

XI.S1 ASME SECTION XI, SUBSECTION IWE

Program Description

10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWE for steel containments (Class MC) and steel liners for concrete containments (Class CC). The full scope of IWE includes steel containment shells and their integral attachments; steel liners for concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. This evaluation covers both the 1992 Edition with the 1992 Addenda and the 1995 Edition with the 1996 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWE and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of steel containments, steel liners of concrete containments, and other containment components for license renewal.

The primary ISI method specified in IWE is visual examination (general visual, VT-3, VT-1). Limited volumetric examination (ultrasonic thickness measurement) and surface examination (e.g., liquid penetrant) may also be necessary in some instances. Bolt preload is checked by either a torque or tension test. IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

The evaluation of 10 CFR 50.55a and Subsection IWE as an aging management program (AMP) for license renewal is provided below.

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 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 the following 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-1232 and IWE-5220 are met; and
- (4) Piping, pumps, and valves that are part of the containment system or that 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 is to 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. 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. The first six examination categories (E-A through E-G) constitute the ISI requirements of IWE. Examination category E-P references 10 CFR Part 50, Appendix J leak rate testing. Appendix J leak rate testing is evaluated as a separate AMP for license renewal in XI.S4.

CATEGORY	PARTS EXAMINED	EXAMINATION METHOD ^a
E-A	Containment surfaces	General visual, visual VT-3
E-B ^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 ^b	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)
^a The applicable examination method (where multiple methods are listed) depends on the particular subcategory within each category. ^b These two categories are optional, in accordance with 10 CFR 50.55a(b)(2)(ix)(C).		

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 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 are 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 are to 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 will be performed in accordance with Inspection Program B. Under Inspection Program B, starting with the time the plant is placed into service, there is an initial inspection interval of 10 years and successive inspection intervals of 10 years each, during which 100% of the required examinations are to be completed. An expedited examination of containment is required by 10 CFR 50.55a in which an inservice (baseline) examination specified for the first period of the first inspection interval for containment is to 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 in item 3 above). IWE-1240 requires augmented examinations (Examination Category E-C) of containment surface areas subject to degradation. A VT-1 visual examination is performed for areas accessible from both sides, and volumetric (ultrasonic thickness measurement) examination is performed for areas accessible from only one side.

5. **Monitoring and Trending:** With the exception of inaccessible areas, all surfaces are monitored by virtue of the examination requirements on a scheduled basis. 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. 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 specifies that (a) examinations performed during any one inspection that reveal flaws or areas of degradation exceeding the acceptance standards are to 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 are to be performed to the extent specified in Table IWE-2500-1 for the inspection interval. Alternatives to these examinations 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. Most of the acceptance standards rely on 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 before the next examination, are documented. Such areas are to 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 are 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 subject to additional examination requirements, and the components are repaired or replaced to the extent necessary to meet the acceptance standards of IWE-3000. For repair of components within the scope of Subsection IWE, IWE-3124 states that repairs and reexaminations are to comply with IWA-4000. IWA-4000 provides repair specifications for pressure retaining components including metal containments and metallic liners of concrete containments. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** When areas of degradation are identified, an evaluation is performed to determine whether repair or replacement is necessary. If the evaluation determines that repair or replacement is necessary, Subsection IWE specifies confirmation that appropriate corrective actions have been completed and are effective. Subsection IWE states that repairs and reexaminations are to comply with the requirements of IWA-4000. Reexaminations are conducted in accordance with the requirements of IWA-2200, and the recorded results are to demonstrate that the repair meets the acceptance standards set forth in Table IWE-3410-1. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** IWA-6000 provides specifications for the preparation, submittal, and retention of records and reports. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** ASME Section XI, Subsection IWE was incorporated into 10 CFR 50.55a in 1996. Prior to this time, operating experience pertaining to degradation of steel components of containment was gained through the inspections required by 10 CFR Part 50, Appendix J and ad hoc inspections conducted by licensees and the Nuclear Regulatory Commission (NRC). NRC Information Notice (INs) 86-99, 88-82 and 89-79 described occurrences of corrosion in steel containment shells. NRC Generic Letter (GL) 87-05 addressed the potential for corrosion of boiling water reactor (BWR) Mark I steel drywells in the "sand pocket region." More recently, NRC IN 97-10 identified specific locations where concrete containments are susceptible to liner plate corrosion. The program is to consider the liner plate and containment shell corrosion concerns described in these generic communications. Implementation of the ISI requirements of Subsection IWE, in accordance with 10 CFR 50.55a, is a necessary element of aging management for steel components of steel and concrete containments through the period of extended operation.

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*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 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 IWB, *Requirements for Class 1 Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 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 IWC, *Requirements for Class 2 Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 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 IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 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 IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

NRC Generic Letter 87-05, *Request for Additional Information Assessment of Licensee Measures to Mitigate and/or Identify Potential Degradation of Mark I Drywells*, U.S. Nuclear Regulatory Commission, March 12, 1987.

NRC Information Notice 86-99, *Degradation of Steel Containments*, U.S. Nuclear Regulatory Commission, December 8, 1986 and Supplement 1, February 14, 1991.

NRC Information Notice 88-82, *Torus Shells with Corrosion and Degraded Coatings in BWR Containments*, U.S. Nuclear Regulatory Commission, October 14, 1988 and Supplement 1, May 2, 1989.

NRC Information Notice 89-79, *Degraded Coatings and Corrosion of Steel Containment Vessels*, U.S. Nuclear Regulatory Commission, December 1, 1989 and Supplement 1, June 29, 1989.

NRC Information Notice 97-10, *Liner Plate Corrosion in Concrete Containment*, U.S. Nuclear Regulatory Commission, March 13, 1997.

XI.S2 ASME SECTION XI, SUBSECTION IWL

Program Description

10 CFR 50.55a imposes the examination requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWL for reinforced and prestressed concrete containments (Class CC). The scope of IWL includes reinforced concrete and unbonded post-tensioning systems. This evaluation covers both the 1992 Edition with the 1992 Addenda and the 1995 Edition with the 1996 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWL and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of containment reinforced concrete and unbonded post-tensioning systems for license renewal.

The primary inspection method specified in IWL is visual examination (VT-3C, VT-1, VT-1C). For prestressed containments, tendon wires are tested for yield strength, ultimate tensile strength, and elongation. Tendon corrosion protection medium is analyzed for alkalinity, water content, and soluble ion concentrations. Prestressing forces are measured in selected sample tendons. IWL specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found.

The evaluation of 10 CFR 50.55a and Subsection IWL as an aging management program (AMP) for license renewal is provided below.

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 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 is to 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. 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. If a coating program is currently credited for managing the effects of aging of concrete surfaces, then the program is to be continued during the period of extended 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 on 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-77. 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, corrosion, and mechanical damage. Tendon wires or strands are also tested for yield strength, ultimate tensile 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 examinations 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 one, three, and five years following the structural integrity test. Thereafter, inspections are performed at five-year intervals. For sites with two plants, the schedule for inservice inspection is provided in IWL-2421. 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 minimum number of each tendon type selected for inspection varies from 2 to 4%. 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. These visual examination methods and testing would identify the aging effects of accessible concrete components and prestressing systems in concrete containments.
- 5. *Monitoring and Trending:*** Except in inaccessible areas, all concrete surfaces are monitored on a regular basis by virtue of the examination requirements. For prestressed containments, trending of prestressing forces in tendons is required in accordance with paragraph (b)(2)(viii) of 10 CFR 50.55a. 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 procedure 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 with 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. The acceptance criteria are qualitative; guidance is provided in IWL-2510, which references ACI 201.1R-77 for identification of concrete degradation. IWL-2320 requires that the Responsible Engineer 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. Quantitative acceptance criteria based on the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R may also be used to augment the qualitative assessment of the responsible engineer. 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 and

elongation, 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.

7. **Corrective Actions:** Subsection IWL specifies that items for which examination results do not meet the acceptance standards are to be evaluated in accordance with IWL-3300 "Evaluation" and described in an engineering evaluation report. The report is 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 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 the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** When areas of degradation are identified, an evaluation is performed to determine whether repair or replacement is necessary. As part of this evaluation, IWL-3300 specifies that the engineering evaluation report include the extent, nature, and frequency of additional examinations. IWL-4000 specifies the requirements for examination of areas that are repaired. Pressure tests following repair or modifications are in accordance with IWL-5000. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** IWA-1400 specifies 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 specified. IWA-6000 specifically covers the preparation, submittal, and retention of records and reports. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** ASME Section XI, Subsection IWL was incorporated into 10 CFR 50.55a in 1996. Prior to this time, operating experience pertaining to degradation of reinforced concrete and prestressing systems in concrete containments was gained through the inspections required by 10 CFR Part 50, Appendix J and ad hoc inspections conducted by licensees and the Nuclear Regulatory Commission (NRC). Recently, NRC Information Notice (IN) 99-10 described occurrences of degradation in prestressing systems. The program is to consider the degradation concerns described in this generic communication. Implementation of Subsection IWL, in accordance with 10 CFR 50.55a, is a necessary element of aging management for concrete containments through the period of extended operation.

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-77, *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*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 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 IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 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 IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants*, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.
- NRC Information Notice 99-10, Revision 1, *Degradation of Prestressing Tendon Systems in Prestressed Concrete Containment*, U.S. Nuclear Regulatory Commission, October 7, 1999.

XI.S3 ASME SECTION XI, SUBSECTION IWF

Program Description

10 CFR 50.55a imposes the inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for Class 1, 2, 3, and MC piping and components and their associated supports. Inservice inspection of supports for ASME piping and components is addressed in Section XI, Subsection IWF. This evaluation covers the 1989 Edition through the 1995 Edition and addenda through the 1996 Addenda, as approved in 10 CFR 50.55a. ASME Code Section XI, Subsection IWF constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, 3, and MC supports for license renewal.

The IWF scope of inspection for supports is based on sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports (ASME Class 1). The sample size decreases for the less critical supports (ASME Class 2 and 3). Discovery of support deficiencies during regularly scheduled inspections triggers an increase of the inspection scope, in order to ensure that the full extent of deficiencies is identified. The primary inspection method employed is visual examination. Degradation that potentially compromises support function or load capacity is identified for evaluation. IWF specifies acceptance criteria and corrective actions. Supports requiring corrective actions are re-examined during the next inspection period.

The evaluation of Subsection IWF as an aging management program (AMP) for license renewal is provided below.

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. According to Table IWB-2500-1, only 25% of nonexempt supports are subject to examination. Supports exempt from examination are the supports for piping systems that are exempt from examination, according to 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. According to Table IWC-2500-1, 7.5% of nonexempt 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 nonexempt support subject to examination were incorporated into Table IWF-2500-1. The revised percentages are 25% of Class 1 nonexempt piping supports, 15% of Class 2 nonexempt piping supports, 10% of Class 3 nonexempt piping supports, and 100% of supports other than piping supports (Class 1, 2, 3, and MC). For pipe supports, the total sample consists of supports from each system (such as main steam, feedwater, residual heat removal), where the individual sample sizes are proportional to the total number of nonexempt 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; Subsection IWF is an inspection program.
3. **Parameters Monitored or Inspected:** IWF specifies visual examination (VT-3) of supports. 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. The visual inspection would be expected to identify 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 nonintegral supports;
- (F1.40) Clearances of guides and stops, alignment of supports, and 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.)

4. **Detection of Aging Effects:** 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; observed degradation is documented if loss of structural capacity is suspected.
5. **Monitoring and Trending:** There is no requirement to monitor or report progressive, time-dependent degradation. Unacceptable conditions, according to IWF-3400, are noted for correction or further evaluation.
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 reexamination of the supports requiring corrective actions 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 before returning to service. Corrective actions are delineated in IWF-3122.2. IWF-3122.3 provides an alternative for evaluation or testing, to substantiate structural integrity and/or functionality. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** To date, IWF sampling inspections have been effective in managing aging effects for ASME Class 1, 2, 3, and MC supports. 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*, 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 IWF, *Requirements for Class 1, 2, 3, and MC Component Supports of Light-Water Cooled Power Plants*, 1989 Edition through the 1995 Edition with 1996 Addenda. The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

XI.S4 10 CFR PART 50, APPENDIX J

Program Description

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."

Appendix J provides two options, A and 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 the Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 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 are 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 B tests described in 10 CFR Part 50, Appendix J, are acceptable methods for performing these LRTs. Leakage testing for containment isolation valves (normally performed under Type C tests), if not included under this program, is included under LRT programs for systems containing the isolation valves.
- 2. *Preventive Action:*** No preventive actions are specified; the containment LRT program is a monitoring program.
- 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 XI.S1 and XI.S2.
- 5. *Monitoring and Trending:*** Because the LRT program is repeated throughout the operating license period, the entire pressure boundary is 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 on the basis of acceptable performance in meeting leakage limits in prior tests. Additional details for implementing Option B are provided in NRC Regulatory Guide 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. The current licensing basis 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, an evaluation is performed 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 the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** When corrective actions are implemented to repair a condition that causes excessive leakage, confirmation by additional leak rate testing is performed to confirm that the deficiency has been corrected. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** Results of the LRT program are documented as described in 10 CFR Part 50, Appendix J, to demonstrate that the acceptance criteria for leakage have been satisfied. 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 the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** To date, the 10 CFR Part 50, Appendix J, LRT 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*,"
U.S. Nuclear Regulatory Commission, September 1995.

XI.S5 MASONRY WALL PROGRAM

Program Description

Nuclear Regulatory Commission (NRC) IE Bulletin (IEB) 80-11, "Masonry Wall Design," and NRC Information Notice (IN) 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," constitute an acceptable basis for a masonry wall aging management program (AMP). IEB 80-11 required the identification of masonry walls in close proximity to, or having attachments from, safety-related systems or components, and the evaluation of design adequacy and construction practice. NRC IN 87-67 recommended plant-specific condition monitoring of masonry walls and administrative controls to ensure that the evaluation basis developed in response to NRC 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 NRC IEB 80-11 program included (1) installation of steel edge supports to provide a sound technical basis for boundary conditions used in seismic analysis and (2) 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.

The objective of the masonry wall program is to manage aging effects so that the evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation. Since the issuance of NRC IEB 80-11 and NRC IN 87-67, the NRC promulgated 10 CFR 50.65, the Maintenance Rule. Masonry walls may be inspected as part of the Structures Monitoring Program (XI.S6) conducted for the Maintenance Rule, provided the ten attributes described below are incorporated.

The attributes of an acceptable Masonry Wall Program are described below.

Evaluation and Technical Basis

1. **Scope of Program:** The scope includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4.
2. **Preventive Action:** No specific preventive actions are required.
3. **Parameters Monitored or Inspected:** The primary parameter monitored is wall cracking that could potentially invalidate the evaluation basis.
4. **Detection of Aging Effects:** Visual examination of the masonry walls by qualified inspection personnel is sufficient. The frequency of inspection is selected 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 warrant the most frequent inspection, because the development of cracks may invalidate the existing evaluation basis.

5. **Monitoring and Trending:** Trending is not required. Monitoring is achieved by periodic examination for cracking.
6. **Acceptance Criteria:** For each masonry wall, the extent of observed cracking of masonry and degradation of steel edge supports and bracing is not to invalidate the evaluation basis. Corrective actions are taken if the extent of cracking and steel degradation is sufficient to invalidate the evaluation basis. An 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 the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** Since 1980, masonry walls that perform an intended function have been systematically identified through licensee programs in response to NRC IEB 80-11, USI A-46, and 10 CFR 50.48. NRC IN 87-67 documented lessons learned from the NRC IEB 80-11 program, and provided recommendations for administrative controls and periodic inspection to ensure that the evaluation basis for each safety-significant masonry wall is maintained. Whether conducted as a stand-alone program or as part of structures monitoring for MR, a masonry wall AMP that incorporates the recommendations delineated in NRC IN 87-67 should ensure that the intended functions of all masonry walls within the scope of license renewal are maintained for the period of extended operation.

References

10 CFR 50.48, *Fire Protection*, Office of the Federal Register, National Archives and Records Administration, 2000.

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.

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

NRC IE Bulletin 80-11, *Masonry Wall Design*, U.S. Nuclear Regulatory Commission, May 8, 1980.

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

XI.S6 STRUCTURES MONITORING PROGRAM

Program Description

Implementation of structures monitoring under 10 CFR 50.65 (the Maintenance Rule) is addressed in Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. These two 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, such that there is no loss of structure or structural component intended function.

Because structures monitoring programs are licensee-specific, the Evaluation and Technical Basis for this aging management program (AMP) is based on the implementation guidance provided in Regulatory Guide 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. Existing licensee-specific programs developed for the implementation of structures monitoring under 10 CFR 50.65 are acceptable for license renewal provided these programs satisfy the 10 attributes described below.

If protective coatings are relied upon to manage the effects of aging for any structures included in the scope of this AMP, the structures monitoring program is to address protective coating monitoring and maintenance.

Evaluation and Technical Basis

1. **Scope of Program:** The 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 or 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 are to be commensurate with industry codes, standards and guidelines, and are to also consider industry and plant-specific operating experience. Although not required, ACI 349.3R-96 and ANSI/ASCE 11-90 provide an acceptable basis for selection of parameters to be monitored or inspected for concrete and steel structural elements and for steel liners, joints, coatings, and waterproofing membranes (if applicable). If necessary for managing settlement and erosion of porous concrete subfoundations, the continued functionality of a site de-watering system is to be monitored. The plant-specific structures monitoring program is to contain sufficient detail on parameters monitored or inspected to conclude that this program attribute is satisfied.
4. **Detection of Aging Effects:** 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 are to be commensurate with industry codes, standards and guidelines, and are to also consider industry and plant-specific operating experience. Although not required, ACI 349.3R-96 and ANSI/ASCE 11-90 provide an acceptable basis for addressing detection of aging effects. The plant-specific structures monitoring program is to contain sufficient detail on detection to conclude that this program attribute is satisfied.

5. **Monitoring and Trending:** Regulatory Position 1.5, "Monitoring of Structures," in RG 1.160, Rev. 2, provides an acceptable basis for meeting the attribute. A structure is monitored in accordance with 10 CFR 50.65 (a)(2) provided there is no significant degradation of the structure. A structure is 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 actions will be identified before loss of intended functions. Acceptance criteria are to be commensurate with industry codes, standards and guidelines, and are to also consider industry and plant-specific operating experience. Although not required, ACI 349.3R-96 provides an acceptable basis for developing acceptance criteria for concrete structural elements, steel liners, joints, coatings, and waterproofing membranes. The plant-specific structures monitoring program is to contain sufficient detail on acceptance criteria to conclude that this program attribute is satisfied.
7. **Corrective Actions:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** Although in many plants structures monitoring programs have only recently been implemented, plant maintenance has been ongoing since initial plant operation. A 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.
- NRC Regulatory Guide 1.160, Rev. 2, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, 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.

XI.S7 RG 1.127, INSPECTION OF WATER-CONTROL STRUCTURES ASSOCIATED WITH NUCLEAR POWER PLANTS

Program Description

Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.127, Revision 1, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," describes an acceptable basis for developing an inservice 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 content of inspection reports. Water-control structures covered by the RG 1.127 program include concrete structures; embankment structures; spillway structures and outlet works; reservoirs; cooling water channels and canals, and intake and discharge structures; and safety and performance instrumentation. RG 1.127 delineates current NRC practice in evaluating inservice inspection programs for water-control structures. The attributes of an acceptable aging management program (AMP) for license renewal are described below.

For plants not committed to RG 1.127, Revision 1, aging management of water-control structures may be included in the Structures Monitoring Program (XI.S6). However, details pertaining to water-control structures are to incorporate the attributes described herein.

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 concrete structures; embankment structures; spillway structures and outlet works; reservoirs; cooling water channels and canals, and intake and discharge structures; and safety and performance instrumentation.
- 2. Preventive Action:** No preventive actions are specified; RG 1.127 is a monitoring program.
- 3. Parameters Monitored or 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 of Aging Effects:** 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 indicates that the available records and readings of installed instruments are to be reviewed to detect any unusual performance or distress that may be indicative of degradation. RG 1.127 describes periodic inspections, to be performed at least once every five years. Similar intervals of five 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 describes 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:** Water-control structures are monitored by periodic inspection as described 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 discuss trending.
6. **Acceptance Criteria:** Acceptance criteria to evaluate the need for corrective actions are not specified in RG 1.127. However, the "Evaluation Criteria" provided in Chapter 5 of ACI 349.3R-96 provides acceptance criteria (including quantitative criteria) for determining the adequacy of observed aging effects and specifies criteria for further evaluation. Although not required, plant-specific acceptance criteria based on Chapter 5 of ACI 349.3R-96 are acceptable. Acceptance criteria for earthen structures such as dams, canals, and embankments are to be consistent with programs falling within the regulatory jurisdiction of the Federal Energy Regulatory Commission (FERC) or the U.S. Army Corps of Engineers.
7. **Corrective Actions:** RG 1.127 recommends that the licensee's inservice 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, the conditions are to be evaluated. 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 the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address 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, it 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: For dam inspection and maintenance, programs under the regulatory jurisdiction of FERC or the U.S. Army Corps of Engineers, continued through the period of extended operation, will be adequate for the purpose of aging management. For programs not falling under the regulatory jurisdiction of FERC or the U.S. 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, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*, Revision 1, U.S. Nuclear Regulatory Commission, March 1978.

XI.S8 PROTECTIVE COATING MONITORING AND MAINTENANCE PROGRAM

Program Description

Proper maintenance of protective coatings inside containment (defined as Service Level I in Nuclear Regulatory Commission [NRC] 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 reduces flow through the sump/drain system. This has been addressed in NRC Generic Letter (GL) 98-04.

Maintenance of Service Level I coatings applied to carbon steel surfaces inside containment (e.g., steel liner, steel containment shell, penetrations, hatches) also serves to prevent or minimize loss of material due to corrosion. Regulatory Position C4 in RG 1.54, Rev. 1, describes 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. The attributes of an acceptable program are described below.

A comparable program for monitoring and maintaining protective coatings inside containment, developed in accordance with RG 1.54, Rev. 0 or the American National Standards Institute (ANSI) standards (since withdrawn) referenced in RG 1.54, Rev. 0, and coatings maintenance programs described in licensee responses to GL 98-04, is also acceptable as an aging management program (AMP) for license renewal.

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."
2. **Preventive Action:** With respect to loss of material due to corrosion of carbon steel elements, this program is a preventive action.
3. **Parameters Monitored or 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 or 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 used. 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 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 the requirements of 10 CFR Part 50, Appendix B, acceptable to address corrective actions.
8. **Confirmation Process:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process.
9. **Administrative Controls:** As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address administrative controls.
10. **Operating Experience:** NRC Generic Letter 98-04 describes industry experience pertaining to coatings degradation inside containment and the consequential clogging of sump strainers. 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 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

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

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*, U.S. Nuclear Regulatory Commission, July 14, 1998.

NRC Regulatory Guide 1.54, Rev. 0, *Quality Assurance Requirements for Protective Coatings Applied to Water-Controlled Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, June 1973.

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