings monitoring and maintenance program that can be credited for managing the effects of corrosion for carbon steel elements inside containment.

To be credited as an acceptable aging management program for License Renewal, a Service Level 1 coatings monitoring and maintenance program must include the attributes discussed below.

EVALUATION AND TECHNICAL BASIS

(1) ***Scope of Program:*** The minimum scope of the program is Service Level I coatings, defined in RG 1.54, Rev. 1 as follows: "Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown." The staff notes that Service Level II and III coatings, as defined in RG 1.54, Rev. 1, may also be managed by this program, and credit can be taken for managing the effects of corrosion for all coated carbon steel surfaces within the scope of the applicant's program.

(2) ***Preventive Action:*** With respect to loss of material due to corrosion of carbon steel elements, this program to monitor and maintain protective coatings is a preventive action.

(3) ***Parameters Monitored/Inspected:*** Regulatory Position C4 in RG 1.54, Rev. 1 states that "ASTM D 5163-96 provides guidelines that are acceptable to the NRC staff for establishing an in-service coatings monitoring program for Service Level I coating systems in operating nuclear power plants..." ASTM D 5163-96, subparagraph 9.2, identifies the parameters monitored/inspected to be "any visible defects, such as blistering, cracking, flaking/peeling, rusting, and physical damage."

(4) ***Detection of Aging Effects:*** ASTM D 5163-96, paragraph 5, defines the inspection frequency to be each refueling outage or during other major maintenance outages as needed. ASTM D 5163-96, paragraph 8, discusses the qualifications for inspection personnel, the inspection coordinator and the inspection results evaluator.

ASTM D 5163-96, subparagraph 9.1, discusses development of the inspection plan and the inspection methods to be utilized. It states "A general visual inspection shall be conducted on all readily accessible coated surfaces during a walk-through. After a walk-through, thorough visual inspections shall be carried out on previously designated areas and on areas noted as deficient during the walk-through. A thorough visual inspection shall also be carried out on all coatings near sumps or screens associated with the Emergency Core Cooling System (ECCS)." This subparagraph also addresses field documentation of inspection results. ASTM D 5163-96, subparagraph 9.5, identifies instruments and equipment needed for inspection.

(5) ***Monitoring and Trending:*** ASTM D 5163-96 identifies monitoring and trending activities in subparagraph 6.2, which specifies a pre-inspection review of the previous two (2) monitoring reports, and in subparagraph 10.1.2, which specifies that the inspection report should prioritize repair areas as either needing repair during the same outage or postponed to future outages, but under surveillance in the interim period.

(6) ***Acceptance Criteria*:** ASTM D 5163-96, subparagraphs 9.2.1 through 9.2.6, 9.3 and 9.4, contain guidance for characterization, documentation, and testing of defective or deficient coating surfaces. Additional ASTM and other recognized test methods are identified for use in characterizing the severity of observed defects and deficiencies. The evaluation covers blistering, cracking, flaking/peeling/delamination, and rusting. ASTM D 5163-96, paragraph 11 addresses evaluation. It specifies that the inspection report is to be evaluated by the responsible evaluation personnel, who prepare a summary of findings and recommendations for future surveillance or repair, including an analysis of reasons or suspected reasons for failure. Repair work is prioritized as major or minor defective areas. A recommended corrective action plan is required for major defective areas, so that these areas can be repaired during the same outage, if appropriate.

(7) ***Corrective Actions:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

(8) ***Confirmation Process:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing confirmation process.

(9) ***Administrative Controls:*** As discussed in the appendix to this report, the staff finds 10 CFR Part 50, Appendix B, acceptable in addressing administrative controls.

(10) ***Operating Experience:*** RG 1.54, Rev. 1 was issued in July 2000. Monitoring and maintenance of Service Level I coatings conducted in accordance with Regulatory Position C4 has no history of operating experience. It is expected to be an effective program for managing degradation of Service Level I coatings, and consequently an effective means to manage loss of material due to corrosion of carbon steel structural elements inside containment.

REFERENCES

American Society for Testing and Materials, ASTM D 5163-96, *Standard Guide for Establishing Procedures to Monitor the Performance of Safety Related Coatings in an Operating Nuclear Power Plant.*

NRC Generic Letter 98-04, *Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,* July 14, 1998.

NRC Regulatory Guide 1.54, Rev. 1, *Quality Assurance Requirements for Protective Coatings Applied to Water-Controlled Nuclear Power Plants,* July 2000.